SRI

Highlights


SRI Guest House at Anupuram

Atomic Energy Regulatory Board

सत्यमेव जयते
GOVERNMENT OF INDIA

Safety Research Institute

Kalpakkam

Atomic Energy
Regulatory Board

SRI Highlights

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Editorial Board

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FOREWORD

Every nuclear facility in the country is required to obtain an authorization from the Atomic Energy Regulatory Board at various stages like site selection, construction, operation and decommissioning. A committee of experts recommends the issue of such authorizations only after a detailed multi tier safety review. Over the years, such reviews are becoming more complex and challenging for the regulators. In the coming decades, DAE is poised for a large growth in various sectors. New types of reactors like VVER, FBR and AHWR will come into operation. More fuel cycle facilities will also be constructed to meet the additional demands for the fuel of these reactors. It is essential that in keeping with the growth of the Department, the Regulatory Board also should augment its expertise. To achieve this objective, AERB started the Safety Research Institute (SRI) at Kalpakkam in February 1999 to carry out and promote safety related research.

The main objective of SRI will be to develop models, methodologies and the knowledge base required for quantitative assessment of the risks associated with the operation of nuclear fuel cycle facilities. The areas of research have been chosen keeping in view their importance to safety assessment of AERB and also after taking into account the work being carried out in BARC and IGCAR. The Institute started with small staff strength and therefore work was carried out only on certain selected topics like probabilistic safety assessment for nuclear power plants and radiation shielding and dosimetry. SRI also promoted an inter-institutional coordinated research programme in the area of atmospheric dispersion studies as part of environmental safety studies. In collaboration with ISRO, a Remote Sensing and Geographic Information System (RS-GIS) data processing facility has been installed at SRI for Environmental Impact Assessment studies for nuclear facilities. Yet another important initiative was to start a depository of safety related computer codes. A large number of well validated codes have been installed and user friendly interfaces have been developed. Since its inception, SRI has also been providing through regular workshops and discussion meetings, an excellent forum where designers, research groups and regulators can come together for formulation and implementation of action programmes aimed at resolving safety related issues. The building along with the infrastructure required for carrying out the research activities of SRI at Kalpakkam and the SRI Guest House at Anupuram have been recently commissioned. Within the short span of four years, SRI has made very good progress and this booklet **SRI Highlights** gives a brief summary of the progress made so far. I am sure that in the years to come the Institute will grow in strength and stature and AERB will find a significant part of the expertise it needs for its regulatory functions within its own organization.

Suhas P. Sukhatme

S.P. Sukhatme

Chairman, AERB

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1. Mission

INTRODUCTION

Atomic Energy Regulatory Board (AERB) has been mandated to review, enforce standards and authorize from safety angle siting, design, construction, operation and decommissioning of nuclear facilities. To carry out this function effectively, AERB is required to equip itself with sound technical infrastructure and to build up a wide knowledge base. AERB's technical independence from the operating units would be best achieved by the regulatory body's own strong research capability in safety related fields. In order to achieve this goal, AERB commissioned the Safety Research Institute (SRI) at Kalpakkam in February 1999. The main objective of SRI will be to develop models, methodologies and knowledge base required for quantitative assessment of risks associated with the operation of nuclear fuel cycle facilities. The institute will also provide an excellent forum where designers, research groups and regulators can come together for formulation and implementation of research programmes aimed at resolving safety related issues.

OBJECTIVES OF SRI

The primary objective of SRI is to build a unique research base in areas which are pertinent to safety functions of AERB in a way complementary to the ongoing R&D activities of the DAE units. These areas include nuclear power plant safety, Fire safety, Industrial and Environmental safety. The main thrust of research and development activities in SRI are:

- to develop models, methodologies and knowledge base required for quantitative assessment of risks associated with the operation of nuclear fuel cycle facilities
- to generate/collect data needed for safety assessment.
- to provide a technical forum for joint research among power plant personnel, research groups and regulatory functionaries in safety related fields.
- to organize regular programmes of technical meetings and training courses for different target groups on a variety of topics for enhancement of safety performance.

In the initial phase, SRI will carry out R&D activities in the following areas.

Nuclear Plant Safety

- Design verification and safety analysis: To develop expertise for safety analysis of the plants under various normal and off-normal situations. To promote validation and up gradation of tools for safety assessment.
- Ageing assessment and life extension of nuclear facilities: To provide methodologies and tools for qualification of plant equipment /component and safety assessment of nuclear facilities.
- Probabilistic safety analysis and reliability engineering: There is a vital need for development of PSA methodologies and their application in regulatory function, generation of database from operating facilities of DAE.
- Safety Code Depository: SRI will organize a depository of all safety related codes often used by the facility designer and safety analysts.
- Studies in radiation shielding and dosimetry for application in both nuclear and medical applications of radiation.

Environmental Safety Studies

- Atmospheric dispersion studies, theoretical and experimental, to develop better models capable of giving realistic estimates of public exposures from operation of nuclear facilities.
- Marine dispersion studies for development of appropriate models for assessment of public exposure from liquid effluents.
- Studies on migration of radioactivity in soil for safety assessment of long-term storage of radioactive waste.
- Formulation of methodologies for Environmental Impact Assessment of nuclear facilities for quantification of risk.
- Application of the state of art techniques like Remote Sensing and GIS for EIA studies.

2. Current Research Activities

2.a PROBABILISTIC SAFETY ASSESSMENT

INTRODUCTION

Nuclear Plant Safety Study is identified as one of the main areas of research and development activity in SRI. Falling within its purview, among other things, is Probabilistic Safety Assessment (PSA) of Nuclear Power Plants (NPP). Research activities performed in PSA methodologies, Reliability Analysis of safety systems and safety support systems and their applications from the reactor safety and regulatory viewpoint are described below. Four specific problems have been addressed so far, Viz.

