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Research reactor instrumentation and control technology

Report of a Technical Committee meeting held in Ljubljana, 4–8 December 1995

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FOREWORD

The majority of research reactors operating today were put into operation 20 years ago, and some of them underwent modifications, upgrading and refurbishing since their construction to meet the requirements for higher neutron fluxes. However, a few of these ageing research reactors are still operating with their original instrumentation and control systems (I&C) which are important for reactor safety to guard against abnormal occurrences and reactor control involving startup, shutdown and power regulation. Worn and obsolete I&C systems cause operational problems as well as difficulties in obtaining replacement parts. In addition, satisfying the stringent safety conditions laid out by the nuclear regulatory bodies requires the modernization of research reactors I&C systems and integration of additional instrumentation units to the reactor. Technical advances in the I&C systems have been rapid in the past years, and this technology should be adapted by the research reactor community. The demands of high level sophistication and reliability to meet various operational and safety requirements are being met by the increased use of microprocessors and personal computers. This requires careful consideration of the research reactor operators in planning how to improve the instrumentation and control for ageing research reactors and in making appropriate selections if involved in building new facilities. In order to clarify these issues and to provide some guidance to reactor operators on state-of-art technology and future trends for the I&C systems for research reactors, a Technical Committee Meeting on Technology and Trends for Research Reactor Instrumentation and Controls was held in Ljubljana, Slovenia, from 4 to 8 December 1995.

This publication summarizes the discussions and recommendations resulting from that meeting. This is expected to benefit the research reactor operators planning I&C improvements.

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1. INTRODUCTION

There are currently 274 operational research reactors in the world and 77% of these reactors were constructed more than 20 years ago. Since their construction, the instrumentation and control (I&C) systems of some reactors have undergone modifications, upgrading and refurbishing, but a number of ageing reactors are still operating with some parts and components of the original system. Satisfying the stringent safety conditions laid out from time to time by the nuclear regulatory bodies requires the modernization of research reactors I&C systems and integration of additional instrumentation units to the reactor. One common example is the need for replacement of old instrumentation system comprising meters and recorders, with the distributed data acquisition and processing network for effective monitoring and control of various process parameters. However, there are no international standards or guides specifically for the design and development features of the I&C systems of research reactors. Most standards are for power reactors, and in the absence of standards for the I&C systems for research reactors, there is a tendency of the regulatory bodies to impose power reactor standards on research reactors. This is not completely appropriate and leads to difficulties. In order to avoid this problem, at least in partial manner, the IAEA is preparing the publication "Safety Practice on Safety Instrumentation of Research Reactors" as a result of the Advisory Group Meetings held in Vienna in 1988, 1995 and 1996. The main purpose of this Safety Practice is to develop in some detail the design requirements of the I&C systems important to safety for research reactors; it also recommends steps to facilitate the design, fabrication and installation of I&C items or systems. This publication will be part of the set of documents developed within the framework of the IAEA Research Reactor Safety Programme that covers all the important areas of research reactor safety such as design (Safety Series No. 35-S1) and operation (Safety Series No. 35-S2).

In addition, many issues must be considered in planning on how to improve the instrumentation and control for ageing research reactors and what the technology and trends in the I&C systems are. In order to clarify these issues and to provide guidance to reactor operators on state-of-art technology and future trends for the I&C systems for research reactors, a Technical Committee Meeting (TCM) on "Technology and Trends for Research Reactor Instrumentation and Controls" was held in Ljubljana, Slovenia, from 4-8 December 1995. The purposes of the meeting were to discuss:

- various I&C systems
- nuclear instrumentation
- non-nuclear (process) instrumentation
- area monitors
- reactor power regulating system
- modifications of I&C system
- state-of-art technology
- future trends

These topics are both very important and relevant in the present day context. The technology of the I&C systems for research reactors has, namely, undergone evolution for decades from vacuum tubes, transistors, microprocessors and PC based units. The demands of high level sophistication and reliability to meet various operational and safety requirements are being met by the increased use of microprocessors and PCs with self-test features and on-line diagnostic facilities to monitor any degradation in performance of the I&C systems. This requires due attention to be put on software quality, testability and its maintainability. Therefore, an adequate quality assurance programme for software and hardware has to be implemented. Due to the lack of real performance data, development changes from hardware to software must be made either partially or with a hardware backup when a full software allocation is made. In the near future, when enough confidence is gained based on performance data from real reactor protection system already operating in some nuclear power plants and research reactors, then the use of a hardware backup will not be required.

This document summarizes the discussions and recommendations resulting from that meeting. It is expected to benefit the research reactor operators planning I&C improvements for ageing reactors and in making appropriate selections if involved in building new facilities.

2. REVIEW OF I&C SYSTEMS

2.1 General Remarks

2.1.1 Basic Concept

The Instrumentation and Control (I&C) of a nuclear research reactor (NRR) is the shell between the operator and the plant.

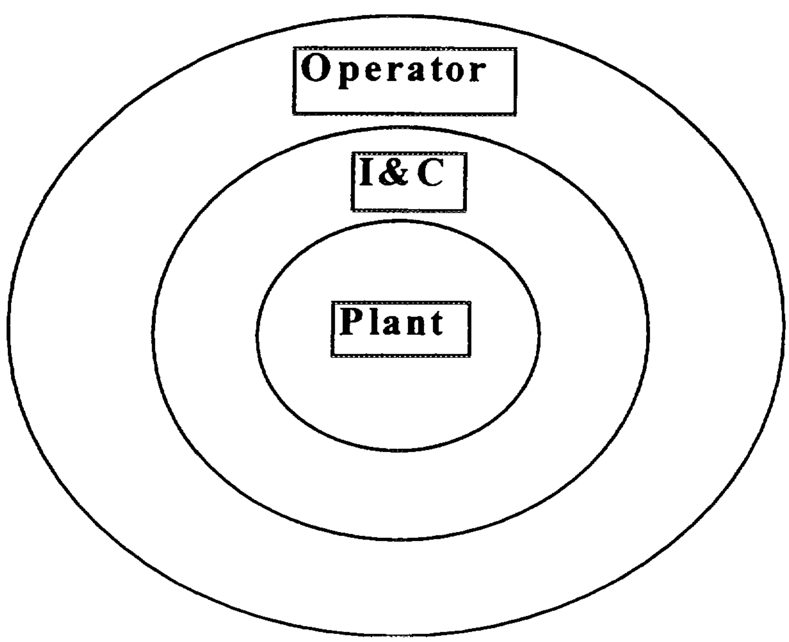

FIG. 2.1: I&C shell between the operation staff and the plant

The main functions of the I&C shell are to provide to the operation staff information about the plant, to run open-loop and close-loop control systems and to process commands from the operation staff. These functions enable the operation staff of a research reactor to perform the following two activities:

- To operate the plant safely and efficiently in all its operational states,
- To take measures to maintain the plant in a safe state or to bring it back into such a state after the onset of either accident conditions or design basis events.
- To supervise specific tasks

The components of this shell are:

- Man-Machine Interface
- Control Consoles, Supervision Desks and Supervision Centres
- Open-loop and close-loop control systems
- Instrumentation racks
- Conditioning units
- Sensors and Actuators

This shell may be considered as comprised by two independent systems. the reactor protection system and the supervision and control system. All the instrumentation used for this shell can be classified in three groups nuclear instrumentation, non-nuclear instrumentation, and radioprotection instrumentation

2.1.2 Functional Groups

The technological process of the plant is divided into systems, most of these systems have to be provided with I&C equipment.

As a result of history and experience or an appropriate functional analysis, these I&C systems are co-ordinated into functional groups The borders of these functional groups are however not always coinciding with the borders of the respective system

The principle of functional groups offers the following advantages

- Design, manufacture, erection and commissioning of functional groups can be carried out in steps and largely independent of each other. It is possible to co-ordinate the individual steps to optimise scheduling and technological details.

- The degree of automation can be selected individually for the functional group, according to the corresponding technological requirements and operational demands

- Subdividing the I&C equipment according to several automated partial processes results in a high availability of the total system According to this structure, there exist no central functions, therefore, faults and disturbances stay to small sectional areas

2.1.3 Applicable Standards

Standards from the nuclear regulatory authority of each country should have mandatory status over all design and development features of the I&C systems of a research reactor

The following international design standards might be used as guidelines for aspects not covered under national standards

- ANSI-ANS-15-15-1978 American National Standard criteria for the reactor safety systems of research reactors
- EPRI-NP-3701-1984 Computer-Generated Display System Guidelines Volume 1 Display Design Volume 2 Developing and Evaluation Plan
- IAEA Safety Series 35-S1 Code for the Safety of Nuclear Research Reactors - Design (1992)
- IAEA Safety Series No. 50-SG-D3: Protection Systems and Related Features in Nuclear Power Plants, A Safety Guide (1980).
- IAEA Safety Series No. 50-SG-D8: Safety Related Instrumentation and Control Systems for Nuclear Power Plants, A Safety Guide (1984).
- IEC-639 Use of Protection System for non Safety Purposes.
- IEC-643 Applications of Digital Computers to Nuclear Reactors Instrumentation & Control.
- IEC-671 Periodic Tests and Monitoring of the Protection System of Nuclear Reactors.
- IEC-674 Safety Logic Assemblies of NPP: Characteristics and Test Methods.
- IEC-880 Software for Computers in the Safety Systems of Nuclear Power Stations (1986).
- IEC-964 Design for Control Rooms of Nuclear Power Plants.
- IEC-965 Supplementary Control Points for Reactor Shutdown without access to the Main Control Room.
- IEC-967 Programmed Digital Computers Important to Safety for NPP.
- IEEE-7-4.3.2-1993 Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations.
- NUREG-0700 Guidelines for Control Room Design Reviews.

- KTA-2201 Seismic Regulations.
- KTA-3501 Reactor Protection System and Monitoring of Engineered Safeguards.

2.1.4 Requirements and Criteria

2.1.4.1 Project and Supply Criteria

2.1.4.1.1 Use of proven technology

All systems and components shall be of proven design with at least one or more identical or similar applications.

2.1.4.1.2 Custom component development

All custom components shall be designed based on previous similar applications and proven technologies. Their design shall be simple and follow a structured development process, including requirement specifications, design specifications, prototypes, simulations, reviews and testing cases.

2.1.4.1.3 Uniformity of equipment

In order to reduce operative and maintenance costs, most components of the I&C systems should be manufactured by the same supplier following proven standards. Some special components and instrumentation may be provided from leading manufacturers with similar previous applications.

2.1.4.1.4 Design documentation

Design specification and design support calculations shall be documented to demonstrate compliance with the requirements at system level. These reports are part of the engineering documentation of the system.

2.1.4.1.5 Verification and validation

Verification and validation of all I&C systems and components shall be carried out at component and system level, via applicable testing procedures, at the system manufacturer location before shipment.

2.1.4.1.6 Units

All system interfaces, displays, panels, instrument readings, recordings and reports should use metric system units (SI units).

2.1.4.2 System General Functional Requirements

2.1.4.2.1 Automatic safety protective actions

Safety protective actions shall be automatic, i.e. they shall be triggered without operator intervention. The operator may initiate manual protective actions; however, he cannot interrupt or interfere with any automatic action.

2.1.4.2.2 Local/remote transfer

Motors, actuators and some other equipment can be controlled from either remote or local control panels. The remote/local control mode is selected by the operator. Local mode is the default for all system conditions.

2.1.4.2.3 Manual/auto transfer

Transfer from manual to automatic mode of operation and vice-versa shall be smooth, without introducing perturbations to the controlled process.

2.1.4.3 System General Reliability Requirements

2.1.4.3.1 Reliability criteria

High reliability and low frequency of maintenance shall be mandatory for all systems. This is the result of adequate system design, introducing redundancy, diversity and physical isolation, plus the use of highly reliable components for each functional unit. For safety systems the "fail-safe" criteria shall also be applied.

2.1.4.3.2 On-line and periodical testing and diagnosis

On-line testing and specific operation/maintenance modes with supervision, testing and diagnosis functions shall be considered for all I&C systems.

2.1.4.3.3 Availability criteria

The current status of engineering allows availability figures greater than 99.95% for all I&C systems. The availability of all and each I&C system shall be evaluated and submitted to the operation staff of the research reactor.

2.1.4.3.4 Independence/Isolation

Any failure in one I&C system shall be isolated to the system itself. This failure should not compromise the behaviour or performance of any other I&C system.

2.1.4.4 System General Maintenance Requirements

2.1.4.4.1 Ease of maintenance

All I&C systems shall be designed and implemented so as to require minimum maintenance. Maintenance activities are simplified by providing self-diagnosis functions and by simplifying access, connections and testing procedures.

2.1.4.4.2 Maintenance by replacement

In order to minimize outage times, all systems should be designed favouring system maintenance by replacement of modules/instruments, instead of by repair.

2.1.4.4.3 On-line repairs

In general, all systems should have electronic modules which can be replaced with the system powered-up and on-line.

2.1.5 I&C Event List

A typical list of events and I&C system actions is presented in the table below. Actions are represented by letters with the following meanings:

- A: Alarm
- I: Interlock
- S: Safety action to trigger a safety actuation system
- E: Evacuation
- V: Emergency Ventilation
- f: failure

TABLE I: I&C EVENT LIST. TRIPS, INTERLOCKS AND ALARMS

SYSTEM	SIGNAL DESCRIPTION	EVENT	
		CONDITION	ACTION
	Start-up neutron flux period	low	A
	Power neutron flux	low	A/I
	Power neutron flux	high	A/S
	Log power neutron flux	low	A/I
	Log power neutron flux	high	A/S
Nuclear	Power neutron flux period	low	A/S
Instrumentation	RPRS wide range linear flux	high	A
	RPRS wide range linear flux	low	A
	RPRS wide range log flux	high	A
	RPRS wide range log flux	low	A
	RPRS N-16 gamma flux	high	A
	RPRS N-16 gamma flux	low	A
	Core temp. difference (T_{out}-T_{in})	low	A
	Core temp. difference (T_{out}-T_{in})	high	A/S
	Core inlet temperature (T_{in})	high	A/S
	Core outlet temperature (T_{out})	high	A/S
	Core pressure difference	low	A/S
	Core pressure difference	high	A/S
Primary	Primary pump discharge pressure	low	A
Cooling	Primary flow	low	A/S
System	Pool water level	low	A/S/E
	Pool water level	high	A
	Pool water temperature	high	A
	Primary water conductivity	high	A
	Flapper valve open in forced flow	yes	A/S
	Flapper valve open in natural convection.	no	A/S
	Fuel cladding failure detector	high	A
	Fuel cladding temperature	high	A
Secondary	Pump discharge pressure	low	A
Cooling	Secondary flow	low	A
System	Water gamma activity	high	A

SYSTEM	SIGNAL DESCRIPTION	EVENT	
Cooling	Air flow rate	low	A
Tower	Water pool level	low	A
Hot Water	Flow rate	low	A
Layer	Layer temperature	low	A
System	Layer temperature	high	A
	Open end gamma activity monitor	high	A/S
Radioprotection	Occupational area gamma monitor	high	A
System	Stack particulate activity	high	A/V
	Stack iodine activity	high	A/V
	Stack noble gases activity	high	A/V
Ventilation	Reactor hall depression	low	A/S
System	Reactor hall depression	high	A
	Air pressure switch (ventilation)	low	A/S
	Seismic accelerations	high	A/S
Other	Liquid effluent pool water level	high	A
Systems	Auxiliary pool water level	low	A
	Auxiliary pool water level	high	A
Power	Mains Power Supply	f	A/S
Supply	Uninterrupted power supply	f	A/S
System	Instrumentation Power supply status	f	A/S
	RPS Neutron Detector check result	f	A/S
	RPS VPLU self-check result	f	A/S
	RPS Trip Unit self-checking result	f	A/S
RPS & SCS	SCS Field Units self-check result	f	A/
Availability	SCS Control Units self-check result	f	A/S
	SCS Superv. Units self-check result	f	A
	SCS Control Net Status	f	A/S
	SCS Supervision Net Status	f	A/S
	SCS peripheral device Status	f	A

2.2 Reactor Protection System

2.2.1 Definition

The Reactor Protection System encloses all electrical and mechanical devices and circuitry involved in generating the initiation signals associated with protective functions that are carry out by the Safety Actuation Systems.

2.2.2 Design Criteria

The following criteria is normally taken into account during the design of the Reactor Protection System:

- **Redundancy and diversification** - In order to achieve the requested system reliability and avoid common failures, these issues are basic design criteria.

- **Reliability** - In general, the reliability of the system should be $\leq$ as 10^{-3} failure/demand. This is normally achieved using the traditional two-out-of-three logic on the three redundant channels provided.

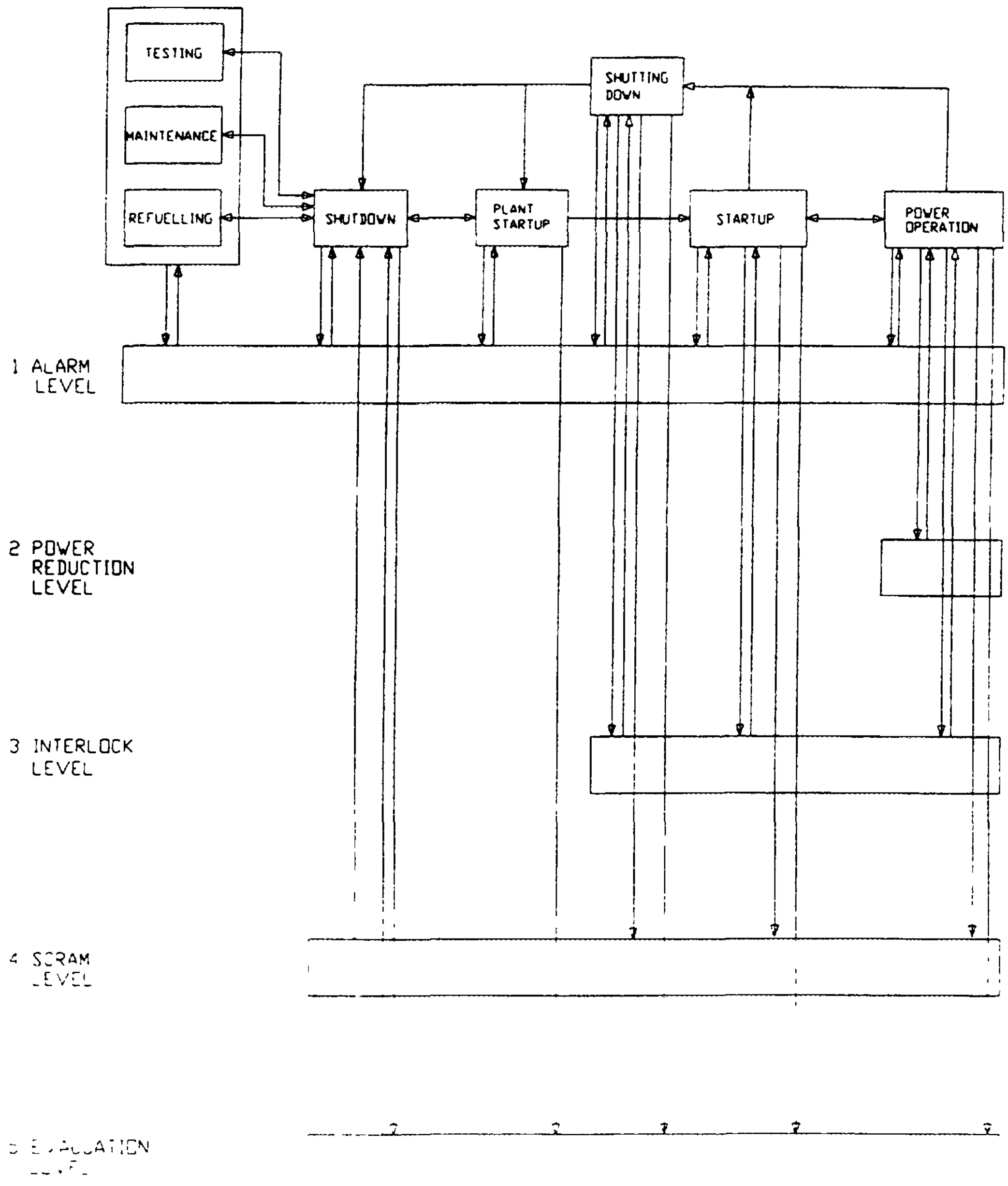

FIG. 2.2: Reactor States and Transition

- **Availability** - In general, a modular design and use of three redundant channels increase the availability of the system
- **Performance** - The response time of the system to any trip event is requested by international standards to be less than 60 ms This time includes the sensing time,

the propagation time through the system, and the actuation time (time of the actuator to switch status).

- **Priority** - Signals from/to safety systems have higher priority of treatment than those signals from/to safety related and non safety systems.

- **Independence** - Physical and electrical isolation of the system enable its fully independence from other systems.

- **Single failure** - Sufficient redundancy and electrical independence shall assure that no single failure results in a loss of any protective function.

- **Fail safe** - The instrumentation of the system shall be designed so that any failure requests the initiation of the corresponding protective action. This criteria can be accomplished using dynamic signals (e.g. pulse train type signals) for normal conditions and static signals for trip conditions. Should an instrument or unit of the system fail, the pulse train disappears requesting trip (fail-safe principle).

- **Common failures** - In general, diversification of paths, physical and functional separation of equipments and channel redundancy are good practices to avoid common failures.

- **Automatic initiation** - All protective actions are automatically triggered by the reactor protection system without operator intervention. The operator may initiate protective actions manually; however, he cannot interrupt or interfere with any automatic action.

- **Manual initiation** - All protective actions are normally triggered automatically but they can also be triggered manually on operator's request. Nevertheless, once a protective action is initiated, manual actions can not prevent or interrupt the normal execution of the required protective action.

- **Periodic inspection, testing and maintenance** - These issues shall be considered in detail. At system-level all tests should be performed on-line i.e. during the operation of the system, with a fallback on 1 out of 2 logic during the test duration. Normally, on-line tests are performed on the trip unit and the nuclear instrumentation modules.

- **Electrical isolation** - All signal paths from and to the instrumentation of the reactor protection system shall have decoupling devices in order to electrically isolate the system. The same is applied to the interconnection of the redundant parts of the system. The protective functions of the system shall not be affected by a failure in an individual channel of the reactor protection system or in any other instrumentation of any safety related or non safety system.

- **Electrical power independence** - Three independent uninterruptible power supplies should be used to satisfy all power requirements of the system, to increase availability of the system, to reduce electrical noises, and to reduce the amount of spurious trips.

2.2.3 Functional Requirements

The objective of the reactor protection system is to trigger the safety actuation systems whenever necessary:

- to protect the integrity of radiation barriers in order to avoid the release of fission products inside or outside the research reactor, and

- to avoid the exposition of reactor personnel to very high rates of radiation due to reactor operation

To achieve these functions, appropriate Safety System Settings (SSS) shall be defined to limit safety variables. Typical safety variables are shown in Table II. Safety variables, which are monitored by the reactor protection system, are typically grouped in different types of channels as shown below.

- Nuclear channels: Start-up and Power channels.
- Thermohydraulic channels
- Radioprotection channels
- Seismic and Special purpose channels
- Availability status variables

TABLE II: TYPICAL SAFETY VARIABLES IN RESEARCH REACTORS

VARIABLE	SSS
Startup Channel Logarithmic Flux	Upper Limit
Power Channel Linear Flux	Upper Limit
Power Channel Logarithmic Flux	Upper Limit
Power Channel Flux Rate	Upper Limit
Core Cooling Flow	Lower Limit
Core Outlet Temperature	Upper Limit
Core Temperature Difference	Upper Limit
Core Pressure Difference	Upper/Lower Limit
Reactor Pool Level	Lower Limit
Flap Valve Status	Open/Close Cooling Mode Dependence
Reactor Pool Dose Rate	Upper Limit
Access Doors to areas with Very High Dose Rates	Unlock
Seismic Acceleration	Upper Limit
Safety Actuation Systems	Unavailable
Main Supply System	Unavailable
Supervision & Control System	Unavailable

Typically the most important barrier to fission product release (in many research reactors the single one) is the fuel cladding. Therefore, values in the previous table, which protect the integrity of this barrier, are defined so that the temperature of the fuel cladding is below a certain upper limit In order to control a temperature the heat source or the cooling conditions shall be monitored

The design of the reactor will define the particular number of safety actuation systems The most safety demanding reactor has the following safety actuation systems

- First Shutdown System
- Second Shutdown System
- Emergency Core Cooling System
- Evacuation Alarm System

The block diagram in Fig.2 3 explains the main functionality required to the Reactor Protection System

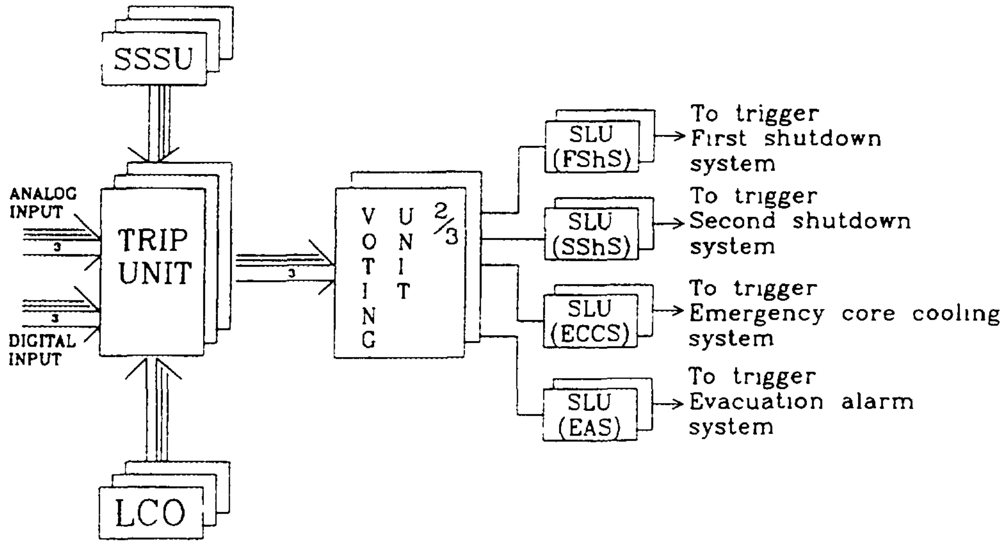

FIG. 2.3: Reactor Protection System Functional Block Diagram.

The reactor protection system is required to fulfill the following main functions

- Trip generation
- Interlock generation
- Operator interface
- Self-check
- Data communication to SCS

2.2.3.1 Trip Generation

Safety variables are electrically conditioned first, by the conditioning units Trip signals are produced by the trip units comparing the conditioned safety variables with appropriate safety system settings These trip signals are voted by the voting units and then processed by the protective logic units to generate the signals which are used to trigger the safety actuation systems

This module monitors analog conditioned safety signals for their rationality, validity, and conformity to defined operational ranges and upon comparing with safety system settings provides binary outputs.

Each binary output is either in a TRUE state or a FALSE state. A TRUE state is produced when the analog conditioned safety signal is outside its allowable operative window i.e. when the analog conditioned safety signal is greater equal than the upper limit of this window or when it is less equal than the lower limit of this window; otherwise a FALSE state is produced. The lower and upper limits of each allowable operative window are defined by one safety system setting and one electrical limit. The electrical limit is a lower limit of the window when the appropriate safety system setting represents an upper limit; or the electrical limit is an upper limit of the window when the appropriate safety system setting represents a lower limit.

The valid electrical range of the analog conditioned safety signals is 0.5 to 4.5 V. That is the lower electrical limit is 0.5 V and the upper electrical limit is 4.5 V. The analog conditioned safety signals are read by the trip unit.

The safety system settings are constants stored in local memory. These constants are read once by the trip unit from the safety setting input unit via the STD-bus through an appropriate digital input/output card plugged into the card cage of the STD-bus system. Each constant is defined by a 16-bit word in BCD format. All constants are read one by one through the same 16-bit input port of the digital input/output card. Hence, the constant to be read is selected by a multiplexor circuit, which receives from the trip unit a constant identifier through the five least significant bits of an 8-bit output port of the same digital input/output card used to read the constant value.

This functional section also generates binary outputs from digital conditioned safety signals. Each binary output is either in a TRUE state or a FALSE state. A TRUE state is produced when the digital conditioned safety signal is equal to zero volts; otherwise a FALSE state is produced when the digital conditioned safety signal is equal to five volts.

The electrical values of the digital conditioned safety signals are either 0 or 5 Vcc. They are read by the trip unit via the STD-bus through appropriate digital input cards plugged into the card cage of the STD-bus system. Each digital conditioned safety signal is hard-wired to one pin (one bit) of one of the 8-bit input ports of the digital input card. Therefore, up to 8 digital conditioned safety signals are read per input operation from one of these 8-bit input ports i.e. digital conditioned safety signals are processed by the trip unit as clusters of up to 8 digital signals each.

The binary outputs produced from analog/digital conditioned safety signals are sent to the voting and protective logic units following a fail-safe approach. Normal conditions (FALSE state) on one signal are represented by a dynamic binary output i.e. a pulse train of logical 0s and 1s on its binary output. On the other hand, during abnormal conditions (TRUE state) on one signal, its binary output is held static at a constant value e.g. electrical zero. These binary outputs are sent by the trip unit to the VPLUs via the STD-bus through appropriate digital output cards plugged into the card cage of the STD-bus system. Each binary output is available at one pin (one bit) of one of the 8-bit output ports of the digital output card. Therefore, up to 8 binary outputs are sent per output operation to one of these 8-bit output ports i.e. binary outputs are processed by the trip unit as clusters of up to 8 digital signals each.

2.2.3.2 Interlock Generation

Safety systems require certain conditions to be satisfied to remain available. Protection interlocks are specified and implemented to prevent erroneous changes in these operational conditions or configurations. These safety interlocks are implemented in the protective logic units of the reactor protection system.

There are four groups of Protection Interlocks. They are:

- Interlocks related to the movement of the safety rods
- Interlocks related to the availability of the safety systems
- Operational configuration changes of the neutron measurement channels
- Operational configuration interlocks related to special systems

The safety interlocks must be met during reactor start-up and maintained throughout the entire reactor operation.

The reactor protection system evaluates the interlock conditions in the same way as the trip conditions described above. Therefore, the protective interlocks are generated by the protective logic units.

These protective interlocks enable on-line configuration changes in the protective logic units and are hard-wired into both safety display and command racks and control rod controllers to constrain movement sequences.

2.2.3.3 Operator Interface Functions

The reactor protection system presents all safety information to the operator through dedicated hard-wired safety display and command racks on the main and secondary consoles. This presentation of safety parameters is independent of the supervision and control system status and enables the operator to evaluate the safety state of the reactor. The operator can manually trigger any safety actuation system from the command panels of the safety display and command racks provided.

2.2.3.4 Self-Check Functions

The trip unit performs several self-check functions, including checks on program timing and execution thread, integrity checks on static data in ROM and RAM, admissibility checks on electronic input levels, admissibility checks of input safety system settings, integrity checks on stored safety system settings in RAM and verification checks on input/output hardware boards.

2.2.3.5 Data Communication to SCS

Safety variables, safety settings, trip signals, and self-check results of the reactor protection system are provided to the Supervision and Control System (SCS), through a one way, electrically isolated interface. This enables the efficient use of reactor instruments, avoiding duplication, while improving presentation by the use of SCS visual display units, and enabling enhanced recording/analysis of safety events.

2.2.4 RPS Architecture

The typical configuration of the Reactor Protection System shown as a block diagram in Fig. 2.3 has been normally developed using hardware components. But the more recent ones are being fully or partially implemented using software components with or without a hardware backup. The use of the hardware backup depends on the amount of functionality allocated to software components. The more functions are allocated to software the more the chances to require a hardware backup. Nevertheless, this depends on the amount of experience in software development with an adequate quality programme and use of adequate structured life cycle for real time safety critical system development, an adequate testing cycle with an appropriate verification and validation programme, an adequate quality assessment plan, and the amount of real performance data of the system. Development changes from hardware to software are necessary but they should be implemented with care. Due to the lack of real performance data, changes must be made either partially or with a hardware backup when a fully software allocation is made. In the coming future when enough confidence is gained based on performance data from real reactor protection systems already operating in some nuclear power plants and research reactors, then the use of a hardware backup will not be required.

Before the final establishment of fully digital RPS, there will be a transition phase, which has already started, of several years, where diverse conventional RPS will be required as a backup system for legal reasons and for gathering sufficient experience and reliability data on the new systems.

The other possibility is to go in steps selecting only part of the functionality required to be implemented in software i.e. a partially digital RPS. For example, the trip unit may be selected for software implementation due to its simplicity. Although module simplicity, this software must be developed with an appropriate life-cycle using a structured methodology for real-time safety-critical system development, an adequate test life-cycle with an intensive verification and validation plan (see Section 5.3.2).

Figure 2.4 puts the reactor protection system into perspective within the I&C system. Figure 2.5 shows a typical hardware architecture of the reactor protection system.

Typically, the following units comprises the reactor protection system:

- Three redundant conditioning units
- Three redundant safety setting input units
- Three redundant trip units
- Two redundant voting units

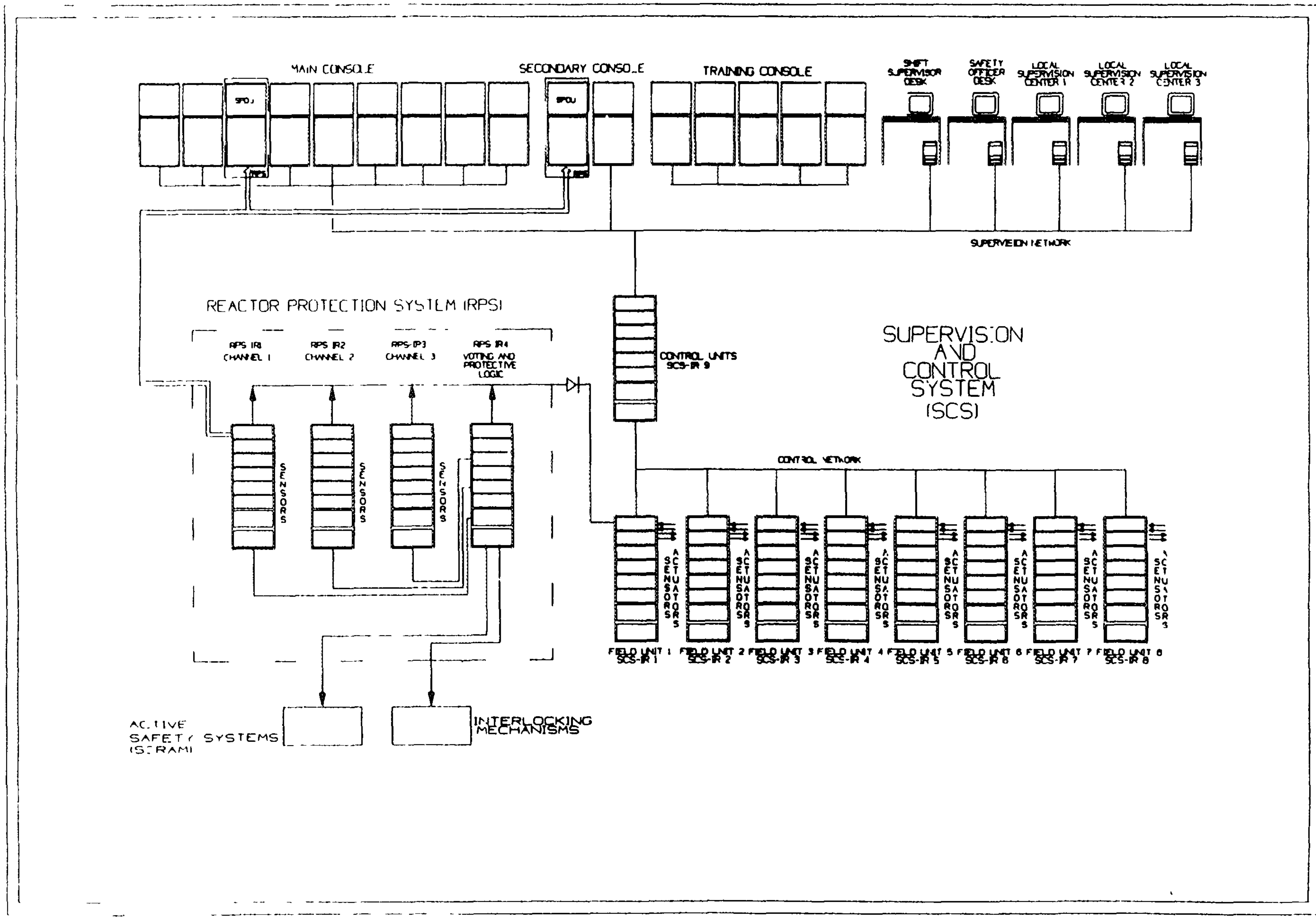

FIG. 2.4: I&C Systems Architecture

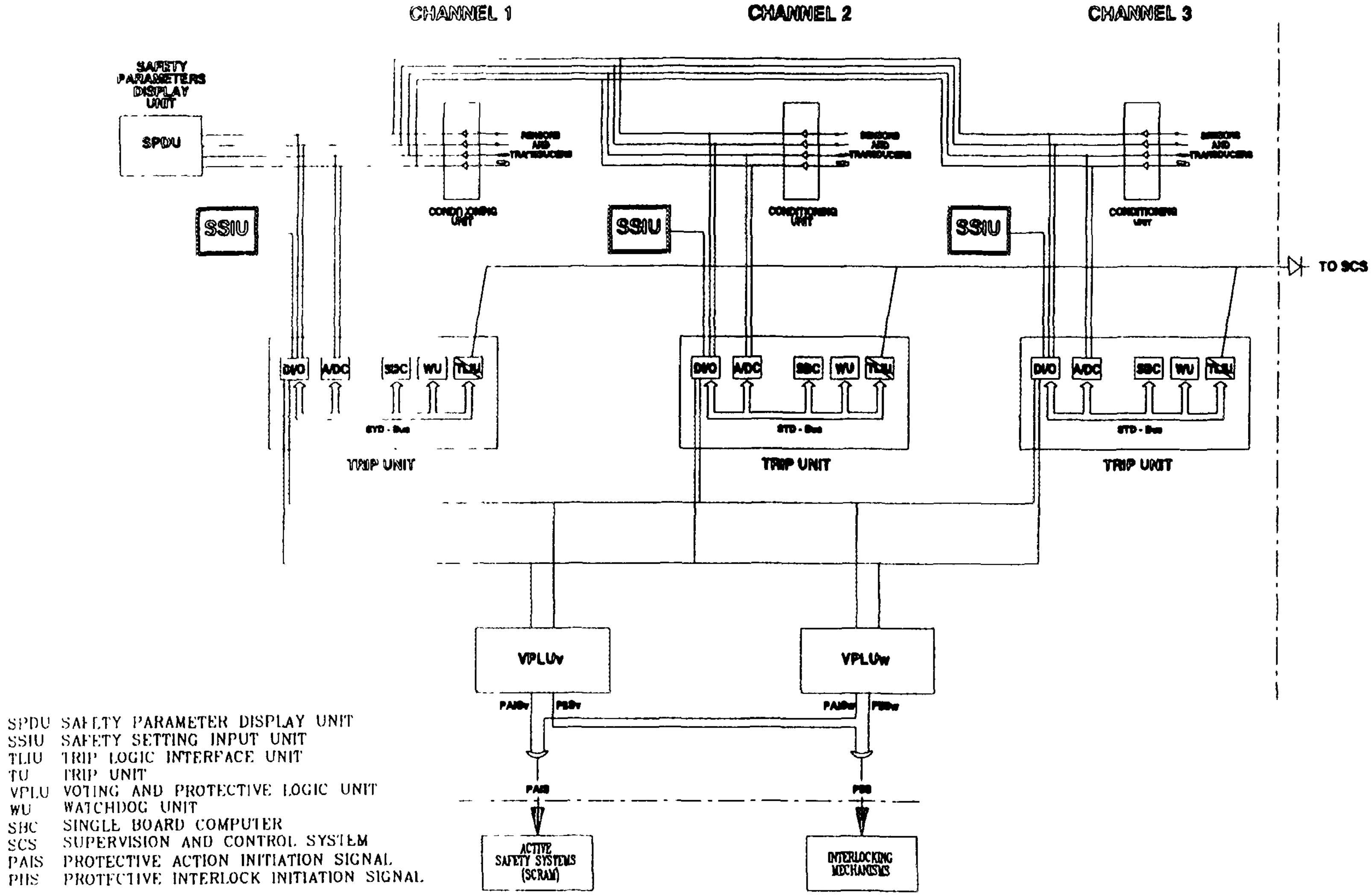

FIG.2.5: Reactor Protection System

- Two redundant protective logic units
- Three redundant interfaces to the supervision and control system, and
- Three watchdog units.

