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Application of personal computers to enhance operation and management of research reactors

Proceedings of a final Research Co-ordination Meeting held in Dalat, Viet Nam, 30 October – 3 November 1995

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FOREWORD

The on-line use of personal computers (PCs) can be valuable to guide the research reactor operator in analysing both normal and abnormal situations. PCs can effectively be used for data acquisition and data processing, and providing information to the operator. Typical areas of on-line applications of PCs in nuclear research reactors include:

- Acquisition and display of data on process parameters.
- Performance evaluation of major equipment and safety related components.
- Fuel management.
- Computation of reactor physics parameters.
- Failed fuel detection and location.
- Inventory of system fluids.
- Training using computer aided simulation.
- Operator advice.

All these applications require the development of computer programmes and interface hardware. In recognizing this need, the IAEA initiated in 1990 a Co-ordinated Research Programme (CRP) on "Application of Personal Computers to Enhance Operation and Management of Research Reactors". The final meeting of the CRP was held from 30 October to 3 November 1995 in Dalat Viet Nam.

This report was written by contributors from Bangladesh, Germany, India, the Republic of Korea, Pakistan, Philippines, Thailand and Viet Nam. The IAEA staff members responsible for the publication were K. Akhtar and V. Dimic of the Physics Section, Division of Physical and Chemical Sciences.

EDITORIAL NOTE

CONTENTS

1. INTRODUCTION

In the past decade, the rapid growth of computer technology has increased the availability of high performance computers at reasonable cost, so it is worthwhile to promote the use of personal computers (PCs) for operation, monitoring and management of research reactors. This, however, requires development of computer programmes and interface hardware. The IAEA recognized this need and initiated a Co-ordinated Research Programme (CRP) on "Application of Personal Computers to Enhance Operation and Management of Research Reactors".

The initial goal of this CRP was to develop software and hardware for use in research reactor operation and management related activities through the application of PCs. The scientific scope comprised monitoring reactor power level, monitoring reactor instrumentation, calibration of neutron detectors, research reactor diagnostic system, on-line data logging, computer aided instructions, maintenance, in-service inspection, basic simulators, etc. The CRP envisaged development of tested and documented codes for worldwide use.

There has been progress worldwide on the digitalization of reactor operation, monitoring and management systems. Complete digital systems of these applications are available, after completion of a process of development — just as was standard practice for the hardwired system that precedent them — and a qualification period lasting several years outside the plants with associated licensing. A number of subsystems are now available for backfitting and upgrading applications.

This new generation of systems makes full use of the inherent advantages of digital information processing based on distributed microprocessor. The benefits claimed include: reduced maintenance and calibration requirements, improved self-testing, better quality information for the operator and increased safety. However, these favourable attributes have their negative counterparts, the most notable being complexity, high sensitivity to any size error, testability and most importantly reliability quantification and common cause failure.

The licensing safety requirements to be met by safety systems are independent of the hardware used. The characteristics of computerized systems deviate greatly from the older, conventional analogue systems in that design and licensing evaluation methods require new kinds of approaches. When nuclear design and licensing principles were originally developed, the impact of computerized technology had not been considered. Therefore, it is appropriate for designers and licensing authorities to devise a policy for the assessment of software in safety systems since software design poses problems that are specific and different from those posed by the design of analogue hardware. The assessment of software reliability remains difficult.

- Software is never error free.
- Full testing of any software of medium complexity is not possible.
- The principle of redundant components does not apply, as all copies of the same software would contain the same design error.

However, it should be noted that there is great potential for nuclear safety enhancement, if computerized safety systems are used correctly, through well designed, engineered, installed and maintained systems.

The CRP was initiated in 1990. It was originally approved for three 3 years, but was extended for another two years, reaching completion in 1995. Initially, the programme comprised 9 research contracts and one research agreement, of which 7 contracts and one agreement were finally completed. During the programme period of 1990-1995, three research co-ordination meetings were organized in which status, progress and achievements of each project were reviewed and future plans were discussed.

2. ACHIEVEMENTS

The objectives of various projects under the CRP varied from on-line monitoring and display of reactor status to detailed thermal hydraulic and neutronic analysis. The research work included the development of hardware and software for PC-controlled data acquisition systems, and analysis and processing of reactor data. Although a wide range of subjects was covered, the CRP could not cover all the desired topics.

List of institutes that participated in the CRP and titles of their research projects:

Institute	*Project Title*
1. Institute of Nuclear Science and Technology, Bangladesh	Development of Data Acquisition System for the 3 MW TRIGA Reactor at AERE
2. Forschungszentrum (KFA) Juelich, Germany	Small Computer Codes for Research Reactor Operation
3. Bhabha Atomic Research Centre, India	On-line Use of Personal Computers to Monitor and Evaluate Research Reactor Parameters
4. Korea Atomic Energy Research Institute, Republic of Korea	Development of Research Reactor Parameter Measuring System Based on PC
5. Pakistan Institute of Nuclear Science & Technology, Pakistan	Development of a Central PC- Based Radiation Level Monitoring and Display System for PARR-1
6. Philippine Nuclear Research Institute, Philippines	Integration of PCs with Instrumentation System of PRR1
7. Office of Atomic Energy for Peace, Thailand	Programme Package for 2-D Burnup Calculations
8. Institute of Nuclear Science and Technique, Viet Nam	Windows User-Friendly Code Package Development for Operation of Research Reactors

The NURESIM (Nuclear Reactor Simulation) software package was also developed by the Institute of Nuclear Science and Technique, Hanoi, (Research Contract No. VIE/5304) in order to provide users a set of tools for training on fundamentals of reactor theories, as well as for performing reactor calculations. The project "Windows User-Friendly Code Package Development for Operation of Research Reactors" is a continuation of the project "NURESIM Lectures on Reactor Physics" which is attached to this report.

2.1. MONITORING THE REACTOR STATUS

One of the major applications of PCs is on-line monitoring and display of reactor signals and information. These signals are acquired from normal instrumentation of the plant and displayed on monitors in the control room and other key locations. This provides the reactor operator and users with the necessary information about the reactor status.

Signals acquired by the computer may include those related with reactor safety and operation from both nuclear and process channels.

The main advantages of using PCs for monitoring the reactor status include:

- PCs are compact and cost effective data logging systems.
- Essential reactor information may be provided to different locations through networking.
- Data can conveniently be stored in the computer memory. This may reduce the use of chart recorders and may result in saving operation and maintenance costs.
- Video and audio alarms may be generated by the computer in case of abnormal conditions.
- Data acquired by the computer is available in a ready form for further processing and evaluation.
- Computer monitoring of a reactor status may improve the appearance of the reactor control room.

The last application was especially attractive and four of the projects were related to on-line monitoring and display of signals.

The main accomplishments of these projects are summarized below:

- The project work at thr Institute of Nuclear Science and Technology, Bangladesh, developed a microcomputer based system for on-line monitoring, display and storage of all operational parameters and information of the 3 MW TRIGA reactor. The system is now running satisfactorily.
- The project undertaken at BARC, Bombay, developed data acquisition and display system for important physics and process parameters of 100 MW research reactor DHRUVA. This system is now in routine use.
- For the upgraded MW Pakistan Research Reactor-1 (PARR-1) a system for centralized, on-line, radiation monitoring has been developed and installed.
- At the Philippine Nuclear Research Institute, Manila, a PC system for data monitoring and analysis of a 3 MW research reactor (PRR-1) was developed. The system was tested with simulation signals.

These examples cover a wide range of reactor power levels and types. The methods used may be applicable to other research reactors.

The hardware needed for implementing reactor status monitoring systems by various projects has the following required features.

(a) Signal conditioning

Signals from instrumentation channels should be conditioned to compatible voltage signals for ADC or digital input system. The conditioning includes amplification of flow level voltage, current to voltage conversion, resistance to voltage conversion, and pulse to pulse conversion.

Proper isolation between the reactor signals and the computer must be ensured such that any malfunction in the computer is not transmitted to the reactor instrumentation. Isolation amplifiers are available commercially to provide full protection to channel output signals. Protection circuits may also be installed at the input to the interface card so as to avoid the possibility of any damage to the interface card and the computer.

(b) Data acquisition

Acquisition of analogue signals has been achieved with the help of analogue-to-digital converter (ADC) cards. The selection of ADC cards is based on the requirements of number of signals, resolution, sampling speed and cost. They should be capable of acquiring analogue, digital and pulse signals.

Another requirement is digital-to-analogue conversion and provision of digital outputs for driving external indicators or meters.

Generally, the number of signals from reactors required for monitoring is large. Therefore, it is necessary to use multiple I/O cards or multiple computers. As an example, PARR-1 and DHRUVA use multiple I/O cards and PRR-1 uses multiple computers with networking.

(c) Computer

Large computer memory and memory devices may be needed for data storage purpose. If additional memory modules are used some battery backup is required. Fast execution speed may not be an essential requirement, as the data processing is minimal. However, this may be a consideration in cases where the number of signals is large and further data processing is required.

(d) Multiple display and networking

Essential reactor status information may be conveyed to various locations. Two schemes have been used for this purpose. In PARR/1 a closed circuit television (CCTV) system of the plant has been used for the distributed display. In this case the video output signal is displayed from the PC. Another scheme has been adopted in DHRUVA where five computers in a local area network provide on-line and off-line information to plant personnel.

2.2. EVALUATION OF REACTOR CHARACTERISTICS

This application involves analysis of data acquired by the PCs. The hardware for signal acquisition is the same as described in the previous section. However, software requirements would depend on each particular application. Standard programming languages can be used for software development both for on-line and off-line applications. There are many parameters to be evaluated by the PC based system but their selection depends on reactor characteristics and on the needs of the institutes.

For example, the heavy water inventory monitoring was a special interest at the heavy water reactor DHRUVA.

In DHRUVA and PARR-1 the data acquisition system developed for reactor status monitoring is also used for evaluating reactor characteristics and performance monitoring. In the case of Korean research reactors, a simple PC system, independent from the reactor monitoring system has been employed for similar purposes.

For the routine experiments of two TRIGA reactors and the commissioning experiments of the new 30 MW HANARO research reactor in the Republic of Korea, a stand alone system, with input signals independent from the reactor operation and safety channels, is configured. This system has been utilized for the criticality approach, real time reactivity measurement, noise analyses, control rod drop time measurement, and thermal power calibration in a natural convection cooled reactor. It replaces conventional counter modules for the criticality measurement, multichannel analyzer and frequency spectrum analyzer for noise analyses, memory oscilloscope or timer for control rod drop time measurement, and recorders or meters for other experiments.

The reactivity meter can use either current or pulse type neutron detectors, covers a wide signal range, and can compensate gamma or neutron source effects.

The noise analyzer can be applied at either critical or subcritical conditions by using multiple neutron detectors simultaneously. One of three representative methods - variance to mean ratio, correlation, and power spectral density methods, can be chosen for real time measurements.

In PARR-1 a new application of real-time signal processing has been used. Statistical analysis of signals from reactor instrumentation channels is done in real-time for evaluation of instrumentation performance. The computer calculates mean value, standard deviation errors, and probability distribution function of the signals and compares these errors with reference errors of nuclear detection phenomena. In case of a malfunction in any part of instrumentation, the signal error exceeds the reference error and the computer generates an alarm. In this way a faulty instrument channel is identified.

In DHRUVA, on-line computation of important physics and process parameters has been achieved. The parameters selected are reactor thermal power, reactivity load due to Xenon, core reactivity balance, heavy water system inventory and performance monitoring of shut-off rods control valve and dump valves. Also off-line application for fuel management, failed fuel detection and location, and stores inventory management have been implemented.

The capabilities of PC for real-time and off-line reactor analysis have been demonstrated in the work described above. A wide range of parametric measurements and analyses were covered. These include; thermal power, reactivity change due to temperature, poison, reactivity measurements and control rod calibration, performance evaluation of instrumentation channels and of shut down devices, approach-to-criticality and kinetic measurement.

There are other parametric measurements and analyses that are not covered in the development work.

2.3. NEUTRONICS CALCULATIONS

A package for analyzing and controlling the burnup behaviour of fuel elements was developed for use on an IBM/PC. The programme package consists of 3 modules, 1 library and 2 input files. The package was written in FORTRAN 77 using NDP FORTRAN compiler version 4.02 Microway Inc.

The first module, PRESIX, prepares the 2 group cross sections for various reactor conditions depending on the burnup history of the fuel elements and core loading pattern. The cross section are stored in XSEC file. The second module, SIXTUS-2, a two dimensional diffusion code reads the cross section data from XSEC file along with the geometry data. It models the reactor core in hexagonal geometry and calculates k_{eff}, neutron flux and power distributions. BURN module performs the calculation of fuel element burnup and stores the burnup history of fuel elements in ELEM.DAT file. The two group cross sections of major components of the core i.e. fuel elements, irradiation channels, etc., are provided by PRESIX.LIB which is generated by WIMS-D4 code.

This code was applied to the follow-up calculation of operation history of the Thai research reactor (TRR-1/1M1). The calculated excess reactivity variation was compared with the measured data, and the difference was marginal.

The package is applicable to the other TRIGA reactors, if a specific library is prepared.

2.4. THERMALHYDRAULIC CALCULATIONS

Heat generated in the fuel plates in the reactor core is mainly transferred by heat conduction across the fuel plates and removed by the force convection of the coolant. To meet the need for heat transfer calculations, the software package HEATHYD (a HEAT-transfer and HYDraulic code) with modules for heat transfer phenomena and fluid flow under steady state operating conditions has been developed for this purpose.

Flow distribution and pressure losses for any arrangement of fuel plates and cooling gaps are determined by iterative hydraulic calculation. Using total mass flow and channel dimensions, the local pressure is determined to calculate local saturation temperature as a criterion for the onset of boiling. Hydraulic calculations are linked to the heat transfer module. To take into account the variation of material properties (density, viscosity, conductivity, and heat capacity) with the local pressure and temperature, the hydraulic and heat transfer modules are linked through outer iteration.

The heat transfer module is based on equations for thermal conduction and Newton's law of cooling, heat removal from the plate surface. The convective heat transfer coefficients are determined by applying a variety of empirical correlations.

The code determines the onset of nucleate boiling and critical heat flux corresponding to flow instability in each individual channel and burnout of fuel plate. Heat flux for the onset of nucleate boiling and critical heat flux are calculated using actual power distribution, coolant velocity, local pressure and saturation temperature at each individual code. Margins to nucleate boiling, flow instability and burnout are also calculated.

The code was verified by thermal hydraulic measurements using instrumented fuel elements in the core of an MTR research reactor. A comparison of the measurements and calculations shows that HEATHYD can predict the phenomena of heat transfer and fluid flow with good accuracy.

2.5. EDUCATION AND TRAINING

Rapid growth of PC use and software engineering has also resulted in the development of software packages for staff training using computer aided simulations.

The NURESIM (NUclear REactor SIMulation) is a software developed to provide a set of tools for training on fundamentals of reactor theory as well as for performing reactor calculations.

The software is written so that the users could apply all parts individually and as a whole. But in fact these parts are loaded into computer memory separately and independently. Because of this feature, the NURESIM may be used on the PC/AT/386 with a memory of 620k.

The software is designed so that it can be used by trainees as well as by qualified staff and analysts. Trainees could find fundamental concepts, and analysts some global calculation codes such as GRACE, PEACO, THERMOS and HEX120. Therefore, the purpose of the NURESIM is not only to be used for education and training, but it could be used as a compilation of databases and codes for making reactor physics calculations. Indeed, the above-mentioned codes have been used for calculating some static reactor physics parameters of the Dalat Nuclear Research Reactor in Viet Nam.

The package contains a number of sections in the text part with figures, flow diagrams and graphics. The text part explains the fundamentals of reactor physics, including basic calculations on reactor physics and heat generation.

Using interactive techniques, the simulation part is intended to enhance the learning process and to provide a flexible tool for training. Users are able to act as a reactor operator by using the computer keyboard and following the reactor behaviour being simulated on the computer display.

The static reactor calculations are performed on the basis of data libraries and input files by running numerical codes which are included in the software package.

Another feature of the NURESIM is its open capacity, so that additional texts, figures, simulation and calculation codes can be easily included.

3. CONCLUSIONS

(1) The CRP has generally achieved its objectives. The programme has contributed in the application of PCs for on-line data acquisition and processing and for evaluating reactor performance. The following benefits were achieved by data analysis for reactor performance evaluation in the CRP projects:

- The PC serves as an expert system. For example, computation of xenon buildup after reactor shutdown with varying operational histories and determination of available time for startup before the reactor poisons out can be easily performed by shift staff themselves without the need for elaborate calculations to be performed by reactor physicists.
- Data handling is much easier by utilizing a PC (graphic displays, storage data, printing and processing).
- A PC enhances the accuracy of the results by minimizing reading error and statistical treatment of many data.
- A PC extends measuring speed and range by fast and continuous sampling of various parameters at the same time. It has also potential for application in several other fields.
- A PC saves operator time and effort and is cost effective.

(2) Small computer codes (SOTRAN, KINIK, HEATHYD, CREMAT, XE, etc.) for performing calculations of fuel burnup, reactivity effects and reactor kinetics have been developed or adapted.

(3) To enhance operation and management of research reactors, software packages have been needed to simulate the complex process of cooling and the mechanisms of heat removal in the reactor core. With the availability of high performance personal computers, activities on the development of numerical thermohydraulic models have been initiated and performed as a major objective of the CRP.

(4) Training of reactor staff has improved through simulation of reactor performance and better understanding of reactor systems.

(5) The individual project achievements have stimulated general interest at the participating institutes. The results may be used by other research reactor centres.

Annex

COMPLETED RESEARCH PROJECTS

XA9846763

DATA ACQUISITION SYSTEM FOR THE 3 MW TRIGA REACTOR AT AERE SAVAR*

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Abstract

A 3 MW TRIGA Mark II research reactor control console has been studied in detail and the channels have been selected for monitoring, display and record using the microcomputer. Information from these channels are fed to the computer through hardware like buffer, AD converter, multiplexer, etc. for continuos display and permanent records using video monitor, printer and diskettes. Besides, the information from the console, other information like operating time, power, total burnup of fuel, operating persons, etc. are also available, With very little modifications in both hardware and software, the data logging system is now running successfully.

INTRODUCTION

A 3 MW TRIGA Mark II research reactor has become operational at Atomic Energy Research Establishment, Savar Dhaka since September, 1986. The reactor is being used mainly for isotope production, neutron radiography, activation analysis and for other experiments of academic use. A triple axes spectrometer has also been installed at the reactor facility.

The Control and Instrumentation group of Reactor Engineering and Control Division is engaged in R & D works in automatic control based system, and control & instrumentation of nuclear reactors. Under this project a microcomputer based system is being designed and developed to automatically monitor and record all operational parameters and relevant information of the reactor in an efficient manner.

The reactor control console has been studied in detail and the channels as per attached list (Annexure 1) have been selected for monitoring, display and record using the microcomputer. Information from these channels will be fed to the computer through hardware like buffer, AD converter, multiplexer,etc for continuous display and permanent records using video monitor, printer and diskettes. Besides, the information from the console, other information like operating time, power, total burnup of fuel, operating persons , etc. will also be available.

* Research carried out in association with the IAEA under Research Contract No. BGD/6360.

DESCRIPTION

The whole system has been shown in the block diagram (Annexure 2). Except the reactor console the remaining hardware system has been provided under this contract by the International Atomic Energy Agency and the Institute of Nuclear Science and Technology at AERE Savar. The incoming signals from the reactor console will be fed to the 8*4 channel multiplexer through a high impedance input buffer. The high input impedance will keep the reactor control system undisturbed from that of the remaining system. The multiplexer will monitor the input at regular intervals and put the information into parallel port of the computer. The personal computer contains video monitor and printer for display and instantaneous records. Diskettes will be used for permanent storage of information.

As seen from the system block diagram, the hardware item include interfacing systems and personal computer with accessories. The interfacing system has been designed and fabricated using locally available components as far as practicable. The computer with accessories and some other components have been provided under this contract by the International Atomic Energy Agency.

INTERFACING SYSTEM

The interfacing system consists of the following :

i) high impedance input buffer
ii) 8 units of 4- channel multiplexer
iii) A/D converter
iv) parallel port

the brief description of which follows :

i) High impedance input buffer

The different channels of the reactor console have been identified as shown in the Annexure-1. To isolate the reactor control system from loading and other disturbances the interfacing system is being provided with high input impedance buffer unit (of the order of 10^{12} ohms). These high input impedance for each channel will provide quite reasonable isolation of reactor electronics for such disturbances. The circuit diagram and other details are shown in C.

ii) A/D converter

Most measurements of dynamic variables are provided by devices that provide information in forms of analog electrical signals. To interface these signals with a computer or digital logic circuit, it is necessary first to perform an analog to digital conversion. Conversion must be such that a unique known relationship exists between analog and digital signals. The transfer function of an ADC is such that some analog voltage is provided as input and the conversion gives a binary number which, later on is used by computer.

The bread board design and testing of A/D converter has been done in the laboratory before connecting with the reactor. AD7581 which is an 8-channel A/D converter has been used. The details are shown in C

iii) Multiplexer

The computer in a data acquisition system periodically samples the values of variables and allows sample variables from many sources to be input to the computer with appropriate programming. If no control function is associated with the system the computer then outputs the information for permanent storage in hard disk/diskettes and display through video monitors and printers

The basic element in multiplexing is essentially a solid state switch. It takes decoded address signal and selects the data from the selected channel by closing a switch connected to that analog input line. The actual switch elements are usually Field Effect Transistor(FET) which have an ON resistance of a few hundreds ohms and an OFF resistance of hundreds to thousands of megohms. 8 units of 4-channel multiplexer have been used in the system. IC chips 4052 have been used for the same.

In general, data acquisition modules accept number of analog inputs from monitored variables called channels as either differential voltage signals(2 wire) or single ended voltage signals (referred to ground) . The computer can then select any one of the channels under programme control for input of data from that channel. Single ended configuration has been used for this system. Details have been shown in C.

iv) Address decoder

This part of the data acquisition system accepts an input from the computer via address lines which serve to select a particular analog channel to be sampled or other components of the interfacing unit. The details are shown in C.

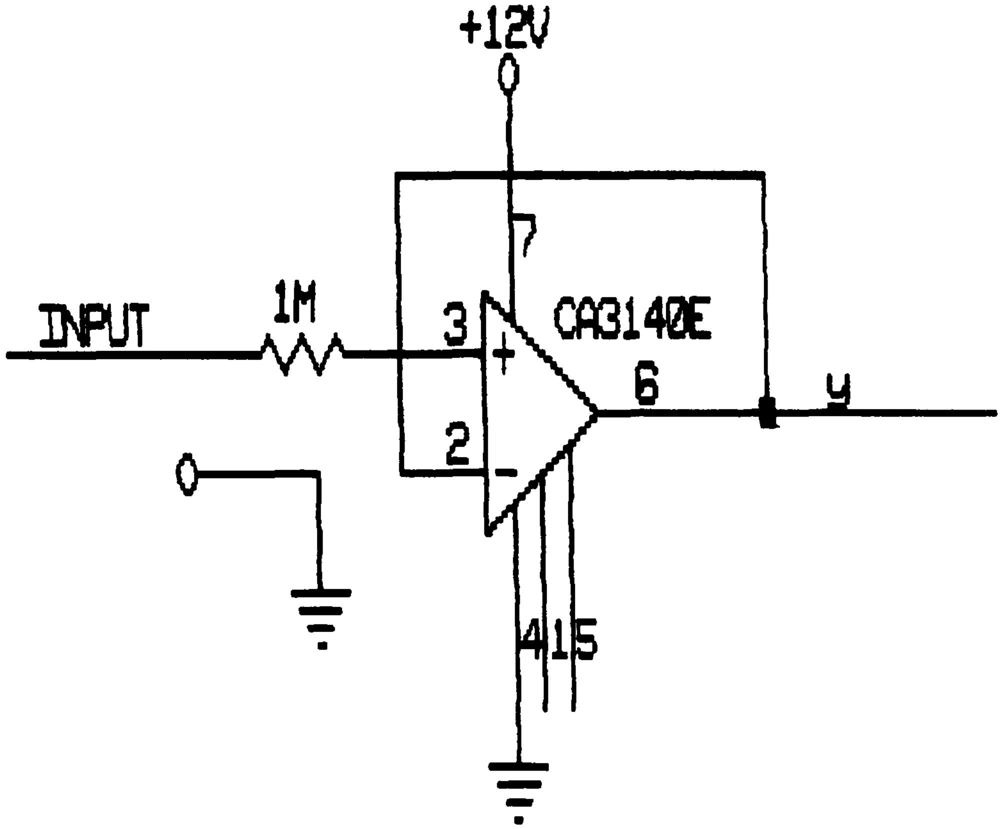
v) Software

The programming for the interface system between the reactor instrumentation and the PC is a time consuming work. The PC being used for this work is IBM compatible using MS DOS 4.01 Version. GW BASIC was initially chosen as the programming language. However, basic software has later been changed and modified to use TURBO BASIC as the programming language. TURBO BASIC has been adopted considering its speed and efficiency. It is to be mentioned here that the Data Acquisition System does not propose any control function and the programme has been developed for the system to monitor, display and store reactor parameters and information. The main portion of the programme have been shown in C.

RESULTS OBTAINED

Input buffer unit

The input buffer circuits developed for use in the Data Acquisition System (DAS) is shown below.



Buffer Unit

The essential feature of the operational amplifier used in this type of circuits is that very little current is drawn by the input. The input current is negligibly small indicating that the input impedance is infinite. This feature ensures that the signals input to the operational amplifier will not affect or load the input source. The input impedance of the circuits using these components is about 10^{12} ohms.

A/D converter

A/D converter AD7581 has been used as shown in the figure **DAS:A/D Converter.** The chip accepts eight analog inputs and sequentially converts each input into eight bit binary words using successive approximation technique. The conversion results are stored in an 8*8 bit dual port RAM. The device runs from a 1 MHz clock. The converter require only a -10V reference and a +5V supply. Start up logic is included on the device to establish the correct sequence.

Multiplexer

The CD4052 dual 4-channel multiplexer has been used in the system. These are bi-directional analog switches allowing any analog input to be used as an output and visa-versa.The switches have low "ON" resistance and low "OFF" leakage current.The devices have an enable input which when high disables all switches to their "OFF" state. The circuit diagram is shown in **figure DAS-1.**

Decoder

The details are shown in **figure DAS: Decoder.**

The circuit diagrams of the complete system have been enclosed as DAS-1, DAS-2, DAS-3, DAS-4 and DAS-5.

Software

Turbo Basic has been used as the programming language. The main programme has been divided into different sections.

a) Data Logging

This part has been developed to record the current operational information of the reactor. The actual mode shows the parameters of the reactor when the reactor is in operation.

The demonstration mode shows different channels of the reactor under simulated conditions.

b) Data Review

This part of the programme enables reviewing of the data already stored in the memory. The recorded data of the reactor can be retrieved and used as and when required using this mode.

c) Printer Routine

This part of the programme enables the use of the printer for making hard copies of information already logged.

d)Graphic demonstration

Using this mode different system of the reactor may be displayed and explained to the visitors and trainees. Systems like horizontal and vertical sections of the reactor, core plan showing the fuel and control rod arrangement, sections of the fuel element, and control rod movement have been included.

CONCLUSIONS

Like any reactor, TRIGA reactor is also under strict regulatory supervision and control of the National Nuclear Safety Committee(NNSC) of BAEC. Therefore, any modification of the reactor system, addition of any equipment/system , etc to the reactor requires clearance from NNSC. Preliminary clearance was thus obtained to test the System using only the calibrating channels of the reactor. These calibrating channels provide simulated signals to all the measuring instruments of the reactor representing actual operational state. DAS was thus tested with the reactor console and was found not to interfere with the reactor electronics by loading the measuring and control instrumentation in any way. On this basis final clearance has been obtained and the system has been connected with the reactor console. With very little modifications in both hardware and software, the data logging system is now running successfully.

ACKNOWLEDGEMENTS

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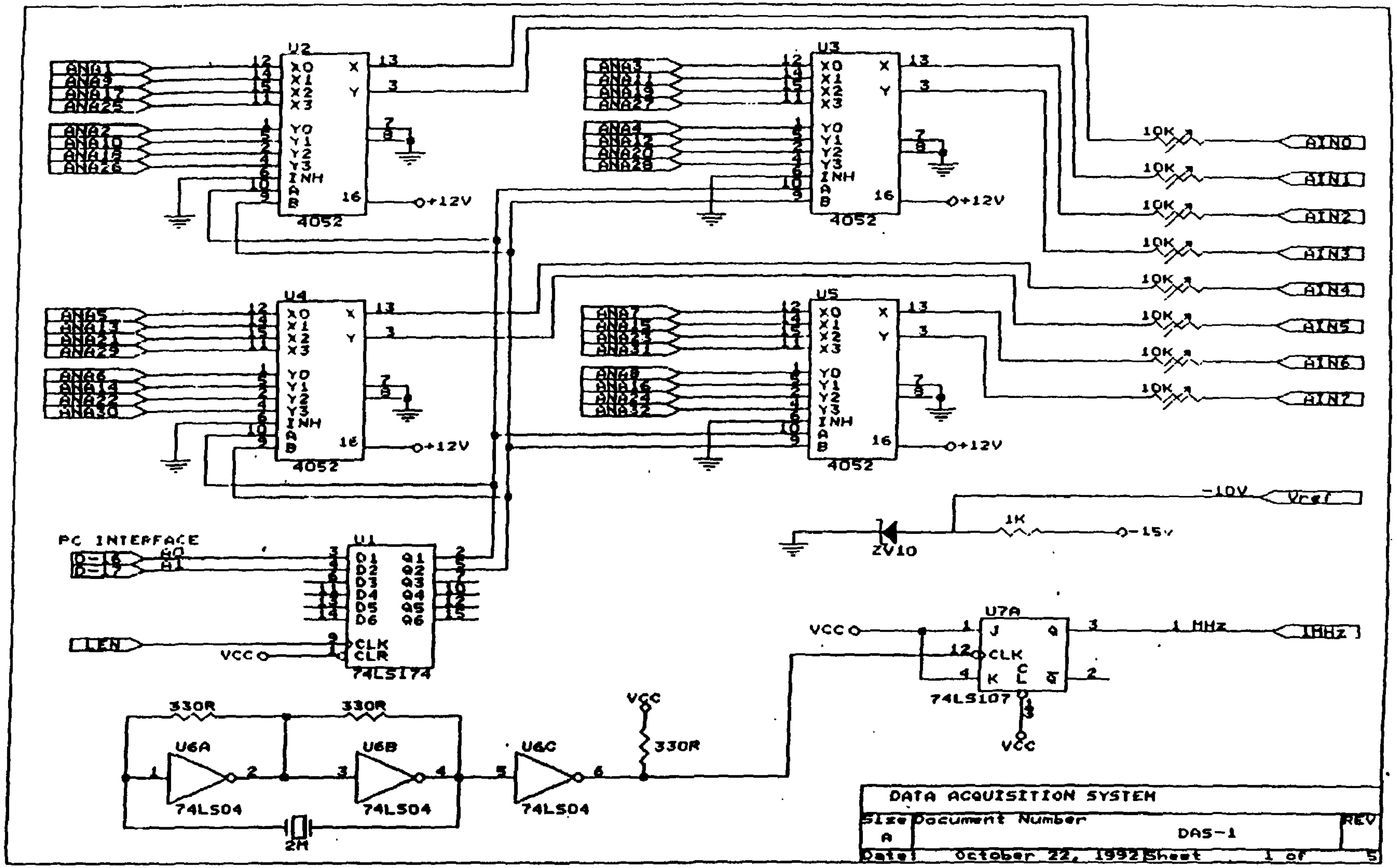
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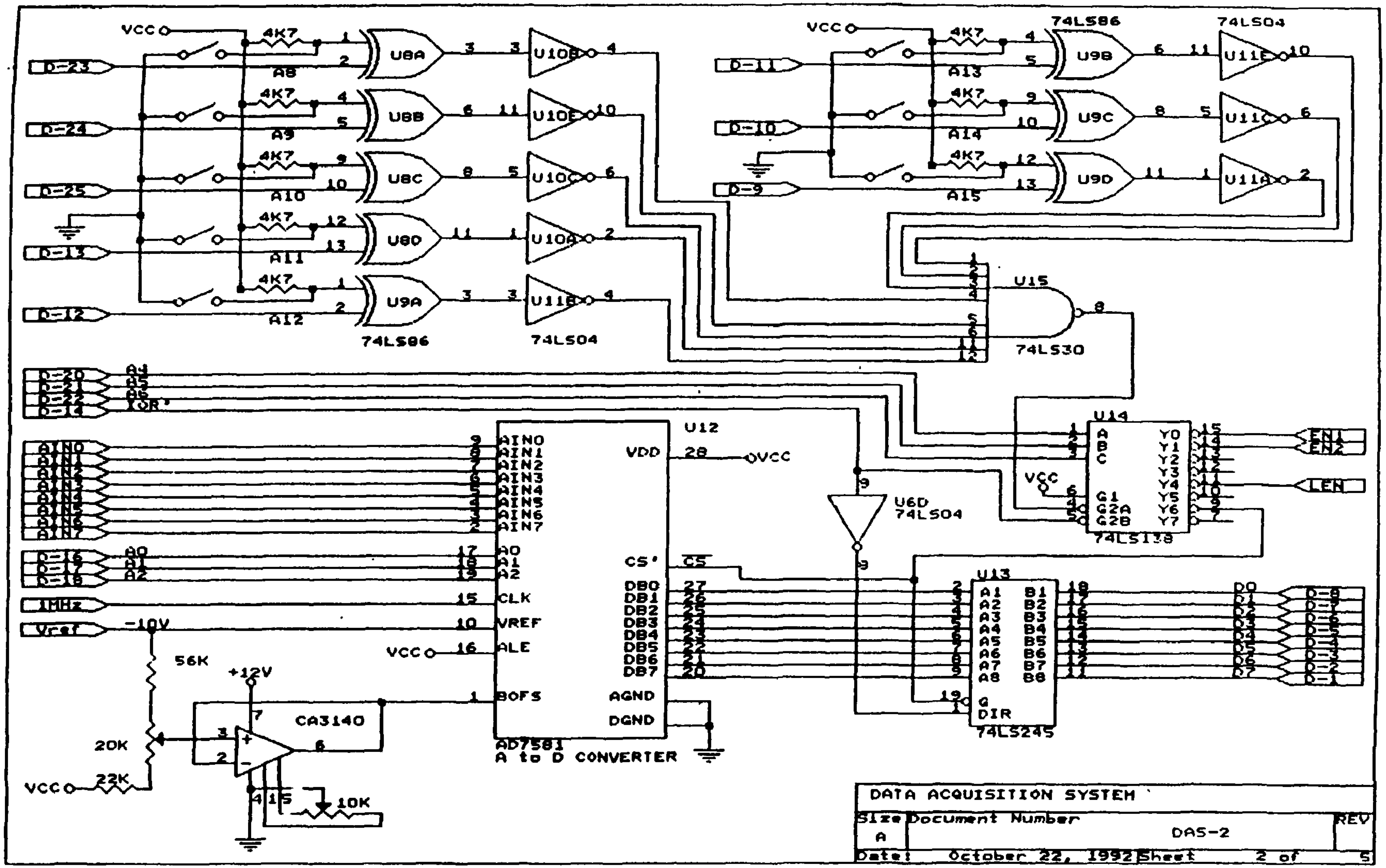

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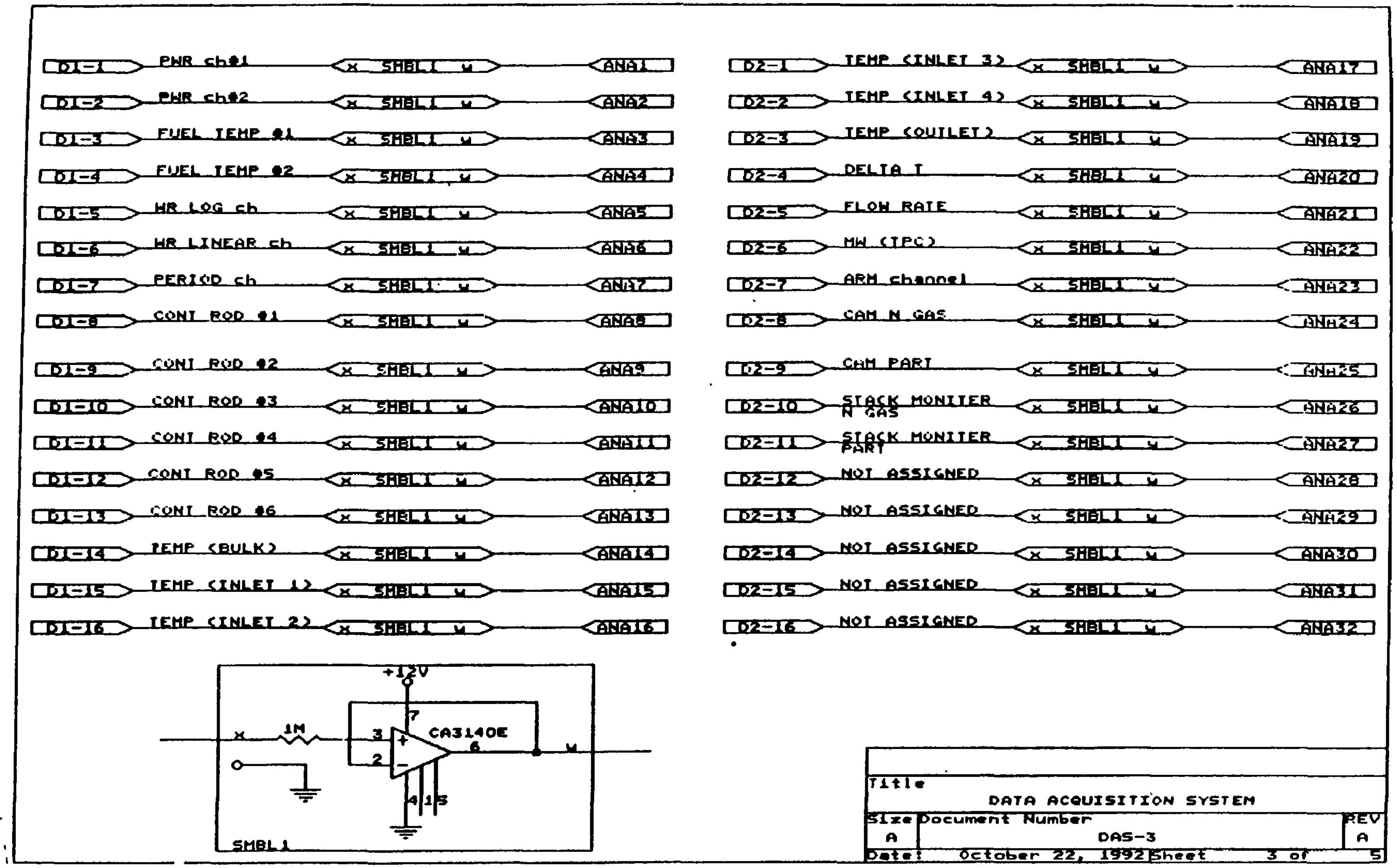
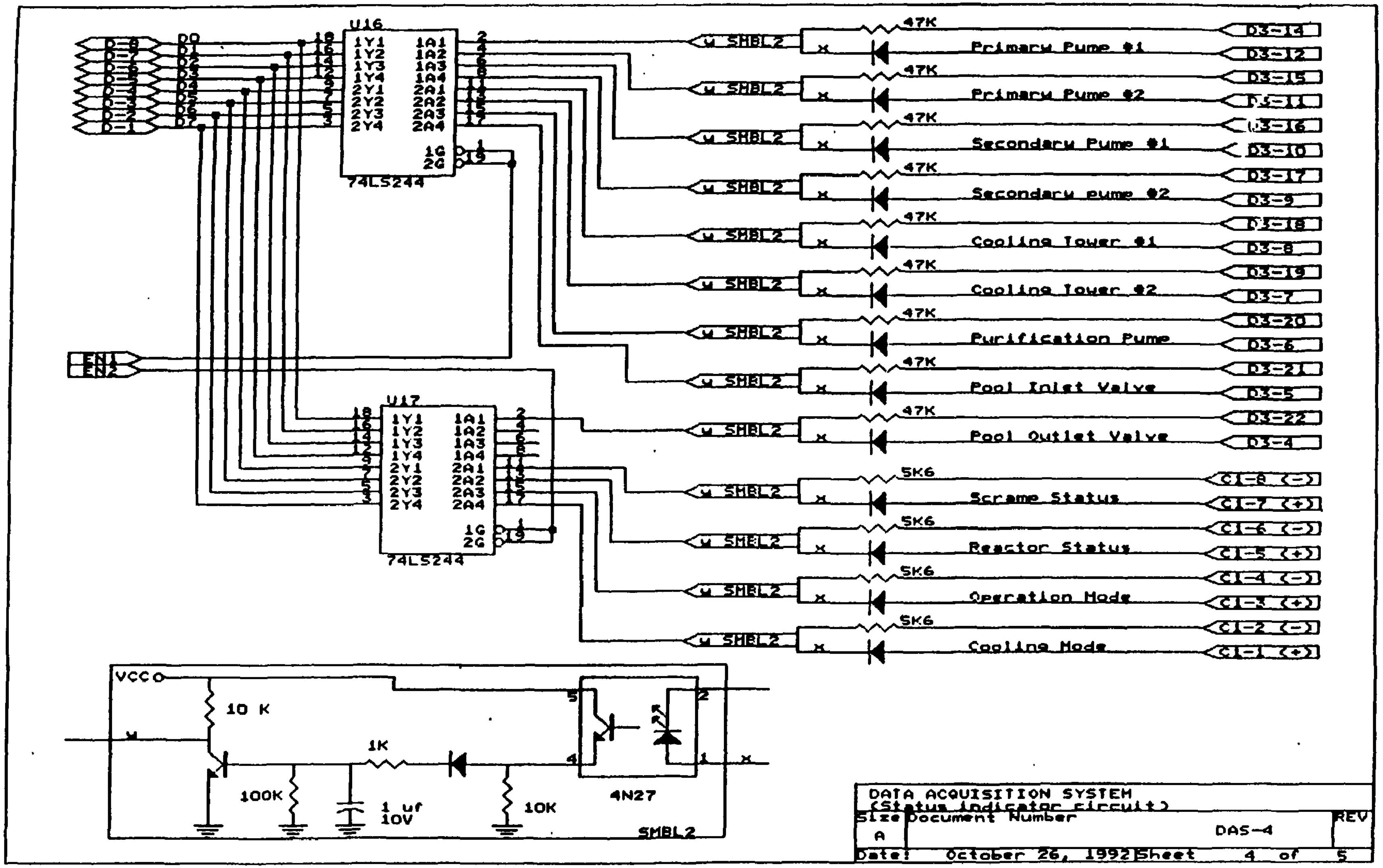


CIRCUIT DIAGRAMS (Cont.)



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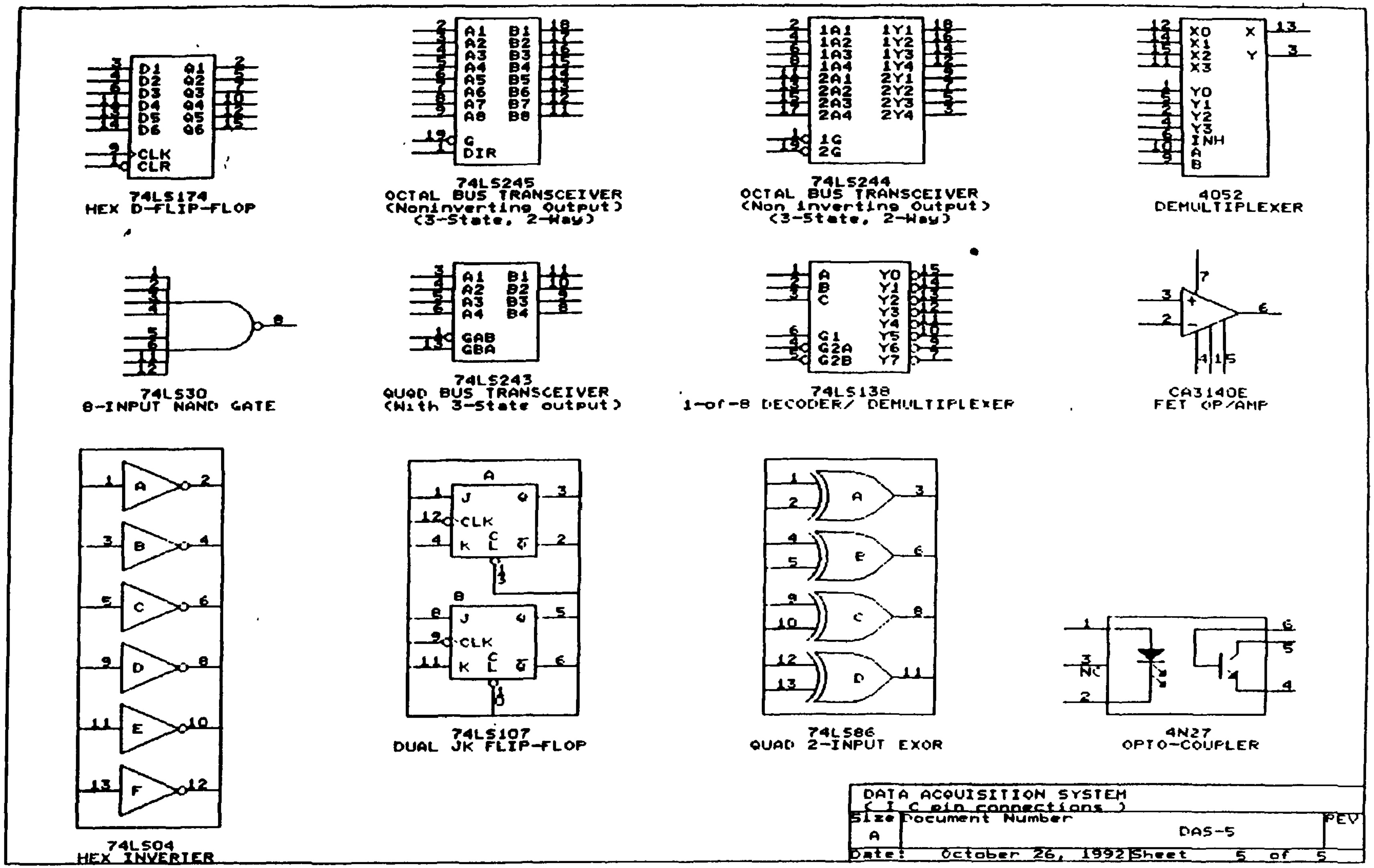

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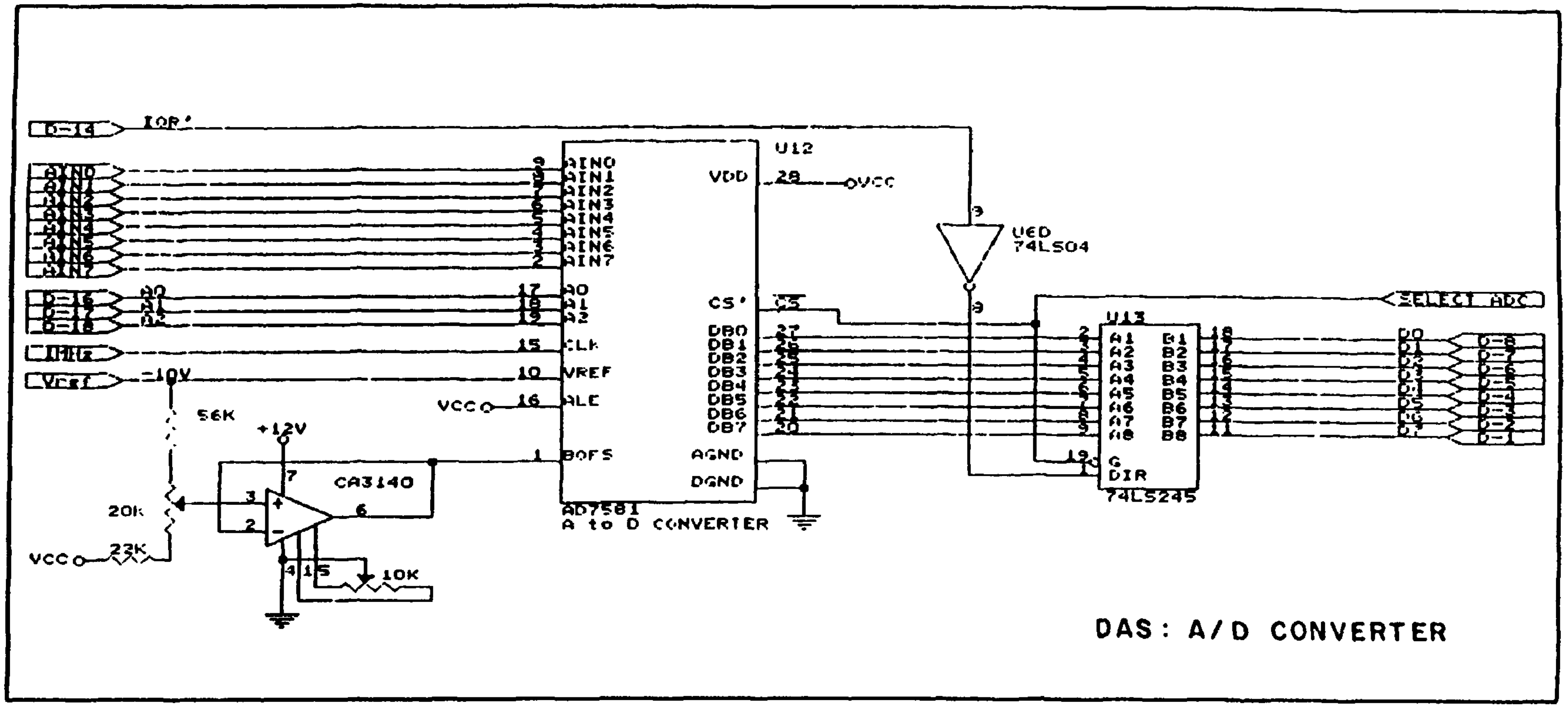

