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**ANALYSIS AND UPGRADE
OF INSTRUMENTATION AND CONTROL SYSTEMS
FOR THE MODERNIZATION
OF RESEARCH REACTORS**

PREPARED AT ADVISORY GROUP MEETINGS
ORGANIZED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
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FOREWORD

The majority of research reactors operating today were constructed more than 20 years ago. Some have undergone modifications, upgrading and refurbishing since their construction and many have been upgraded in power to meet requirements for higher neutron fluxes. A number of aging reactors, however, are still operating with their original instrumentation and control (I & C) systems or modified systems retaining some of the original equipment or components.

Worn and obsolete I & C systems cause operational problems as well as difficulties in obtaining replacement parts. In addition, there are the increasing demands of safety requirements and authorities. Rapid developments in I & C systems in the last 20 years provide motivation to the research reactor operators to replace their obsolete systems.

In order to assist research reactor operators planning I & C systems improvements or replacements, the IAEA convened an Advisory Group Meeting during 24 to 27 September 1984 in Vienna to discuss new developments, requirements, recommendations and experiences regarding I & C systems. A subsequent meeting was held during 26 to 29 August 1986 to review the draft report prepared by the participants of the first meeting. The Agency is grateful to these participants for their contributions to this publication.

EDITORIAL NOTE

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1. SCOPE

This document provides assistance in the review and planning process for the upgrade of instrumentation and control systems (I and C systems) and related safety features of the reactor protection system for research reactors. In the interest of safety a need was realized to evaluate the performance of outdated I & C systems. An advisory group was assembled to develop guidelines and to provide recommendations for the upgrade of I & C systems. The recommendations on I & C systems upgrade contained in this document were developed by the advisory group using as guidelines the established safety criteria and operating standards for research reactors.

2. INTRODUCTION

I & C systems of research reactors worldwide are in most cases twenty years or more in age. There are still in existence vacuum tube electronic I & C systems that are very difficult to maintain as many replacement parts are not available. In many cases these old systems are taken to their end of life before it is realized that they can no longer be repaired. To alleviate this condition it is apparent that considerations must be made to modify these systems and upgrade them. These research reactors were also built before there were nuclear power programs, thus they have not benefitted from the development of the presently used safety criteria. In addition, many research reactors have undergone power upgrading without any changes to their I & C systems. Technical advances in reactor I & C systems have been rapid in the past ten years and this technology, and the safety standards that have evolved, should be adapted by the research reactor community.

The process of improving or replacing I & C systems in research reactors is not a simple task and must be accomplished in such a manner as to provide safety of operation. If the upgrading of the I & C systems and related safety features is to be done in conjunction with the renewal of other systems or a power increase, it would be helpful to refer to IAEA TECDOC-214, "Research Reactor Renewal and Upgrading Programs".¹ Several sample cases are provided in the document. During

the conversion of many research reactors from HEU fuel to LEU fuel, many reactor owners will consider complete upgrade programs to include I & C systems, reactor fuel, power level, experimental facilities and support systems.

3. ASSESSMENT OF SAFETY

3.1 Safety Standards

With the increasing number of operating nuclear power reactors throughout the world, a need and strong effort to develop basic safety criteria and operating standards for power reactors has evolved. In addition to national codes and standards for nuclear power reactor I & C systems,^{2,3} the IAEA has published two safety guides for power reactors.^{4,5} The principles of safety for reactor operation contained in these documents are helpful in the assessment of safety; however, in many cases they do not directly apply to research reactors.

Since less emphasis has been placed on the development of standards for research reactors, there is a tendency for licensing authorities to impose power reactor standards upon research reactors.⁶⁻⁹ To aid member states the Agency published in 1984 Safety Series Handbook No. 35, "Safe Operation of Research Reactors and Critical Assemblies".¹⁰ This document is very general and does not contain specific standards or codes for research reactor I & C systems. The only existing set of rules, standards or guidelines for I & C systems is contained in the U.S. Standard ANSI/ANS-15.15-1978, "Criteria for the Reactor Safety Systems of Research Reactors."¹¹ This standard applies to reactors licensed by the United States Nuclear Regulatory Commission (USNRC). For other research reactors worldwide, the owner/operator and the associated regulatory agencies have to look for other applicable documents or develop new criteria.

3.2 Safety Review

A general safety review of operating research reactors and a more detailed review of I & C systems important to safety should be performed from time to time. Since most of the research reactors are relatively old and have original equipment in their safety related design features,

it is frequently realized that there is a necessity for a safety review.^{12,13} The following are taken from the IAEA Safety Series No. 35¹⁰ and are directly related to the general safety review:

"-2.2 Despite regulatory supervision, the owner/operator is directly responsible for safe operation of the reactor."

"-2.4 The major areas in which the regulatory body may be engaged in carrying out its responsibilities include the review and approval of safety documentation, including approval of limits and conditions important to reactor safety."