2.2.4.1 Conditioning Units (CDU)

The ConDitioning Units (CDU) have sensor-specific boards and modules to perform signal amplification, electric level conversion, signal checking and processing in order to be able to output within prescribed electronic ranges analog or digital signals to the trip units. The types of sensors interfaced by these conditioning units cover nuclear instruments, radioprotection monitors, switches and all process and ventilation instrumentation.

There are three redundant conditioning units, each one works with only one of the three redundant trip units. Hence, there is a direct one-to-one correlation between each conditioning unit and each trip unit.

2.2.4.2 Safety Setting Input Unit (SSIU)

The Safety Setting Input Units (SSIUs) are three redundant devices, one for each trip unit. Each SSIU has a front panel, which provides as many sets of thumbwheels as Safety System Settings (SSS) are required by its related trip unit. Appropriate values for these SSS are defined by an authorised operator through these thumbwheels and they are read into RAM only once every time the trip unit is power-on. Each SSIU has a panel-door, which is mechanically locked in its closed position, to allow no changes of its SSS while its related Trip Unit is operating. In case this mechanical protection is overridden, the unlock SSIU shall power-off its related trip unit.

The trip unit includes a validation module to check once after power-on the admissibility of the safety system settings read from the safety setting input unit and a verification module to check periodically the integrity and consistency of the safety system settings in RAM.

2.2.4.3 Trip Unit (TU)

Each trip unit is built with the following industrial high-quality board- and systems-level computer products.

- **STD-Bus Single Board Computer.** One intended for embedded applications where performance, size, and reliability are high priorities shall be employed. Its microprocessor and its large on-board selection of the most commonly needed peripheral devices enable it to operate as an effective stand-alone single board computer. It carries the PROM burned with the stand-alone software of the trip unit.

- **STD-Bus Digital Input Boards.** They should be reliable general purpose STD-bus interface boards, which provide an adequate number of 8-bit parallel input ports. It is used by the trip unit to get digital signals from its unshared single allocated conditioning unit and its unshared single allocated safety setting input unit.

- **STD-Bus Digital Output Boards.** They should be reliable general purpose STD-bus interface boards which provide an adequate number of 8-bit parallel output ports. It is used by the trip unit to send digital signals to both voting units of the reactor protection system, its unshared single allocated safety setting input unit, and its unshared single allocated watchdog unit.

- **STD-Bus Analog Input Cards.** They should be reliable, high speed, at least 12-bit analog to digital convertors. In order to avoid complex data-acquisition methods like interrupt mode or DMA mode, each analog input should be read by the trip unit processor in polled mode. They are used by the trip unit to get analog signals from its unshared single allocated conditioning unit.

- **STD-Bus Backplane.** It is physically comprised of the card cage, connectors, and backplane printed circuit card. It provides the common connection across which all STD-bus products plugged into the card cage communicate.

2.2.4.4 Voting Units

There are two redundant hardware voting units working in parallel. Each voting unit is comprised of voters. There are as many voters per voting unit as the amount of output signals from one trip unit.

Each voter outputs a digital binary signal, called the voted signal. This voted signal is produced by applying logic two out of three to the three redundant digital binary inputs to the voter. These three redundant digital inputs are produced by the three redundant trip units of the reactor protection system.

For each voter, logic two out of three means that the voted signal is true only when at least two of the three redundant input signals are true; otherwise is false.

2.2.4.5 Protective Logic Units

There are two redundant hardware Protective Logic Units working in parallel. The outputs of these two redundant units are voted using logic one out of two at the actuator/interlock device level.

Voted signals are used to generate the Protective Actions Initiating Signals (PAIS) and the Protection Interlock Initiating Signals (PIIS). PAIS control whether to trigger or not the four safety actuation systems i.e. the First Shutdown System (SCRAM), the Second Shutdown System (Gadolinium Injection), the Chimney Water Injection System (CWIS) and the Evacuation Alarm System (EAS). PIIS control the internal configuration of the protective logic units and the interlocks to the safety actuation systems.

In order to supervise the operation of the voting and protective logic units, Safety Parameters (SP) are sent to both Safety Parameter Display Units (SPDU) in the main and secondary consoles and to the Supervision & Control System (SCS) via appropriate interface units.

2.2.4.6 Supervision & Control System (TLIU)

The Supervision and Control System (SCS) supervises and controls different safety related systems and systems not important to safety of the research reactor.

Following the defence-in-depth concept, it also supervises data related to the safety systems, in order to check Limiting Conditions for safe Operation (LCOs), which are established to provide acceptable margins between the normal operating values and the safety system settings during all operational states of the reactor. Therefore, data used or generated by the trip units are read by the SCS from its three redundant one-way TLIUs, which are updated periodically by their appropriate trip units.

These TLIUs provide shared contiguous I/O address space with optical isolation between the trip unit and the SCS. The optical isolation provides an electrical barrier between the trip unit and the SCS, so decoupling both systems. Hard built-in read/write characteristics of the shared data area are such that it can be only written by the trip unit processor and only read by the SCS.

2.2.4.7 Watchdog Unit

The Watchdog Unit (WU) monitors the correct operation of the trip unit processor by checking that the Trip Unit Software (TUS) is running within correct loop timing and in correct execution sequence.

It shall be triggered periodically by the trip unit processor, in order to confirm that the TUS is running within a correct loop timing.

It shall also receive from the Trip Unit a sequence of expected digital codes within each cycle of the TUS. The reception of these sequence of expected digital codes enables the WU to confirm that the TUS is sequentially executing the coded-in expected milestones and therefore confirm that the TUS is following an expected execution thread.

A single failure either to strobe the watchdog timer within a set time period or to receive the expected digital code forces the WU to hold static the Watchdog Trip Signal (WTS). By keeping this signal static, the WU is requesting actuation of the First Shutdown System (SCRAM). This request is reset only when the trip unit is power-on.

2.3 Supervision & Control System

2.3.1 Design Criteria

Selection of the hardware architecture of the Supervision and Control System should be based on the following design criteria

- Simplicity
- Reliability
- Performance
- Availability
- Maintenance
- Expandability
- Standardisation
- Modularity & Flexibility

2.3.1.1 Simplicity

Maximum simplicity of the system is achieved by using

- low number of hardware components with low diversity of types within the system
- simple interconnection structure between system components due to its horizontal logic structure
- low number of software components due to the use of similar platforms on each processor
- low software complexity due to the reduced number of communication paths between software components

2.3.1.2 Reliability

High reliability of the system is achieved by

- developing the software using an adequate structured methodology for real time systems development
- applying an intensive verification and validation program under an adequate testing life-cycle
- using highly reliable hardware components

2.3.1.3 Performance

High performance of the system is achieved by

- applying, wherever possible and necessary, parallel execution of different specific tasks on different units
- using processors with a high processing and numeric calculation capacity
- using a high-speed communication network
- using a well distributed database in order to increase the speed of data recovering

2.3.1.4 Availability

High availability of the system is achieved by

- high availability to fulfil the required valued for operation
- using electrical power from an uninterruptible power supply
- using two redundant communication paths to implement the required communication networks
- using two redundant hard disks on control units to ensure a high data availability

2.3.1.5 Maintenance

Low maintenance cost is achieved by

- using a modular design where maintenance of hardware components is done by replacement Mean time to repair is very low (about 5 minutes) and is done during the operation of the system with only temporary local function degradation
- using low number of hardware components with low diversity of types
- using standard components and diagnostic tools to detect a problem and to determine the error cause
- using on-line software modules to automatically detect and correct errors during the operation of the system

2.3.1.6 Expandability

High grade of expandability of the system is achieved by using a modular design both in software and hardware Modular design enhances the following expandability features

- Quantitative increase possibility e g addition of signals, control loops, new functional requirements, etc
- Simplicity to increase/decrease software modules e g simulator training implementation is a matter of adding only the models of the plant

2.3.1.7 Standardisation

High standardisation of the system is achieved by

- assuring availability of all required hardware components in the market
- using market structured communication interfaces
- using well known and structured software support

2.3.1.8 Modularity and Flexibility

High grade of modularity and flexibility of the system is achieved by using a modular design both in software and hardware Modular design enhances the following features

- Ability to easily handle possible modifications on the working load, and possible variations, modifications, replacements and extensions on the configuration of the system
- Easy implementation of future changes in the interfaces of the system
- Hardware architecture applicable to different environments
- Further functional division in different specific and interconnected subsystems, enabling their integral supervision and control
- The concept of modular field units within a distributed system is easily adaptable to the reactor process Because the reactor process to be supervised and controlled is comprised of different independent functional subprocess which are distributed in the reactor building

2.3.2 Functional Requirements

2.3.2.1 The Supervision and Control System performs several functions which are described in the following sub-sections.

2.3.2.2 Alarm Handling

The supervision and control system should present all information related to system alarm status

The alarm handling system shall detect alarm activation and clearance at different levels They can be categorised according to process requirements

- Alarm prioritisation according to alarm relevance
- Alarm masking on cause-effect relationship
- Chronological alarm storage
- Alarm differentiation using color

The system shall provide several notification-means to alert operators on an alarm condition, as message presentation on monitors, indicators on console, graphical presentation in system context, audible indications

The operator shall acknowledge all activated alarms An explanatory message is available for each one

All alarm condition events, alarm clearing, and/or acknowledge actions are stored chronologically, along with the time of occurrence as well as some additional information, in the historical database, and in a permanent recording device

2.3.2.3 Report Generation

The supervision and control system should have a printed report generator management module This module should be able to generate spreadsheets with fixed text along with process and operators' data

Report generation should be triggered by operator request or periodically by an automated procedure at specified time intervals, or on specific events such as shift rotation, experiment completion, etc

2.3.2.4 Event Recording

The system should have the capability to record different kinds of events produced during control and supervision of different plant processes. They might be stored chronologically in the database.

2.3.2.5 Plant Control

Reactor operators shall have direct and simple access to the control commands on different plant processes.

Software modules included in the plant control system should be able to handle: regulating control loops, interlocking logic, execution of logical sequences, and manual control commands.

Individual control commands, which are generated by operators to activate plant equipment actuators, have to travel from the supervision level to the field level.

2.3.2.6 Supervision Interface

The system provides the operator a global or partial plant status display by means of several diagrams and/or tables, mimics, histograms, graphs, alphanumeric and auditive information, etc.

The system supervises operator actions during component manipulation, such as valve opening/closing, pump start up/shutdown, parameter or setpoint changes, command sequences, etc., providing guidance to operator actions.

The supervision and control system displays information in the reactor operating console to help reactor operation during normal or abnormal conditions, providing operators clear and concise information. This information display is arranged hierarchically in the video display units, according to functionality and priority.

2.3.2.7 Administrative Data Management (ADM)

The supervision and control system software has an auxiliary module dedicated to processing administrative data called Administrative Data Management (ADM.) Capabilities of this software module are database management for the following data:

- Frequently used data: daily reports, data sheets, checklists
- Safety system behavioural analysis
- Actual operational status (components out of service)
- Periodic maintenance, testing, and inspection
- Facility upgrade data
- Failures during operation, events and incidents related to safety
- Radioactive inventory
- Personnel functions and training
- Radiation exposure and medical checkups of personnel
- Quality assurance audits and testing
- Start up reports, including testing

- Decommissioning data
- Communication with regulating boards

2.3.2.8 Limitation Functions

The limitation system has limit control mechanisms that protect the plant from operational perturbations that could occur if operational parameters were allowed outside limit ranges. Should these limits be exceeded, the limitation system will try to bring the reactor back to normal operating range This is generally accomplished by means of reactor power reduction.

These limitations prevent undesirable protection system activation resulting in a SCRAM with complete insertion of all control rods, thus increasing plant availability.

2.3.2.9 Reactor Power Regulation System

The reactor power regulation system is implemented as a software module within the Supervision and Control system. This system, when active, is responsible of regulating reactor power to its setpoint, compensating reactivity changes (temperature effects, Xenon, sample insertion, fuel burnup, etc.) and to perform power changes to new levels upon modification of the power setpoint

This system takes two input channels: one is a Wide Range Fission Chamber (WRFC), covering the range from source level to 125% full power, another is a N-16 Gamma Ionization Chamber, for power range, which effectively measures the reactor global core power The output control signal drives the movement of the regulating rod Since the safety compensated ionization chamber readings from the reactor protection system are available at the supervision and control system, then these readings and the reading from the WRFC are adjusted using appropriate correction factors to provide a better estimate of the average core power This is important because the objective of the reactor power regulation system is to keep at a reference value the average core power, not a local power

The control algorithm uses for short term response the WRFC channel and, for long term power response, the N-16 Chamber input

The WRFC provides a fast input based on local flux measurement, whereas the N-16 Chamber provides a delayed, cooling-flow sensitive integral value of the core power output

This type of control improves availability and safety conditions at high powers, by preventing power overshoots when the WRFC may be partially shielded/under-reading by spatial configurations or by skewed flux shapes

This system can be overridden by operator manual control at any time

2.3.2.10 Data Communication

The supervision and control system allows for transmission of real-time data and operational/historic database information to external nodes, to satisfy requirements from administrative, isotope production or scientific staff

2.3.2.11 Status and Self-Diagnosis

These SCS has a set of routines to verify overall system performance and to diagnose its functional status. This is informed to reactor protection system and to operators.

2.3.3 System Architecture

Once it has been accurately and unambiguously defined *what* it is requested to the system, the process of defining *how* the system must be designed within the design constraints can start. At this stage of the development the type and number of hardware units shall be defined and selected. These selection depends strongly on the functionality required. Function allocation shall also be performed i.e. it shall be clearly defined the work (functions) allocated to each hardware unit.

For many well-known reasons found in different publications on the subject and current trend on nuclear power plant and research reactor design, it is recommended that the supervision and control system of a research reactor should be designed as or upgraded to a computer-based distributed system. Therefore, the hardware units mentioned above should correspond to industrial computers, single board-computers with adequate boards following the STD-Bus standard or VME standard, or adequate PLCs (e g SPS Speicher Progrmmierbare Steuerungen).

Based on the functional requirements and specific design features described below, different alternatives must be analysed and then the most appropriate architecture for a computer-based distributed supervision and control system shall be selected Therefore, there is no unique architecture for all research reactors It must be carefully selected

Units of a computer-based distributed systems are physically distributed in the nuclear installation These units are hierarchically organised in multi processing-levels according to the type of tasks allocated to them Between these processing-levels communication networks must be considered

The supervision and control system integrates the main console, secondary console, training console, supervision desk, radioprotection desk, local supervision centres, instrumentation racks containing processing hardware and software, wiring and networking and plant instrumentation Fig 2 4 shows the general architecture of the supervision and control system and its relationship to the reactor protection system

The SCS is a distributed computer-based system, with a general architecture of three hierarchical processing levels

- Supervision level
- Control level
- Field level

and two communication levels

- Supervision Communication Network level
- Control Communication Network level

2.3.3.1 Processing Levels

A computer-based distributed I&C system may have many processing levels

2.3.3.1.1 Supervision Level

All man-machine interactions between operators and plant instrumented systems take place at this level. This level comprises all supervision units These units are used for reactor process supervision, command, and plant data recording and management These units are located in the main console, secondary console, training console, supervision desk, radioprotection desk, and local supervision centres.

The main supervision functions are: alarm presentation, global or partial plant status visualisation through several diagrams, operator notification of ongoing transients, operations or mode selection, manual equipment control (pumps, valves, etc)

There are two operating modes in the supervision level interactive and monitoring. In the interactive mode, all functions and supervision capabilities for start up, operation, maintenance and refuelling are available In the monitoring mode only essential functions to display global plant status at selected building areas are enabled

The Supervision Units allocate all information management functions as historic database, relevant signals, alarms, operations and sequences

The Supervision and Control System can also access safety parameters whose main function is to help control room operators acquire a quick plant status estimation

This supervision also includes information to allow operators estimate reactivity control, core and primary system cooling conditions, radioactive control and containment status

2.3.3.1.2 Control Level

All reactor process control functions take place at this level This level comprises all control units These units are in charge of data collection and transmission, control algorithms execution and process interlocks verification

Each control unit is connected to a set of field units, through the control network, conveying process data and actuation signals Each control unit has access to all signals collected by all connected field units, this information conforms the real-time database allocated to that control unit

Each control unit can access real-time databases in other control units, to obtain process variables collected by other field units This way, all control units can access any required process signal values

Output variables (driving signals for process actuators) are handled similarly Each control unit has command signals for all output signals belonging to the field units connected to it These output values are also stored in the real-time database for that unit

2.3.3.1.3 Field Level

This level comprises all field units. These units are in the boundary of the system and constitute the link between control units and sensor/actuator devices. Input data are acquired from plant sensors and processed here, before sending the information to upper levels in the hierarchy. Output data coming from control units is sent to appropriate plant actuators.

Signal conditioning takes place at this level for all acquired signals through field sensors, and for all actuation signals (field actuator excitation). This conditioning includes scaling, linearization, local validation and some degree of reflexive control.

The interface between field units and the process is implemented by means of signal conditioning modules (scaling and electric conversion) for both data input and actuator excitation. For each particular case, galvanic isolation can be provided between field units and sensors or actuators.

All process variables required by the Supervision and Control System take place at the field unit level.

The Reactor Protection System, being a completely independent system, handles exclusive input/output signals. Most of these signals are nevertheless, also relevant to the supervision and control system. Therefore, they are read from the reactor protection system using a one-way custom communication channel with optical isolation.

One possible functional distribution of the Field Unit is shown in the following table

TABLE III: FIELD UNITS. FUNCTIONAL DISTRIBUTION

FIELD UNIT	INTERFACE TO
Field Unit 1	Reactor Protection System
Field Unit 2	Control Rods and Reactor Hall Systems
Field Unit 3	Ventilation Systems
Field Unit 4	Physical Security Systems
Field Unit 5	Electrical Systems
Field Unit 6	Experimental Facilities
Field Unit 7	Process Systems
Field Unit 8	Auxiliary Process Systems

2.3.3.1.4 *Field Instrumentation*

The sensors and actuators which are connected to the supervision and control system are safety-related sensors. They are typically no redundant and their signals are of different types: on/off, digital, analog. These sensors can be connected directly or through signal transmitter to the conditioning modules of the field units. Signal transmitters could be intelligent in which case, they can access the communication channel with the control units directly Sensor configuration and distribution is exclusively dependent on the kind of process requiring supervision or control capabilities

2.3.3.2 Communication Levels

The supervision communication network level connects all Supervision Units and Control Units

The control communication network level connects all control units and field units

Both communication networks are independent Each of them is implemented with double redundancy to increase its availability The speed required for both communication networks and therefore the type of communication network depends on the amount of data to be transferred by them

2.4 Control Rooms

Control rooms of a research reactor are the supervision and control centres of the plant They concentrate all the equipment of the I&C system, which enable the operating staff of the research reactor

- to operate the plant safely and efficiently in all its operational states,
- to take measures to maintain the plant in a safe state or to bring it back into such a state after the onset of either accident conditions or design basis events, and
- to supervise specific tasks

Only the control room design issue on control room I&C components is mentioned here, i.e.

- main console
- secondary (emergency) console
- training console
- supervision desk
- radioprotection desk
- instrumentation racks

Another important issue is the Man-Machine Interface (MMI) design. Excellent guidelines to develop man-machine interfaces for nuclear facilities are provided in the EPRI report EPRI-NP-3701.

The Man-Machine Interface (MMI), which belongs to the I&C of the plant, allows the operation staff to supervise and control the plant. The MMI is comprised of hardware and software interface units. First MMI designs were fully implemented using hardware units, but the current practice and tendency is to include more and more software units on new designs or plant upgrades.

These interface units are physically distributed across the plant buildings. They are divided in local or remote interface units. Local interface units are located next to the system or equipment under supervision and/or control. These local units are used by the local operators during the operation of the system or equipment; and by engineers during the independent commissioning of the system or equipment. Remote interface units are located in specific control rooms or control positions and grouped in appropriate control consoles, supervision desks or supervision centers. These remote units are used by console operators. shift supervisors. radioprotection officers and reactor head to supervise and control the operation of equipments, systems, functional groups and/or the plant and by appropriate staff during the reactor commissioning. Functional. safety. and cost/benefit criteria shall be analyzed before deciding about local/remote characteristics of all and each interface units.

Other important control room design issues, such as:

- Staff and operating principles
- Relationship with other control centres
- Location, environment and protection
- Layout
- Communication systems

are not considered here, because they are out of the scope of the present report. Issues on functional and ergonomic design of control rooms are explained in detail on reports such as NUREG-0700 and IEC-964.

2.4.1 Main Console

The current practice on new designs and upgrades of main consoles of nuclear research reactors is to locate on this console all the equipment needed by the operator to operate the plant safely and efficiently in all its operational states and to take measures to maintain the plant in a safe state or to bring it back into such a state after the onset of either accident conditions or design basis events.

All these equipments are located in the main console to reduce the work-load to the operator. This became possible since the introduction of CRT monitors, which enables to display huge amounts of information in one rack of the console. The layout of the console depends on the specific operational requirements and the design, size and characteristics of the instrumentation and controls used.

With the introduction of PC based monitoring, PLC or a software-based distributed I&C system a large number of conventional recorders, indicators, switches, alarm windows, lights etc. are eliminated in the case of non safety related systems, as these parameters are incorporated in software based systems. This has resulted in considerable reductions of number of co-operative consoles and vertical panels. A majority of operation and information is implemented through mimics on the touch screens.

The main console is based on a modular design and consists of nine console racks, forming a continuous and comfortable workplace. Each console rack, on its top surface include indicator panels or CRT monitors for information presentation, plus custom command panels for dedicated keyboards, push-button groups, track-balls and communication/command equipment sets. Internally, room, connections and anchoring points are available for cabinets, such as supervision computers, electronic

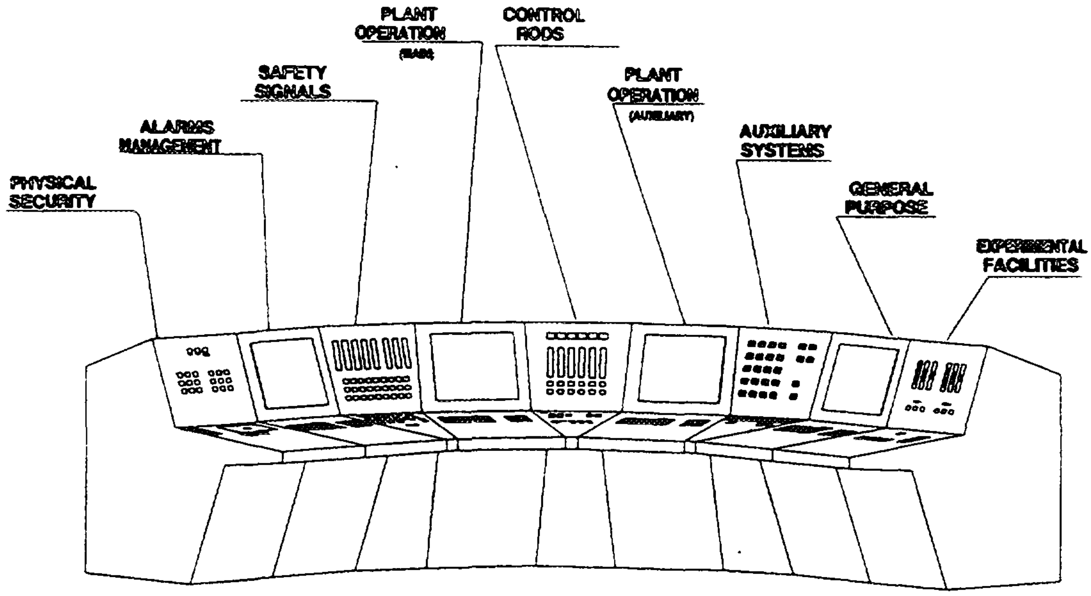

FIG. 2.6: Main Console - Architecture and Functions

TABLE IV: MAIN CONSOLE RACKS. BASIC FUNCTIONS AND RELATED SYSTEMS

CONSOLE RACK	BASIC FUNCTION	SYSTEM
Console Rack 1	Physical Security	PPS, ITS, CCTV
Console Rack 2	Alarm Management	SCS
Console Rack 3	Safety Signals	RPS
Console Rack 4	Plant Operation (Main)	SCS
Console Rack 5	Control Rods	SCS
Console Rack 6	Plant Operation (Auxiliary)	SCS
Console Rack 7	Auxiliary Systems	SCS
Console Rack 8	General Purpose	SCS, ITS, PS
Console Rack 9	Experimental Facilities	SCS

interface units and power supply units. An integral armrest, for written handwork, contours the inner border of the console.

The ergonomic design of the main console is based on nuclear power plant concepts. Mechanically, the main consoles is designed under a frame and panel concept, including lateral force resistant features, for seismic loading, built to seismic mechanical requirements.

In particular, console rack three (third from the left standing in front of the console) is a Safety Display and Command Rack, which belongs to the reactor protection system. Other racks of the main console belong to the supervision level of the supervision and control system. The third rack is hardwired, and is functionally and electrically independent from the other console racks. Its indicator panel implements the Safety Parameter Display Unit, which presents the status of the neutron and safety thermohydraulic instrumentation and the discrete (on/off) safety signals involved in the safety logic. Its command panel includes the manual triggering, interlock and reset push-button of the protection and active safety systems. This rack in the main console includes all the equipment that is required to shutdown the reactor and to keep the reactor in a safe shutdown condition.

Account should be taken of the need for sufficient redundancy and diversity for the I&C used in the main console, and physical and electrical isolation of the main console.

2.4.2 Secondary (Emergency) Console

All the equipment needed by the operator to shutdown the reactor and to keep the reactor in a safe shutdown condition is located in the main console. However, they shall be duplicated in the secondary console. Because access to these equipments in the main console may become unfeasible or the operating personnel may be forced to evacuate the main control room due to emergency conditions.

To avoid unfamiliarity problems to the operator with the equipment layout, the secondary console shall be an identical replica of a section of the main console. Therefore, racks three and four of the main console are duplicated to build the secondary console. These racks are called in the secondary console, one and two, respectively. Their basic functions and I&C systems are shown in Table V.

TABLE V: SECONDARY CONSOLE RACKS. BASIC FUNCTIONS AND RELATED I&C SYSTEMS

CONSOLE RACK	BASIC FUNCTION	SYSTEM
Console Rack 1	Safety Signals	RPS
Console Rack 2	Plant Operations	SCS

Rack 1 (first from the left standing in front of the console) is the Safety Display and Command Rack, which belongs to the reactor protection system. In all its engineering aspects, it is equivalent to rack three in the main console: hardwired, functionally and electrically independent, its indicator panel implements the Safety Parameter Display Unit, its command panel includes the manual triggering, interlock and reset push-button of the protection and active-safety systems. This rack in the secondary console includes all the equipment that is required to shutdown the reactor and to keep the reactor in a safe shutdown condition.

Rack 2 (second from the left standing in front of the console) includes a supervision unit of the supervision and control system, with the same man machine interface capabilities as rack four in the main console. This rack in the secondary console is not essential either to shutdown the reactor or to keep the reactor in a safe shutdown condition.

Account should be taken of the need for sufficient redundancy and diversity for the I&C used in the secondary console, and physical and electrical separation from the main console

2.4.3 Supervision Desk

The supervision desk has one supervision unit with one CRT and one dedicated keyboard This unit, which belongs to the supervision level of the supervision and control system, enables the shift supervisor to monitor the same information as that displayed on the operator's console, but no control functions are provided

2.4.4 Radioprotection Desk

The radioprotection desk is identical to the supervision desk The main function allocated to the supervision unit in the radioprotection desk is to display information from area monitors and radioprotection equipment. Command functions are limited and enable the safety officer only to trend data downloading, operational parameters uploading, reconfiguration and alarm resetting related to radioprotection monitors, such as Area Monitors, Gaseous Effluents Monitors, and Failed Fuel Monitors. No other control/command functions are provided to control the operation of any other equipment of the plant.

2.4.5 Training Console

The goal of the training console is to train new members of the operating personnel and to teach students different topics on nuclear reactor control To achieve this goal, the training console reproduces the main functionality of the main console Given the proper authorisation, all plant supervision information is made available under the same man machine interface, but with limited control capabilities

The training console is a replica of the main racks of the main console It excludes the functions associated with physical security and auxiliary and experimental facilities All devices to monitor the plant during shut-down, start-up and normal power operation, including transients and accidents, are included Reactor Power Regulation Control is identical to that of the main console, but can effectively operate only under authorisation

The training console is based on a modular design and consists of five console racks, as those of the main console The first rack of the training console is a replica of the second rack of the main console, and so on with the following racks Therefore, the fifth rack of the training console is a replica of the sixth rack of the main console Their basic functions and I&C systems are shown in Table VI

TABLE VI: TRAINING CONSOLE RACKS. BASIC FUNCTIONS AND RELATED SYSTEMS

CONSOLE RACK	BASIC FUNCTION	SYSTEM
Console Rack 1	Alarm Management	SCS
Console Rack 2	Safety Signals	SCS
Console Rack 3	Plant Operation (Main)	SCS
Console Rack 4	Control Rods	SCS
Console Rack 5	Plant Operation (Auxiliary)	SCS

All racks are connected to the supervision and control system. The Safety Display and Command Rack i.e. rack one of the training console is driven by software modules of the supervision and control system. This software-driven rack neither receives nor sends signals to the reactor protection system, but its interface is identical to rack two of the main console.

2.4.6 Instrumentation Racks

Instrumentation racks used to be located mainly in the control room but current practices are to distribute them along the reactor and auxiliary buildings following the concepts for a distributed I&C system. Some or none of them may be located in the control room. This will depend on specific requirements for each particular research reactor.

Instrumentation racks may be located in suitable separate areas from the control room for case of on-line maintenance, replacement, calibration and routine electronic testing of modules.

In general, the instrumentation racks in a distributed I&C system have typical functions as shown in Table VII and belong to different levels and divisions of this I&C system as shown in Fig. 2.4.

TABLE VII: INSTRUMENTATION RACKS. BASIC FUNCTIONS

INSTRUMENTATION RACK	FUNCTION
RPS-IR1	Channel 1 RPS Units
RPS-IR2	Channel 2 RPS Units
RPS-IR3	Channel 3 RPS Units
RPS-IR4	Voting and Protective Logic Units
SCS-IR1	Field Unit 1 - Interface to RPS
SCS-IR2	Field Unit 2 - Control Rods and Reactor Hall Systems
SCS-IR3	Field Unit 3 - Ventilation Systems
SCS-IR4	Field Unit 4 - Physical Security Systems
SCS-IR5	Field Unit 5 - Electrical Systems

INSTRUMENTATION RACK	FUNCTION
SCS-IR6	Field Unit 6 - Experimental Facilities
SCS-IR7	Field Unit 7 - Process Systems
SCS-IR8	Field Unit 8 - Auxiliary Process Systems
SCS-IR9	Control Units

Mechanically, each instrumentation rack is a heavy welded frame structure, with front and rear access doors, interior supports cabinets supports (typically 19 inches cabinets) and connector plates for wiring. Its design includes structural reinforcement features for seismic resistance in both lateral directions. Power supply and ventilation are below the floor level, through ducts and conduits.

During operation, the operator access points to the instrumentation racks are limited to the Safety System Setting Input Unit (Fig. 2.5). These units are three redundant hardware panels to define the Safety System Settings for each channel of the reactor protection system. Therefore, these units are included in three redundant racks, e.g. RPS-IR1, RPS-IR2 and RPS-IR3 (Fig. 2.4). No other access points are considered during operation.

Video and keyboard interfaces are used during commissioning, testing and maintenance of each trip unit, field unit and control unit of the I&C architecture. These access points should be available only to authorised personnel.

3. TECHNOLOGICAL OPTIONS

In order to operate a reactor, many variables have to be measured and monitored. Some of them are considered to be vital for the safety of the reactor, and their measurement has to be fast and reliable. What follows is a description of the technological options actually used to measure, process, display and record the variables in research reactors.

3.1 Nuclear Instrumentation

In research reactors the neutron flux has to be monitored from a low level to maximum power level. The detected flux is normally subdivided into three intervals, source range (below 10^4 nv, equivalent to 10^{-3} percent of nominal power), intermediate range (from 10^3 to 10^9 nv, or 10^{-4} to 10^2 percent of nominal power), and power range (from 10^7 to $2x10^9$nv, or 1 to 200 percent of nominal power) as shown in Fig. 3.1. In the source range interval, due to the low rate of events occurring within the detector, pulse counting mode is traditionally used, whereas the current mode is adopted in the intermediate and power range intervals. The following equipment is currently available for detection and measurement of the neutron flux in the above three intervals.

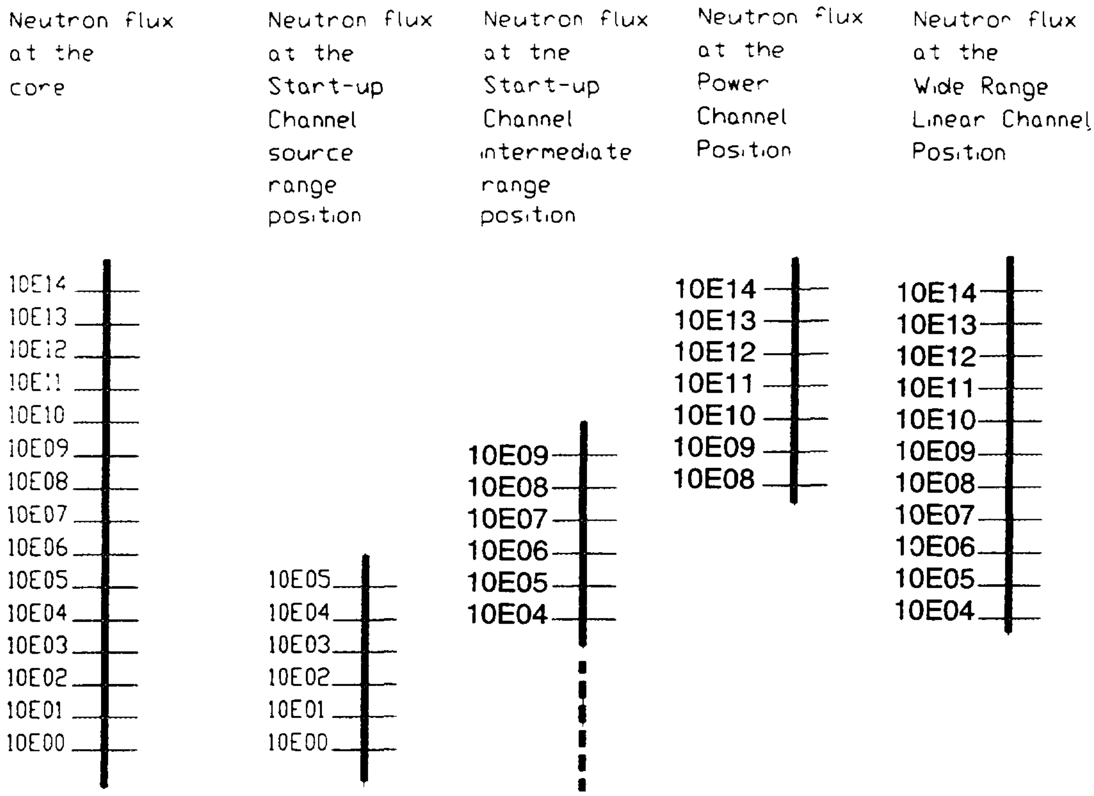

FIG. 3.1: Neutron Flux Measuring Ranges

3.1.1 Neutron Detectors

The main reason to monitor neutron flux in a reactor is that it is proportional to the power density, and this is the variable which we are concerned about. There are mainly five types of neutron detectors, BF_3 proportional counters, Boron (^{10}B) lined detectors, fission chambers, ^{3}He proportional counters, and self powered neutron detectors Two other instruments are widely used to monitor the reactor power, calorimeters, used to monitor power density, and N-16 detectors, used to monitor integral reactor power

Detectors are usually located in fixed positions about or around the reactor core, and the quantity measured is often the leakage flux, i e , the flux exterior to the core The detector shall be placed in instrument ports that are leak-tight so that they will not be subjected to moisture A purge of inert gas in these ports is also often provided in order to prevent the buildup of corrosive substances Since detectors are located in intense gamma ray fields, which may degrade cable insulation, mineral insulators are preferred such as are provided with integral-lead detectors These cables are procured at the time the detector is ordered, and must be sized for length at that time

If the detector is sensitive enough, it is recommended to place it as far as possible from the core This will reduce radiation hazard to the detector, and also limit the effect of gamma induced current

In general, neutron and radiation detectors may be designed for any of three modes of operation current mode, pulse mode and mean square voltage (MSV) mode. The latter is also called "Campbelling" mode In any detector the incident radiation is converted to electric charge The three modes of operation refer to the manner in which this charge is collected and processed The mode of operation of a particular system depends on the specific design of the detector and its external electronics

- Current mode A current measuring circuit is placed across the terminals of the detector, and the average current is measured by the system Current mode operation is the most common operation mode It is used for measuring the power of the reactor and also in most portable survey meters (see Section 3 1.2 4)

- Pulse mode An RC circuit is placed across the terminals of the detector and the voltage produced in the circuit is measured Pulse mode is used in a research reactor mainly during start-up because it provides spectral information, used to distinguish between neutrons and gamma radiation Pulse mode detectors are also useful in criticality experiments in which it is desirable to plot the inverse counts versus the mass of ^{235}U in the core

- MSV or Campbelling mode The output of the current mode operation is a varying current which can be regarded as a sum of a steady-state component and a time varying component. The MSV circuit blocks the steady-state component and squares the amplitude of the varying component. The resulting signal is proportional to the square of the charge that is created by each incident particle of radiation, thus enhancing the difference between types of radiation. MSV mode is used primarily for nuclear reactor instrumentation as a means of measuring neutrons in high gamma background. Campbell instrumentation may be used to measure the neutron flux in a reactor from source range to power range with a single channel. (see Section 3.1.2.5)

Since the insertion and withdrawn of control rods on top of the core changes the shape of the neutron flux, care must be taken in selecting the right position for the neutron detectors. Periodic calibration and cross checking of neutron detectors is highly recommended.