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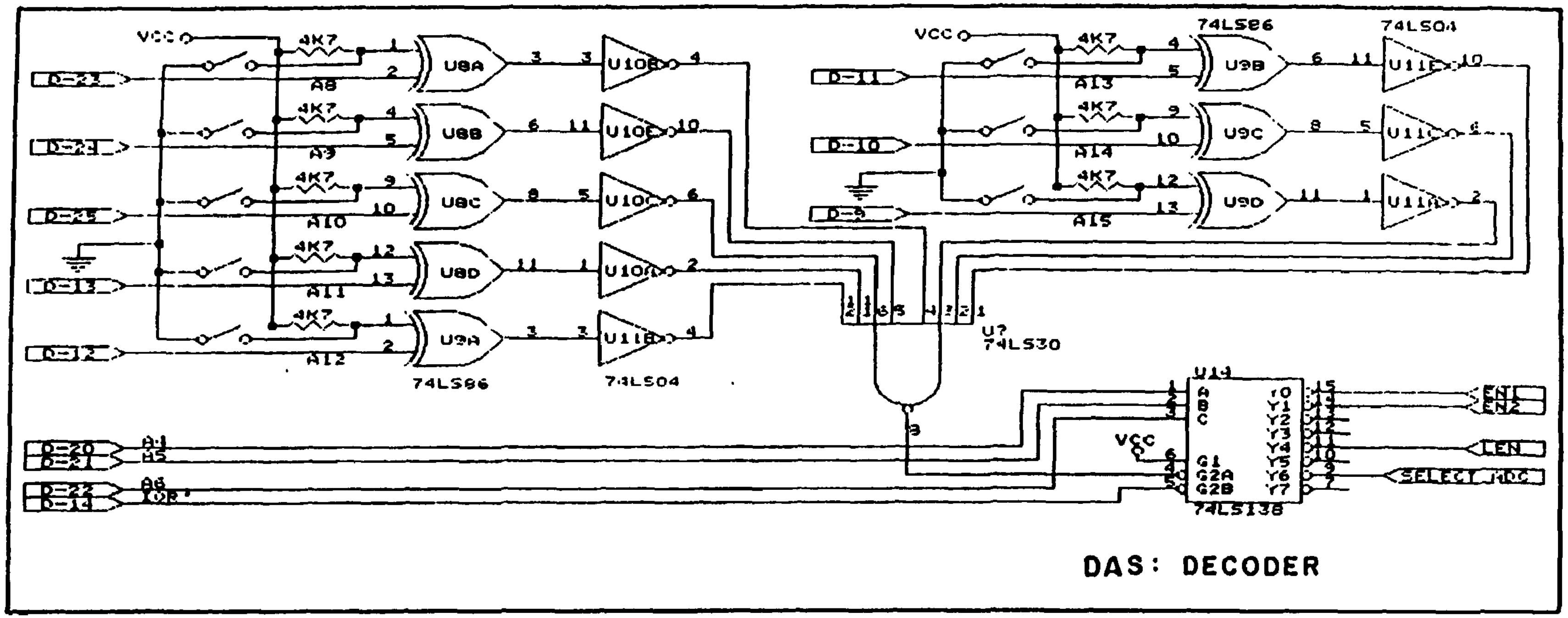


DAS: DECODER

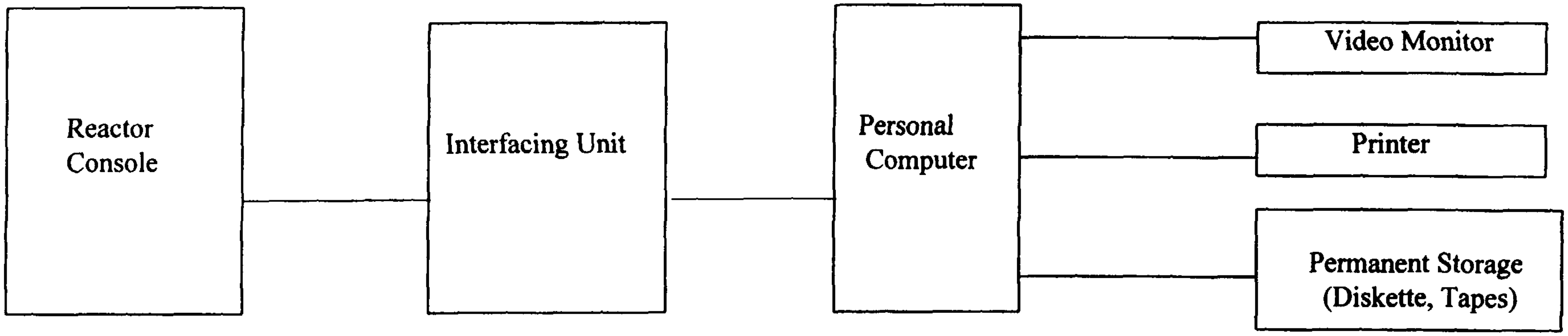
Annexure I

FULL SCALE DEFLECTION (FSD) CURRENTS AND VOLTAGES OF CONSOLE READOUT METERS

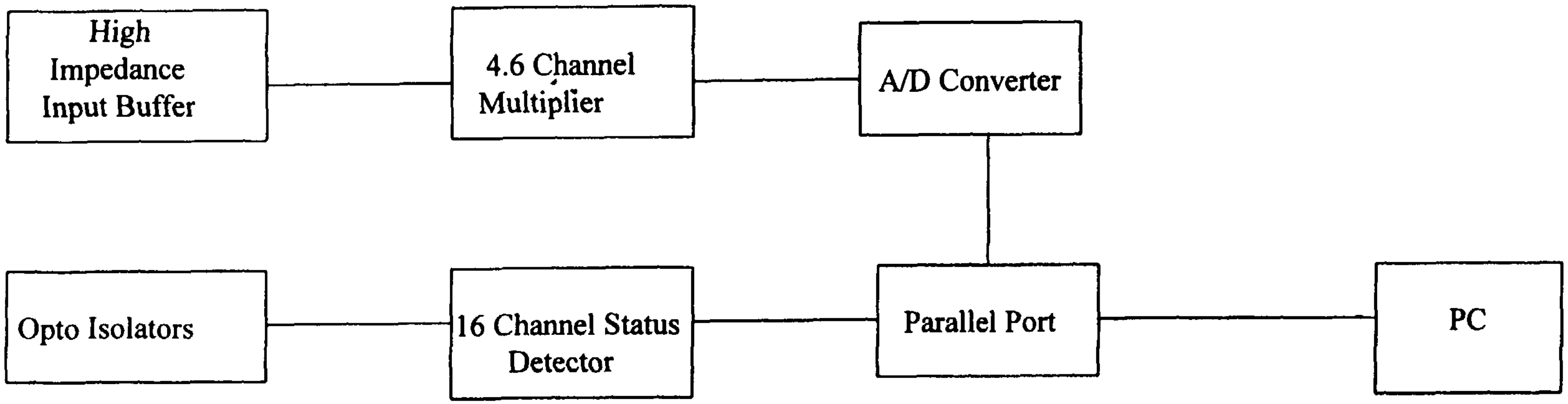
Sl Nos	Channel	Scale	F S D mA	V
1.	Power Channel 1	110 % of 3 MW	1	11
2.	Power Channel 2	110 % of 3 MW	1	11
3.	Fuel Temp. 1	1000°C	1	10
4.	Fuel Temp. 2	1000°C	1	10
5.	WR Log Channel	10^{-8} to 110 %	1	10
6.	WR Linear Channel	110 %	1	10
7.	Period Channel	(-)30 -∞ (+)3 sec	1	10
8.	Control Rod Pos.	000 to 999	-	-
9.	Bulk Water Temp.	60°C	1	10
10.	Temp Core Inlet 1	50°F to 120°F	1	10
11.	Temp Core Inlet 2	50°F to 120°F	1	10
12.	Temp Core Inlet 3	50°F to 120°F	1	10
13.	Temp Core Inlet 4	50°F to 120°F	1	10
14.	Core Outlet Temp	50°F to 120°F	1	10
15.	Del-T	20°F	1	10
16.	Flow	0 to 4000 GPM	1	10
17.	MW (TPC)	5 MW	1	-
18.	ARM Channels	10^{-1} to 10^{4} mR/h	1	100mV
19.	Stack Monitor			
	a) Noble Gas	10 to 10^{6} CPM	5 mV	
	b) Particulate	10 to 10^{6} CPM	5 mV	

Annexure II

DATA ACQUISITION SYSTEM FOR THE TRIGA REACTOR



SYSTEM BLOCK DIAGRAM



INTERFACING UNIT

XA9846764

NUMERICAL FLUID FLOW AND HEAT TRANSFER CALCULATION (HEATHYD CODE)*

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Abstract

Under normal operating condition, heat transfer from the fuel plate to the coolant occurs by convection phenomena. In case of force convection, the rate of the heat being transferred is proportional to the temperature difference between plate surface and coolant temperature. In most research reactors of MTR type the coolant flow is turbulent that results in an enhancement of the heat transfer. The matter of fluid flow analysis for the reactor core is the determination of flow rate and pressure losses resulting mainly from irreversible process of friction and velocity and height change.
Hydraulic calculation is made using iterative method. The results of this calculation are provided for heat transfer analysis by the HEATHYD code. The physical and mathematical model of the heat transfer of HEATHYD includes the equation for thermal conduction and Newton's law of cooling.
After completing experimental verification of the HEATHYD model, it was applied to calculate the thermohydraulics of a plate type MTR. It was assumed that the core consists of 48 fuel elements generating a total power of 30 MW. The coolant flow and heat transfer for the hot channel was analyzed. In the hot channel case, the saturation temperature has been partly exceeded by the clad surface temperature with the result of subcooled boiling. In the case of the average channel, the clad temperature does not reach the saturation temperature.
The results of calculations show that the safety margin to flow instability represents the limiting parameter regarding safe design and operation.

I. STEADY STATE CORE THERMOHYDRAULICS

I. 1 Fluid Flow in channels

Under normal operating condition, heat transfer from the fuel plate to the coolant occurs by convection phenomena. In case of forced convection, the rate of the heat being trasferred is proportional to the temperature difference between plate surface and coolant temperature. The value of the proportionality factor, convective heat transfer coefficient, depends on the flow conditions including coolant physical properties. In most research reactors of MTR type the coolant flow is turbullant that results in an enhancement of the heat transfer.
The matter of fluid flow analysis for the reactor core is the determination of flow rate and pressure losses resulting mainly from irreversible process of friction and velocity and height change.

* Research carried out in association with the IAEA under Research Agreement No. GFR/6478.

Pressure Drop of coolant

The conservation law for the main flow determined the velocitiy change resulting from the cross section change. The corresponding pressure loss is calculated by applying Bernoullis equation. By neglecting the pool water velocity, the pressure drop is expressed as follows:

$$\Delta P_d = \delta u^2 / 2$$

The total pressure drop through a fuel element consists of losses in upper and lower boxes of the fuel element and the friction pressure drop on the surface of fuel plates. These losses are in addition to the losses due to the change in height between core inlet and outlet. Main part of the pressure drop results from the fricition loss in the cooling channels. Two different approaches for the calculation of friction losses are being outlined. The most plate-type research reactors are designed for subcooled core flow under normal operation condition. The friction pressure loss is calculated from the following formula /1/:

$$\Delta P_f = 4 f (L_c / D_e) (\delta u^2 / 2)$$

For turbulent flow existing in the coolant channels, the friction factor f, is determined by using /2/:

a) Blasius correlation:

$$f = 0.0791 / R_e^{0.25} \qquad \text{for} \quad 5000 < R_e < 51094$$

b) Colebrook-white correlation:

$$f = 0.651 / Re^{0.0825}$$

Contraction and expansion of flow paths of the coolant while entering and leaving cooling channels result in pressure losses that depend on flow rate, channel geometry and dimensions of the end boxes.

For a given incompressible coolant, the velocities before and after channel exit or inlet can be related by:

$$U_o / U = A_c / A_o$$

The entrance and exit pressure losses can be calculated from the followng expression:

$$\Delta P_{en} = K_{en} (\delta u^2 / 2)$$

$$\Delta P_{ex} = K_{ex} (\delta u^2 / 2)$$

Where:

$$K_{en} = (1/ß - 1)^2 + 0.05 = 0.63 + 0.37 (A_c / A_o)^2$$

and, Re defines the Reynolds number according to:

$$R_e = \frac{\delta \, . \, V \, . \, D}{\mu}$$

Where

δ = Density of coolant

V = velocity of coolant

D = hydraulic diameter

μ = kinematic vislosity

One additional contribution to the pressure loss (or change) results from the change of height between core inlet and outlet and is given by the product of specific weight of the coolant and change in height. For downward flow its contribution is negative.

Iterative Hydraulic Calculation

Because of interrelation of coolant velocity and pressure loss, hydraulic calculation is made using iterative method. Starting with the average coolant velocity resulting from the total mass flow and channel area, the pressure drops due to friction, velocity change and contraction is determined. Using individual channel dimensions, the total core flow rate is split into channels flow rates,whereupon, the individual coolant velocities are calculated and used in return to determine the pressure drop of the coolant at each channel. After calculation of the total pressure loss the coolant velocity is subsequently determined applying the correlation between velocity and channels pressure drop. At the end of each iteration step a convergency test regarding channel coolant velocity and total mass flow is performed. In case of high convergency being defined by the maximum number of the iteration steps, hydraulic calculation is terminated and results of the final iteration step including flow distribution and pressure drops are compiled as main hydraulic parameters. Fcr the local pressure values the saturation temperature is finally calculated according to /3/ :

$$T_S = -167 \ln [1.05 - \ln (226P) / 167 / 0.065]$$

The results of the hydraulic calculations are provided for heat transfer analysis in the second part of the HEATHYD code. To take into account the variation of the parameter like viscosity and density with the temperature, the hydraulic and heat transfer part are linked via outer iteration that will be discusses subsequent to the description of the heat transfer routine.

I. 2 Heat Transfer Model of HEATHYD

Heat being generated in the fuel plates is mainly transfered by heat conduction across the fuel plates and removed by the forced convection of the coolant. The physical and mathematical model of the heat transfer of HEATHYD includes the equation for thermal conduction and Newton's law of cooling. The equation of heat conduction for slab geometry of the fuel plates is expressed in the following form and numericaly solved using iteration method.

$$q = - k \cdot A \cdot \frac{dT}{dX}$$

In this equation, q represents the heat flux (rate of energy being released from the clad surface A) and k is refered to as the thermal conductivity of the fuel and clad material. For the determination of the clad surface temperature, the rate of the heat transfer to the coolant is calculated applying Newtons cooling formula of the form:

$$q = A \cdot h\,(T_S - T_C)$$

In this equation, the convective heat transfer coefficient depending on the physical properties and flow condition (hydraulics) of the coolant. Ts and Tc have been used as cladsurface and coolant temperature respectively. The variation of hydraulic parameters, like velocity or channel flow rate requires a linkage of fluid flow to the heat transfer part of HEATHYD. This feedback mechanism as depicted in Fig. 1 is mathematically modelled through outer iteration. By this procedure, the results of the heat transfer calculations are provided to the hydraulic part and vice versa. T_S, T_C, velocity and pressure are the parameters that are mainly exchanged between the two parts.

The calculation of heat transfer coefficients takes place in the subroutine ALPHA using the following relationship:

$$h = \frac{k}{D} Nu$$

Nu ist referred to the Nusselt number as a dimensionless parameter characterizing both the physical properties of the coolant and dynamic characteristics of the flow. For the turbulant flow in the cooling channels of the core, different empirical correlations for Nu are available. All correlations are expressed as a function of Reynolds number R_e ,specifying hydraulic condition and prandtl number Pr, for physical properties of the coolant.

I. 3 Correlations for Heat Transfer coefficient

Convective one phase heat-transfer model of HEATHYD includes three different correlations for h. It is chosen by an option parameter named WEG.

1) Hausen correlation /3/:

$$Nu = 0.037 \left[1 + 0.333 \left(\frac{D}{Lt}\right)^{2/3} \right] (Re^{0.75} - 125)\, Pr^{0.42} \cdot \left(\frac{\mu c}{\mu s}\right)^{0,14}$$

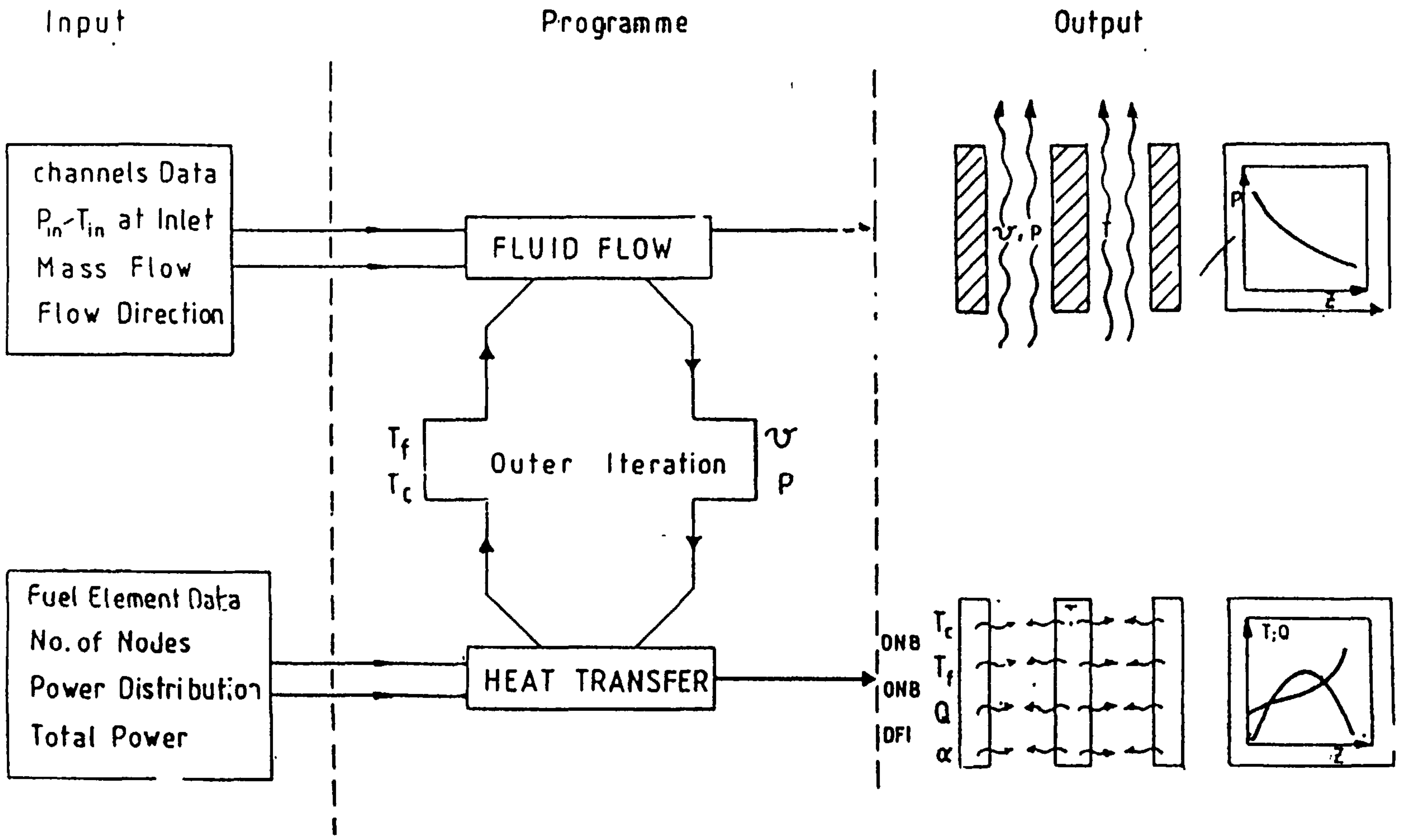

Fig. 1: Flow diagram of the HEATHYD thermohydraulic code

where are:

D	=	hydraulic diameter of the cooling channel
L_t	=	total length of fuel plate including the lenght of the meat free zone
$\mu_{c,s}$	=	viscosity at the coolant and surface temperature
Re	=	Reynolds number
Pr	=	$u \cdot c_p/k$
k, c_p	=	thermal conductivity and specific heat of the coolant

2) Sieder-Tate correlations /4/:

$$Nu=0.037Re^{0.8} \cdot Pr^{0.33} \left(\frac{\mu c}{\mu s}\right)^{0.14}$$

3) Dittus-Bollter correlation /5/:

$$Nu = 0.023\, Re^{0.8} \cdot Pr^{0.4}$$

I. 4 Physical Properties of Water

Physical properties as a function of the coolant temperature are assumed to be polynomial of up to 6 orders as follows:

For the specific heat

$$C_p = C_o + C_1 T + C_2 T^2 + \dots\dots + C_6 T^6$$

For the conductivity

$$k = k_o + k_1 T + k_2 T^2 + \dots\dots + k_6 T^6$$

For the dynamic viscosity

$$\mu = \mu_o + \mu_1 T + \mu_2 T^2 + \dots\dots + \mu_6 T^6$$

For the density

$$R = 1002.9 - 0.15838T - 0.00284\, T^2$$

I. 5 Onset of Nucleate Boiling

Nucleate boiling occurs at a surface temperature above T_{sat} by a quantity $T_s - T_{sat}$. Under ONB conditions, the clad surface temperature over which nucleate boiling occurs, is determined as a function of local coolant pressure (P^z) and surface heat flux, by the correlation developed by Bergles and Rohsenow /5/:

$$T_s = T_{sat} + 5/9\, (9.23\, q/P_z{}^{1.156})\, P_z{}^{0.0234}/2.16)$$

Rearranging above equation, the heat flux at ONB is given by:

$$q_{ONB} = P_z^{1.156} [\,1.8\,(T_s - T_{sat})\,]/\,(0.46\;P_z^{0.0234})\;/9.23$$

This correlation is applicable down to the low pressure rangecharacterizing the operating conditions of research and test reactors.

The actual axial location at which ONB will ocur depends upon the axial heat flux distribution, coolant velocity and pressure drop along the channel. For the simplicity, heat flux at ONB is calculated conservatively by using the worst combination of parameters i.e. ONB occurs at the channel exit with the lowest pressure, saturation temperature and highest coolant temperature rise. This correlation is used to determine the heat flux resulting in nucleate boiling. In case of reaching or exeeding heat flux for ONB, the temperature calculation is performed by using separate correlation given as follows:

$$T_s - T_{sat} = (43.1 - 0.11\,T_{sat})\,q^{0.3}$$

where T_{sat} and T_S are saturation and surface temperature (°C) and q is referring to the heat flux (MW/m^2). To determine the margin to nucleate boiling, the local heat fluxes are compared with q at ONB and the ratios are printed out by HEATHYD as a function of axial points and channels.

II. MODEL OF CRITICAL COOLING

II. 1)Flow instability

The phenomenon of flow instability is a result of interaction between pressure drop and coolant flow in heated channels. For the heated channels, the pressure drop as a function of mass flow deviates from m^2-dependency at low flow rates and shows a minimum. Before the minimum, any decrease in the flow rate results in an increase of the pressure drop with the consequence of low local pressure and saturation temperature. The minimum in the pressure drop/mass flow kurve depends on flow characteristics and heat flux. The determination of critical heat flux at the onset of flow instability has been experimentally inverstigated by whittle and Forgan /6/ for the coolant channels conditions exiting in MTR. They measured the mass flow, exit temperature and pressure drop corresponding to minima in the pressure drop -vs- flow rate kurve for subcooled water flowing (upward and downward) in norrow heated channels (width 2.54 cm, thickness 0.14 to 0.32 cm, and length 40 to 61 cm) under the followong conditions:

$$1.2 \le P_{exit} \le 1.7 \text{ bar}$$

$$83 \le L_H / L_D \le 190$$

where

L_H = heated length of channel

L_D = heated equivalent diameter of the channel

= 4 (channel flow area / channel heated perimeter)

= $4\, t_w W / (t_w + W_h)$

Based on these measurements the following correlation was proposed:

$$R = (T_{out} - T_{in}) / (T_{sat} - T_{in})$$

where

$$R = 1 / (1 + n\, D_H / L_H)$$

and n is bubble detatchment parameter.

Values of n suggested by different authors are ranging from 12 to 37. A value of n = 25 was determined as best fit for Whittle and Forgan data. For different values of n, R is plotted as a function of L_H / D_H in Fig. 2. Safety margin S_f to the onset of flow instability is defined as follows:

$$S_f = R\,(T_{sat} - T_{in}) / (T_{out} - T_{in})$$

where R is referring to as a parameter depending on design and T_{out} ist the coolant outlet temperature. According to this expression an increase of heat production or decrease of inlet subcooling (T_{sat} - T_{in}) aggravates flow instability and lessen the safety margin. Using the expression for S_f, the HEATHYD code calculates the safety margin to the onset of flow instability which is printed out for each cooling channel. Heat flux corresponding to flow instability can be calculated from coolant temperature rise i.e. R (T_{sat} - T_{in}) , specific heat, and flow rate. The average heat flux at onset of flow instability is expressed in term of velocity, channel geometry, temperature, and fluid properties /7/:

$$q_{OFI} = 0.05 \quad [\, R\delta\, C_p \;\; t_w\, (W\, t_w / W_H L_H)\, U\, (T_{sat} - T_{in})\,]$$

In this correlation the effect of channel entrance loss, which is a stabilizing factor for the system, is not included. The amount of the heat transfer at OFI depends on pressure through saturation temperature, T_{sat} . Since pressure drop characteristics are not required, the accuracy of the prediction does not depend on two phase correlations (subcooled void fraction, pressure drop, and heat transfer coefficient). All two phase effects are included in parameter „n“, and flow instability is intimately related to pressure drop. The pressure drop depends on the local water quality, which follows from the axial heat distribution.

II.2 Departure from Nueleate Boiling (DNB)

As a result of the heat flux increase, small bubbles are formed and collapsed after leaving the fuel plate surface. Because of agitation process, the heat transfer to coolant becomes hig-

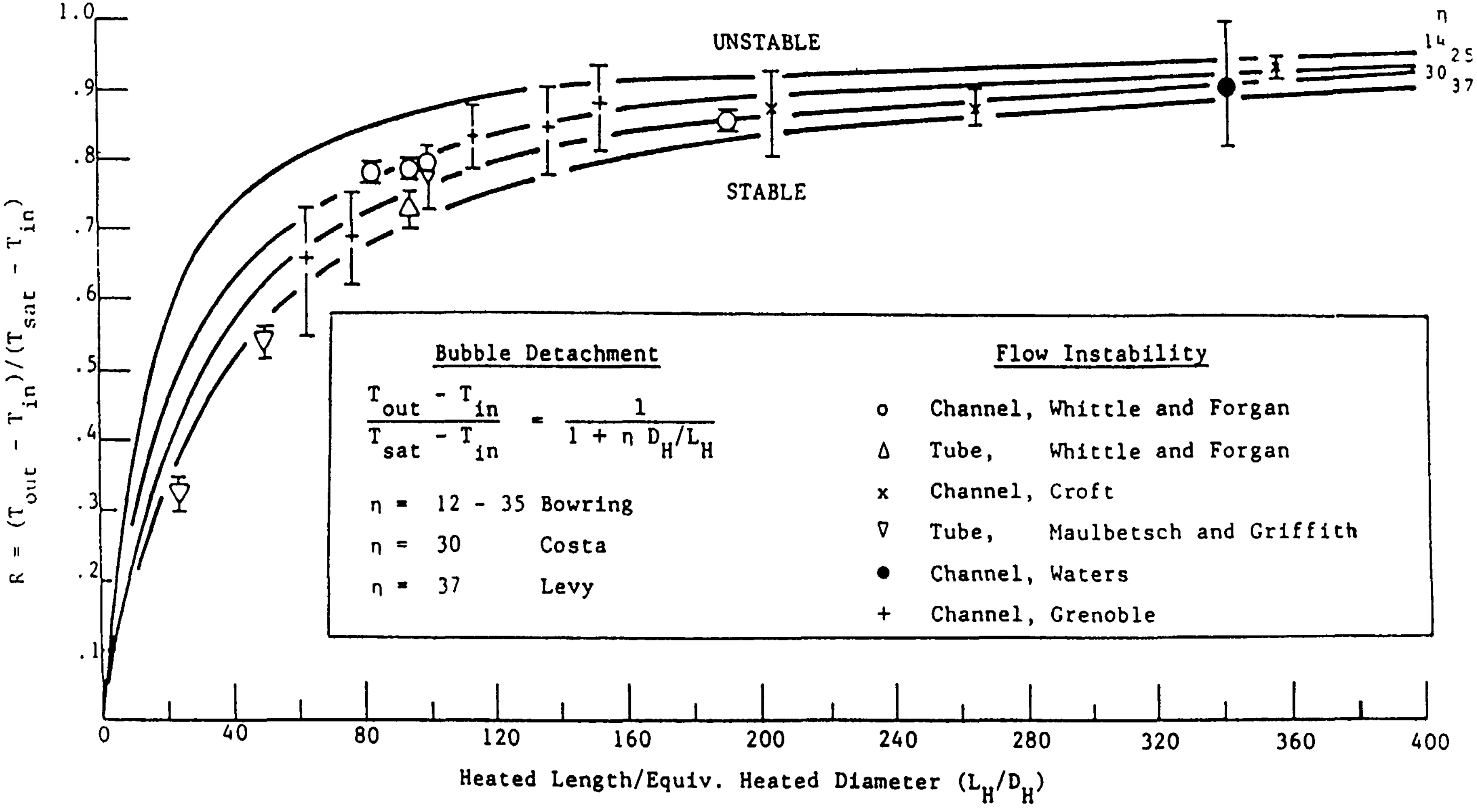

Fig. 2: Correlation for flow instability and bubble detachment/7/

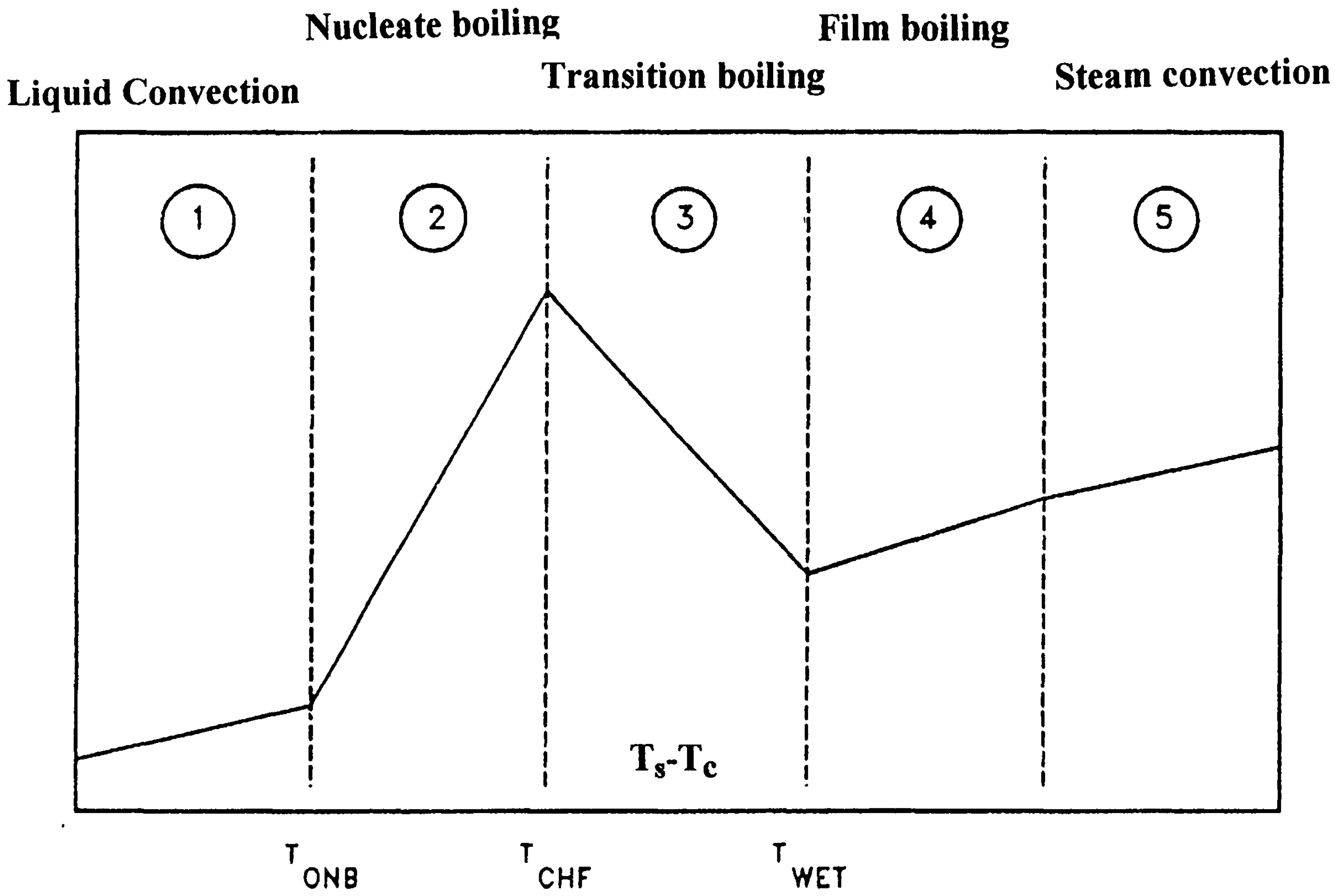

Fig. 3: Heat flux versus temperature difference for different flow regime

her. Due to further increase of power the bubble formation rate becomes higher than the collapsing rate so that they begin to clump in the region adjacent to the heating surface which affects the heat transfer rate. In case of covering of plate surface, the thermal linkage of the heating surface to the coolant is interupted. Under this condition the rate of heat transfer drops as a function of temperature difference between surface and fluid temperature (film boiling condition).

The heat transfer coefficient in this regime is by orders of magnitude lower than in the corresponding region before critical heat flux so that a rapid rise of the surface temperature takes place. The variation of the heat transfer rate to coolant with the temperature difference has been depicted in Fig.3. According to this figure, if the surface heat flux exceeds a certain magnitude an unstable transition from Nucleate Boiling to Film Boiling occurs. For this reason, the determination of the critical heat flux at which Departure from Nucleate Boiling (DNB) happens is important. For the safe and optimum operation, the actual heat flux should not exceed the critical value. For this aim the minimum ratio of critical heat flux to that existing at the fuel plate is defined as the DNB ratio. The critical heat flux depends on surface condition, physical properties and flow conditions. Most of the DNB correlations have been developed for the heated tubes. Two DNB correlations for the round tubes (Labunstov) and narrow rectangular channels (Mirshak) are applicable in low pressure range. Both correlations have been involved in the HEATHYD code and are optionally used /6, 7/

Labunstov Correlation :

The Labunstov correlation is based on experimental data from several sources. These data cover a wide range of velocity and pressure, but all have positive subcooling at the channel exit. Labunstov observed that the burnout heat flux varies with the pressure, coolant velocity and the magnitude of subcooling at the exit, and that these fluxes are virtually independent of the length, diameter, and configuration of the cooling channel. The effect of the channel dimensions becomes pronounced only for diameters less than 2 mm. According to Labunstov critical heat flux is given by :

$$q_c = 145.4\ G(p)\ [\ 1+2.5\ U^2/G(p)\ ]^{1/4}\ .\ (1+15.1\ Cp.\Delta Tsub\ /\lambda\ P^{1/2})$$

where

$$G(p) = 0.9953\ p^{0.3333}\ (1 - P/Pc\ )^{4/3}$$

and

$$\Delta t_{sub} = T_{sat} - T_{in} - \Delta T_c$$

The above relation is valid within the parameter ranges given below

Steam quality :	negative-0
Velocity :	0.7 to 45 m/sec
Pressure :	1.0 to 200 bar absolute
Subcooling (Δtsub) :	0 - 240 ° C
q_c :	116 to 5234 w/cm2

Mirshak Correlation :

The Mirshak correlation is based on data for annular channels with heated tube diameter of 1.27 cm and 2.03 cm and rectangalar channels (with channel width of 6.4 cm, heated strip width of 5.08 cm, channel thickness from 0.3 to 0.58 cm). For both test sections, only one side of the channel was heated. All data correlated have positive subcooling at the channel exit. Acoording to Mirshak, critical heat flux is given by

$$q_c = 151\ (1+0.1198\ U)\ (1+0.00914\ \Delta t_{sub})\ (1+0.19\ P)$$

where

$$\Delta t_{sub} = T_{sat} - T_{in} - \Delta T_c$$

The above correlation is valid within the parameter ranges given below :

Steam quality :	negative
Velocity :	1.5 to 13.7 m/sec
subcooling (Δt_{sub}):	5 to 75 C
Pressure :	1.72 to 5.86 bar absolute
Eqivalent diameter :	0.53 to 0.17 cm
qc :	284 to 1022 w/cm^2

In both the Labunstov and Mirshak correlation, the burnout heat flux depends on the water subcooling and vice versa. For this reason the determination of the critical heat flux requires an iterative procedure. As a first step, the coolant subcooling is calculated for channel geometry and coolant flow using corresponding physical properties of water. By substituting into the Mirshak or Labunstov correlation, the critical heat flux for the next step of iteration is determined and vice versa. The iteration is continued until the convergency criteria specified by the number of iteration are fullfilled.

III. Application of the HEATHYD code

After completing experimental verification of the HEATHYD model, it was applied to calculate the thermohydraulics of a plate type MTR. It was assumed that the core consists of 48 Fuel elements generating a total power of 30 MW. For this power level a total flow rate of 50.4 m3/h (for each fuel element) was assumed. As inlet coolant temperature, 40 °C and a pressure of 2 bar were applied as main initial thermohydraulic data(Table 1).

In addition to the more relevant case that is defined as the average core, the coolant flow and heat transfer for the hot channel was analysed.The main results of the HEATHYD calculation have been compiled on Table 2 for the average and hot channel case at nominal power and overpower of the 114 %, respectively. In all cases, hydraulic calculations show 4.08 m/s for the coolant velocity and 0.44 bar as total pressure loss. The result of the heat transfer part depends on the total power. In case of overpower and hot channel, it amounts to 3.56 including axial radial and engineering factor. The results of all calculations have been depicted in the following figures. In Fig. 4 and Fig. 5, the clad surface and coolant temperatures have been represented for four cases as a function of the active length of the fuel plate.

TABLE 1. MAIN DATA OF A MTR CORE USED IN CALCULATION WITH HEATHYD

Steady state power level	30
Fuel element dimensions (mm)	76 1 x 80 5 x 873
Number of fuel lemen in the core (a) Standard fuel.elements (b) Control elements	 40 8
Number of fuel plates in (a) Standart fuel elements (b) Control elements	 21 15
Fuel plate dimensions (mm)	1 30 x 70 75 x 625
Fuel meat dimension (mm)	1 54 x 62 75 x 600
Cladding thickness	0 38
Water channel thickness (mm)	2 55
Total primary system flow rates (m3/h)	3120
Flow rates through the fuel element (m3/h)	50 40
Direction of the flow rates through core	Downwards
Average power generated pler fuel element (MW)	0 5462
Radial factor	3 12
Coolant inlet temperature	40*
Pressure at channel inlet (bar abs)	2*

* Assumption

TABLE 2. RESULTS OF FLUID FLOW AND HEAT TRANSFER CALCULATION USING HEATHYD CODE

PARAMETERS	NOMINAL POWER AVERAGE / HOT		OVER POWER AVERAGE / HOT	
Coolant velocity (m/sec	4,08	4,081	4,080	4,082
Pressure drop:				
(a) dynamic (bar)	0,082	0,083	0,082	0,083
(b) gravitation (bar)	0,059	0,059	0,059	0,059
(c) friction (bar)	0,028	0,273	0,276	0,272
(d) inlet (bar)	0,024	0,024	0,024	0,240
Total pressure drop across core (bar)	0,442	0,438	0,442	0,438
Pressure at channel exit (bar)	1,555	1,559	1,555	1,559
Saturation temperature at channel exit (°C)	112,46	112,53	112,54	112,54
Coolant temperature rise across the channel (°C)	10,46	33,14	12,09	37,79
Peak clad temperature (°C)*	73,93	134,40	78,30	145,60
Average heat flux (w/cm^2)	38,03	118,65	43,36	135,26
Peak heat flux (w/cm^2)	62,67	195,56	71,46	222,93
Margin to Onset of Nucleate Boiling (ONB)	2,59	0,80	2,28	0,69
Safety margin:				
(a) to Departure from Nucleate Boiling (DNB)	7,47	2,27	6,53	1,96
(b) to Onset of Flow Instability (OFI)	5,24	1,68	4,59	1,47

*** Using Haussen II Correlation**

In the hot channel case, the graph includes the saturation temperature that has been partly exceeded by the clad surface temperature with the result of subcooled boiling. The subcooled region is specified by exceeding the heat flux at onset of nucleate boiling (margin to ONB 0.69). In case of the average channel, the clad surface temperature does not reach the saturation temperature that indicates one phase liquid cooling as Fig. 4 shows.

The variation of the heat flux at ONB and DNB with the height of the cooling channel has been shown in Fig. 6, Fig. 7 and Fig. 8 for hot and average channel at nominal and over power condition. By comparison, the actual heat flux has been also included in the figures. According to the results, the critical heat flux corresponding to departure from nueleate boiling is not reached. Because of low ONB value (margin amounts to 0.69), the cooling of the hot channel at over power level of 114 % occurs under the condition of subcooled nueleate boiling.

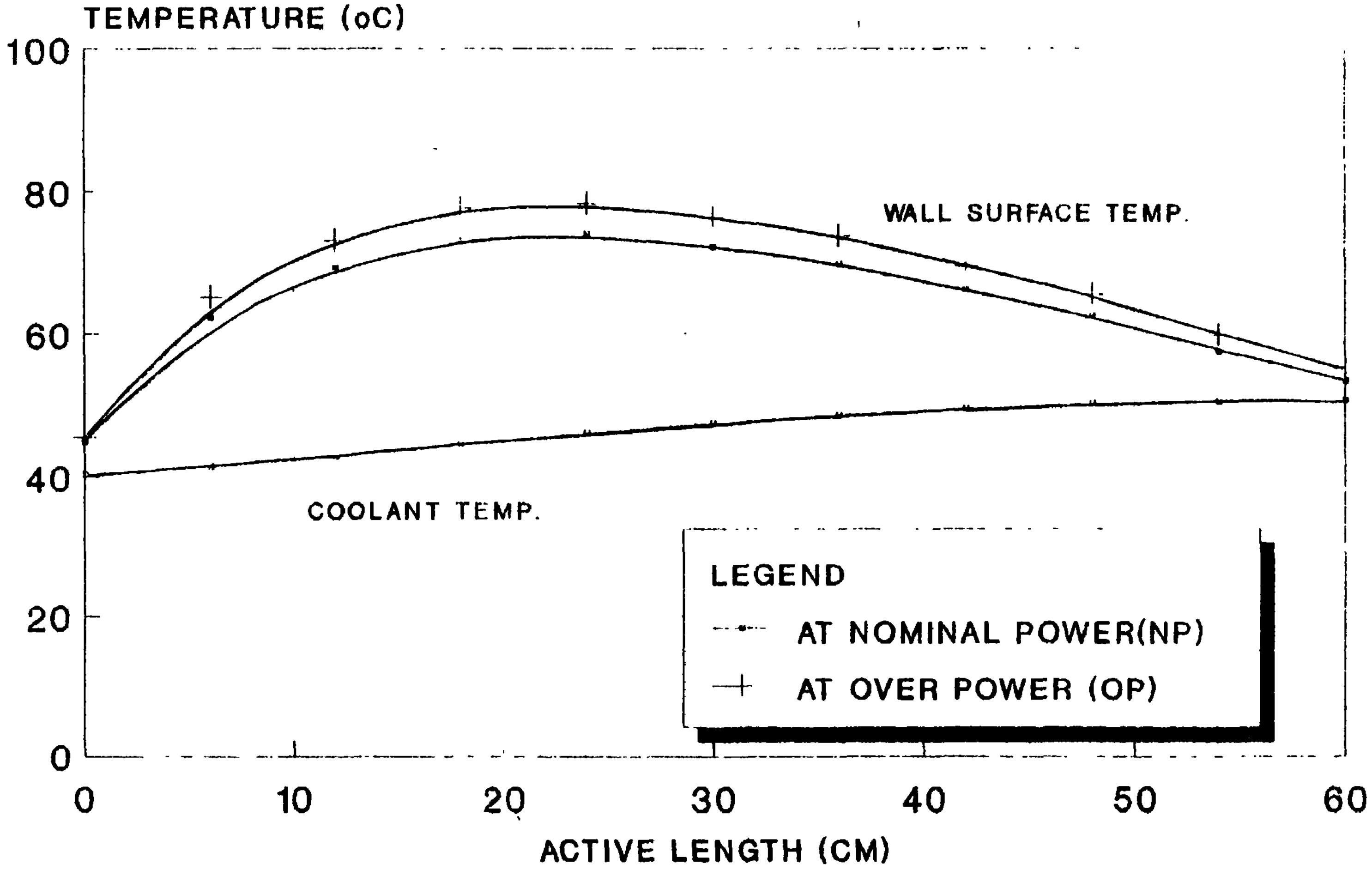

Fig. 4: Axial distribution of the temperature of the fuel plate and coolant

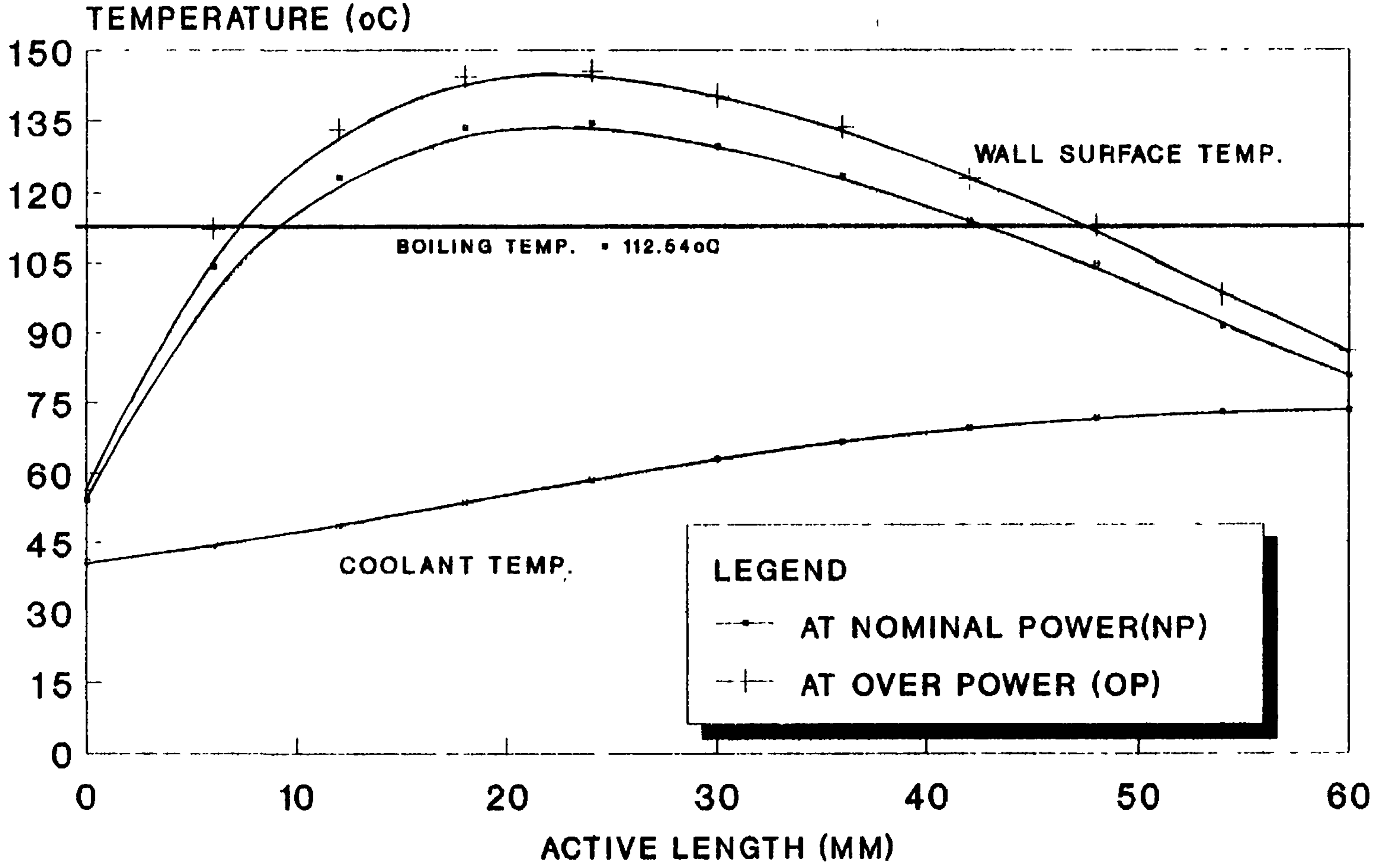

Fig. 5: Distribution of the temperature of the fuel plate and coolant for hot channel case

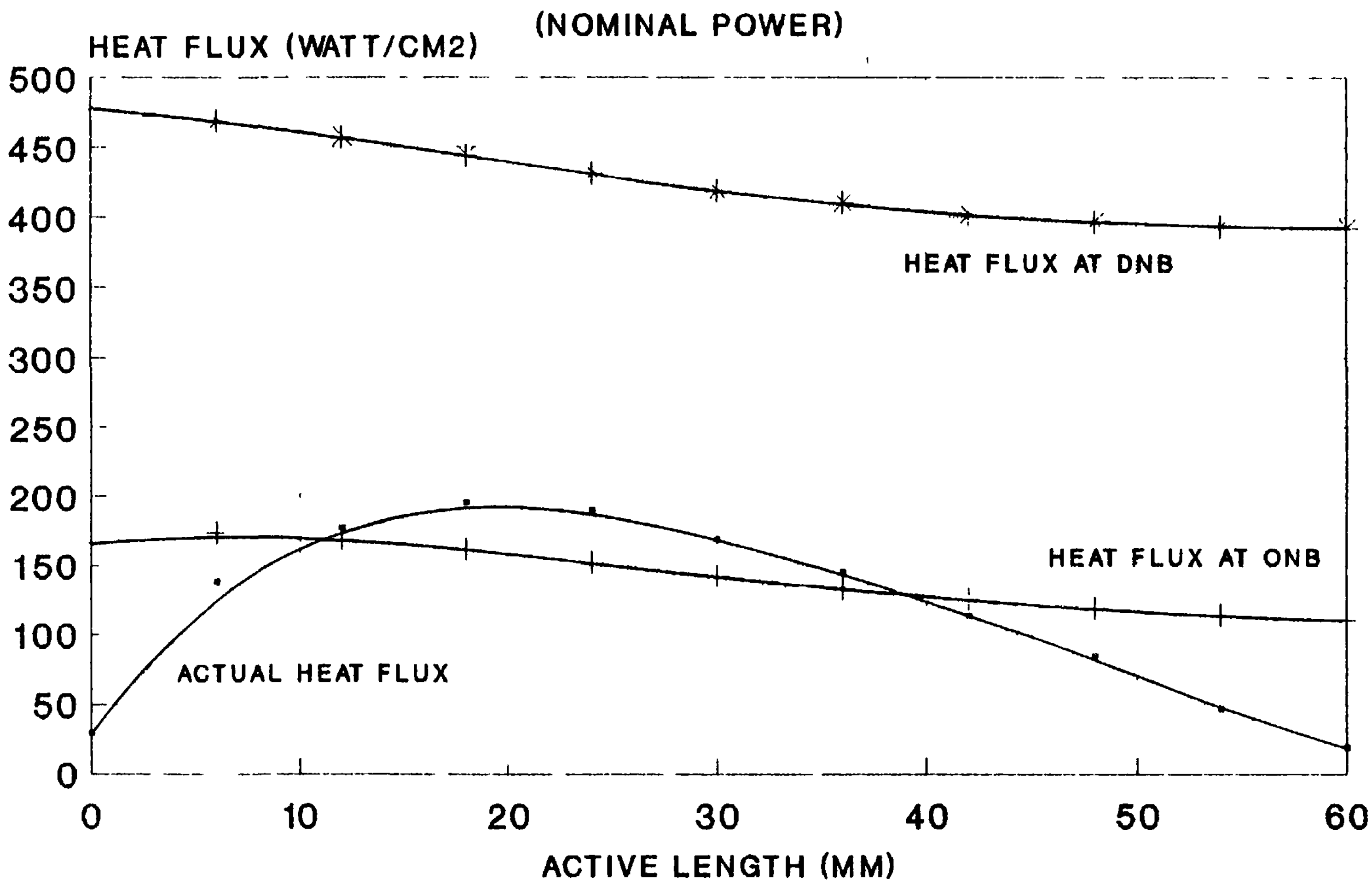

Fig. 6: Distribution of the temperature of fuel plate and coolant for hot channel case