"-3.4 The safety report (or parts thereof) may be updated occasionally and the need for updating should be reviewed periodically. The need for such updating may be recommended by the operating organization or by the regulatory body."

"12.1 Modifications and experiments having safety significance should be submitted for review and approval by the regulatory body."

The following are basic requirements in the review of I & C systems.

- (a) Seek out those I & C systems directly related to reactor safety.
- (b) Review the performance of I & C systems against existing safety criteria.
- (c) The review and assessment of improvement programs should be performed by an independent body. If there is not a qualified review group available to conduct the review then it is recommended that the reactor owner/operator contact the Agency for assistance.

There are many reasons for conducting a safety review of I & C systems. The following are the more obvious ones:

- (a) Incidents or accidents demonstrating in a drastic manner that the present I & C system should be replaced soon.
- (b) Recommendation of independent experts evaluating the I & C system performance to replace the system or parts of the system.
- (c) Aging of the systems resulting in a loss of performance (increased needs for repair or large differences from calibration standards).

- (d) Lack of spare parts.
- (e) Large changes in safety philosophy (e.g. redundancy, diversity, self-monitoring, etc.) being introduced.
- (f) Large changes in safety standards.
- (g) Upgrading the reactor or making other changes, e.g. core conversion from HEU to LEU^{15,16}.
- (h) Expiration of the license and renewal requirements.

4. MODIFICATIONS AND MODERNIZATION

4.1 General

The spectrum of research reactors covers the complete range from zero power critical assemblies to reactors operating at power levels over 100 megawatts. Research reactors vary greatly in design and operational requirements and are not immune to accidents. Keeping this in mind it becomes difficult to develop a universal safety approach which encompasses the complete requirements for safety systems of these reactors. However, some of the general requirements of I & C systems can be clearly identified.

4.2 Available Technology

4.2.1 Neutronic Channels

In research reactors the neutron flux has to be monitored from a low level to maximum power level. The detected flux is normally sub-divided into three groups, the source range (less than 10^4 nv), the intermediate range (10^3 nv to 10^6 nv) and the power range (10^5 nv to 10^{12} nv). The following equipment is currently available for detection and measurement of neutron flux in the above three ranges.

(a) Detectors

BF_3 counters with high sensitivity are suitable for source range detection depending on reactor startup with or without a neutron source. Boron lined counters can also be used with a startup neutron source.

Fission chambers with a coating of U-235 as the active material and having a sensitivity in the region of 1 cps/nv are suitable for

intermediate range. Ionization chambers are suitable for use in the startup range, intermediate range and in the high power range. Compensated ionization chambers can be deployed whenever gamma background is very high. Other detectors which can be used in research reactors include He^3 counters and self-powered incore detectors of various types. He^3 detectors are highly sensitive and are very useful for use in research reactors. However, they are more expensive than BF_3 counters. Self-powered incore detectors employed in reactors provide a good indication of the local burnup of the fuel. Since the operating power levels in research reactors are usually low and burnup is not optimized, incore detectors are used only for research work and not for safety instrumentation. The high voltage power supplies selected for the detecting system should ensure that the detector operates in the stable region. Selection of connecting cables to the detectors should take into account the fast neutron flux in the detection region as well as the noise and immunity to microphonics.

(b) Preamplifiers

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

- (i) charge sensitive preamplifiers,
- (ii) current mode preamplifiers, and
- (iii) voltage sensitive preamplifiers.

The charge sensitive preamplifiers are normally used with BF_3 counters to integrate the charge from the detector. These work satisfactorily with input capacitance in the order of 500 to 100pF. The reliability of counting with charge sensitive preamplifiers is limited to 10^5 pulses per second.

Current mode preamplifiers can provide reliable counting up to 10 cps.⁸ These are specially suited for use with fission counters.

Voltage sensitive preamplifiers with low gain have high input impedance and low output impedance. These are useful for use with scintillation detectors, and also for matching of cable impedance.

(c) Campbell instrumentation

The use of Campbell instrumentation enables the measurement of neutron flux levels from source range to power range with a single instrument. These are hence useful for application in research reactors. However, extensive use of these channels has not become popular in safety applications.

(d) High voltage supplies

The polarizing voltage needed for the detectors depends on the specific type. For BF_3 counters, voltages of up to 3000V are needed whereas in fission counters, the voltages are 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 1mA. 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 of about 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 in the 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 a detector to the amplifier by a suitable technique such as injection and detection of a known AC signal is used.

(e) Trip units

The 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 and have a provision for locking after the levels are set. The trip circuits should employ semi-conductor electronics with adequate noise immunity and built-in hysteresis to avoid spurious trips.

Currently available semi-conductor technology, though extremely reliable, is often backed up by output relays in series to doubly ensure the safety of the reactor.

(f) 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.

4.2.2 Area Monitors

The area monitors 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 area monitors can be used as applicable.