3.1.1.1 BF_3 proportional counters are highly sensitive to neutrons, and usually are used in a pulse counting mode to monitor the neutron flux and the reactor period (rate at which the neutron flux changes) in the source (start up) range. Even it is not usual, BF_3 detectors can also be used as compensated ionization chambers, to monitor the neutron flux level and the reactor period in the intermediate range.

3.1.1.2 Boron (^{10}B) lined detectors are used in both, pulse counting mode (as counters) and DC current mode (as ionization chambers). As counters they are used in the source range, and as ionization chambers they can be used either as intermediate or as power range detectors. Usually when used in the intermediate range they are gamma compensated ionization chambers, and when used in the power range they are uncompensated ionization chambers.

3.1.1.3 Fission chambers with a coating ^{235}U are suitable for both, source and intermediate ranges. In the source range it is used in pulse counting mode, and in the intermediate range it is used with both, pulse counting mode and campbelling mode.

Special features of the utilization of fission chambers is their application to the wide range monitoring system, where a single detector is effectively used with a combination of pulse counting mode and campbelling mode to monitor the reactor operation for the range of about ten decades, from start up to power. Miniature size fission chambers are also available for in core use. In case of wide range monitoring systems, regenerative fission chambers, with a mixed coating of ^{235}U and ^{238}U are currently applied in power reactors, and can also be used in research reactors.

^3He proportional counters are very high sensitive to neutron flux, and can be used in the start up range, however, because of their high cost, they are mainly used for experimental purposes.

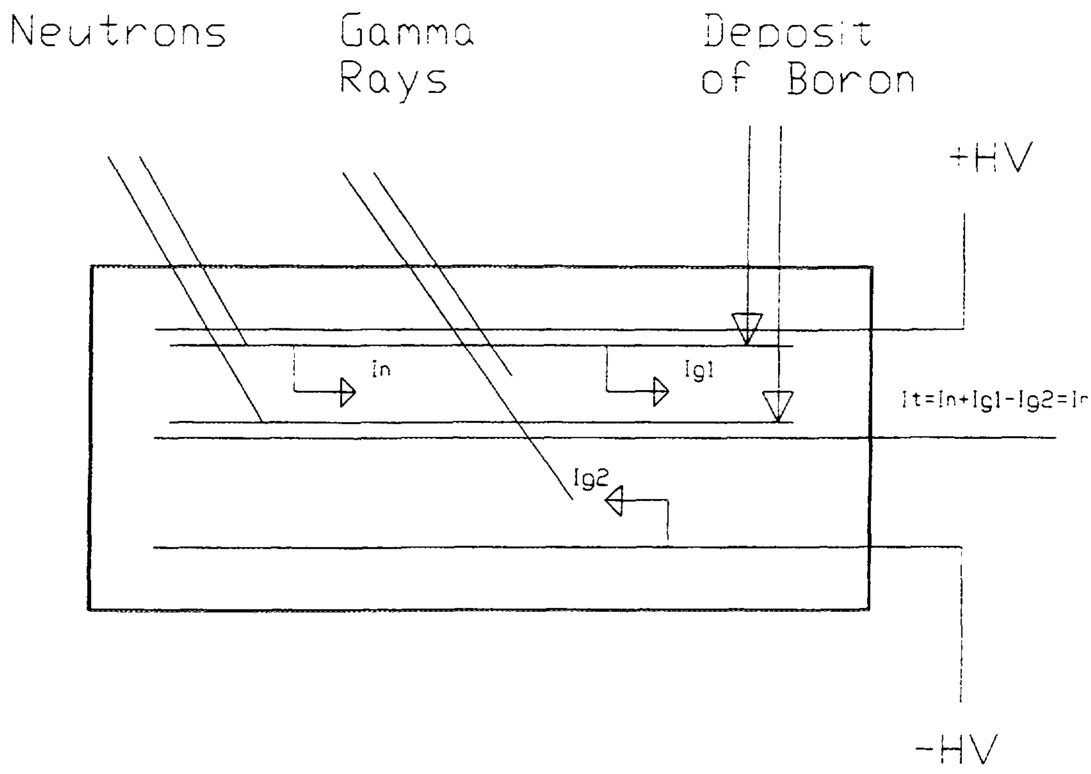

FIG. 3.2: Compensated Ionization Chamber

3.1.1.4 Self powered neutron detectors (SPND) utilize a β decay current that is emitted from a neutron activated material, usually rhodium or vanadium, rhodium being the most used. The time response of this type of detector is fully dependent on the half life of its β decay reaction, and in general it is used to monitor neutron flux level during stationary conditions. Because of their small size, SPNDs are suitable for in-core use, specially for flux mapping and flux monitoring at irradiation positions. One variation of the SPND, known as prompt SPDs uses capture gamma rays to generate the current. In this case photo electrons and compton electrons are emitted by gamma rays (prompt capture gamma rays or fission gamma rays). Because they are fast and sensitive to fission gamma rays, they are mainly used to monitor power density. The most common materials used in prompt SPDs are Pt and Co. It is important to notice that β decay SPNDs are also sensitive to gamma rays, and that prompt SPDs are also sensitive to neutron activation, so, one must be careful when using these detectors, in order to properly understand the measured signal

3.1.1.5 Calorimeters - Even if it is not a neutron detector, calorimetric methods are useful for reactor density power monitoring A typical sensor is a gamma thermometer, in which gamma rays are absorbed increasing the temperature difference between a point close to a region surrounded by vacuum and a point in contact with the reactor coolant Since within the reactor the gamma flux is proportional to the fission rate, therefore, proportional to the power density, the temperature difference can be easily measured by a differential thermocouple, and related to the reactor power density

These gamma thermometers are widely used in power reactors, and can also be used in research reactors to monitor power density.

3.1.1.6 Nitrogen 16 activity is produced by the activation of oxygen-16 within the reactor core, through the reaction: $^{1}n + ^{16}O \rightarrow ^{16}N + ^{1}p$.

The ^{16}N level is therefore proportional to the reactor integral flux, or the integral power. Hence a detector that is located near a reactor coolant discharge line can be used to monitor the ^{16}N activity level, and hence the reactor power. This approach provides a useful way for corroborating the signals obtained from neutron detectors and calorimeters (heat balances). However, it should be recognized that the power signal from a N-16 detector will be delayed relative to the true core power by the time required for the primary coolant to flow from the reactor core to the detector location.

3.1.2 Nuclear Channels

Nuclear channels for research reactor applications can be modular or integrated. In integrated channels, with exception of the pre-amplifier, all electronic circuits necessary to measure and monitor the neutron flux level and period are incorporated in only one cabinet, in such a way that a single failure will require the complete channel to be serviced. Modular channels are made up of independent modular instruments, like amplifiers, power supplies and ratemeters, and any single failure will require only the replacement of the failed modulus. One family of standard modules widely used in research reactors instrumentation systems is the Nuclear Instruments Modules (NIM) which can be used for control and safety applications.

3.1.2.1 Preamplifiers

Neutron flux counting channels have to be incorporated in all research reactors. In order to avoid noise pick-up, the initial processing electronics (preamplifiers) have to be located at a short distance from the detector. However, preamplifiers are not located directly on the counters, to avoid radiation damage to its components. Three types or preamplifiers are available for use with detectors operating in the pulse counting mode:

- charge sensitive preamplifier,
- current mode preamplifier, and
- voltage sensitive preamplifier.

Charge sensitive preamplifiers are normally used with BF_3 detectors to integrate the charge from the detector. They work satisfactorily with input capacitance in the order of 1 pF. The reliability of counting with charge sensitive preamplifier is limited to 10^5 pulses per second.

Current mode preamplifiers can provide reliable counting up to 10^8 counts per second. They are specially suited for use with fission chambers.Voltage sensitive preamplifiers, with low gain, have high input impedance and low output impedance. They are useful for use with scintillation detectors, and also for matching of cable impedance.

3.1.2.2 Linear Amplifiers

Preamplifiers are used in nuclear channels in which the variable of interest is the rate of pulses occurring within the detector. Since they are installed in places with high radiation fields, their main purpose is to increase the pulse amplitude with a minimum of electronic components. Therefore, the output pulse of the preamplifier is not suitable for count rate meters, and additional signal conditioning is necessary. Linear amplifiers provide the necessary means to discriminate the undesirable noise and to properly shape the pulses, making them compatible with counters, ratemeters and spectroscopy analyzers. Conventional solid state amplifiers are an essential part of reactor protection systems, NIM modules are preferred, but many research reactors I&C systems have integrated nuclear channels. Both of them have shown to be reliable and safe.

3.1.2.3 Count Ratemeters

Count ratemeters are considered the final processing unit in a nuclear channel which uses pulses to monitor neutron flux. As part of source range channels, they transform a given rate at which pulses are being counted into a continuous signal, proportional to the measured rate. The continuous signal is compared to pre-set values to allow the withdraw of safety/control rods, if a minimum count rate is present, or to trip the reactor, if a maximum value is achieved before the intermediate range channels are active. The continuous signal is also used to monitor and record the neutron flux. Usually count ratemeters also incorporate a period meter circuit, which gives the rate with which the count rate changes. Since the count rate is proportional to the neutron flux, the period signal is an indication of the rate with which the neutron flux changes, and is used to trip the reactor whenever a "too low" value is achieved.

As in the case of linear amplifiers, count ratemeters can be a NIM module or an electronic block within an integrated system, and when it is part of the reactor protection system, conventional solid state components are preferred.

3.1.2.4 Current Meters

Current meters are the signal conditioning units for neutron detectors operating in DC current mode. Due to the high range in which DC current mode detectors are used, about six decades, usually research reactor I&C systems have two types of current meters, logarithmic and linear. Logarithmic current meters are used to monitor the neutron flux in the intermediate range, where precision is not the main requirement, since the reactor is far away from an unsafe condition. Linear current meters, covering two or three decades, are used to monitor the neutron flux in the power range, where precision becomes an issue. Regarding the type of equipment, in research reactor I&C systems, solid state current meters follow the line adopted for the source range channel. If the source range channel uses NIM modules, the other channels also use NIM modules, which means that the current meters are also NIM modules, on the other hand, if the source range channel is of the integrated type, then the current meters are electronic block within an integrated system. Multirange linear current meters with manual or automatic range switch, although available, are not recommended for safety applications.

3.1.2.5 Campbell Instrumentation

The use of Campbell instrumentation enables the measurement of neutron flux levels from source range to power range with a single detector. Wide-range channels, as they are known, are hence useful for application in research reactors. Because of the high pulse amplitude of fission chambers, Campbell instrumentation is specially suited to monitor the reactor in the start-up region, when the channel operates in the pulse counter mode. However, extensive use of these channels has not become popular in safety applications. There are some reported cases using microprocessor controlled wide-range channels, but it is only considered as an operational channel, without safety relevance.

3.1.2.6 High Voltage Power Supplies

The polarizing voltage needed for the detectors depends on the specific type. For BF_3 detectors, voltages up to 3000V are needed whereas in fission chambers, the voltages are usually limited to 800V. In the case of ionization chambers, the voltage depends upon the flux at the detector, and can extend up to 1500V. Power supplies normally have a current rating of 1 mA. In the case of detectors operating in the counting mode, the design of the power supply has to take into account the plateau length of the detector, which normally is 100V. Electronic regulators employing AC to DC converters operating in the frequency range of 4 ~ 20 kHz are normally used. The high voltage transformers in the power supply are shielded and encapsulated for noise-free operation as well as satisfactory performance in humid conditions. The ripple for high voltage power supply is limited to 10 mV. The technology for high voltage power supply is well understood and is good enough for extensive use in all safety channels of research reactors. A continuous check on the condition of the cable connecting the detector to the amplifier by a suitable technique such as injection and detection of a known AC signal is recommended.

3.2 Radiation Monitors

Radiation monitors are divided into three main categories: area monitors, air monitors and process monitors.

3.2.1 Area Monitors

Area radiation monitors are used to ensure that the radiation levels in the reactor complex are within acceptable limits, and provide trip/alarm signals depending on the set level. The following monitors can be used as applicable:

- Low-level gamma monitor

  This monitor employs a G.M. counter as detector, that covers 3 logarithmic ranges usually starting at 10 μ Sv/hr (1 mR/hr). The use of microprocessor units allows the use of more than one G.M. counter at the same unit, in order to increase the measuring range. The use of two G.M. counters allows to cover the range from 10^{-5} to 0.1 Sv/hr (1 mR/hr to 10 R/hr).

- High-level gamma monitor

  The detector for this instrument is a gamma ionization chamber, and has 3 logarithmic ranges extending from 0.01 to 1 Sv/hr (1 R/hr to 100 R/hr).

- Area neutron monitor

  This instrument uses a REM/n counter as a detector, and normally has 3 logarithmic ranges extending from 10^{-5} to 10^{-3} Sv/hr (1 mRem/hr to 100 mRem/hr). The basic detector is a BF_3 counter surrounded by a moderator/attenuator assembly (rich in hydrogen) to provide a weighted dose response from .025 eV to 14 MeV of neutron energy.

3.2.2 Air Radiation Monitors

Air radiation monitors are used to monitor particulate and gaseous activity in the reactor building air and at the building exhaust system. Particulate activity is monitored using paper filter where the particulate material is retained. Usually they also have a small gas volume which is used to detect and monitor airborne fission products, and are equipped with alarm circuits which can be used to produce an automatic isolation and shutdown of the air handling system. Sometimes a charcoal active filter is present in the monitor, to allow the detection of iodine, in case of fuel failure. Beta-gamma scintillation detectors, proportional counters and GM detectors are commonly used in particulate monitors. The monitoring of the building and exhaust air for gaseous activity normally involves the use of a gamma scintillation detector counting of Argon-41 in the exhaust stack and integral counting of gases within the reactor building. These systems normally have a large gas monitoring volume (10 to 30 liters), where a beta scintillation detector is used to allow the detection of noble gas fission products.

3.2.3 Process Radiation Monitors

Process radiation monitors are used to monitor radioactivity in process systems. like in the primary coolant system and in the purification system. The objective of process monitors is to assure that the radioactive material is not getting out of its barriers (for example failure of fuel elements). and that the purification system has not been saturated. and still has the capability to retain activation products.

Other possible points for process radiation monitor is on the secondary side of the main heat exchanger, to assure that there is no leaking in it. and effluent water monitoring. Usually process radiation monitors employ gamma scintillation detectors. If a research reactor is designed so that its core is in a closed vessel, then the gas (air or cover gas) located above the core tank's primary coolant can be sampled continuously by a beta-gamma detector, to provide a very sensitive means for detecting incipient failures of fuel element cladding.

3.3 Non-Nuclear (Process) Instrumentation

3.3.1 Scope

Non-nuclear instrumentation, comprises those process measurements that do not involve neutron or gamma parameters (Fig. 3.3).

Non-nuclear instrumentation includes both safety signals and safety related signals. Safety signals are redundant and are sent to the Reactor Protection System (RPS). Safety related signals are sent to the Supervision and Control System (SCS).

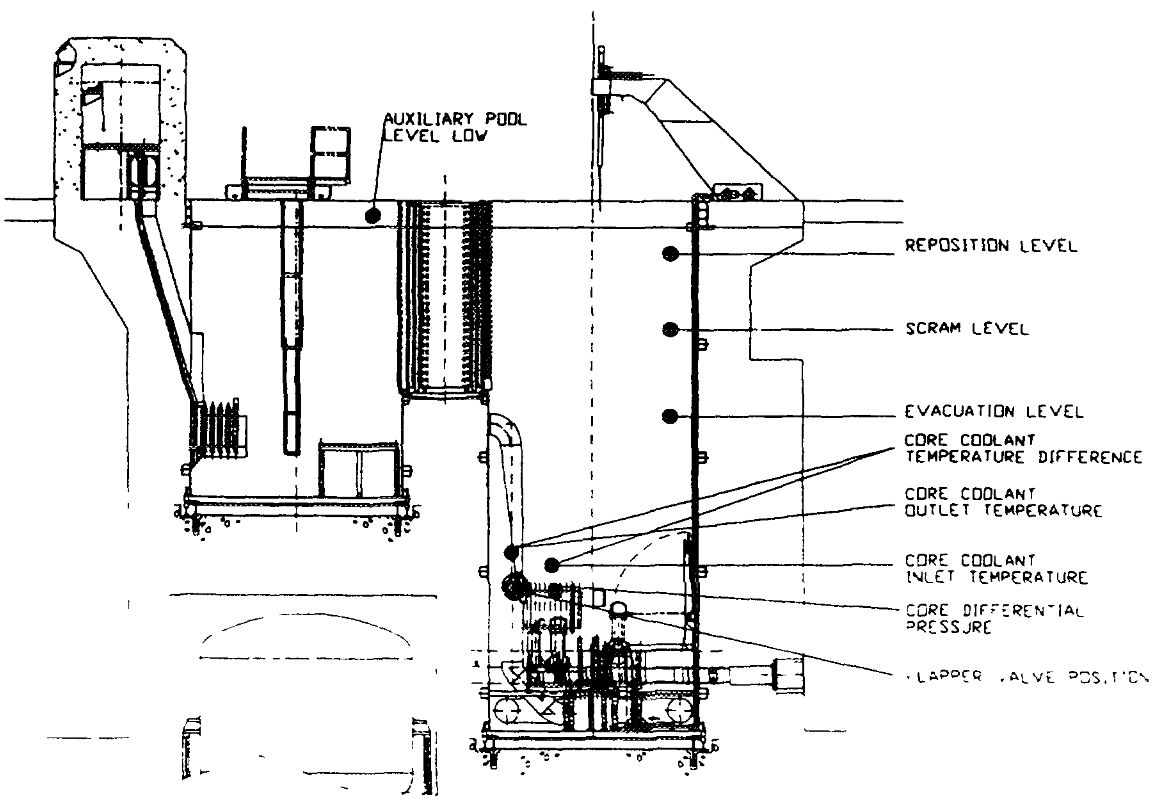

FIG. 3.3: **Reactor Core and Primary Cooling Circuit Instrumentation**

3.3.1.1 Safety signals

The following systems supply signals to the RPS.

- Reactor cooling system instrumentation
- Control assemblies mechanisms instrumentation
- Seismic instrumentation
- Ventilation system instrumentation

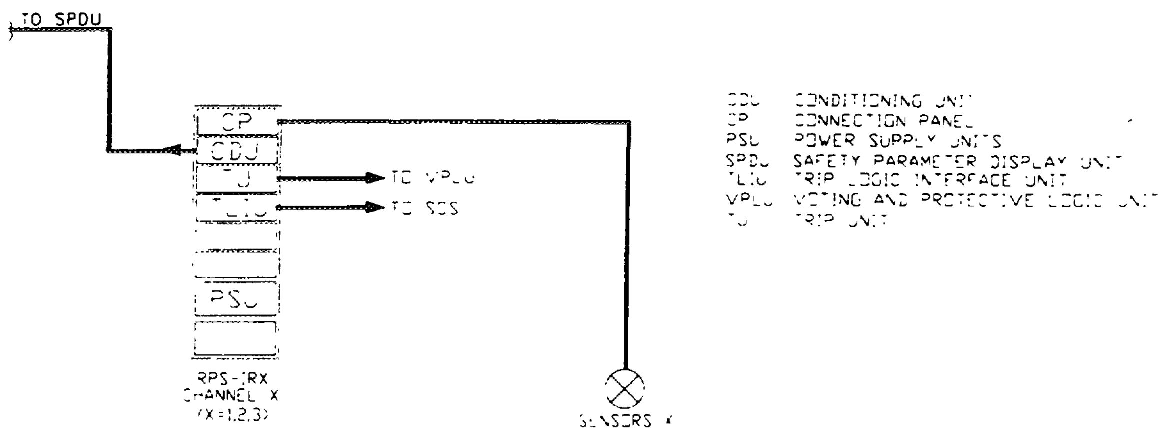

FIG. 3.4: Non-nuclear Instrumentation- Safety Signals

3.3.1.2 Safety related signals

Systems detailed below provide signals to the Supervision and Control System :

- Compressed air system instrumentation
- Reactor cooling system instrumentation
- Auxiliary pool cooling system instrumentation
- Secondary cooling system instrumentation
- Fire detection system
- Long term spent fuel element storage pool (water level)
- Experimental facilities (if any)
- Hot water layer system instrumentation
- Reactor water treatment system instrumentation
- Demineralized water system instrumentation
- Active effluents system instrumentation
- Supervised area ventilation system instrumentation
- Reactor hall ventilation system instrumentation
- Non restricted area ventilation system instrumentation
- Ventilation system water supply instrumentation
- Pneumatic transport system instrumentation

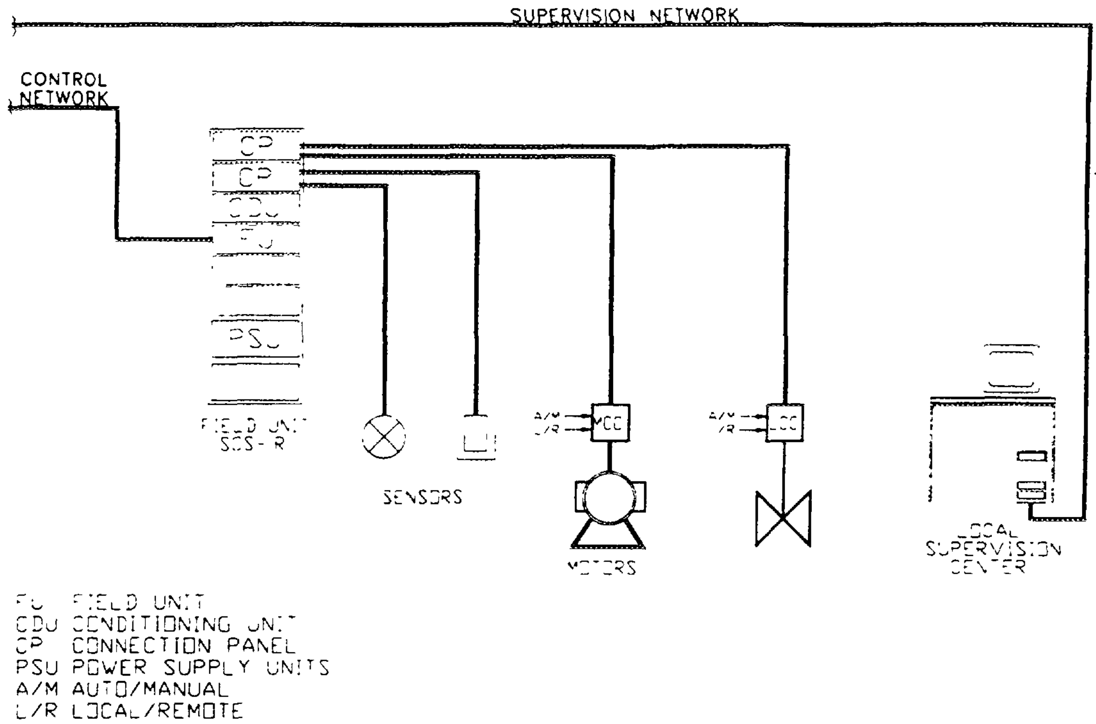

FIG. 3.5: Non-nuclear Instrumentation- Safety Related Signals

3.3.2 Goals

3.3.2.1 Safety goals

Triplicate sensors measure safety variables. These signals are sent through three physically and functionally separate channels, thus avoiding failures due to common cause.

In the Reactor Protection System these safety signals are compared to Safety System Settings to generate their corresponding trip signals, which are then processed in a 2 out of 3 coincidence logic.

This scheme allows to carry out tests and maintenance tasks in one channel at a time, falling back to 1 out of 2 logic, without losing system availability.

3.3.2.2 Control goals

Non-nuclear instrumentation includes manual or automatic actions that modify the working conditions of auxiliary systems . Most of the control loops are carried out by the Supervision and Control System, with some local controllers for ventilation and special equipments.

3.3.2.3 Supervision goals

After being measured, reactor and process parameters are monitored, recorded and displayed for supervision purposes through the Supervisory and Control System at the Main Control Room, the Secondary Control Room, the Training Room and the Local Supervision Centers for the Reactor Island, and at the Supervision Centers of other external process-related facilities.

3.3.3 Field level instrumentation description

The following operation modes are adopted for on-off field devices actuation:

- **Remote manual mode**

  The different on-off actuating devices distributed along the Reactor Plant can be individually commanded from the Control Rooms. For this purpose, a remote/manual selector is available in front of local panel.

  Local priority is always assured by hardware.

- **Local manual mode**

  In this mode on-off devices may be individually actuated from the field. For this purpose local command boards are available.

- **Automatic mode**

  The different on-off devices may be actuated in Automatic Mode when this option is selected by means of an AUTO-MANUAL selector available at the local command board.

3.3.4 On-off devices operation description

- **Motors control center**

  The Motors Control Center (MCC), is electrically connected to a Field Unit (FU).Auto/manual and local/remote selectors are available in front of the Motor Control Center panel. The Auto/manual and local/remote status is sent to the Supervision and Control System through the Field Units.

  Whenever local operation is desired, the final device may be operated by means of push buttons available in front of Motor Control Center panel. Local command priority over remote command is assured by design.

- **Field Units (FU)**

  The Field Unit of the Supervision and Control System receives and sends command and status signals from the Motor Control Center. Signal conditioning and analog to digital conversion is made at the Field Unit.

- **Local Control Centers (LCC)**

  On-off devices, other than motors, may be operated from a local control centers (LCC) which are located in the process rooms. Auto/manual and local/remote selectors are available in front of Local Control Center panel.

  The Auto/manual and local/remote status is sent to the Supervision and Control System through the Field Units. Whenever local operation is desired, the final device may be operated by means of push buttons available in front of panel. Local command priority over remote command is assured by design.

- **Local Supervision Centers (LSC)**

  A supervisory facility is available in three locations of the Reactor Island: ventilation system room, ground floor process room and auxiliary building. Each Local Supervision Center contains on operator interface with a video monitor, keyboard and printer, that can be used given the appropriate password authorization.

  Only supervision functions can be made at this centers. No control or command functions are available.

3.3.5 Process Instrumentation Description

3.3.5.1 Scope

As any industrial facility, research reactors and critical facilities have many process variables to be measured. Table 1 is a typical list of process variables measured in research reactors. What follows is a discussion of the available sensors used to measure process variable, mainly temperature, pressure, level and flow.

TABLE 1: LIST OF TYPICAL PROCESS VARIABLES MEASURED IN RESEARCH REACTORS

Reactor inlet and outlet temperature	Primary and secondary flow
Reactor core, ΔP	Water level
Reactor room underpressure	

3.3.5.2 Temperature Measurements

There are three main families of temperature sensors: thermocouples, resistance thermometers and thermistors. Some old reactors still have bimetalic thermometers (binary or with local readings), but it is suggested that they be replaced in order to allow the operator to follow closer any temperature transient from the control room. Other temperature sensors, like semi-conductor thermometers, consisting of doped germanium sensors, have a complex resistance temperature relationship and are useful only for very low temperature measurements.

3.3.5.2.1 Thermocouples

Thermocouples can be classified into two main types: noble metal and base metal. Noble metal thermocouples, like platinum and platinum-rhodium, are used for high temperature measurements, usually in the range of 600 to 1600°C. They are chemically inert and highly suitable in oxidizing atmospheres, however, it is important to notice that platinum and rhodium are very sensitive to gamma and neutron radiation (as seen elsewhere in this document, they are the recommended materials for Self Powered Detectors). Base metals, like chromel and alumel are used for temperature measurements in the range of 130 to 1100°C. Although they do not show the same linearity as noble metal thermocouples, and they have higher sensitivity as compared to noble metal thermocouples, they need less sophisticated electronics. Studies have shown that chromel-alumel thermocouples are most suitable for applications close to the reactor core, where the gamma and neutron fluxes are very high. When the sensors must cover a temperature range of as high as 1000°C it is recommended to use nicrosil-nisil type thermocouples, that have become popular in the field of high temperature measurements. Usually they respond faster to temperature transients than other thermometers, and should be considered when sensor response time is an important parameter to safety.

3.3.5.2.2 Resistance Thermometers

Resistance thermometers, which usually employ platinum elements, have very high sensitivity, linearity and precision. They can be used safely up to temperatures of 500°C, however, one should consider that they are sensitive to neutron and gamma radiation, and in general they have larger time constant then thermocouples, so this must be taken into account when they are employed in the reactor protection system.

3.3.5.2.3 Thermistors

Opposite to resistance thermometers which have positive resistance coefficient (resistance increase with temperature), thermistors have negative resistance coefficient. Thermistors have much higher sensitivities as compared to platinum resistance elements. They need simple processing electronics, which results in higher system reliability. Because of their small size, they also have smaller time constant than thermocouples, however, since they are highly non-linear sensors and because their susceptibility to radiation damage, as well as ageing, they are not recommend for applications in safety systems.

3.3.5.2.4 Thermowells

In order to make maintenance activities easier, thermowells are welded in the pipe (or tank) where the temperature is to be measured. Then the thermometer is installed in the thermowell, using some thermal material to make good contact between the thermowell and the thermometer. It is important to notice that the use of the thermowell increases the response time of the thermometer to changes in the temperature of the process being monitored. If this information is important for safety, one should consider the possibility to avoid the use of the thermowell, installing the thermometer directly in contact with the process.

3.3.5.3 Flow Measurements

In those reactors that use forced circulation to remove the heat produced in the core, coolant flow measurement is usually a safety parameter, since it gives a fast indication of degradation of the capability to remove the energy produced in the reactor. This enables shutting down the reactor faster than compared to coolant temperature measurements, besides, usually the temperature sensors are located in the outlet piping, and if the coolant flow is lost, the temperature of the coolant close to the reactor core will be very different from the point where the temperature sensor is located.

There are several ways to measure and monitor coolant flow. The existence or flow rate in a closed system may be deduced by measuring or monitoring circulating pump operation, when flow in the system is forced, however, since we can have pump cavitation, and some valve closing the flow channel, this method should only be used as a backup to direct flow measurement.

Flow measurement can also be deducted by correlating thermal noise in the flow channel. The basis for this technique is the measurement of transport time of the noise pattern (caused by local turbulence and bubble collapse) between two thermocouples located a few pipe diameters apart in the fluid. In this case accuracies in the range of 3 to 5% are possible. The use of ultrasonic transducers allows another possible correlation, and in this case no penetration in the pipe is necessary, and, over a limited range of flow, we can have accuracies in the order of 5%. Another possible correlation is the measurement of ^{16}N activity between two distinct points in the coolant system piping. This technique is actually under study, and can be used to attend the requirement of functional diversity for flow measurement.

In research reactors that use liquid metal coolant, flow can be measured either by use or an electromagnetic flow meter or by an ultrasonic flow meter. The electromagnetic flow meter generates a voltage proportional to the coolant flow, and the ultrasonic flow meter deduce the flow by measuring the sound transmission time between the transmitting and receiving rod pairs. The accuracy of measurement in both cases is around 10%.

Despite the several techniques of flow measuring, most of them are used just to validate the technique, and as a redundant way to assure that there is coolant flowing in the system, because the most common type of flow meters use differential pressure transmitters to measure the difference in pressure caused by orifice plates or venturi tubes placed in the channel. Accuracies of the order of 3% are easily achievable, and the instruments are simple and extremely reliable, therefore suitable for safety systems. It is also important to notice that, depending on the geometry of the piping of the cooling system, the differential pressure can be measured in other locations, like in an elbow type curve, where we have a high differential pressure between the inside part and the outside part of the elbow (this is a common way to measure flow in power reactors, with the advantage that we do not put a restriction in the cooling system). Another location commonly used to measure coolant flow is the reactor core itself. The reactor core is an restriction to the flow, and it causes a differential pressure which is increased by the column of water equivalent to the length of the reactor. The coolant flow is measured directly on the reactor core and this signal, known as core ΔP, is usually used in the reactor protection system.

When very precise information is needed a turbine flow meters is used. It consists of a turbine that is installed inside of the pipe line and that rotates as the fluid passes through it. An electromagnetic pickup, located on the outside of the pipe, indicates the angular velocity of the turbine, which is proportional to the flow rate. An disadvantage of turbine flow meters is that they are installed inside of the pipe, and since they are more complex than an orifice plate, they can cause more problems for maintenance.

3.3.5.4 Liquid Level Measurements

The most common method of measuring liquid level is to measure the differential pressure between the bottom of the vessel compared to the free vapor space above the liquid. A second technique is the bubble, tube method, wherein the pneumatic back pressure exerted on a bubble tube which bubbles gas through the liquid, is directly proportional to the liquid level. This method is not recommended for reactor pool water level, since the oxygen inside of the bubbles can be activated by the neutron flux.

Because many research reactors are old. they still use the primary methods of liquid level measurement which consists of a float switch actuated by a mechanical arm. or a magnetic reed switch actuated by a magnet imbedded in a float. Since many of the float switches use a glass bulb with mercury, to make electronic contact between two wires when the float rotates, and because this type of instrument is a binary instrument and does not allow the operator to monitor continuously the liquid level, it is recommend that installations using these instruments should be replaced with more modern and reliable instruments.

Other methods are available to measure liquid level. One of them employs an ultrasonic transducer located at the bottom of the tank which generates pulses and receives them back after a delay proportional to the liquid level by reflection from the interface between the liquid level and the air space above.

If the reactor has a closed cooling system, it is recommended to use a thermal conductance type level meter composed by an electric heater and a differential thermocouple, to measure the liquid level inside of the reactor vessel. This method does not have good accuracy to determine precisely the level position, however, it is the safest way to assure that the reactor core is covered by the cooling liquid.

Finally, in systems that use liquid metal, the level can be "detected" by a point contact resistance tube, or by inductive sensors

3.3.5.5 Pressure Measurements

Since most of the research reactors are open type reactors, usually only two values of pressure are measured, the "underpressure" in the reactor room, and the pressure in the pneumatic system. When the reactor has a closed cooling system, the pressure in the cooling system is also measured. Pressure transmitters are divided into two parts, the sensor (called primary element), and the conditioning circuit. The primary element is an elastic mechanical component which is used to transform pressure (or differential pressure) into displacement (proportional to the pressure), and the conditioning circuit is used to transform displacement into an electronic signal. Usually the transmitter is classified according to the principle used to convert the displacement into an electronic signal. According to this classification pressure transmitters can be classified as strain gage, force balance, capacitive, or wire resonant.

All of them are reliable, and can be used in research reactors, even in the safety system. What follows is a brief explanation of the different types mentioned above.

- Strain gage transmitters have strain gages located on both sides of a membrane. As the membrane deflects, under the action of pressure, one of the gages is stretched (its resistance increases), and the other is contracted (its resistance decreases). The use of a Whetstone bridge allows the measurement of the change in resistance, which is proportional to the displacement, consequently proportional to the pressure variation.

- In force balance transmitters the displacement caused by the pressure variation is mechanically propagated to a ferrite disc that is part of a differential transformer. The displacement of the ferrite disc generates a DC current which passes through a magnetic field, generating a "magnetic force" which tries to return the ferrite disc to its original position. The circuit is such that the DC current is proportional to the displacement of the ferrite disc, consequently proportional to the original displacement, and to the pressure variation.

- In capacitive pressure transmitters, the primary element is a membrane that has metal in its inner part, and that is a central plate of a two capacitor system. As the membrane displaces to one side, it increases the capacitance of one capacitor, and decreases the capacitance of the other, therefore, the capacitance of the system changes. There is a linear relationship between the capacitance and the displacement.

- Wire resonant transmitters use the principle that a stretched wire has a resonant frequency that is proportional to the length of the wire. With the use of an elastic material the length of the wire becomes proportional to the applied pressure, resulting in a resonant frequency proportional to pressure. Other types of transmitters exist, like inductive and piezoelectric, but because they are not very reliable, and present some maintenance problems, are not common in research reactors.

3.3.5.6 Other Process Field Instruments

Since some of the operational research reactors were constructed more than 30 years ago, it is recommend that surveillance instruments be installed to monitor in primary coolant pump. Inductive accelerometers and displacement sensors are widely used in nuclear power plants, and in the industry in general, and have been proven to be reliable. By monitoring pump vibration, one can anticipate failure conditions and make the necessary repairs well in advance.

In regions with seismic history, accelerometers shall be used to detect any seismic activity, and shutdown the reactor when necessary.

The use of on-line conductivity meters is recommended to monitor the water quality continuously, however the information must be complemented with periodic off-line analysis, performed at appropriated laboratories. The analysis must consider conductivity, pH and a detailed chemical and radiochemical composition.

3.3.5.7 Process Instrumentation Signal Conditioning Systems

Since research reactor do not have many process variables connected to the reactor protection system, it is recommend that those variables that are part or the safety system should be independent, i.e., it has its own conditioning unit with respective power supply. Microprocessor based systems can be used, but only for those variables, and functions that are not safety related.