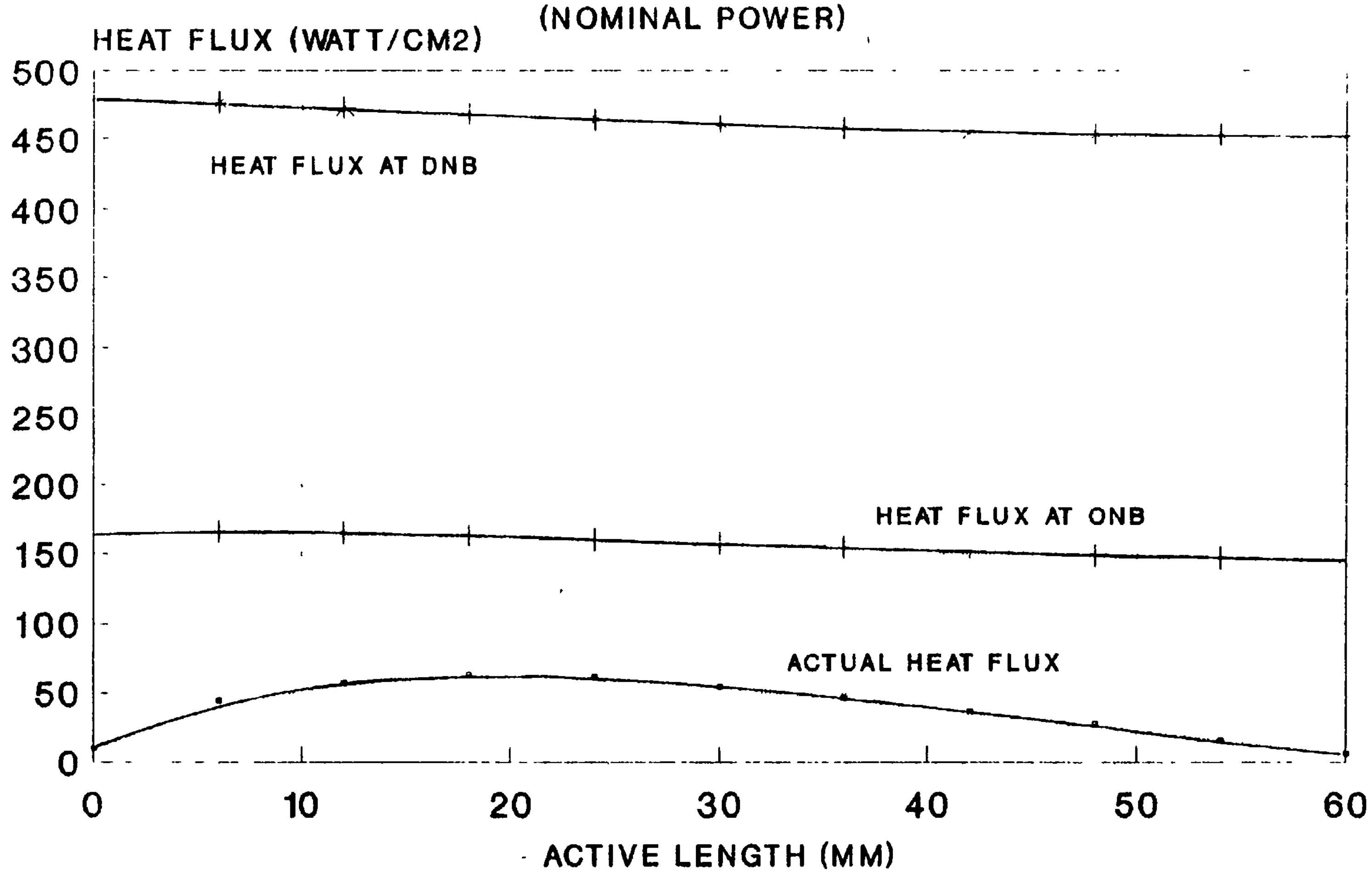

Fig. 7: Axial variation of heat flux at ONB and DNB in comparison to actual heat flux

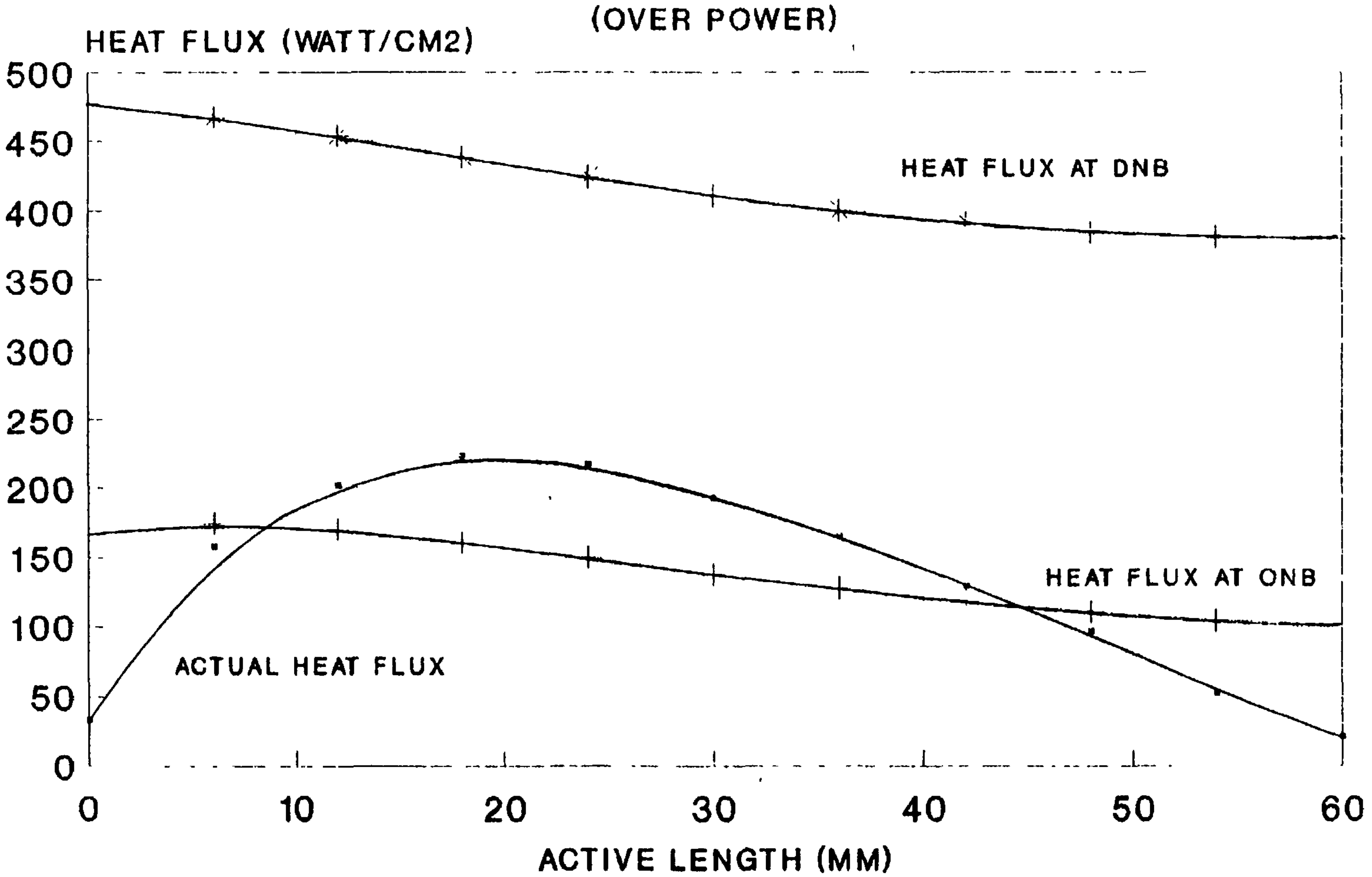

Fig. 8: Axial variation of heat flux at ONB and DNB in comparison to actual heat flux for hot channel case

The results of calculations show that the safety margin to flow intability represents the limiting parameter regarding safe design and operation. Accordingly, the minimum margin to onset of flow instability amounts to 1.47.

NOMENCLATURE

Symbol	Definition	Units
A_c	Total Water Cross section Area in an Element	cm^2
A_O	The Crossectional Area of the End Box Imediatily Beyond the Channel Entrance / Exit	cm^2
C_P	Specific Heat of Water	KJ/Kg.C°
D_e	Equivalent Hydraulic Diameter	cm
E	Youngs Modulus of Elasticity	bar
f	Friction Factor	(Dimensionless)
f_a	Axial Peak-to-Average Heat Flux Ratio	(Dimensionless)
f_r	Radial Peak-to-Average Power Ratio	(Dimensionless)
G	Mass Flux (= pU)	Kg/m^2sec
h	Film Heat Transfer Coefficient	$w/cm^2C°$
H_{co}	Effective Fuel Plate Length for Heat Transfer	cm
k	Heat Conductivity of Water	w/m.C°
K_{en}	Entrance Pressure Loss Coefficient	(Dimensionless)
K_{ex}	Exit Pressure Loss Coefficient	(Dimensionless)
L_c	Length of Fuel plate (Coolant Channel)	cm
N_f	Number of Fueled Plates	(Dimensionless)
P_z	Pressure of Coolant at any Point „z"	bar abs
P	Pressure at Channel Exit	bar abs
P_c	Critical Pressure of Coolant	bar abs
ΔP_{en}	Pressure Loss at Channel Entrance	bar
ΔP_{ex}	Pressure Loss at Channel Exit	bar
ΔP_f	Pressure Loss Through Channel due to Fricition	bar
ΔP_d	Dynamic Pressure Loss	bar

NOMENCLATURE (cont)

ΔP_F	Total Pressure Loss	bar
P_r	Prandtls Number	(Dimensionless)
q	Local Heat Flux	w/cm^2
q_a	Axial Average Heat Flux	w/cm^2
q_{ONB}	Heat Flux at Onset of Nucleate Boiling	w/cm^2
q_{OFI}	Heat Flux at Onset of Flow Instability	w/cm^2
q_c	Burnout (Critical) Heat Flux	w/cm^2
Q	Volumetric Flow Through the Element	m^3/hr
R_e	Reynolds Number	(Dimensionless)
T_{in}	Water Temperature at Core Inlet	C°
T_{out}	Water Temperature at Core Outlet	C°
ΔT_c	Water Temperature Rise in the Coolant Channel	C°
T_{sat}	Saturation Temperature of Water	C°
T_s	Clad Surfache Temperature	C°
T_{fl}	Fluid Temperature	C°
ΔT_{sub}	Water Subcooling	C°
t_m	Fuel Meat Thickness	cm
t_p	Fuel Plate Thickness	cm
T_w	Water Channel Thickness	cm
U	Water Velocity in the Channel	m/sec
U_o	Water Velocitiy just beyond the Channel	m/sec
V_{crit}	Critical Velocity	m/sec
W	Water Channel Width	cm
W_h	Effective Fuel Plate Width for Heat Transfer	cm
W_p	Total Plate Width of Chord of Curved Plate	cm
z	Axial Location along the Channel	cm
λ	Heat Vaporization	KJ/Kg
μ	Dynamic Viscosity of Water	Pa.sec
ρ	Denstiy of Water	Kg/m^3
υ	Poissons Ratio	(Dimensionless)
υ'	Kinematic Viscosity of Water	cm^2/sec

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ON-LINE USE OF PERSONAL COMPUTERS TO MONITOR AND EVALUATE IMPORTANT PARAMETERS IN THE RESEARCH REACTOR DHRUVA*

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Abstract

The on-line use of Personal Computers in research reactors, with custom made applications for aiding the operators in analyzing plant conditions under normal and abnormal situations, has become extremely popular. A system has been developed to monitor and evaluate important parameters for the research reactor DHRUVA, a 100 MW research reactor located at the Bhabha Atomic Research Centre , Trombay. The system was essentially designed for on-line computation of the following parameters: reactor thermal power, reactivity load due to Xenon, core reactivity balance and performance monitoring of shut-down devices. Apart from the on-line applications, the system has also been developed to cater some off-line applications with Local Area Network in the Dhruva complex. The microprocessor based system is designed to function as an independent unit, parallel dumping the acquired data to a PC for application programmes.
The user interface on the personal computer is menu driven application software written in 'C' language. The main input parameters required for carrying out the options given in the above menu are : Reactor power, Moderator level, Coolant inlet temperature to the core, Secondary coolant flow rate, temperature rise of secondary coolant across the heat exchangers, heavy water level in the Dump tank and Drop time of individual shut off rods.

1 INTRODUCTION

The On-line use of Personal Computers (PC) in Research reactors, with custom made applications for aiding the operator in analysing plant conditions under normal and abnormal situations, has become extremely popular. PC can be effectively used for data acquisition and for processing the acquired data in providing information to the operator with user-friendly displays.

The typical areas of on-line applications of PC in nuclear research reactors include :

- Acquisition and display of data on process parameters.
- Performance evaluation of major equipment and safety related components.
- Fuel management.
- Computation of Reactor physics parameters.
- Failed fuel detection and location.
- Inventory of system fluids.

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* Research carried out in association with the IAEA under Research Contract No. IND/6046.

A system has been developed to monitor and evaluate important parameters for the research reactor DHRUVA, a 100 MWt research reactor located at the Bhabha Atomic Research Centre, Trombay. Dhruva is a natural uranium fuelled, heavy water moderated and heavy water cooled reactor with a maximum thermal neutron flux of 1.8 X 10^{14} n/cm^2/sec. The reactor is intensely utilised for basic and applied research, radio-isotope production, material testing and man-power development and training. The reactor has a vertical core and the coolant heavy water flows from bottom to top through individual coolant channels where the fuel assemblies are housed. The primary coolant circuit consists of three identical heavy water recirculating loops. Heavy water inside the reactor vessel serves as moderator & reflector and the heavy water in the coolant, moderator and the reflector regions is intermixed. Reactor power is controlled by controlling moderator level in the reactor vessel. Demineralised water is used as secondary coolant in a closed loop. The secondary coolant is cooled by sea water in a separate set of heat exchangers. A simplified flow sheet of the coolant system is shown in Fig - 1.

2 OBJECTIVE

The objective of the project is to develop a PC based on-line process and reactor parameter monitoring system for Dhruva to serve as an operator aid. The system was essentially designed for on-line computation of the following parameters :

- Reactor thermal power.
- Reactivity load due to Xenon.
- Core reactivity balance.
- Heavy water system inventory.
- Performance monitoring of shut-down devices.

Apart from the on-line applications mentioned above, the system has also been developed to cater to some off-line applications with Local Area Network in the Dhruva complex.

3 SYSTEM DESCRIPTION

The system acquires data on various process parameters required for computation of the desired functions, stores and manipulates the acquired data and presents the required information through an easy-to-use interface. The microprocessor based system is designed to function as an independent unit, parallelly dumping the acquired data to a PC for application programmes.

The main system comprises of two nos. of Front-end bins for acquiring and further transmission of process data to PC through a serial link, Bin-1 catering to all process parameters and Bin-2 for performance monitoring of shutdown devices. An IBM

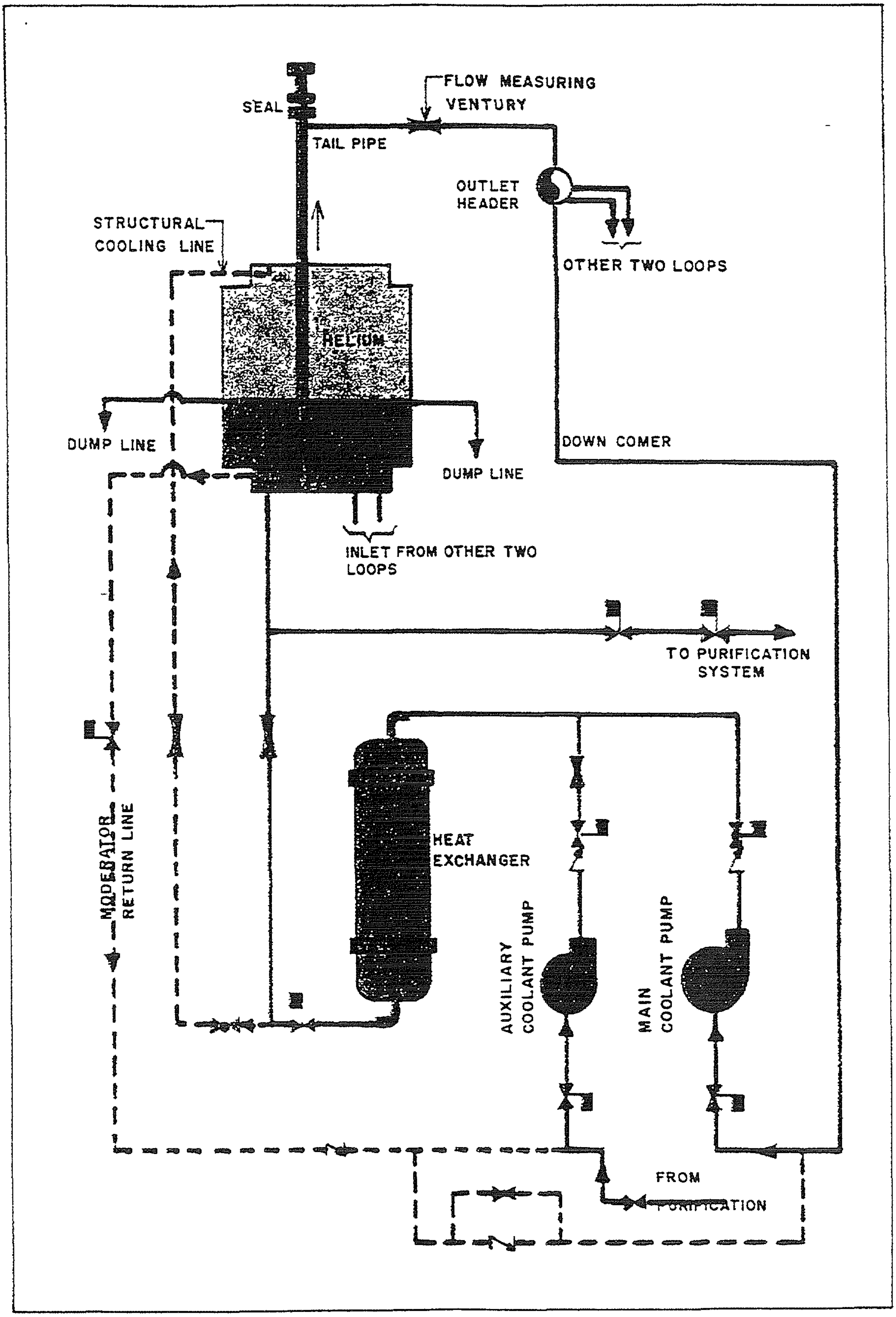

Fig - 1 SIMPLIFIED FLOW DIAGRAM OF DHRUVA MAIN COOLANT SYSTEM compatible PC/AT-486 is used as the computer to acquire the on-line data from the Front-end system with a cartridge tape drive of 250 MB provided for monthly back-up of the data. To make the on-line/off-line information available to other users as well, Local Area Network (LAN) in the reactor complex has been installed. Five nos. of PC/ATs based on Intel 80386 have been used as the nodes of the network. The PC in the control room forms a node on the network having all the application software which can be accessed from any other node with appropriate access control features incorporated. All the above PCs are networked using Novel DOS-7 operating system in Peer-to-Peer mode. Each PC connected as a node to the network is a workgroup of its own but can also cater to the other users if required. Schematic of the network is given in Fig -2. Block diagram of the front-end system(Bin # 1) is shown in Fig - 3 and the scematic of front-end system for status /performance monitoring of shut-down devices(Bin # 2) is indicated in Fig - 4.

i) Reactor thermal power.

Thermal power of the reactor is computed by calorimetric calculations using the reactor coolant system and other process system parameters. The PC based system is developed to provide on-line thermal power readings based on heat transferred to the secondary coolant system with appropriate corrections incorporated.

The application software on the PC after computing the thermal power not only displays the values but also gives the graphical display of thermal power, and core coolant inlet temperature for the last 72 hours.

ii) Reactivity load due to Xenon.

Information regarding reactivity load due to Xenon at any instant is important towards assessing the Xenon over-ride time available to restart following a reactor trip/shutdown depending on the excess reactivity available. Subsequent to poisoning out, information is needed about the poison-out period during which the Xenon will decay to a sufficiently low level such that the reactor can be restarted. Since determination of reactivity load due to Xenon involves solution of complex mathematical expressions which require expertise of reactor physicists, operations personnel are put to a disadvantage at odd times. The PC based system has therefore been developed for providing reactivity load due to Xenon build-up/decay for any given operating history of the reactor. In addition, operations personnel can also interact with the system for obtaining information on a possible power manoeuvre scheme before shutdown to maximise the available shut-down time without poisoning in order to attend to certain short duration shutdown jobs. For convenience of interpretation, reactivity load due to Xenon is expressed in terms of moderator level in reactor vessel.

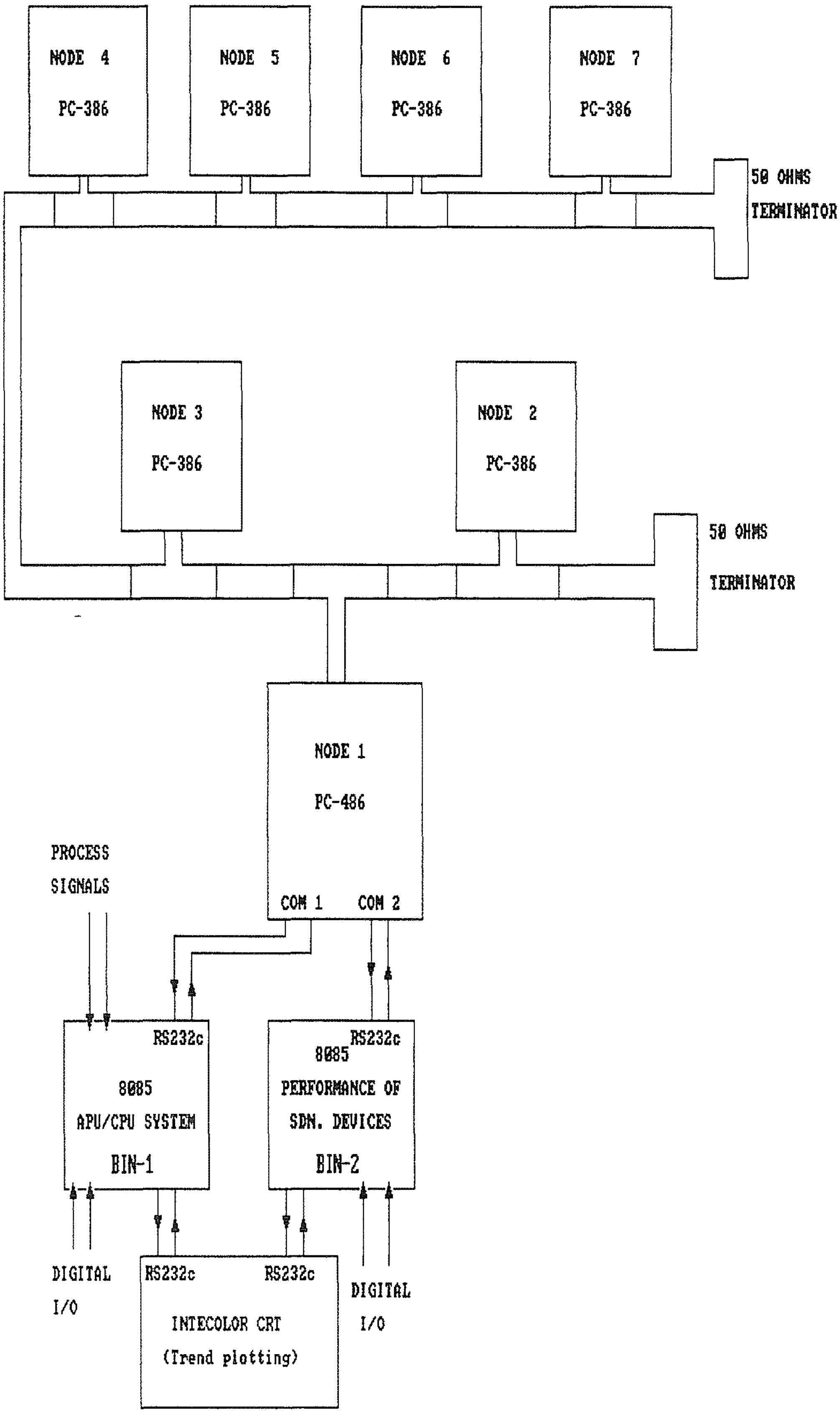

Fig - 2 SCHEMATIC DIAGRAM OF THE NETWORK

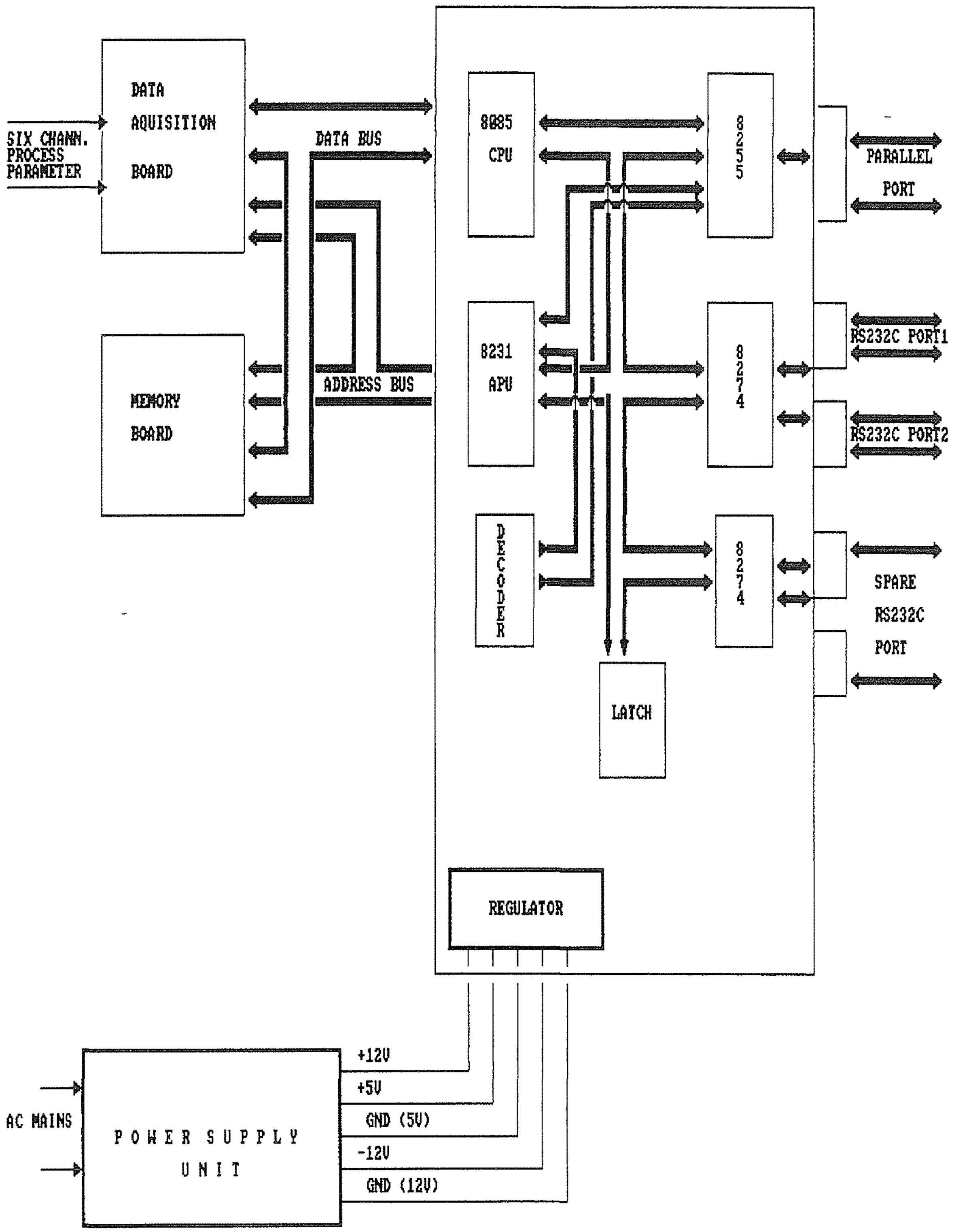

Fig - 3 MAIN BLOCK DIAGRAM OF FRONT-END SYSTEM(Bin No. 1)

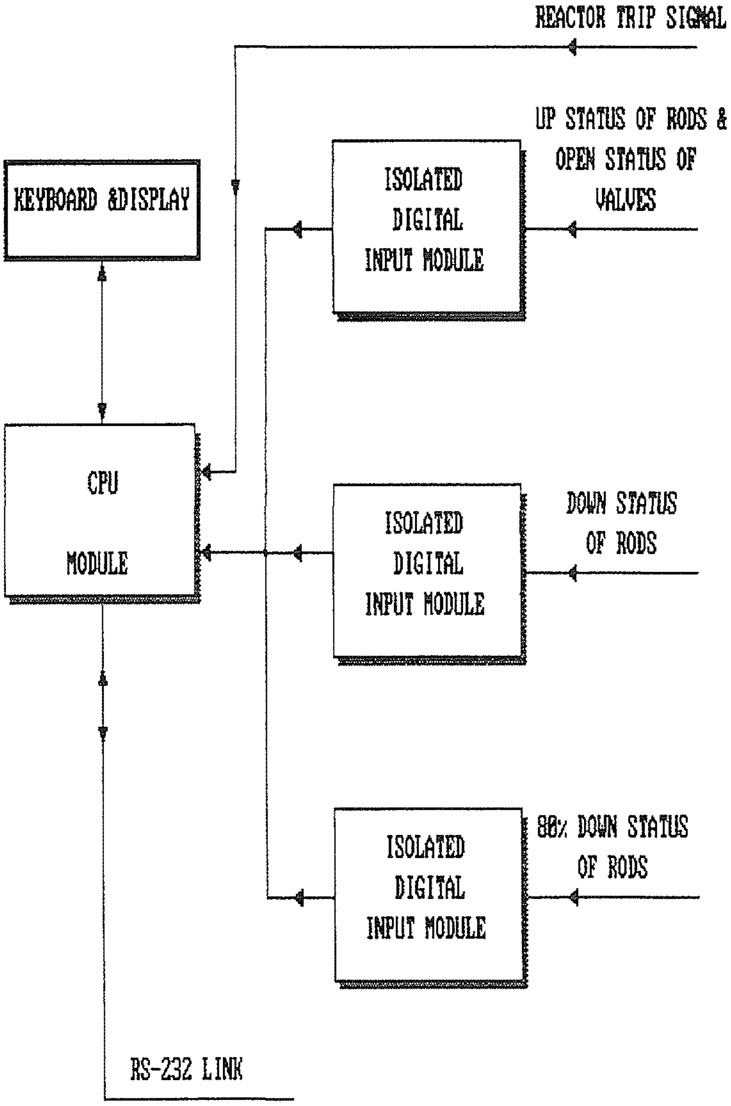

Fig - 4 SCHEMATIC OF FRONT-END SYSTEM FOR STATUS/PERFORMANCE MONITORING OF SHUT-DOWN DEVICES(Bin.No. 2)

The system periodically acquires information on the moderator level, reactor power and moderator inlet temperatures to calculate the operating Xenon and Iodine load and keeps updating these values. Whenever the reactor trips or when the user initiates the power manoeuvre routine, the system acquires the values of Xenon and Iodine at that instant and predicts the value of Xenon at a given time subsequently with further proposed changes in reactor power taken into account. Using the value of Xenon at any time the corresponding value of moderator height is calculated. By comparing this with the maximum attainable moderator level in reactor vessel, it is possible to find Xenon over-ride time and xenon poisoned out time.

iii) Core reactivity balance.

In a reactor like DHRUVA there are primarily two types of reactivity loads which affect the critical height of the moderator in the reactor.

Firstly, it is the operational effects arising because of the reactor thermal power itself as also the temperature rise in the various regions of the core. Operational reactivity effects also include poisoning due to fission products like Xenon and Samarium. Secondly, it is the reactivity effect due to the various experimental/irradiation assemblies present in the reactor core.

In computing the reactivity balance, effort is made to account for all the reactivity effects mentioned above to interpret the criticality as reflected in the observed critical height of the reactor. The computed reactivity effect may not agree precisely with observed height and the error will be reflected in the unassigned reactivity. However, under all operating conditions it is necessary to ensure that the error remains within specified limits and in case of any discrepancy beyond the specified limits, the same is investigated. This information is derived from the operating parameters viz. the reactor thermal power, moderator level and core inlet temperature. These parameters are continuously acquired by the front-end system and fed to the PC through a serial link. The user friendly PC based programme provides the reactivity balance sheet. The reactivity effects due to experimental loads are obtained from physics computations and are entered off-line as and when necessary.

iv) Heavy water system inventory.

For all the operating states of the reactor, a strict vigil has to be maintained on the total inventory of heavy water in the system. Abnormal loss of inventory is indicative of possible degradation of the primary coolant system pressure boundary. Also, the operability of reactor at any power with different operating and experimental reactivity loads is mainly decided by the maximum pumpable moderator height into the reactor vessel which can be computed by knowing the total inventory of heavy water available in the system.

This system continuously acquires the information on heavy water levels in the various storage tanks of the system and transfers the processed signal to PC through a serial port. In the PC the application software displays all the tank levels and the corresponding inventory and computes the total inventory by adding the fixed inventory in the system piping after incorporating appropriate temperature corrections. Based on the total inventory, the maximum pumpable height in the reactor vessel is computed.

The system updates the inventory every two minutes and maintains the history for the past 72 hrs.

v) Performance monitoring of Primary shut-down devices.

The primary shut-down system of DHRUVA consists of 9 cadmium shut-off rods. When a reactor trip is generated by the protection system, the electro-magnetic clutches holding the shut-off rods above the active region of the core, are de-energised and the shut-off rods fall rapidly into the core under gravity and spring induced initial acceleration.

For satisfactory performance of shut-off rods, the drop time should be within the specified time limits. The shut-down devices are provided with limit switches to indicate their position status. The system is designed to monitor this status, measure the time of travel on a reactor trip and to maintain history of performance data.

Front-end system periodically acquires the required analog process signals and after conditioning and digitisation, the signals are stored in the memory of the front-end system. This data is transferred to the PC through a Serial communication link. It can also operate in stand-alone mode by computing and displaying all the computed parameters.

4 APPLICATION SOFTWARE

The user interface on the personal computer is menu driven application software written in 'C' language. The memory resident programmes written will be active as long as the main computer is on and keep capturing the data from both the serial ports every 2 minutes. The data thus captured is stored in PC files which is in turn available for the on-line as well as off-line application software.

The main input parameters required for carrying out the options given in the above menu are Reactor power, Moderator level, Coolant inlet temperature to the core, Secondary coolant flow rate, temperature rise of secondary coolant across the heat exchangers, heavy water level in the Dump tank and Drop time of individual shut off rods. All these data items are stored continuously in two files. The programme when invoked will access the required data from these files and does the individual operation/function as demanded. It keeps updating the current status of any calculated parameter by scanning the data in the file every 2 minutes.

Apart from the on-line applications mentioned above, the system has also been developed to cater to three off-line applications Viz. *Stores inventory programme, Fuel* *assembly history programme and Reactor daily performance report* . All these packages have been developed in FOXPRO environment with user-friendly pop-up menus and help levels at every stage. These programmes are accessible from all the nodes connected on the LAN. Provisions also exist to incorporate extra features in this system in future by augmenting the capacity of the Front-end system. The proposed applications include logging of important system parameters such as individual fuel channel coolant outlet temperature, channel coolant flow and channel power output etc..

5 CONCLUSIONS

The use of PC as an operator aid has been found to be highly beneficial for operation and maintenance related tasks and has a potential for applications in several areas. The system has considerably reduced the burden on the part of the operator by providing the necessary information in an organised manner through easy-to-use interfaces. Though the hardware and software developed for the system is specific to Dhruva reactor, the concept and the method of approach can be suitably modified and utilised for other applications as well.

ACKNOWLEDGEMENTS

Implementation of this project is the result of a great deal of time and effort put in by many individuals and their knowledge, experience and dedication were invaluable in making the system functional. The project team would like to record its special appreciation for the following officers of Reactor Group, BARC who made significant contributions to the project.

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U.S. Kulkarni	: Design & Engg. Services section.
R.S. Sharma	: Reactor chemistry section.

DEVELOPMENT OF RESEARCH REACTOR PARAMETER MEASURING SYSTEM BASED ON PC*

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Abstract

Development of research reactor parameter measuring system based on a personal computer was carried out at the Korea Atomic Research Institute. The work includes the interface construction between a PC and reactor instruments, experimental logic development, programming and test. At first, various reactor experiments were reviewed to identify signal type, dynamic range and number of channels. The following programmes were developed: multi-counter, multi-scaler, reactivity computer, control rod timer and thermal power calibrator. The programmes multi-counter and multi-scaler are basic tools for many experiments, such as criticality measurements, reactor start-up operation, shielding experiments, health physics experiments, random neutron process analysis, etc. The personal computer replaces many conventional counter modules and multi-channel analyzers for counting and scaling experiments and the programme for a control rod drop time replaces the memory oscilloscope. The power calibration programme fully automates this experiment offering accuracy and convenience. The reactivity computer can accommodate virtually all reactor power monitors.
With the modern personal computers, the reactor staff can obtain more accurate results and their experimental field can be expanded.

1. Introduction

The modern personal computer (PC) offers good features for the research reactor experiments for which computer application could not be justified for long time because of its high price, and research reactor staffs have been encouraged to utilize it for their reactor analysis and experiments. Nowadays, computerization is common trend in every field and many utilities are commercially available at reasonable cost. For the case of research reactor experiments, however, a low cost and ready-made computerized system is not available yet since its demand is very limited, which means that reactor staffs should develop such system by themselves but majority of them are not specialists in the computer. From this point of view, it is considered that developing a reactor parameter measuring system based on the PC is worthwhile.

The work includes the interface construction between the PC and reactor instruments, experimental logic development, programing, and test.

At first, various reactor experiments were reviewed to identify signal type, dynamic range, number of channels, *etc.*, and then commercial I/O boards were surveyed to choose proper one. So far, the effort has been concentrated on the use of PC counter because its application is not so popular in general compared to the use of analog to digital converter (ADC) while it shares very important parts in the research reactor experiments especially for a new reactor commissioning. As an ADC board has few counter channels, however, a pure counter board which can accept up to nine pulse channels, should be self designed.

Developed programs are multi-counter, multi-scaler, reactivity computer, control rod drop timer, and thermal power calibrater.

The PC multi-counter and multi-scaler are basic tools for many kinds of experiment such as criticality measurement, reactor startup operation,

* Research carried out in association with the IAEA under Research Contract No. ROK/6050.

shielding experiments, health physics experiments, random neutron process analysis, *etc.* As a PC replaces many conventional counter modules and several multi-channel analysers (MCA) for counting and scaling experiments, it offers lots of advantages in cost and effectiveness, especially for those experiments using several counters simultaneously.

The reactivity computer measures real time variation of reactivity. It can be used virtually with all neutron detectors for the reactor power monitoring simulataneously. In order to cover the startup channels, numerical methods for inverse point kinetics are accutely reviewed and tested in detail to verify error trend and to search the best algorithm. The program made for the test itself can be utilized for the simulation of other reactivity measurements to verify source of error and its trend - such as the gamma background effect when an uncompensated ionization chamber is used, the source effect, conventional rod drop method, *etc.* The reactivity computer can be used at any reactor power level from startup to power range. So far, it has been utilized for the control rod calibration, reactivity coefficient measurement, *etc*.

The program for control rod drop time replaces the memory oscilloscope for this experiment, and offers operators conveniency in documentation and reporting.

The power calibration program fully automatizes this experiment offering accuracy and conveniency.

Thanks for the computerization of experiments, reactor staffs can obtain more accurate results, expand their experimental field, save their time, and unburden their effort a lot in measurement, data logging and reporting.

2. PC Interface

Research reactor experiments include many kinds of different sensors and instruments which have several different signal types and dynamic range. The required sampling speed is also very different from one experiment to another. Therefore, the basic requirements for the interface, are enough analog and pulse channels, high speed, high precision, wide band, high reliability, and easy feature for calibration. There are many commercial I/O boards satisfying these requirements except for the combination of analog and pulse channels. An ADC board has only few or no counter/timers. Therefore, a commercial ADC board is chosen but the counter board is self designed.

The interface system established for the reactor experiments, basically consists of an analog I/O board and a counter one.

Considering low level signal acquisition such as thermocouples, differential input system instead of single ended one, is adopted even though it reduces channels to a half. The number of analog channels can be expanded up to 124 by adding analog channel expansion pannels.

It has a 12-bit ADC which has the maximum throughput rate of 50 KHz, and programable amplifiers of gain 1, 10, 100 and 500, which can cover at least 4-decades of signal band. It also has a board temperature sensor. The combination of high gain and board temperature sensor permits direct connection of thermocouples, but the ADC speed is remarkably reduced at high gain (100 or 500) and the zero offset becomes unstable when channel expansion pannels are used.

In addition to the analog input system, it has two 12-bit DACs, 16 digital I/O (DIO), and two 16-bit counter/timers. While DACs are not required so far, two counter channels are not sufficient. Therefore, these channels have been utilized only for the timer and a pure counter board is added. As DIOs are assigned to the analog channel expansion pannels they cannot be used for actual DIO. DIOs are not required so far.

Its calibration and range selection are done by software, which offers lots of convenience.

The counter board is basically composed of three Intel 8254 counter/timer chips which has 10 MHz speed, and can receive up to nine pulse channels. Should more pulse channels be required it can be easily expanded.

The PC for this work can be IBM-PC/286, /386, /486 or compatibles with the DOS operating system.

ASSEMBLER and PASCAL were chosen for the programing languages considering program readability and efficiency. ASSEMBLER is used for the interface programs dealing with direct I/O functions and for time critical routines. PASCAL is the main language used for various reactor parameter calculations based on the data received by ASSEMBLER routines. Basic interface routines provided by the I/O board supplier, cannot be used for this work since they do not support the system with analog channel expansion pannels. Therefore, basic programs for analog data acquisition as well as the counter/timer acquisition and control, should be self-programed.

3. Development of Computerized Reactor Experiments

3.1. Multi-Counter

Many reactor experiments are depended on counter(s). This program is to replace conventional counter modules by a PC. It can display up to eight counter data on the CRT screen. Therefore, when eight channels of pulse inputs are fed to this system it is just like eight conventional counters are working. Furthermore, it has several advantages compared to using counter modules. There is no dwell time between each counting step, user can watch several past records, he can get dead time corrected results, and he can save wanted data for further analysis.

Fig. 1 shows an example of CRT screen while multi-counter is working. Current counting line is updated at every 1/50 second, and it is not interrupted by key input requests such as counting time change, *i.e.*, counters are flying while the user is typing in. There are four functions of counting time set, reset, data save and end of work. Counting time can be set by 1/50 second precision. Data can be saved continuously or selectively,

F1(ctrl-T) F4(ctrl-R) F5(ctrl-S) F8(ctrl-E)
Set Time:100 Reset Save data Exit:

T(sec)	Counter1	Counter2	F.C-1	F.C-2	BF3-1	BF3-2	BF3-3	BF3-4
10.00	1000	1000	1000	1000	1000	1000	1222	551
cps	100.01	100.01	100.01	100.01	100.00	100.00	122.20	55.10
100.00	10000	10000	10000	10000	10000	10000	12015	5689
cps	100.01	100.01	100.01	100.01	100.00	100.00	122.02	56.89
100.00	10000	10000	10000	10000	10000	10000	12103	5674
cps	100.01	100.01	100.01	100.01	100.00	100.00	121.03	56.74
100.00	10000	10000	10000	10000	10000	10000	12050	5666
cps	100.01	100.01	100.01	100.01	100.00	100.00	120.50	56.66
100.00	10000	10000	10000	10000	10000	10000	12129	5598
cps	100.01	100.01	100.01	100.01	100.00	100.00	121.29	55.98
100.00	10000	10000	10000	10000	10000	10000	12142	5657
cps	100.01	100.01	100.01	100.01	100.00	100.00	121.42	56.57
100.00	10000	10000	10000	10000	10000	10000	12093	5631
cps	100.01	100.01	100.01	100.01	100.00	100.00	120.93	56.31
32.68	3268	3268	3268	3268	3268	3268	3943	1834

Fig. 1. An Example of CRT Display While Multi-Counter is Working

including each counting time interval and the time from the start of the experiment. Data saved or to be saved, are displayed by the reverse mode for its easy identification. The data screen scrolls upward when it reaches the bottom.

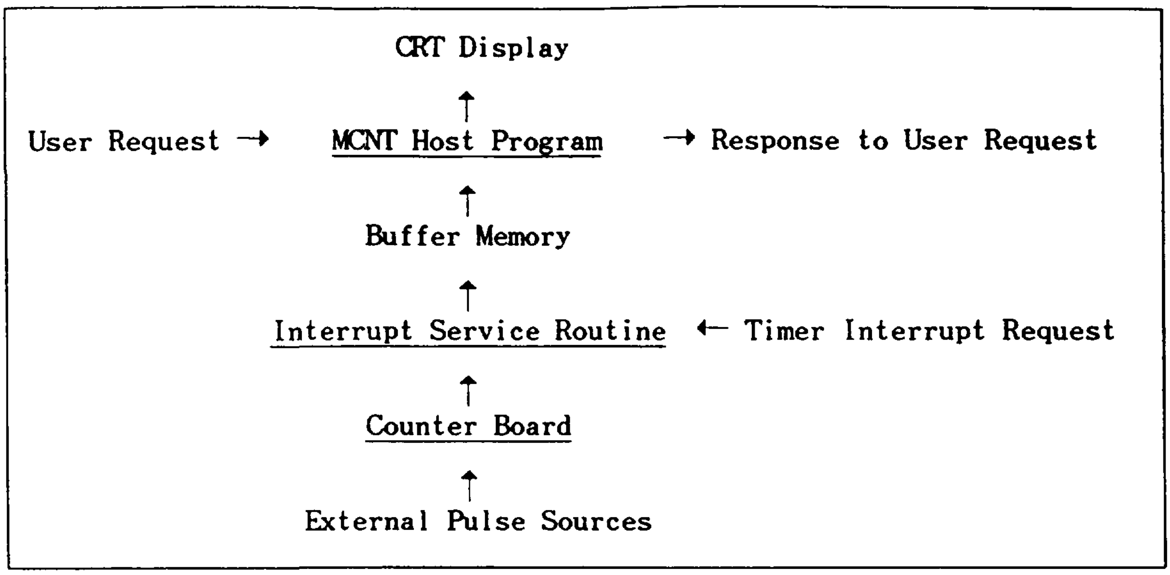

Fig. 2. The Working Concept of PC Multi-Counter

The basic working concept of the multi-counter is shown in Fig. 2. The timer interrupt request is generated at every 1/50 second. Its interrupt source which can be used for this function, can be a timer of the PC system itself, one on the counter board, or regular external triggering pulses. Should the time be kept very precisely, higher priority interrupt source should be used. For the case of IBM PC clone, the system timer interrupt (IRQ0) has the highest priority available to the user. Therefore, the PC system timer interrupt is utilized as a time keeper in this program.

The interrupt service routine whose functioning is synchronized with the interrupt request, latches all working counters at the same time, reads counter register values, and stores them in the circulating buffer memory. This function is common to other programs utilizing counter board except the sampling frequency and number of bits assigned to a count value. For the case of this multi-counter, full 16-bits are used for one count value, and the maximum count rate that can be measured is $2^{16} \times 50$/second ~ 3 Mcps.

The host program checks and reads buffer memory, calculates counts, corrects dead time effect, displays counted data, and responds to the aforementioned user requests comming through the keyboard.

3.2. Multi-Input Multi-Channel Scaler

It is to use a PC for the random neutron process measurement covering up to four neutron counters. There are various methods of the random neutron process measurement such as correlation, variance to mean ratio, count probability, dead time methods, *etc.* Each method measures neutron pulse sequence information and obtains reactor dynamic parameters, but the instrumentation or analysis method of each case is different.

The success of these techniques is dependent on the time resolution of individual count(s) and amount of data. The time resolution should be comparable to the prompt neutron life time (~ μ second in a fast reactor and ~ 100 μ seconds in a thermal reactor), and the data amount should be sufficient enough to ensure statistical reliability. There have been several experimental techniques to satisfy these requirements depending on the instrumentation and reactor type. However, if the count sequence is measured with precision and is saved in the mass storage then numerical simulations for most techniques are possible. The conventional method of this approach is utilizing a sufficiently long record of detector signal on the magnetic tape, which is to be analysed by the computer.

Nowadays, a PC is equipped with enough mass storage and speed for the neutron pulse sequence measurement. The feasibility of PC counter application to the above mentioned approach, was tested at the Korean TRIGA Mk-II reactor using single detector, and its results were analysed by several different methods[1,2]. This test encourages that a PC counter can be a good tool for the pulse sequence measurement in the thermal reactor. For the fast reactor application, however, a special I/O board should be designed for very fast scaling.

The system is now upgraded to accommodate up to four detector signals simultaneously. Its maximum scaling speed is dependent on the number of counters and PC speed, but it can be as fast as 80 μ seconds when an IBM-AT is used, which is fast enough to determine the prompt neutron decay constant of usual thermal reactor. It also has automatic switching feature from the fast to slow scaling for the delayed neutron effect measurement. For the case of fast scaling, the number of bits to save each count value is minimized to relax huge memory requirement since the count probability during one very short time is very low. Current program can scale up to 192 K channels if one or two detectors are used and up to 96 K for three or four detectors. For the case of slow scaling, data are stored in the 16-bit array and channel size is 72 K/(number of counters). As the dwell time between each scaling is only for the data saving to the hard disk, data loss is minimized.

Fig. 3 shows a result obtained by the variance to mean ratio (VTMR) analysis for the scaled data at Korean TRIGA Mk-II reactor with two neutron counters. The same data can be analyzed by correlation, count probability, FFT method, *etc.* Should this experiment be done by the conventional method, it needs two MCAs and a computer. Each MCA scales neutron pulse and sends data to the computer. Nowadays, a PC board type MCA is commercially available and the external data transfer from the MCA to the PC which requires more time than the pulse scaling, is not needed. But a MCA board can receive only one pulse line and the MCA channel size is limited.