(a) Low level gamma monitor

This monitor employs a GM counter as the detector and usually contains 3 logarithmic ranges extending from 1 mR/hr to 100 mR/hr.

(b) High level gamma monitors

The detector for this instrument is an ionization chamber. Generally 3 logarithmic ranges extending from 1 R/hr to 100 R/hr are provided.

(c) Area neutron monitors

This instrument uses a REM/n counter as a detector and normally has 3 logarithmic ranges extending from 1 mRem/hr to 100 mRem/hr. The basic detector is a BF_3 counter surrounded by a moderator/attenuator assembly to provide a weighted dose response from .025 ev to 14 Mev of neutron energy.

4.2.3 Facility Air Monitoring System

The facility air monitoring system provides for the monitoring of particulate and gaseous activity in the reactor building air and the building exhaust system. Particulate activity is monitored using a moving filter tape continuous air monitor (CAM). These monitors are equipped with alarm circuits which can be used to produce an automatic isolation and shutdown of the air handling system. Beta-Gamma scintillation detectors, proportional counters and GM detectors are commonly used in particulate monitors. The monitoring of building and exhaust air for gaseous activity normally involves the use of a gamma scintillation detector to provide for photopeak counting of Argon-41 in the exhaust stack and integral counting of gases within the reactor building. These systems normally have large gas monitoring volumes of 10 to 30 litres. Fission product monitors are often incorporated in the facility air monitoring system. CAM's having a very small gas volume are used to detect airborne fission products. The failure of fuel can also be detected by continuous water monitors for the presence of fission products in reactor coolant systems.

4.2.4 Temperature Measurements

Temperature in reactor systems are sensed by thermocouples or resistance temperature detectors. Thermistor sensors are finding favor for industrial use but have not found extensive application in reactor systems due to a lack of experience.

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 temperatures in the range of 600 to 1600°C. These are chemically inert and highly suitable in oxidizing atmospheres. Base metal thermocouples like chromel and alumel are being used for temperature measurements in the range of -130°C to 1100°C. These have higher sensitivity as compared to noble metal thermocouples, thus needing less sophisticated electronics. Resistance thermometers, which almost universally employ platinum elements, have very high sensitivities and can be safely used up to temperatures of 250°C. Resistance thermometers, however, have a large time constant, in the order of a few seconds and one should take this into account when these are employed in

reactor safety systems. Thermistors have much higher sensitivities as compared to platinum resistance elements and hence only need simple processing electronics resulting in higher system reliability. However, their susceptibility to radiation damage as well as aging has not been fully established in a reactor environment. Their resistance vs. temperature characteristic is also highly non-linear. However, this is a promising area where large-scale application is possible to effect the economy. Semi-conductor thermometers consisting of doped germanium sensors have a complex resistance-temperature relationship and are useful for very low temperature measurements. However, these do not have much relevance for use in reactor safety systems.

4.2.5 Flow Measurements

In certain types of research reactors having high power outputs, flow measurements provide one of the best safety parameters. Flow sensing gives a fast indication of impending temperature excursion of the fuel as compared to temperature measurements of the coolant at its outlet. This enables shutting down of the reactor well in advance as compared to temperature sensing.

The most common type of flow meters are differential pressure instruments operating from venturi or orifice plates. Accuracies in the order of 3% are easily achievable. The instruments are simple and extremely reliable.

Flow can also be measured by correlating thermal noise in the flow channel. The basis for this technique is the measurement of transport time of the noise pattern between two thermocouples located a few pipe diameters apart in the fluid. Accuracies in the range of 3 to 5% are possible. Correlation techniques can also be applied with ultrasonic transducers, in which case no penetration in the pipe is necessary. Accuracies are in the order of 5% over a limited range of flow.

In research reactors using liquid metal coolant, flows can be measured either by use of an electromagnetic flow meter or by an ultrasonic flow meter. The electromagnetic flow meter generates a voltage increasing with flow by the application of a magnetic field. The ultrasonic flow meters measure the flow by monitoring the transmission

time between the guide rod pairs. The accuracy of measurement in both these cases is around 10%.

4.2.6 Liquid Level Measurements

The primary means of liquid level measurement and control is the use of a float switch actuated by a mechanical arm or the magnetic reed switch actuated by a magnet imbedded in a float. In large reactors liquid level measurements in the calandria and the other important containers in a reactor system are of paramount importance for ensuring safety. The most common method of sensing liquid level is by the measurement of differential pressure at the bottom of the vessel compared to the free space above the liquid. Another important technique is the bubble tube method wherein the back pressure exerted on a bubble tube which bubbles gas through the liquid gives a measure of the liquid level. The third method employs an ultrasonic transducer located at the bottom of the tank which transmits pulses and receives them back after a delay proportional to the liquid level by reflection from the interface between the liquid level and the gas space above. The liquid levels in a metal system can also be detected by the point contact resistance tube or by inductive sensors. All the above methods have been employed in research reactors and are providing reliable performance.