3.3.6 Conventional Measurements Description

3.3.6.1 Core Coolant Inlet Temperature

Core inlet coolant temperature is measured with triple redundancy at the inlet coolant collector.

Detectors used are DIN 43760 "A" Class Pt-100. Maximum allowable deviation according with DIN 43760 is of +/- 0.13 ohm.

Each core inlet coolant temperature measurement assembly includes two Pt-100 detectors. The first detector is used for inlet coolant temperature measurement. The second detector is used for core coolant differential temperature.

Pt-100 temperature detectors are connected to RTD and to current converters. These converters provide a 4 to 20 milliampere output signal. The current signals provided by the RTD detectors are wired to the Reactor Protection System.

3.3.6.2 Core Coolant Outlet Temperature

The core coolant outlet temperature is measured by three detectors, of 43760 "A" Class Pt-100. Maximum allowable deviation according with DIN 43760 is of +/- 0.13 ohm.

Each core outlet coolant temperature measurement assembly includes two Pt-100 detectors. The first detector is used for outlet coolant temperature measurement.

The second detector is used for core coolant differential temperature.

Pt-100 temperature detectors are connected to RTD and to current converters. These converters provide a 4 to 20 milliampere output signal. The current signals provided by the RTD detectors are wired to the Reactor Protection System.

3.3.6.3 Core Coolant Differential Temperature

Detectors used are Pt-100 thermoresistances. Core outlet coolant temperature for differential temperature is measured at the control absorber guide box level. Three redundant measuring channels are used for differential temperature measurement.

3.3.6.4 Core Coolant Differential Pressure

Differential pressure through the core is measured by means of three redundant differential pressure transmitters.

Transmitters used are of the capacitive sensor type and develops a 4-20 milliampere current output.

Maximum inaccuracy due to linearity, hysteresis, temperature effects, pressure variation effects or other effects must be equal or lower than 0.2% of full span.

Parts in contact with reactor primary water are of stainless steel. Body material is plated carbon steel.

3.3.6.5 Reactor Tank Water Level

The following discrete water levels are detected by means of Single Pull Double Throw (SPDT) contacts output transmitters:

- Pool water level very high
- Pool water level low.
- Scram level (three redundant detectors)
- Evacuation level (three redundant detectors).

3.3.6.6 Primary Circuit Flapper Valves Position

Flap valve position is detected by means of duplicated detectors. Detectors used are of the reed-switch type. Closed valve position is detected by two detectors.

3.3.6.7 Auxiliary Pool Water Level

Overflow and reposition level are detected both in the reactor tank and auxiliary pool.

3.3.7 Ventilation System Instrumentation Criteria

3.3.7.1 Scope

Instrumentation criteria stated here will be used for Reactor Island controlled area instrumentation, non-controlled areas and other services where automatic ventilation systems are required.

3.3.7.2 Ventilation System Instrumentation Architecture

The ventilation control system is based on distributed acquisition and control units, which are all connected to a ventilation central unit. This central unit concentrates all ventilation system parameters, process alarm status, logic interlocks and sequential controls related with automatic functioning. The ventilation central unit communicates with the Reactor's Supervision and Control System, which receives all supervision information and may execute remote commands on the ventilation system.

Injection and extraction units, such as valves, dampers or hermetic seal flapper valves, are actuated through a system-wide interlock logic, to assure correct total system behavior in any required operational condition.

Manual operation of the principal components such as ventilators, dampers and hermetic seal flapper valves is available so to allow maintenance and start -up operations.

3.3.7.3 Field Sensors

The ventilation control system takes inputs from analog sensors and switches. The analog sensors are used for measurement of:

- differential pressure
- temperature
- humidity
- air flow

In addition , field switches provided status information on:

- damper position
- valves position
- motors parameters status

3.3.7.4 Local Indicators

Local indicators will be located in the field to allow verification the general status of the different variables needed for local operation, such as differential pressure, temperature, air flow and others.

Local differential pressure U-type manometers are provided to evaluate filter status.

Cooling units are instrumented by bourdon manometers and local temperature indicators.

3.3.7.5 Valve Actuators

Valve actuators will be electrically actuated. In cases when higher actuation power is needed, high pressure air-actuated pistons will be used.

3.4 Control Rod

3.4.1 Control Rod Drive Mechanisms

Most research reactors are controlled by the insertion of a neutron absorber (control rod) into the reactor system or are controlled by the adjustment of the moderator level. Control rods require linear mechanisms which should be fail-safe, reliable, fast acting and radiation resistant.

A control rod drive mechanism basically consists of a drive motor coupled through gearing and a electromagnetic holding device to the neutron absorber or reflector. On the receipt or a scram signal, the electromagnet de-energizes, thereby allowing the control element to be inserted into or out of the reactor core under gravity to introduce negative reactivity. The insertion by gravity is often assisted by a spring to obtain the initial acceleration to reduce the insertion time. The delay between the receipt of a scram signal and the actuation of the device is dictated by the release time of the electromagnet as well as the initial accelerating force provided by the backup spring. The electromagnetic delay time can be minimized by providing suitable air gaps, employing a laminated core, as well as by the provision of suitable resonating circuits to bring the magnetizing current to zero very fast without causing damage to the coil. The facility for partial release of neutron absorber for on-line testing purposes is being currently provided in some reactors. This is done by temporarily reducing the current supplied to the rotary electromagnetic clutch for one rod at a time.

The movement of the neutron absorber is suitably damped by a shock absorber at the end of travel to prevent damage to the control rod. Vane type rotary hydraulic dashpots with oil reservoir are commonly used as shock absorbers. Double vane can be used to balance forces on dashpot shaft arising from high oil pressure during damping action. Detailed theoretical analysis as well as experimental investigations are necessary to optimize the rod drop dynamics.

The speed with which the control rod is driven in the direction of increasing reactivity is limited by proper choice of gearing between the drive motor and the linear drive. Of particular interest is the pneumatic pulse rod system for pulsing TRIGA reactors. These rods have rapid withdraw capability by the use of a piston assembly and the application of higher pressure air to the piston.

Since the reactivity of control rods is non-linear it is possible to incorporate programmed drive speed arrangements to reduce startup time of the reactor. This involves drive electronics, and one has to ensure that the failure of the electronics does not lead to an uncontrolled rate of reactivity addition.

Depending on design, shutdown rod drives for research reactors should be operated in a proper sequence so as to minimize the rate of reactivity addition.

Different designs for converting rotary motion of a motor to the linear motion of the neutron absorber are in existence. These include rope and pulley mechanisms, rack and pinion drives and screw and nut mechanisms with split nuts for scramming. Use of linear induction motor or a linear synchronous motor eliminates use of a rotary prime mover thereby providing more reliability. However, these mechanisms are larger in size for the same load capacity and are not very popular for reactor applications. Rope and pulley mechanisms permit flexible layout of drive mechanisms with respect to the neutron absorber and can be accommodated within a very small space. While using rope and pulley mechanisms, proper tools are necessary for retrieval of neutron absorber assembly subsequent to incidental rope snapping.

Full scale life-testing of control rod drive mechanism is necessary to establish reliability. Microprocessor based data acquisition systems are being employed during testing to facilitate condition monitoring and to predict maintenance requirements.

Depending on application, seismic qualification of control rod drives may become necessary.

3.4.2 Control Rod Position Indication

The position of a control rod or a shutdown device should be known at all times of reactor operation to ensure the availability of the device when a scram command is given. Limit switches which indicate that the shutdown device is in the maximum reactivity position will meet all normal requirements. Direct indication of the shutdown element position has been successfully obtained by the actuation of limit switches by push rods moved by the element itself. Magnetic reed switches can be also used for this purpose. Redundant limit switches are normally used for reasons of reliability. Synchos, encoders, potentiometers and RDVTs can provide additional information on intermediate position of the shutdown device. A tacho-generator can provide useful information on the velocity of shutdown device during scram.

3.4.3 Magnet Power Supply

Most of research reactors use gravity to insert the control elements when fast shutdown is required. The electromagnetic clutch is supplied with electric current by the magnet power supply, which also acts as a fast cut-off device when a scram is initiated by the protection system.

The magnet power supply is considered in many designs as the major safety device, since it is a junction point, in which all protection systems and logic units converge to perform the safety act of reactor shutdown.

Besides reliability, "fail-safe" design and all other standard safety requirements, the magnet power supply shall also meet the following requirements:

- fast response to a trip signal. The required cut-off time of the magnet power supply is based on the safety analysis of a postulated reactivity accident,

- Independence between the safety circuits of the individual electromagnets.

In order to avoid spurious trips, the magnet power supply should be connected to essential or guaranteed electrical boards, which may be provided with redundant power sources to achieve the required reliability.

3.5 Trip And Test Units

3.5.1 Trip Units

Trip units provide a positive trip signal with a fail safe feature when a certain measured parameter level goes beyond the preset range. The trip level should be continuously variable with a provision for locking after the levels are set.

Available technology offers a minimum of four types of safety logic systems, based on diverse logic elements, for use in reactors. The relay logic which has been tried out extensively in the safety systems of many research and power reactors has established its reliability and availability beyond doubt. However, relay logic systems are bulky, consume large amounts of power, are susceptible to environmental conditions, and are very slow acting, requiring several milliseconds from the receipt of the scram signal to the provision of tripping action. Though this is an apparent disadvantage, the delay serves very effectively to filter out fast spurious signals. Relay logic can also cause electromagnetic interference when many relays open simultaneously.

Hardwired solid state logic is the second alternative for a trip system. The use of discrete chips in combined logic form will result in a system which is effective, reliable and consumes much less power consumption than relay systems. Triplication of the system enhances the reliability. However, precautions have to be taken at the gate level, to ensure that scram signals are not generated by spurious signal pick-up in the system. To avoid spurious trips, circuits using semi-conductor electronics need adequate noise immunity and built-in hysteresis. Currently available semi-conductor technology, though extremely reliable, is often backed up by output relays to doubly ensure the safety of the reactor.

Programmable solid state logic, the third alternative, has not been used extensively in research reactors so far. However, the rapid development of computer controlled systems will soon result in the implementation of solid state logic systems for reactor applications. The use of programme controlled logic cards are standard items in industry and integration of these cards in the required fashion will provide a suitable logic system for research reactors. As for other cases, the programme controlled logic will have to be triplicated to ensure reliability.

Finally, the use of duplicated or triplicated microprocessor systems for the safety channels of a research reactor is likely to be effective and economical. However, microcomputers have not been tried out in large numbers in reactor systems. Their introduction into safety systems poses many questions regarding reliability and unsafe failures, and as reported, when used, have presented some operational problems.

Considering the many problems not yet solved which have been shown to exist in relation to existing software, it seems that the software reliability problem will only be alleviated through improvements in procedures employed prior to the testing phase. Essentially, no regulations or approved industry standards exist that give guidance in applying such a procedure to computer based systems used in research reactor protection systems.

In all safety systems using semiconductor logic, or semiconductor components, unsafe failure cannot be ruled out. It is therefore necessary in such systems to provide continuous reliable impulse testing to monitor performance of the system.

3.5.2 Test Units

The test units provide signals for the testing of the complete electronics of the safety channels. Highly stable voltage sources are readily available and are more than adequate for the generation of accurate calibration signals. The Campbell channels are tested with sine wave signals with harmonic distortion limited to 1%. The integration of the test units in the safety electronics should take into account the availability of the safety system when a trip channel is being tested. When a trip channel is being tested in a two out of three system, the other two channels should be active, for protection of the reactor.

3.6 Displays, Alarms and Data Recorders

3.6.1 Displays

Computer-based systems are well suited to handling complex information gathering, analyzing and display tasks. However, it is often difficult to justify their integrity for use in high reliability safety systems. Computer-based systems may be used for automatic reactor control purposes to good advantage, but hard wired systems are to be preferred for safety duty (e.g. alarms or protection) because of their easier (and therefore cheaper) qualification for such applications. The use of a mimic diagram including the principal components of the installation is recommended.

3.6.2 Alarms

Audible and visual warning devices may be very useful for alerting the operator that a safety level is being approached, or that some action is required. The radiological monitoring systems are typical examples. It is customary to have such warnings both, at the reactor control desk and in the reactor confinement building.

It is generally desirable to have audible and visual alarms preceding the trip level on any safety system parameter. These should enunciate in the control room since this alerting signal may give the operator time to perform whatever remedial procedure is available to him to alleviate the need for a reactor trip. Warning signals are often given on power level, reactor period, high radiation level, safety devices of experiments, and any safety device concerned with a critical assembly within a reactor or an experiment.

It is not necessary to require safety system redundancy (duplication) in alarm systems, however, it is necessary to keep reliable alarm (warning) systems, by the use of appropriate buffering systems if necessary. Alarm systems can not compromise the functionality of safety systems.

3.6.3 Data Recorders

It is desirable to have recorders to record certain safety-related conditions of the reactor. It should be noted that any actions which may be initiated by data recorders will not be available if the recorders are not in operation; and, in general, it is considered unwise to depend on recorders for safety action. Good administrative control is required to ensure that they are operating during any event.

Data recorders, or loggers, should be considered to be only safety related systems at best and viewed as operator aids, rather than systems qualified for safety duty. Data logging systems may be analogue or digital, and modern computer based systems can provide a comprehensive data analyzing and display facility.

3.6.4 Control absorbers instrumentation

Control absorbers instrumentation includes safety signals and safety related signals.

Safety signals are detected by means of three redundant channels and are sent to the Reactor Protection System. Safety related signals are sent to the Supervision and Control System.

3.6.4.1 Safety Signals

The following parameters are measured:

- Control absorber locked to control assembly electromagnet
- Control absorber in bottom position

3.6.4.2 Safety related signals

The following parameters are measured and controlled:

- Continuous control absorber position
- Step motor electrical drive signals
- Control absorber in top position

Continuous control absorber position is detected by a Helipot type resistance.
The Helipot device is directly coupled to the control absorber mechanical driver.
A continuous signal is developed and sent to the Supervision and Control System.

3.6.5 Seismic instrumentation

Seismic Instrumentation comprises seismic switches for safety purposes and seismic intensity meters for post-event evaluation.

3.6.5.1 Seismic switches

The Reactor Protection System includes three independent triggering channels, which are based on triaxial seismic switches. The trip signals are the direct result of the seismic switches output, which have setpoints according to their location in the reactor building.

The effect of the seismic signals is to initiate reactor SCRAM, and are also indicated in the Supervision and Control System.

3.6.5.2 Peak accelerometers

In the event of an earthquake occurring at the site, two peak accelerometers are provided to enable post-event decision making. The maximum local acceleration in each of the three axis is recorded, and provides indication of the magnitude of the event.

The accelerometers locations are preliminary defined at basement level, to obtain indication of ground movement, and at reactor hall, where amplification effects may be apparent by comparison.

4. REACTOR PROTECTION AND POWER REGULATING SYSTEMS

The Nuclear Instrumentation comprises the Neutron Flux Instrumentation and a Nitrogen-16 Gamma Channel. Neutron flux measurement instrumentation supplies information on physical parameters to the Reactor Protection System and the Reactor Power Regulating System, describing the state of the reactor regarding neutron production and its evolution.

The neutron flux measurement system comprises three independent measuring channels based on fission counters, a wide range fission chamber and compensated ionization chambers which are able to keep track of neutron flux for the whole operation range of the reactor. There is an overlap between the channels, as well as an adequate margin over Full Power (FP) for the follow-up of eventual power excursions (see Fig. 3.1).

The Reactor Protection System (RPS) includes two separate channels to cover the neutron excursion over the whole operation range of the reactor (from source level to 125% of full power level). These channels are:

- **Start-up Channel.**
- **Power Channel.**

In order to comply with safety and reliability requirements, each channel type of the Reactor Protection System (startup channel and power channel) is constituted by redundant channels entirely independent from the detector up to the electronic measuring modules.

Under normal conditions the Reactor Protection System operates with the required redundancy. Should a failure occur in one channel, the resulting signal at the end of that channel is coincidental with the corresponding signal requiring a protective action in order to reduce as much as possible the required coincidence over the rest of the system. ("Fail Safe" criteria).

The Reactor Power Regulating System (RPRS) controls reactor power from source level to 125% of full power level, using neutron and gamma instrumentation ·

- Wide Range Linear Channel
- ^{16}N Linear Channel.

4.1 Start-up Channel

The Start-up Channel consists of three identical operating channels. The channels consist of several entirely independent detectors and electronic measuring modules designed to comply with the required reliability and safety criteria. Under normal conditions, the system thus operates with the required redundancy.

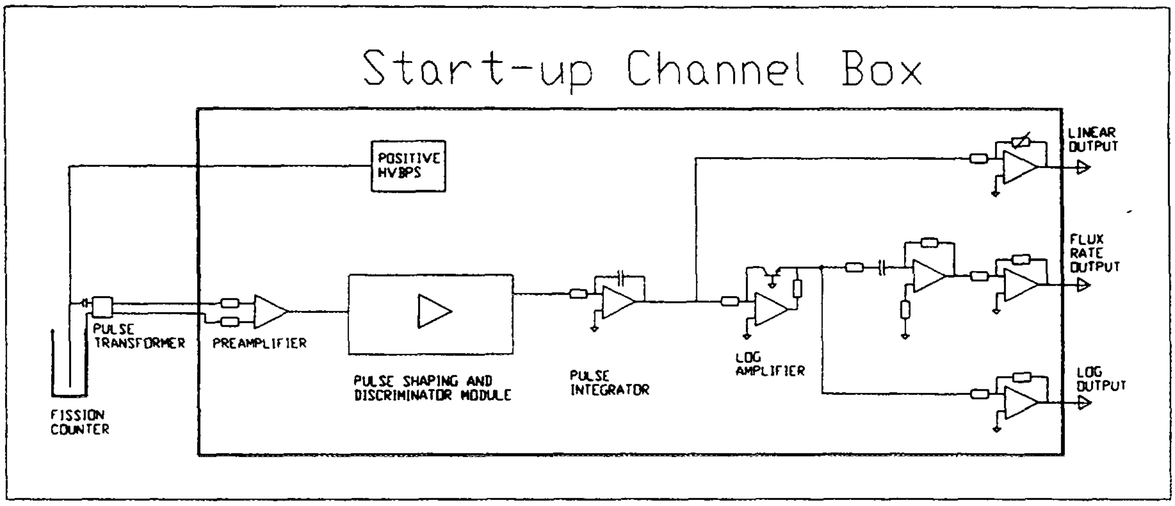

FIG. 4.1: Start-up Channel

The Start-up channel has both log and linear outputs over its whole operating range. This channel is fed into the Reactor Protection System and also into the Reactor Power Regulating System through a functional and electrical isolated interface that assures absolutely independence of the Reactor Protection System.

4.1.1 Function

The Start-Up Channel measures the evolution of neutron flux from the earliest stages of reactor operation (neutron source level), up to 5 measurement decades below full power.

In order to encompass its vast range of measurement, each start-up neutron detector can be move upward and downward to regulate its height relative to the reactor core. During reactor start-up, as the neutron power increases, the detector is lifted before saturation is reached. When operating in the Start-up range and in power reduction operation, the detector is lower when the counts per seconds reach a lower limit. For a given detector position, measurements are performed from source level (3 to 100 cps), up to 30000 cps.

4.1.2 Design Criteria

- The channel is capable of reading neutron flux variations from +14 %/sec to -6 %/sec.
- Sensors have low sensitivity to gamma radiation.
- Sensors and related elements are electrical insulated from their containers.
- Sensor-originated signals have sufficient level and are transmitted by preamplifiers located beyond the radiation field of the core, with minimal interference.
- The fission chamber's containers are mounted on trolleys and displaced under remote control from the pool open-end. The detector positions are varied in accordance with the reactor's operating level.
- The correct functioning of each channel can be monitored from the Control Room.

4.1.3 Neutron Detectors

Fission Counters are used as detectors in start-up channels. They are housed in containers from which they are electrically insulated.

Fission signal cables, as well as all other related elements in the high exposure area, remain unaffected by radiation.

4.1.4 Neutron Source

A neutron source of Am-Be will be provided for the start-up of the reactor to assure a minimum count rate at the Start-up channel detector positions.

4.2 Power Channel

The Power Channel consists of three identical channels. The channels consist of several entirely independent detectors and electronic measuring modules designed to comply with the required reliability and safety criteria. Under normal conditions, the system thus operates with the required redundancy.

The Power Channel has a log output over its whole operating range. In addition a linear output is available in the last decade to provide a better precision at full power operation.

4.2.1 Function

The Power Channel measures the reactor's neutron flux from 5 decades below full power to 125% of full power level.

4.2.2 Design Criteria

- Capability to measure neutron flux variations from +14 %/sec to -6 %/sec.
- Sensors housed in containers, inside the reactor tank, with electrical insulation between sensors and containers.
- Sensors have low sensitivity to gamma radiation.

The correct functioning of each channel can be monitored from the Control Room.

4.2.3 Neutron Detectors

Compensated Ionization Chambers with ^{10}B as converter and gamma compensation are used in the power channels.

These detectors have good neutron sensibility and low sensibility to high gamma levels. This is achieved by electrical compensation within the chamber itself (about 1-2%).

The current to be used lies in the range of 10^{-10} to 10^{-3} A.

Nuclear-used insulate cables are used for signal extraction and high voltage extension.

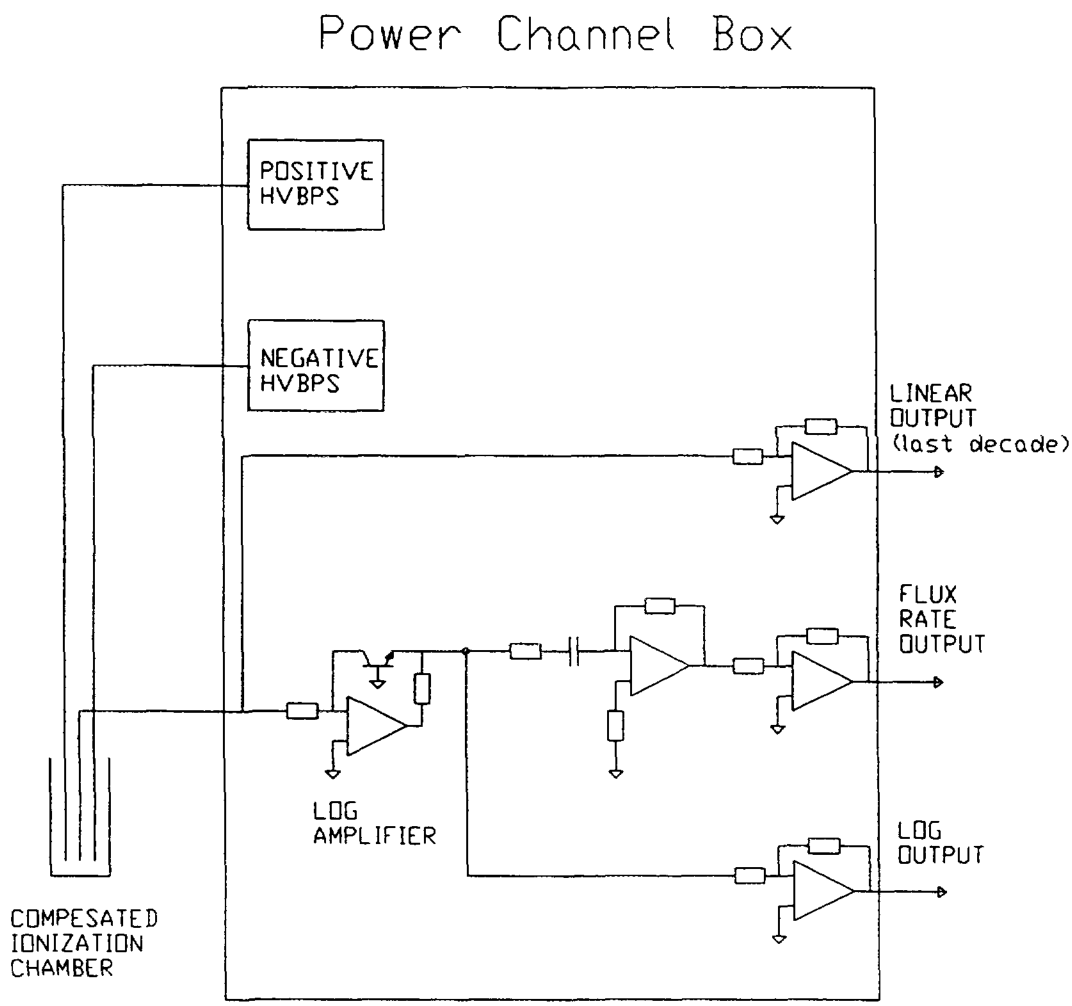

FIG. 4.2: Power Channel

4.2.4 Detector Containers

Containers prevent environmental effects from weakening the detector's insulation and shield them from the intense gamma radiation field. The design criteria are the following:

- Detectors housing are aluminum and other materials compatible with high radiation levels.
- The container-sensor-cable ensemble can easily be separated for replacement or maintenance purposes.
- If necessary, sensors can be withdrawn from the reactor pool for replacement or maintenance tasks. Seals and appropriate couplings are provided.

The whole assembly is tight to prevent moisture leaks.

4.3 Wide Range Linear Channel

The Wide Range Linear Channel (WRLC) consists of an unique neutron flux measuring channel. The channel provides a measure of the neutron flux over 10 decades combining the outputs of two separate Pulse and Campbell processing modules that are feed with a wide range fission chamber.

The Wide Range Linear Channel have both linear and log outputs over its whole operating range.

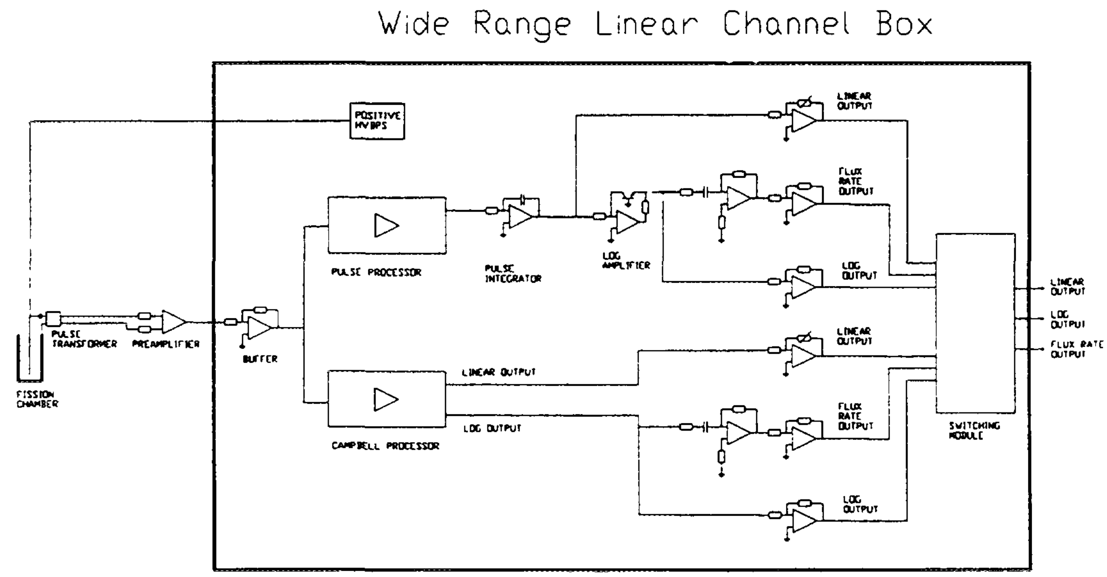

FIG. 4.3: Wide Range Linear Channel

4.3.1 Function

The Wide Range Linear Channel measures the reactor's neutron flux from the source level to 125% of full power level. It provides the neutron flux evolution to the Reactor Power Regulating System (RPRS) to allow an automatic control of the reactor power.

4.3.2 Design Criteria

- Capability to measure neutron flux variations from +14 %/sec to -6 %/sec.
- Sensor housed in a container, inside the reactor tank, with electrical insulation between the sensor and the container.
- Sensor-originated signals have sufficient level and are transmitted by a preamplifier located beyond the radiation field of the core, with minimal interference.
- Low sensitivity to gamma radiation of the measurement channel.
- The correct functioning of the channel can be monitored from the Control Room.

4.3.3 Neutron Detector

- A Fission Chamber suitable for operation in both Pulse and Campbell modes is used as the detector in the Wide Range Linear Channel.

4.4 Nitrogen-16 Linear Channel

A ^{16}N Gamma linear channel is used for global core power measurement, for use in the Rector Power Regulation System. This channel measures the high energy gamma radiation of the ^{16}N produced by the transit of the primary coolant through the core. As the coolant activation is directly proportional to the spatial integral of the neutron flux, is its an indirect global measure of the core power. This measurement includes a delay due to coolant transit from core to detector. The power regulation can then be achieved using this global measurement with lower error than with spatially localized neutron detectors. This channel complements the WRLC as input channel to the RPRS.

4.4.1 Function

The ^{16}N Channel measures the gamma activity of the primary coolant . with linear output to be used covering the range from 0.1% to 125 % of full power level. It provides the output signal to the Reactor Power Regulating System.

4.4.2 Design Criteria

- Capability to measure ^{16}N gamma flux in the range 10^{-5} Sv/h to 10 Sv/h, with a resolution better than 10^{-5} Sv/h.
- Sensor housed in a container, parallel to primary coolant pipe outside the reactor tank at dry location.
- Auto-range linear output over 6 decades.
- Sensor signals output transmitted to Regulating Channel Field Unit.
- Self-test capability at Field Unit level plus channel tests performed by the Supervision and Control System.

4.4.3 Gamma Detector

A sealed high pressure gas ionization chamber is used as gamma detector, with linear current output over 6 decade range.

5. MODIFICATION OF I&C SYSTEMS

5.1 Need for Modifications

The majority of research reactors operating today were put into operation some 20 years ago, some of them undergone modifications and refurbishing since their construction and many have been upgraded in power to meet requirements for higher neutron fluxes with or without any changes to their I & C systems. A number of old reactors are still operating with some parts and components of the original system. Old I&C systems cause operational problems as well as difficulties in obtaining replacement parts. In addition, there are the increasing demands of safety requirements.

Technical advances in I & C systems have been rapid in the past years, and this technology should be adapted by the research reactor community.

The introduction of improved safety instrumentation on existing and future nuclear facilities is a continuous and evolving process. For the old facilities, the main objectives is to provide improved systems to be consistent with modern safety standards, to cope with equipment obsolesce and to permit improved economical paramcters.

Several technical requirements must be fulfilled while making any modifications or modernization of I & C systems. These are:

- safety requirements,
- operational requirements,
- budget constrains (including minimization of outage time).

The safe, reliable and economical operation of existing research reactors must be ensured and therefore, careful consideration must be given to the modernization of the old I & C systems.

5.2 Licensing Requirement: Safety Assessment of Safety System

Changes to or upgrades of I & C systems which are important to safety, especially to safety systems, must be proposed in writing and verified by the licensee before such modifications are implemented. Changes to non-safety I&C systems may be made at the discretion of the reactor operators, provided that the licensee agrees that there will be no impact on safety.

Results of a safety review should meet recent regulatory safety criteria. It is also a requirement that the best available methods and tools should be used for the safety assessment.

Some regulatory authorities requires a periodic safety assessment of the system no matter whether any modifications have been made. For this case the best available assessment techniques should be used for the old system, and fulfillment of recent criteria should be proved by the new technique.

In order to comply with requirements of licensing authorities a thorough safety assessment is needed. This should confirm with the basic objectives as is described in IAEA Safety Standard, Safety Series No. 35-S1 (Code on the Safety of Nuclear Research Reactors: Design):

> "The safety of the reactor shall be analyzed and evaluated to demonstrate, that it is adequate - the results of the safety analysis shall be reflected in the Safety Analysis Report".

The evaluation of the safety of a reactor includes analysis of the reactor to a range of postulated initiating event, such as: malfunction or failures of any components, operator error, external events which could lead either to anticipated operational occurrences or to accident conditions. Unavailability limits for certain safety systems (or components) can be established to ensure the required reliability for the performance of safety functions. Main considerations for such an analysis are as follows:

5.2.1 Consideration of the Human Factor

Historically the focus of reliability analysis has been on component failure or success. This neglects the fact that in most systems human interaction and interface with the equipment is an important and often critical element in the overall success or failure of that system. Consequently, the system can only be modeled accurately if the human element is considered.

The inclusion of human factors in nuclear reactor system reliability analysis is a fairly recent concept. As such, the methodology and data available are not as refined as those that are used to model the contributions of equipment and components to system reliability.

5.2.2 Qualitative Analysis

Qualitative reliability analysis are used to identify possible ways in which a system can fail. The calculation can result in all combinations of components and human failures that lead to safety (protection) system failure, which prevents the safety system to shut-down the reactor upon request. For this analysis a top-down logic model, known as a Master Logic Diagram (MLD), similar to a fault-tree, could advantageously be used with a top event of a safety system failure upon request. The results of this analysis can be used to prove the fulfillment of the important single failure design criteria.

5.2.3 Quantitative Analysis

The quantitative analysis uses what is known or assumed about the failure probability of individual component parts and failure characteristics of the system to predict the failure probability of the system. A mathematical model for system success or failure is used as a function of failure rates, repair rates, test intervals, mission time, system logic and surveillance test schedules. In practice, the validity of quantitative results may be limited by the quality and quantity of the data; however useful comparative result and sensitivity analysis do not depend on the availability of extensive data. The protection systems of nuclear facilities are designed to satisfy reliability requirements. Targets are laid down to satisfy requirements for frequency of any single accident leading to an uncontrolled release. From the knowledge of the expected frequency of the initiating events, it is possible to arrive at designed targets for the protection system.

There are several reliability calculation techniques for hardwired system of any complexity. Computer codes are available using fault-tree analysis, and quantitative figures can be obtained for the safety system if reliability data for components are available. Techniques for modeling Common-Cause Failures are available in these codes; sensitivity analysis for components failure is included, and conclusions for fulfilling single failure criteria and sensitivity to human failure can also be drawn.

Documents that are basically tutorial and have been prepared to provide the user with the basic principles that are needed to conduct a reliability analysis of safety systems are: "Guide for General Principles of Reliability Analysis of Nuclear Power Station Safety Systems", ANSI/IEEE, Std 352, IAEA Safety Series No. 35-S1 and IAEA Safety Series No. 35-G1.

By applying the principles, systems may be analyzed. results may be compared with reliability objectives, and the basic for decisions may be suitably documented.

5.3 Safety System Containing Software

Work has been in progress on the digitalization of reactor protection system. Complete digital systems for 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 application.

This new generation of systems make 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 overall increase of safety. However, these favorable 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 requirement 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 device 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.

The main safety concerns are as follows:

- **Software is never error free;** its main cause is design error.
- Full testing of any software of medium complexity is not possible (the goal of testing is to find errors in the software, and not to show that the software is correct). As software is not subject to degradation through wear (as are mechanical and electrical components) the reliability depends upon the type and number of undiscovered errors remaining in the developed system. These errors may remain undetected for long periods of time until the right input conditions exist for them to be resolved. The very large number of possible test cases needed to exhaustively test means, that exhaustive testing is not practicable in all but the most trivial of cases.

- **Principle of redundant components does not apply**, as all copies of the same software would contain the same design error. There is possibility for a common cause failure if the exact design package is installed in each redundant channel and train of safety systems. It is possible that there exists a failure that could occur at the same time and could fail in the same manner in multiple safety trains. Such a failure would effect the ability of the safety system to perform its intended function. Furthermore, since computerised system design uses common information highways and/or can handle multiple input functions, a single computer failure in one train could effect a number of available trip functions, thereby reducing the availability and functional diversity of the existing analogue design.

However, it should be noted that there are great potentials for nuclear safety enhancement, if computerised safety systems use is applied correctly, through well designed, engineered, installed and maintained systems. The effects of the concerns discussed can be reduced to acceptable levels, provided the principal factors for defence against these concerns are addressed. Principal factors are:

- acceptable provisions should be taken to ensure that **high software reliability** is achieved. This can be performed by a Validation and Verification procedure.
  Since programming errors may propagate, errors during development can be a serious problem for computerised system. This can be reduced to a problem of how to achieve the reliability of software through the development process. The idea is that it might be easier to review and validate the various phases of the development process, which in the software case is a purely intellectual process, then validate the end product with all its complexity when it is complete.

- Another provision can be to provide a **diverse backup** (without software). The backup system does not need to cover the whole of the primary computer function.

- **Combination** of the above two methods.

A number of national and international standards dealing with the production of software for systems important to safety and also for systems not important to safety are being prepared. However, there are the formulation of requirements and the software engineering methods for meeting those requirements which make it very difficult to establish fitness for purpose of software. These are areas of active technical development. Perceptions of these problems are becoming clear, and ways of solving at least some of them are gradually emerging. In this context reference is given to the IAEA document "Software Important to Safety in Nuclear Power Plants", Technical Reports Series No. 367.

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Annex

PAPERS PRESENTED AT THE MEETING

XA9745519

EXPERIENCE, STATUS AND FUTURE OF THE COMPUTERIZED REACTOR INSTRUMENTATION AT THE TRIGA REACTOR VIENNA

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Abstract

The paper describes the 33 years old history of the instrumentation of the TRIGA reactor Vienna and focuses on the present computerized instrumentation installed in 1992. The experience of three years of operation is discussed and some of the failures are analyzed. Potential future problems both with soft- and hardware as well as with spare part supplies are analyzed.

1 HISTORY OF THE TRIGA VIENNA INSTRUMENTATION

The TRIGA reactor Vienna initially went critical on March 7, 1962 with an instrumentation supplied by the vendor, General Atomics. The reactor instrumentation fulfilled the requirements of the late '50 and early '60, being an electronic tube type system with a double pen recorder for linear- and log power. The block diagram is shown in fig.1. This instrumentation was used until 1968 when it became obsolete and was replaced by a transistorized instrumentation supplied by AEG / Germany. Again it fulfilled the safety requirements of the late '60 and early '70. It consisted of a hard- wired, relay type system with a console and a separate electronic cabinet (fig.2). The system worked properly until 1992, but in the late '80 a lack of spare parts due to ageing of components became obvious. Negotiations for a new instrumentation started in 1990 and bids from General Atomics, Siemens and an Hungarian company were received. The contract was awarded to General Atomics as they offered a complete instrumentation covering both steady state- and pulse operation. The installation took place during six weeks in summer 1992.