Counter-1 in the figure is a fission chamber located at the upper part of the core in the central irradiation hole, and counter-2 is also a fission chamber but located near the outer surface of graphite reflector, which is used as the startup power monitoring channel. The fast scaling was by about 215 μ seconds of Δ for about 1,000 seconds and slow scaling was by about 27.6 milli-seconds for about 4,000 seconds. VTMRs were calculated for all possible grouping of $n\Delta$, *i.e.*, $(1, 2, \ldots, n)\Delta$, $(2, 3, \ldots, n+1)\Delta$, ... groups but not for the simple $(1, 2, \ldots, n)\Delta$, $(n+1, n+2, \ldots, 2n)\Delta$, ... ones, so as to enhance data precision(see Fig. 4). While it generates very precise VTMRs its computing time is too long. For the case of an IBM-AT, it takes several nights for one experimental data analysis.

Dot lines in the figure represents fitted results only for the fast scaled data assuming all delayed neutron terms are linear function of T. Solid lines are fitted for all data to the exact VTMR equation including all delayed neutron terms, and for all variables related with VTMRs - counting efficiency (counts/fission), reactivity, Pu-239 fission portion (assuming fissions are occured only by U-235 and Pu-239), neutron generation time and effective delayed neutron fraction.

It shows something like discontinuity between the fast and slow scaled data which is caused by the different time band of measurement. This trend is amplified if the measuring time is shortened or VTMRs are the smaller

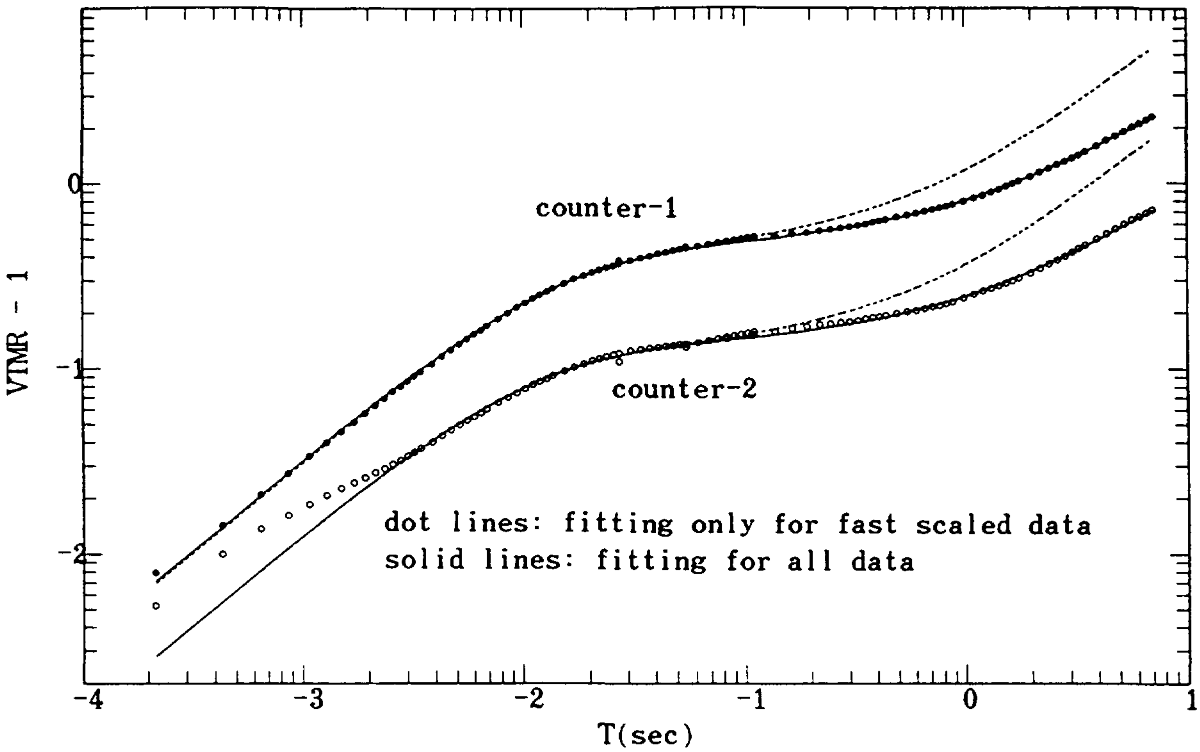

Fig. 3. An Example of VTMR Analysis for the Data Measured by the PC Multi-Scaler

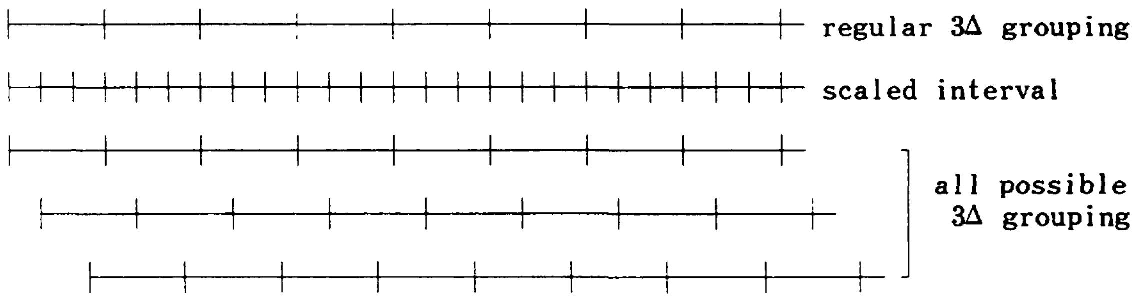

Fig. 4. Data Grouping Concept for VTMR Calculation(T = 3Δ)

by the lower counting efficiency or the higher subcriticality. For the case of this experiment, The counting efficiency of each detector is rather low (less than 3×10^{-5}/fission), but the reactivity (~ -0.1 $) is rather close to the critical status.

As it can scale multiple detector signals simultaneously it is a very useful tool for the analyses of cross-correlation and cross power spectral density (CPSD) especially.

3.3. Real Time Reactivity Measurement

The method of real time reactivity measurement by computer was first proposed in 1959, but its first application was in 1965. At that time and for many years after then, a digital computer had been too expensive to use it for reactivity measurement. Therefore, analog computers had been popularly used for this purpose.

Nowadays, thanks for the recent rapid development of the PC, a low cost PC offers good features as a reactivity computer. The digital computer solves reactivity equations numerically - not by the hardware as the analog computer does but by the software. Its advantages are higher accuracy, easier use, flexibility in data handling, low cost, provisions for the expansion of detector channels, and wide signal dynamic range. Therefore, many digital reactivity computers have been developed and commercial ones are also available.

The usual reactivity computer utilizes one current detector signal fed through the ADC. The counter signal can also be used by similar manner if the square pulse train coming from the discriminator is converted to DC signal by count rate to voltage converter (CRVC). Its advantage is that only an ADC board can be used for counter channel(s). But a computer can receive multiple neutron signals by any combination of analog and pulse if it is equipped with an ADC and counter board (if the number of counter channel is one or two, only an ADC board can be enough since the usual ADC board has few counter channels).

Expanding number of reactivity measuring channels is simple but use of neutron pulse counter raises some difficulty because of the statistical nature of pulse-counting. Counting requires a certain time interval. As the shorter time interval results in the less counting which has the poorer precision, the counting time interval cannot be shortened enough unless the counting rate is very high. At very high count rate, however, the dead time effect is very significant and it is not so easy to correct its effect since its characteristic in actual counting system is very complicated.

The dead time effect can be relaxed by adding a dead time correction circuit at the CRVC and using faster response components but it still has limitation. Actual counting system has at least four or five parts connected in serial - detector, preamplifier, amplifier, discriminator and/or counter - and each has different dead time and characteristics (extendable or non-extendable). Thus it is almost impossible to make a fixed relationship which will be used for the design of an accurate dead time correction circuit. Furthermore, the dead time at very high count rate varies sensitively with the change of discriminator level setting and background gamma level.

When the PC counts pulses directly, aforementioned difficulties still remains unresolved but the dead time effect can be corrected by software instead of dead time correction circuit. Correction by software permits sophisticated function between the count rate and reaction rate. If the experimentally measured dead time is fitted to proper function such as polynomial, it can be easily applied to the program. Its feasibility was tested by single detector[2] at very high count rate. It was successful but its results was rather sensitive to the dead time, which requires very accute dead time information.

If the counting interval is longer, the count rate during the experiment can be lowered to relax dead time effect, but it might cause other problems by the enlongated time interval itself in solving difference equations and the integrated power representation during that interval.

The concept of neutron density (or fission power) variation measured by the count rate is slightly different from that sampled by an AD conversion of the current detector signal. For the latter case, the current is based on sufficient reaction rates in the detector, and the sampled data can be successfully assumed to represent the power at that time. However, the count rate measurement needs at least a certain time interval to count pulses, and its result represents the time integrated power (or averaged power) during that interval but not the spot value. The usual power monitoring by the neutron counter is based on the assumption that this interval is short enough compared to the power variation, but it can be a source of error in the real time reactivity measurement.

The use of counter also imposes strong possibility that the experiment is to be accomplished at the neutron source range (or startup range) where the neutron source is effective. Its effect is the stronger when the count rate is the lower but it can be neglected at the power range where current detectors can be used for the power monitoring. In order to verify these effects - rather long sampling interval, integrated power during the sampling interval in lieu of spot value and source effect - to the reactivity calculation, a series of numerical simulation was carried out for various forms of finite difference approximation of reactivity equations to search an optimum one. It was concluded that the source effect can be corrected by source multiplication measurement, and the effect of integrated power is very small. The effect of rather long time interval can also be minimized by using proper finite difference equation(FDE).

Annex A describes this numerical simulation to test the sensitivty on the time interval and to search an optimum form of FDE. The searched FDE gives very accurate reactivity value even when the sampling interval is 1 second. This equation has been utilized for the actual reactivity computer using multiple neutron counters.

Fig. 5 is an example of reactivity measuring experiment using two fission counters. It can treat up to nine counter signals. Reactivity

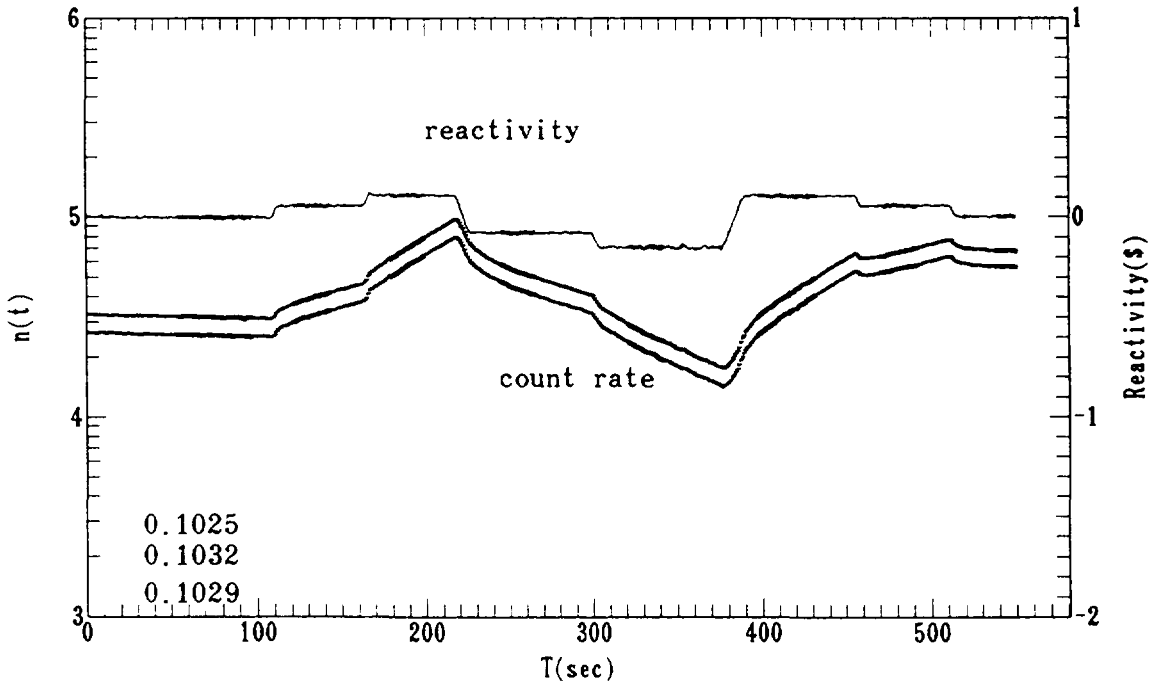

Fig. 5. An Example of Real Time Reactivity Measurement Using PC-Counter

calculation and data display take place once every second, but the actual count rate values used for reactivity calculations, are averaged ones of adjacent two seconds. Lower part curves in the figure drawn by log scale, represent count rate variation, and upper two curves(they looks like as if one line because they are too close) depict reactivity variation. If it reaches the right end of the screen, the screen is swapped half. Thus, the screen always displays at least about 300 seconds' data except when it is just beginning of the experiment. The time averaged reactivity value while it is constant, can be read in digital form by very simple key board manipulation. Those values displayed at the left-bottom corner of the screen are recently averaged reactivity values (unit: $) of each counting channel and the average of all channels(last one). Thicker parts of lines in the figure indicate that the averaging process is(was) occured at that time period. All data related to the experiment are saved in the hard disk, and the experiment can be revived if data checking is needed.

3.4. Control Rod Drop Time Measurement

The control rod drop time can be measured if there is any tool to monitor the drop initiation and termination signals with clock. A memory oscilloscope is a typical tool which can be used for this purpose.

A PC equipped with an ADC can be a better tool for this purpose. The logic is very simple but it needs a program from which drop time can be obtained conveniently. The program made through this work, reads both signals of drop initiation and termination, and graphically displays them on the CRT screen. The user locates cursors at the points of drop initiation and termination and the program displays rod drop time. All data related with this experiment are saved and the experiment can be revived. Fig. 6 shows a sample result of control rod drop time measurement.

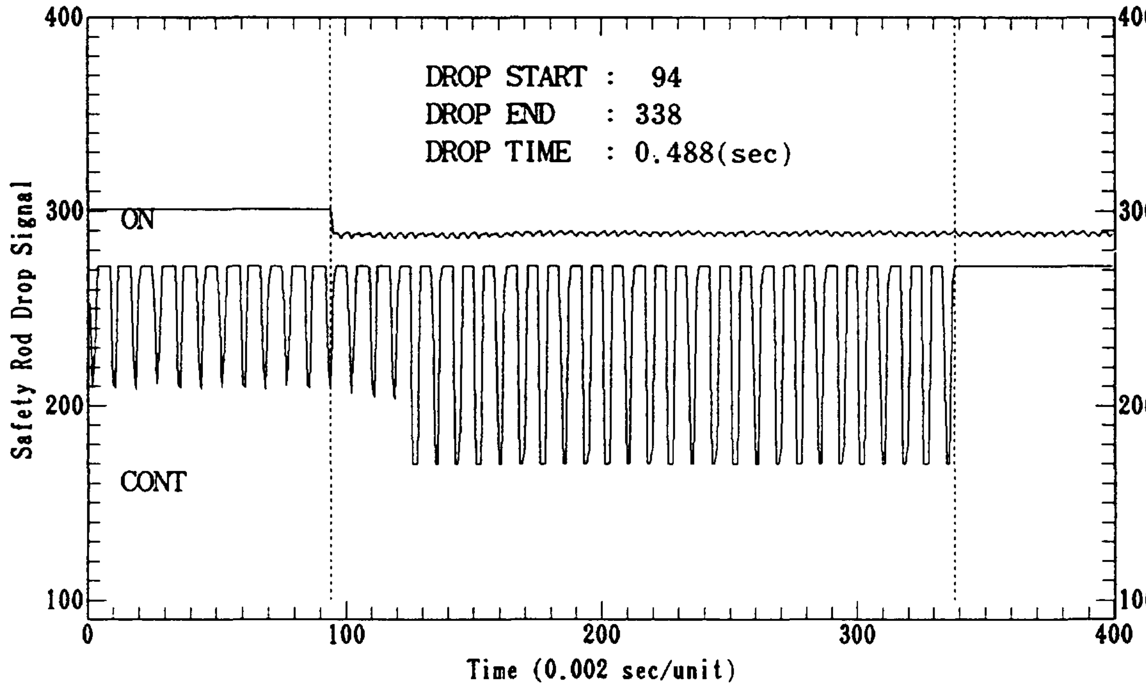

Fig. 6. A Sample Result of Control Rod Drop Time Measurement

3.5. Thermal Power Measurement

For the case of natural convection cooled TRIGA reactors, the reactor thermal power is calibrated by measuring pool water temperature rise. A PC can measure the temperature and power monitor values during the experiment, fit temperature rise, calculate thermal power, and give calibration value for each power monitor.

Fig. 7 is an example of this measurement in Korean TRIGA Mk-III reactor. In this case, eight thermocouples for water and one for pool concrete body temperature measurement, are used. The experiment starts when the water temperature is lower than the concrete temperature by a certain value and automatically terminates when the water temperature rises by about twice of this value.

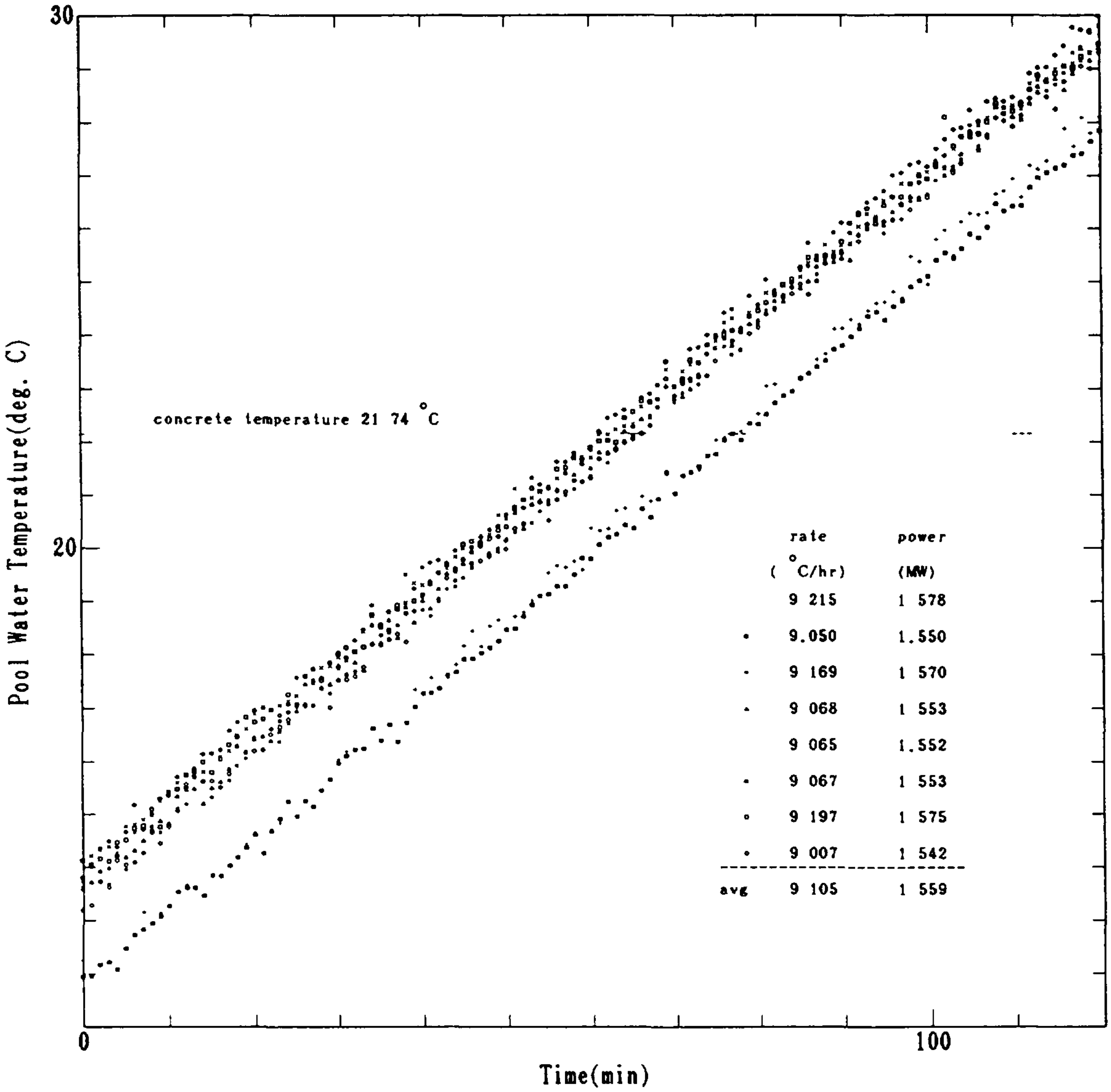

Fig. 7. An Example of Thermal Power Measurement

4. Conclusions

A PC system which can be applied to most of important research reactor experiments is successfully developed. It includes multi-counter, multi-scaler, reactivity computer, control rod drop time measurement and power calibration. So far, it has been focused on identifying and solving any problems or difficulties when counters are used[3,4], since the analog data acquisition is relatively well known.

The multi-counter and multi-scaler are basic tools for many kinds of research reactor experiments which replace conventional counter modules and MCAs.

The reactivity computer can accommodate virtually all reactor power monitors. As the startup channels can also be used, it can give reactivity values at the earliest stage of new reactor commissioning and can be used in critical assemblies as well.

The control rod drop time measuring program which replaces the use of memory oscilloscope, helps operators make a neat report for this experiment with graphic.

The power calibration program fully automatizes this experiment offering improved accuracy and conveniency.

As many conventional instruments are replaced by a PC and experiments are computerized, it is cost effective, improves accuracy, expands experiment field, and unburden operators' duty.

The system will be improved further covering operational data acquisition and control functions as well as some other experiments,

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Annex A

NUMERICAL SIMULATION OF REACTIVITY MEASUREMENT

1. Introduction

If the reactivity is measured by a digital computer the reactivity equations should be converted to finite difference form. In such case there can be several forms of finite difference equations.

In general, the shorter sampling interval gives the more accurate result, but the sampling speed is sometimes limited by some reasons. While a digital computer is used for the experiment, data saving for further analysis or revival of experiment is one of the merit which should not be missed. If the data size is too large, however, data handling itself is not so easy even though the modern PC has farely good auxiliary memory. One should consider that the faster sampling generates more data and requires larger memory. And when the neutron detector is pulse counter the sampling speed is also limited by the statistical nature of counting itself.

Therefore, it is worthwhile to testing each finite difference form of reactivity equation whether it gives accurate result by resonable sampling interval. This test can be accomplished numerically. At first, we assume reactivity variation simulating an actual experiment, estimate neutron density variation by point kinetics, and calculate reactivity by inverse point kinetics. By comparing calculated reactivity with initially assumed values, we can select the best one.

This test requires two procedures - solving point kinetics and inverse point kinetics. This report describes numerical models tested for this purpose. Its main concern has been how to get more accurate result by longer time interval.

2. Numerical Solution of Point Kinetic Equations

The point kinetic equations formulated by reactivity (ρ) and neutron generation time (Λ), are as following;

$$\frac{dn}{dt} = \frac{\rho - \beta}{\Lambda} n + \Sigma \lambda_i C_i + s \qquad (1)$$

$$\frac{dC_i}{dt} = \frac{\beta_i}{\Lambda} n - \lambda_i C_i \qquad (2)$$

with initial conditions of known $n(0)$ and $C_i(0) = \beta_i n(0)/\Lambda\lambda_i$.

One of the safe approach to convert Eq.s 1 and 2 to finite difference form, is trapezoidal integration from t to $t+\Delta$ (Δ is time interval).

$$n(t+\Delta) - n(t) = \{[\rho(t+\Delta) - \beta]n(t+\Delta)/\Lambda + [\rho(t) - \beta]n(t)/\Lambda + \Sigma \lambda_i C_i(t+\Delta) + \Sigma \lambda_i C_i(t) + s(t+\Delta) + s(t)\}\Delta/2$$

$$n(t+\Delta)\{1 - [\rho(t+\Delta) - \beta]\Delta/(2\Lambda)\} = n(t) + \{[\rho(t) - \beta]n(t)/\Lambda + \Sigma \lambda_i C_i(t+\Delta) + \Sigma \lambda_i C_i(t) + s(t+\Delta) + s(t)\}\Delta/2 \qquad (3)$$

$$C_i(t+\Delta) - C_i(t) = \{(\beta_i/\Lambda)\,[n(t+\Delta) + n(t)] - \lambda_i[C_i(t+\Delta) + C_i(t)]\}\Delta/2$$

$$C_i(t+\Delta)[1 + \Delta\lambda_i/2] = C_i(t)[1 - \Delta\lambda_i/2] + (\beta_i/\Lambda)\,[n(t+\Delta) + n(t)]\Delta/2 \qquad (4)$$

As Eq.s 3 and 4 are simple linear simultaneous equations and $C_i(t+\Delta)$ can be expressed only by one unknown variable $n(t+\Delta)$ from Eq. 4, they can be explicitely solved if $C_i(t+\Delta)$s in Eq. 3 are replaced by those from Eq. 4. However, in such case Eq. 3 becomes very complicated. Therefore, an iterative approach is used as following;

Assume $C_i(t+\Delta)^0 = C_i(t)$

→ Calculate $n(t+\Delta)^0$ [Eq. 3]

→ Iterate: $C_i(t+\Delta)^j$ [Eq. 4] → $n(t+\Delta)^j$ [Eq. 3]

no

$|[n(t+\Delta)^j - n(t+\Delta)^{j-1}]/n(t+\Delta)^j| <$ Convergence Limit

yes

End of iteration

The primary requirement to solve Eq.s 3 and 4 is that $1 - [\rho(t+\Delta) - \beta]\Delta/(2\Lambda)$ in the left hand side of Eq 3 should be greater than 0, otherwise $n(t+\Delta)$ has negative value or diverges

$$1 - [\rho(t+\Delta) - \beta]\Delta/(2\Lambda) > 0 \qquad (5)$$

Above condition is satisfied whenever $\rho \leq \beta$, *i.e.* prompt critical or subcritical. When it is prompt supercritical the primary requirement is that $\Delta < 2\Lambda/(\rho - \beta)$.

Above requirement is just to prevent failure for solution but does not guarantee accuracy. Of course, the shorter Δ gives the more accurate result, but Δ should not be too short to give resonable accuracy.

The sensitivity of accuracy on Δ is tested and its results are depicted in Fig. 1. At first, the reactivity variation versus time is assumed as the top figure which simulates zig-zag movements of control rod. Its ramp rate is ±\$0.05/second and reactivity range is

$\pm$\$0.2. Neutron density n(t) variation due to the reactivity swing is calculated by different Δs of 0.01, 0.05, 0.2 and 1.0 second.

The top figure also shows n(t) variation calculated by 0.01 second time interval. Next three figures are relative differences of n(t) by 0.05, 0.2 and 1.0 second to that calculated by 0.01 second.

Results by 0.05 second show less than 0.03 % difference just after the reactivity transient and almost no difference after 0.25 second from the transient. Results by 0.2 second show less than 0.06 % difference just after the reactivity transient and the difference decreases gradually. Results by 1.0 second show very slowly increasing trend of difference as reactivity swing is repeated, but differences are less than 0.5 % within 150 seconds.

Above results indicate that n(t) variation during usual reactivity measuring experiment can be accurately estimated by iterative solution of Eq.s 3 and 4 if Δ is around 0.1 second.

3. Numerical Simulation of Reactivity Measurement

As n(t) variation can be accurately estimated by previous procedure, this value can be used instead of measured n(t) for the simulation of reactivity measuring experiment.

If Eq. 1 is rearranged to calculate reactivity from measured n(t), it is changed to following reactivity equation:

$$\frac{\rho(t)}{\beta} = 1 + \frac{\Lambda}{\beta}\frac{1}{n}\left(\frac{dn}{dt} - \Sigma\lambda_i C_i - s\right) \qquad (6)$$

As n(t) is measured by the computer $\rho(t)$ is easily calculated if $C_i(t)$s are known. Traditionally, $C_i(t)$ is calculated by following Eq. 7 and the combination of Eq.s 6 and 7 has been called as the inverse point kinetics.

$$C_i(t) = (\beta_i/\Lambda)\int_{-\infty}^{t} n(t')exp[-\lambda_i(t-t')]dt' \qquad (7)$$

If it is assumed that $C_i(t)$ is known and $C_i(t+\Delta)$ is to be calculated, Eq. 7 can be modified to,

$$C_i(t+\Delta) = exp(-\lambda_i\Delta)\left[C_i(t) + (\beta_i/\Lambda)\int_0^{\Delta} n(t+\tau)exp(\lambda_i\tau)d\tau\right] \qquad (8)$$

where, τ = t'-t.

The form of finite difference equation for Eq. 8, depends on how to treat integration term(let's it be I) in the right hand side.

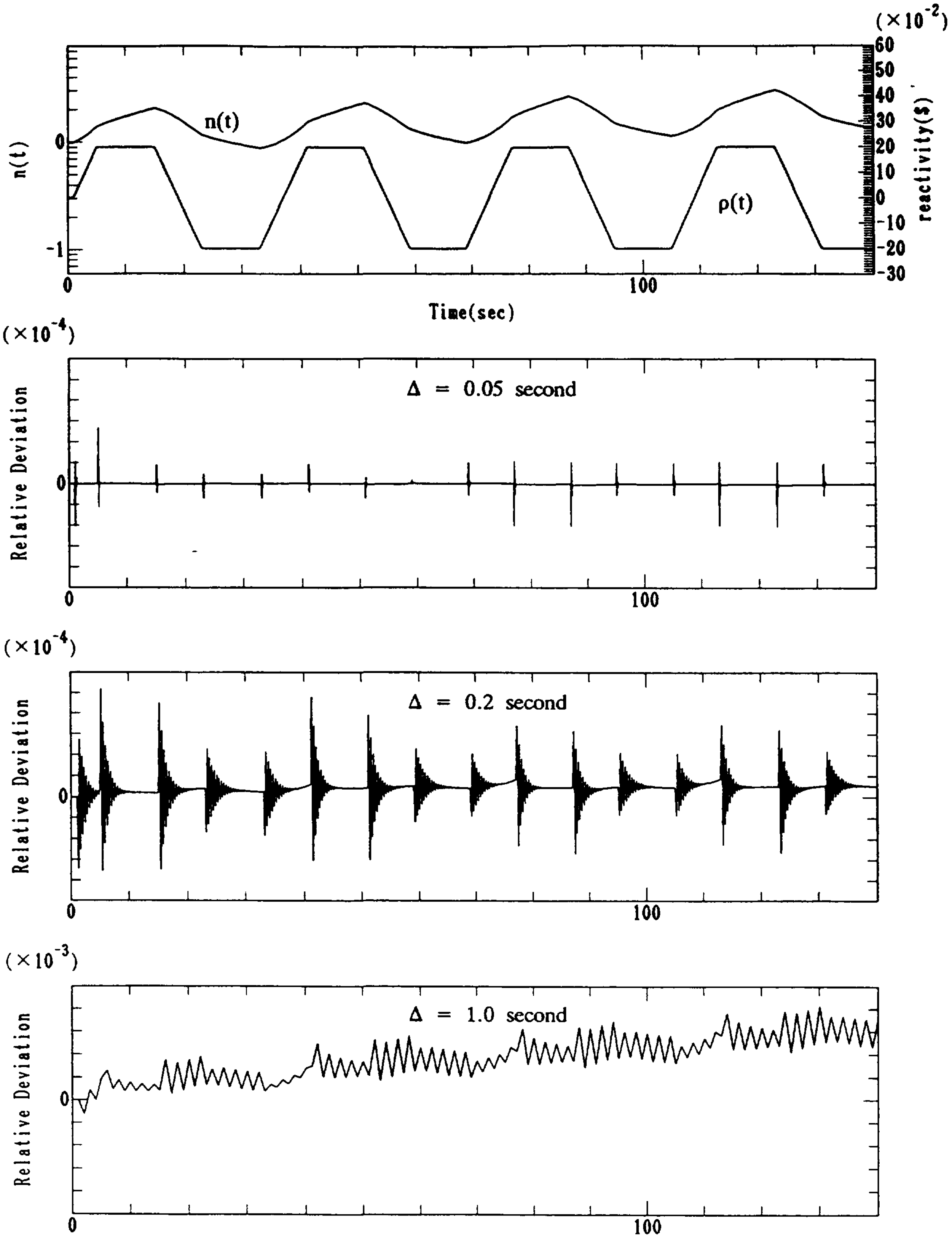

Fig. 1. Trend of n(t) deviation in finite difference approach to point kinetics

Table 1. Maximum Error of Each Case
(Δ in second and error in \$)

Δ	Case 1	Case 2	Case 3	Case 4	Case 5
0.2	-1.98E-4	-1.97E-4	1.04E-3	-1.83E-4	-1.83E-4
0.5	-3.23E-4	-3.19E-4	1.02E-2	-1.58E-4	-1.78E-4
1.0	6.37E-4	9.55E-4	4.58E-2	3.60E-4	4.61E-4

Case 1. Assume $n(t+\tau) = n(t)\, exp(\alpha\, \tau)$

As $n(t+\Delta) = n(t)\, exp(\alpha\, \Delta)$,

$$\alpha = ln[n(t+\Delta)/n(t)]\, /\Delta \qquad (9)$$

$$I = [n(t+\Delta)exp(\lambda_i\Delta) - n(t)]/(\alpha + \lambda_i) \qquad (10)$$

$$\rho(t+\Delta)/\beta = 1 + \{\alpha - [\Sigma\, \lambda_i C_i(t+\Delta) + s(t+\Delta)]/n(t+\Delta)\}/(\beta\, /\Lambda) \qquad (11)$$

Case 2. Assume $n(t+\tau) = n(t)(1 + a\tau)$

As $n(t+\Delta) = n(t)(1 + a\Delta)$,

$$a = [n(t+\Delta)/n(t) - 1]/\Delta \qquad (12)$$

$$I = \Big[n(t+\Delta)exp(\lambda_i\Delta) - n(t)\{1 + a[exp(\lambda_i\Delta) - 1]\}/\lambda_i\Big]/\lambda_i \qquad (13)$$

$$\rho(t+\Delta)/\beta = 1 + [a\, n(t) - \Sigma\, \lambda_i C_i(t+\Delta) - s(t+\Delta)]/[n(t+\Delta)\, (\beta\, /\Lambda)] \qquad (14)$$

Case 3. Assume $n(t+\tau)exp(\lambda_i\tau) = n(t)(1 + b\tau)$

$$I = \Delta[n(t) + n(t+\Delta)exp(\lambda_i\Delta)]/2 \qquad (15)$$

This assumption is just for the convenience of integration, but b cannot be consistent because it is dependent on λ_i.

Eq. 7 is not the only one form of equation to calculate $C_i(t)$s. Eq. 2 which describes the delayed neutron precursor balance, can be used directly. In this case Eq. 2 is integrated from t to $t+\Delta$ in order to calculate $C_i(t+\Delta)$ from the previous step value $C_i(t)$.

$$C_i(t+\Delta) - C_i(t) = \int_t^{t+\Delta} (\beta_i/\Lambda)\, n(t') - \lambda_i C_i(t')dt' \qquad (16)$$

The variation of $C_i(t')$ during the integration interval can be assumed to be linear since its variation is slow, but n(t) variation can be assumed to be exponential or linear as those cases of 1 or 2.

Case 4. Assume $n(t+\tau) = n(t)\, exp(\alpha\,\tau)$

With the same conditions of case 1,

$C_i(t+\Delta)[1 + \Delta\lambda_i/2] = C_i(t)[1 - \Delta\lambda_i/2] + (\beta_i/\Lambda)\,[n(t+\Delta) - n(t)]/\alpha$ (17)

When $n(t+\Delta)$ is almost the same value with $n(t)$ the last term of right hand side approaches to 0/0, which causes overflow error or rather big error in division calculation. Therefore, should $\alpha\Delta$ is very small, the last term can be approximated to $exp(\alpha\Delta) = 1 + \alpha\Delta$, which results in the same approximation with case 5. When $|\alpha\Delta| \leq 0.01$, the difference between $exp(\alpha\Delta)$ and $1 + \alpha\Delta$ is less than 10^{-6}.

Case 5. Assume $n(t+\tau) = n(t)(1 + a\tau)$

With the same conditions of case 1, the finite difference equation for delayed neutron precursors is the same with Eq. 4.

Above five cases are tested for different Δs, and results are depicted in Fig.s 2 ~ 4 and summarized in Table 1. Only case 3 resulted rather big error but the other cases are not so much different. If only the maximum error is compared, all cases except case 3 show almost the same result when $\Delta = 0.2$ second but when Δ is 0.5 or 1.0 second cases 4 and 5 results about a half error of cases 1 and 2. In general, case 4 shows the best result followed by cases 5, 1 and 2. These trends indicate that Eq. 2 is more proper than Eq. 7, and the exponential approximation of n(t) during the time interval is better than the linear approximation.

If case 4 is used and the sampling time interval is 1.0 second then the maximum error is about 0.04 cents which is much smaller than other errors caused by n(t) measuring, delayed neutron data, noise, *etc.*

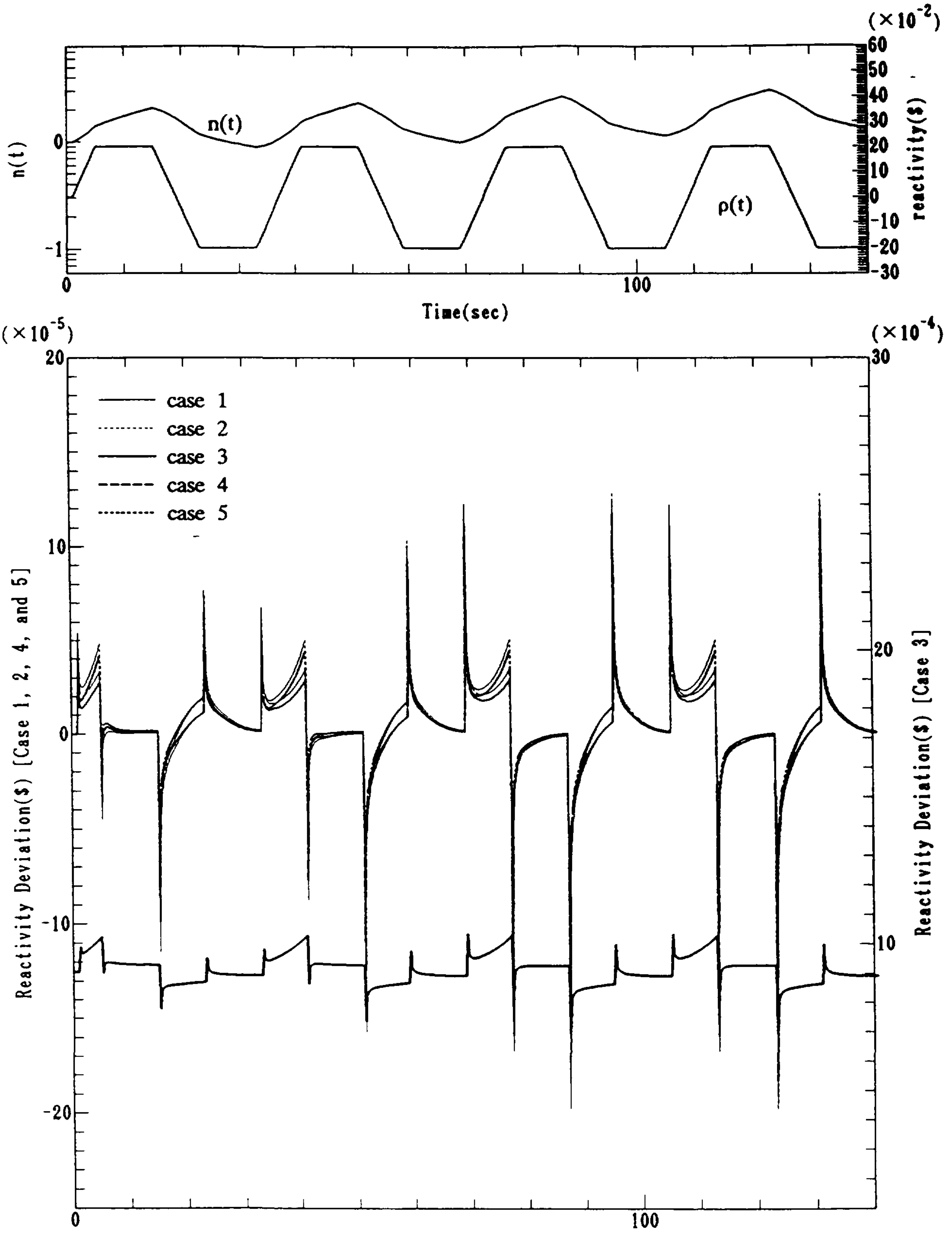

Fig. 2. Reactivity error of each case(Δ = 0.2 second)

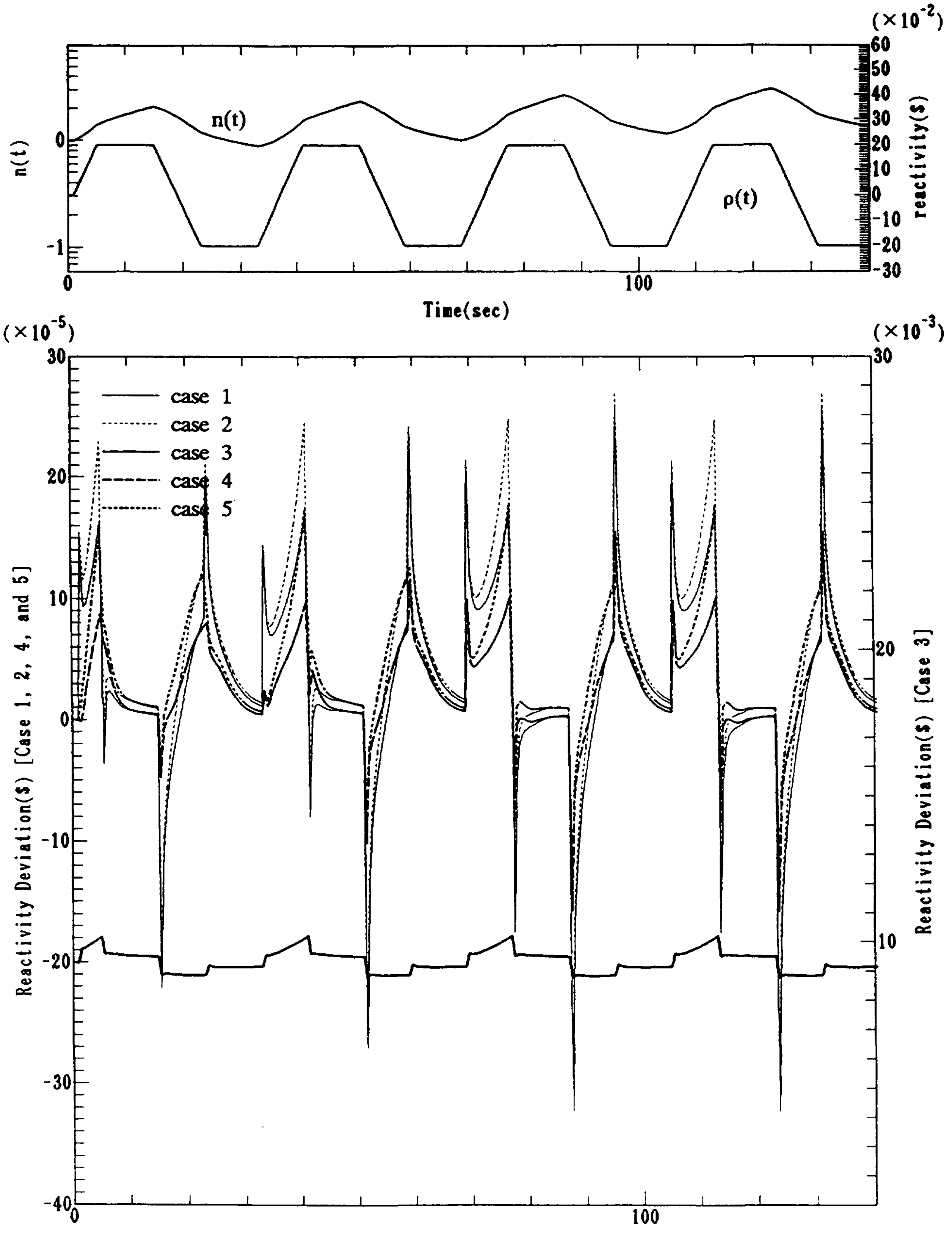

Fig. 3. Reactivity error of each case(Δ = 0.5 second)

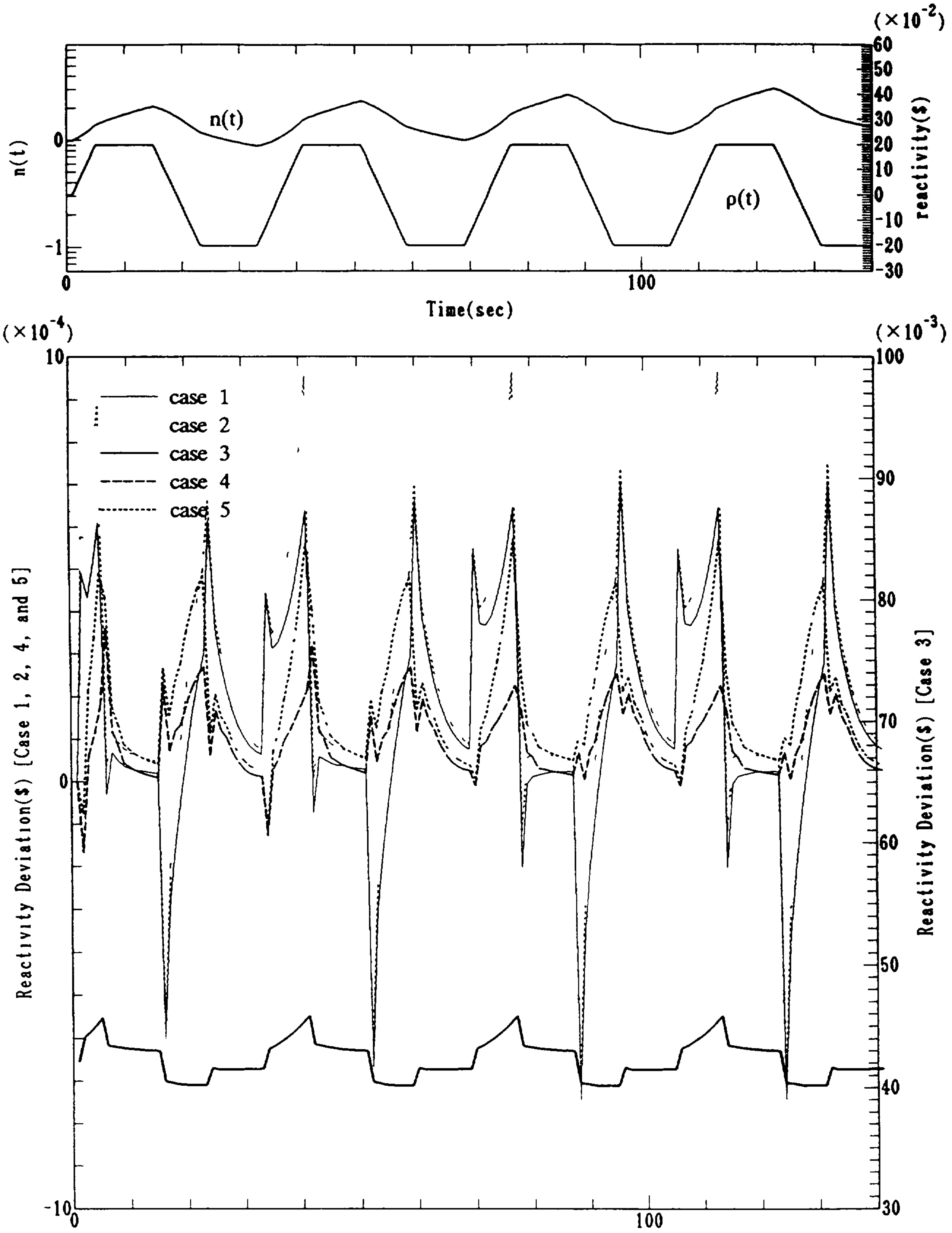

Fig 4 Reactivity error of each case(Δ = 1.0 second)

DEVELOPMENT OF A CENTRAL PC-BASED SYSTEM FOR REACTOR SIGNAL MONITORING AND ANALYSIS*

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Abstract

A personal computer based system was developed for on-line monitoring, signal processing and display of important reactor parameters of the Pakistan Research Reactor-1. The system was designed for assistance to both reactor operator and users. It performs three main functions. The first is the centralized radiation monitoring in and around the reactor building. The computer acquires signals from radiation monitoring channels and continuously displays them on distributed monitors. Trend monitoring and alarm generation is also done. In case of any abnormal condition the radiation level data is automatically stored in computer memory for detailed off-line analysis. In the second part the computer does the performance testing of nuclear instrumentation channels by signal statistical analysis, and generates alarm in case the channel standard deviation error exceeds the permissible error. Mean values of important nuclear signals are also displayed on distributed monitors as a part of reactor safety parameters display system. The third function is on-line computation of reactor physics parameters of the core which are important from operational and safety points-of-view. The signals from radiation protection system and nuclear instrumentation channels in the reactor were interfaced with the computer for this purpose. The development work was done under an IAEA research contract as a part of coordinated research programme.

1. INTRODUCTION

Pakistan Research Reactor-1 (PARR-1) was converted to Low Enriched Uranium (LEU) fuel and its power upgraded from 5 MW to 10 MW. The power upgradation demanded enhanced capability for monitoring of radiation levels and core parameters for reactor and personnel safety [1]. To meet these requirements a personal computer (PC) based system, interfaced with PARR-1, was developed for on-line, centralized monitoring and evaluation of information related to radiation levels in and around the reactor building, performance testing of nuclear instrumentation channels and calculation of reactor physics parameters. The project work was performed under IAEA research contract No. 6048/RB and its extensions as a part of the coordination research programme on the application of personal computers to enhance the operation and management of research reactors.

The PC-based system performs three main functions:

a) On-line acquisition of radiation signals from radiation protection channels and processing of the digitized data.

b) Acquisition of signals from nuclear channels in real-time and error analysis of these signals using statistical techniques.

c) On-line calculation and display of reactor physics parameters using data acquired from nuclear channels.

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* Research carried out in association with the IAEA under Research Contract No. RAK/6048.

Real-time graphic display of radiation levels in and around the reactor are provided on video terminals placed at key locations in the reactor hall. The display is in the form of trend monitoring and histograms. The computer continuously compares actual radiation levels with the alarm settings and generates 'alert' and 'action' alarms if any radiation signal exceeds the preset limit. Under alarm conditions the data is automatically stored on the mass storage device of the computer and is available for detailed off-line analysis.

For channel performance evaluation, the computer calculates and displays mean values, statistical errors and error distributions of signals originating from nuclear instrumentation channels. In case the standard deviation error of a particular signal exceeds the reference error by a fixed margin, an alarm is generated indicating some malfunction in the channel performance.

The reactor parameters computed by the computer are used in the prediction of control rods position at criticality during reactor startup, measurement of control rod worth, reactivity calculations and computation of thermal power.