4.2.7 Control Rod Drives

Most research reactors are controlled by the insertion of a neutron absorber or a neutron reflector into the reactor system. Some reactors also have fuel followed control rod mechanisms. These mostly require linear moving mechanisms which are reliable, fast-acting and radiation resistant.

A control rod drive mechanism basically consists of a drive motor coupled through gearing and an electromagnetic holding device to the neutron absorber or reflector. On the receipt of a scram signal, the electromagnet deenergizes 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 insertion time. The delay between the receipt of a scram signal and the actual operation 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 electro-magnetic delay 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 accelerating spring design should provide for the maximum electromagnetic holding force obtainable for the holding device, which depends on the maximum space available. 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. 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 of the pulsing TRIGA reactors. These rods have rapid withdrawal capability by use of a piston assembly and the application of higher pressure air to the piston space between its down position and fully up position. When used in steady state operation these rods have a continuous air supply to the piston which keeps the rod assembly in contact with an electrically driven rod drive assembly. Since the reactivity of control rods is highly non-linear, it is possible to incorporate programmed drive speed arrangements to reduce the 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 by providing a second similar system. Depending on the 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 the rotary motion of a motor to the linear drive of a shutdown mechanism are in existence. These include rope and pulley mechanisms, rack and pinion drives, and screw and nut mechanisms with split nuts for scrambling. Use of a linear induction motor or a linear vernier synchronous motor eliminates the 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.

4.2.8 Shutdown Device Position Indication

The position of a shutdown device (e.g. a control rod) should be precisely known during 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. Selsyns, potentiometers, encoders and magnetic reed switches can provide additional information on intermediate positions of the shutdown device. Direct indication of the shutdown element position has been successfully obtained in many reactors by employing push rods actuated by the element itself.

4.2.9 Emergency Cooling Systems

For larger research reactors, one has to take into account the probability of a loss of coolant accident (LOCA). Upon detection of a LOCA, the system should be able to provide identification of the location of the break in the cooling circuit. It should also actuate emergency cooling systems. The LOCA detecting system should be independent of all other systems. Prevention of spurious actuation can be ensured by providing multiplicity of safety channel signals to actuate the emergency cooling system.

4.3 Safety Logic Systems

The 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 can also fail in an unsafe manner, though very rarely. Relay logic is slow acting, requiring several milliseconds from the receipt of the scram signal to the provision of tripping action. Though this is an apparent disadvantage, this delay serves very effectively to filter out fast transients which might have caused unwarranted tripping of the reactor. Relay logics are bulky, consume large amounts of power and are susceptible to environmental conditions like dust and humidity if sealed relays are not used. 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 and reliable. Triplication of the system enhances the reliability. The power consumption for solid state logic is very low. 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. Solid state logic 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 control. The use of program 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 program controlled logic will have to be triplicated to ensure reliability. The program logic control has not been used extensively for reactor safety systems so far.

And 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 control systems. Their introduction into safety systems poses many questions regarding reliability and unsafe failures.

Considering the many problems not yet solved which have been shown to exist in relation to software testing, 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. For safety systems of nuclear power reactors, U.S. and International standards for computer controlled systems have been approved.^{23,24}

In all safety systems using semiconductor logic, or semiconductor components, unsafe failures cannot be ruled out. It is therefore necessary in such systems to provide continuous reliable impulse testing to monitor the performance of the system. No safety system should be approved for field use unless this aspect is incorporated.

4.4 Safety Criteria

The subjects to be addressed when considering the modernization of an existing research reactor are much broader than simply the selection of the design and modification itself. A basic review of the reactor safety system as a whole should be undertaken in order to ensure that not only are the operational parameters improved, but also the safety and integrity of the whole system is maintained.

Particular emphasis should be placed on currently-used safety criteria. The safety criteria for I & C systems and related safety features for research reactors are summarized as follows.

4.4.1 Administration

- (a) The reactor should be operated and maintained only by trained and authorized staff following written instructions.
- (b) Plans should be prepared for periodic testing and inspection to ensure that the safety status of the reactor is maintained.
- (c) A detailed safety report including a comprehensive fault analysis should be provided before reactor operation begins.

4.4.2 Safety Measuring Instrument Channel

- (a) All safety measuring channels must be independent of each other.
- (b) No single channel failure can produce a failure of more than one channel.
- (c) A fail-safe design which considers common failures (loss of ion chamber power supply, loss of power supply to instruments, open and short circuits in cabling, failure of active components, etc.) should be provided.

4.4.3 Protection System

- (a) The reactor protection system shall have the capability of monitoring and processing safety-relevant variables and initiate protective actions in order to prevent intolerable consequences or accidents.