2 PRESENT STATUS

The original instrumentation design proposed by GA (fig.3) was considerably modified to meet the stringent requirements of the licensing authorities, the final design is shown in fig.6. Although one wide- range channel is microprocessor controlled, it is only considered as an operational channel without safety relevance.

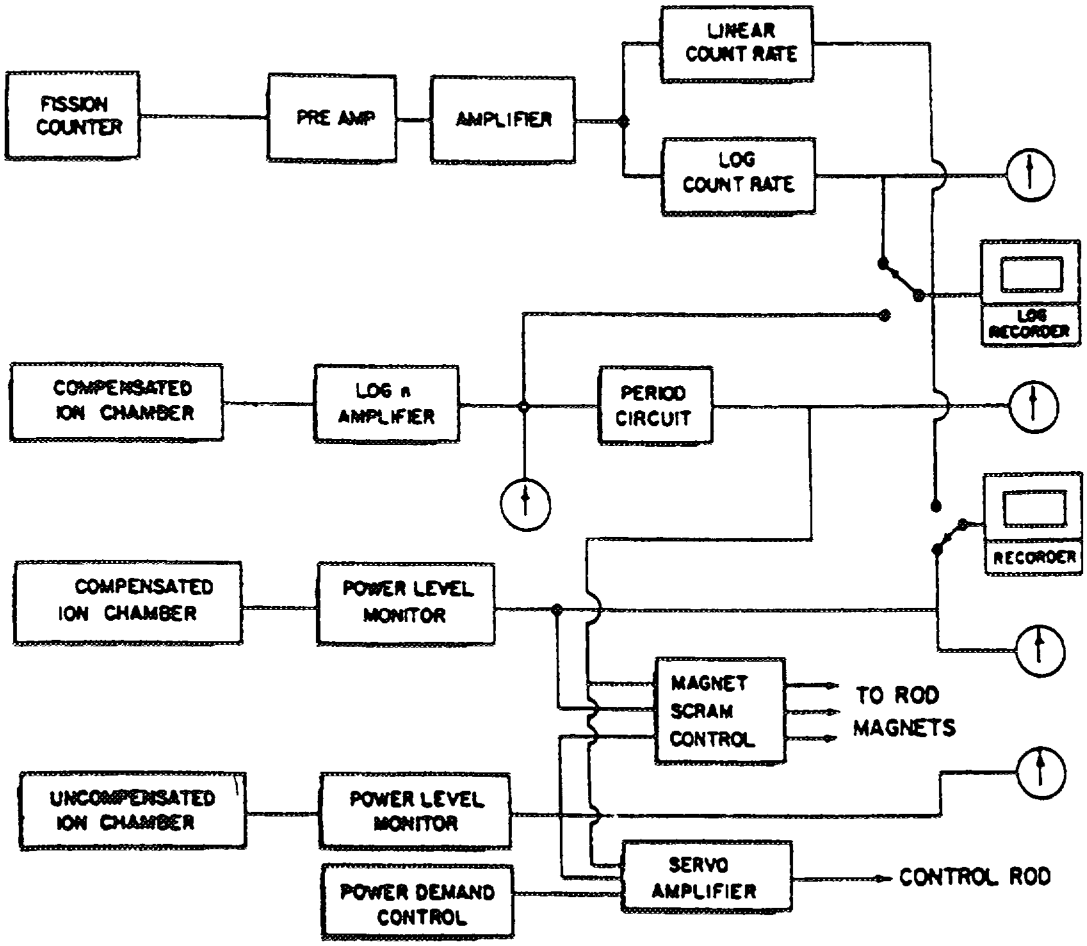

Fig. 1: Block diagram of the instrumentation and scram system

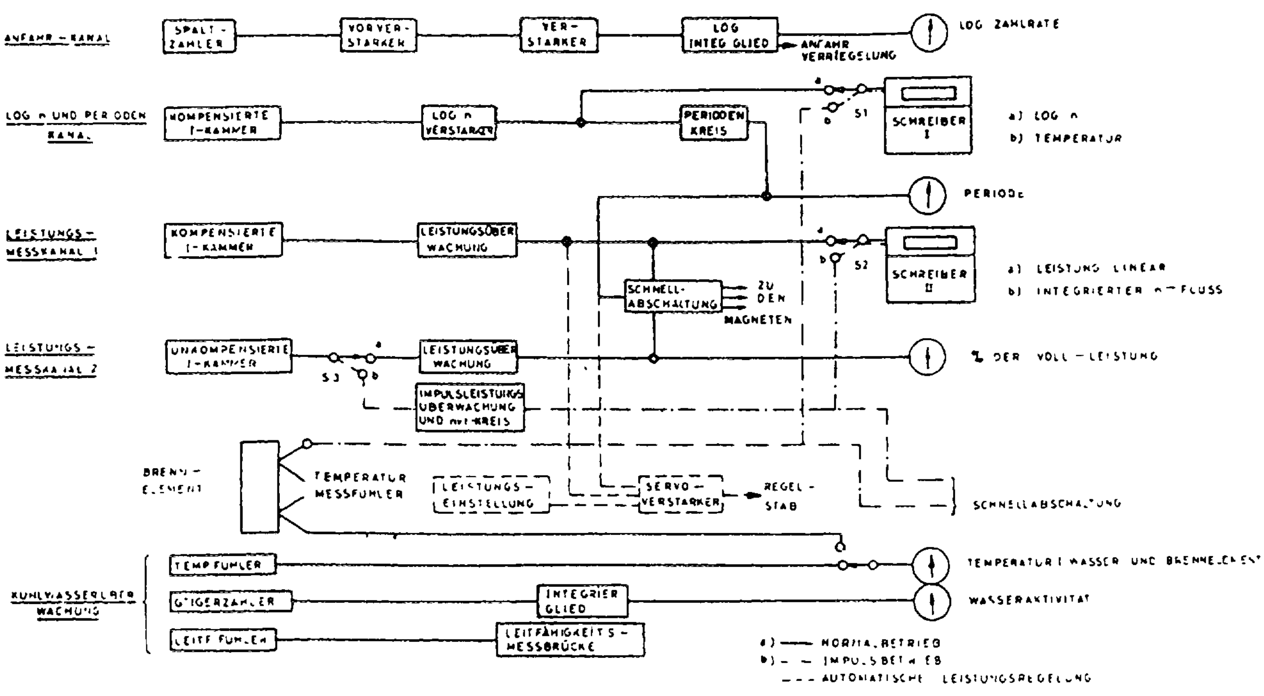

Fig 2. Block diagram of the instrumentation 1968 - 1992

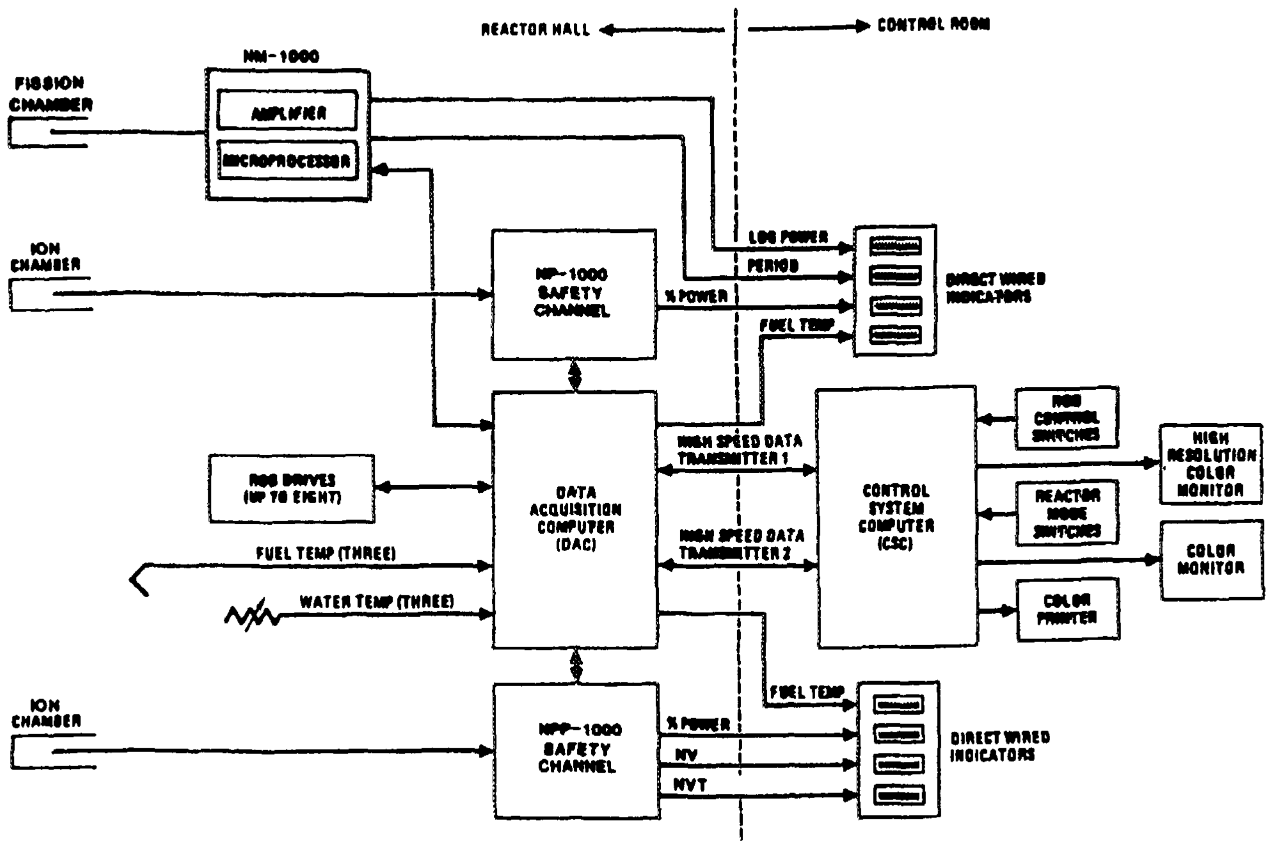

Fig. 3: Block diagram of the reactor instrumentation as offered by GA in 1990

MAIN ERRORS: November 1992 to December 1995

Date	Cause	Consequence
Sep 92	Thunderstorm	ROM of wide- range channel destroyed
Oct 92	Contacts of microswitches	Problems with rod- drives
Dec 92	EMI due to motor	Period- SCRAM
Jan 93	Bad electrical contact	Temperature- SCRAM
Mar 93	Temperature- effect	Problem with safety channel
Mar 93	Broken capacitor	Transformer died
Mar 93	Broken resistor	Purifier flow low
Mar 93	Loose wires	No magnet current
Apr 93	Reversed wires	Primary cooling system affected
Apr 93	Short circuit	Fuel temperature low
June 93	Console Power ON/OFF	Power supply of DAC failed
June 93	Loose wire in DAC	SCRAM
Sep 93	DAC- Timeout	SCRAM
Jan 94	HD connector loose	CSC failure
Apr 94	DAC- Timeout	SCRAM
May 94	DAC- Timeout	SCRAM
June 94	HD failure	CSC failure
July 94	NM1000 constants wrong	Prestart check failed
July 94	Wrong software version	Pulse not recognized
Sep 94	HD failure	AUTO- Mode does not work
Nov 94	NMP2 not adjusted	Prestart check failed
Dec 94	Connector loose	Prestart check failed
May 95	Relay died	Prestart check failed
Dec 95	NM1000 problem with constants	Prestart check failed sometimes

Fig. 4

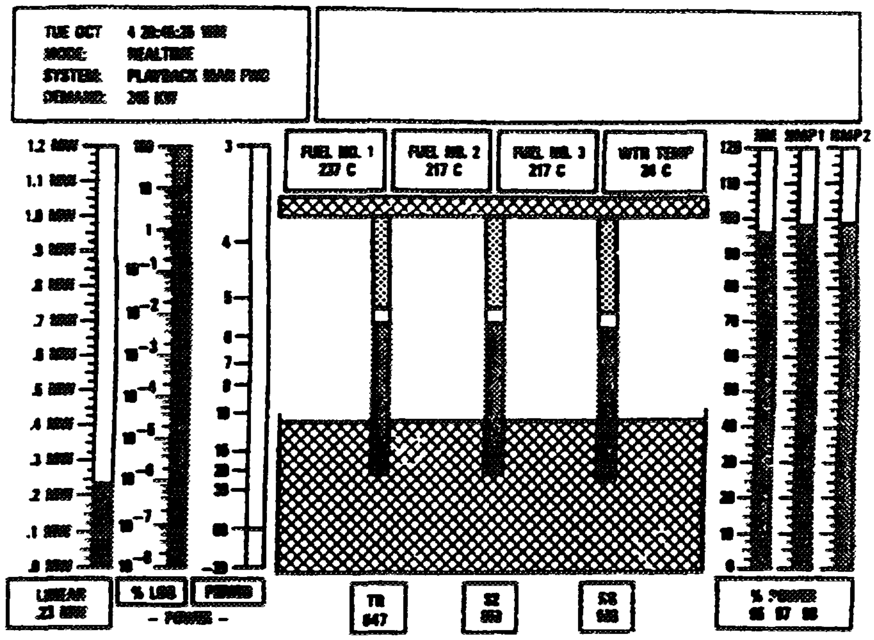

Fig. 5a: Graphic CRT

```
11/03/92*********************VIENNA STATUS WINDOW***********************12:44
* NM1000 Power                5.77e-3 W   * Prim Flow m3/h              .1
* NM1000 Log Power            2.30e-6 %   * Sec Flow m3/h               .0
* NM1000 % Power                    0 %   * Purif Flow m3/h            2.1
*                                         * Inlet Cond uS/cm           1.6
* NMP CH                            0 %   * Outlet Cond uS/cm           .2
* NMP PH                            0 %   *
*                                         * Pool Level Extr Low A       OK
* Fuel Temp #1                     24 C   * Refill Pool                 OK
* Fuel Temp #2                     23 C   *
* Fuel Temp #3                     24 C   * Aerosolmonitor M 17     5.6e+1 cpm
* Fuel Temp #4                     23 C   * Vent PH M 15            1.0e-1 cps
* Fuel Temp #5                     25 C   * Vent CH M 16            1.8e+2 cps
* Fuel Temp #6                     27 C   * Tank Top M10 e-2 mSv/h  9.9e-3
*                                         *
* Pool Temperature               25.9 C   * Trans Drive Position         0 mm
* Prim Out Temp SCRAM            22.0 C   * Shiml Drive Position         3 mm
* Prim In Temp ALARM             27.6 C   * Reg Drive Position          29 mm
*                                         *
* Current Pulse Number           1190     * Reactor Hall Underpr        OK
*                                         *
* One Kilowatt Interlock           No     * NMP PH & CH Power Chec     .00 %
* Rod Withdrawal Prohib            No     * Magnet Supply Voltage 2.037e+1 VDC
*                                         *
****************************************************************************
```

Fig. 5b: Status CRT

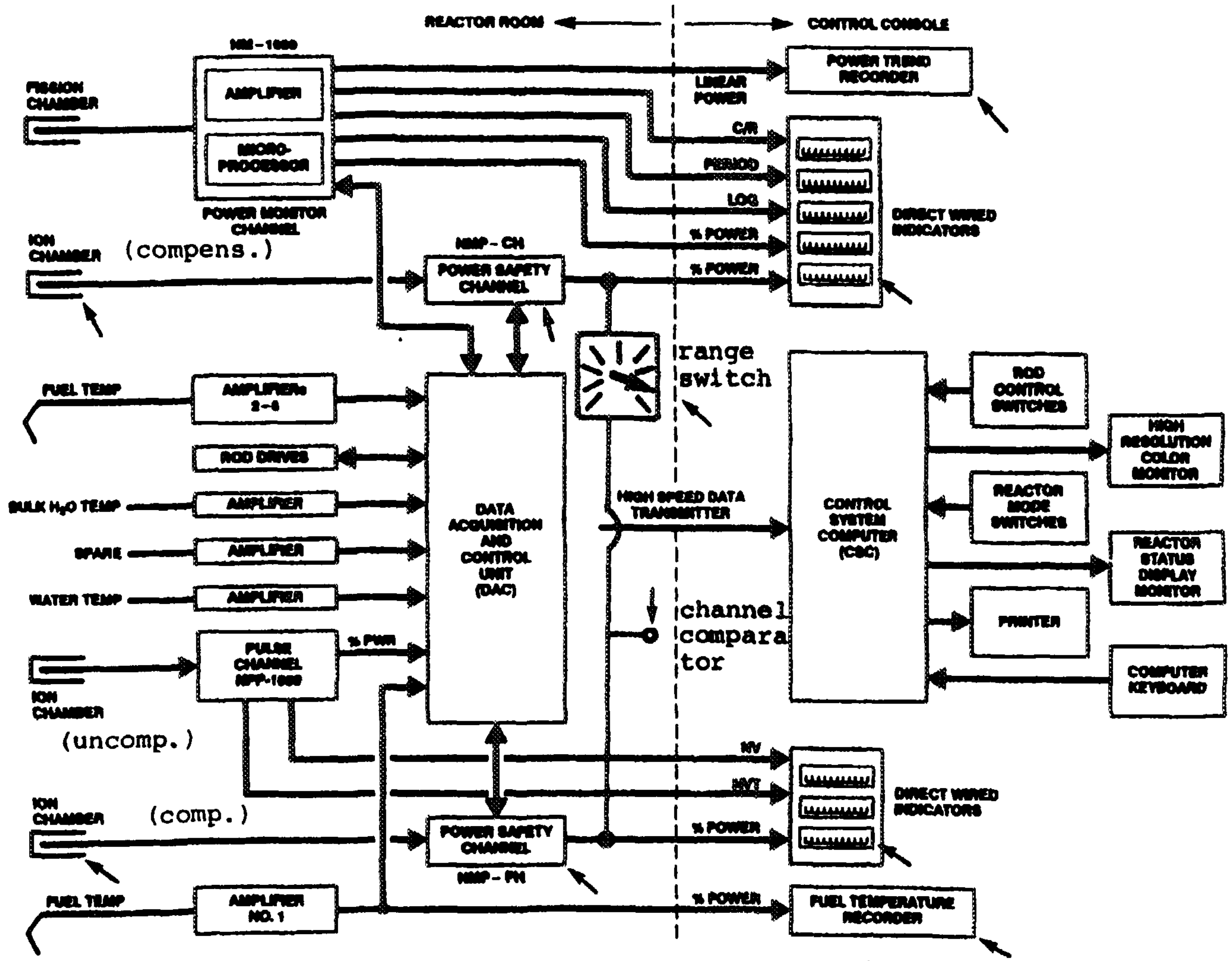

Fig. 6: Block diagram of the new reactor instrumentation

All main signals which are used for the reactor protection system are hard wired. As there are for example two parallel independent linear channels (one out of two logic), the water temperature and the high voltage of all channels. Two fuel elements are equipped with three NiCr- Ni thermocouples each. Those signals are added to the reactor protection system in the way of a one out of six logic.

The computerized part of the I&C system is used to display the information about the condition of the reactor and to handle the control rod drives in all operational modes.

The data display is shown on one graphic CRT and on one status CRT (fig.5). The arrows in the block diagram in fig.6 show the instruments which where modified for the TRIGA Vienna compared to the original proposal. Additional features added for the Vienna instrumentation are:

- a range switch
- a strip chart recorder for linear power
- a strip chart recorder for fuel temperature during pulse operation
- a comparator for the two independent linear channels

With these features the reactor instrumentation is in operation since summer 1992.

3 OPERATION EXPERIENCE

During the last 3 years of operation a number of failures occurred which made the reactor inoperable for about 30 days in total. Some of the errors were localized within a short time, other errors were more difficult to overcome. In example in February 1994 a computer drive in DAC failed and a new drive loaded with the Austrian DAC software was ordered from GA. In June 1994 the drive in CSC computer failed and again a new drive loaded with the Austrian CSC software was ordered from GA. Apparently both the DAC and CSC software which was used in Vienna before both failures were not the same as the new software loaded by GA, therefore problems arose with pulse operation. In fact since June 94 pulse operation was not possible due to this problem. Finally after many phone calls and faxes the problem was solved and the system is fully operational again.

Further we observed that the system is quite sensitive to room temperature. During summer 1994 we some times had control room temperatures of

35°C, and some channels in the DAC cabinet had surface temperatures of above 55°C. We had already removed the front and backdoor of the cabinet before and installed two ventilators on the top of the cabinet, however it is strongly recommended to improve air circulation in the DAC. Also the microprocessor of the NM-1000 is very temperature sensitive, the channel constants are erased above 30°C.

Another problem occurred the last days. The prestart test fails sometimes with the information:

%Power= 9.46e+0 should be 9.97e+0

To solve this problem the operators made several resets on the computer, until the prestart test is completed without error. After this the reactor can be operated normally.

Finally we had realized that there is no document available which allows the operator simple verification of software or to adapt certain parts of the software to local needs. In example the procedures to change date and time in the status window, to change the denomination of sensors in the status-, alarm- and scram windows or to access the pulse number or the kWh value in case of necessary changes, all this has to be individually asked for. This is a time consuming and unnecessary burden for both the operator and the supplier. In fig.4 there is a summary of instrument related problems since summer 1992.

4 EVIDENT PROBLEMS

The instrumentation design originates from the mid '80 and most of the electronic equipment especially the console computer and the DAC computer were already outdated at the time of installation. Due to the rapid development in data acquisition technology the problem of spare part availability becomes imminent.

- problems with HDD
- practically no access and documentation of software
- even slight changes in the configuration are difficult as there are no documents available

5 CONCLUSION AND OUTLOOK

The new instrumentation has the following advantages:

- spare parts of the electronical circuits are available
- the safety system is more sophisticated and
- the prestart test results can be printed for documentation

A disadvantage is that there exists no documentation of the software and the NM1000, especially the importance of the various channel constants are not very clear.

The most serious problem in the moment is the hard disk, which is an old 40MB drive. This device has a short life time and it is expected that this spare part is no more available in the near future. So in the beginning of next year we will try to replace this drive by an AT- bus hard disk. This type of hard disk has a much higher life time, and will be available as spare part for the next years.

XA9745520

REACTOR INSTRUMENTATION EXPERIENCE AT IPEN-CNEN/SP

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Abstract

The Instituto de Pesquisas Energeticas e Nucleares (IPEN) in Sao Paulo started in 1977 development of nuclear instrumentation. In 1982 two groups were established, one responsible for maintenance of electronic equipment and the other, the Nuclear Instrumentation Section (NIS), for new projects, such as a light water moderated critical facility (IPEN/MB-0). The NIS had the responsibility to execute all necessary activities to develop the "nonconventional" instrumentation systems: control, protection and radiation monitoring for the IPEN/MB-01 facility. After the commissioning of the critical facility the NIS group started the development of instrumentation systems for a power reactor. As a first step an overall review of the qunality assurance programme was implemented. In parallel the development of self powered detectors was carried out.

1. INTRODUCTION

There are three main nuclear research centers in Brazil, Instituto de Pesquisas Energéticas e Nucleares (IPEN-CNEN/SP), located in São Paulo, Instituto de Engenharia Nuclear (IEN), located in Rio de Janeiro, and Centro de Desenvolvimento de Tecnologia Nuclear (CDTN), located in Belo Horizonte, Minas Gerais. IPEN operates a 2 MW(th) MTR type reactor (IEA-R1) and a 100 W critical facility (IPEN/MB-01), IEN operates a 200 W Argonauta type reactor (ARGONAUTA), and CDTN operates a 100 KW Triga MARK-I type reactor (IPR-R1). In this presentation I will discuss the existing experience at IPEN-CNEN/SP, in the field of reactor instrumentation.

2. FROM 1957 TO 1977

IEA-R1 reactor was the first nuclear reactor in the south hemisphere, as a result of the program "Atoms for Peace". The operation of the reactor started on 1957, when it reached criticality for the first time. Since 1957 until 1973 there was an "electronic division" responsible for two main tasks: to develop small projects, in order to attend the necessities of the researchers, and to provide maintenance for all the instruments in the Institute, mainly the ones related to the reactor. In 1973 the Direction of the Institute noticed that the technical staff had the capability to develop some instruments and started a nationalization program. However, by that time importation was easy, and since the economy of Brasil was growing, it was difficult to maintain the same professionals working together for long periods. For these reasons the nationalization program was too slow, and when the prototypes were finished, they were already obsolete. Another event that contributed to slow down the nationalization program was the renewing of the reactor instrumentation. The original instrumentation was suplied by Babcock & Wilcox, and installed in 1957, when the reactor went critical for the first time. Figure 1 shows the original control panel of the reactor. In 1974 a new instrumentation system was bought

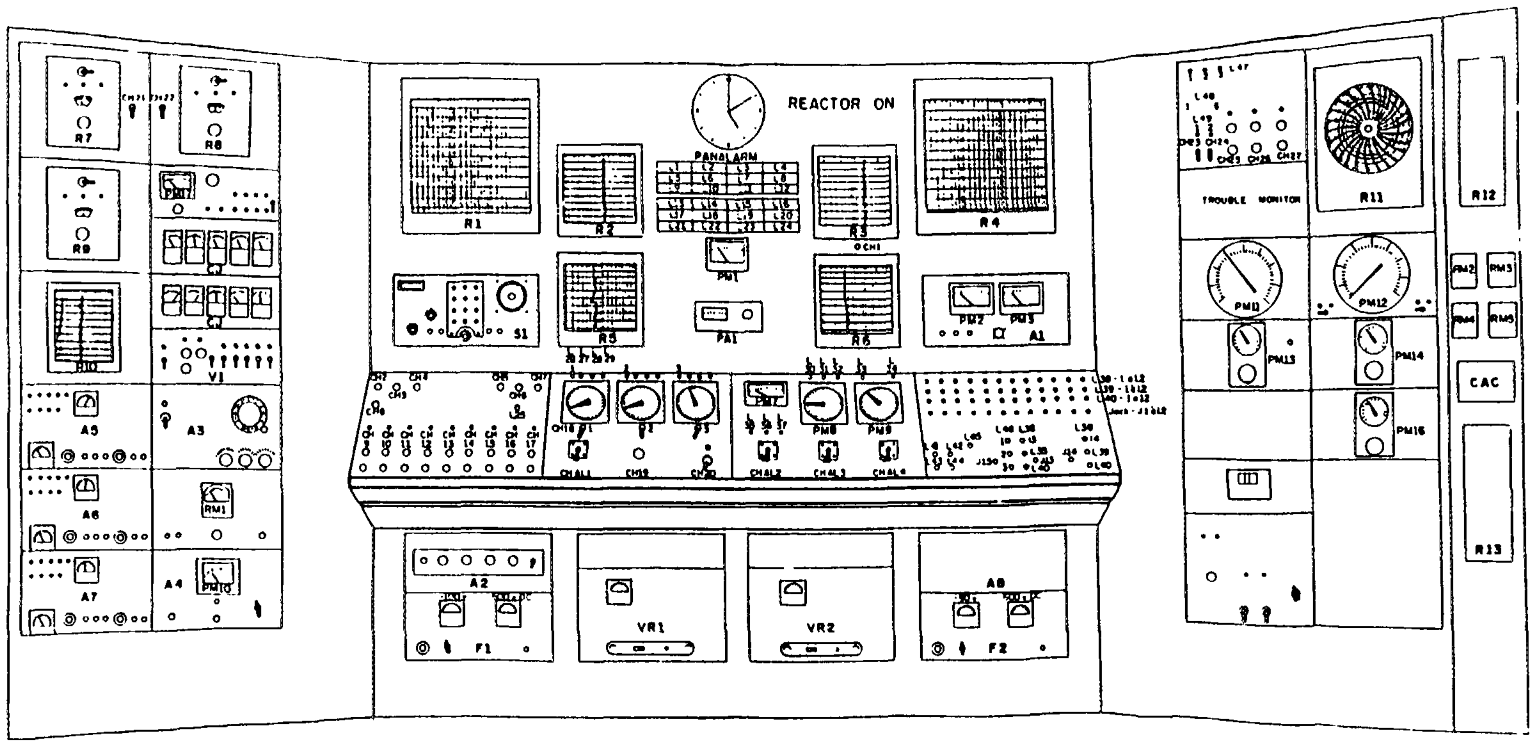
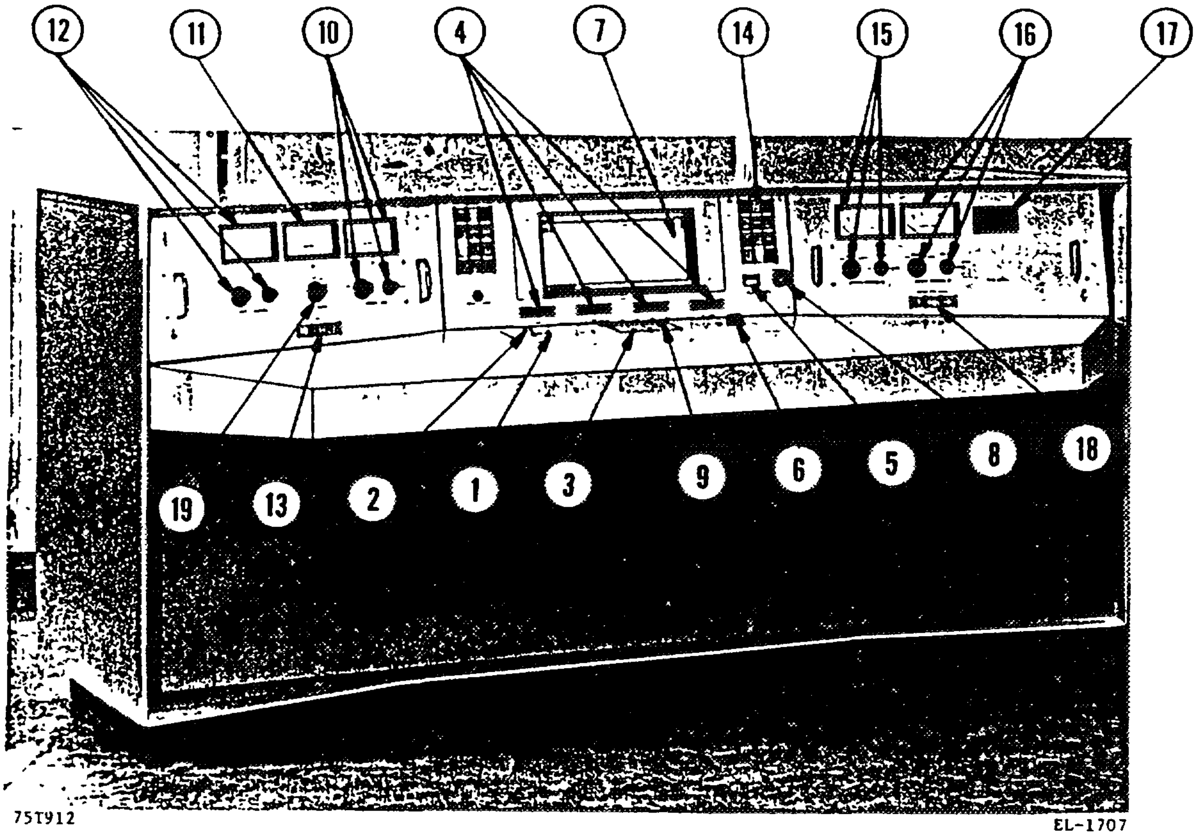


FIG. 1. Original control panel of the IEA-R1 reactor.



1-POWER ON switch
2-MAGNET POWER/OFF key switch
3-CONTROL ROD switches
4-ROD POSITION digital indicators
5-MANUAL/AUTOMATIC switch
6-REACTOR POWER switch
(for strip chart recorder)
7-Dual pen strip chart recorder
8-FLUX CONTROL dial(% demand)
9-SCRAM BAR
10-PERIOD meter
11-LOG-POWER meter
12-SAFETY CHANNEL 1 meter
13-TRIP/ALARM indication
14-ALARM indication
15-SAFETY CHANNEL 2 meter
16-SAFETY CHANNEL 3 meter
17-N-16 meter
18-TRIP indication
19-AUDIO range switch

FIG. 2. Operator's console (IEA-R1 reactor).

from General Atomic, a U.S. company. It was composed by a new set of rod drive mechanisms, process and area radiation monitors, safety channels, scram circuitry, instrumentation panels and a new operator's console, as shown on figure 2. The new system was installed in 1976, and excluding the radiation and neutron detectors, all other field instruments were maintained as originally installed. It is important to note that all activities (design, construction and installation) were performed by GA technicians. The only contact IPEN engineers had with the system was during some "follow up" visits to GA, during the training sessions promoted by GA for maintenance purposes, and during the installation of the system, when they worked toghether with GA engineers.

Actually four neutron detectors are used to monitor and control the reactor power level, and the period of the reactor. The four detectors are: one wide range type detector, one compensated ionization chamber, and two non-compensated ionization chambers. The wide range detector is a U_{235} fission chamber, and connected to its electronic circuit forms a Campbell channel. It is used to monitor the reactor power from start up up to 120 percent of the nominal power. It is also used to monitor the period of the reactor in the start up region (below 1 percent of nominal power). The compensated ionization chamber is a B_{10} linned detector. Toghether with its electronic circuit it forms a linear channel, which is used to control the reactor in the automatic mode. The non-compensated ionization chambers are BF_3 detectors, which are used to monitor the reactor power from 1 up to 120 percent of the nominal power. Together with their electronic circuit they form what is called the safety channels (safety 2 and safety 3, since the wide range monitor is also known as safety 1). Figure 3 shows the relative location and the operational interval of the several neutron detectors. The safety channels, plus the Campbell channel, are used to trip the reactor in a 2 out of 3 logic circuit, whenever the reactor power reaches 110 percent of its nominal value.

Other signals used to trip the reactor are: high radiation level (in the reactor pool or in the room of experiments), low core ΔP, low primary collant flow, high core outlet temperature, bridge position indication, core header position indication and beam ports indication.

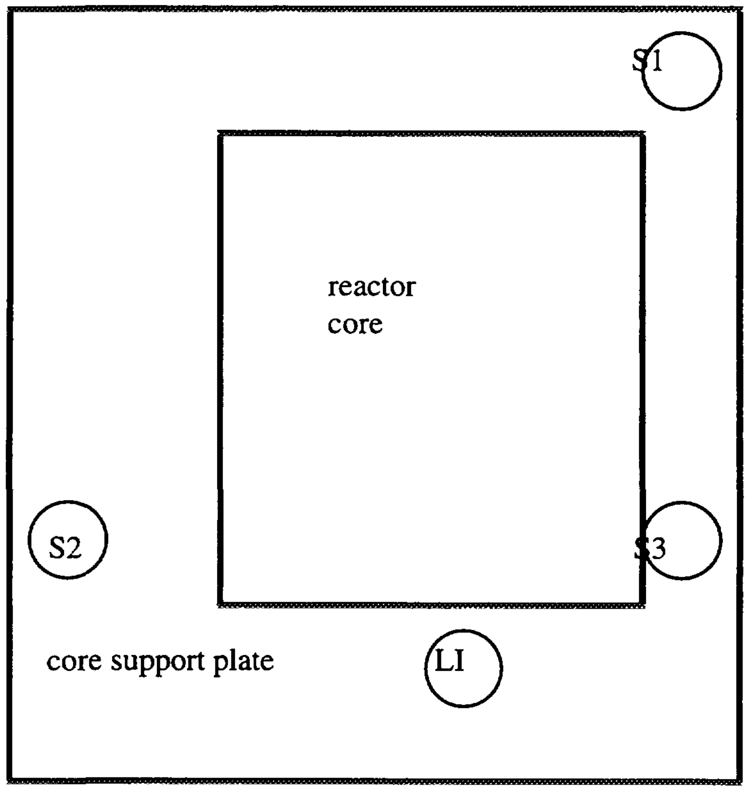
A N-16 channel, using a gamma ionization chamber is used as a redundant indication of the reactor power. It is used mainly for calibration of the neutron detectors.

In the early 70s, IPEN, by that time named Instituto de Energia Atomica (Atomic Energy Institute), joined a U.S. program to study high temperature gas cooled reactors (HTGR). Within this program the reactor engineers of IPEN made a conception and a specification of a graphite moderated, air colled critical assembly, named RPZ (Zero Power Reactor). The major componentes of the facility were bought, and some delivered. The instrumentation system, including the nuclear channels and radiation monitors, was bought from GA, and the operator's control panel was assembled by a brazilian company, in a joint activity with IEA's engineers. After the signature of the technical cooperation agreement between Brazil and Germany, to develop and transfer nuclear technology, the brazilian HTGR program started to slow down, and was definitely abandoned in the late 70s. To make some use of the instrumentation bought, all itens become "spare parts" for the IEA-R1 instrumentation system, and have been used as such until now.

3. FROM 1977 TO 1982

In 1977 IPEN started having some difficulties with importation, and the nationalization program was reformulated. A working group was formed, including some international colaborators, with the especific mission to develop nuclear instrumentation. Very soon some equipments were prototyped like NIM modules (bin, power supplies, spectroscopy amplifier, counter/timer, printer controller, current pre-amplifier, pulse pre-amplifier, single channel analyzer, etc..); health physics instruments (personnal dosimeters; environmental monitor; and portable monitors); and some general application instruments like noise analysis amplifier, high pass filter for noise analysis, level controller, etc.

At the beginning, the production was only for internal consumption, but soon other institutions wanted to buy the instruments, and in 1982, from a production of 73 itens, 32 were sold to other institutions, including several universities. By the end of that year, in addition to the 73 produced, other 103 were in final phase for conclusion (depending of imported components), and a contract had been signed with a private company to produce and commercialize some of the instruments.