The computer-based system was installed in the control room of PARR-1. The basic system architecture is shown in Fig. 1. It will eventually replace most of the functions of the old PDP 11/23 plant computer. The system was designed as user-friendly and provides the operator and reactor users a quick overview of the status of the reactor and its instrumentation, alongwith the radiation levels. Some other main advantages are:

i) The data storage capability of the computer provides the means for archiving and off-line data analysis in abnormal condition. The computer controlled system minimizes the use of chart recorders, thus saving on unnecessary recording and frequent mechanical maintenance problems

ii) The channel performance evaluation algorithm, displays the mean values of important nuclear parameters. This part therefore also serves as the plant safety parameters display system (SPDS).

iii) Based on signal error analysis a faulty instrumentation channel may automatically be identified by the computer which facilitates maintenance scheduling.

iv) On-line reactor parametric calculations, result in a better knowledge of the behaviour of reactor core and also save time and labour.

2. INTERFACING OF REACTOR INSTRUMENTATION SIGNALS WITH PC

The radiation protection system of PARR-1 consists of 9 radiological channels measuring radiation levels in different active areas in and around the reactor building. Most of these channels were interfaced with the PC. Other nuclear and process channels which could be connected with the computer are:

- Startup Countrate A and B
- Startup Period A and B
- Linear Flux A and B
- Logarithmic Power A and B
- Logarithmic Power Period A and B

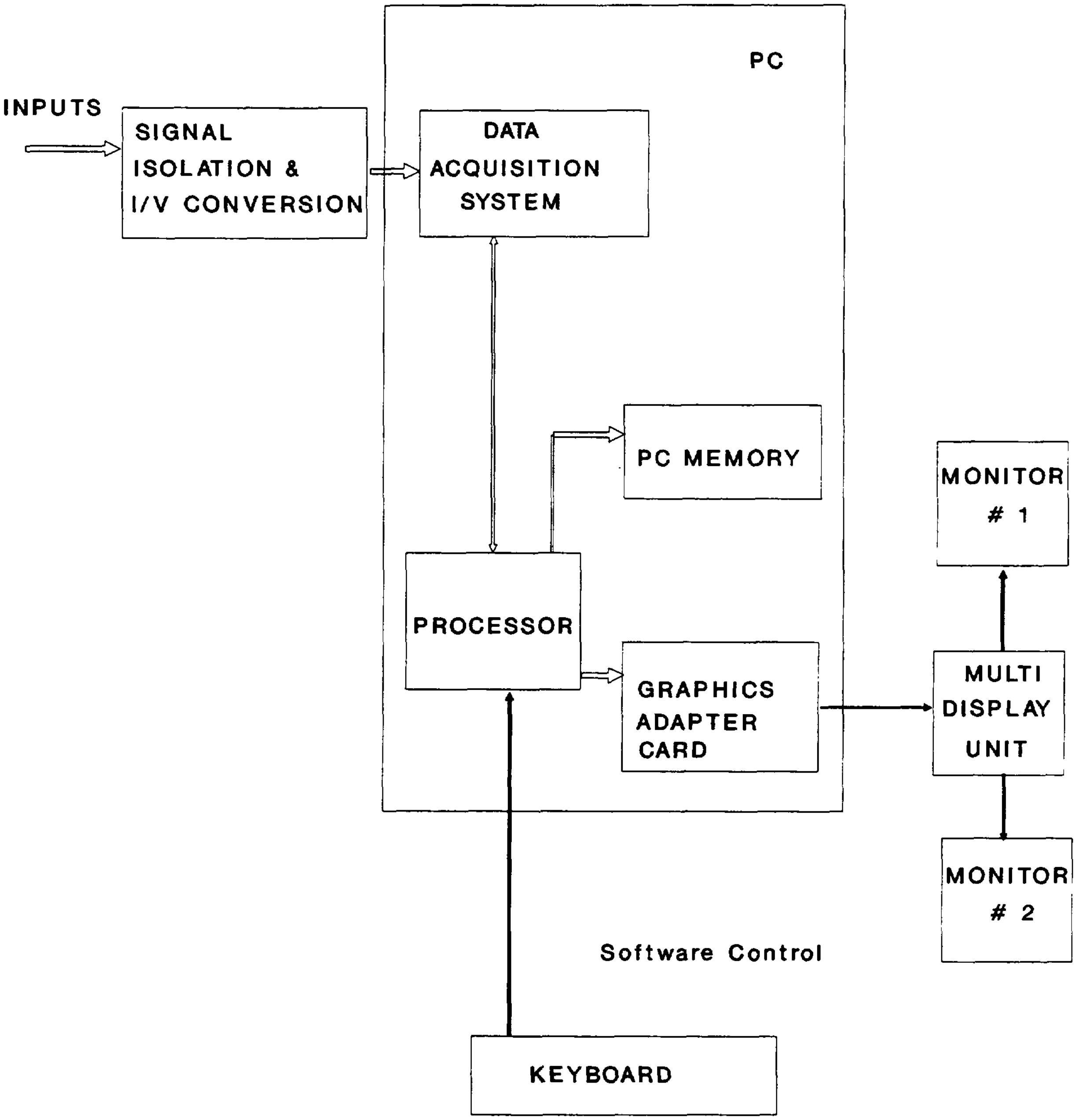


FIG. 1. Block diagram of basic system architecture.

- Nitrogen-16 Power Channels A, B and C
- Coolant Temperature A and B
- Coolant Flow Channels A, B and C

Additional electronics will be required to expand the capacity of the interface card if it is desired to connect all the above mentioned channels.

2.1. Signal isolation

One of the essential requirement of interfacing of reactor signals with the computer is that any malfunction in the computer must not be transmitted to the reactor instrumentation. This was achieved by providing proper isolation between the reactor signals and the computer interface. Conditioning of the signals from the instrumentation channels was carried out prior to their input to the computer. Isolation amplifiers [2] with auxiliary protection circuits were used for this purpose to provide full protection to channel output signal even shorts to ac line voltages from the computer side, or if the signal connection to the computer is inadvertently grounded. Protection circuits were also installed on the input to the interface card, which are particularly important for signals originating from the field transmitters so as to avoid the possibility of any damage to the interface card and the computer.

2.2. Data acquisition system

Acquisition of analogue signals was performed with the help of an analogue-to-digital converter (ADC) card, plugged into the personal computer. The selection of ADC card was made on the basis of high resolution, fast A/D conversion speed, low sampling error and multiple signals acquisition. The ADC card used in the system had 16-input and 12-bit resolution capability with high sampling rate [3]. Its other features include a variable conversion gain and low sampling error. The specifications of data acquisition card are described in the next section on system hardware.

For a signal amplitude of 0 to 10V, a 12-bit resolution yields a sensitivity of 2.4 mV, which is good for detecting small changes in the signal amplitude. A maximum sampling rate of 4000 samples per second can be achieved in BASIC language. Higher sampling rates of 20,000 samples per second can be obtained using special command modes of the ADC card, or with C-language programming. Proper selection of the sampling rate also depends upon the time constant of instrumentation channels. At low current ranges the channel response is very slow (in seconds), but at higher values of input signal, the channel has fast (milliseconds) response time. Based on the experimental measurements in the laboratory, a typical sampling rate of 10 samples/s/channel was selected. The programmable gain feature of the ADC card was very helpful in selecting proper range of the input voltage signals.

2.3. Data acquisition software

Special data acquisition software was developed which performs the following functions:

- Sequentially select input signals for digitization (polling scheme)
- Set the sampling rate and number of samples (block length) for each signal
- Set the ADC gain to be applied to each input signal
- Perform analogue-to-digital conversion of the selected signal and store the digitized data in an array in computer memory

The data acquisition software has been developed in Quick Basic and C- languages [4]. However, as will be discussed later, because of low sampling rate involved, the fast speed execution by C-language is not an essential requirement, and compiled Basic language is effective for most data acquisition and processing purposes.

3. SYSTEM HARDWARE

3.1. Computer specifications

Model	PC/AT Compatible
Processor Type	INTEL 80386-16
Math Coprocessor	80387-16

ROM BIOS	Phoenix VER/1.00
Memory	640 kB extended up to 2 MB
Clock	25 MHz.
Serial Ports	Two RS-232/C
Disk Drives	60 MB hard disk, two floppy drives
I/O slots	Five 8-bit and three 16-bit
Video Adapter	VGA and EGA
Monitor	Colour

3.2. Multiple display hardware

To provide multiple graphic display of radiation levels at different locations in the reactor hall, a signal driver circuit was developed. The driver, which can be used with CGA, EGA and Hercules (monochrome) adapter cards, allows two or more monitors for IBM PC and compatibles to be driven by a single video signal. The multiple display circuit consisted of several octal, tristate, non-inverting buffers. The buffers had low power dissipation and were permanently enabled by an on-board 5V dc power supply. The TTL levels from the computer video signal were applied to the inputs of the buffers and the outputs were fed to the video monitors.

The multiple display circuit was useful for short transmission distances between the computer and the monitors. However, over large distances the signals were corrupted with noise and the images could not be reproduced accurately. A second approach for centralized monitoring and display was adopted using the close circuit TV (CCTV) system of the plant. In this scheme, the composite video signal from the CGA display card of the personal computer was fed to one of the camera inputs of the CCTV control and switching unit, which then distributed the signal over the existing cables to various monitors. By the use of CCTV system, displays are now available at many locations as compared to the multidisplay card developed earlier requiring extensive cabling in the reactor building. Provision of additional display points is also possible in the system as new connections can simply be added to the CCTV system.

3.3. Isolation amplifiers

Isolation amplifiers with the following specification are mounted on the measuring channels.

Input:	0... 10V
Input impedance:	10M ohm
Output:	0...20mA / 0...10V
Max. load resistance:	800 ohm
Isolation voltage:	±1500V
Power supply:	24Vdc
Small signal bandwidth:	400Hz

3.4. Data acquisition card

Analog input channels	16 single ended or 8 differential
Resolution	12-bits
Accuracy	0.01% of reading ±1bit
Full scale	±10,±5,±2.5,±1,±0.5

Maximum overvoltage	±30V
A/D type	Successive approximation
Conversion time	15μs. max / 12μs. max.
Linearity	±1 bit
Power supply	+5V / 1A, +12V / 2mA, -12V / 18mA

4. RADIATION MONITORING

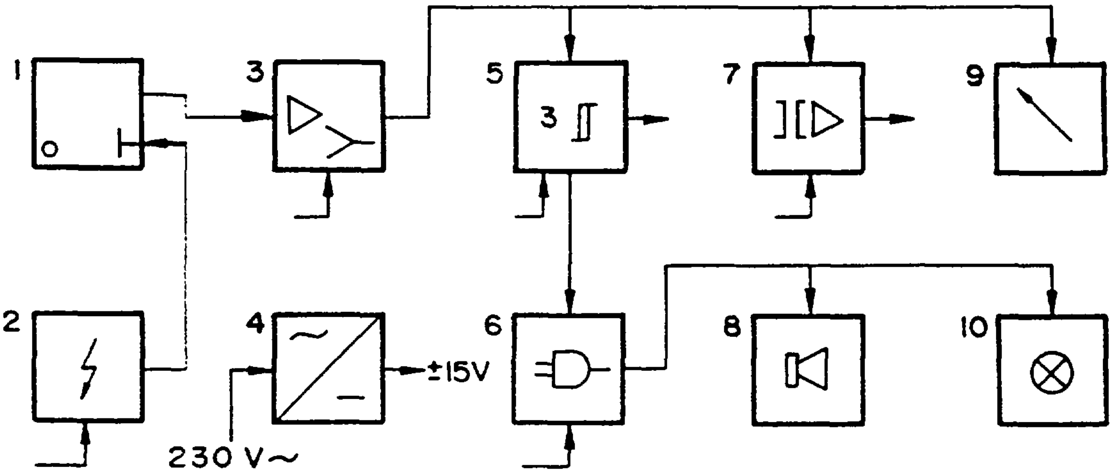
The radiation monitoring channels in PARR-1 use gamma ionization chamber as detector and provide dc voltage output proportional to the logarithm of detector current. The measuring range of all channels is 1 Sv/h. The configuration of a typical radiation monitoring channel is shown in Fig. 2.

Various steps involved in the processing and display of radiation monitoring signals by the computer are described in the following sections.

4.1. Data processing and display algorithm

Routines for data processing and archiving of radiation signals were developed in Quick Basic language [5]. The flowchart for a typical signal acquisition and display algorithm for radiation monitoring is shown in Fig. 3. Following steps are involved in the data processing algorithm.

a) Conversion of A/D levels into engineering units
The data after A/D conversion is in the form of ADC levels, which is transformed into radiation level units by applying appropriate conversion formulae for the particular instrumentation channel.



1. GAMMA CHAMBER
2. HV POWER SUPPLY
3. LOG AMPLIFIER
4. POWER SUPPLY
5. ALARM UNIT
6. CONTROL MODULE
7. ISOLATOR
8. HOOTER
9. INDICATOR
10. FLASHER

FIG. 2. Radiation monitor with ionization chamber.

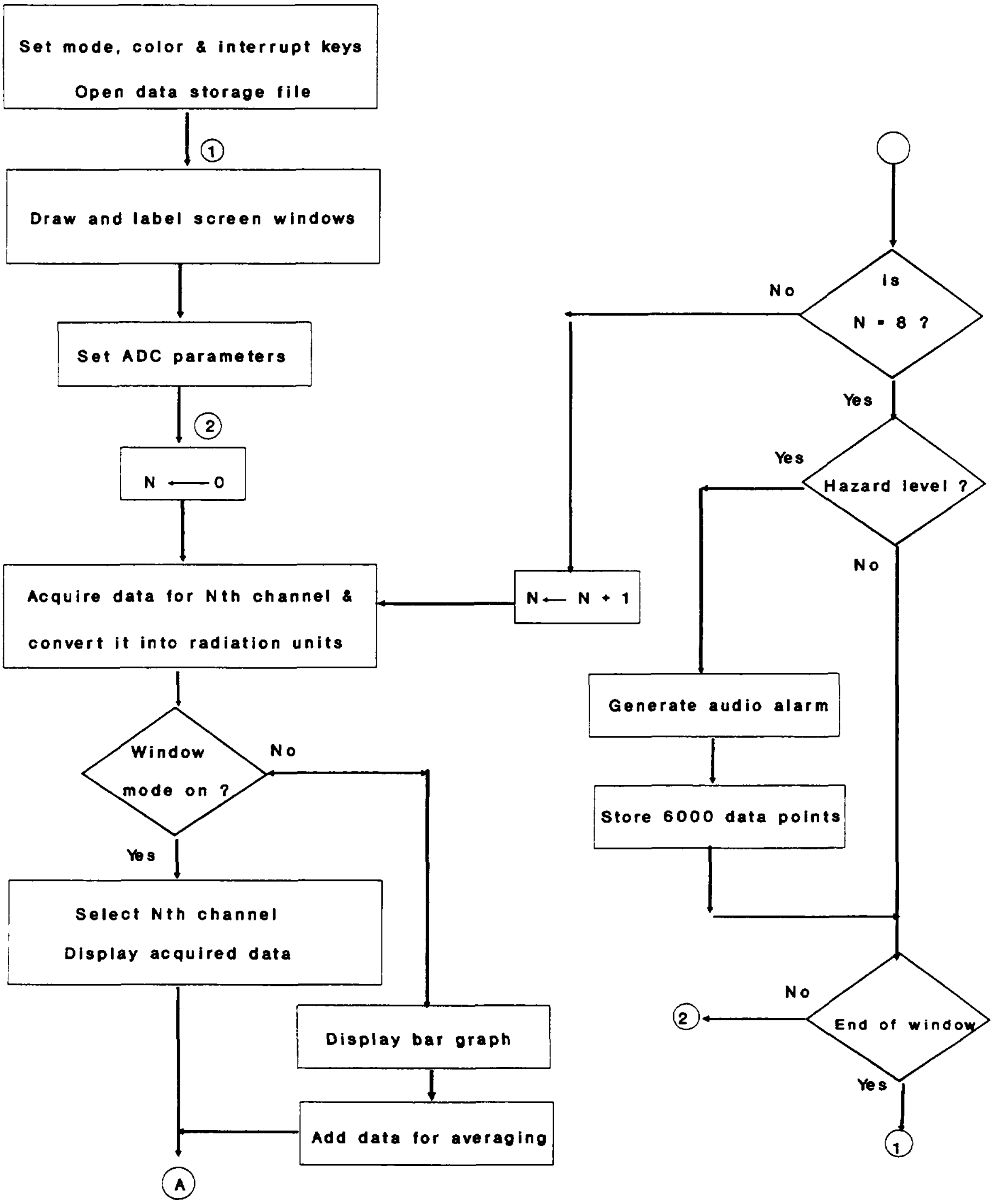


FIG. 3. Flow chart of signal acquisition and display algorithm.

b) Data smoothing
Nuclear radiation signals have inherent random fluctuations present around their mean value, and the digital conversion of analogue signals also causes sampling error. The two errors may result in relatively large standard deviation. To minimize such an error, signal averaging, both simple and moving, is performed on the digitized data.

c) Alarm generation
Data from each channel is compared with the alarm settings of that channel. In case the alarm setting is exceeded, the computer generates appropriate video and audio alarms.

d) Data storage and archiving

It is necessary to store the radiation level data, specially in an alarm condition. A possible utilization of such data is in post-accident analysis. Since the storage of data on computer hard disk is a relatively slow process for on-line applications, efficient storage was achieved in the algorithm by creating virtual disk in the computer RAM, which are simulated disk drives that use a portion of the computer memory as storage medium. The data is stored in the form of a random access file, which is compact and can be accessed quickly which results in considerable saving of computer execution time. Two modes of data storage were implemented in the algorithm; a) automatic storage by the software in case the radiation level of any channel exceeds a predetermined limit, and b) storage on operator's command by pressing a pre-assigned key. The capacity for data storage in both cases is about 10 minutes. Once the virtual disk for data storage is full, a backup file is made automatically on the hard disk and the original file is overwritten with new data.

4.2. Graphic display

Two modes of signal display were implemented through software, i.e. window type display mode and bargraph. Both modes have their own special features and the operator can toggle between the two modes interactively by pressing a pre-assigned function key. In the window display or trend monitoring mode, the screen is divided into four windows where each window displays a different radiation signal in real-time. Such a display is shown in Fig. 4 where the vertical axis indicates the signal amplitude calibrated in logarithmic scale over a span of seven decades and the time information is indicated on the horizontal axis. As the trace displaying the current radiation value reaches the extreme right end of the window, the display is reset for a fresh record. The total time for display of one record depends upon the sampling frequency selected and can vary from 10 seconds to 3 minutes. Each graphic window is labeled according to the radiological channel whose signal is currently displayed along with the alarm setting of the respective radiological channel. In case the signal value of any radiological channel exceeds the preset level, an audio-alarm is generated and a visual alarm indication appears on the corresponding window. The alarm can be reset by the operator but the visual indication remains till the alarm condition persists.

The histogram display mode shows the average radiation doserate for the last 5 seconds. The mean value of 50 data points of each signal is computed and displayed in the form of a dynamic bargraph. The vertical side of the display indicates the signal mean value and is calibrated in logarithmic scale (seven decades) and the horizontal side is the sequential channel number. The bargraph mode is reset for a fresh display after about 5 seconds. A printout of a typical histogram is given in Fig. 5.

5. PERFORMANCE ANALYSIS OF INSTRUMENTATION CHANNELS

For the performance analysis, the computer sequentially acquires the signal from each instrumentation channel, and performs statistical analysis on this data. Signal errors and probability distribution functions are computed and compared with the reference errors.

Standard deviation error of the signal from a nuclear channel gives a handle to determine the performance of the channel. In practice, nuclear detection phenomena have statistical fluctuations around their mean value [6]. The measuring instrumentation also introduces additional error due to the electronic noise. In case of some malfunction in any part of the channel (from the nuclear detector to the output stage), the standard deviation of the channel signal will exceed the nuclear error by a wide margin. Also a zero value of signal mean and standard deviation would indicate an open connection somewhere in the channel circuit.

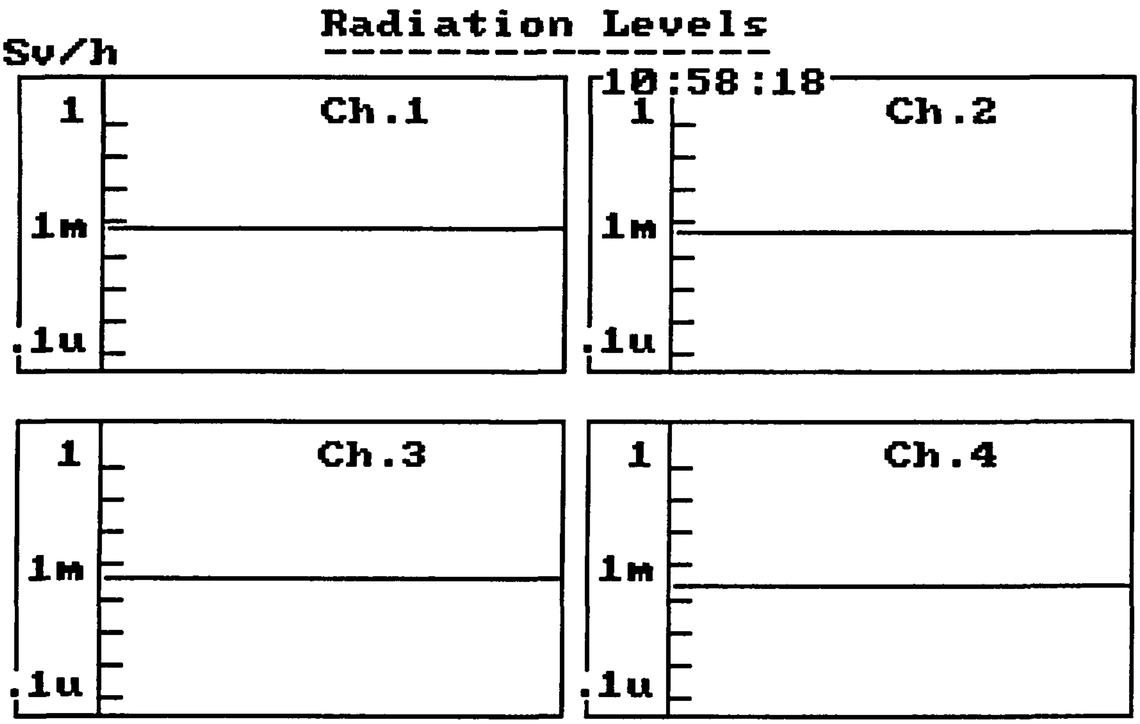


FIG. 4. Window display.

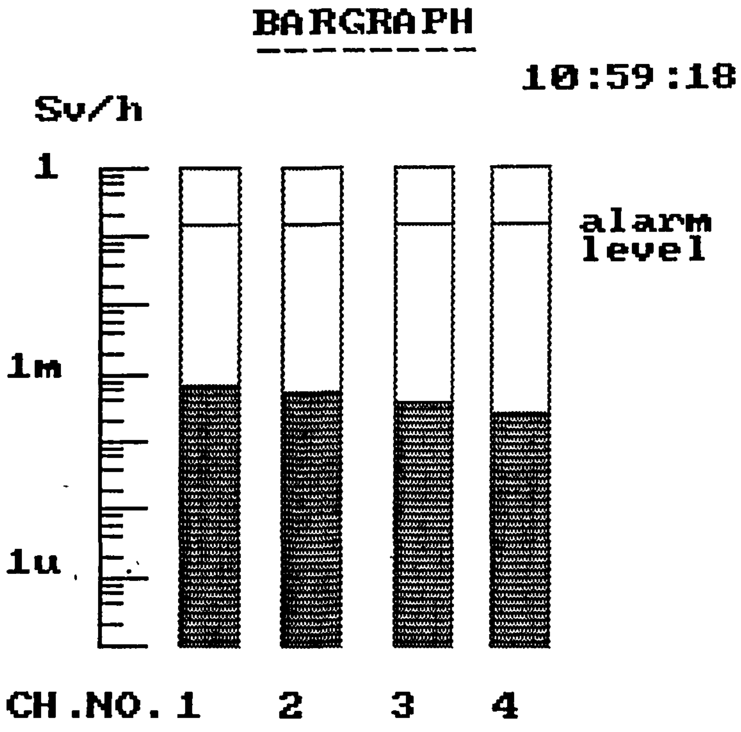


FIG. 5. Histogram display.

Another function of interest for monitoring the nuclear channel performance is the signal probability distribution function [7]. If the full-width-half-maximum of the probability distribution curve exceeds $\pm 3\sigma$, or if the curve has large tails, it is then an indication of increased channel noise. The probability distribution function can also be used to differentiate systematic errors from random errors. For unbiased channel output (no systematic error), the peak of the probability curve would occur at the signal mean value. Any persistent deviation of the peak from the mean value would indicate a positive or negative systematic fault in the channel behavior.

Another main advantage of this method is that the mean values of all nuclear signals are available for display by the computer. A compact information display of these important parameters constitutes the safety parameters display system, which is a requirement in power plants.

5.1. Algorithm for Channel Performance Monitoring

The algorithm developed for channel performance analysis, uses essentially the same data acquisition routines as employed in radiation monitoring algorithm. The number of input signals monitored for channel testing are however much greater than those utilized in radiation monitoring. A total of 12 signals originating from various nuclear instrumentation channels are acquired. These signals are essentially the same as those listed in section 2.

5.1.1. Data acquisition

The flow chart of the channel testing algorithm is shown in Fig. 6. The computer starts a data acquisition and analysis cycle by acquiring first N data points of the first instrumentation channel (Startup Countrate Channel A) signal. The digitized data is stored in an array and data analysis routines are called upon. Once the data analysis of these N points is complete, the data from the next channel is acquired and analyzed. This process continues till all channel signals are analyzed in a sequence. Once the cycle is complete, the channel number counter is reset and the computer starts a new cycle.

5.1.2. Data processing

The digitized data obtained by A/D conversion of each input signal is analyzed by program software. The main functions of the data processing algorithm are as follows:

i. Convert the raw data into engineering units, such as, reactor power (MW), period (s) and countrate (cps) using proper channel transfer function relationships.

ii. Calculate signal mean value, standard deviation and reference errors of N data points of each channel.

iii. Compute the probability distribution function and determine Full-Width-Half-Maximum (FWHM) of the probability function (optional).

iv. Compare the standard error of channel signal with the reference errors of nuclear detection phenomena and generate alarms in the following conditions.

   a) The standard error, σ exceeds nuclear detection error by a fixed margin.
   b) FWHM of the probability function exceeds $\pm 3\sigma$.
   c) Average signal value is zero (open or grounded signal condition).

The optimum sampling interval for signal digitization was set at 0.2 second, corresponding to a sampling frequency of 5 Hz. This value of sampling frequency was a trade-off between maintaining high speed and reasonably low scatter in digitized data. In order to obtain good statistics, the total number of data points for each signal (N) were set at 50. The total data

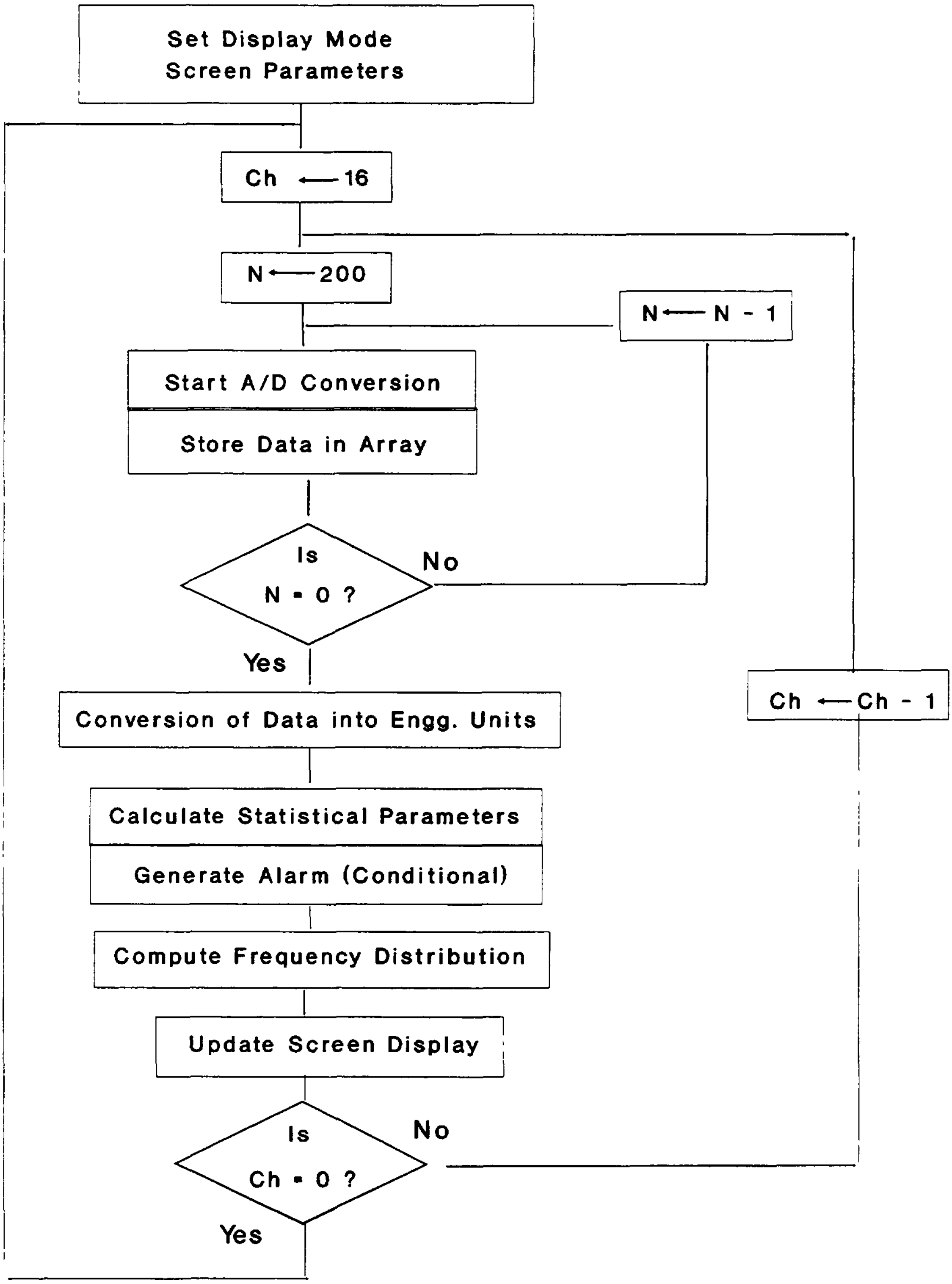


FIG. 6. Flowchart of channel performance testing.

acquisition time for one channel signal is therefore 10 seconds. The total time for one complete cycle for acquisition and analysis of twelve signals is about 2 minutes. It has been observed that the data analysis time was negligible compared to the data acquisition time. In order to avoid unnecessary computation, the frequency distribution plotting is performed by the computer only in case of alarm generated by high standard deviation errors, or by the operator's command.

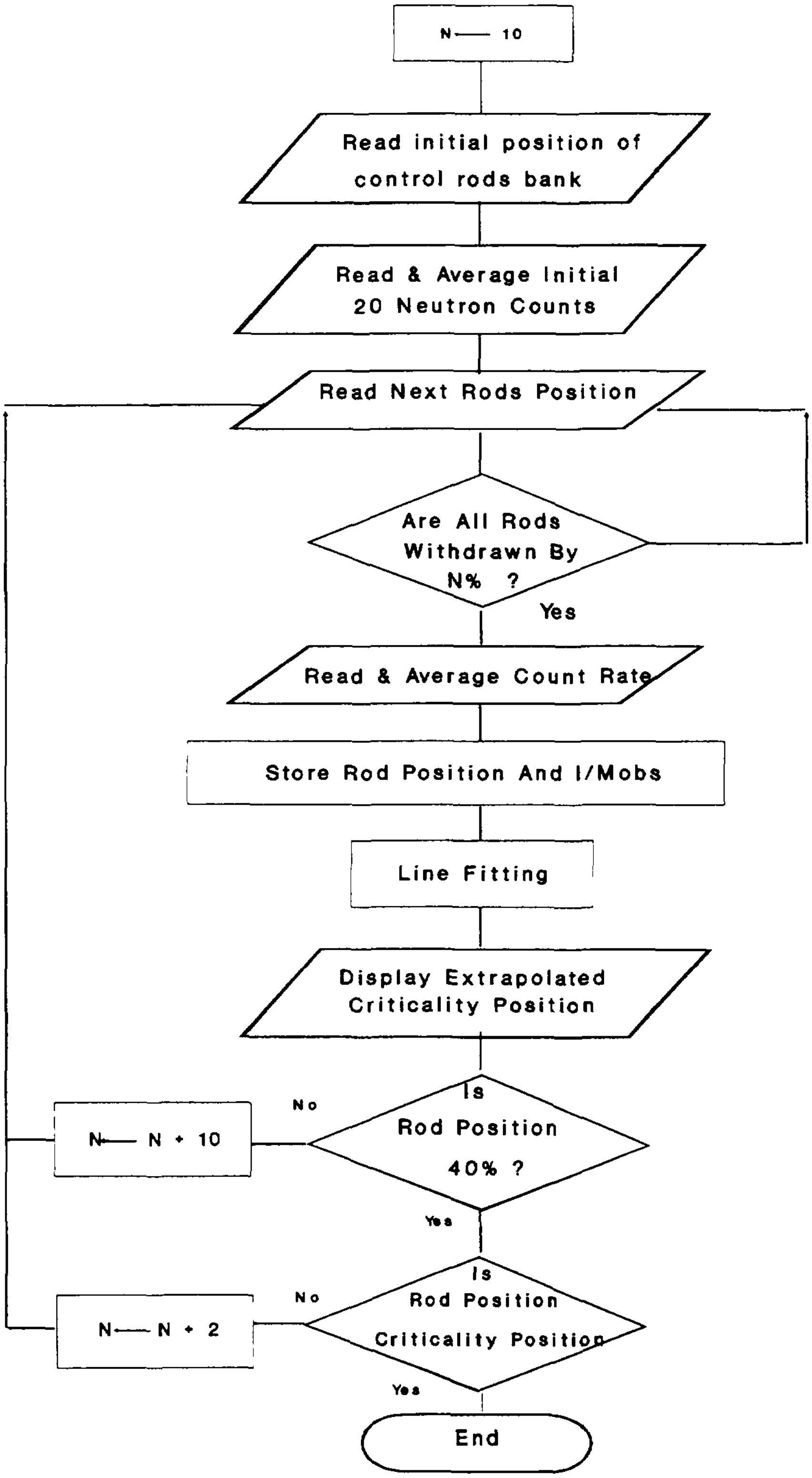


FIG. 7. Flowchart of approach to criticality algorithm.

6. ON-LINE COMPUTATION OF REACTOR PHYSICS PARAMETERS

A number of algorithms were developed which performed on-line computation of important reactor physics parameters using in-situ information available from reactor instrumentation. These algorithms are described in the ensuing sections.

6.1. Approach-to-criticality

During each reactor startup, it is essential to have prior knowledge of either the value of the fuel loaded in the core to achieve criticality, the amount of control rods withdrawal from the core, or the moderator level for which the reactor will become critical. In PARR-1, the operator relies on the last criticality position of control rods as the reference position during each reactor startup. This is however not valid after fresh fuel loading, or any reactivity change due to experiments, etc., prior to reactor startup. An algorithm was developed in which the computer predicted estimated critical rod withdrawal positions during each reactor startup, by analyzing the current neutron count data and the control rod positions.

The approach-to-criticality is determined by the computation of inverse multiplication factor. The relative neutron multiplication, M, at any level of subcriticality is governed by

$$M = C / C_o \qquad (1)$$

where, C is the neutron countrate measured by a detector monitoring neutron flux in the core and C_o the unmultiplied countrate from extraneous neutron source at shutdown state. As the reactor approaches criticality, the neutron multiplication, M tends to go to infinity. It is therefore more convenient to plot $1/M$, which goes to zero at criticality, against the loading parameter. The loading parameter may be the control rod withdrawal, moderator level or fuel load.

6.1.1. Approach-to-criticality algorithm

The flow diagram of the algorithm is shown in Fig.7. The signals acquired in the algorithm are the neutron countrate from the two startup channels and the position indication signals from all shim rods. The program is initiated prior to the reactor startup. Initially, the computer acquires 50 values of neutron countrate from the two startup channels at shutdown stage, and computes C_0. The program then monitors the withdrawal position of all rods. As soon as the rod bank is withdrawn by 10 %, the computer acquires the current value of neutron countrate. The ratio *1/M* is then computed and the program determines extrapolated critical rods position by fitting a line between the current and previous values of *1/M*. This process is repeated in 10% rod withdrawal steps and the estimated control rod position at criticality is regularly displayed on the monitor screen for information to the operator. As the estimated withdrawal position approaches 40%, the execution cycle is repeated after every 2% withdrawal step till criticality is achieved.

6.1.2. Graphic display

The experiment on approach to criticality is graphically displayed on the monitor screen in real-time. Control rod position is shown on the x-axis while the inverse multiplication factor is displayed on the y-axis. The position of control rods at criticality is updated after each new iteration. With the help of this information the operator can determine the next safe increment of control rod withdrawal. Both video- and audio-alarms are generated when criticality is achieved.

6.2. Reactivity calculation

Reactivity is one of the most important nuclear parameters in an operating reactor. Continuous monitoring of reactivity can be accomplished with the help of a reactivity meter. Such a reactivity meter had been developed earlier on the plant PDP-11/23 computer [8]. Due to the difficulty in maintaining the old PDP computer the reactivity meter was installed using the PC. Another related algorithm developed on the PC was for the control rod reactivity worth determination based on positive period technique. The flow diagram of this algorithm is shown in Fig. 8.

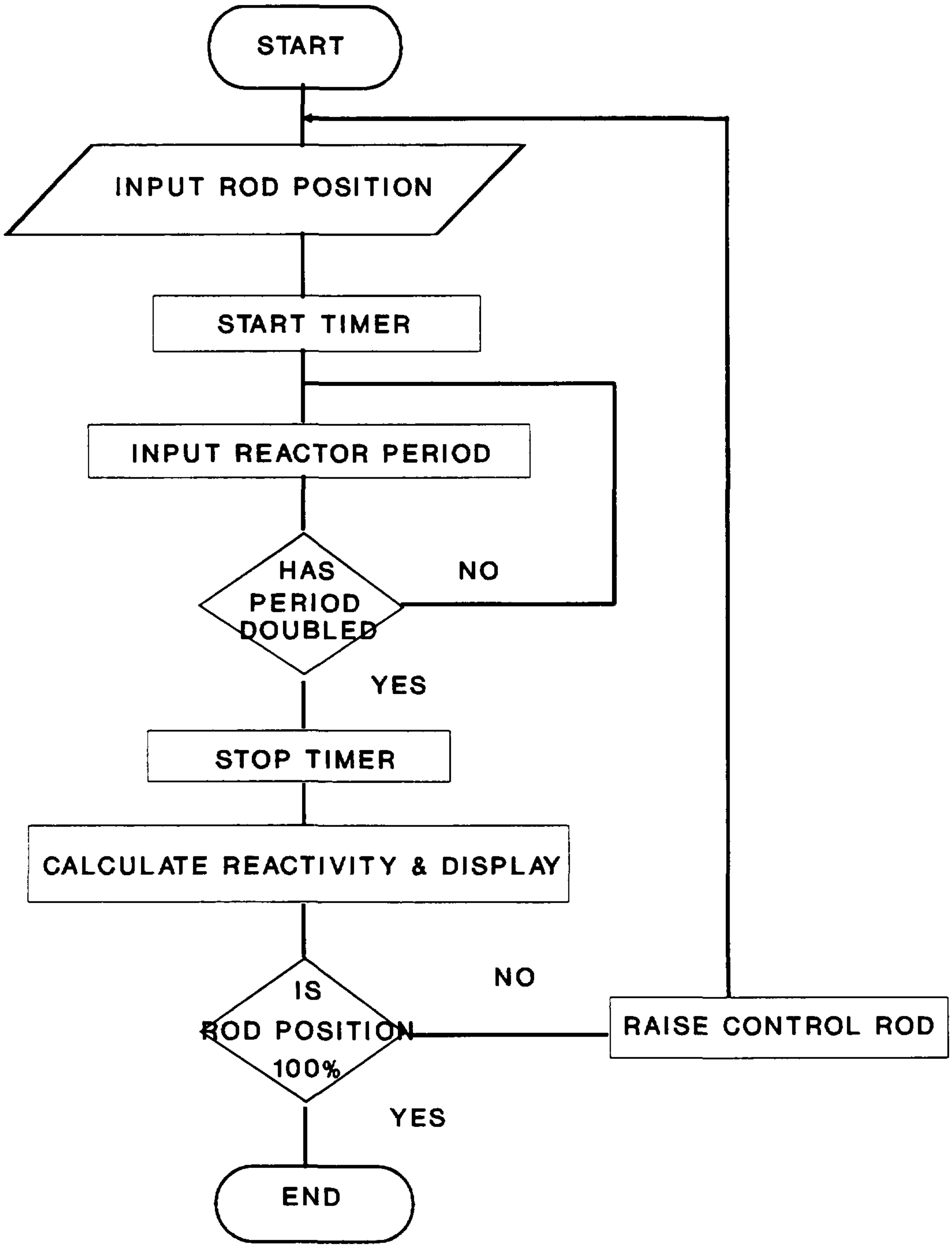


FIG. 8. Flowchart of control rod calibration algorithm.

In actual measurement of control rod worth by the computer, the reactor was initially operated in a steady state. A positive ramp reactivity input was then applied by withdrawing one control rod by a predetermined amount. The computer measures stable reactor period in two ways. It acquires reactor period from the power period channel and takes an average of about 50 successive period values. The other method is the measurement of time required by the reactor power to approach e-times the value of initial power. This measured reactor period was then converted into reactivity by a subroutine which uses in-hour equation for reactivity computation. The reactivity value was plotted versus the rod position over the entire length of control rod and the reactivity worth of one control rod was obtained.

6.3. Computation of thermal power

Reactor thermal power is calculated using standard heat balance equation [9]

$$P = mC_p\Delta T \qquad (2)$$

where,

P = reactor power (kW)
m = mass flowrate (kg/s)
C_p = specific heat of the coolant (kJ/kg-^{0}C)
ΔT = coolant temperature difference across the core (^{0}C)

Thus if the measured values of coolant mass flowrate, and the net increase in the coolant temperature while passing through the core are available, reactor power can be calculated directly with the help of Eq. (2).

The flow diagram of the reactor power calculation algorithm is shown in Fig. 9.

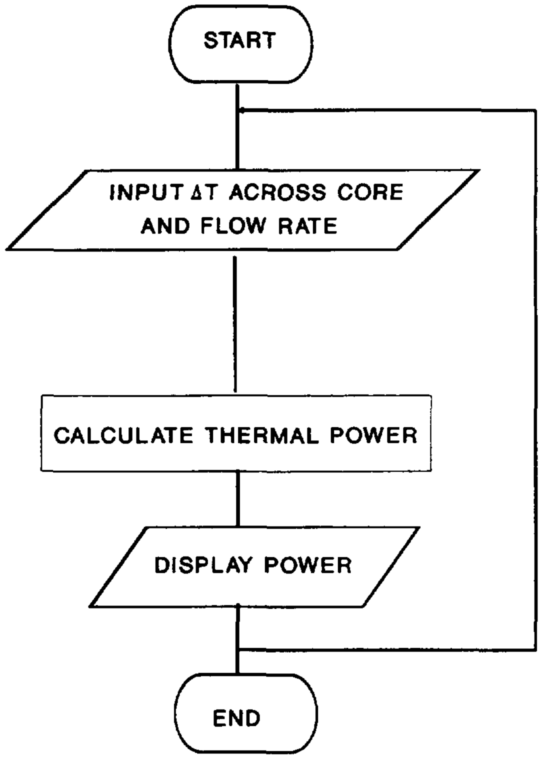


FIG. 9. Flowchart of thermal power computation algorithm.

7. CONCLUSION

The computer-aided signal monitoring, processing and display system has been installed in the control room of PARR-1, besides the main operation console. This is a compact, economical system that constantly updates important reactor data and informs reactor operator and users about the reactor status. In this way the application of computer has considerably enhanced the plant monitoring capabilities. Figs. 10-11 show the photographs of the computer and one TV monitor located in the reactor user area at the beam port floor.

A typical printout of the instrumentation performance analysis algorithm is shown in Fig. 12. The error computation is not accurate for reactor period signals, which are differential in nature and have large fluctuations around the mean value. The mean-value of the period signals is however useful for reactor operator. There have been many instances where the computer has detected a noisy instrumentation channel or detector, and the problem was corrected by maintenance. It was observed that by replacing the detector signal cables by double shielded cable in the startup channels, the scatter in neutron counts signal reduced by a factor of two. The time constant of the log countrate meter was also adjusted with the help of the system to obtain minimum error.

In one application of the PC-based system, the performance of newly modified automatic reactor controller was studied. The error bands of linear flux channel signal and rod position signals were analyzed in automatic control mode, and compared with the errors in the manual mode. Optimum adjustment of PID settings of the reactor autocontroller was done by noting the minimum error bands. Signal error computation thus provided a simple and effective means for this job.

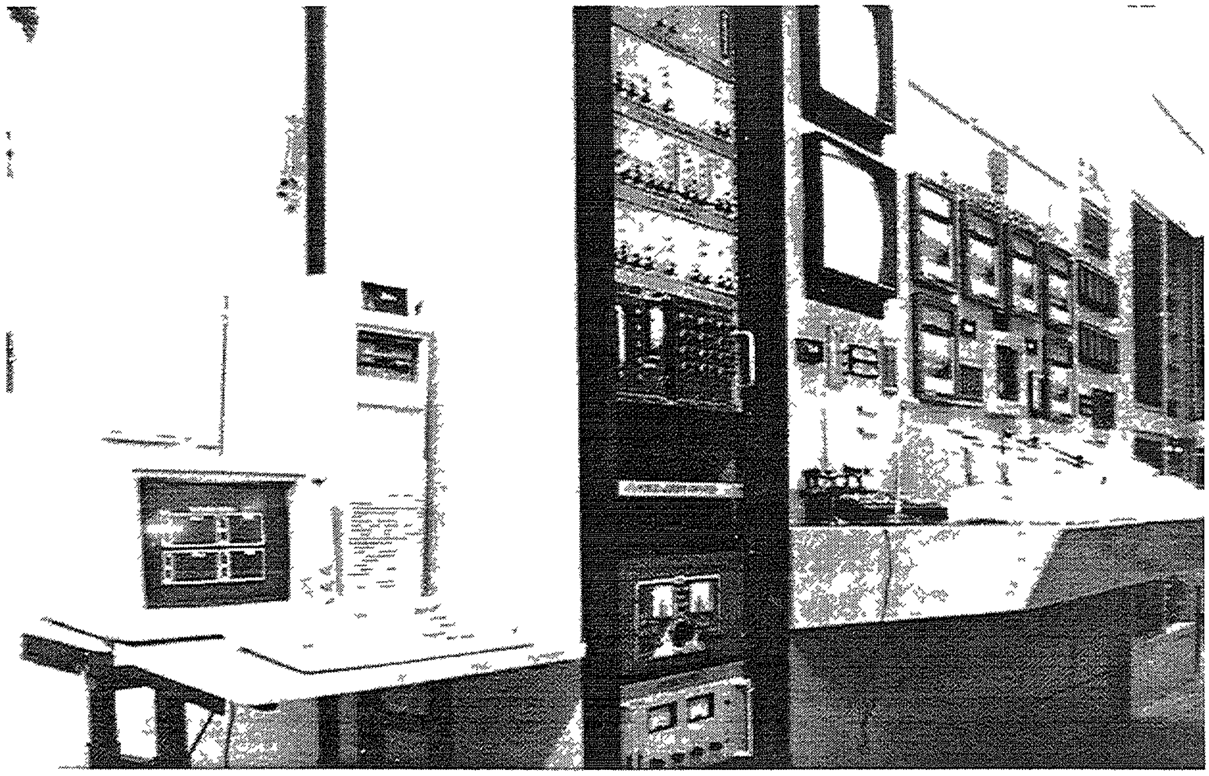


FIG 10 Personal computer and control room of PARR-1

It is planned to extend the capabilities of reactor parametric calculations of the computer-based system to include the following additional reactor parameters.

- Automatic correction in neutron power based on N-16 and thermal power.
- Measurement of control rod drop time.
- Reactor safety measurements, e.g., time behaviour of neutron flux and coolant temperature following reactivity transients.
- Integration of in-core flux mapping data for fuel burnup calculations

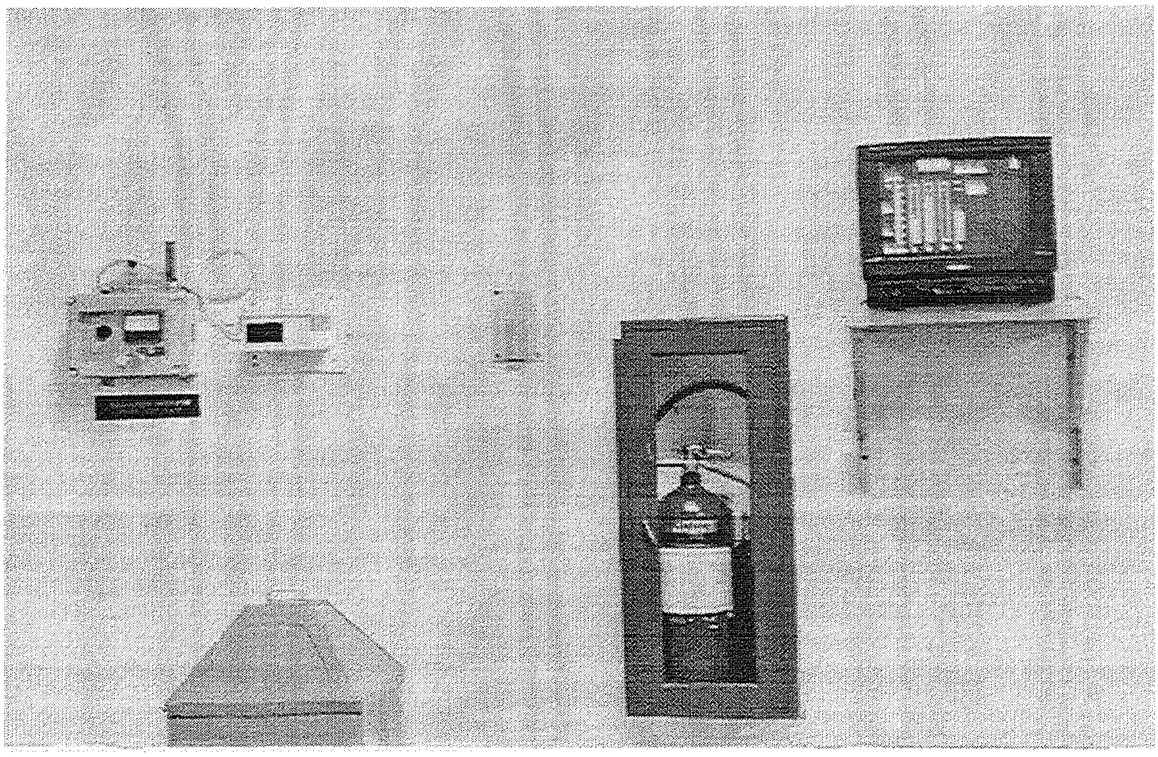


FIG. 11. TV and radiation monitor located in the reactor hall.