- (b) When any fault develops in the system, the failure should result in the shutdown of the reactor. Testing methods which verify operability of the protection system periodically should be incorporated.
- (c) The protection channels shall be used purely for the purpose of the specific safety action intended and shall function independent of other systems.
- (d) There should be a multiplicity of protection channels preferably operating on diverse principles with at least two being capable of producing a reactor shutdown.
- (e) The protection system should be reliable and fast-acting and upon the receipt of single or simultaneous signals initiate safety action (a popular scheme for simultaneous signals being two out of three logic).
- (f) The safety system could have one or more over-power channels (commonly referred to as safety amplifier channels) which should saturate only at three to four times the permissible operating power. These channels will provide shutdown when reactor power reaches a selected level above the permissible limit.
- (g) When defined limits of the relevant process variables are reached, the reactor protection system shall initiate the fast shutdown (scram) of the reactor and when applicable provide secondary protective actions. The initiation limits shall provide a margin of safety against the upper safety limit (such as fuel safety limit).
- (h) A period scram should be considered for the startup flux measuring channel as well as the power flux measuring channel.
- (i) The reactor protection system shall automatically initiate the required protective action. It is allowed in exceptional cases to make use of "warning alarms and take manual actions instead of automatic initiation if sufficient time is available before protective action is needed. Manual interventions such as interruption or resetting of protective actions shall be possible; however, they should be allowed only if intervention does not reduce overall safety of the reactor.
- (j) The reactor protection system shall remain functional during all operational states and following a failure-inducing event.

- (k) The reactor protection system should be designed to be self-monitoring. The non-self monitoring parts shall be capable of being tested during shutdown phases and, if required, also during normal operation. It shall be insured that the monitoring system is operational during testing. Localized failures detected in the reactor protection system shall be displayed, e.g. by visual and audible alarms in the control room or visual indications on the instrument.
- (l) The components of the reactor protection system shall be reliable. Certified instruments or instruments approved by general experience shall be used. The instrument shall be capable of withstanding specified environmental and operational conditions.
- (m) Physical separation of redundant systems, such as installation in separate cabinets, is required if damage to redundant systems would prevent the initiation of protective actions.
- (n) Cables of redundant systems should be spatially separated. If this is not possible, measures shall be taken for preventive fire protection.
- (o) The components of the reactor protection system shall be clearly marked for identification.
- (p) For large research reactors emergency power should be supplied to the reactor protection system and those components that require air cooling shall be provided ventilation from the emergency power supply system.

4.4.4 Control and Protection System Interlocks

- (a) Rod withdrawal may be possible only if the startup channels detect a multiplied flux of a startup neutron source (low flux signal interlock).
- (b) Reactivity addition rate should be limited to ensure safe operation (period scram, limited rod withdrawal speed, etc.).
- (c) If controlled safety rod withdrawal is used, interlock must be provided to ensure proper order of rod withdrawal.
- (d) Manual trip should be provided.
- (e) Trips should not be self-resetting.

4.4.5 Routine Operational Testing and Checks

- (a) Rod drop times and rod withdrawal times should be tested routinely (preferably by a built-in system).
- (b) Electronic functional test of all the instrument channels must be provided before each startup.
- (c) All the important safety parameters should be monitored and displayed.

4.5 Design and Project Planning

Several requirements must be fulfilled while making any modifications or modernization of I and C systems. These are mainly safety requirements, operational requirements and budget constraints (including minimization of the reactor outage time).

Drawing up a detailed and accurate design and project planning will help greatly to fulfill these requirements.

Initial considerations for modification of the reactor I & C systems are as follows.

- (a) Specify the participation of reactor operation and maintenance personnel in the modification and the requirements of manufacturers or engineering companies.
- (b) Determine if the modifications are to be made in one step or in several phases.
- (c) Make provisions for further modifications.
- (d) Establish the level of involvement of the regulatory body.

The various phases of major modification or modernization projects must be clearly identified. These are mainly:

- (a) Planning: the plan should contain a detailed time schedule identifying all phases of work, duration of each phase, and the various time constraints.

- (b) Basic Design: description of the scope of the modifications, the concept of the new instrumentation channels, control functions, safety functions, ... the contribution of manufacturers or engineering companies should be decided at this phase.
- (c) Detailed Design: possibly developed by manufacturing or engineering companies. At this phase, all details of the new system shall be specified. The interface with the existing equipment shall be clearly defined and the new components chosen and specified. If the collaboration with manufacturers or engineering companies has been decided, it is of major importance that the reactor operation and maintenance personnel remain involved in the design work as they will have to take over the responsibility of the system at a later stage. Furthermore, it is highly recommended at this phase to inform the regulatory body or their representative of the planned modifications. Often their preliminary comments will lead to a change in design or a modification to the program.
- (d) New Core Calculation: could be needed when the modifications involve different control systems.
- (e) Amendment to the Safety Analysis Report: shall be made in conjunction with the detailed design. The regulatory body approval will be needed before the modification project can be initialized.
- (f) Detailed Purchasing Specifications: once the detailed design has been completed, purchasing specifications can be written for all components of the new system.
- (g) Call for Tender: obtain bids toward selection of a construction contractor.
- (h) Purchasing: a follow-up of the equipment to be manufactured shall be made to ensure that the requirements and the time schedule are followed.
- (i) Reactor Shutdown and Dismantling of Replaced Equipment: all parts to be dismantled shall be clearly identified. Dismantling procedures could be required (decontamination of some parts may be required).
- (j) Installation of New Equipment: it is recommended that the operation and maintenance personnel collaborate on the installation of the new system. It could be vital to proper training and knowledge of the new equipment.