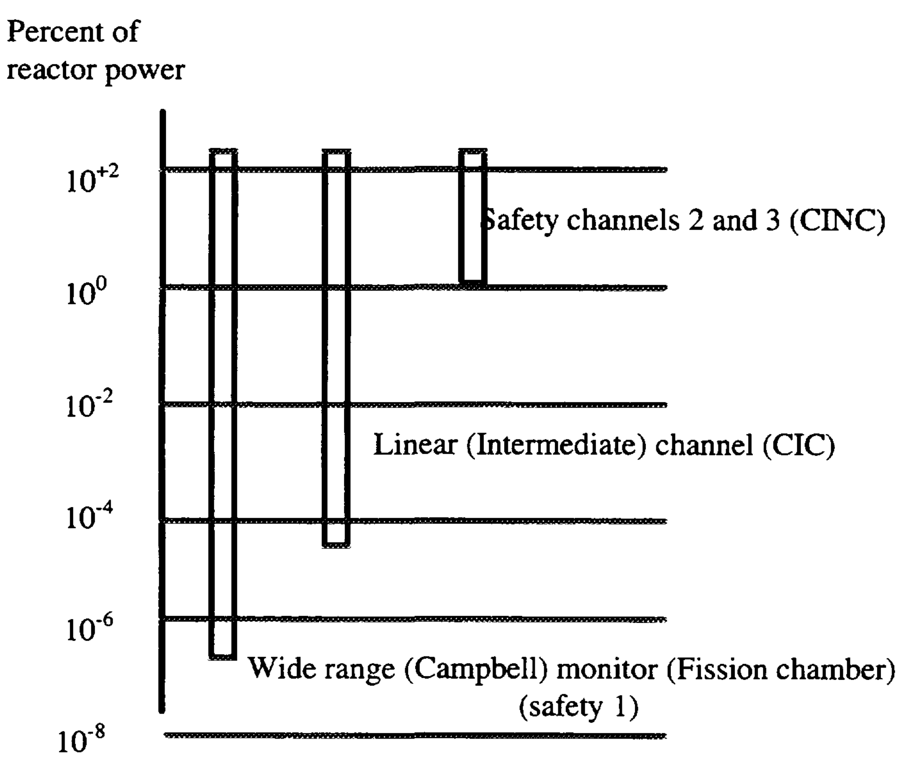


FIG. 3. Neutron detector operational intervals (IEA-R1 reactor).

4. FROM 1982 TO 1988

In 1982 Brazil started a national independent program to develop nuclear technology, now following the line of PWR reactors. Within this program two major laboratories were to be developed, a high pressure, high temperature thermal-hydraulic loop, and a light water moderated critical facility, named IPEN/MB-01. When the project started it was noticed that, due to the increasing quantity of instruments in the Institute, and because of economic problems, all electronic engineers were involved with maintenance activities, leaving no time for new projects. Then it was decided to create two groups, one responsible for maintenance and the other responsible for new projects. The group responsible for maintenance was also responsible for giving technical support for researchers that needed some especific equipment, usually small adaptations. The group responsible for new projects became the "Nuclear Instrumentation Section (NIS)", part of the "Instrumentation and Control Division".

The I&C Division had the mission to coordinate all activities related to control and instrumentation of the critical facility IPEN/MB-01, including the electrical systems, and the NIS had the mission to execute all necessary activities to develop the "non-conventional" instrumentation systems, with one basic orientation: to obtain the highest possible index of nationalization. Here it is important to notice that the term "non-conventional" means any variable not found in the common industry (like neutron flux and period, radiation level, control rod position and drivers, etc.). The responsability for the "conventional" instrumentation was left to an engineering company, that had great experience in design and construction of control panels for oil refineries, and for the industry in general.

Since the NIS had, by that time, only 4 engineers (2 with 8 years of experience, and 2 with 2 years of experience), and about 6 technicians, several areas had to be worked out in parallel, as: planning, training (of new professionals), construction of a new laboratory (for development and production), and development of a program of quality assurance (including internal procedures for design, construction, testing, production, intallation and commissioning). During the years of 1983 and 1984 an old building was adapted, giving place to the Nuclear Instrumentation Laboratory, and we started the development of the “non-conventional” instrumentation for IPEN/MB-01 reactor.

4.1. Design and development of “non-conventional” instrumentation for IPEN/MB-01 reactor

The "non-conventional" instrumentation was divided into three systems: control (including rod drive mechanisms and the operator’s console), protection and radiation monitoring, as described as follows.

Control system

The control system was designed to allow continuous monitoring of tne process variables, and to maintain them within pre-defined limits. It was sub-divided into three subsystems : process control, data acquisition and reactivity control. Included in the control system is the operator’s console, from where the operator can monitor the information from the nuclear channels, that belong to the protection system.

The process control subsystem was developed to allow control of the moderator level and temperature, in order to perform reactor physics experiments. It was made up of "conventional" instruments, which means ΔP transmitters (to measure level), thermocouple sensors (to measure temperature), and a single loop controller.

The data acquisition subsystem was developed to acquire operational data, in order to make the daily records. It did not have the "experimental" purpose. It consists of a microcomputer IBM-PC compatible, plus an A/D unit with capacity for 32 analog signals and 32 digital signals. It is important to notice that by that time the government of Brazil was establishing the rules for development of digital systems in the country, so, we did not have suppliers for digital parts, and for this reason we had to develop our own A/D unit, which was composed by a mother board (following the STD bus), a multiplexer, a 12 bits A/D converter, an I/O digital interface, a RS422 comunication interface, and the necessary power supply.

The reactivity control subsystem was developed to allow insertion and removal of safety and control elements. There are two elements for safety and two for control, and only one element can be moved at a time. For each element (safety/control), the subsystem has a driving mechanism, a power source driving unit, a relative position indicator unit, and an absolute position indicator unit. Figure 4 shows the driving mechanism.

Protection system

The protection system was designed to avoid any unsafe condition. It was subdivided into two subsystems, the nuclear detection subsystem and the interlock subsystem. The nuclear detection subsystem is used to monitor neutron flux level and period. It is composed of 8 nuclear channels to monitor the neutron flux from start up to 100% of full power (100 watts), including comparators and isolation amplifiers. Three channels are used in the start-up region, and the others in the intermediate and power regions. In each region we have three measurements of the neutron flux (power) and three measurements of the period. The nuclear channels are complemented by two linear channels, used (alternatively) to control the reactor in automatic mode. Figure 5 shows the relative location of the detectors, and the operational interval of them.

The nuclear channels were designed and built by the engineers of IEN (another nuclear research institute of Brazil), that had a well consolidated program to develop such instruments.

As explained before, by that time the government of Brazil was establishing the rules for development of digital systems in the country. This means that we did not have enough experience with digital systems, and for this reason all the protection system, including the nuclear channels, was built with analog circuits and electro-mechanical relays. Figure 6 shows the front view of one typical nuclear channel.

The interlock subsystem was designed to avoid any unsafe operations, and to promote the scram of the reactor at any time an unsafe condition is detected. It consists of "conventional" electro-mechanical relays plus 7 buttons distributed throughout the installation to promote the scram of the reactor. The signals used to scram the reactor are: loss of electrical power, including individual power for any nuclear channel, neutron high flux and low period, high radiation level, loss of pneumatic system, low underpressure inside reactor room, and any open door (to access the reactor room).

Since we did not have enough reliability data regarding the relays produced in Brazil, we had to develop a qualification program for them. It consisted of testing a representative sample of the relays bought as follows: 500.000 operations at 120% of nominal currente, normal ambient conditions, followed by the aplication of the same current, continuously, during 24 hours, at 70 °C. The criteria for approval was zero failure in the closed condition, and response time (to open) lower than 100 mseg.

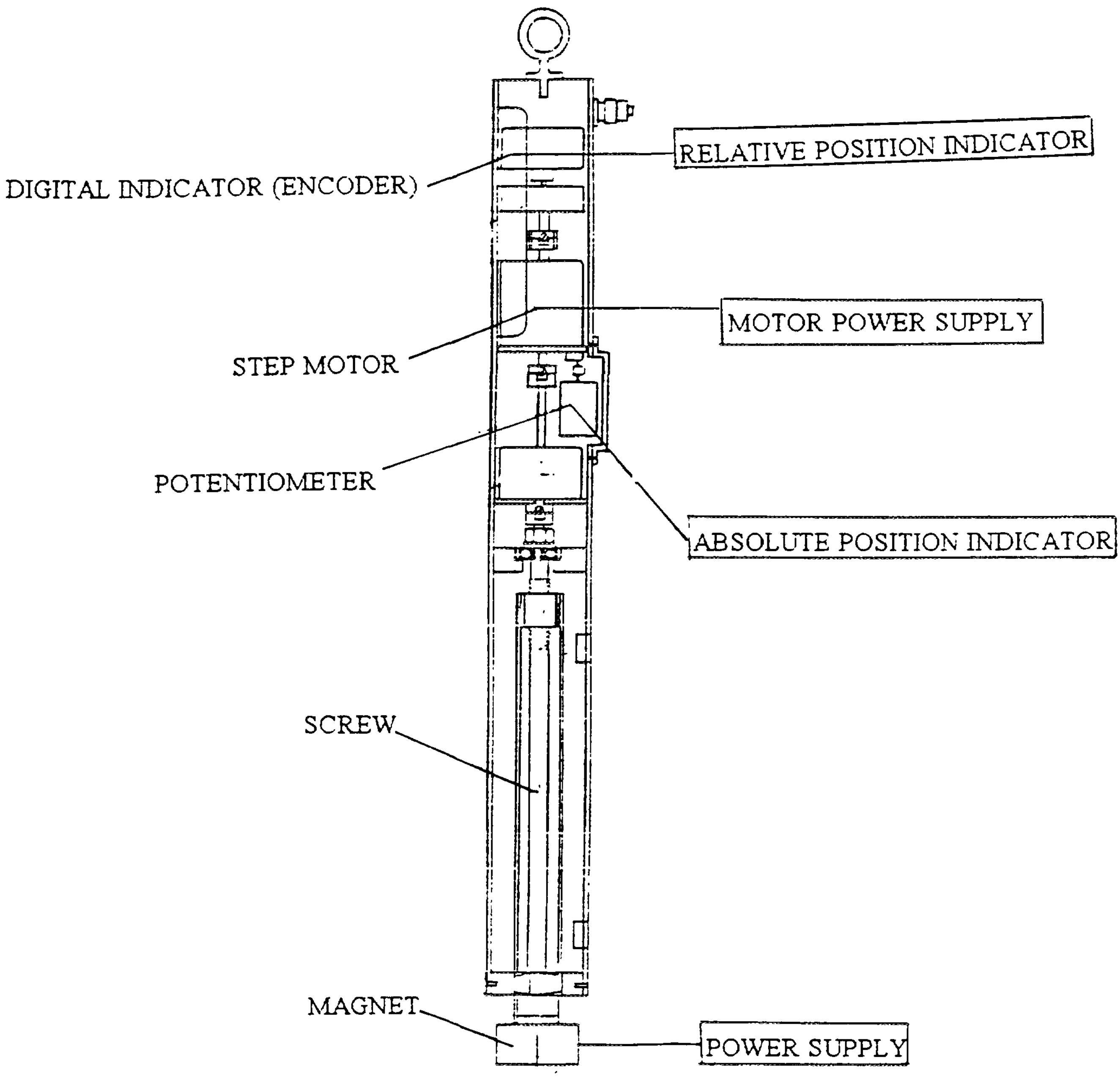


FIG. 4. Rod drive mechanism (IPEN/MB-01 reactor).

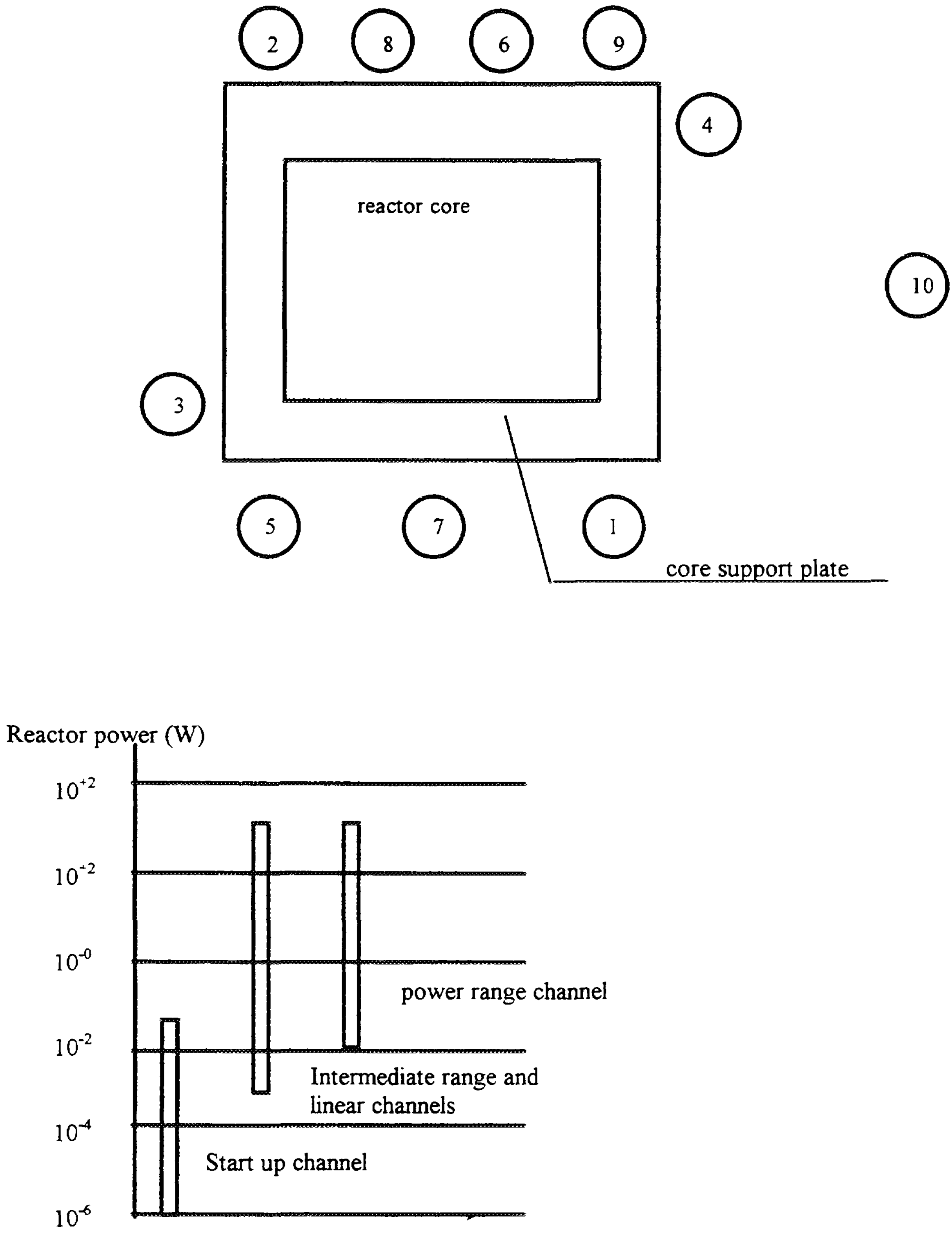


FIG. 5. Location and operational interval of neutron detectors (IPEN/MB-01 reactor).

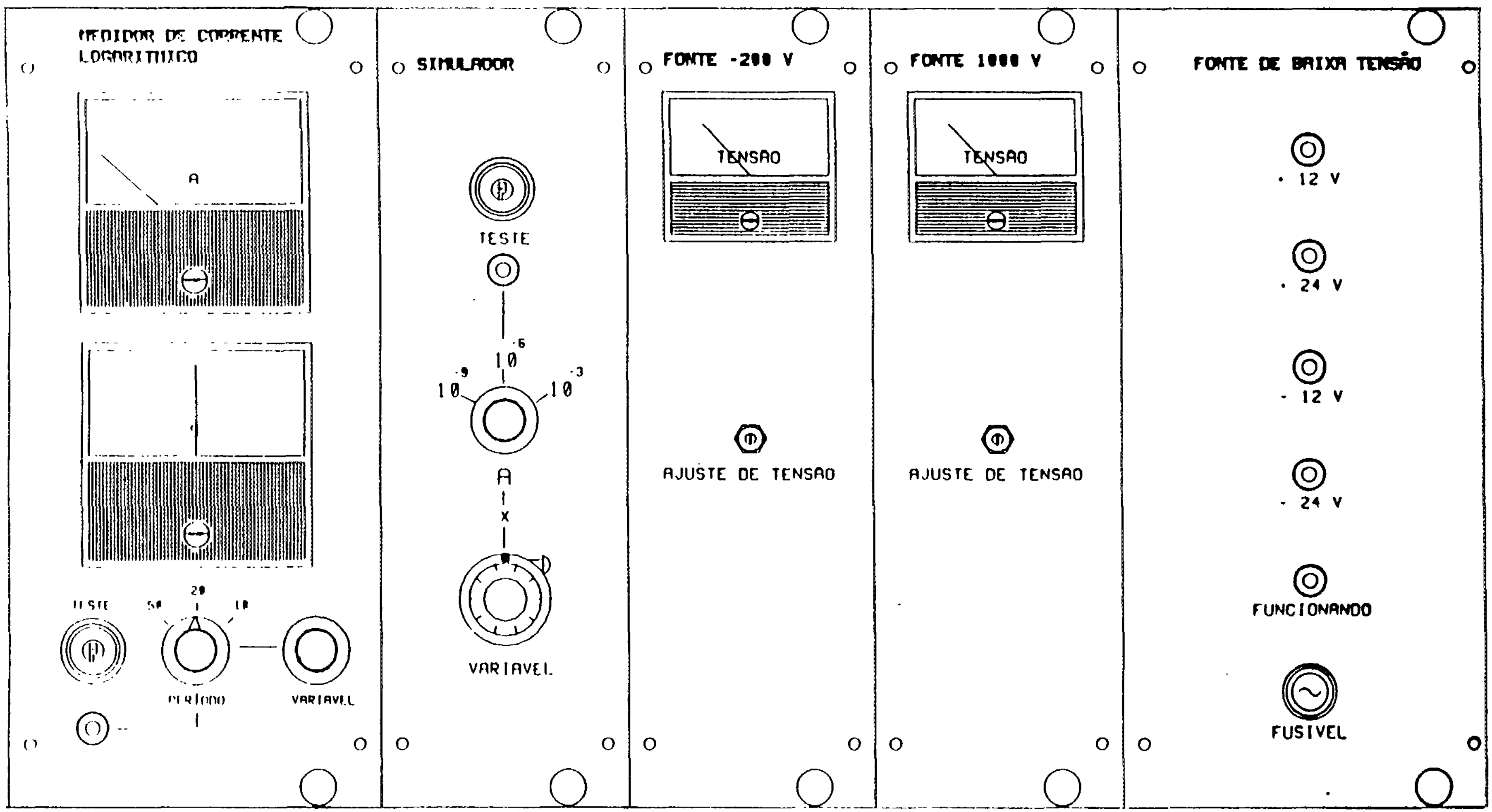


FIG. 6. Front view of a typical nuclear channel (IPEN/MB-01 reactor).

In order to qualify the instruments of the protection system, we also had to study and develop techniques to measure the response time of the process and nuclear instrumentation, in order to verify that they were within acceptable values.

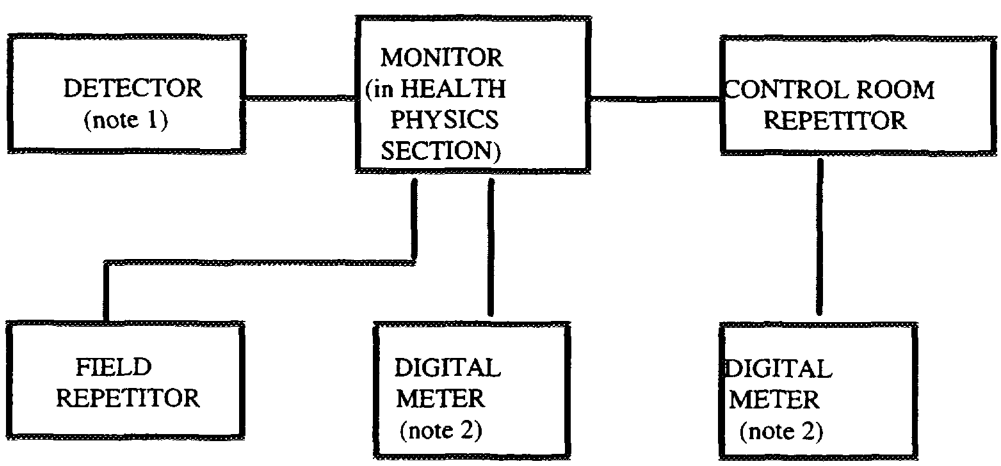
Radiation monitoring system

The radiation monitoring system was designed to allow continuous monitoration of radiation levels inside the critical facility. It is composed by 27 radiation monitoring channels, being 14 area monitors, 8 process monitors, 3 air monitors, 1 hand and foot monitor and 1 monitor of the "portal" type.

As a comon practice, all area monitors have a detector, a main monitor, locate at a panel in the Health Physics section, and two repetitors, one in the control room, to inform the reactor operator, and the other close to the detector. Figure 7 shows a schematic of an area monitor.

Excluding the nuclear detectors, all instrumentation and control systems were designed, developed, produced, installed and commissioned by brazilian engineers and technicians.

After 6 years of work, including about two years spent rebuilding the Nuclear Instrumentation Laboratory, in november 1988 the reactor IPEN/MB-01 reached it's criticality for the first time.



note 1 : The detector can be a G.M , with it's associated pre-amplifier, or an ionization chamber
note 2 : The digital meter can be connected to 5 monitors

Fig.7. Typical radiation monitor channel.

5. FROM 1988 UNTIL NOW

It is important to say that the activities that started in 1982 were the result of an agreement between the National Nuclear Energy Commission, and the Brazilian Navy. According to the agreement, the technical staff of both institutions started working together, with a common objective: to develop nuclear technology. So, after the commissioning of the critical facility, we started the development of another instrumentation system, now for a power reactor, which required the instrumentation to be classified as nuclear class (1E). Since we did not have procedures to develop class 1E instrumentation, we had to make an overall review of our quality assurance program, including a new methodology to develop digital electronic systems.

In parallel with the review of the quality assurance program, we developed an internal capability to construct and characterize self powered detectors (SPD), and started to develop new prototypes of electrometers, to use together with the SPDs, radiation monitors and a new generation of nuclear channels.

At the end of the first semester of 1995 we started noticing that, although the operator's console of IEAR1 reactor is about 20 years old, some of the process instruments, and transmitters, are about 40 years old, and a new modernization program started to be implemented. Actually we are working on the schedule for the modernization program, which is expected to be completed in 1999.

MAIN REFURBISHMENT ACTIVITIES ON ELECTRONIC AND ELECTRICAL EQUIPMENT FOR THE FRG-1 RESEARCH REACTOR

XA9745521

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Abstract

As GKSS intends to operate the research reactor FRG-1 safely and reliably for many years to come, the plant is constantly refurbished and upgraded both in the interests of safety and operational reasons. The following electronic and electrical systems have been replaced or improved since 1990:

-Information and signalling systems
-Emergency power plant (permit applied for)
-External and internal lightning protection system
-Reactor protection system (in part)
-Safety lighting
-Alarm and staff locating system
-Control room telephone system
-Closed-circut television system
-Beam tube controls
-Storage plant for radioactive liquid waste
-Ambient dose rate measuring system
-Meteorological measuring system
-Controi and measuring system for the primary cooling circut
-Conrol rod drives
-Control rod control system
-Soft start for the secondary pumps
-Control and switching devices for the emergency power plant
-Trailing cable installation for the reactor bridge
-Main-voltage distribution systems / cable routes

As GKSS intends to operate the research reactor FRG-1 safely and reliably for many years to come, the plant is constantly refurbished and upgraded both in the interests of safety and for operational reasons. Table 1 shows the electronic and electrical plant and systems that have been replaced since 1990. Some of these measures are described in detail below.

Information and signalling systems

The information and signalling system, the control room and the control desk for the research reactor FRG-1 were replaced in 1990. The aim was to considerably improve the operating crews' overview of the status of the plant by enabling optimum presentation of information and messages in mosaic mimic diagrams and on visual display units integrated in the console. In April 1991 it was decided to install a further independent information and signalling system for all the auxiliary facilities in order to make additional information available to the operating crews and handle modifications and additions in a more flexible manner.

Plant and Systems Replaced since 1990	
*	-Information and signalling systems
*	-Emergency power plant (permit applied for)
*,**	-External and internal lightning protection system
*	-Reactor protection system (in part)
*	-Safety lighting
	-Alarm and staff locating system
	-Control room telephone system
	-Closed-circut television system
*	-Beam tube controls
	-Storage plant for radioactive liquid waste
	-Ambient dose rate measuring system
	-Meteorological measuring system
	-Control and measuring system for the primary cooling circut
*	-Conrol rod drives
**	-Control rod control system
	-Soft start for the secondary pumps
	-Control and switching devices for the emergency power plant
	-Trailing cable installation for the reactor bridge
	-Main-voltage distribution systems / cable routes
	* covered in this paper, ** to be completed in 1996

Table 1

The information and signalling systems shown in Fig. 1 are divided into the systems research reactors, test facility and auxiliary facilities. The components, which include the operating and monitoring system (OS 262), the automation system (AS 230) and the long-term memory, are connected to each other by bus 1 in the case of the "research reactor" system and by bus 2 for the "auxiliary facilities" system. The test facility, which can be switched on to bus 1 or bus 2, corresponds with both systems.

The binary signals arriving from the reactor are received by the automation system (AS 230) by way of binary input units, processed according to the function in the program memory and sent out as messages by output devices. The messages are announced by a flashing light on the annunciator panel, signalised audibly by a horn, displayed on the screen and registered on the message recorder. Every signal change at a binary signal input creates a binary word. The binary word contains the signal address, an identifier "signal arrived" or "signal disappeared" and the date and time and is sent as a telegram via bus 1 to the OS 262 operating and monitoring systems and the long-term memory.

The analogue signals arriving from the process are received cyclically by analogue input devices, digitised with 10 bit resolution, labelled with the date and time and also sent as a telegram via bus 1 to the OS 262 operating and monitoring systems and the long-term memory.

Using the light pen attached to the monitor it is possible to call up various views of the plant and display them on the monitor. Fig. 2 shows the primary cooling circuit of the research reactor with the reactor in Pool 1, the primary piping system, the delay tank, the

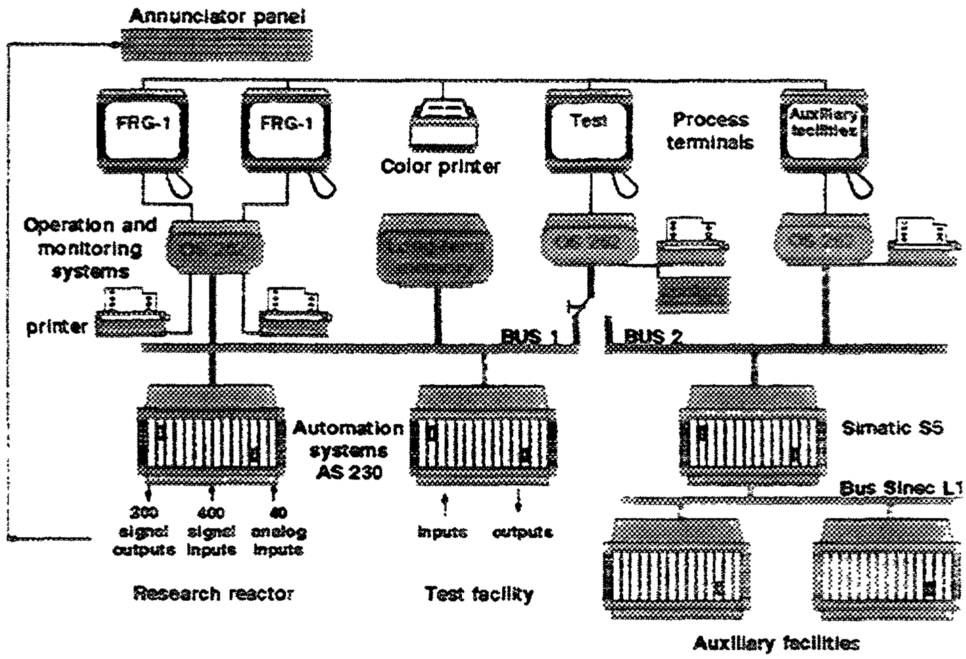

Fig.1: Information and signalling systems

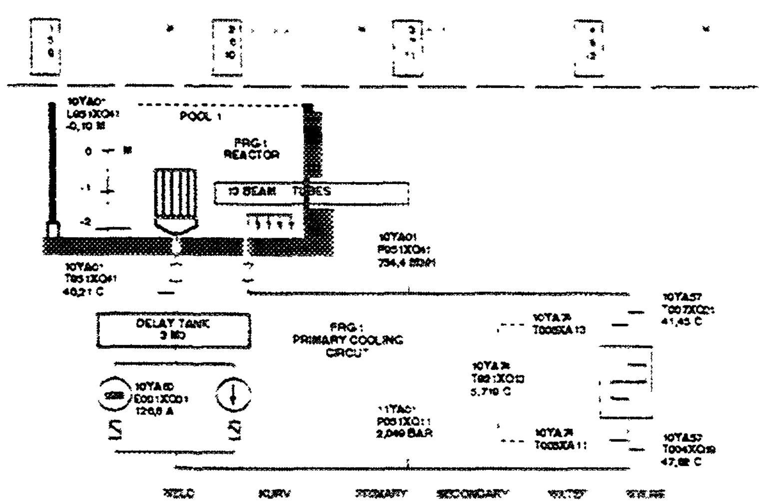

Fig.2: Monitor display, primary cooling circuit

primary pumps, the heat exchanger and the shutoff valves. The analogue measurements such as temperature, motor current, pressure, flow, and the difference in temperature across the heat exchanger are shown on the monitor display at the place where the measuring point is situated in the system.

Fig. 3 shows part of the redesigned control room from the viewpoint of the FRG-1 operator. The mosaic mimic diagram in the background shows the pool with the two research reactors FRG-1 and FRG-2, the water-cooling circuits and cleaning circuits, the reactor safety panels and the beam tube controls. At the top, the annunciator panels and television monitors are integrated. The console for the FRG-1 operator with the two process terminals can be seen in the foreground.

New emergency power plant

It is planned to construct a new emergency power plant for the reactor. The new emergency power plant is to replace the existing system, parts of which are 30 years old and faultprone; it will also provide emergency power for the planned new physical protection installations. At the beginning of 1989, KWU-Siemens was commissioned to draw up plans. The results of this planning are to be seen in Fig. 4.

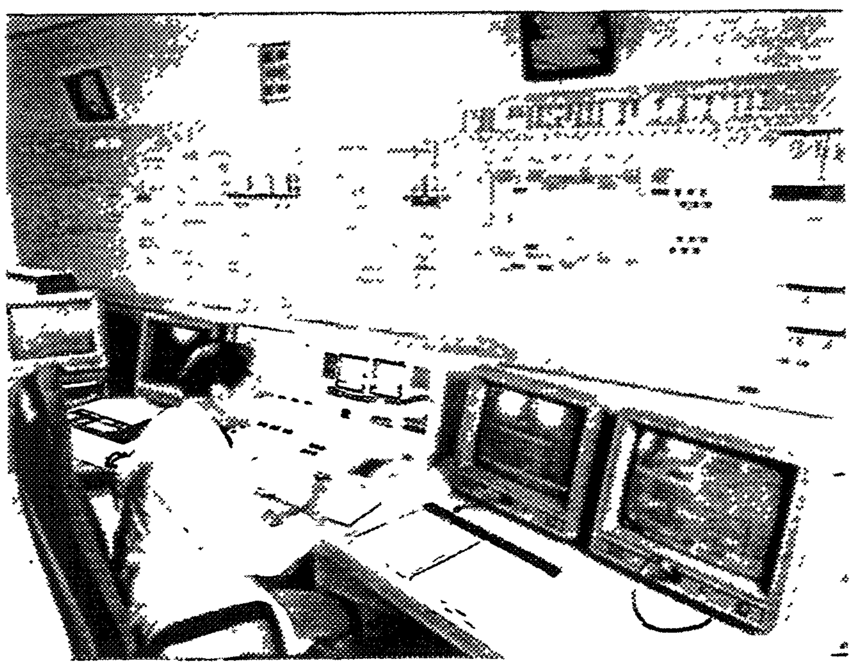

Fig. 3 Control room

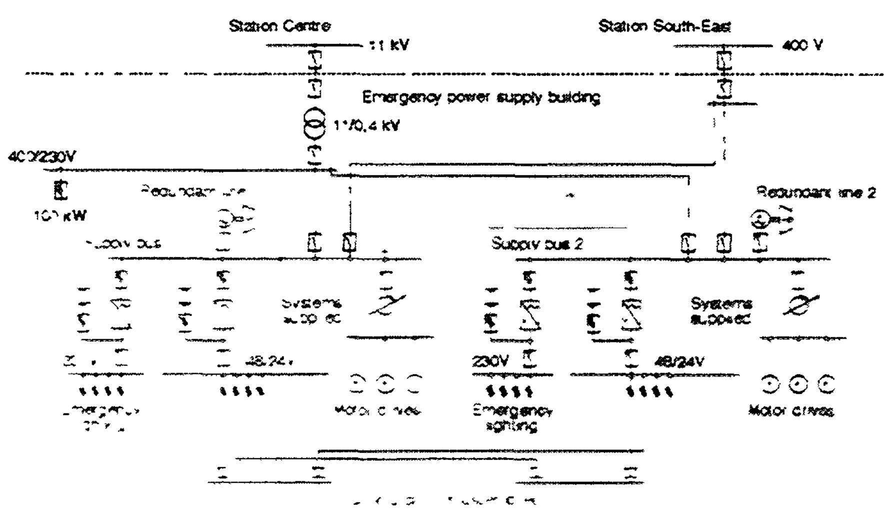

Fig.. 4: Emergency power supply

The power plant is designed as a two-line (redundant) system with two sets of diesel generators and the appropriate switchgear and control mechanisms and a third set of diesel generators as a standby. In normal operation the electric energy is fed into the emergency supply buses from Station Centre via an 11 kV line and an 11 / 0.4 kV transformer. If repairs are necessary a second feed is provided from Station South-East, but with lower capacity. The equipment requiring emergency power is connected to the emergency supply buses.

Voltage-sensitive equipment (e.g. motors, actuators) is supplied from two voltage-regulated distributors. The voltage is stabilised on each of the two lines by a constant voltage transformer which is fed from the emergency supply bus. By isolating this starting criterion for the diesel generators from the starting voltage and breakdown torque of the motors it is possible to adjust the diesel-starting criterion to a value that is low enough to largely prevent unintentional starting of the generators as a result of voltage depressions in the mains. All the other equipment requiring emergency power, including such that is not important from the point of view of safety, is supplied directly from the emergency supply buses.

Each emergency bus supplies a 230V and a 24V d.c. system with batteries and a rectifier. The 24V d.c. systems supply the control systems of the emergency power loads, the diesel generator sets and the switchgear. The 230V systems supply the emergency lighting and the three-phase switchgear.

The power requirement for each redundant line is approximately 220 kW. Taking cable losses and the degree of efficiency of the generator into account, the power required from the diesels is approximately 260-300 kW. The emergency power plant is to be installed in the emergency power supply building shown in Fig. 5, which is designed for two separate lines. The lines are separated from each other in accordance with Fire Protection Class F90. Two rooms on the ground floor house the three-phase and d.c. switchgear, the rectifiers and the control system. A third room houses the control system for the standby diesel (diesel generator 3). The batteries and the three diesel fuel tanks are installed in the basement.

The outside walls, ceiling and basement floor of the building have steel reinforcements and are designed to meet the internal lightning protection requirements of Protection Zone1. The building itself and the mechanical and electrical installations will be earthquake proof (SMK Scale I=7).

In May 1990 the application for a permit was submitted to the licensing authority. Since the authorities demand that the emergency power plant be earthquake-proof but have not yet issued specifications on the nature of the earthquake, which have yet to be decided, it is uncertain when a building permit will be issued.

External and internal lightning protection

In 1989 GKSS commissioned the preparation of plans for an external and internal lightning protection system ensuring that the reactor, and especially the reactor protection system that carries out the functions necessary in an operational emergency, are effectively protected against damage by lightning. These plans take into account all the measures that have been elaborated in national and international bodies in accordance with the state of the art and will provide the FRG-1 reactor with a lightning protection system that can be extended and is sure to meet the requirements of the future. They include the introduction of a protection zone model of the kind usual in EMC technology (EMC = electromagnetic compatibility).

Fig. 6 shows a simplified version of the lightning protection zone model. The reactor building is protected against direct lightning strike by an external system with air terminal rods and horizontal conductors, down leads and a equipotential bonding ring type. This equipment passes lightning strikes down into the earth. In conjunction with the external

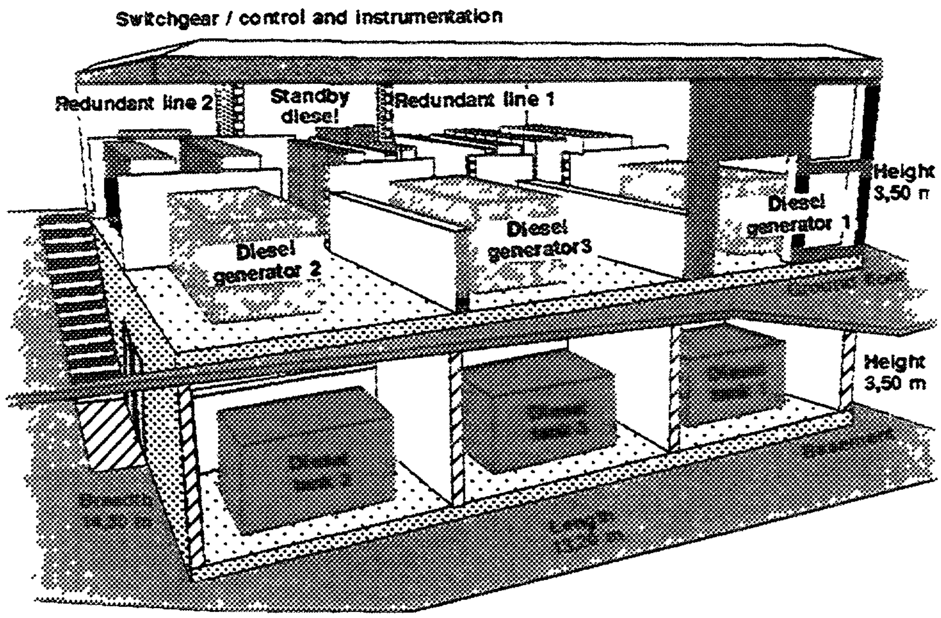

Fig. 5: Emergency power supply building

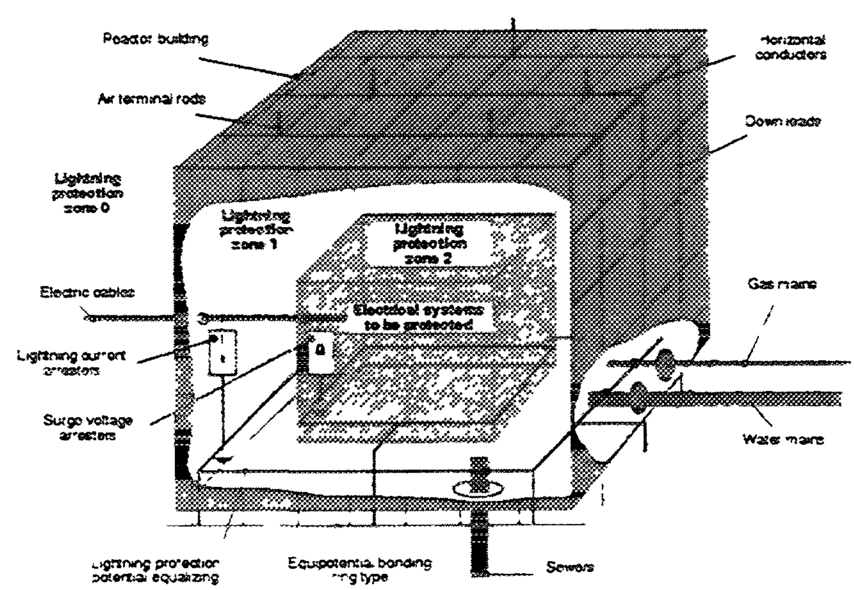

Fig.6 Lightning Protection Zones

lightning protection system, the walls of the building enclose the systems inside and have the effect of a Faraday cage. Outside is Lightning Protection Zone 0, inside is Lightning Protection Zone 1. At the interface between Zone 0 and Zone 1, all electric cables are connected to a arresters which passes lightning stroke current and overvoltage into the ground. All metal pipes such as gas and water mains, sewers etc. and metal constructions are connected to each other and to earth by an equipotential bonding conductor (lightning protection potential equalising). This means that lightning current in pipes and other constructions is also earthed.