CHANNEL MONITOR 12:10:29 08-07-1995
Mean Reactor Power: 9 MW

CHANNEL	MEAN VALUE	ST. DEVIATION (%)	REF. ERROR (%)	STATUS
STARTUP A: LCR	8449.0 cps	0.9	1.09	OK
STARTUP B: LCR	1837.8 cps	2.2	2.33	OK
LOG A: PERIOD	-80.2 s	14.4	11.17	OK
LOG B: PERIOD	-101.00 s	10.10	9.95	OK
LOG A: POWER	8.52 MW	0.14	0.04	OK
LOG B: POWER	8.85 MW	0.23	0.04	OK
LIN A: POWER	9.26 MW	0.30	0.04	OK
LIN B: POWER	9.40 MW	0.43	0.03	OK
N-16 A: POWER	9.04 MW	0.53	0.03	NOISY
N-16 B: POWER	8.91 MW	0.29	0.04	OK

FIG. 12. Channel testing printout.

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INCORPORATION OF PERSONAL COMPUTERS IN A RESEARCH REACTOR INSTRUMENTATION SYSTEM FOR DATA MONITORING AND ANALYSIS*

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Abstract

The research contract was implemented by obtaining off-the shelf personal computer hardware and data acquisition cards, designing the interconnection with the instrumentation system, writing and debugging the software, and the assembling and testing the set-up. The hardware was designed to allow all variables monitored by the instrumentation system to be accessible to the computers, without requiring any major modification of the instrumentation system and without compromising reactor safety in any way. The computer hardware addition was also designed to have no effect on any existing function of the instrumentation system. The software was designed to implement only graphical display and automated logging of reactor variables. Additional functionality could be easily added in the future with software revision because all the reactor variables are already available in the computer. It would even be possible to "close the loop" and control the reactor through software.

It was found that most of the effort in an undertaking of this sort will be in software development, but the job can be done even by non-computer specialized reactor people working with programming languages they are already familiar with. It was also found that the continuing rapid advance of personal computer technology makes it essential that such a project be undertaken with inevitability of future hardware upgrading in mind.

The hardware techniques and the software developed may find applicability in other research reactors, especially those with a generic analog research reactor TRIGA console.

INTRODUCTION

Research Contract 6049/R2/RB between the International Atomic Energy Agency (IAEA) and the Philippine Nuclear Research Institute (PNRI) began on 1 August 1990. It sought to incorporate Personal Computers (PCs) with the Instrumentation System of the Philippine Research Reactor (PRR-1) for data monitoring and analysis. At the time, the PRR-1 Instrumentation System was fully analog, containing no microprocessors of any kind. After several extensions, the research contract ended on 14 April 1994.

There are many benefits in the installation of computers in nuclear reactor instrumentation systems. Data collection could be automated and made more reliable, data display could be more effective with video graphics screens, and computed reactor parameters could be displayed in real time. Some control functions could even be performed by computers.

Unfortunately, computers were just too big and expensive to be integrated in a cost-effective way with the instrumentation of small research reactors until a few years ago. The invention of microprocessors and the evolution of PCs into consumer items changed that. Not only did prices plummet because of mass production, but the power of small computers grew exponentially under market pressure.

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* Research carried out in association with the IAEA under Research Contract No. PHI/6049.

Microprocessors were used with instrumentation almost as soon as they were invented. By the time PCs became consumer items, scientific laboratory equipment commonly had internal microprocessors or were controlled by external PCs. Some researchers had gone as far as connecting PCs to nuclear research reactors. Usually the connection was for a limited purpose, consisting of a data link to one or a few instrumentation channels. These small-scale connections did not have a significant impact on the reactor instrumentation system.

In contrast, entirely microprocessor-based research reactor instrumentation systems offered by some vendors were revolutionary, intended as they were to be complete replacements for existing systems. However, the large capital investment required was discouraging to most reactor owners who had analog instrumentation still in excellent working order. A complete replacement was also certain to require extensive retraining of the reactor operators. It was also almost certain that there would be teething problems with a new unproven system, which could cause reactor reliability to initially decrease.

The PRR-1 Instrumentation System was still fairly new in 1990, having been upgraded with a standard analog console from General Atomics when the PRR-1 was converted to a TRIGA reactor. Complete replacement with a microprocessor-based system was not of any interest then. However, it was realized that it was possible to obtain the benefits of computerization inexpensively and without replacing the existing system, which also meant there would be no need for extensive operator retraining and no risk of decreased reliability.

Low-cost standard PCs were the key. These PCs could be passively linked to every existing instrumentation channel and to every important electrical switch. The analog instrumentation would continue to function as before. The computers initially would be used only for automatic recording and graphical display of reactor data. The benefits there alone would justify the PCs.

However, since all the variables monitored by the instrumentation would be available in the PCs, future applications could go far beyond just data recording and display. These extended applications would be based on doing complex computations on reactor data in real time. For example, just with additional software, the PCs could perform a reactivity calibration of one or more control elements during a single reactor start-up, something which used to take hours or days of tedious experiments or required expensive auxiliary instrumentation.

It would even be possible to "close the loop" and let the PCs assume some control functions. For example, the analog feedback loop connecting reactor power and period to the regulating rod (usually put on line to automatically maintain reactor power) could be implemented in software if the PCs were allowed to control regulating rod position. However, this step would go beyond passive connection to the instrumentation system and could not be taken lightly.

It was realized that a project to incorporate computers in such a large scale with the PRR-1 Instrumentation System would require considerable knowledge of computers, both in hardware and in software. Before PCs, very few reactor instrumentation people would have had the specialized knowledge. After the explosion in PC use, the information needed by non-computer-specialized people to do hardware and software development became widely available. By the nineties, a reasonably experienced, competent, and technically curious instrumentation person probably already had enough PC knowledge to begin such a project, and whatever additional knowledge he needed to finish the project he could get easily.

Research Contract 6049/R2/RB was based on the conviction that such a project was desirable and was meant to prove that it was at last feasible.

THE END-PRODUCT

The end-product of the research contract is described here. The description should perhaps appear nearer the end, but it would be easier to follow the rest of the report if it is already known what the end-product looks like.

Data Display

Three video screens controlled by separate PCs display all the variables monitored by the PRR-1 Instrumentation System. The screens are labeled Center, Left, and Right. See Figures 1, 2, and 3.

Digital variables (those which have only a few discrete states, such as on/off) have text-only displays. Analog variables (those which can have any value in a continuous range) have text displays showing current values and also have scrolling plots showing their history, similar to strip-chart recorder outputs. A few analog variables that rarely change have text-only displays. Other exceptions are the control element positions, which have bar-graph representations of the current positions instead of plots of previous positions.

Figures 1, 2, and 3 show test values, not actual reactor-derived values. (Note that all the plots are sinusoidal.) Although the figures are in black-and-white, the actual screens are in color. Color intensities are also reversed to print better on paper. (Dark screen colors are printed light, and vice-versa.)

The display colors of a variable change according to the value of the variable in comparison with one or more setpoints. Five sets of colors are defined:

Condition:	*Background Color:*	*Text and Plot Point Color:*
Normal	Blue	Gray
Warn	Bright Blue	White
Alarm	Gray	Black
Scram	White	Bright Blue
Emergency	Red	Gray

The bars representing control element positions are exceptions. Each bar is in red if the corresponding control element is attached to the drive, and in background color if the control element is unattached and left in the core as the drive changes position. (The variables displayed are actually drive positions, not control element positions.)

The screen background is predominantly blue when everything is normal. The background colors were chosen to show an abnormal condition conspicuously, even at a distance where the text is not readable.

Related variables are grouped together. For example, the variables monitoring the state of the electrical power supply are all at the upper right corner of the Center Screen. Similarly, the variables moni-

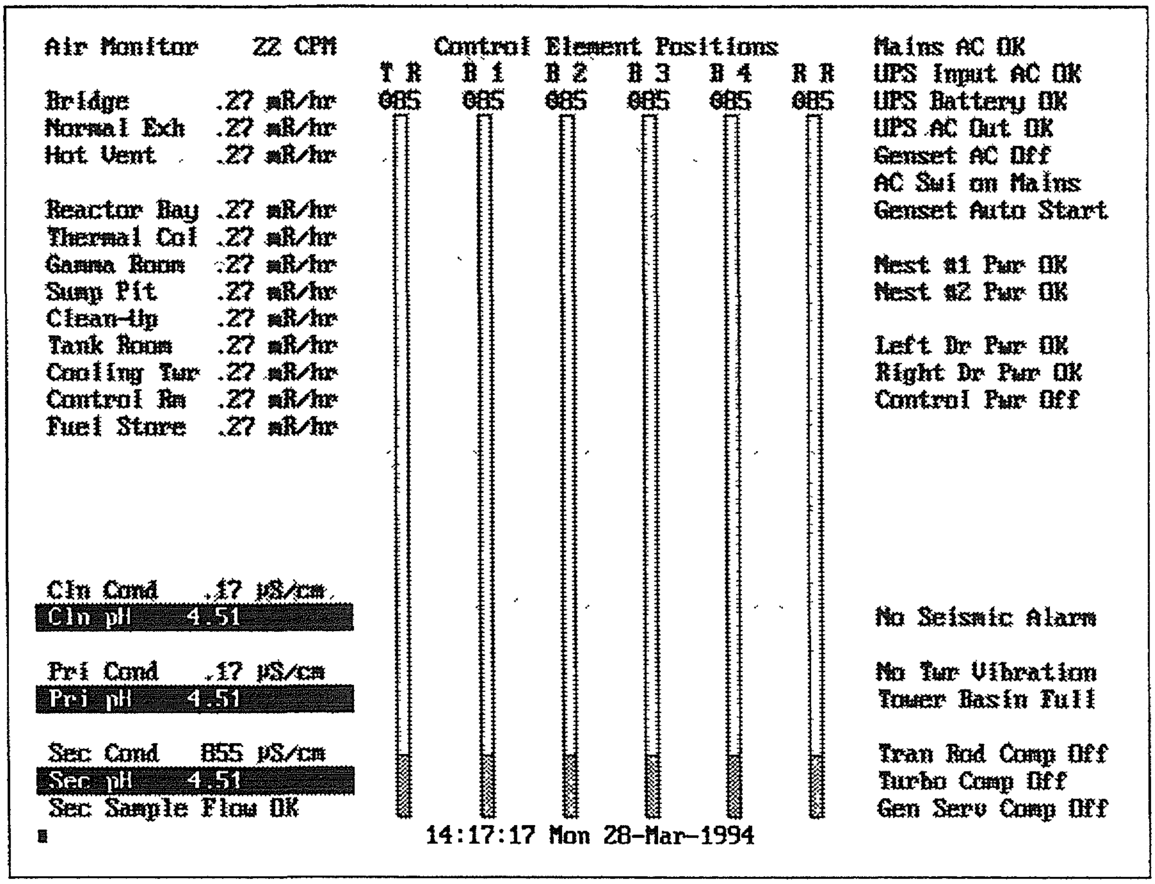


Figure 1. The Center Screen

toring the state of the mechanical parts of the forced cooling system are grouped together at the right side of the Right Screen.

Some variables affect the display of others. For example, the setpoints of Linear Power #2 and Fuel Temperature #2 in the Right Screen are different depending on whether the reactor is at NATURAL COOL MODE or FORCED COOL MODE.

The PCs sample data continuously in a program loop, as fast as the hardware allows. Even a slow PC can take at least a hundred data sets per second, with each data set containing a sample of every variable in the PRR-1 Instrumentation System connected to the PC. However, the screens are not updated every time a data set is taken.

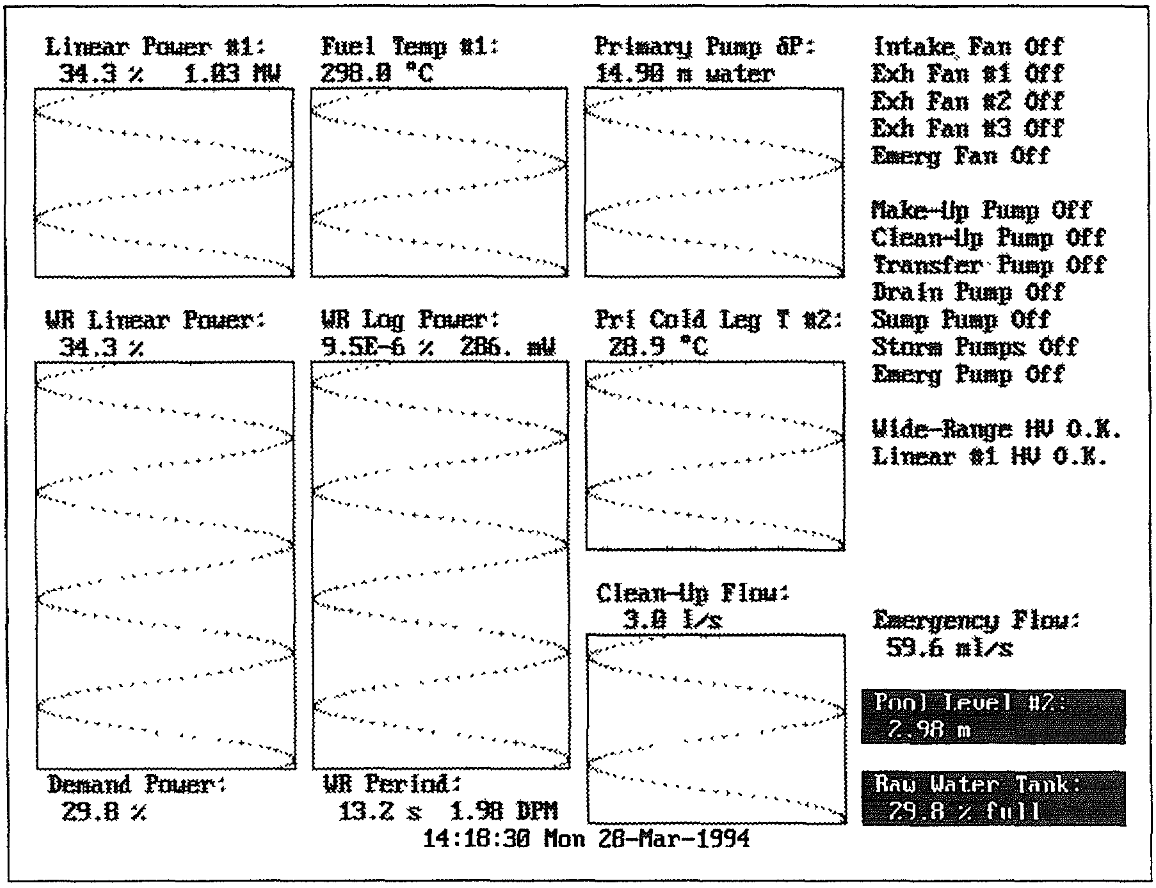


Figure 2. The Left Screen

A PC updates its screen when it detects that a state change has occurred. A state change is defined to occur whenever any digital variable changes, or whenever any analog variable crosses a setpoint, or whenever any control element position changes by 50 units. If no state change occurs within a programmed time interval, the PCs update their screens anyway. The update time interval is nominally 100 PC BIOS ticks (about 5.5 seconds), but may be programmed longer for low-performance PCs or shorter for high-performance PCs.

Each screen also shows the time and date of the last update at the bottom center. The three PCs controlling the screens communicate with each other. The Center PC acts as a master timekeeper and sends its own time and date to the other PCs so that all three PCs are synchronized.

An exception to the update routine can occur in the Right PC. When this PC detects that the reactor is in PULSE MODE and that the transient rod has been fired, it immediately concentrates on the Pulse Power Channel and ceases sampling all other variables. The channel is sampled at a rate of 3,200 Hz, or one sample every about 0.3 milliseconds. Sampling continues until the designated storage space in computer memory is exhausted in about ten seconds or until the PC has detected that the transient rod has fallen back into the core and ended the pulse.

The PC then locates the pulse profile in the series of samples. (A typical pulse is only about 20 milliseconds wide, occupying about 70 consecutive samples out of a complete set of as many as 32,768

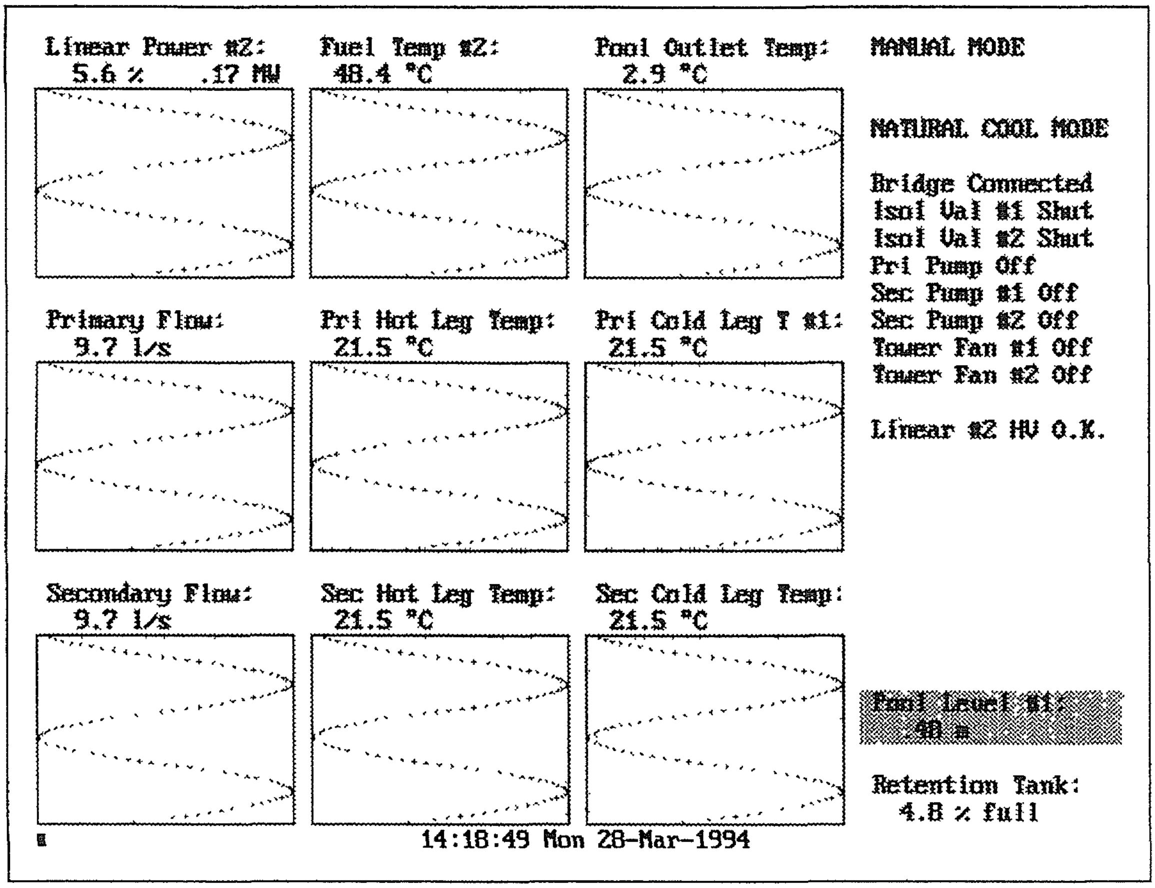


Figure 3 The Right Screen

samples.) The pulse profile is plotted on the screen (in the space usually occupied by the Linear Power #2 channel, which the reactor instrumentation disables in PULSE MODE) and the PC resumes its regular screen update routine.

The graphical display of the pulse profile has no equivalent in the regular PRR-1 Instrumentation System, which indicates only the peak power and pulse energy released, both calculated with analog circuits. (The Right Screen also displays these variables in PULSE MODE.) The pulse profile can be seen with the standard analog TRIGA console only with an expensive high-speed recorder, which is not usually attached to the Pulse Power Channel.

A small block character at the lower left corner of each screen blinks on and off to assure the reactor operator that the PC is operating normally and has not hung up for any reason. Warning messages (for example, for a failed disk-write operation) may appear at the bottom of the screen.

Data Logging

Each PC is forced to save the current data set to memory when a state change occurs or when a programmed time interval passes without a state change. The save time interval is nominally 200 PC BIOS ticks (about 11 seconds), and is not necessarily the same as the screen update interval

The memory requirement for data saves is very modest, requiring only 58 bytes per data set in the Left and Right PCs, and 108 bytes in the Center PC. When the PC has stored 1,000 data sets in memory, it moves all the data to a disk file, to be preserved even if the PC is turned off. A special case occurs in the Right PC after a reactor pulse has occurred. The whole set of sampled pulse data containing the pulse profile is saved immediately to disk after the pulse ends.

In order to keep the saved disk files from overwriting each other, a disk file is given a unique file name based on the time of day, and files are placed under a new subdirectory for every day of the year. The file name also codes the source PC and the type of data contained (regular or pulse).

A saved disk file is 58,068 bytes long in the Left and Right PCs and 108,068 bytes long in the Center PC. (The extra 68 bytes are occupied by a file header.) Each disk file represents as much as 3 hours of history. A single 120-megabyte hard disk (a common size today) can hold 8 months of data in the Left and Right PCs and 4 months in the Center PC. Gigabyte-size hard disks which will be practical in the very near future can hold years of data. In practical use, the data would be archived in more permanent media and the hard disk cleared every few weeks or so.

The high data density is possible because data sets are saved in raw form and in binary format. In this form, data files are readable only with the aid of a computer program. A simple reader was developed, but it is also possible to write more sophisticated programs which will process many disk files from all three PCs at one time and extract specific information.

THE HARDWARE

The PRR-1 Instrumentation System

In order to understand the way PCs had to be connected to the PRR-1 Instrumentation System, it is first necessary to understand the way the system was built.

As part of TRIGA conversion of the reactor, the PRR-1 Instrumentation System was completely redesigned and rebuilt in 1984 with modern components (but without computers of any kind). Because of the way the project was implemented, the system can be divided into two parts: the part that was provided by General Atomics (GA), and the part that was provided by PNRI. The split is physically obvious: the system has two enclosures, the console for the GA part and another instrumentation cabinet for the PNRI part.

The GA Part

A generic analog TRIGA console monitors and displays reactor power, period, fuel temperature, and pool water temperature. Most of these variables are measured by more than one channel. Although there are no specific provisions for connection to the outside world, the measured variables are available as 0 to +10 volt signals from circuit board pin-outs.

The console also includes drive controls for the reactor's four safety blades, a regulating rod, and a transient rod. Blade and rod drive positions are coded as proportional 0 to +10 volt signals and displayed by voltmeter circuits. The console has logic circuits to interlock the operability of the drives with conditions set by the power channels according to the operational mode selected. The console also has a servo system linking the regulating rod drive with one of the power channels.

The console houses the reactor's safety shutdown circuit, which accepts signals from the power and temperature channels and automatically drops the safety blades into the core when warranted. The circuit also accepts external relay-contact shutdown signals.

The circuits in the console can be classified into three groups: two safety groups and a control group.

The safety groups encompass those channels which can trigger the reactor's safety shutdown circuit (also known as the scram circuit). The safety-related variables (reactor power and fuel temperature) are each measured by two independent channels, one in each group. To guard against common-mode failure, the safety groups are mechanically as well as electrically separated from each other, as far as they can be while mounted in the same console. The electrical components are in left and right console drawers. Some instrumentation channels are not linked to the safety shutdown circuit but are located in either the left or right drawers without electrical isolation, and are counted as part of one or the other safety group.

The control group includes everything else, especially the blade and rod drive controls. These circuits do not provide signals to the shutdown circuit. These circuits are in the center section of the console, away from the safety circuits.

The PNRI Part

The rest of the PRR-1 instrumentation measures water temperatures, flow rates, pressures, conductivities, and pH in both primary and secondary loops of the cooling system, in the emergency cooling system, and in the clean-up and make-up systems, also water levels in the pool and various storage tanks; radiation levels beyond the shielding; and the condition of the AC electrical power supply

Most of the variables are measured using conventional current-loop transmitter and receiver modules. The transmitter modules are field-mounted close to the transducers sensing the variables, while the receiver modules are mounted in racks in a cabinet adjacent to the console. The receivers convert their inputs to 0 to +10 volt signals which are sent to display modules (panel meters) and a multi-pen strip-chart recorder.

The safety-related channels have alarm modules, which compare the signals from the receivers with setpoints, and which connect to the shutdown circuit in the console through relays. Each safety-related variable is measured by two independent channels.

Some variables are available only as on/off signals, typically from mechanical switches. These signals drive indicator lamps or relays in the cabinet.

The cabinet also has an annunciator panel providing audible and visual warning of abnormal conditions. The annunciator is driven by alarm modules or by external switches or relay contacts.

The cabinet also houses the push-button controls of numerous electrical motors connected to pumps and blowers in the reactor's cooling system and ventilation system. Some of these motor controls are interlocked using relay logic with the variables measured by channels in the cabinet, as well as with the controls in the console.

As in the console, the circuits in the instrumentation cabinet can be classified into two safety groups and a control group, which are mechanically and electrically separated. The safety channels are mounted in the left and right wings of the instrumentation cabinet. Those control circuits which are also housed in the left or right wings are electrically separated from the safety circuits in such a way as to prevent acting as a bridge between any two safety channels.

The combined safety groups in both the GA part and the PNRI part will be referred to in this report as the Left System and the Right System. The combined control groups will be referred to as the Center System.

The variables monitored by the PRR-1 Instrumentation System are listed in Appendix A.

Off-The-Shelf PC Hardware

It was kept in mind that the research contract should use off-the-shelf PC hardware whenever possible, and the fabrication of special circuits should be minimized. The whole point of the research was to investigate whether the work could be done with inexpensive standard PC hardware, and by people without extreme specialization in computers.

The Basic PC Hardware

It was obvious that the PCs to be used had to be IBM-PC/XT/AT compatibles. Only for this type of PC were all the following true:

1. Hardware was inexpensive and widely available from many suppliers;
2. The computer architecture was open and non-proprietary, with information necessary for programming and for connecting auxiliary hardware widely known and available;
3. The PC performance was powerful enough for the purpose

Nevertheless, there were sub-types and options within the IBM-compatible family from which choices had to be made. The choices frequently were decided only by cost. The cost of better performance came down drastically during the duration of the research contract, so that the hardware in use at the end was very different from those in use at the beginning.

The initial PC hardware was quite modest: XT-class PCs with 640 kilobytes of memory, a 20-megabyte hard disk, a Hercules-type monochrome display, and one or two serial ports. In the end, however, the project was using 80386SX-class PCs running at 33 MHz, 2 megabytes of memory, a 120-megabyte IDE-type hard disk, color VGA displays, and two serial ports.

At the time the research contract began, only PCs with 8088-type motherboards operating at 12 MHz were affordable, which were usable but slow. Later, 80386-type motherboards operating at 33 MHz came down to about the same price level, and were substituted for the originals for a considerable gain in speed. Similarly, initially only video displays with monochrome graphics were affordable. Later, color VGA displays came down in price and were substituted for the original displays, leading to a great improvement in the quality of data presentation. A similar situation occurred with the hard disk drives: 120-megabyte IDE-type hard disks were eventually substituted for the initial 20-megabyte ST-506-type hard disks, improving storage capacity six times and speed ten times at about the same cost.

Upgrading the PC hardware was painless and could be done step-wise because new components generally were "downward-compatible"; i.e., old hardware and software could run with new hardware. The only exception was the video system, where software written for monochrome graphics displays would not work with color VGA displays. The difficulty was anticipated and avoided by writing the display software in such a way that it was easily upgraded as well.

The Data Acquisition Hardware

The crucial hardware was the data acquisition interface between the PCs and the instrumentation. It was necessary to digitize the analog signals from the instrumentation before it can be processed by the PCs. Even the on-off signals from the instrumentation which were already digital needed transformation, since they were at voltage levels incompatible with the PCs. It was not necessary to fabricate the necessary circuits because many vendors offered data acquisition interfaces for PCs. These typically contained Analog-to-Digital (A/D) Inputs, Digital Inputs (D/I), Digital-to-Analog (D/A) Outputs, and Digital Outputs (D/O) in one circuit board.

It was a matter of selecting the best-suited board. The following were the major considerations:

1. *Cost.* The total cost for the data acquisition interface should not be more one or two thousand US dollars, meaning that each board should not cost more than a few hundred since several were needed. Of course the research contract had limited funding, but it was also realized that using a data acquisition interface costing several times the cost of a PC (only a few hundred dollars itself) was against the spirit of the project.
2. *Accuracy.* The digitized signals should not be less accurate than the source analog signals. A/D conversion was available with 8-bit, 12-bit, or 16-bit outputs. An 8-bit output implied an accuracy of 1 part in 256, or about 0.4% with bipolar signals and 0.8% with the unipolar signals actually available from the PRR-1 instrumentation. A 12-bit output implied 1 part in 4096, or about 0.05% accuracy with the PRR-1 signals. A 16-bit output implied 1 part in 65536, or about 0.003% accuracy with the PRR-1 signals. The analog PRR-1 instrumentation channels typically had an accuracy of 0.5%, which meant that only 12-bit or 16-bit A/D conversion was acceptable. The 12-bit interfaces were clearly preferable, since the 16-bit interfaces cost about twice as much.
3. *Physical form.* Interfaces were available in stand-alone boxes, which were independently powered and typically connected to a PC through the serial port. These could be connected to any PC, even non-IBM-compatibles. Interfaces were also available as internal plug-in cards, communicating with the microprocessor directly through the PC bus. A card could be used only with a specific type of PC. The internal cards were preferable, since they simplified wiring and were generally less expensive than the stand-alone interfaces.
4. *Availability of accessories.* It was desirable that accessories such as opto-isolators, relay outputs, and wiring adaptors be inexpensively available ready-made from the same supplier. Some design and fabrication work could be saved.

The PCL-812PG Enhanced Multi-Lab Card from B&C Microsystems was chosen as the basic data acquisition card. In one card made to plug into the standard PC bus, there were 16 12-bit A/D channels, 16 D/I channels, 2 D/A channels, and 16 D/O channels. The following accessory cards were also used: the PCLD-780 Screw Terminal Board, the PCLD-782 16-Channel Opto-Isolated D/I Board, and the PCLD-785 16-Channel Relay Output Board. The general specifications of the data acquisition cards are given in Appendix B.

These cards were used throughout the duration of the research contract. Upgrading was not attractive. The cost of better components did go down, but not to the same degree as that of the more common PC components. Also, upgrading the data acquisition interface would have required extensive rewriting of the software, unlike that of the other parts of the PC.

Specially-Built Hardware

Very few special-purpose hardware items needed to be fabricated. One of these was an opto-isolated connector between the RS-232C serial ports of the PCs, functionally equivalent to a null-modem cable. The need for this circuit is explained in the next section, and its schematic is shown in Appendix C.

Buffer amplifiers were also needed between a few signals in the instrumentation and the data acquisition cards. No specific design is given in this report; many designs based on operational amplifiers would have worked, since there was no exceptional requirement.

System Design

The following constraints were voluntarily imposed when the hardware system was conceptualized:

1. There must be no reduction in safety. There must be no interference with the safety shutdown system, no interference with signals in safety-related channels, and no interference with interlocks or alarms.
2. The separation between the two safety groups and those between the safety groups and the control group must be preserved
3. There must be no degradation of the existing system. There must be no signal interference; there must be no excessive loading of signal circuits; the isolation of AC, high-voltage, and other power circuits from signal circuits must be preserved, and no existing channel or function must be removed or disabled.
4. The PC installation must be considered temporary, at least initially. All changes to the existing system must be reversible, and any additions must be removable.
5. The PCs will not be allowed to directly control the reactor, at least initially.

The constraints determined the following features of the hardware system design:

1. *Three PCs had to be installed.* Each PC hosted only the data acquisition cards associated with one particular group. Four PCL-812PG cards were installed to accommodate all the variables in the PRR-1 Instrumentation System, one each in the PCs in the Left and Right Systems and two cards in the PC in the Center System.
2. *The cross-connections between PCs had to be opto-isolated.* The PCs had to send digital signals to each other. Some connections were handled through the data acquisition cards (see next item). There were also connections through the serial ports of the PCs, for which a special opto-isolated RS-232C link was designed and built.
3. *Each input digital signal had to pass through an opto-isolating and conditioning circuit.* The input digital signals were supplied by switches or relay contacts at voltage levels inappropriate to the data acquisition cards. The opto-isolation that had to be provided had the added benefit of allowing the connection of a digital signal to a data acquisition card not in the same group. The standard opto-isolated D/I accessory boards of the data acquisition cards were used.
4. *Each input analog signal tapped from a high-impedance point in the PRR-1 instrumentation had to be buffered.* Nearly all the take-off points for the analog inputs of the data acquisition interface were already buffered, since they were intended to service panel meters or recorders. There were only a few exceptions that needed buffering. The added buffers were designed to use common operational amplifiers available in integrated circuit form, and powered by the DC supplies available in each of the groups in the PRR-1 Instrumentation System. No opto-isolation was necessary, since analog signals were never connected to a data acquisition interface card not in the same safety or control group.
5. *The D/O and D/A output signals available in the data acquisition interface cards were available but were not used for the time being.* These signals can be used in the future for control purposes, after some wiring and software changes.
6. *The PCs and their displays should be provided with electromagnetic shielding.* The PCs were never actually installed within the same cabinets housing the PRR-1 Instrumentation System, but if they were, it would have been necessary to protect the existing instrumentation channels against radio-frequency interference from the PC motherboard and cards and magnetic interference from the PC display. The protection could have been provided with special enclosures around the PC components. It was not believed the usual light sheet-metal and plastic PC cases would have provided enough protection.

THE SOFTWARE

In keeping with the off-the-shelf philosophy of the project, the software was written to run with the standard BIOS (in ROM) and MS-DOS (in disk) used by nearly all IBM-compatible PCs. A custom-built operating system might have had theoretical advantages in performance, but PC hardware was already so advanced that slow software was not disabling. In any case, the skill to write a custom operating system was not available.

The software was written to require at least an XT-type BIOS and MS-DOS version 3.3. Eventually the software was run with an 80386-type BIOS and MS-DOS 5.0, but the new features of the up-to-date operating system were not used. Also, the software was originally written to work with a monochrome graphics display without assistance from the BIOS, but was later re-written to work with a standard VGA display, using the BIOS in the VGA display adapter.

The software initially developed was intended to implement only data recording and display PC usage could be greatly expanded with an extension of the software, with very little additional hardware

Unlike the hardware, which was almost entirely off-the-shelf, the software had to be entirely custom-written Although most of the project's money went to hardware, most of the time went to writing and debugging the software. Eventually almost 1.5 megabytes of source code was written and tested The undertaking was manageable only by breaking it down into smaller tasks which were handled one at a time The descriptions which follow are given generally in the same order as the tasks they apply to were handled

Programming Languages

First, the programming language had to be chosen. Since the software would have to deal directly with the hardware to get data in and out of the PCs, at least part of the software would have to be written in a low-level language such as Assembler. However, it would be much easier to write the software to manipulate the data (after it was inside the PCs) with a high-level mathematically-oriented language such as Fortran It would also be easier for other people to modify or extend the software in the future if it was written in Fortran, the *lingua franca* of scientific computing.

It was decided to use Microsoft Fortran 4 1 and Turbo Assembler 2.0 (referred to as just MS-Fortran and Assembler later in this report). Standard Fortran has no capability to deal directly with hardware, but it can call subroutines written in another language. Subroutines in Assembler could be written to conform with the calling and variable-passing conventions of the Fortran compiler After separately compiling the source codes and linking the various object files formed, single executable files could be created which seamlessly integrate hardware and data manipulation in one package.

The use of an intermediate-level language such as C was an alternative to the use of the Fortran-Assembler combination, but it was not adopted Both hardware manipulation and data processing could be done in an intermediate-level language, but not as easily as with a specialized language for each. Moreover, source code in C is notoriously difficult to follow by anybody but the writer, which will pose a problem to future software revision. Furthermore, the programmer (the Chief Scientific Investigator) already had much experience working with the Fortran-Assembler combination, but almost none with C

The following compile options of Microsoft Fortran 4.1 were used:

/AL *Use large memory model.* Program code and data can each be larger than 64 kilobytes

/FPi *Compile with emulator library.* The program will use the math co-processor if it is installed, but an emulator library will allow the program will run even without the math co-processor

/4Yd *Turn on compile-time warnings about undeclared variables.* The source code must explicitly declare all variables This option promotes code clarity and aids debugging

/4Nt *Allow names up to 31 characters long.* This option allows improved source code readability

No special feature of Turbo Assembler 2 0 was used

The Data Acquisition Card Driver

Attention was next paid to the problem of providing a software driver for the data acquisition hardware. The support software provided by the vendor with the PCL-812PG card was examined. The software was found to be a set of libraries intended to be linked with the user's object files. The libraries support object code produced by some common C and Pascal compilers. The libraries contain subroutines which provide control over the data acquisition card and also the means to transfer acquired data to the calling program. Unfortunately, the software set did not include a Fortran-compatible library.

The first software writing task was to write an MS-Fortran-compatible library in Assembler Using the user's manual and the disassembled library code as guides (the vendor did not supply the source code), a driver was written to provide the same functions for MS-Fortran as provided by the vendor's drivers for C and Pascal. The driver worked well, but after some experimentation, it was found that the functions were not comprehensive enough for this application, and the subroutine calling convention was too unwieldy.

That first driver was not actually used Nevertheless, the driver is useful for simple applications, and the source code is included in this report for the benefit of PCL-812PG card owners who may wish to use Fortran in the same way they use C or Pascal with the card.

A completely different software driver for the PCL-812PG card was written next, also in Assembler, which was the driver actually used in the project. The driver was written as a Terminate-and-Stay-Resident (TSR) COM-type DOS program. The driver is referred to hereinafter as the PCLTSR driver

The PCLTSR driver is a program intended to be run before the main program, and which will keep itself in memory. The PCLTSR driver captures one of the unused software PC interrupts, through which the main program calls a driver function. The interrupt-driven function-calling method is similar to the way the BIOS and DOS operates, and the PCLTSR driver in effect becomes a part of BIOS and DOS after it is run. Any program has access to the functions after the driver is loaded

The following PCLTSR driver functions are defined:
00 - reset driver
01 - start interrupt-driven scan
02 - return interrupt-driven scan status
03 - return interrupt-driven scan A/D data
04 - return D/I data
05 - return driver time
06 - set D/A data
07 - set D/O data
08 - set interrupt-driven conversion gain
09 - set pacer timer divisors
0Ah - start DMA-driven scan
0Bh - return DMA-driven scan status
0Ch - return DMA-driven scan A/D data

The program needs conversion to the COM form after compilation into executable code, for example by the EXE2BIN utility, if the compiler cannot do the conversion directly. The COM file is only 3600 bytes in size, but it allocates and retains an additional 64 kilobytes of memory to use as a buffer for Direct-Memory-Access (DMA) data transfers.

Assembler Subroutines For MS-Fortran

Routines written in MS-Fortran cannot make interrupt calls, so they can have no direct access to the PCLTSR driver Subroutines callable by MS-Fortran were written in Assembler and compiled to create object files which were later linked with the MS-Fortran object files These subroutines can make interrupt calls and provide the link to the PCLTSR driver.

Similarly, Assembler subroutines were prepared to allow MS-Fortran routines to deal with the standard PC hardware: keyboard, VGA display, serial ports, and disk drives The Assembler subroutines handle the hardware with the interrupt function calls provided by the standard BIOS and MS-DOS, except in cases like the serial ports where BIOS or MS-DOS support is poor or nonexistent In these cases the Assembler subroutines program the hardware registers directly.

Subroutines in Assembler were also written to provide access to some useful BIOS and MS-DOS functions, such as those related to the PC time-of-day A special subroutine in Assembler was also written to calculate a Cyclic Redundancy Checksum (CRC) for a byte string (The CRC is used during serial port communication.) The calculation could be done in MS-Fortran, but runs much faster in Assembler.

The compiled code for all these subroutines take up only about 12 kilobytes of the executable file after linking.

MS-Fortran Routines

General Routine

The rest of the software was written in MS-Fortran There were only four executable files (aside from the COM file for the PCLTSR driver). One EXE file for each of the Center, Left, and Right Systems for data logging and display, and one EXE file to be run off-line to read the disk files created by the other EXE files Numerous source files went into each EXE file, however

The files related to data logging and display files are referred to as LOGGER files in the rest of this report. The executable files were actually named LOGGER_C.EXE, LOGGER_L.EXE, and LOG-

GER_R.EXE. (This pattern of naming files to identify the System was used many times.) The files related to data reading are referred to as READ_LOG files; the executable file was named READ_LOG.EXE.

The MS-Fortran LOGGER and READ_LOG source files are given in Appendices H through L. Appendix H contains source code common to all. Appendices I, J, K contain LOGGER source code for the Center System, Left System, and Right System, respectively. Appendix L contains READ_LOG source code

The *.FOR files in the Appendices are listings of MS-Fortran routines. The *.INC files are meant to be inserted during compilation in places designated by $INCLUDE statements in some of the *.FOR files. The *.INC files contain declaration statements which must be consistent throughout the program. The LOGGER routine is similar in the Center, Left, and Right Systems. The general routine is described in the following paragraphs in this section. Variations are described in a separate section for each System. The READ_LOG routine is described in its own section.

LOGGER has a main routine ($MAIN) which calls subroutines to check and initialize the hardware, and then transfers control to a subroutine (ACQUIRE_DATA) which is the real core of the program. When it regains control, $MAIN calls subroutines to reset the hardware and exits back to DOS

ACQUIRE_DATA initializes some variables, then enters a loop. The loop begins by blinking the on-screen activity indicator. Then the keyboard is checked for the special keypress (Ctrl-Shift-ESC) that commands a program exit. If the keypress was made, the subroutine breaks out of the loop and returns control to the main program. The keypress is the only way to quit LOGGER short of a reset or turning off the PC. In the future, other keypresses may be programmed at this point to direct the action of the program.

If the exit keypress was not made, ACQUIRE_DATA continues in the loop by obtaining new data from the PCL-812PG card. The data is processed to determine if a state change has occurred. The loop also checks if the save time interval has passed. The old data is saved to memory if any of the two is true; otherwise the old data is discarded as the new data replaces the old data. If memory is full, its contents are saved to disk and memory is flushed.

The loop then continues to process any serial port message, and synchronizes time and date with the other PCs. The loop then enters the screen update routine.

The screen is updated only if a state change has occurred or the update time interval has passed. The screen is not updated every time the loop goes through one cycle because the video hardware is very slow compared to the rest of the PC, and the time taken by too frequent screen updates would cause some loss of reactor data.

ACQUIRE_DATA then goes back to the start of the loop to begin another cycle. More routines may be added in the future to ACQUIRE_DATA before the loopback, limited only by the ability of the PC to complete the routines fast enough to avoid an appreciable loss of reactor data.

ACQUIRE_DATA is primarily a control routine. The work is actually done by subroutines. Many subroutines are common to the Center, Left, and Right Systems, but those dealing with acquired data are specialized to one System. Each analog variable has a unique subroutine with two entry points: one to check for a state change (PROCESS_xxxx), and another to update its part of the screen (UPDATE_xxxx). The digital variables are handled as a group by two subroutines: PROCESS_DI_DATA and UPDATE_DI_DISPLAY

Center System Routine

The Center System has two PCL-812PG cards, unlike the Left and Right Systems which have only one each. It was decided not to implement the more advanced methods of data transfer (interrupt-driven and DMA-driven) in the second card, since this would need another IRQ and another DMA channel in the PC in addition to those assigned to the first card. The second card is therefore not as powerful as the first card. The routines in the Center System use the PCLTSR driver to access the first card, just like in the Left and Right Systems, but they call Assembler subroutines to access the second card directly.

The Center System serves as the master timekeeper. It sends synchronizing time signals to the Left and Right Systems through the serial ports (The Center System's COM1 is connected to the Left System's COM1, and the Center System's COM2 is connected to the Right System's COM1.) The Center System also keeps track of the save time interval and broadcasts a master save-data signal.

The Center System acts as a master controller in another way: it accepts state-change signals from both of the other Systems, and it makes sure all the other Systems get a save-data signal whenever any System signals a state change This ensures all data is saved when a state change occurs anywhere, preserving a "snapshot" of the entire PRR-1 Instrumentation System at that instant

Left System Routine

The Left System routine does not differ greatly from the general routine.

Right System Routine

The Right System is the only one that uses the DMA-driven data transfer mode of the PCL-812PG card. The Center and Left Systems use interrupt-driven data transfer for the first card, and the Center System uses program-driven data transfer for the second card The Right System also uses interrupt-driven data transfer for its own card, but only if the reactor is not currently pulsing

When the reactor is pulsing, only DMA-driven data transfer is fast enough to capture the pulse profile. However, the PCL-812PG card can service only one A/D channel in DMA-driven mode, which is why data acquisition is suspended for all other analog variables in the Right System during a pulse.

The Right System detects that the reactor is pulsing through digital variables which indicate that the console's Operations Mode Switch is in PULSE LO or PULSE HI, and that the transient rod has been fired. The loop in ACQUIRE_DATA should be able to detect the event within milliseconds after the operator has fired the rod, before the transient rod has even reached full travel. ACQUIRE_DATA then switches the PCL_812PG card to DMA-driven mode, and skips all routines in the loop aside from checking the digital variables to determine when pulsing has terminated. Pulsing ends when the transient rod falls back into the core, after the ten seconds the pulse mechanism allows it to stay in the fired position.

ACQUIRE_DATA displays the pulse profile and saves the DMA-acquired data immediately to disk after a pulse. ACQUIRE_DATA then resumes its regular loop.

Routine To Read LOGGER Files

LOGGER creates two types of disk files. Data from all variables is saved in a *.LOG file. Pulse data is saved in a *.DMA file. To conserve disk space and to preserve accuracy, raw binary data from the PCL-812PG card is saved, not the processed data displayed on the PC screens The READ_LOG program was written to read *.LOG files off-line independently of LOGGER, possibly in another PC entirely.

The READ_LOG routine begins by displaying a list of valid *.LOG files in the default disk drive and directory and asking the user to select one. READ_LOG identifies valid *.LOG files through the header included by LOGGER in each file. The header also includes information identifying the System which saved the file (Center, Left, or Right) and the date and time the file was saved. This information is also displayed by READ_LOG to help the user make a selection.

READ_LOG opens the file selected by the user and displays the first data set it contains on the screen, processed to show "real" values, not raw binary values. The user can page up or down in the file, or jump to a specific data set. READ_LOG also knows about the setpoints associated with each variable, and color-codes the display to indicate an abnormal condition. There are separate display subroutines for the Center, Left, and Right Systems because these save different data

READ_LOG was written only to browse through the data. Other programs may be written to process an accumulation of disk files as a historical database of reactor operation, extracting more complex information as needed

TESTING

When the research contract started, the intent was to do actual installation in the PRR-1 Instrumentation System and use the PCs during regular operation. It was not foreseen that the PRR-1 will be shut down indefinitely because of problems related to aging. At the time the research contract ended, the PRR-1 was still shut down for repair.

Although the Instrumentation System was not one of the reactor sub-systems causing problems, it was not operational throughout the duration of the research contract. The reactor fuel was unloaded, and the in-core parts of the instrumentation (including the control element drives) were disassembled and removed in order that the reactor pool could be emptied and its liner rehabilitated.

It was therefore not possible to actually install the PCs in the PRR-1 Instrumentation System and do a real test. The best that could be done was to put together the complete PC set-up outside the PRR-1 Instrumentation System, and test to make sure the hardware and software works, and that the set-up performs as designed. The wiring to the instrumentation would be missing, but input signals could be simulated with other voltage sources. Because the PC relationship with the PRR-1 Instrumentation System was meant to be passive from the beginning, the lack of a physical connection did not prevent a complete work-out of the set-up. The complete PC set-up was tested and debugged in this way

The PC set-up performed as designed. The data acquisition hardware worked as expected, and the PCs established communications with each other and functioned as expected, although not until after much software debugging.

Of-course, table-top testing cannot reveal all possible problems. In particular, there may be undiscovered interactive problems with the PRR-1 Instrumentation System. (Some shielding against radio and magnetic interference is already anticipated.) The PC set-up will be installed and actually used when the PRR-1 Instrumentation System is reactivated. It is expected that any such problem will be minor and manageable. The greater problems of working out the hardware design and writing and debugging the software can be considered solved.

CONCLUSIONS

The research contract successfully showed that inexpensive off-the-shelf PC hardware can be connected to a research reactor instrumentation system with minimal modifications to both PCs and instrumentation, allowing comprehensive computerized data monitoring without compromising reactor safety.

It was found that most of the effort in an undertaking such as this will go towards software development and debugging. Hardware knowledge is essential, but the effort expended in software is much less straight-forward and is much more time-consuming than the effort expended in hardware.

It was discovered that technological advances in PC hardware are so rapid that by the end of software development, the original PC hardware will be obsolete and may be advantageously replaced with newer hardware with the same cost. Replacement is made easier by the general downward-compatibility of new PC hardware. Software should be written with eventual hardware upgrading in mind.

It was proven that even an ancient programming language like Fortran can be used in a project such as this, provided that the language is fortified with subroutines written in Assembler. The language combination may be the natural choice for non-computer-specialized reactor people who wish to write and maintain their own PC software, and whose main experience in programming is in Fortran.

The research contract failed to acquire real operating experience because of the inoperable state of the PRR-1. The basic conclusions are not affected, however, and real experience may be gained in the future after the PRR-1 is rehabilitated.

It should be noted that the hardware was designed to connect with a generic TRIGA analog console, of which there are many world-wide. The design developed is readily adaptable to any of these consoles, and with only a little more work, to any other instrumentation system with similar electrical separation between redundant safety circuits.