- (k) Cold Tests and Calibration: the tests of the new system and its integration into the final reactor instrumentation and control system shall be performed, based on tests and calibration procedures developed during the detailed design phase.
- (l) Comissioning: the approval of the regulatory body shall be needed before a hot startup. The tests and calibration procedures for the hot startup should also be submitted for approval.
- (m) Hot Startup: final tests and calibrations shall be performed prior to startup of the reactor and a stepwise increase to full power.
- (n) Final Approval: shall be issued taking into account the test results. Approval of the regulatory body shall be obtained prior to normal operation.
- (o) Training of Operation and Maintenance Personnel: the training should start as early as possible. Involving the operation and maintenance personnel as early as the design phase is highly recommended.

4.6 Development of Specifications

The end of the design work is the writing of detailed specifications. These will apply directly to the modifications or modernization and address new philosophy, technology, components and operational techniques. The main areas and topics to be considered in development of specifications are:

- (a) description of the new system (as part of an existing facility) and design features;
- (b) regulatory requirements (applicable codes, rules and regulations, relations with the regulatory body);
- (c) quality assurance and quality control;
- (d) operational constraints;
- (e) installation and repair constraints;
- (f) local environmental conditions and service conditions;
- (g) detailed description of each subsystem or circuit, including component data sheets;
- (h) definition of the interfaces with the remaining equipment;
- (i) tests to be performed (workshop tests, cold tests on-site, calibrations, hot start-up tests);
- (j) additional design work to be performed by suppliers;

- (k) documentation requirements;
- (l) surveillance requirements;
- (m) training of personnel; and
- (n) required spare parts.

4.7 Review and Approval of Modifications, Specifications and Amendment to the Safety Analysis Report

A close collaboration between the owner/operator, the design consultant and the regulatory body during the design state is essential in order to have a reliable review of the design project documents for approval. Documents which should be prepared for the approval of the design are:

- (a) system specifications including the technical realization and design criteria for the safety system,
- (b) safety analysis of the design,
- (c) installation and commissioning program, and
- (d) project time schedule.

Any other document requested by the regulatory body should be prepared and submitted for approval. Any safety related modification or replacement of the I & C system is not to be realized without approval of the design by an independent regulatory body.

4.8 Implementation

4.8.1 Procurement

There are several ways to realize an I & C system modification or replacement project depending on the technical and economical resources of the facility. The alternatives for upgrade are a complete upgrade on a 'turn key' basis, partial upgrade (nuclear channels, control system, safety system, etc.), a 'mixed' upgrade project with several subcontractors and a coordinating body (e.g. an independent consultant, the facility itself) or a gradual upgrade. No recommendation can be given as to the selection of these alternatives; each one has to be considered separately. Independent of the alternative chosen it is of vital importance that representatives of the reactor facility participate in

the implementation. Licensed equipment or equipment manufactured to specific standards should be used in I & C systems. A thorough specification of the equipment is important in the selection of possible vendors and moreover for an accurate technical and economical evaluation of quotations. To obtain comparable quotations, close negotiations with the possible vendors is preferable. Important documents which should be submitted to the vendor when requesting quotations are:

- (a) detailed specification of the equipment and description of the safety system,
- (b) licensing and quality demands,
- (c) delivery requirements,
- (d) description of the reactor facilities and control room, and
- (e) preliminary control console lay-out.

After the vendor or vendors have been chosen, a detailed contract is drawn between the contractor and the owner/operator. In many cases the vendor and the contractor are one and the same.

4.8.2 Installation

As a part of the design an installation program is prepared and verified with the contractor(s). The aim of a detailed installation program is to minimize installation costs and reactor shutdown periods. Here good timing is an important factor. A complete replacement of the I & C system may bring about a longer shutdown period because the old instrumentation has to be taken out of operation and replaced. A partial or a gradual upgrade program gives shorter shutdown times since the reactor can be operated with the old system. One solution towards obtaining a shorter shutdown time is to operate the reactor with the old I & C system for as long as possible during the installation and partly during the commissioning tests. This is possible only if the new control console and equipment are installed independent of the old console. The installation procedure should be supervised by the owner/operator or by an independent body.

4.8.3 Testing

To minimize the testing period on site it is recommendable to test the equipment as thoroughly as possible before delivery. Functional and

environmental tests of the nuclear channels should be performed and the results approved before acceptance and installation of the equipment. The function of the safety system and individual components of the control system shall be tested before delivery by simulating the input parameters and measuring the output parameters of the system.