Further protection zones can be set up inside the building. At the interface between the zones, all electric cables are connected to surge voltage arresters and all metal constructions and pipes are connected to each other by equipotential bonding conductors.

Fig. 7 shows the lightning protection zone model as applied to the reactor facilities. The buildings belonging to the reactor system (coloured yellow, Lightning Protection Zone 1) are protected against damage from lightning current and overvoltage by an system consisting of rods, horizontal conductors and down leads, connection of the electric cables to lightning current arresters and surge voltage arresters, and equipotential bonding conductors.

Protection Zones 1A and 2 are created within Zone 1 by means of further surge voltage arresters for the electric cables and equipotential bonding conductors. These zones contain the installations that are important to the safety of the research reactor.

Reactor protection system
The reactor protection system shown in Fig. 8 monitors important process parameters of the research reactor on a redundant and diverse basis. If a process parameter ex-ceeds or falls below the set limit (on two out of three modes), the reactor is automatically shut down and further protective functions are initiated if necessary. In order to improve the operational safety of the research reactor still further, a diversified "flow measurement device for the secondary cooling circuit" to assist "temperature measurement in the primary circuit" was added to the protection system, and the three binary transmitters for the measuring point "water level, basement" were replaced by three analogue, self-monitoring measuring channels. It is also planned to replace the three redundant measuring channels for "water level in the pool" by three channels that would not fail in an operational emergency.

Emergency and safety lighting
Fig. 9 shows the basic structure of the overall lighting system for the reactor facilities. The overall lighting system consists of normal lighting, emergency lighting and safety lighting. If the normal lighting fails (power cut), the emergency diesel generator (SSA) is started automatically and supplies the emergency lighting system with electric power after about 8 seconds.

The safety lighting, that has to be installed in addition to the general lighting, is part of the personal protection system. It is battery-maintained for 60 minutes and switched on within 0.5 seconds if the supply from the mains fails. This makes it possible to finish any

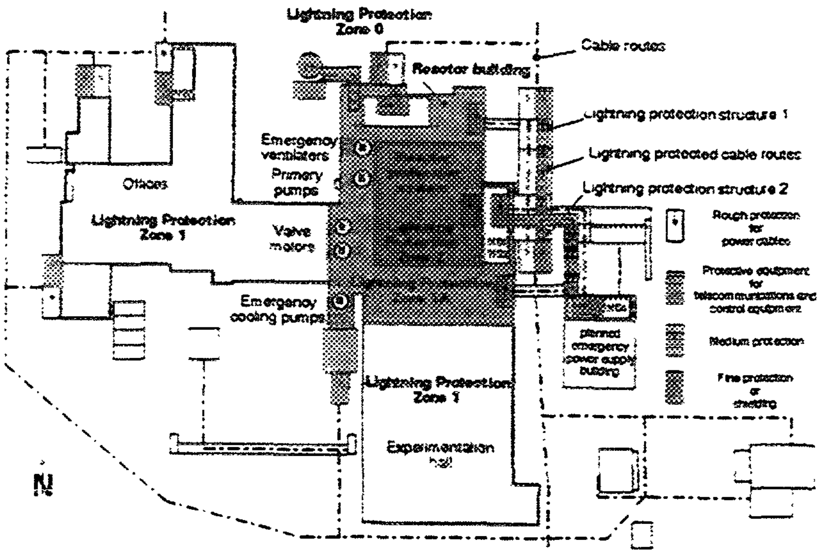

Fig.7 Lightning protection systems for the reactor facilities

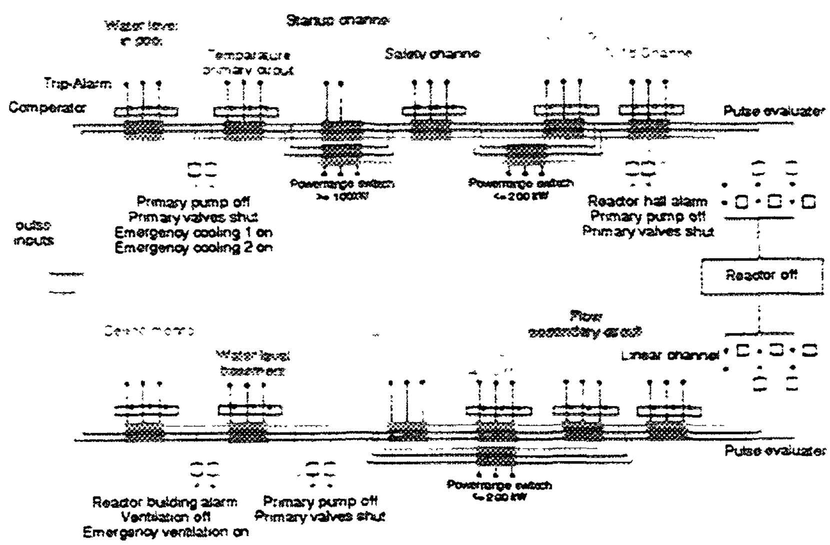

Fig.8 Reactor protection system

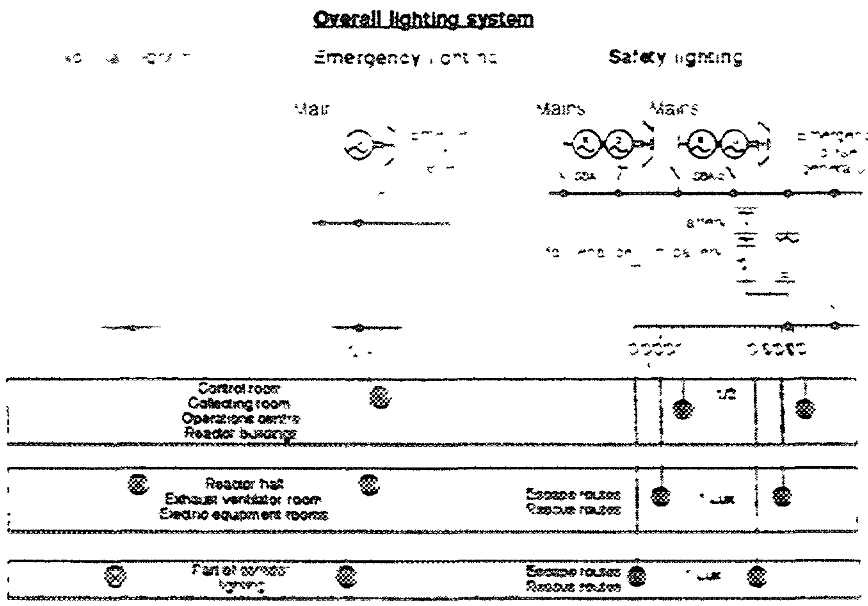

Fig.9 Lighting of reactor facilities

necessary work and leave the premises safely. As the present system no longer conforms to the current regulations, new safety lighting including the installations for the reactor building and the Hot Laboratory was planned in accordance with the latest regulations. The new system has already been constructed; installation will be completed in 1995.

Beam tube controls

FRG-1 has 12 beam tubes, which are shown in Fig. 10. Through the beam tubes the neutrons pass from the reactor core through the shield to the experimentation hall, where they are used for neutron beam experiments with the lead gate open. The beam tubes consist of the tube with the tube flooding system, the collimator with the collimator flooding system and the control mechanism for the lead gate.

For safety reasons and to enable a quick overview of the operational status, the structure of the beam tubes and the status messages are displayed on mosaic mimic diagrams in the control room and the experimentation hall. The signal lamps and the key-operated switches for "enabling" the controls are shown in Fig. 11. Each beam tube, lead gate and collimator has signal lamps of different colours for "Leak", "Full", "Open", "Shut", "Enabled" and "Empty", and each has a keyoperated switch for enabling the controls.

When the experiments were set up again in 1990 it became necessary to reconstruct the control system for the beam tubes as well. To ensure reliable functioning, contactor equipment was used for the control section and a programmable controller of proven design for the signalling section.

Control rods

FRG-1, shown in Fig. 12, is situated in a water pool some 9 metres deep and consists largely of the reactor core with its fuel elements, the frame for the reactor core, the bridge and the control rods. The control rods are made up of a control element with a fork-type absorber in the reactor core, a connecting piece to hold the absorber, an extension tube housing an extension rod, and the control rod drive for moving the absorber in and out.

It is no longer possible to replace some of the parts of the control rod drives, which have worked reliably for years. For this reason, and to eliminate some recognised weaknesses, it became necessary to redesign the control rod drives as well.

Fig. 13 shows the redesigned control rod drive. The forked absorber is coupled to the control rod drive by an electromagnet. The electromagnet with the coupled absorber is moved up and down by the 25 x 25 mm rectangular tube, the threaded rod and the motor gear assembly according to the direction of rotation of the motor. The extent of travel, which does not exceed 670 mm, is displayed in the control room via the torque indicator. The travel, the absorber coupled to the control rod drive and the restraint are supervised by limit switches. Moreover, the old relay control system was replaced by a programmable controller.

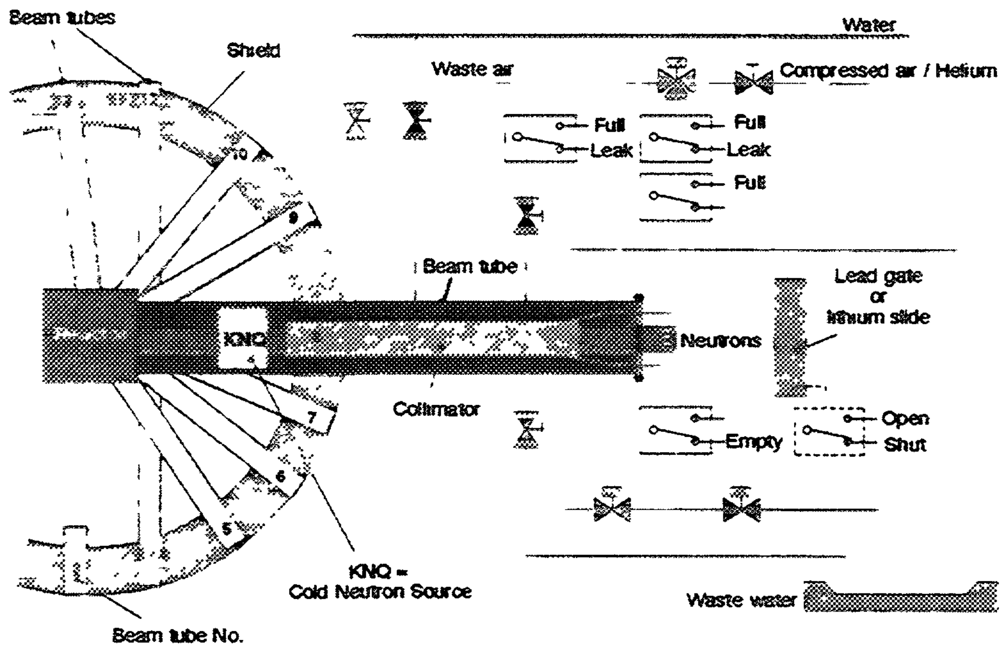

Fig.10 Beam tube controls

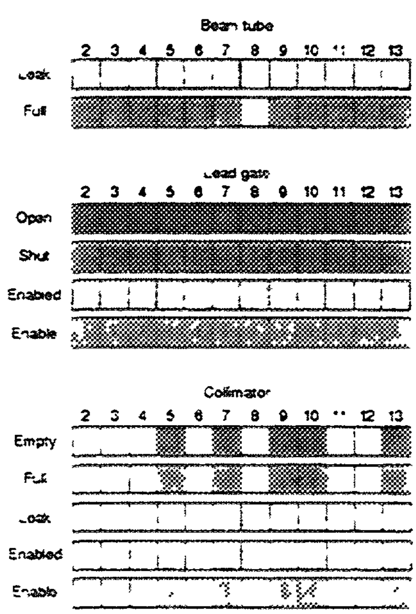

Fig.11 Beam tube messages

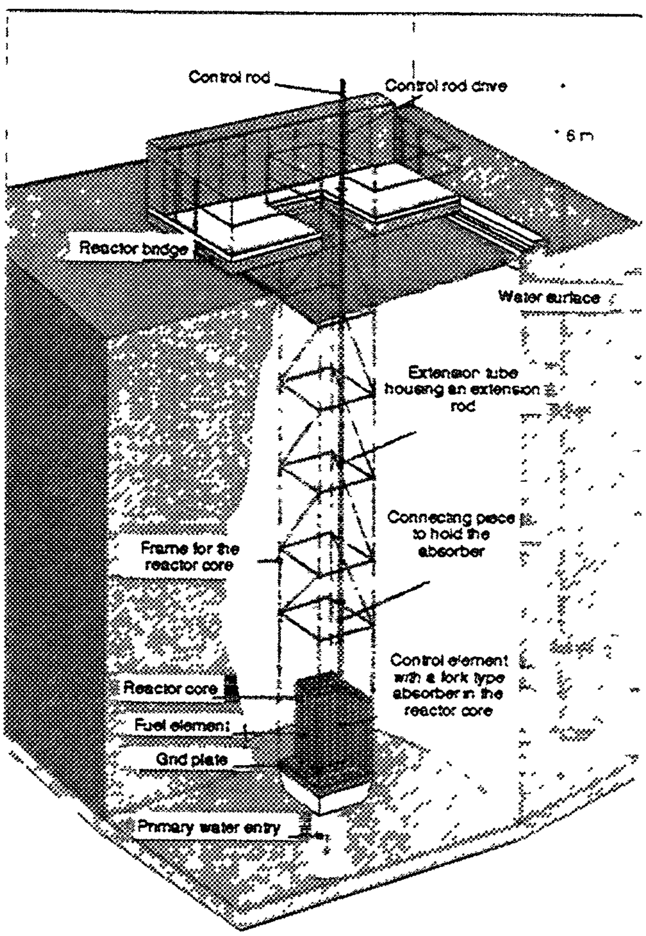

Fig.12 Control rod in the reactor

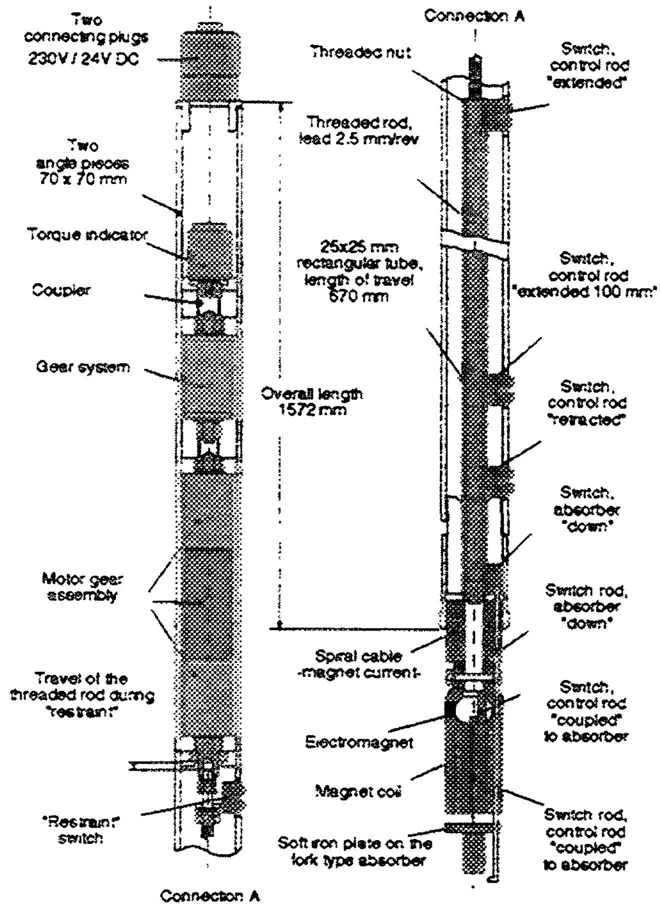

Fig.13 Control rod drive

Final remark

Optimum operation of the reactor is a necessity for continuous research work with neutrons. Comprehensive planning work was necessary in order to minimise the down time of the reactor by installing new equipment and systems and upgrading the existing ones. Computer-assisted work schedules were made out for the purpose, and most of the work was carried out during the normal times at which the reactor was shut down. Additional downtime of about five days per year was necessary for upgrading the plant and systems listed in Table 1.

The cost amounted to approximately 4.5 million DM, not including the new emergency power plant and GKSS' own contribution. With the new emergency power plant it is about twice as high.

Further improvements are already at the planning stage

XA9745522

INSTRUMENTATION AND CONTROL SYSTEMS OF THE JRR-3M

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Abstract

The instrumentation and control system of the JRR-3M consists of its constituent systems of neutron instrumentation, process instrumentation, reactor power control, reactor protection, engineered safety feature stating and process radioactivity monitoring The system is designed and constructed under the laws, standards and criteria of those days with a satisfactory quality assurance program A specific feature of the system is an extensive introduction of computer systems with a process computer and a management computer This contributes to lightening operators' loads satisfactorily

1. INTRODUCTION

The Japan Research Reactor No 3 (JRR-3) had been designed and built as the third research reactor in Japan with fully domestic technology for the first time Since its first criticality in 1967, it had been utilized for many neutron beam experiments and irradiation experiments In 1985, the JRR-3 was shut down, and its modification work was started so as to satisfy demands from users and researchers for a more highly performance research reactor

The design work for the modified Japan Research Reactor No 3 (JRR-3M) had begun in 1985 After the completion of conceptual design and final design, a hearing by the Science and Technology Agency began and an application for permission for alteration of the JRR-3 was accepted in 1984 The JRR-3M reached first criticality in March 1990 Thereafter the JRR-3M has been operated satisfactorily

This report introduces the structure and functions of the instrumentation and control systems of the JRR-3M

2. SPECIFIC DESIGN FEATURES

Although a design of the instrumentation and control (I&C) system of the JRR-3M is almost same as that of the other former research reactors, it is characterized by an extensive introduction of computer systems This brings about an effective reactor operation and lightening of the load on reactor operators As the design of JRR-3M reactor systems adopted basic ideas of the Guide for Safety Design of Light Water Nuclear Power Reactor Facilities, the design of I&C systems also followed it

The computer system of JRR-3M consists of a management computer system and a process computer system The management computer system works mainly for reactor-operation-data logger and plays some roles to print out the operation data, display a trend graph of process variables and store the data The process computer system works to support start-up and shut-down procedures of the reactor and manage data from process instrumentation

3. I&C SYSTEM STRUCTURE AND FUNCTIONS

The structure of I&C system of the JRR-3M is presented in Fig 1 The outlines of the constituent systems are as follows

1) Neutron Instrumentation System

The neutron instrumentation system consists of a neutron measurement and control system and a safety protection system It is designed to measure the neutron flux not only during the reactor operation but also during the reactor shut-down period to have necessary information for reactor operation and safety protection The neutron measurement and control system consists of two start-up channels and two linear power monitoring channels The safety protection system consists of two logarithmic power monitoring channels and two power level safety channels

2) Process Instrumentation System

The process instrumentation system of the JRR-3M is consists of an instrumentation and control system and a safety protection system It is designed to measure process variables, such as flow rate, temperature, water level, pressure and so on, to provide information which is necessary for reactor operation and safety protection The measuring methods adopted in the system are of conventional ones but safety important components used are specified as "reactor grade"

3) Reactor Power Control System

The reactor power control system consists of a reactivity control system, the linear power monitoring channels of the neutron instrumentation system and the reactor shut-down (control rod system) It is designed to automatically operate control rods to compensate reactivity changes caused by loading and unloading of irradiation samples, accumulation of Xenon, changes of coolant temperature, burn-up of fuels and so on in reactor operation

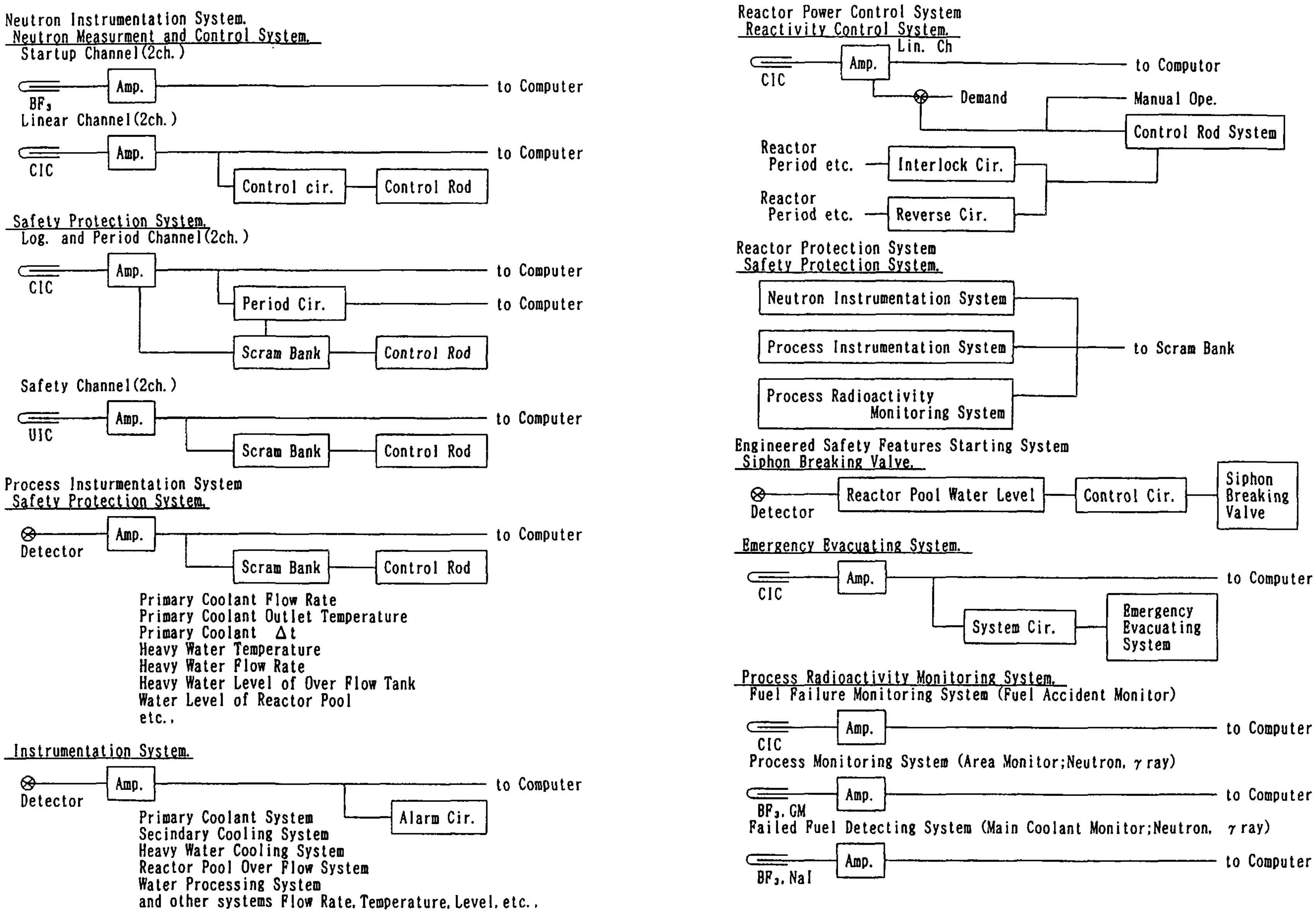


FIG. 1 The structure of I&C system.

4) Reactor Protection System

The reactor protection system is designed to automatically operate the control rod system and shut down the reactor in safety in the case of anticipated operational occurrences and accidents. The reactor protection system is fully covered with the safety protection system and is operated by signals from the neutron instrumentation system, the process instrumentation system and the process radioactivity monitoring system.

5) Engineered Safety Features Starting System

If there is the possibility that a damage or a trouble of the reactor components causes to damage the reactor fuels and furthermore release the radionuclides to the atmosphere, the engineered safety features starting system must operate features which protect the fuel element and mitigate the release of radionuclides. The system is activated by the process instrumentation system and the process radioactivity monitoring system and automatically operates siphon break valves, an emergency evacuation system and some isolating valves in a ventilation system.

6) Process Radioactivity Monitoring System

The process radioactivity monitoring systems consist of a process monitoring system and a failed fuel detector system

4. SAFETY DESIGN

The I&C of JRR-3M is designed under the basic philosophy as follows. This philosophy is based on the Guide for Safety Design of Light Water Nuclear Power Reactor Facilities in Japan.

-It is designed with redundancy so that a single failure or any of the components or channels that comprise the system does not result in loss of safety functions of the system.
-It is designed such that the important components or channels that comprise the system are separated each other taking into account the independence between them thereby preventing loss of safety functions.
-It is designed to be functionally separated each other so that the system does not lose its safety functions.
-It is designed to allow the reactor facilities to be settled in a state of safety eventually in case of driving power loss.
-It is designed such that the components or channels that comprise the system are fed by the fail free power.
-It is designed such that the components or channels that comprise the system are capable of being tested in general.

5. SYSTEM DESIGN AND COMPUTER SYSTEM

The I&C system of the JRR-3M is considered to be designed such that the man-machine-interface lightens the load of the reactor operators

The reactor control room is apart from the reactor building and is out of the radioactivity control area A computer system is highly introduced, and layouts of the instruments and the color graphic panels in the control room are considered so as to improve the operability of the components of not only the instrumentation and control systems but also the reactor cooling systems Photograph 1 shows a control console and instrumentation racks in the control room, and Fig 2 the layout of control console

In the center of the control console, there are a control unit of the control rods and a panel that shows control rod positions and power monitoring channels outputs There are two display monitors for process monitoring in each side of the center console To the left of the console is a ITV monitor to observe inside of the reactor room, etc To the right is a CRT for displays and monitoring of the management computer system

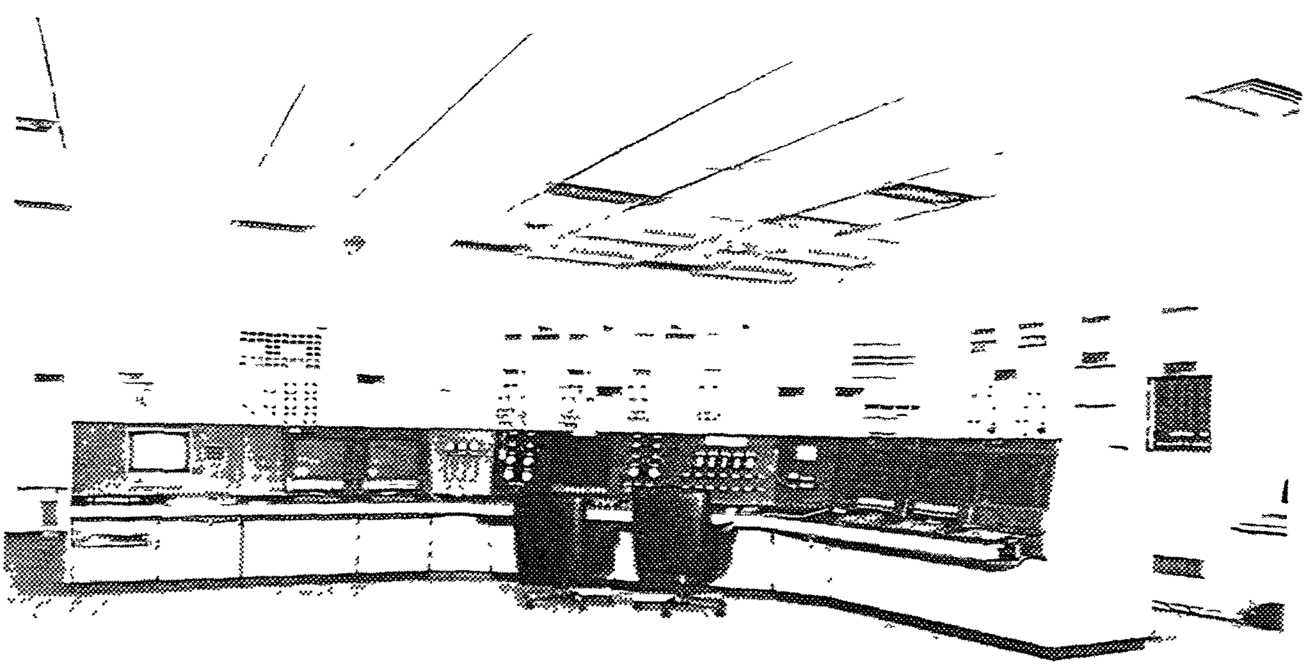
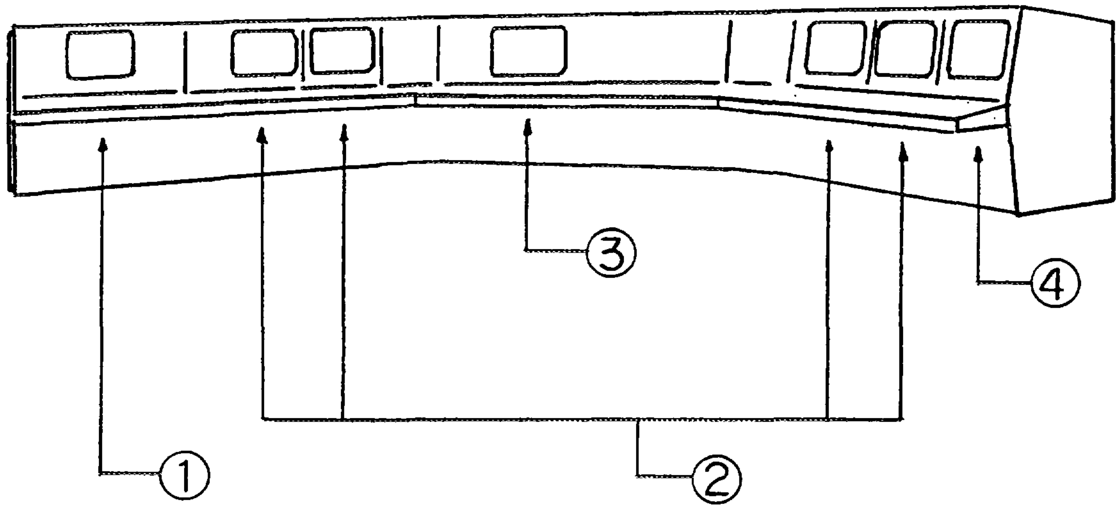


PHOTO. 1 Control console and instrumentation racks in the control room.



① ITV monitor; Inside view of the reactor pool and the reactor room, doorway of the reactor room and so on.

② Displays and monitoring of process variables; Trend graphs of the process variables, Status of the equipments e.g. run or not, alarm setpoints.

③ Main Display ; Reactor power and rod positions.

④ Displays and monitoring of management computer system; Technical data (reactivity, neutron flux, thermal power, etc.), trend graphs of the past operation cycles, print out of operation data.

FIG. 2 The layout of control console.

The computer system of the JRR-3M consists of the process computer system and the management computer system. The process computer system operates and monitors the all instrumentation and control systems. By using this system, it is possible to remotely operate pumps, valves and so on in the reactor cooling systems, change alarm settings and display start-up and shut-down procedures. The management computer system works mainly for operational data logger. Other functions of the system are as follows: technical calculations, long-term storing of the operation data, print out of the operation data in regulated formats, automatic storing of the operation data in a case of a reactor scram and fault tree analysis(FTA) in a case of alarming. Figure 3 shows a simplified diagram of the computer systems.

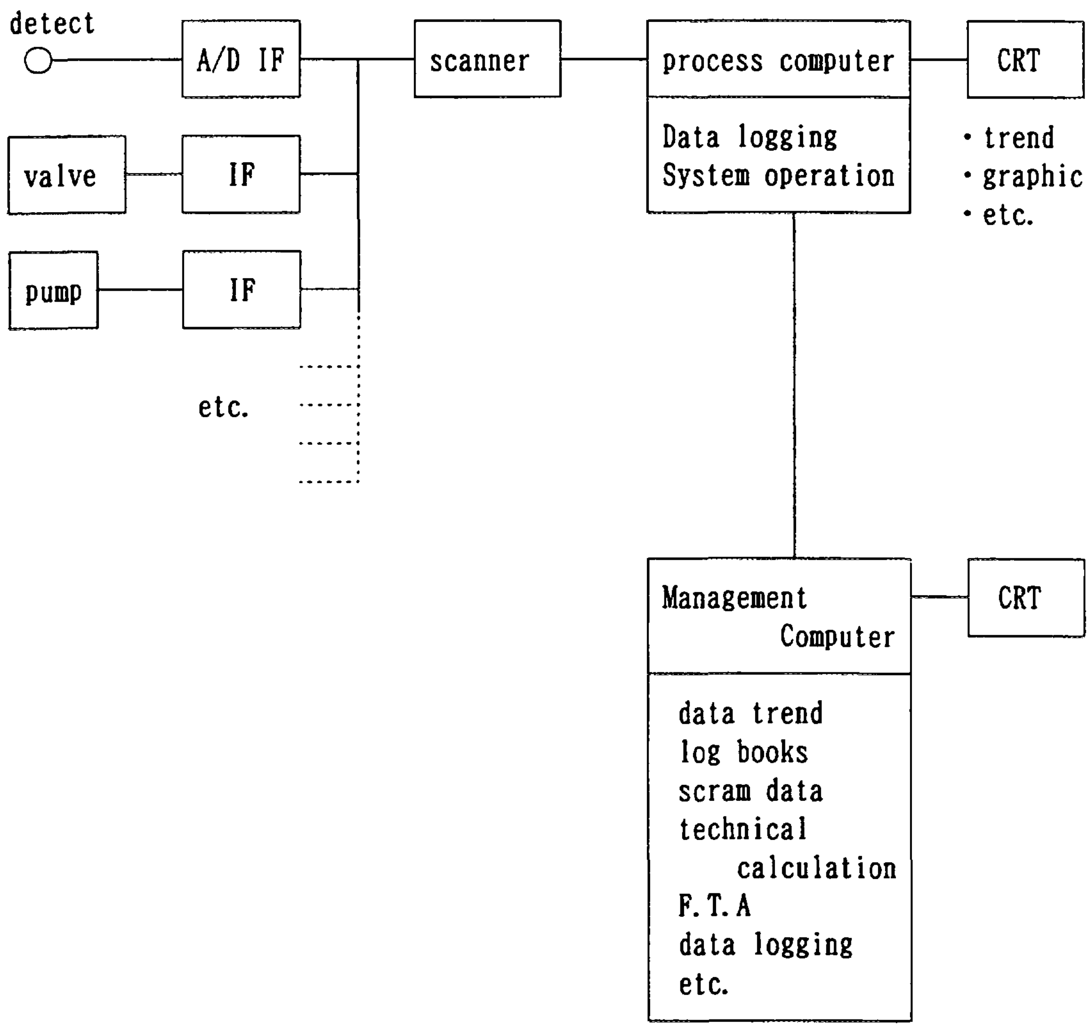


FIG. 3 Simplified diagram of the computer system.

6. QUALITY ASSURANCE

A quality assurance activity for modification of the JRR-3M has three main processes.

The first QA process is achieved by JAERI and investigates adequacy of design, safety design and others comparing with laws, standards and criteria. It has two stages of investigation: investigation in the department and in the institute.

The second QA process is achieved by the regulatory body: the Reactor Regulation Division, the Reactor Safety Bureau in STA. The process is necessary to be admitted to build and modify a reactor. Furthermore, in order to get the admittance, the Reactor Regulation Division inquires an investigation to the Atomic Energy Commission and the Atomic Energy Safety Commission organized by the Japanese government and is reported back the findings of the committees.

The last QA process is achieved by the regulatory body, contractors and JAERI The regulatory body investigates the design and methods of construction in regard to the important features and components described in the law for the regulations of nuclear source material, nuclear fuel material and reactors In order to set about construction work, it is necessary to obtain sanction on it The design and methods of construction are inspected in three stages inspection by contractors, by JAERI and by the regulatory body The other features and equipment are inspected by contractors and JAERI following the respective QA program

7. CONCLUSION

The JRR-3M is designed and constructed under the latest knowledge prepared in Japan The instrumentation and control systems of the JRR-3M are characterized by the extensive introduction of computer system The computer system contributes to lightening the load on reactor operators and safety operation

THE DALAT REACTOR INSTRUMENTATION AND CONTROL SYSTEM AFTER RENOVATION

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Abstract

The Instrumentation and Control System (ICS) of the Dalat 500 kW research reactor was designed and manufactured by the former Soviet Union and has been put into operation since November 1983.

After ten years of reactor operation, the system has revealed some features of ageing and obsolescence. As its components and equipment were produced in the 70's, some integrated circuits (IC) and equipment were hard to find on the market and as a consequence maintenance work of the system became a big difficulty. To solve this problem the Dalat Nuclear Research Institute has asked for assistance from the IAEA through the TC Project, VIE/4/010 - "Renovation of the Dalat Reactor Instrumentation and Control System". The project has the following main objects:

- Reassessing the ICS by a general inspection on reliability, ageing of electronic components and equipment and spare parts intended for a 10-year usage. These reassessments will serve as base to lay out detail duties for the project implementation.