Appendix A

VARIABLES MONITORED BY THE PRR-1 INSTRUMENTATION SYSTEM

Variable	*Nature*	*Voltage*	*Location*
Pulse Mode	Digital	+15 VDC	Right Console
Cooling Mode	Digital	+15 VDC	Right Console
Wide-Range Log Power	Analog	0 to +10 VDC	Left Console
Wide-Range Linear Power	Analog	0 to +10 VDC	Left Console
Wide-Range Period	Analog	0 to +10 VDC	Left Console
Linear Power #1	Analog	0 to +10 VDC	Left Console
Linear Power #2	Analog	0 to +10 VDC	Right Console
Demand Power	Analog	0 to +10 VDC	Center Console
Pulse in Progress	Digital	+15 VDC	Right Console
1 kW Permissive	Digital	TTL level	Left Console
Pulse Peak Power	Analog	0 to +10 VDC	Right Console
Pulse Energy	Analog	0 to +10 VDC	Right Console
High Voltage (Wide-Range)	Digital	TTL level	Left Console
High Voltage (Linear Power #1)	Digital	TTL level	Left Console
High Voltage (Linear Power #2)	Digital	TTL level	Right Console
Fuel Temperature #1	Analog	0 to +10 VDC	Left Console
Fuel Temperature #2	Analog	0 to +10 VDC	Right Console
Pool Water Temperature	Analog	0 to +10 VDC	Right Console
Safety Blade #1 Position	Analog	0 to +10 VDC	Center Console
Safety Blade #2 Position	Analog	0 to +10 VDC	Center Console
Safety Blade #3 Position	Analog	0 to +10 VDC	Center Console
Safety Blade #4 Position	Analog	0 to +10 VDC	Center Console
Regulating Rod Position	Analog	0 to +10 VDC	Center Console
Transient Rod Position	Analog	0 to +10 VDC	Center Console
Safety Blade #1 Magnet Contact	Digital	6 3 VAC	Center Console
Safety Blade #2 Magnet Contact	Digital	6 3 VAC	Center Console
Safety Blade #3 Magnet Contact	Digital	6 3 VAC	Center Console
Safety Blade #4 Magnet Contact	Digital	6.3 VAC	Center Console
Regulating Rod Magnet Contact	Digital	6.3 VAC	Center Console
Transient Rod Air	Digital	6 3 VAC	Center Console
Rod Withdrawal Prohibit	Digital	TTL level	Center Console
Rod Control Power	Digital	110 VAC	Center Console
Left-Hand Drawer Power	Digital	110 VAC	Left Console
Right-Hand Drawer Power	Digital	110 VAC	Right Console
Primary Hot Leg Temperature	Analog	0 to +10 VDC	Right Cabinet
Primary Cold Leg Temperature #1	Analog	0 to +10 VDC	Right Cabinet
Primary Cold Leg Temperature #2	Analog	0 to +10 VDC	Left Cabinet
Secondary Cold Leg Temperature	Analog	0 to +10 VDC	Right Cabinet
Secondary Hot Leg Temperature	Analog	0 to +10 VDC	Right Cabinet
Primary Flow Rate	Analog	0 to +10 VDC	Right Cabinet
Secondary Flow Rate	Analog	0 to +10 VDC	Right Cabinet
Clean-Up Flow Rate	Analog	0 to +10 VDC	Left Cabinet
Emergency Flow Rate	Analog	0 to +10 VDC	Left Cabinet
Primary Pump Differential Pressure	Analog	0 to +10 VDC	Left Cabinet
Clean-Up Water Conductivity	Analog	0 to +10 VDC	Center Cabinet
Primary Water Conductivity	Analog	0 to +10 VDC	Center Cabinet
Secondary Water Conductivity	Analog	0 to +10 VDC	Center Cabinet
Clean-Up Water pH	Analog	0 to +10 VDC	Center Cabinet
Primary Water pH	Analog	0 to +10 VDC	Center Cabinet
Secondary Water pH	Analog	0 to +10 VDC	Center Cabinet
Pool Level #1	Analog	0 to +10 VDC	Right Cabinet
Pool Level #2	Analog	0 to +10 VDC	Left Cabinet
Raw Water Tank Level	Analog	0 to +10 VDC	Left Cabinet

Variable	*Nature*	*Voltage*	*Location*
Retention Tank Level	Analog	0 to +10 VDC	Right Cabinet
Seismic Detector	Digital	+24 VDC	Annunciator
Bridge Position	Digital	+24 VDC	Annunciator
Air Radiation Level	Analog	0 to +VDC	Air Monitor
Bridge Radiation Level	Analog	0 to +4 VDC	Area Monitor
Reactor Bay Radiation Level	Analog	0 to +4 VDC	Area Monitor
Thermal Column Radiation Level	Analog	0 to +4 VDC	Area Monitor
Gamma Room Radiation Level	Analog	0 to +4 VDC	Area Monitor
Normal Exhaust Radiation Level	Analog	0 to +4 VDC	Area Monitor
Hot Vent Radiation Level	Analog	0 to +4 VDC	Area Monitor
Sump Pit Radiation Level	Analog	0 to +4 VDC	Area Monitor
Clean-Up Demineralizer Radiation Level	Analog	0 to +4 VDC	Area Monitor
Tank Room Radiation Level	Analog	0 to +4 VDC	Area Monitor
Cooling Tower Radiation Level	Analog	0 to +4 VDC	Area Monitor
Control Room Radiation Level	Analog	0 to +4 VDC	Area Monitor
Fuel Storage Radiation Level	Analog	0 to +4 VDC	Area Monitor
Primary Pump On	Digital	220 VAC	Motor Control
Secondary Pump #1 On	Digital	220 VAC	Motor Control
Secondary Pump #2 On	Digital	220 VAC	Motor Control
Emergency Pump On	Digital	220 VAC	Motor Control
Make-Up Pump On	Digital	220 VAC	Motor Control
Clean-Up Pump On	Digital	220 VAC	Motor Control
Transfer Pump On	Digital	220 VAC	Motor Control
Drain Pump On	Digital	220 VAC	Motor Control
Sump Pump On	Digital	220 VAC	Motor Control
Storm Pumps On	Digital	220 VAC	Motor Control
Tower Fan #1 On	Digital	220 VAC	Motor Control
Tower Fan #2 On	Digital	220 VAC	Motor Control
Intake Fan On	Digital	220 VAC	Motor Control
Exhaust Fan #1 On	Digital	220 VAC	Motor Control
Exhaust Fan #2 On	Digital	220 VAC	Motor Control
Exhaust Fan #3 On	Digital	220 VAC	Motor Control
Emergency Fan On	Digital	220 VAC	Motor Control
Transient Rod Compressor On	Digital	220 VAC	Motor Control
Turbo Compressor On	Digital	220 VAC	Motor Control
General Service Compressor On	Digital	220 VAC	Motor Control
Isolation Valve #1 Position	Digital	110 VAC	Motor Control
Isolation Valve #2 Position	Digital	110 VAC	Motor Control
Cooling Tower #1 Vibration	Digital	+24 VDC	Annunciator
Cooling Tower #2 Vibration	Digital	+24 VDC	Annunciator
Tower Basin Low Level	Digital	+24 VDC	Annunciator
Low Secondary Sample Flow	Digital	+24 VDC	Annunciator
Mains AC Power Fail	Digital	+24 VDC	Annunciator
UPS In AC Power Fail	Digital	+24 VDC	Annunciator
UPS Battery Low	Digital	+24 VDC	Annunciator
UPS AC Output Fail	Digital	+24 VDC	Annunciator
Generator AC On	Digital	+24 VDC	Annunciator
Transfer Switch To Generator	Digital	+24 VDC	Annunciator
Generator Start-Up Manual	Digital	+24 VDC	Annunciator
Nest #1 Power Fail	Digital	+24 VDC	Annunciator
Nest #2 Power Fail	Digital	+24 VDC	Annunciator

Appendix B

GENERAL SPECIFICATIONS OF DATA ACQUISITION HARDWARE

PCL-812PG Data Acquisition Card

- Sixteen single-ended A/D input channels Software-programmable bipolar input ranges ±5V, ±2.5V, ±1 25V, ±0 625V, ±0 3125V Accuracy 0 015% of reading ± 1 bit Linearity ± 1 bit. Overvoltage ±30V maximum continuous
- HADC574Z 12-bit successive approximation A/D converter Maximum A/D sampling rate 30 kHz in DMA mode
- Three A/D trigger modes software trigger, programmable pacer trigger, external TTL-compatible pulse trigger
- Three A/D data transfer modes by program control, by interrupt routine, or by DMA transfer
- Sixteen TTL-compatible D/I channels.
- Two 12-bit monolithic multiplying D/A output channels Output range of 0 to +5V or 0 to +10V with on-board voltage reference, other ranges possible with external reference AD7541AKN D/A converter Linearity: ±0.5 bit Output drive ±5 mA max. Settling time 30 microseconds
- Sixteen TTL-compatible D/O channels.
- Three programmable 16-bit timer/counter channels in Intel 8253 device. Two channels permanently connected to on-board 2 MHz clock as programmable pacer, one channel free for user application Pacer programmable from 35 minutes per pulse to 0.5 MHz External gate TTL-compatible
- PC interrupt channel jumper-selectable between IRQ 2 or IRQ 7 Interrupt enabled through software
- PC DMA channel jumper-selectable between channels 1 or 3 DMA enabled through software
- PC I/O requirements. 16 consecutive addresses, with base address DIP-switch selectable for address lines A8 to A4
- PC slot requirement. full-length 8-bit XT-type
- Signal connectors Five 20-pin post headers, one each for A/D, D/I, D/A, D/O, and counter

PCLD-780 Screw Terminal Board

- Universal screw terminal board to provide convenient signal connection to PCL-812PG card Contains 40 screw terminals Connects to any two PCL-812PG post headers with two 20-pin flat cables

PCLD-782 Opto-Isolated D/I Board

- Sixteen current-input channels in on-board screw terminals Input current 80 mA max to each channel 4N25 opto-isolator devices used Withstanding voltage 1500 VDC Input conditioning circuits installable on-board
- Sixteen output channels connect to D/I post header of PCL-812PG through 20-conductor flat cable Each channel buffered by voltage comparator

PCLD-785 Relay Output Board

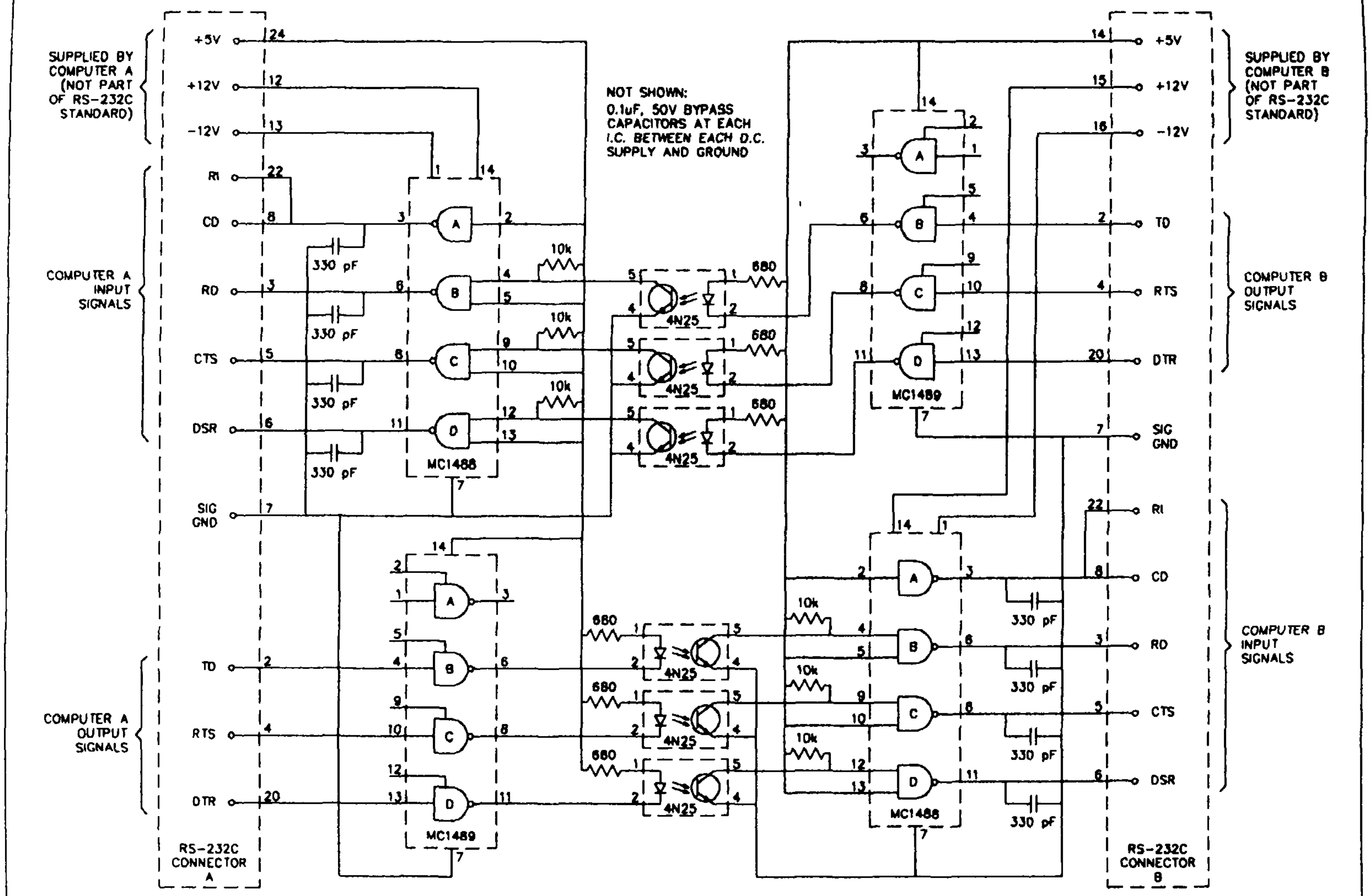
- Sixteen SPDT relays controlled by D/O from PCL-812PG card Relay contacts available in on-board screw terminals Contact rating: 120V AC/DC, 1 A. Breakdown voltage: 500V AC/DC minimum Insulation resistance 100 Mohm typical Total switching time: 10 msec typical

- Power supply: +12 VDC jumper-selectable as external or from PC through PCL-812PG card

- Connects to D/O post header of PCL-812PG card with 20-conductor flat cable.

Appendix C

SCHEMATIC OF OPTO-ISOLATED NULL MODEM CONNECTOR



NOTE: *Communications may not be reliable at a baud rate above 19,200.*

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PROGRAM PACKAGE FOR 2D BURNUP CALCULATION*

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Abstract

The program package for 2 dimension burnup calculation was developed for TRIGA Mark III reactor The package consistes of 3 modules PRESIX, SIXTUS-2, and BURN, 1 library, and 2 input files PRESIX module prepared cross sections for diffusion calculation SIXTUS-2 module, a two dimensional diffusion code in hexagonal geometry, calculates k_{eff}, neutron fluxes and power distributions BURN module performs the burnup of fuel elements and stored the result in the ELEM DAT file PRESIX LIB is two group cross section library for major reactor core components prepared using WIMS-D4 code PRES INP, the first input file, reads information on reactor power and core loading pattern ELEM DAT, the second input file, is prepared for specific TRIGA reactor and dependent on operation history To verify the reactor model and computational methods, the calculated excess reactivities were compared to the measurement The results are in good agreement

1. INTRODUCTION

The Thai Research Reactor 1/Modification 1 (TRR-1/M1) is TRIGA Mark III At the beginning, the core was loaded with 8 5/20 standard fuel elements The fuel management and burnup calculation of TRR-1/M1 were performed by hand calculation or by using TRIGAP, one dimensional diffusion code[1] which gives quite accurate results for homogenous core Afterwards, The OAEP has planned to upgrade the performance of TRR-1/M1, especially, improving the fuel economy by replacing the 8 5/20 standard element by 20/20 LEU fuel element The LEU fuel has higher uranium concentration, the power peaking as well as maximum temperature is expected to be higher than the homogenous standard fuel element core In this case, 2-dimensional calculation is more appropriate for calculating the burnup and fuel management The purpose of this study is to develop a program package for 2-D Burnup calculation of TRIGA Mark III reactor, especially for mixed core reactors

Program package was developed for IBM/PC 386 machine or higher using NDP FORTRAN compiler ver 4 0 2 Microway Inc The program package consists of 3 modules PRESIX, SIXTUS-2 and BURN, one library, PRESIX LIB (the effect group constants of the materials used in the reactor) , 2 input files PRESIX INP (core loading pattern and FLAG for Xenon correction), ELEM DAT (fuel element data) More details on modules, library and files are briefly described in Sec 2

To verify the reactor model and computational methods, the cross section library was prepared for TRR-1/M1 at 1 MW operation The calculated reactivity of the beginning of each core loading was compared to the measured reactivity The effective axial buckling was

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* Research carried out in association with the IAEA under Research Contract No THA/6704

adjusted until the calculation is in good agreement with measurement. the calculated excess reactivities of core loading no. 1-7 were compared to measured excess reactivities In general, results are in good agreement. It can be concluded that the 2 D package can be used for burnup calculation of TRIGA reactor, especially, TRR-1/M1 with LEU/Standard mixed core.

2. STRUCTURE OF PROGRAM PACKAGE

The program package consists of 3 modules, 1 library and 2 input files which are presented as a flow chart in Figure 1 are briefly described as following ·

2.1. PRESIX Module

PRESIX module prepares cross section for SIXTUS-2 module by reading (1) dependent input data, e.g. reactor power, core loading pattern, and number of fuel elements of each type from 'PRES.INP' file (2) fuel element burnup history from 'ELEM.DAT' file and (3) group constants for major materials used in the reactor from PRESIX.LIB If the fission products poisoning is considered. The corrections are calculated as equation (1)

$$\Delta\Sigma^x = \Delta^x(\tau_0) + \frac{\Delta^x(\tau_1) - \Delta^x(\tau_0)}{\tau_1 - \tau_0}(BUP - \tau_0) \qquad (1)$$

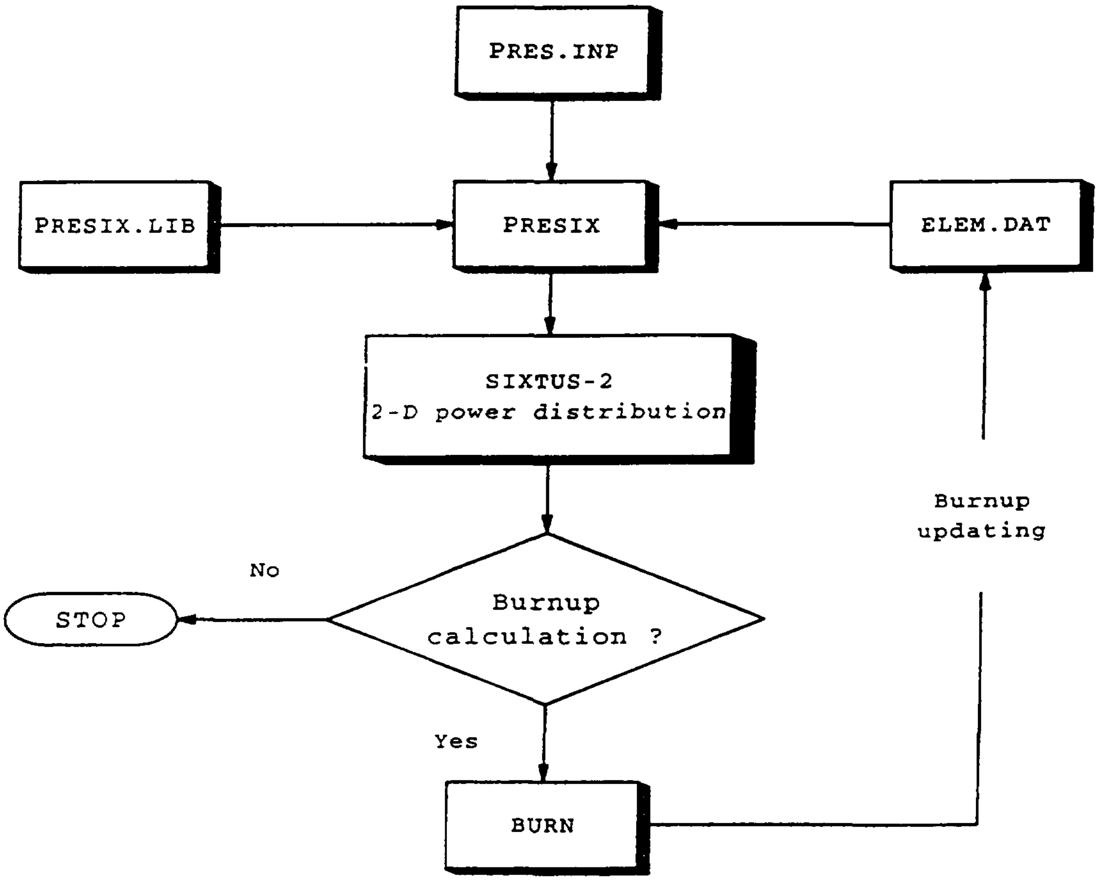


Fig. 1. Flowchart of Program Package

Where BUP	=	current burnup in % of U-235
$\Delta\Sigma^x$	=	Xenon correction to 2-group constants for unit cell
$\Delta^x(\tau_0)$	=	Difference between group constants at full power, equilibrium xenon and zero power, no xenon at burnup equals τ_0
$\Delta^x(\tau_1)$	=	Difference between cross sections at full power, equilibrium xenon and no xenon at burnup equals to τ_1

If the burnup of the fuel element falls between two steps in the library, the effective unit cell cross sections are obtained by linear interpolation as shown in Figure 2.

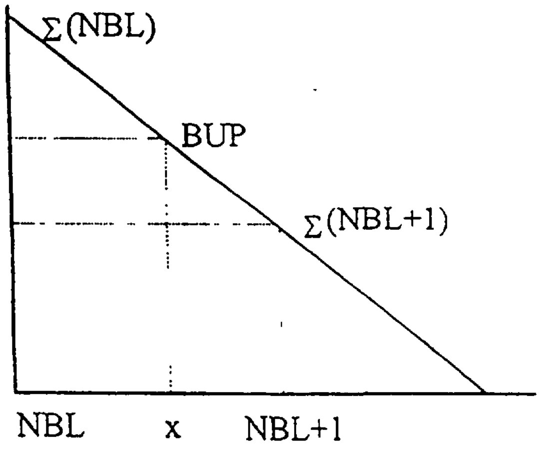


Fig. 2. Linear interpolation of cross sections

Cross sections at any burnup, x

$$\Sigma(x) = \Sigma(NBL) + \frac{\Sigma(NBL+1) - \Sigma(NBL)}{BUR(MN,NBL+1) - BUR(MN,NBL)}(BUP(NODE) - BUR(MN,NBL)) \quad ....(2)$$

Where

BUP(NODE) = Current burnup of element at NODE

BUR (MN,NBL) = Burnup of fuel element type MN, at burnup step BNL in the 'PRESIX.LIB' library

BUR (MN,NBL+1) = Burnup of fuel element type MN, at burnup step BNL+1 in the 'PRESIX.LIB' library

Then 'PRESIX' module arranges the cross sections in the form for SIXTUS-2 module as follows:

Position	**Group 1**	**Group 2**
1	Σ_{f1}	Σ_{f2}
2	Σ_{tr1}	Σ_{tr2}
3	Σ_{a1}	Σ_{a2}
4	$\upsilon_1\Sigma_{f1}$	$\upsilon_2\Sigma_{f2}$
5	Σ_{t1}	Σ_{t2}
6	$\Sigma^{2\to1}$	0
7	0	$\Sigma^{1\to2}$

2.2. SIXTUS-2 Module

SIXTUS-2 is a two dimensional multigroup diffusion theory code in hexagonal geometry [2]. Originally, the code is written in FORTRAN EXTENDED version 4.8 for CDC series 6000 and CYBER computers. Later on, VAX version is also available. In this work, the VAX version is converted and modified to run on PC 386 machine by using NDP FORTRAN 77 compiler. Some subroutines were modified to be able to run on PC machine. Three new subroutines namely GRAPHIC.FOR, PGRAPH.FOR, and GLIB.FOR were developed for display and print out graphic mode. After verifying with the reference data to ensure the correctness and accuracy of the code, SIXTUS-2 code is used as a module in the program package. The sample input files for SIXTUS-2 is listed in APPENDIX A. The manual of SIXTUS-2/PC is listed in APPENDIX B

2.3. BURN Module

BURN module reads the input from PRES.INP and ELEM.DAT. Normalized power distributions of fuel elements are obtained from SIXTUS-2 module and stored in a file named 'POWER.NOR'. Then accumulative burnup or burnup. BU, of j^{th} fuel element during time step d_l can be calculated from [3]

$$BU(t_n)_j = \sum_{l=1}^{n} \Delta BU(d_l)_j = \sum_{l=1}^{n} (P_j^l)(d_l)_j \qquad \text{...........(3)}$$

$$= \sum_{l=1}^{n} (NP_j^l)(\bar{P})(d_l) \qquad \text{...........(4)}$$

Where $t_n = \sum_{l=1}^{n} d_l$

P_j^l = Power of j^{th} fuel element

$\overline{P}$ = Average power

NP_j^l = Normalized power of j^{th} fuel element at time step d_l

2.4. PRESIX.LIB Library

Before preparing the cross section library, the sensitivity study has been performed. The reactivity changes when the reactor is brought from zero power to full power are due to the temperature increase of (1) fuel meat, (2) clad, and (3) water. At 1 MW, the average temperature in the fuel meat is estimated to be 230 °C [4]. The results of standard element unit cell calculation using WIMS/D4[5] are shown in Table I.

TABLE I. REACTIVITIES CHANGES DUE TO TEMPERATURES

	Meat	Clad	Water	δk/k (%)
Reference °C	20	20	20	-
Increase fuel temp. only °C	235	20	20	-2.50
Increase clad temp. only °C	20	100	20	0.0
Increase water temp. only °C	20	20	54	+0.004
			Total	-2.496

Since the major change in reactivity dues to temperature of fuel meat, the temperature of fuel elements are estimated from ref.[4] and temperature of clad and water are kept constant at 100 °C and 54 °C, respectively.

	LEU	STD	LEU follower	STD follower
Fuel temp. °C	267	230	263	225
Clad temp. °C	100	100	100	100
Water temp. °C	54	54	54	54

The PRESIX.LIB is prepared for 1 MW TRIGA. It consists of fuel element (20 wt %, 20% enrichment, Er as burnable poison), standard fuel element (8.5 wt %, 20% enrichment), LEU fuel follower element and standard fuel follower element. It contains effective two group constants for all major types of fuel and nonfuel unit cells in TRR-1/M1 reactor. All unit cell have the same volume. Fuel unit cell contains fuel rod and water. For non fuel, the super cell approximation is used i.e. central non fuel rod surrounded by six fuel rods and water. The model for fuel and non-fuel unit cell is shown in Figure 3. All unit cell cross sections were calculated using WIMS/D4 with group boundary at 1.07 eV The cross sections of fuel elements are prepared used as-built weights and are tabulated as function of burnup up to 40%.

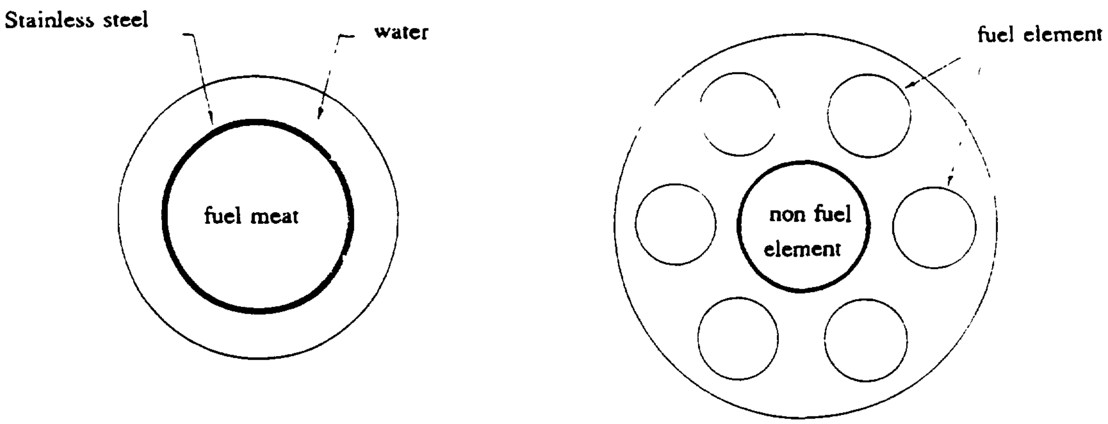


Fig. 3. Model for unit cell calculation

The data in the library are written in the following orders :

(1) Cross section for LEU, Standard, LEU fuel follower and Standard fuel follower elements at each time step are written as follows :

$$D_1 \qquad \Sigma_{a1} \qquad \Sigma_{12} \qquad \upsilon_1\Sigma_{f1}$$

$$D_2 \qquad \Sigma_{a2} \qquad \Sigma_{21} \qquad \upsilon_1\Sigma_{f2}$$

(2) Cross sections of non-fuel cells are written in the following orders : transient rod, water cell, dry irradiation channel, graphite element, graphite reflector, water reflector, neutron detector, and wet irradiation channel.

(3) burnup step in percent of U-235 for LEU, Standard, LEU fuel follower and standard fuel follower elements

(4) number of fission neutrons/fission for each group .

(5) percent burnup at burnup equals τ_0

(6) Xenon correction factor : $\Delta^x(\tau_o)$

(7) Xenon correction factoı (slope) : $\dfrac{\Delta^x(\tau_1) - \Delta^x(\tau_o)}{\tau_1 - \tau_o}$

2.5 PRESIX.INP File

PRESIX.INP file contains all independent input data : thermal power of the reactor, number of fuel elements of each type, burnup time step, core loading patterns and FLAG for xenon correction. All input data must be written in the prescribed order and format.

Card no. 1	format A80 : jobid
Card no. 2	free format
	power : Power - thermal power in kW
	bzsq : effective axial buckling
Card no. 3	free format
	nl : no. of LEU fuel elements in core
	nstd : no. of STD fuel elements in core
	nfl : no. LEU fuel follower elements in core
	nfs : no.of STD fuel follower elements in core
Card no. 4	format A80 : any comment
Card no. 4	format 6(16,212)
	idel (i) : element identification
	ixe (i) : FLAG for xenon correction

2.6 ELEM.DAT File

ELEM.DAT is a file prepared for specific TRIGA reactor and dependent on the operation history reactor. It contains the following data for each element : identification number of the element, type of the element, and history of the burnup of fissile element in MWD and % burnup of U-235.

For characterizing different type of fuel or nonfuel element, the identification numbers are used as follows :

1 LEU fuel element
2 Standard fuel element
3 LEU fuel follower
4 Standard fuel follower
5 Transient rod
6 Water cell
7 Dry irradiation channel
8 Graphite element

9 Graphite reflector
10 Water reflector
11 Neutron detector
12 Wet irradiation channel

Identification number for fuel elements are given by General Atomic Co., the manufacturer of TRIGA fuel elements. It is not necessary to distinguish between different non fuel elements. Therefore, for the nonfuel elements, the same identification number for characterizing the type of those elements are used.

The data are written in the prescribed orders and formats.

Card no. 1 format A80 : comment

Card no.2 format (i6,f8.2)
num : numbers of fuel and non-fuel elements in the library
t : burnup time step

Card no.3 format A80: comment

Card no.4 format(f12.3,20x,f12.3)
eacc : accumulated burnup (MWD)
ecurr : current burnup (MWD)

Card no.5 format A80: comment

Card no. 6 format(4x,i6,5x,i3,3(6x,f9.5))
id : identification of element
ntype : type of fuel/non-fuel element
bmwdl(n) : burnup of n^{th} element (MWD)
bupl(n) : burnup of n^{th} element (% of U-235)

2.7. How to run the package

The package can be run using batch file 'PACKAGE.BAT' as follows

PRESIX

SIXTUS INP = TRR.INP OUTA = TRR.OUT XSEC = XSEC R

BURN

To run PRESIX module needs PRES.INP, ELEM.DAT, and PRESIX.LIB, respectively. The cross sections are stored in 'XSEC' file.

To run SIXTUS module needs 'TRR.INP' file and 'XSEC' (cross sections) file, the output is stored in 'TRR.OUT'

To run BURN module needs 'PRES.INP', ELEM.DAT, and POWER.NOR files, respectively.

Examples of files and PRESIX.LIB are listed in APPENDIX A.

3. VERIFICATION OF THE PACKAGE

3.1. Fission product poisoning

The fission products generated in the reactor during operation in the megawatt power range, or higher, poison the core significantly. Xenon-135 is the principal fission product poison. The chain of production of Xe-135 is shown in Figure 4.[6]

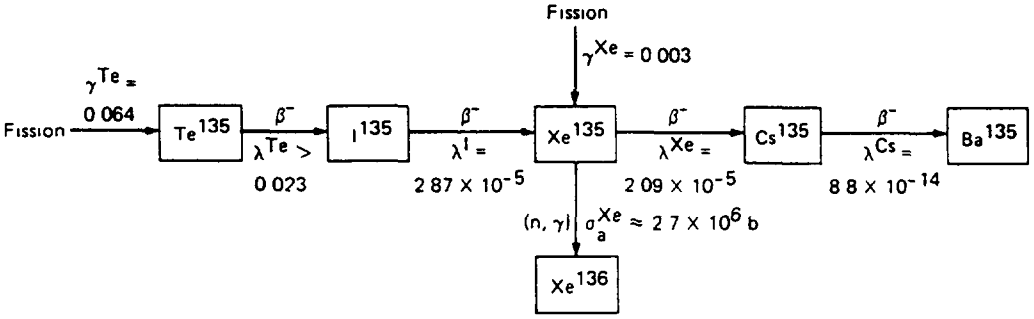


Fig. 4. The ^{135}Xe fission product chain

Where γ^{Te} and γ^{Xe} are the fission yields of tellurium and xenon
Assuming one group neutron and constant flux, the production of I-135 and Xe-135 are

$$I(t) = \frac{\gamma^I \Sigma_f \Phi}{\lambda^I}\left[1 - \exp(-\lambda^I t)\right] + I(0)\exp(-\lambda^I t), \quad \ldots\ldots. (5)$$

$$X(t) = X(0)\exp(-(\sigma_a^{Xe}\Phi + \lambda^{Xe})t) + \frac{\gamma \Sigma_f \Phi}{\sigma_a^{Xe}\Phi + \lambda^{Xe}}\left[1 - \exp(-(\sigma_a^{Xe}\Phi + \lambda^{Xe})t)\right]$$

$$-\frac{\gamma^I \Sigma_f \Phi - \lambda^I I(0)}{\lambda^I - \lambda^{Xe} - \sigma_a^{Xe}\Phi}\left[\exp(-(\sigma_a^{Xe}\Phi + \lambda^{Xe})t) - \exp(-\lambda^I t)\right], \quad (6)$$

Where $\gamma = \gamma^I + \gamma^{Xe}$

After shutdown the reactor, the changes of I-135 and Xe-135 are as following

$$I(t) = \frac{\gamma^I \Sigma_f \Phi(0)}{\lambda^I}\exp(-\lambda^I t) \quad (7)$$

$$X(t) = \frac{\gamma \Sigma_f \Phi(O)}{\sigma_a^{Xe}\Phi(O) + \lambda^{Xe}}\exp(-\lambda^{Xe} t) + \frac{\gamma^I \Sigma_f \Phi(O)}{\lambda^I - \lambda^{Xe}}\left[\exp(-\lambda^{Xe} t) - \exp(-\lambda^I t)\right] \quad (8)$$

Normally, the reactor is operated 7 hours per day, 5 days a week, at the power of 1 MW
To study the effect of Xenon during operation of TRR-1/M1, 2 cases are considered

(1) continuous operation for 5 days and shutdown for 2 days during the weekend
(2) operation for 7 hours during the weekday and shutdown for 2 days during the weekend

Both cases, the xenon effect was calculated assuming average neutron flux is 1 x10^{13} n/cm^{2}s^{-1} The result of xenon effect is shown in Figure 5 The solid dots represent xenon effects during operation and white dot represented the Xenon effect during shutdown At equilibrium xenon, the reactivity was approximately 1.6 % δk/k It is shown in Figure 4 that the average Xenon concentration for such operation approximately 50% of equilibrium value

3.2. Burnup calculation

It is difficult to model the actual xenon operating history, because the reactor does not operated continuously Also xenon has only small effect on power shape and consequently on burnup distribution in a small reactor like TRIGA reactor The simplify the calcualtion, the burnup calculations were performed at no xenon condition. The model of calculation is shown in Figure 6 Nodes 1-127 represents the reactor core Nodes 128-331 represents the water reflector There are 5 control rods namely transeint rod(TR), regulating rod(REG), Shim1 rod (SH2), Shim2 rod (SH2), and Safety rod (SAFE) The calculated reacitvity of the beginning of each core loading was compared to the measured reactivity The efffective axial buckling was adjusted until the calculation is in good agreement with the measurement The calculation of core loading no 1-7 were performed and results are summarized in Table II The comparison of calculation to the measurement for core loading no. 1-7 are shown in Figures 7- 10, respectively

TABLE II. CALCULATED EXCESS REACTIVITY($) AS A FUNCTION OF BURNUP (MWD)

Core no.	step	1	2	3	4	5	6	7
1	MWD	0	3	20	40	61.23		
	K_{eff}	1.02935	1.02899	1.02402	1.02045	1.01718		
	ρ_{ex}	4.073	4.025	3.351	2.863	2.418		
2	MWD	61.23	80	100	120	137.53		
	K_{eff}	1.03436	1.03170	1.02957	1.02766	1.02614		
	ρ_{ex}	4.745	4.3893	4.102	3.845	3.639		
3	MWD	137.53	160	180	200	224.54		
	K_{eff}	1.02029	1.01841	1.01668	1.01470	1.01160		
	ρ_{ex}	2.841	2.582	2.344	2.070	1.638		
4	MWD	224.54	250	271.99				
	K_{eff}	1.02237	1.01932	1.01684				
	ρ_{ex}	3.126	2.708	2.366				
5	MWD	271.99	290	310	330	350	370	391.87
	K_{eff}	1.03412	1.02835	1.02620	1.02381	1.02160	1.01943	1.01720
	ρ_{ex}	4 713	3.938	3.647	3.328	3.020	2.682	2.416
6	MWD	391.87	410	430	455.98			
	K_{eff}	1.02379	1.02205	1.01994	1.01708			
	ρ_{ex}	3.319	3.082	2.793	2.414			
7	MWD	455.98	480	500	520	540	566.12	
	K_{eff}	1.02472	1.02211	1.02032	1.01868	1.01726	1.01522	
	ρ_{ex}	3.446	3.090	2.845	2 620	2.424	2.142	

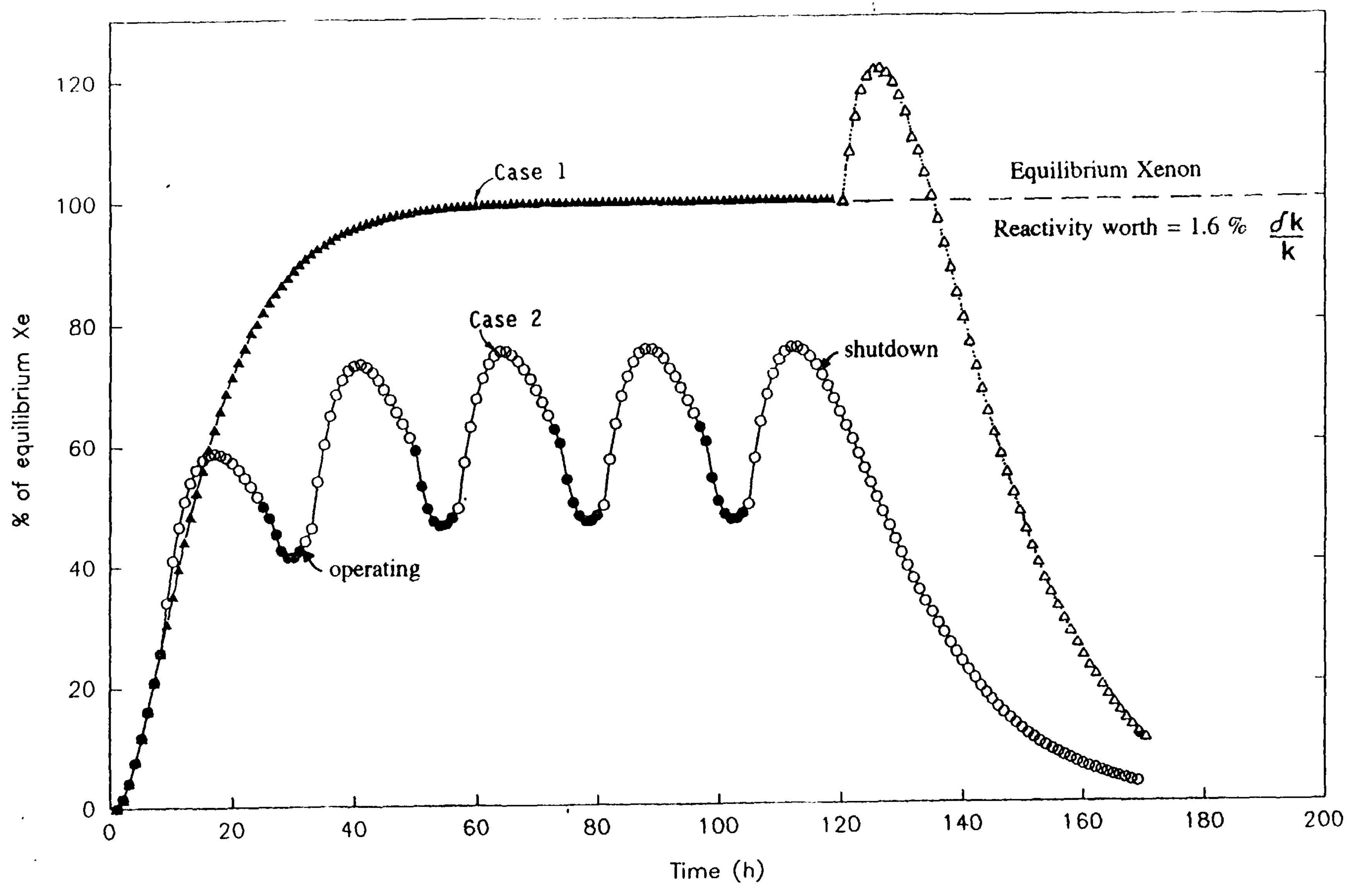


Fig. 5. Xenon buildup and decay

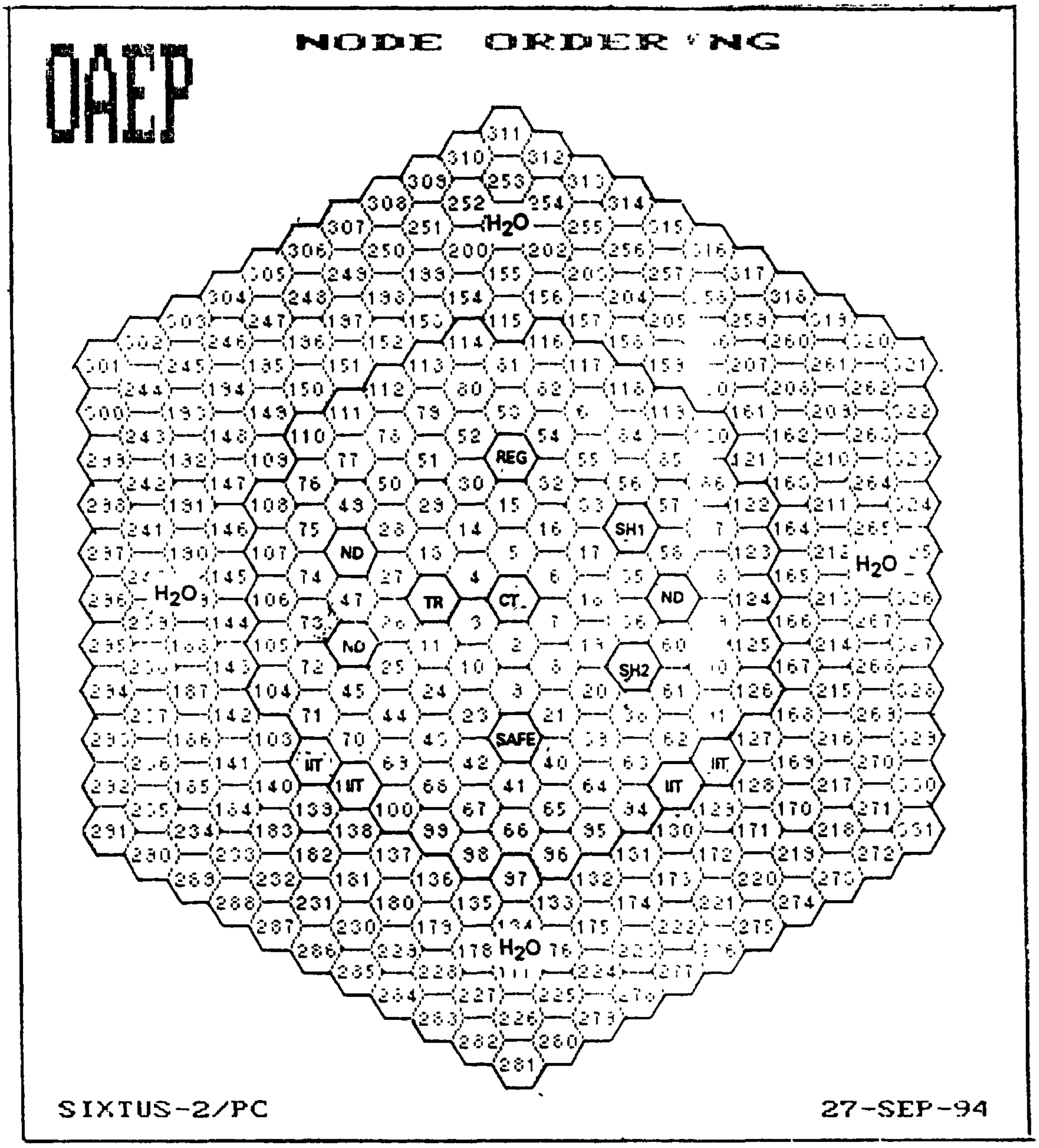


Fig. 6. Model of core calculation

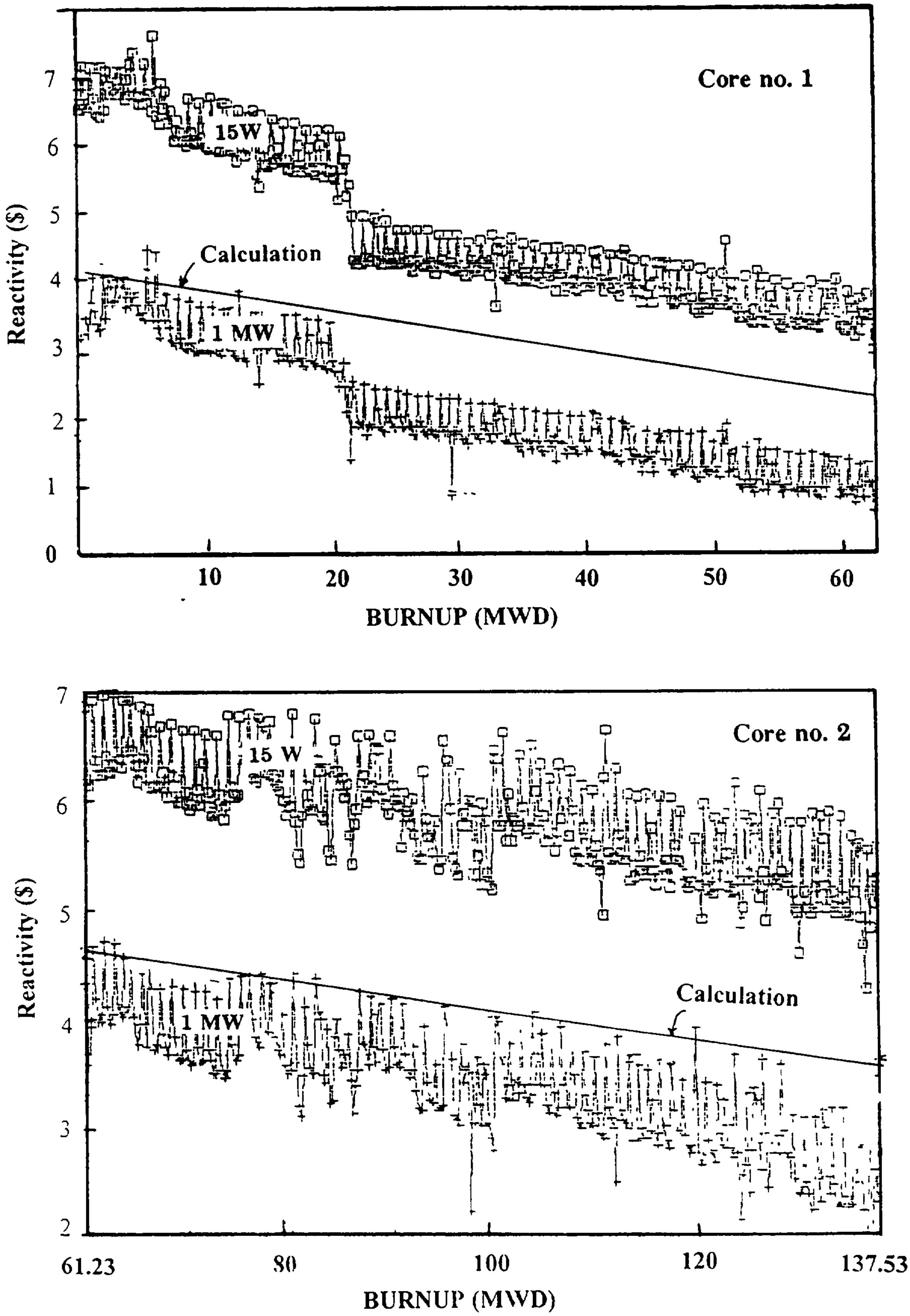
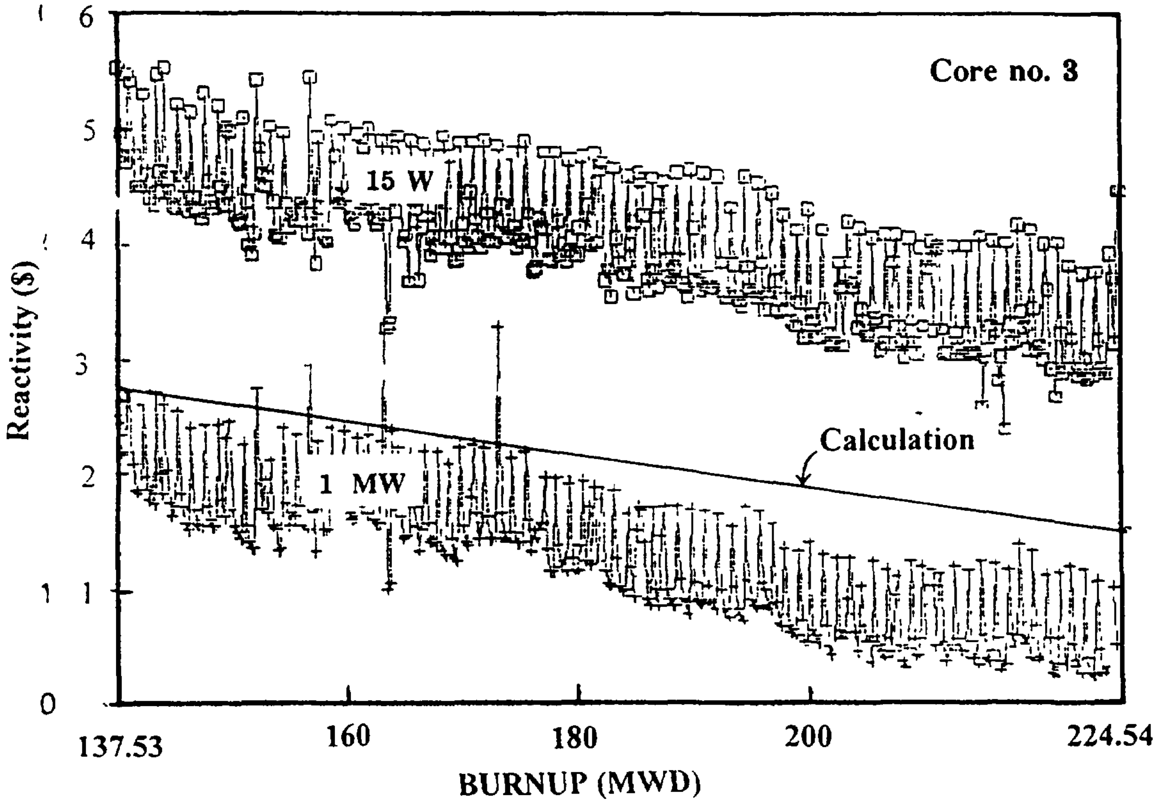