For the final commissioning tests an adequate test program shall be prepared for the purpose of demonstrating that the design objectives have been achieved. This shall be submitted to the regulatory body for review and approval. Accordingly, the test program and associated procedures should be prepared in advance by the owner/operator in cooperation with the designers, vendors and contractors.

Commissioning tests should be arranged in functional groups and, where applicable, in a logical sequence. No test sequence shall proceed unless the required previous steps have been completed successfully. Documentation covering the scope and sequence of these tests shall be prepared in appropriate detail.

It is recommended to use independent standardized and calibrated equipment to measure the most important reactor parameters during the initial tests. The reactor operating staff should participate in the commissioning tests.

Close liaison should be maintained between the regulatory body and the owner/operator throughout the commissioning program. In particular, the results of tests directly affecting safety shall be available to the regulatory body. After successful commissioning tests the reactor shall be operated under normal operating conditions with increased reactor operation staff involvement for a designated testing period. The behaviour of the system shall be carefully observed and documented during this period. A final performance report is normally required to be submitted to the regulatory body.

4.9 Documentation

Documentation is required for installation of equipment, facility modifications, procedural changes, and testing and should include as a minimum the following.

- (a) Verification that new equipment and its installation meets codes and standards that were specified in the design of the system. If there are differences they should be documented and changes approved.
- (b) Verification that the I & C system satisfies the following design basis of the system:
 - (i) overall functional performance requirements of the system have been met;
 - (ii) variables or combination of variables that are to be sensed to provide protective actions for each design basis event are properly measured and an instrument response is provided;
 - (iii) calculated range and rate of change of the sensed variables have been verified by testing;
 - (iv) limiting safety system settings for each sensed variable in all modes of operation have been established;
 - (v) maximum permitted response times of the system needed to accomplish all the safety tasks have been measured; and
 - (vi) reliability criterion for each safety task has been satisfied.
- (c) Design verification programs:
 - (i) quality analysis,
 - (ii) failure analysis, and
 - (iii) reliability analysis.
- (d) Electrical drawings (as built) provide circuit logic, schematics, number and location of sensors, racks, panels, set-point adjustments, operator controls, displays, manual provisions and system-test provisions.
- (e) Bench test results and system tests of the following:
 - (i) range, span and expected accuracy for each instrument,
 - (ii) expected response times of the safety system components, and
 - (iii) normal safety system settings for each sensed variable.
- (f) Final test results and verification of the qualification, functional performance and any other special requirements of the I & C system.

4.10 Final Approval

To obtain an operating license for the upgraded reactor, final approval by the regulatory body is needed. The following material should be submitted for final review and approval:

- (a) detailed system drawings including changes and corrections (as built drawings),
- (b) changes to the Safety Analysis Report,
- (c) functional and environmental test documents,
- (d) commissioning test documents (results of component testing),
- (e) test operation report (verification of system performance),
- (f) maintenance and regular test program, and
- (g) operator training program.

After receiving the final approved documents from the regulatory body, the reactor can be operated with the new I & C system.

5. CONCLUSIONS

In conclusion there are several requirements in the successful planning and implementation of an I & C system upgrade program. The periodic safety evaluation of I & C systems and related safety features using accepted safety criteria to identify needed improvements is the single most important requirement. Such a review insures that safety of operation will not be compromised. Continued reference to safety standards and technical specifications is essential. Communication and agreements between the owner/operator, the vendors and the contractor is also essential for a successful project. Involvement of the reactor operations staff in the testing and performance verification is necessary to insure proper training and confidence of performance of duties and proper maintenance and surveillance of new systems. The key to the upgrade project is a complete detailed plan.

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DEFINITIONS

The following definitions have been extracted from IAEA documents and ANSI/ANS Standards. Items marked with an asterisk were taken from IAEA documents.

*Accident Conditions - Substantial deviations from operational states which are not expected to occur during the operating life of the reactor but which could lead to the release of significant quantities of radioactive materials or significant exposure of personnel.

*Anticipated Operational Occurrences - All operational processes deviating from normal operation which are expected to occur during the operating life of the reactor and which, in view of appropriate design provisions, do not cause any significant damage to items important to safety or lead to accident conditions.

Calibration - The determination of response of an instrument over its range so that its output can be correlated, with acceptable accuracy, to true values of the measured parameter.

*Channel - An arrangement of interconnected components within a system that initiates a single output. A channel loses its identity where single output signals are combined with signals from other channels, such as from a monitoring channel or a safety actuation channel.

Channel Calibration - An adjustment of the channel such that its output responds with acceptable range and accuracy to true values of the parameter which the channel measures.

Channel Check - A qualitative verification of acceptable performance by observation of channel behaviour. This verification may include comparison of the channel with expected values, or other independent channels or methods of measuring the same variable.