- Renovating some electronic boards/blocks/sub-systems which mainly affect the system reliability in using more modern IC and equipment.

This paper will be devoted to the main characteristics of the renovated system.

DESCRIPTION OF THE SYSTEM

The instrumentation and control system of the Dalat research reactor can be divided into four main sub-systems as follows:

- Neutron Flux Control sub-System - NFCS
- Control Console Display sub-System - CCDS
- Control Logic sub-System - CLS
- Process and Instrumentation sub-System - PIS

1. NEUTRON FLUX CONTROL SYSTEM - NFCS

The NFCS of the Dalat research reactor has the following functions:

- Measuring reactor power and period;

- Giving out analog signal proportional to the unbalance between reactor and setting powers. This signal is transmitted to the CLS for automatic reactor power regulation.

- Giving out alarm and scram signals on power and period of each measuring channel and failure signals of these channels. These signals are led to the CLS to be elaborated and to emit acting alarm and scram signals.

The reactor neutron flux is measured in 3 ranges:

- Source Range (SR): $(10^{-8} - 10^{-2})$%Pn (Pn = 500 kW)
- Intermediate Range (IR): $(10^{-3} - 10)$%Pn
- Power Range (PR): (1 - 120)%Pn.

To control neutron flux of the reactor 9 neutron detectors are used: 6 fission chambers with gamma compensation, KNK-15 and 3 ionization chambers with gamma compensation, KNK-3. The KNK-15 chambers operating in pulse mode are used for 3 measuring channels of source range and 3 measuring channels of intermediate range. The KNK-3 chambers in current mode are used for 3 measuring channels of power range. These chambers were positioned in dry vertical tubes outside the graphite reflector.

Because neutron flux acquisition and processing are based on the "2 out of 3" selection principle, the system has three identical electronic units - IAPU1, IAPU2, IAPU3 (Information Acquisition and Processing Units).

The layout of the neutron flux control system is illustrated in *Fig. 1*.

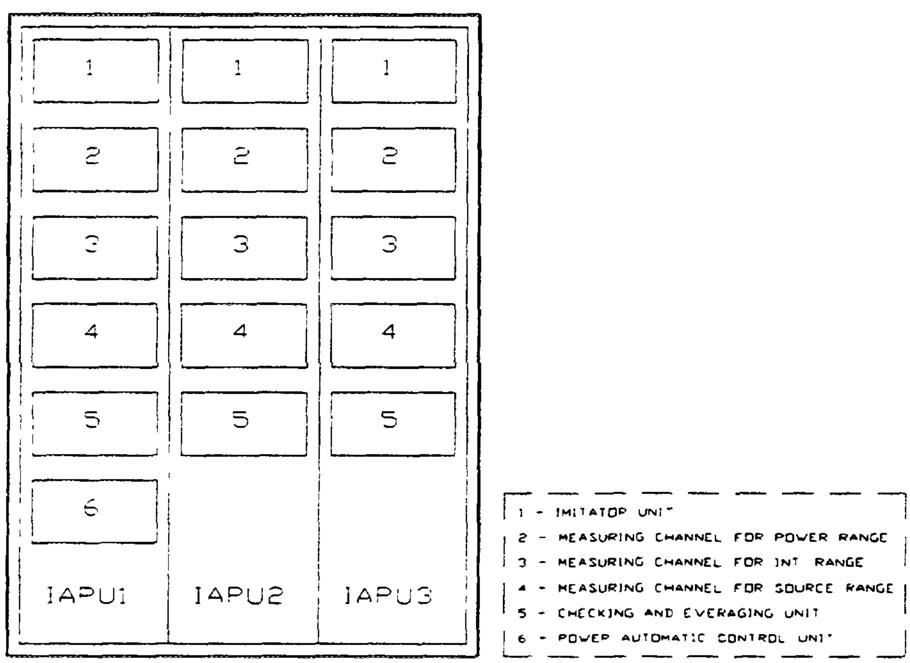

Fig.1. Position of the electronic units of the NFCS

Each IAPU contains an imitation block used for testing the working capability of the unit; 3 measuring channels connected to 3 neutron detectors; a checking and averaging block. The measuring channels give out power signals (in logarithmic scale for source and intermediate ranges and in linear scale for power range), period signals, range signals in which the reactor is operating, alarm and scram signals on power and period.

The checking and averaging block is used to check the operation status of the unit, to emit failure signals and the beginning and ending signals of the range in which the reactor is running; to average power and period values; to give out alarm and scram signals. These signals are transmitted to the CLS and the CCDS.

Fig. 2 presents a typical block diagram of a measuring channel in the source and intermediate ranges. *Fig. 3* is for a channel in the power range.

In particular, an automatic regulating block is added to the IAPU1 in order to automatically regulate the reactor power. Reactor power signal is compared with the power value set at the reactor control console. The unbalance value is sent to the power regulating block in the CLS.

Renovation of some important boards of the NFCS is included in the project VIE/4/010. The design obeyed some principles such as keeping unchanged the old mechanical standard and the technical specifications of the old boards, but making them more reliable.

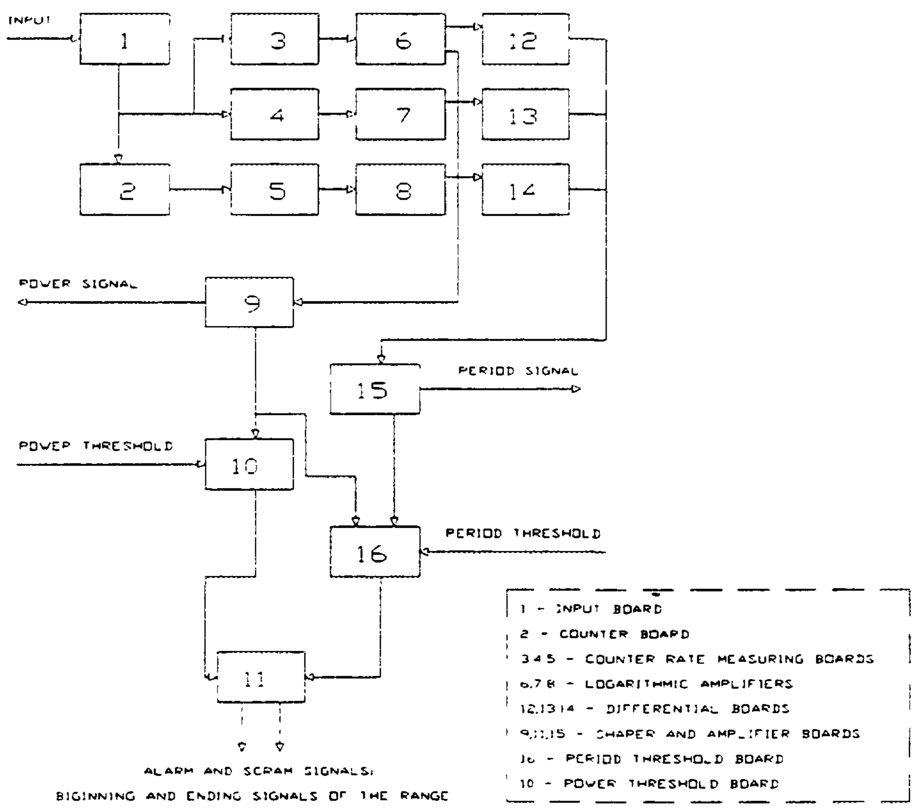

Fig.2. Block diagram of the measuring channel in the source and intermediate ranges

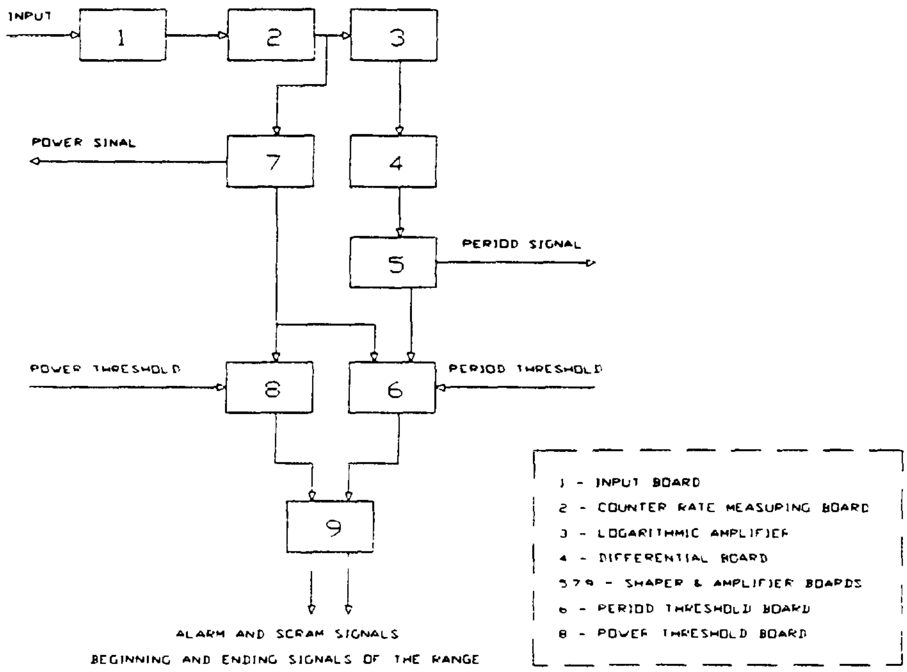

Fig.3. Block diagram of the measuring channel in the power range

2. CONTROL CONSOLE DISPLAY SYSTEM - CCDS

The reactor parameters display system is used for:

- Presenting on a monitor screen by digit and line modes important parameters of the reactor such as: power and period values of 9 measuring channels in three ranges; the averaged value of reactor power in each range; beginning and ending values of each range; safety threshold values of reactor power and period; reactor negative period value.

- Indicating on a digital indicator the averaged values of reactor power and period in the power range.

- Recording reactor power and period on a 2-pen recorder type SK12-3.

- Managing and saving data in a PC hard-disk.

Because of low reliability, low quality of indication, the reactor control console display system was totally designed and constructed according to the project program.

The block diagram of the display system is illustrated in *Fig. 4*. The system consists of:

- The isolation amplifier block designed for electric isolation between the NFCS (Transmitter) and the CCDS (Receiver).

- The analog multiplexer block (64 to 16 lines). In our case there are only 57 analog signals to be elaborated.

- The active filtering block is used to filter noises accompanying input analog signals.

- An add-on industrial multifunction board - PCL 812PG with a 16-input 12-bit ADC, 16 digital inputs, 16 digital outputs.

- An industrial PC computer with super VGA monitor is the central unit of the system for controlling acquisition, processing, presentation, management and saving data.

All electronic boards of the system were designed on the EUROCARD standard.

The software of the system was written in Pascal language. Data are saved in hard-disk 120 MB or in floppy-disk 1.2 MB.

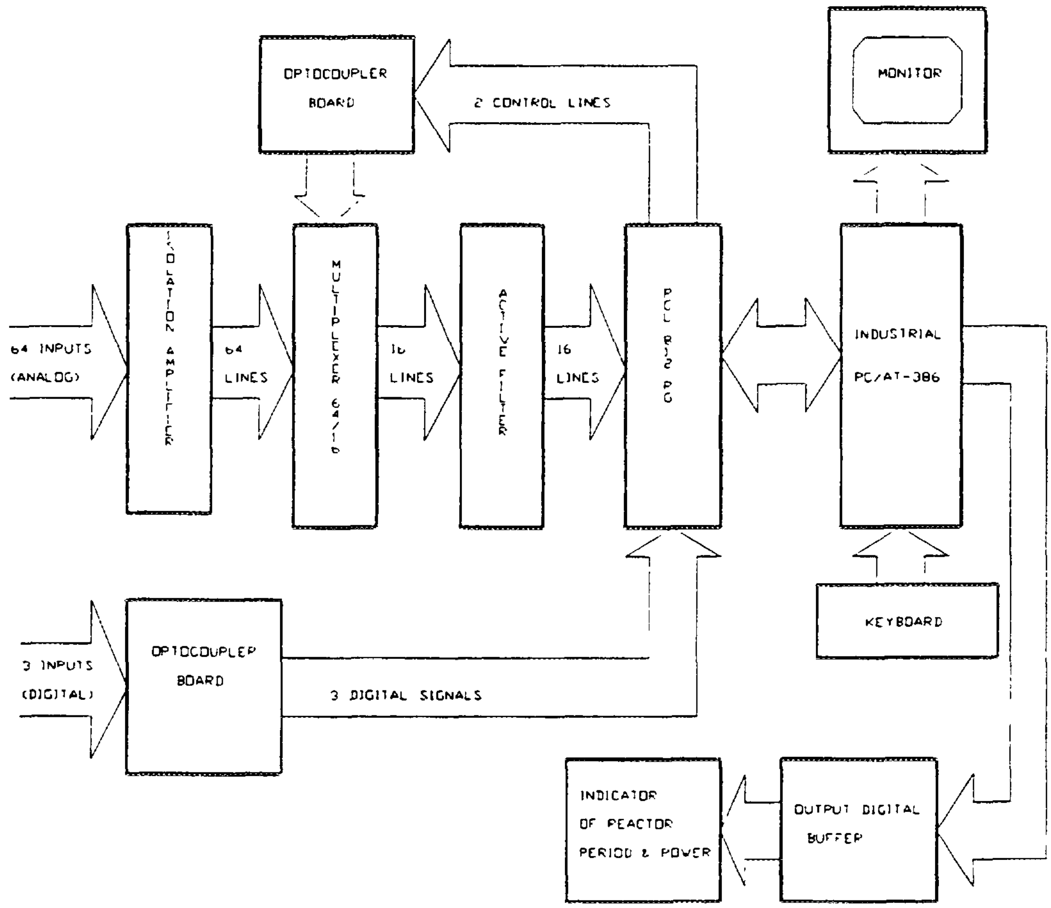

Fig.4. Block diagram of the reactor display system

3. CONTROL LOGIC SYSTEM - CLS

The CLS has the following functions:

- Controlling the reactor in manual and automatic modes.

- Shutting down the reactor at power or period scram signal, scram signals of the PIS or failure of the city electricity network.

- Indicating on control console status information of the ICS.

To control the reactor 7 control rods are used: 2 safety rods (SR), 4 compensating rods (CR) and one automatic regulator rod (AR).

The signals that go to the CLS are:

- Control signals in the manual mode from the control console.

- Alarm and scram signals on period in the intermediate and power ranges from the NFCS.

- Alarm and scram signals on power in the power range from NFCS.

- Failure signals from IAPUs of the NFCS.

- Signal proportional to the unbalance between reactor and setting powers from the IAPU1 of the NFCS.

- Alarm and scram signals on flow-rates of primary and secondary loops and scram signal on the reactor water level from the PIS system.

The safety and compensating rods made of B_4C move in wet stainless steel tubes with a 27 mm outer diameter and 1 mm thickness. The theoretical worth of each rod is about 3.2%, moving speed is 3.4 mm/s. In the manual mode, CR rods can be withdrawn step by step with a step length of 20 mm. The automatic rod made of stainless steel has 0.4 % worth and 20 mm/s maximum moving speed.

Servo-drives of SR and CR are of the same type and consist of a 48 V DC motor, magnetic clutch, braking mechanism, rod way switches, rotating drum and potentiometer for position indication. The servo-drive of AR has a 110 V AC motor, magnetic clutch, rod way switches, rotating drum and potentiometers. AR speed depends on the unbalance value between reactor and setting powers. The moving range of the control rods is 650 mm.

In a case of scram or failure of the city electricity network, currents fed to coils of magnetic clutch in servo-drives of SR and CR are cut off. These rods freely drop into the reactor active core.

To increase accuracy and reliability of rod position indication, the indicators and electronic board for them have been replaced. The indicators are of a digit and bar-graph type - INDICOMP A2000.

To control SRs, CRs and AR, there are respectively the SR control logic block, CR control logic block and AR control logic block.

3.1. Control logic block for safety rods

The SR control logic block has the following functions:

- Controlling the movement of two safety rods.

- Processing logically input scram signals and creating the shut-down signal. This signal creates SRs dropping and is led to the CR and AR control logic blocks to cause CRs' dropping and to force AR moving down with maximum speed.

- Processing logically input alarm signals to emit audio alarm and lighting signals and prohibit moving up of CRs and AR.

The block diagram of the SR control logic block is illustrated in *Fig. 5*.

Obeying the "2 out of 3" principle, the SR control logic block has three identical Logic Processing Blocks - LPB1, LPB2, LPB3.

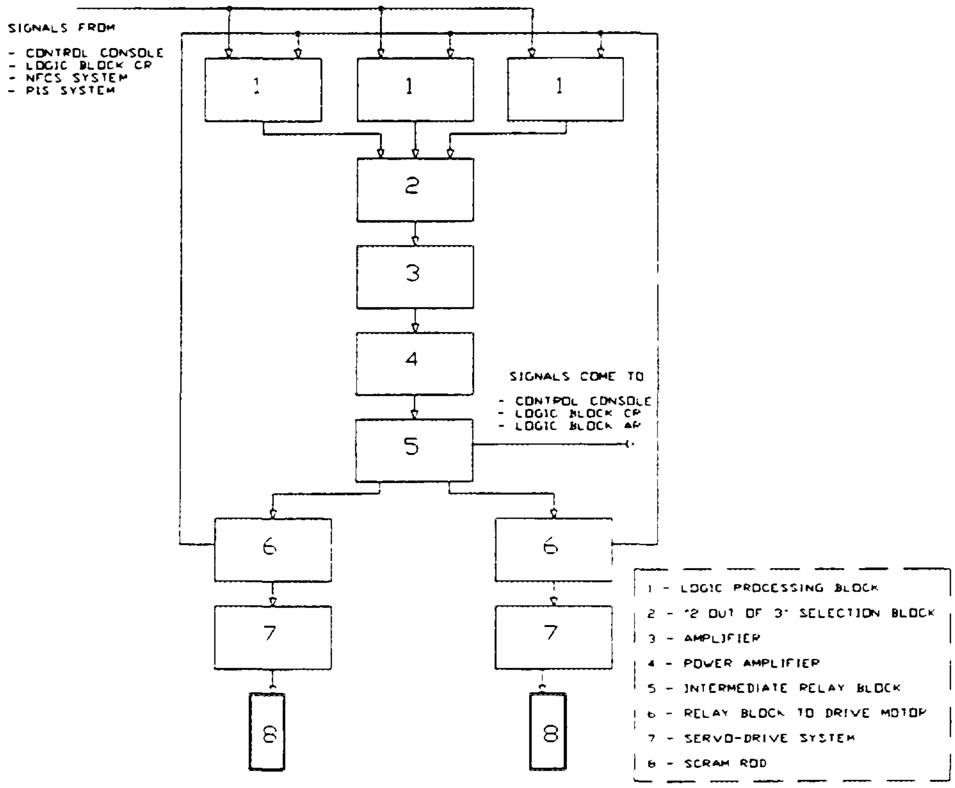

Fig.5. Block diagram of the safety rods control logic block

Input signals of the SR control logic block are *(Fig. 6):*

- From the NFCS: alarm and scram signals created in IAPUs on period in the intermediate and power ranges (TwIR1,2,3, TwPR1,2,3, TsIR1,2,3, TsPR1,2,3); alarm and scram signals elaborated in IAPUs on power in the power range (PwPR1,2,3, PsPR1,2,3); signals of range in which the reactor is operating (IR, PR); failure signals of IAPUs (W1,2,3).

- From the control console: scram signal by pressing the scram button B1; reset signal from RESET button; halt signal from HALT button; up and down signals in the manual mode.

- Two scram signals from two buttons B2 and B3 on the reactor platform.

- From the PIS: alarm and scram signals on flow-rates of the primary and secondary cooling loops (Fw1, Fs1, Fw2, Fs2); alarm and scram signals on reactor water level (Lw and Ls).

- Signals on top and bottom positions of the CRs and AR from their servo-drives.

Signals from LPBs are led to 2/3 selection boards. Output signals of these boards are continuously transmitted to amplifier boards and then to power amplifier boards. Output signals of power amplifiers are used to drive relays in intermediate relay boards. These relays will control SRs, CRs and AR and drive indication lamps on the control console.

Functional diagram of reactor protection unit is presented in *Fig. 6.*

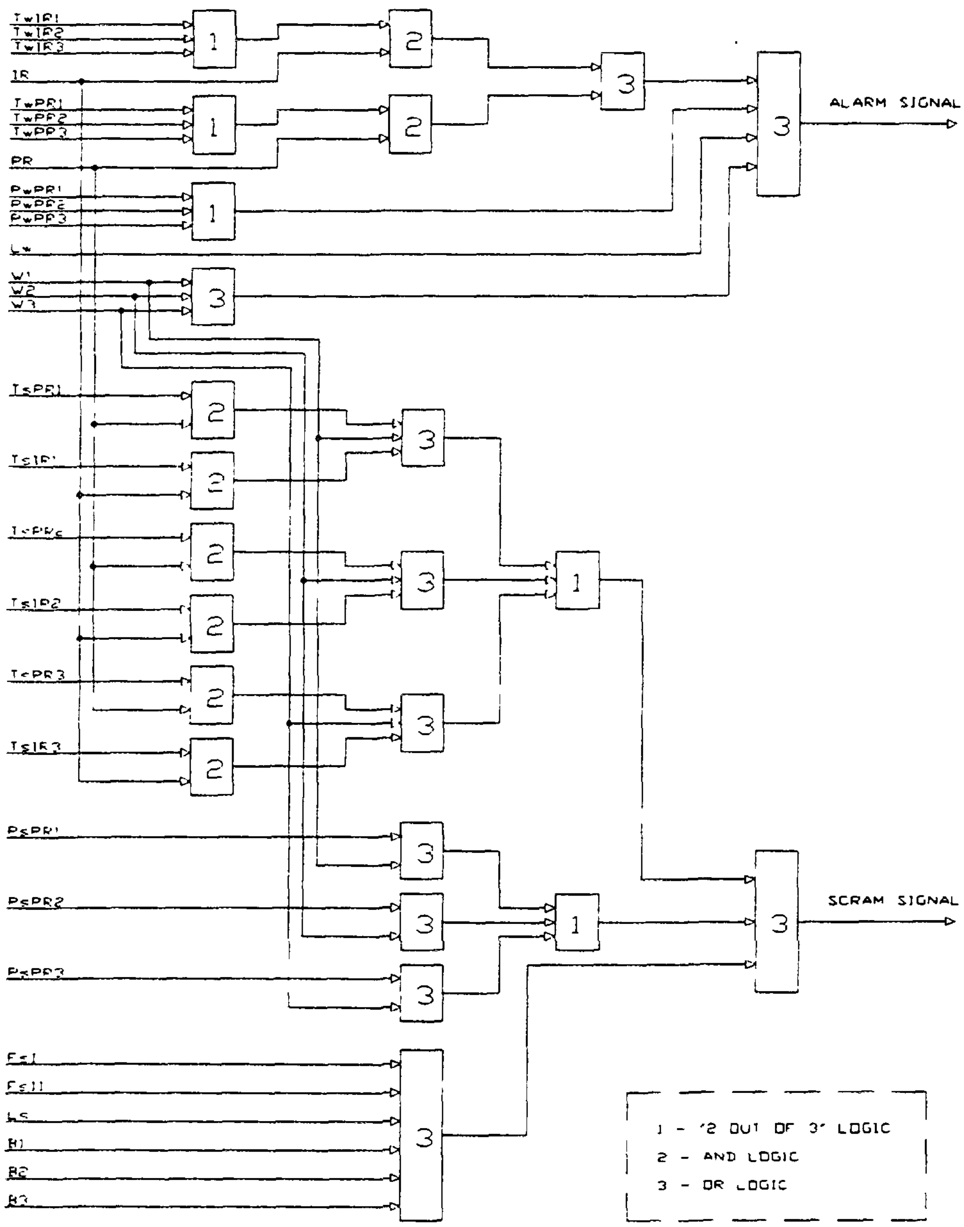

Fig.6. Functional diagram of reactor protection unit

In the framework of the project, all relays in intermediate relay boards and transistors in the amplifier boards have been replaced by items with a higher quality and reliability. Decreasing currents fed to LED indication and consequently decrease in power consumption in all electronic boards has an important meaning in the reliability improvement of the CLS.

3.2. Control logic block for compensating rods

The CR control logic block has the following functions:

- Controlling movement of four compensating rods.

- Dropping CRs in case of scram.

- Giving out signals on CRs' status and transmitting them to SR and AR control logic blocks and to the control console for indication.

Intermediate relays, power amplifier boards have been renovated or replaced.

The block diagram of the CR control logic block is illustrated in *Fig. 7.*

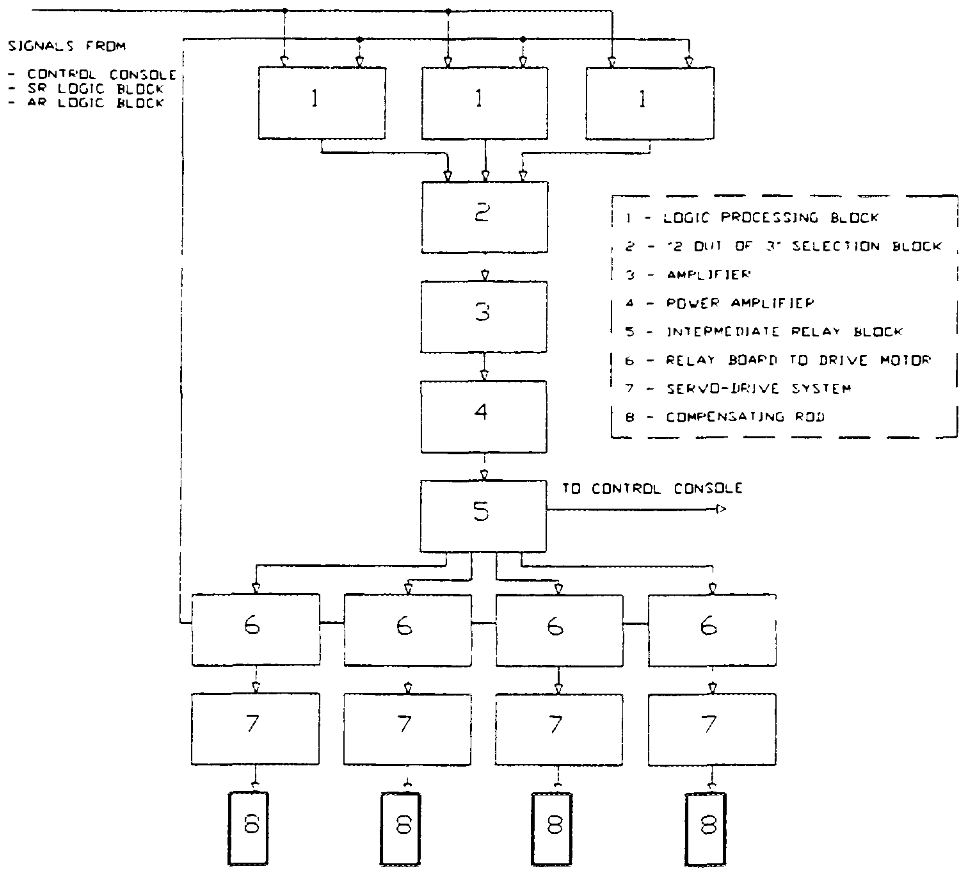

Fig.7. Block diagram of the compensating rods control logic block

3.3. Control logic block for automatic regulator rod

The AR control logic block has the following functions:

- Regulating reactor power in the automatic and manual modes.

- Giving out signals on AR position for presentation on the control console and transmitting them to the CR control logic block.

- Forcing the AR to move down with maximum speed in case of scram.

To control the servo-drive of the automatic regulator rod, the magnetic amplifier is used.

The block diagram of the AR control logic block is illustrated in *Fig. 8.*

As in the SR and CR control logic blocks, intermediate relays and power amplifiers of the AR control logic block have been renovated or replaced.

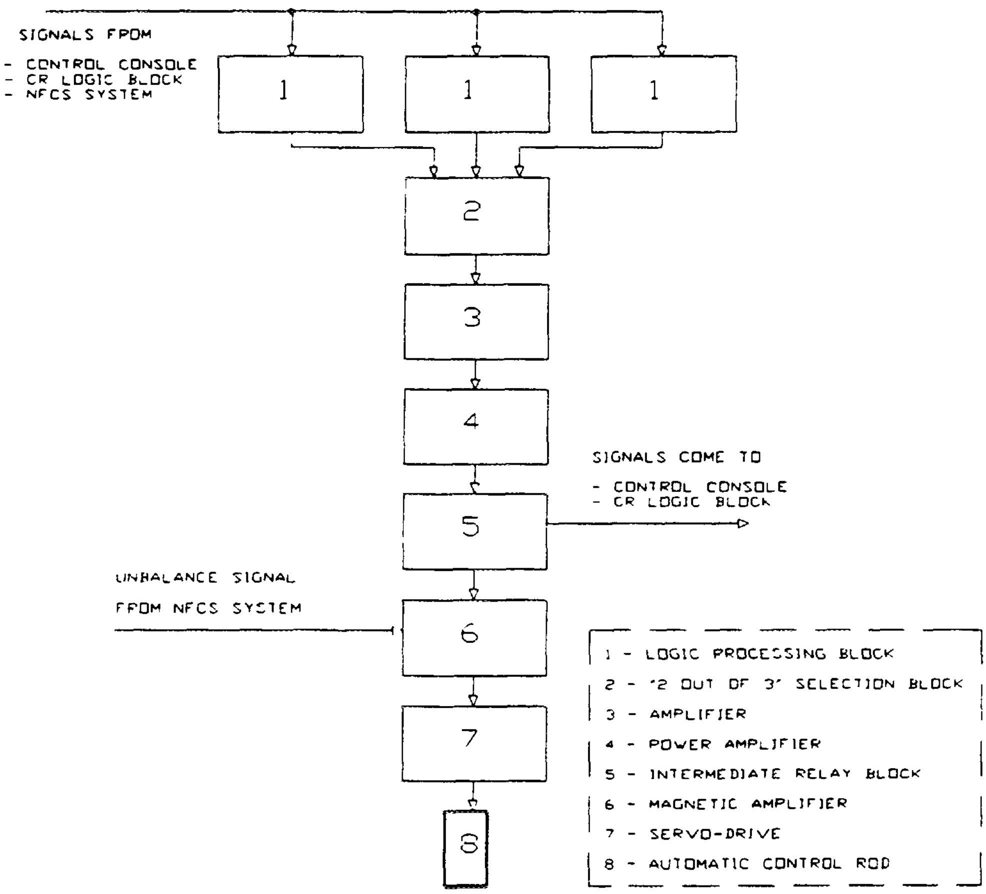

Fig.8. Block diagram of the control logic block for automatic regulator rod

4. PROCESS AND INSTRUMENTATION SYSTEM - PIS

The PIS of the Dalat reactor has the following functions:

- Measuring and recording/indicating the reactor technical parameters such as: temperature at various points in the reactor tank and at the heat exchanger; lost water levels in the reactor and other tanks and in the reactor hall sump; water flow-rates of the primary and secondary loops; air flow-rates from the reactor tank space and in the reactor stack; pressure at some points on the primary and secondary loops; conductivity of reactor water before and after passing the water purification system of the primary loop.

- Creating alarm and scram signals on water level of the reactor tank and water flow-rates of the primary and secondary loops and sending of these signals to the CLS to shut down the reactor.

- Creating alarm signals on the other technical parameters of the reactor and sending of these signals to the control console for lamp indication.

The PIS has been totally renovated in the frame of the project VIE/4/010.

4.1 Temperature measurement

There are 10 temperature measuring points: 5 points in the reactor tank, 4 points at the inlet and outlet of the heat exchanger of the primary and secondary cooling loops, one outdoor temperature point.

In order to measure temperature in the reactor tank, five resistance temperature detectors (RTD) Pt-100 type are used. These RTD are connected to transmitters, TEU-7 type. The output current of the transmitters has an industrial standard value of (4-20 mA) used for driving a recorder or indicator. The temperature measuring loop at the inlet and outlet of the reactor active core with a range of 0-100 °C, consists of a 2-pen recorder type SK12 and the temperature measuring loops for the other points with a range of 0-100 °C contain the digital indicators type 3.1/2 digit LED DMP.

The temperature measuring loops for the inlet and outlet points of the primary and secondary loops at the heat exchanger consist of four RTD type Pt-100, four transmitters type TEU-7 and two 2-pen recorders with alarm value monitoring function. The temperature measuring range is 0-50 °C.

The measuring loop for outdoor temperature with a range of 0-50 °C has an RTD type Pt-100, a transmitter type TEU-7 and a digital indicator type 3.1/2 digit LED DMP.

A typical temperature measuring loop is presented in *Fig.9.*

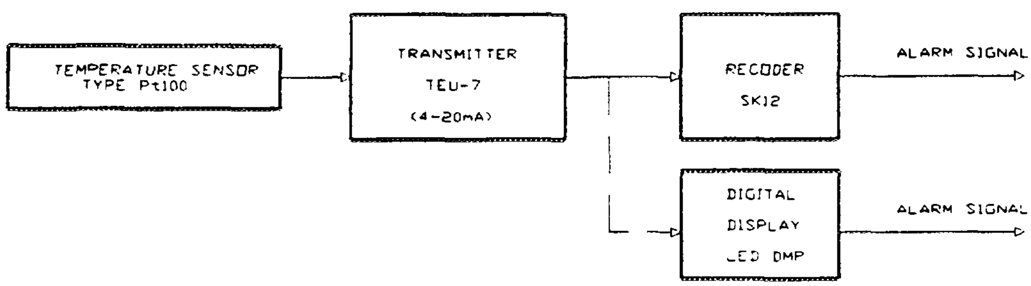

Fig.9. Diagram of a typical temperature measuring loop

4.2. Flow-rate measurement

The measuring loops for water flow-rates of the primary and secondary cooling loops consist of two diaphragms of the Soviet type, two difference pressure transmitters of the ARK-500 type and a 2-pen recorder with alarm value monitoring function. The water flow-rate measuring ranges of the primary and secondary loops are respectively 0-63 m^3/h and 0-100 m^3/h. Whenever the water flow-rate becomes lower than a setting value, the recorder emits a scram signal. This signal is transmitted to the CLS to shut down the reactor. In order to check the operation status of the measuring loop before reactor start, one electronic circuit with switches and potentiometer was designed. An operator can check alarm and scram functions of the loop by turning the potentiometer to create a simulation flow-rate. One switch will lock the 48 V DC power supply for feeding the control rod servo-drives. Then, raising the reactor power is prevented.

The measuring loop for water flow-rate in the water filtering system has a sensor/transmitter supplied by RS company with a range of 0-2.5 m^3/h, a current to voltage converter and a digit indicator type 3.1/2 digit LED DMP.

The measuring loops for air flow-rate exhausted from the reactor tank space and of the reactor stack consist of two sensors of Soviet type Venturi V-2NZ, two difference pressure transmitters of Soviet type DM-E1, amplifiers and indicators. The air flow-rate measuring ranges from the reactor tank space and the reactor stack are respectively 0-3200 m^3/h and 0-32000 m^3/h.

4.3. Water level measurement

The gross measuring loop for lost water level in the reactor tank contains a pressure sensor pipe sank in the reactor tank with a deepness of 4 m (measuring range 0-4 m), a pressure difference transmitter type ARK-200, an electronic circuit for emitting alarm (water level lower than 30 cm from full level) and scram (water level lower than 60 cm from full level) signals and a digit indicator type 3.1/2 digit LED DMP. The alarm and scram signals are transmitted to the CLS and to the indication circuit to emit flashing light and bell sound. To check scram creation of the CLS for the reactor water level, one electronic circuit with switches, potentiometer and comparators was designed. The potentiometer simulates the lost of water reactor. In order to prevent the starting of the reactor, one switch locks the 48 V DC power supply when the loop is in checking mode.

The fine measuring loop was installed to strictly and accurately control the reactor water level. The loop has a pressure sensor pipe sank in the reactor tank with a deepness of 20 cm (measuring range is 0-20 cm), pressure difference transmitter type ARK-200 and digit indicator type 3.1/2 digit LED DMP.

The measuring loop for water level in the reactor back-up tank contains a pressure sensor pipe sank in the tank with a deepness of 1 m (measuring range is 0-1 m), pressure difference transmitter type ARK-250 and digit indicator type 3.1/2 digit LED DMP.

4.4. Water level detection

Equipment for water level detection used in the PIS are of Soviet type CYC. These equipment operate on capacitance measurement between detection rod and tank shell. The PIS has the following water level detection equipment:

- Equipment for detection of low, high and overflow water levels in the reactor hall sump. The overflow signals create flashing light and bell sound. The signal on low water level in the sump is used to halt the exhausting pump.

- Equipment for detection of low and high water level in the spent fuel storage.

- Equipment for detection of high water level in the expansion tank.

- Equipment for detection of low water levels in the cooling tower of the secondary cooling loop.

4.5. Water conductivity measurement

Reactor water conductivity is measured by on-line equipment with sensors installed at the inlet and outlet of the reactor water filtering system. The equipment is JENWAY-4080L type with measuring range of 0-200 μS/cm.

4.6. Pressure measurement

In order to control the operation status of pumps, valves, mechanical and ion filters in the primary and secondary loops, pressure meters have been installed at some measuring points. The pressure meter at the outlet of the primary loop pump has an alarm contact to emit signal when the pressure at the outlet is lower than a preset value. The signal switches on a "NO COOLING WATER FOR PUMP" lamp.

CONCLUSION

The renovation of the Dalat reactor instrumentation and control system obeyed the following principles:

- Soviet Union basic design principles for the system must be kept unchanged.

- Renovated boards/blocks/sub-systems must meet the technical requirements as well as safety criteria.

- Designing electronic boards/blocks/sub-systems and using ICs and equipment should improve the system reliability.

In the renovation of the reactor ICS, the radiation protection problem was more stringently resolved. The radiation monitoring system was improved, using a PC computer which displays and records measurement results at 12 positions in the reactor hall and the control room. A new air radiation monitoring system was installed for controlling stack air before releasing it to the atmosphere. Besides above activities, the reactor instrumentation and control group has also carried out the design and construction of some computer-based systems such as a reactor physics parameters measuring system, a fuel element surface temperature measuring system and a protocol system of important parameters of the reactor.

The renovated instrumentation and control system of the Dalat reactor has been commissioned and put into operation since November 1993. The system has been working with high reliability during the two years of its operation.

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