Fig 7 Measured excess reactivities of core no 1 and no 2 as a function of burnup compared to calculated excess reactivities



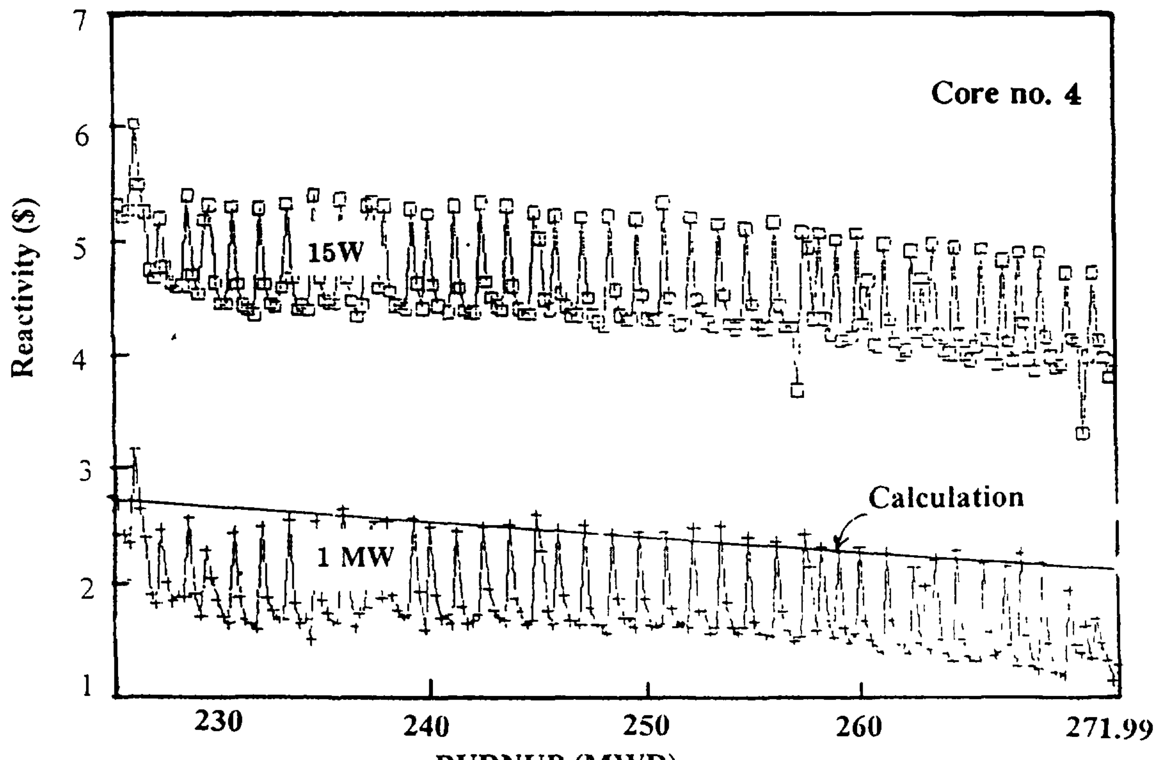


Fig. 8 Measured excess reactivities of core no.3 and no 4 as a function of burnup compared to calculated excess reactivities

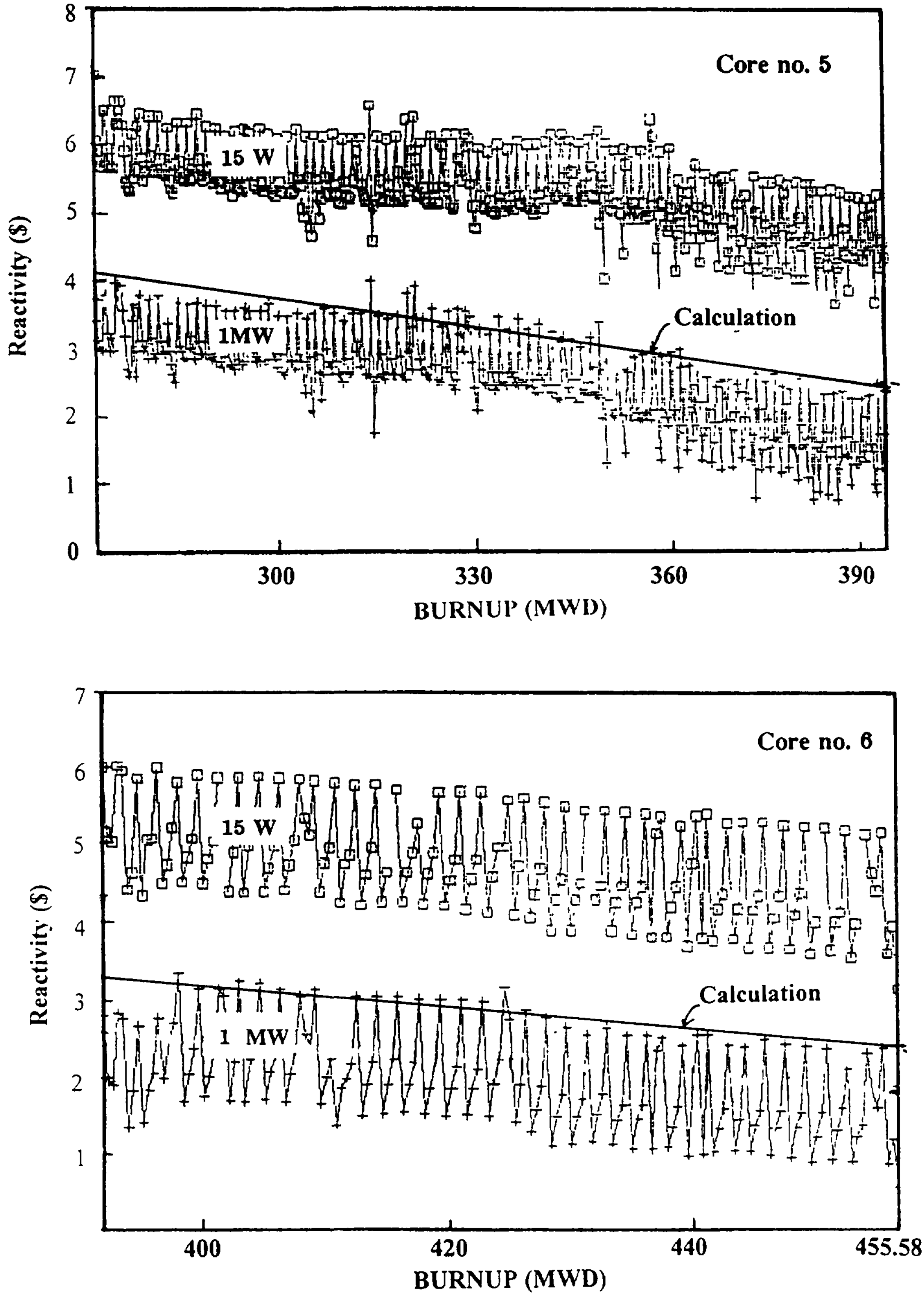


Fig 9 Measured excess reactivities of core no 5 and no 6 as a function of burnup compared to calculated excess reactivities

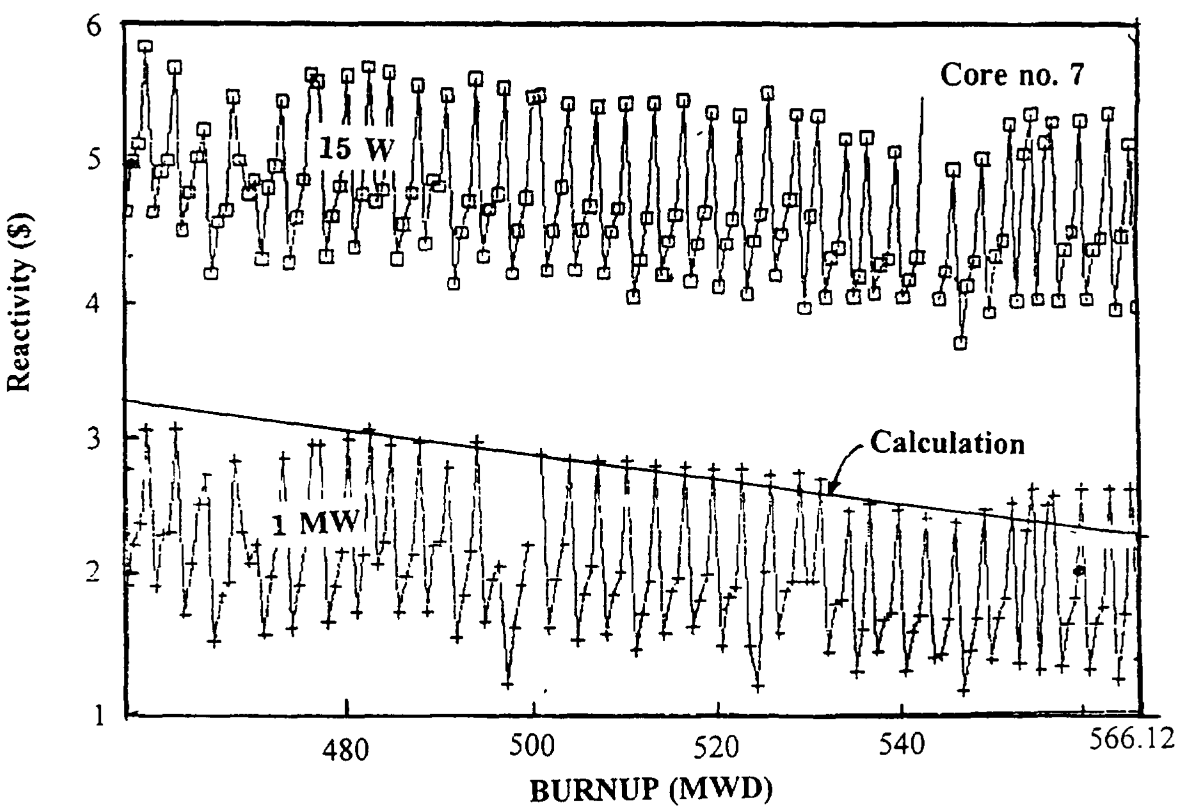


Fig. 10. Measured excess reactivities of core no. 7 as a function of burnup compared to calculated excess reactivities.

4. CONCLUSION

In general, the calculations are in good agreement with the measurements except core loading no.1. At the beginning of core no.1 up to 20 MWD , the results are still in good agreement , then a sudden drop of measured reactivities were observed. According to the data recorded in the log-book, this drop coincides with the re-calibration of control rod at the period. The total control rod worth after re-calibration is approximately $ 1 smaller then before. The drop is expected due to part of control was inserted during the calibration.

It can be concluded that the 2 D Package can be used for the burnup calculation of TRR-1/M1. In this study, the PRESIX.LIB, cross section library was prepared for 1 MW operation of TRR-1/M1. If the package is intended to use for other TRIGA reactor which may be operated at different power or has different type of fuel elements or different non-fuel elements. In this case, the new cross section library should be prepared for the specific reactor. The package is easy to use and give good prediction of the power peaking, especialy for mixed core core configuration, and burnup calculation.

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WINDOWS USER-FRIENDLY CODE PACKAGE DEVELOPMENT FOR OPERATION OF RESEARCH REACTORS *

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Abstract

The content of the project was to developed:

1 MS Windows interface to spectral codes like THERMOS, PEACO-COLLIS, GRACE and burn-up code.
2. MS Windows C-language burn-up diffusion hexagonal lattice code.

The overall scope of the project was to develop a PC-based MS Windows code package for operation of Dalat research reactor. Various problems relating to neutronic physics like thermalization, resonance treatment, fast spectral treatment, change of isotopic concentration during burn-up time as well as burn-up distribution in the reactor core are considered in parallel to application of informatics technique. The developing process is a subject of the concept of user-friendly interface between end-users and the code package. High level input features through system of icon, menu, dialog box with regard to Common User Access (CUA) convention and sophisticated graphical output in MS Windows environment was used. The user-computer interface is also enhanced by using both keyboard and mouse, which creates a very natural manner for end-user

1. RESULTS OBTAINED

I. MS Windows Interface to Spectral Codes

I.a. Introduction:

The set of codes THERMOS-PC, PEACO-COLLIS-PC, GRACE-PC (from now on we call THERMOS, PEACO-COLLIS, GRACE) has been adapted to PC under the IAEA Research Contract No 5304/R0/RB Here are some brief descriptions for the set of codes

THERMOS solves the integral transport equation for the thermal energy range in a cylindrically symmetric medium In the code, white, perfectly reflective and black boundary conditions can be applied The Brown-St. John model, the Nelkin model and the Koppel-Young model can be used for calculating the scattering kernels of hydrogen and deuterium bound in water The scattering model for oxygen and graphite is the free-gas model In the code, there is a special treatment for hydrogen bound in polyethylen The scattering is assumed to be isotropic in the

* Research carried out in association with the IAEA under Research Contract No. VIE/7950.

laboratory system but an optional transport correction can be applied too. The transport kernel is calculated on the base of a first flight probability technique. The spectral space-averaged parameters prepared by THERMOS will be provided to the code GRACE for full spectral lattice calculation

PEACO code solves the problem of resonance absorption of neutron in nuclear reactors. There are two main cases in the resonance treatment. The first one is the case where the resonances are considered to be theoretically well-separated, that is, level-spacing of resonances are much larger than Doppler width, this treatment has been applied to fertile materials at neutron energies below about 10KeV and has been considered to be suitable for most thermal reactors The second case occurs when the resonances are not well separated and/or when there are overlapping resonances due to different materials. This case is important for the fast and immediate reactors. Some approximations concerning the neutron flux behavior are proposed in this code.

PEACO code consists of two main parts of calculation. The resonance cross-sections calculated from resonance parameters over the resonance energy ranges is done by c des LINEAR, RECENT, SIGMA1, FEDGROUP and supplied by a binary file(PEO1.DAT input file in this version of PEACO) as a library needed for PEACO-MAIN (in short, we call as PEACO). Therefore, in this PC-based version, the code PEACO-COLLIS (in short we call as COLLIS) is considered to be the first part of the code PEACO, which prepares collision probabilities in the geometry of cylindrical rod or slab at specified values of the total cross section of resonance absorption compositions and will be used as representative points for collision-probability interpolation later The PEACO-MAIN (in short we call PEACO), the second part of the code PEACO, solves the neutron slowing down and calculates various averaged values such as effective resonance cross-sections, using the resonance cross-sections and input file prepared by COLLIS. The flux distribution and various averaged or integrated quantities are calculated by PEACO and will be provided to the code GRACE for full spectral lattice calculation.

GRACE code solves the fast neutron spectrum equation in slab geometry with the critical thickness (Keff = 1). In this case the real cell geometry is not reflected in the code. The cell characters are reflected here only in material composition and through the thermal and resonance date, which are submitted to the input of GRACE by THERMOS and PEACO codes.

According to our experience in exploitation of this set of codes, we find that this code system has some limitations

1) Each code can only run separately, therefore users have to prepare input data for GRACE manually from the output data of THERMOS and PEACO-COLLIS. This needs a professional knowledge of the users
2) Some limitations incorporated with FORTRAN-Language running in DOS should be removed by application of MS Windows high-level features.
3) The existing code system has no ability to create group constants for fuel element as a function of burnup.

Therefore, we are developing a burnup code called HASOB which is a combination of codes THERMOS, GRACE and DOT. The code DOT is developed to calculate the change of isotopic concentration vs. burnup given neutron flux and depletion time as well as initial isotopic concentration. The MS Windows code HASO-W is developed to interface input/output of the new code system consisting of THERMOS, PEACO-COLLIS, GRACE and HASOB.

I.b-Results:

1. Development of the code DOT. The burnup equations are solved for U-235 and U-238 chains, which enable to calculate isotopic concentration as a function of time, given constant neutron flux. At present, the following isotopes are considered: U-235, U-236, U-238, Np-239, Pu-239, Pu-240, Pu-241, Pu-242, and an average fission product FP. The code DOT is written in FORTRAN. The results of testing the code DOT is shown in Appendix 1. This work has been done.
2. Modifying reading and writing commands in THERMOS, PEACO-COLLIS and GRACE codes to automatically preparing input data for GRACE. This work has been done.
3. Combination of THERMOS + GRACE + DOT into a unique code named HASOB. In this version of HASOB, we still assume no spectral correction in microscopic cross-sections for isotopes concentration calculation. The code HASOB is developed in such a way that users can prepare input data for neutron thermalization and fast spectral treatment in the same way just like prepare input data for THERMOS and GRACE codes as individual. This work is in progress to investigate if the absence of library of some isotopes born in U-235 and U-238 chains can be negligible in low burnup calculation.
4. Develop HASO-W as a MS Windows interface in mixed language FORTRAN-C to control input preparation and output treatment for the set of codes THERMOS-PC, PEACO-COLLIS-PC, GRACE-PC and HASOB. We have used a system of menu, dialog box, on-line help and a built-in editor to enable the users to interact with the above codes as individual or as the whole. Using HASO-W, therefore, the users can run non-Windows codes from MS Windows environment. This work is in progress.

II. Development of MS Windows C-Language Burn-up Hexagonal Diffusion Code.

The set named HEXBW of codes HEXIN and HEXBURN is written in C-and FORTRAN-languages respectively for MS Windows in order to calculate Keff, power and burnup distribution in the hexagonal lattice core. The multi-group two-dimensional diffusion equation is solved by using finite difference method and acceleration techniques for inner and outer iterations. In the development of HEXBURN, we use a FORTRAN hexagonal diffusion code derived from India for reference and then we modify and further develop it to meet our needs. In the first stage, we develop HEXBURN in FORTRAN to run in MS DOS, which is the main code of the set and which can provide burnup distribution for each burnup zone and fuel element in the reactor core. In the second stage, we modify HEXBURN to run in MS Windows by using Microsoft FORTRAN version 5.1 combined with C-language. The code HEXIN is developed for input data preparation for HEXBURN. We use both parallel and consequent approaches to develop the set of two codes in order to accelerate the process and assurance validation of the codes. The works of the development process are as follows:

1. Modifying input data preparation of Indian hexagonal code for simulation of Dalat research hexagonal lattice core with different core symmetry configuration of 30, 60, 90 and 180 degree. This work has been done.
2. Developing the subroutine BURNUP for calculation of burnup distribution. The code DOT has been modified to be a routine in the subroutine BURNUP for calculation of isotopic concentration change given neutron flux and depletion time. The power distribution is normalized according to a given reactor power, then the real neutron flux is calculated and spectral-averaged for each burnup zone. The burnup distribution and the change of isotopic concentration are calculated for burnup zones. The change of fuel group constants as function of burnup is built in the code HEXBURN. Some results obtained on testing

HEXBURN for burnup calculation of Dalat research reactor are presented in Appendix 3. This work has been done.

3. Studying methods of control rods treatment and searching the critical states for obtaining neutron flux distribution for burnup calculation is being carried out.

4. Developing HEXIN in MS Windows C-language. All preparation of input data for HEXBURN will be done by the users in MS Windows environment by using a system of menu, dialog box, high-level color graphic reactor core simulation editor. Therefore, users can draw hexagonal reactor core with an appropriate system of row and column index using mouse and 16-color pallet. All other HEXBURN code's control and geometry data are given by users through a user-friendly system of menu and dialog box. We are using all of our informatics knowledge in co-operation with other outside experts to deal with various problems in developing such a very flexible input preparing code. Some input screens of HEXIN are shown in Appendix 2. This work has shown encouraging results and would need more skillful workforce.

5. Although several attempts has been tried to re-write HEXBURN in C-language, the test version has not been completed due to a complicated structure of this large code. The difficulty has been successsfully overcome by modifying the code to run in MS Windows as a mixed C + FORTRAN - Windows application. Therefore, the code can exploit the advantages of MS environment as it is our main purpose. The problem of matching of HEXIN and HEXBURN in terms of input/output interfaces is being carried out.

2. CONCLUSIONS

1. Development of a PC-based MS Windows Code Package (in the concept of user-friendly interface between end-users and the code package) is very useful not only for operation of Dalat research reactor but also for other similar research reactors. The code package would provide the reactor operation staff a usefull tool without requirement of qualified knowledge in mathematics and physics structure of the code package.

2. The combination between FORTRAN and C-language in MS Windows gives us an ability of construction of user's friendly interface between end-user and code package in reactor physics. Such a combination in using FORTRAN programming, depletion calculation and C-language programming in MS Windows has shown to be a promising approach to solve the above mentioned limitations of the existing spectral code system.

3. Development of HEXBW code including the HEXIN and HEXBURN in C- and mixed C + FORTRAN-languages respectively is proven to be very efficient and effective from the point of reactor physics view. It enables to evaluate the power/burnup distribution, as well as the reactor cycle in a very user-friendly way. The code HEXIN developed here is a typical example of the use of MS Windows with a system of menu, dialog box, combo box and color graphic screen. That facilitates the input preparation for a FORTRAN/C-language code without requirement of a professional knowledge of the code from the user.

4. The use of both parallel and consequent approaches to develop the code package in order to accelerate the process and assurance validation of the package has shown to be an appropriate and efficient way in development of rather large and complicated reactor physics package.

5. The co-operation with other outside informatics experts to deal with various problems in integration of reactor physics, computation and informatics is needed for developing the package.

6. The concept of PC-based user-friendly interface would be considered to be a new approach in re-engineering processes in development and exploitation of research reactor physics and thermohydrolics code systems.

7 Validation of the code package against experiment and Benchmark problems must be carried out in the next year. It requires more workforces and time to correct any undetected errors in the code package. The absence of library of some isotopes born in U235 and U238 chains may be negligible in low burnup calculations However, this assumption must be further carefully examined The effect of control rods on the anti-symmetry of axial neutron distribution in the core needs more efforts to be well established into HEXBURN

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APPENDIXES – THE RESULTS OBTAINED ON TESTING CODES

Appendix 1.

RESULTS OBTAINED ON TESTING CODES

Number of Isotopes is 9 Neutron Flux is 2 00E+12n/cm2s
Total Depletion Time is 5,000 0 days
Total Number of Time Steps is 20, Time Step is 250 days

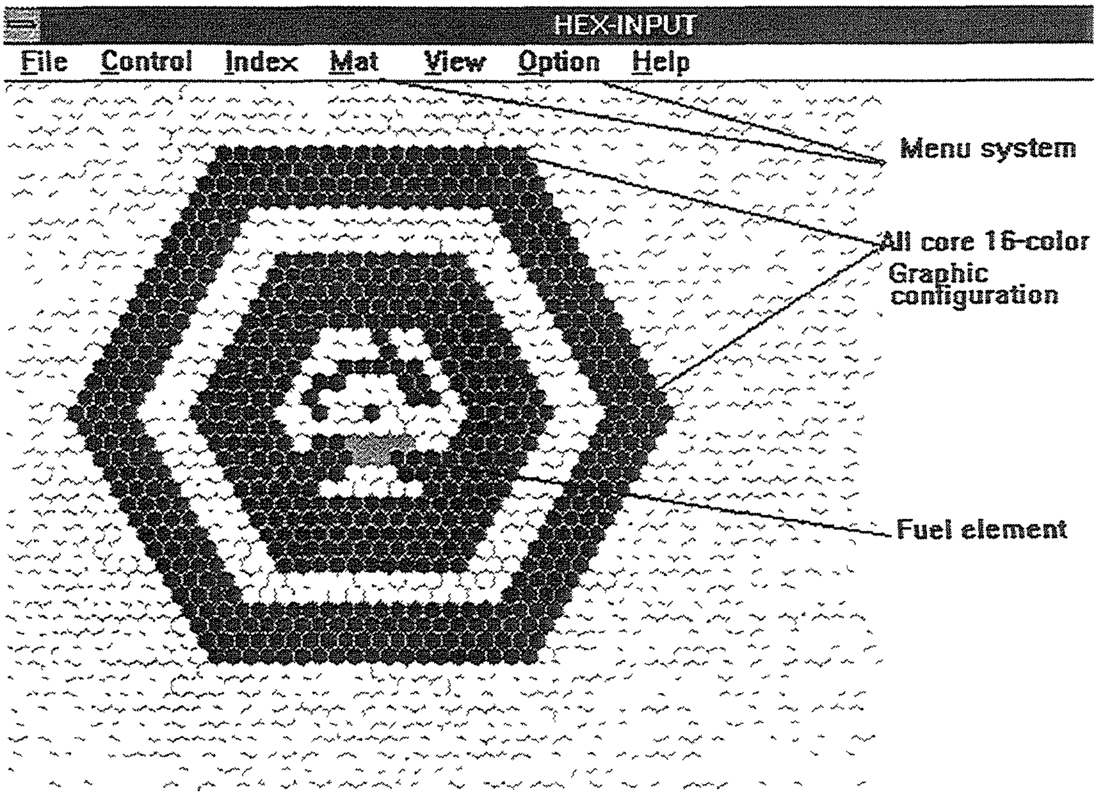
Table 1 Number Densities of Isotopes as a Function of Burnup

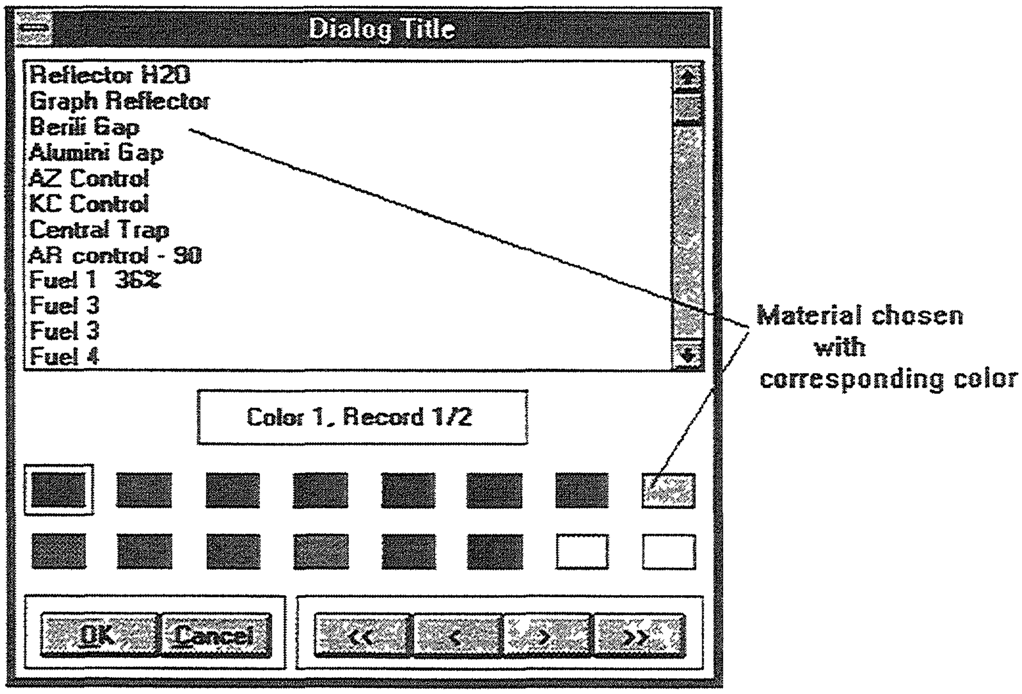
(1 E24 nucl /cm3)

Step	Burnup	U235	U236	U238	Np239	Pu239	Pu240	Pu241	Pu242	FP
0	0 0	1249-2	0 0	2220-2	0 0	0 0	0 0	0 0	0.0	0 0
2	4 95	1187-2	1020-4	2216-2	2856-7	4008-5	4846-7	4008-9	3252-11	5158-4
4	9 65	1129-2	1989-4	2212-2	2850-7	7695-5	1879-6	3054-8	5016-10	1006-3
6	14 11	1073-2	2910-4	2207-2	2845-7	1106-4	4079-6	9760-8	2427-9	1473-3
8	18 36	1020-2	3784-4	2203-2	2840-7	1413-4	6993-6	2187-7	7318-9	1917-3
9	20 41	9941-3	4204-4	2201-2	2837-7	1557-4	8690-6	3028-7	1145-8	2130-3
12	26 24	9213-3	5402-4	2195-2	2829-7	1950-4	1463-5	6599-7	3367-8	2739-3
14	29 88	8758-3	6150-4	2191-2	2824-7	2183-4	1920-5	9909-7	5947-8	3120-3
15	31 64	8538-3	6510-4	2189-2	2821-7	2292-4	2165-5	1185-6	7653-8	3304-3
17	35 02	8116-3	7202-4	2184-2	2815-7	2495-4	2683-5	.1633-6	1204-7	3657-3
19	38 23	7715-3	7860-4	2180-2	2810-7	2681-4	3234-5	.2158-6	1792-7	3992-3
20	39 78	7521-3	8176-4	2178-2	2807-7	2767-4	3520-5	2449-6	2149-7	4153-3

Appendix 2.

SOME SCREENS OF THE CODE HEXIN





Appendix 3.

SOME RESULTS OBTAINED ON TESTING THE CODE HEXBURN (Version FORTRAN)

DALAT RESEARCH REACTOR CONFIGURATION OF 1/2 REFLECTIVE SYMMETRY RUN 9.00PM 12 DEC. 1994
LATTICE PITCH IS 3.5 CM.
POWER PER HIGH UNIT IS 4.16667 Kw
NUMBER OF BURNUP STEPS IS 4
TOTAL BURNUP TIME IN FULL POWER IS 456 DAYS.

MAP FOR ZONE NUMBERS IN RADIAL PLANE :

```
12 12 12 12 12 12 12 12 12
12 12 12 12 12 12 12 12 12 12
12 12 9 9 9 9 9 9 9 9 9 12 12
12 12 9 9 9 9 9 9 9 9 9 12 12
12 12 9 9 9 8 8 8 12 8 9 9 9 12 12
12 12 9 9 8 10 10 15 15 10 10 8 9 9 12
12 12 9 9 7 10 5 5 14 5 5 10 7 9 9 12 12
12 12 9 9 7 10 6 4 1 1 4 6 10 7 9 9 12 12
12 12 9 9 7 10 5 4 11 3 11 4 5 10 7 9 9 12 12
12 12 9 9 8 10 5 4 11 2 2 11 4 5 10 8 9 9 12 12
12 12 9 9 7 12 5 4 11 2 16 2 11 4 5 13 9 9 9 12 12
```

MAP FOR MATERIAL NUMBERS :

```
1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 2 2 2 2 2 2 2 2 2 1 1 1
1 1 1 2 2 2 2 2 2 2 2 2 1 1 1
1 1 2 2 2 3 3 3 1 3 2 2 2 1 1 1 1
1 1 1 2 2 3 10 10 12 12 10 10 3 2 2 1 1
1 1 2 2 12 10 12 12 1 12 12 10 12 2 2 1 1
1 1 2 2 12 10 6 11 10 10 11 6 10 12 2 2 1 1
1 1 2 2 12 10 12 11 10 11 10 11 12 10 12 2 2 1 1
1 1 2 2 3 10 12 11 10 3 3 10 11 12 10 3 2 2 1 11
1 1 2 2 12 1 12 11 10 3 7 3 10 11 12 8 2 2 2 1 1 1
```

POWER DISTRIBUTION NORMALIZED :

```
0 0 0 0 0 0 0 0 0 0 0 0 0
0 0 0 0 0 0 0 0 0 0 0 0 0 0
0 0 0 0 915 971 1127 1124 953 886 0 0 0 0
0 0 856 809 854 1079 0 1061 826 771 803 0 0 0
0 0 0 844 797 0 992 1219 1210 968 0 748 773 0 0 0
0 0 917 880 919 1044 1313 1462 1293 1009 868 802 798 0 0
0 0 0 1026 994 1110 1457 0 0 1424 1060 908 805 0 0 0
0 0 501 0 527 540 666 0 0 0 644 504 424 0 0 0 0 0
```

BURNUP CALCULATION RESULTS :
CORE AVERAGE FISSION CROSS-SECTION IS 1.256734E-02
FUEL AVERAGE MICRO FISSION CROSS-SECTION IS 8.019722E-23
ALL FUEL ELEMENT AVERAGE FLUX IS 2.177106E+13
AVERAGE BURNUP IS 7.899008 %

BURNUP DEPTH (IN %) OF EACH FUEL ELEMENT :

7.17 7.59 8.93 8.91 7.45 6.94
6.87 6.36 6.84 8.57 0.0 8.43 6.63 6.07 6.45
6.77 6.28 0.0 7.97 9.44 9.37 7.78 0.0 5.90 6.22
7.34 6.92 7.36 8.37 10.12 11.51 9.97 8.11 6.96 6.32 6.42
8.02 7.94 8.89 11.17 0.0 0.0 10.93 8.50 7.27 6.34
8.00 0.0 8.40 8.66 10.27 0.0 0.0 0.0 9.94 8.10 6.81 0.0

ZONE BURNUP DISTRIBUTION :

MAT.	ZONE MATERIAL	FLUX	ISTEP	TIME(day)	BURNUP(%)
1	10-FUEL 3	.6607E+13	4	456.0	9.40
2	3-BERILI GAP	.1589E+14	4	456.0	.00
3	11-FUEL 3	.8025E+13	4	456.0	11.51
4	11-FUEL 3	.5565E+13	4	456.0	8.28
5	12-FUEL 4	.5080E+13	4	456.0	7.52
6	6-KC CONTROL	.2293E+13	4	456.0	.00
7	12-FUEL 4	.4651E+13	4	456.0	6.78
8	3-BERILI GAP	.8048E+13	4	456.0	.00
9	2-GRAP REFLECTOR	.6728E+13	4	456.0	.00
10	10-FUEL 3	.4693E+13	4	456.0	6.78
11	10-FUEL 3	.7423E+13	4	456.0	10.46
12	1-REFLECTOR H2O	.1120E+13	4	456.0	.00
13	8-AR CONTROL-90	.1700E+13	4	456.0	.00
14	1-REFLECTOR H2O	.8148E+13	4	456.0	.00
15	12-FUEL 4	.3111E+13	4	456.0	7.63
16	7-CENTRAL TRAP	.1154E+14	4	456.0	.00

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NURESIM LECTURE ON REACTOR PHYSICS (VISUAL AIDS)*

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Abstract

The purpose of the NURESIM software (NUclear REactor SIMulation) is to be used as a computer guide in quick view of the texts and pictures in the fields of nuclear reactor physics This software is designed so that it can be used by users of different knowledge levels Students could find here elementary concepts, researchers - important calculation codes as GRACE, PEACO, THERMOS, HEX120

The NURESIM software is composed of four parts units, pictures, simulations and calculations In the terminology of IAEA-TECDOC-314 (1984) the first three parts may be classified as a level 2 of sophistication IFM code package "Code package useful as a first introduction for nuclear engineers" The last one (calculations) is classified as a level higher Details about each part are explained in Paragraph 2 A users guide is in Paragraph 3

2. DESCRIPTION

2 1 UNITS

The text part of the NURESIM contains the following units

0 Introduction
1 Nuclear Reactions
2 Neutrons
3 Nuclear Reactors
4 Research Reactors and Reactors of Nuclear Power Plants
5 Nuclear Power Plant (NPP)
6 Physical Processes in Actual Reactors
7 Poisoning
8 Temperature Coefficient
9 Reactor Calculations
10 Reactor Kinetics - The Basic Concepts
11 Kinetic Equations
12 Reactivity and Kinetics Parameters
13 Reactor Control
14 Thermohydraulics
15 Nuclear Reactor - Heat Source in NPP
16 Reactor Transient Response (1)
17 Reactor Transient Response (2)
18 Compensated Response Calculations
19 Reactor Transfer Functions and System Stability (1)
20 Reactor Transfer Functions and System Stability (2)
21 Reactor Transfer Functions and System Stability (3)

Materials for this part are drawn from [1-12]

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* Research carried out in association with the IAEA under Research Contract No VIE/5304

2.2. PICTURES

The NURESIM has the illustrated pictures. Each picture is named by pair of numbers: the first one is a number of unit, the second - its order inside unit

U1P1 Scattering and Fission Scheme
U1P2 Energy Spectrum of Secondary neutrons
U1P3 Scheme illustrated real time process and fragment location in nuclear fission
U3P1 Reactor Core and Unit Cell
U3P2 Typical Model of Unit Cell
U4P1 Reactor of the World's First NPP
U4P2 RBMK-1000 Reactor
U4P3 Calder-Hall Reactor
U4P4 Candu Reactor
U4P5 WWER-1000 Reactor
U4P6 ISIS Research Reactor
U4P7 Neutron Source Reactor II (PHRENIC)
U4P8 ADIBKA-1 Reactor
U4P9 TRIGA-MARK-II Reactor (LENA)
U4P10 ANNA Reactor
U4P11 Fast Research Reactor
U5P1 Classification of NPP
U5P2 General View of NPP
U6P1 U-235 burn-up and Pu build-up
U7P1 Reactor Poisoning Upon Shutdown
U9P1 Neutron flux distribution in a reactor with a reflector and with a flattened core
U12P1 Power transient in a start-up incident
U15P1 Reactor Heat Transfer System
U17P1 Response to reactivity variation designed to produce fast, monotone power increase to predetermined constant power level; $l = 10^{-4}$ sec (reactivity $\$A\Sigma a_i e^{-\lambda(t-t_1)}$, t_1 - peak moment)
U17P2 Response to reactivity variation designed to produce fast, monotone power increase to predetermined constant power level; $l = 10^{-4}$ sec (reactivity $\$A\Sigma a_i e^{-\lambda(t-t_2)}$, $[t_1, t_2]$ - peak period)
U17P3 Instantaneous $\alpha(t) \equiv [dn(t)/dt] / n(t)$ as a function of reactivity for U-235 systems with various prompt-neutron life time and for various ramp rates
U17P4 Instantaneous α a function ramp rates (Prompt-neutron lifetime, $l = 10^{-4}$ sec.)
U17P5 Instantaneous α a function ramp rates (Prompt-neutron lifetime, $l = 10^{-6}$ sec.)
U17P6 Instantaneous α a function ramp rates (Prompt-neutron lifetime, $l = 10^{-8}$ sec.)
U17P7 Computed response for reactivity ramp rate $\delta k = 10^{-4}$ sec^{-1} with various superimposed reactivity steps in a system undergoing spontaneous fission at $t = 0$ (Prompt-neutron lifetime, $l = 10^{-5}$ sec.)
U17P8 Computed response for various reactivity ramp rate in a system undergoing spontaneous fission at $t = 0$. (Prompt-neutron lifetime, $l = 10^{-5}$ sec.)
U17P9 Instantaneous α versus time following various positive and negative step changes in reactivity. (Prompt-neutron lifetime, $l = 10^{-4}$ sec.)
U17P10 Nonequilibrium response of $\alpha(t)$ and $n(t)$ to large step changes in reactivity; U-235 system with $\beta = 0.0065$
U18P1 Compensated response to a step reactivity change of \$1.20 ($\delta k_o = .0078$) in U-235 systems characterized by an energy shutdown coefficient
U18P2 Compensated response to a step reactivity change of \$1.00 ($\delta k_o = .0065$) in U-235 systems characterized by an energy shutdown coefficient
U18P3 Calculated transient response for various reactivity steps and heat loss constants (R) in U-235 systems; $\delta k(t) = \delta k_o - B_\varepsilon \int_o^t E(t')e^{-R(t-t')}dt'$
U18P4 Calculated transient response for various reactivity steps and heat loss constants (R) in U-235 systems; $\delta k(t) = at - B_\varepsilon \int_o^t E(t')e^{-R(t-t')}dt'$
U19P1 Reactor-transfer-function amplitude response for U-235 and U-238 systems.
U19P2 Reactor-transfer-function phase shift response for U-235 and U-238 systems
U19P3 Reactor-transfer-function amplitude response for Pu-239 and Pu-240 systems
U19P4 Reactor-transfer-function phase shift response for Pu-239 and Pu-240 systems.
U19P5 Reactor-transfer-function amplitude response for U-233 and Th-232 systems.
U19P6 Reactor-transfer-function phase shift response for U-233 and Th-232 systems.

U19P7 Comparison of computed transfer functions and experimental data for representative intermediate, and thermal U-235 systems
U19P8 Comparison of computed transfer functions with experimental transfer function data, a Pu-239 metal critical assembly.
U19P9 Behavior of the reactor transfer function amplitude and phase for various degrees of subcriticality in U-235 systems.
U20P1 Reactor-transfer-function (amplitude) at full and zero powers
U20P2 Reactor-transfer-function (phase shift) at full and zero powers.

2.3. SIMULATIONS

The NURESIM's user could have a chance to act as a reactor operator using computer keyboard and you observe some reactor events simulated on computer display.

1. XENON

Some fission products have the first resonance near the thermal region or directly in it, hence, their thermal neutron absorption cross-section are very high. Of the greatest importance among them is Xe-135 having resonance at the energy of 0.084 eV and cross-section 3.15×10^6 barns averaged for Maxwell spectrum at normal temperature. An unusually large Xe-135 cross-section even when the concentration is negligible which is typical of short lived fission products gives rise to essential neutron absorption. The fact that Xe-135 is formed through intermediate fission products I-135 results in the instability of the reactor operation and great negative reactivity after high flux thermal neutron reactor shutdown

The Xe-135 effects on the neutron balance is called radioactive poisoning since radioactive Xe-135 vanishes, in the long run, after the chain reaction stops

With the burnout lacking, one would estimate Xe-135 saturation concentration by radioactive I-135 equilibrium. This equilibrium does not hold in the neutron flux, where Xe-135 has the maximum permissible concentration and I-135 build-up is unlimited. When the reactor is shut down and the neutron flux is equal to zero, the equilibrium between I-135 and Xe-135 atoms is reestablished, I-135 concentration reduces at once, Xe-135 concentration increases for some time The maximum concentration is reached within the time interval t_{max} = 11.3 h.

One of the optimal shut-down problems is formulated as the following: At the highest neutron flux (highest power) how to shut the reactor for the least shut-down time and so that the reactor poisoning always is lower than the certain permissive level.

The XENON demonstrates the xenon pit and optimal shut-down control.

2. ROS - Reactor Operation Simulation

The ROS is intended to simulate some of the most important and simple cases of reactor kinetics qualitatively. It is necessary to solve inhour equation to get reactor period Then you can choose control rod disposition The aim here is demonstration of some features concerned with kinetics aspect, so that the reactor is always supposed to be at critical state and other parameters such as the core height, number of fuel elements, core radius and so on may be neglected. Suppose that one rod is placed at the bottom and the other is withdrawn all the way where one of these rods is to be calibrated Total worth of the rod depends on its position, radius (r) and material (A)

Using arrow keys (of keyboard), you can withdraw the first rod for calibration. If the reactivity insertion speed is large, i.e you press control key while power level does not reach up its return level or it is nearly the maximum allowable level, then the reactor must be shut off automatically, otherwise, there could be an incident

3. Heat Flow

This NURESIM's function lets you see heat flow in light-water reactor

4 TRESDEN - Transient RESponse of neutron DENsity

The TRESDEN demonstrates the kinetics equation solutions described by G R Keepin for uncompensated response calculations

The general kinetics equations with common notations have the form

$$\frac{dN(t)}{dt} = \frac{\rho - \gamma\beta}{\Lambda} n + \Sigma \lambda_i C_{ieff} + S$$

$$\frac{dC_{ieff}(t)}{dt} = \frac{\gamma\beta}{\Lambda} n - \lambda_i C_{ieff}$$

where $C_{ieff} = \gamma_i C_i(t)$, $n(t)$ is the fundamental-mode neutron density at time t, and S is an external neutron source

Using the Laplace transform leads to the desired integral solution for n(t) in terms of Λ

$$n(t) = n(0) + \Sigma_j (Bj/\Lambda) \int_o^t e^{S_j(t-t')} \rho(t') n(t') dt' + \Omega_o(t)$$

with $\Omega_o(t)$ defined as

$$\Omega_o(t) = \int_o^t [\gamma_s S(t') + \Sigma_i (\lambda_i C_i(0) - \beta_i n(0)/\Lambda) \gamma_i e^{\lambda_i t}] \Sigma_j Bj e^{S_j(t-t')} dt'$$

and S_j, B_i are tabulated values

The TRESDEN gives the numerical solution of the equations without external source S

5 AUTOSYS - AUTO control SYStem

The AUTOSYS simulates the flowchart of automatic control system You can observe the following example stages

- The reactor power is monitored by means of four ex-core ionization chambers and eight in-core neutron detectors,
- The signals from in-core detectors and power indicators are transferred to compactors,
- The signals are amplified by amplifiers and transferred to summators, then
- The summators give signals to withdraw or insert the control rods for normal reactor operation

6 FEEDBACK

The FEEDBACK gives compensated response to ramp function reactivity in U-235 system taken from G R Keepin

Reactivity ramp has a form

$$\delta r(t) = At - B \int_o^t N(t) dt$$

You can choose tabulated values A and B to learn about interesting model

7 FUEL

The FUEL lets you observe two fuel procedures Fuel pin fabrication and Fuel reloading process

2 4 CALCULATIONS

Realizing an idea of IDE (Integrated Development Environment), the NURESIM contains the calculation codes GRACE, PEACO, THERMOS and HEX120 You can use them for the serious reactor calculations The NURESIM includes the EDIT from DOS Ver 6 2 to edit the input files

GRACE

The GRACE solves the fast neutron spectrum equation in slab geometry with the critical thickness (k=1) So the real cell geometry is not reflected in the code The cell characters are reflected here only in material composition and through the thermal and resonance data, which are submitted to the input by THERMOS and PEACO codes

The GRACE needs the input files GRACE01 SL, GRACE11 SL, GRACE99 SL and gives the output files GRACE02 TAM, GRACE08 TAM, GRACE10 TAM, GRACE KQ

See [13] for deeper understanding the GRACE theory, or [14] for users guide

PEACO

The PEACO solves the neutron slowing-down and calculates the various averaged values such as effective resonance cross-sections You can consider a heterogeneous system which may be homogeneous in special case and may generally be infinite lattice of square or hexagonal cells, cylindricalized lattice or clustered type fuel element.

The PEACO needs the input files PEO1.DAT, PEO8 DAT, PEO10.DAT and gives the output files PEO6.KQ, PEO20 KQ, PEO76 KQ Using the file PEO20 KQ after some simple calculations by hand, you can obtain the constants having the format according to the input of the GRACE

See [15] for deeper understanding the PEACO theory, or [16] for users guide

THERMOS

The THERMOS solves the integral transport equation for the thermal energy range in a cylindrical symmetric medium. White, perfectly reflective and black boundary conditions can be applied For calculating the scattering kernels of hydrogen and deuterium bound in water, the following models can be used The Brown-St. John model, the Nelkin model and the Koppel-Young model The scattering model for oxygen and graphite is the free-gas model A special treatment for hydrogen bound in polyethylene is built in the code The scattering is assumed to be isotropic in the laboratory system, but also an optional transport correction can be applied The source calculated by the code is assumed to be of the form 1/E spectrum or it can be specified by users. The transport kernel is calculated on the basis of a first flight probability technique The set of algebraic equations is solved by an iteration method in a multigroup-multishell approach

It is recommended to give the name of a data file JOB TM and the name of a result file THERMO OUT

See [17] for deeper understanding the THERMOS theory, or [18] for users guide

HEX120

In HEXAGA the group equations are approximated for a uniform 60-degree triangular mesh using a seven-point difference formula at the points of intersection of triangular mesh lines The linear system of finite difference equations is solved by means of the AGA two-sweep iterative method The HEX120 is a version of HEXAGA for 120 symmetric geometry

See [19] for more details on the AGA method, or [20] for users guide

3. USERS GUIDE

3 1 HARDWARE AND SOFTWARE REQUIREMENTS

To install NURESIM, the following hardware and software are required

- An IBM or compatible personal computer
- An EGA or VGA colour monitor
- DOS 3.1 or higher
- Co-processor (It is necessary for reactor calculation functions)
- Free Conventional MEM of 620 K or more (it is recommended to use CONFIG SYS with files = 40, device = himem sys, dos = high)

3.2. INSTALLATION

1 Insert the NURESIM (3 1/2") diskette in into the appropriate disk drive

2 At the "C \>" prompt, change the disk drive to "A " (or "B " depending on your system set up)

3 At the prompt, type "install" (without the quotes) and press <Enter> or <Return> key The installation will automatically begin.

3 3. FILES

The NURESIM has 4 file groups

1 Executable files
NURESIM EXE is the main program
GRACE EXE, PEACO EXE, THERMO EXE and HEX120.EXE are reactor calculation codes
EDIT COM is an editor taken from DOS Ver 6 2

2 Text files
There are 23 text files named "contents" and "unit" with extensions numbered from 0 up to 21 They have an ASCII format You can edit these files if it is necessary (See section 2.1 above)

3 Picture files
There are 48 picture files which have name with extension "pic" (See section 2 2 above)

4 BGI subdirectory
You need BGI subdirectory with a file named EGAVGA BGI to go NURESIM that is written in the PASCAL

3 4 HOW TO USE NURESIM

After NURESIM.EXE started, Menu Bar appears at the top and Status Line at the bottom of the screen as the following

File	Unit	Physics	Kinetics	Dynamics	Others

F1 Help	F4 Open	Alt-F4 Close	Alt-X Exit

The options File has the suboptions

Help	F1
Open	F4
eXit	Alt-X

Unit

New	F3
Open	F4
Close	Alt-F4
picView	F5
Next	F6

Physics

caLc	Alt-L	⟶	Fast	F7
Xenon	F12		Resonance	F8
			Thermal	F9
			Global	F10
			Edit	F2

Kinetics

Ros	Alt-R

Dynamics

Heat flow	Alt-H
Transient response	Alt-T

Others

Autosys	Alt-A
feedBack	Alt-B
fuEl	Alt-E

Therefore, press

F1 For help

Alt-X Exit to DOS

F3 Open Contents for direct selection of the unit.
Press F3, move the cursor to a title of necessary unit and press <Enter> to choose
Press <Esc> to deactivate cursor

F4 Open the units by turns
The opened units compose a cycle.

Alt-F4 To close unit (delete unnecessary unit from the cycle)
Remember to close unnecessary units to economize the computer memory

F5 Activates cursor for direct selection of the picture to display
When there is a note "See U* Pic" in the opened unit you could move cursor to the character U and press <Enter> to display the corresponding picture
Press any key to return unit. Press <Esc> to deactivate cursor

F6 Display opened units by turns

Alt-L Reactor Physics Calculation (using GRACE, PEACO, THERMOS, HEX120)

F2 Running DOS 6 2 Editor

F12 Simulates xenon poisoning problem

Alt-R Running ROS program (Reactor Operation Simulation)

ROS is a complete part of NURESIM in which one can practice as a reactor operator and observe on computer screen the positions of the control rods (longitudinal and transverse cross-section) and power display

Alt-H To demonstrate a heat flow circuit in Light-water-cooled reactor and observe a heat removal cycle in a primary circuit, heat and temperature distributions in reactor core

Alt-A To study the flow chart of reactor auto-control system

Alt-T Running TRESDEN to get a neutron density for a transient response

Alt-B To see a self-limiting power excursion

Alt-E To observe the fuel process Fuel Pin Fabrication and Fuel Replacement Operation

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