Channel Test - The introduction of an input signal into the channel to verify that it is operable.

*Commissioning - The process design which reactor components and systems, having been constructed, are made operational and verified to be in accordance with design assumptions and to have met the performance criteria; it includes both non-nuclear and nuclear tests.

*Competent Authority - A national authority designated or otherwise recognized as such by the Government for a specific purpose.

*Common-Cause Failure - The failure of a number of devices or components to perform their functions, as a result of a single specific event or cause.

Design Basis Event - Anticipated operational occurrences (such as the loss of coolant flow or a reactivity excursion) which are used to determine the specific design requirements for the reactor safety system.

Engineered Safety Features - Features of a unit, other than reactor trip or those used only for normal operation, that are provided to prevent, limit, or mitigate the release of radioactive material.

Function Test - The introduction of an input signal into an instrument to verify that it is operable.

*Items Important to Safety - The items which comprise: (a) those structures, systems and components whose malfunction or failure could lead to undue radiation exposure of the site personnel or members of the public; (b) those structures, systems and components which prevent anticipated operational occurrences from leading to accident conditions; (c) those features which are provided to mitigate the consequences of malfunction or failure of structures, systems or components.

*Logic - The generation of a required binary output signal from a number of binary input signals according to predetermined rules or the equipment used for generating this signal.

*Maintenance - The activity of keeping all equipment in good operating condition, including both preventive and corrective (or repair) aspects.

Measuring Channel - The combination of sensor, lines, amplifiers and output devices which are connected for the purpose of measuring the value of a process variable.

*Normal Operation - Operation of a reactor within specified limits and conditions including startup, power operation, shutting down, shutdown, maintenance, testing and refueling.

Operable - A system or component is operable when it is capable of performing its intended function in a normal manner.

Operating - A system or component is operating when it is performing its intended function in a normal manner.

*Protective Action - Protection system action calling for the actuation of a particular safety actuation device.

*Protective Device - A safety device designed and installed to act to ensure that one or more safety limits are not violated.

*Protection System - A system which encompasses all electrical and mechanical devices and circuitry, from sensors to actuation device input terminals, involved in generating those signals associated with the protective function.

Protective Instrument System - That part of the Reactor Safety System which is the total of all protective instrument subsystems necessary to sense the occurrence of all Design Basis Events and to initiate the operation of the safety shutdown equipment. Provisions for manual initiation based upon the decision of the reactor operator are included.

Protective System - A safety system designed and installed to act automatically to ensure that one or more safety limits are not violated.

*Quality Control - Quality Assurance actions which provide a means to control and measure the characteristics of an item, process or facility in accordance with established requirements.

Reactor Owner - The individual or organization designated to have overall responsibility for the reactor facility. In the case of licensed reactors, the licensee or his designee.

Reactor Safety System - The reactor safety system is the composite of the safety interlocks, as defined, the protective instrument system, as defined, and the safety shutdown equipment, as defined.

*Redundancy - Provision of alternative (identical or diverse) elements or systems, so that any one can perform the required function regardless of the state of operation or failure of any other.

*Regulatory Body - Either a body authorized by the Government independent of the operating organization or a body or committee of the operating organization that is independent of the reactor management.

Research Reactor - A device designated to support a self-sustaining neutron chain reaction for research, developmental, educational, training or experimental purposes, and which may have provision for the production of nonfissile radioisotopes.

*Safety Action - A single action taken by a Safety Actuation System.

Safety Actuation System - The collection of equipment required to accomplish the required Safety Action when initiated by the Protection System.

Safety Function - A specific purpose that must be accomplished for safety.

Safety Interlock - An interlock: (a) which functions to limit the magnitude of a Design Basis Event and is assumed to remain operable in the accident analysis for the facility; (b) which, if subjected to unsafe failure, would allow the occurrence of a potentially serious event that has not been explicitly encompassed by the design basis for the reactor safety system.

*Safety Limits - Ultimate limits assigned to important process variables or parameters which, if exceeded, could result in undue personnel exposure or release of undue amounts of radioactivity.

Safety Measuring Channel - A measuring channel in the reactor safety system.

*Safety-Related Instrumentation and Control System - Those I & C systems important to safety which are not included in Safety Systems.

Safety Shutdown Equipment - That equipment necessary to execute, upon receiving a signal from the protective instrument system, the shutdown of the reactor by rapid reactivity reduction, for safety (non-routine) reasons.

Safety System Support Features - The collection of equipment that provides services such as cooling, lubrication, and energy supply required by the Protection System and the Safety Actuation Systems.

Shall, Should and May - The word "shall" is used to denote a requirement; the word "should" to denote a recommendation; and the word "may" to denote permission, neither a requirement nor a recommendation.

Unsafe Failure - Any malfunction such that the unit (i.e. module, channel, subsystem, system, or piece of equipment) is no longer operable. A malfunction which results in the immediate execution of the protective action of the unit is not an unsafe failure.

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