1. Statistics of Loss of Offsite Power (LOSP) at Kalpakkam
2. Estimation of Station Blackout Frequency
3. Reliability Analysis of Safety Systems of PFBR
4. Reliability Demonstration Testing of Components of Nuclear Power Plants

STATISTICS OF LOSS OF OFF-SITE POWER AT KALPAKKAM

Loss of off-site power (LOSP) statistics specific to a given site is an important input to the reliability analysis of safety systems of Nuclear Reactors situated at the site. The data reported by Madras Atomic Power Station (MAPS) over a period of fifteen years from 1984 to 1998 (Fig.1), have been compiled and analyzed for statistical parameters like, frequency of power failure, longest duration of power failure, maximum number of failures in a year, power unavailability, power failure frequency-duration-correlation, etc. and are given below.

(i) Average frequency of LOSP i.e., failure rate (λ) is 3.8 / year.
(ii) Mean Time To Failure (MTTF) is 96 d.
(iii) The frequency of time taken to restore power is depicted as a histogram in Fig.1.
(iv) Mean Down Time (MDT) for offsite power is 39 min., which has not decreased over the years.
(v) The repair rate (γ) is calculated as 1.6 h^{-1}
(vi) Availability of off-site power, A = 0.99974
(vii) Based on the analysis, the steady state unavailability is found to be $U(\propto)=2.6 \times 10^{-4}$, which implies an availability of 99.97% as expected.

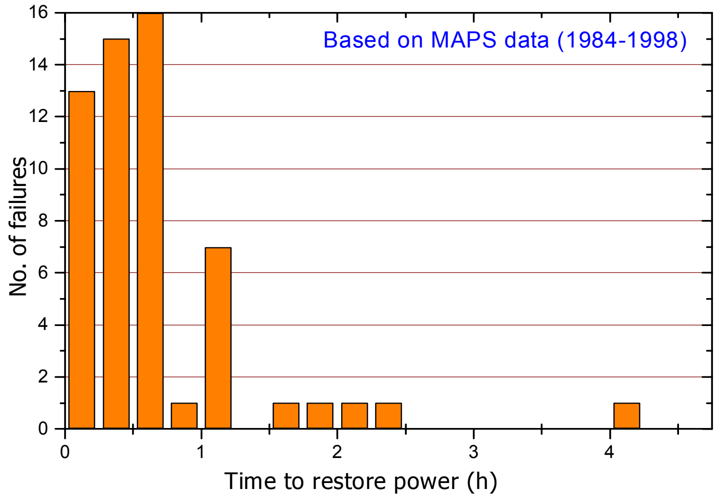


Fig. 1: Frequency Distribution of Failure Duration

LOSP Frequency and Duration Relation

The frequency of LOSP is given in Fig. 2. It depicts the frequency that LOSP recovers in time t or more. An exponential fit to the observed data for time < 2 h and a Weibull fit for the remaining duration have been made as given below:

$$Y = a.e^{-bx}\text{, where a = 3.8 and b = 1.57}$$

$$Y = a.e^{-bx^{c}}\text{, where a = 3.8, b = 2.146 and c = 0.475}$$

Although the fits are good, it would be good to have a physical basis for the fitted function. Further, considering the scarcity of data, instead of a series of stretched exponentials, a single function would be preferable. A study of the nature of the relationship of off-site power failure duration and its frequency leads to a power law dependence, which has been confirmed by performing an extreme value analysis. The extreme value distribution is also used to extrapolate beyond observed LOSP durations.

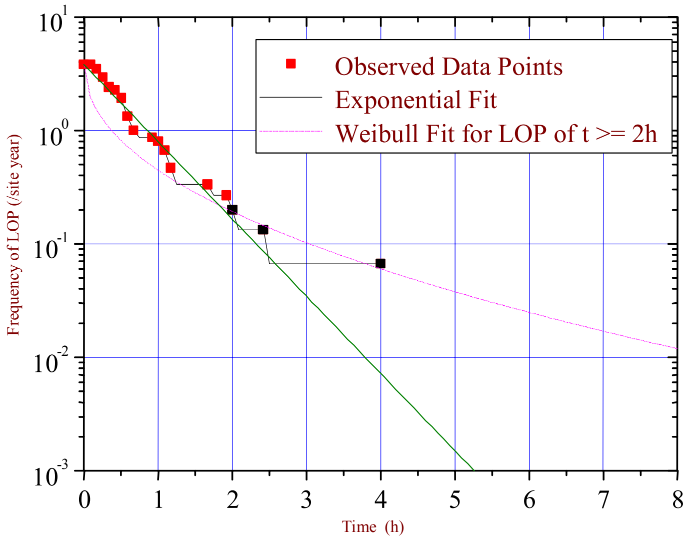


Fig. 2: Frequency of Loss of Offsite Power Exceeding Specified Duration

ESTIMATION OF STATION BLACKOUT FREQUENCY

The Station Black Out (SBO) is defined as the failure of Class IV and Class III power systems of a plant (AC power). Its frequency and duration determine the design of alternate sources of power for a reactor so that the probability of core damage is minimized.

The frequency of SBO is given by the product of frequency of class IV power failure (of duration ≥ t), Unavailability of class III power and the probability of non-recovery of class III power in time t.

Frequency of SBO for Prototype Fast Breeder Reactor (PFBR) and Fast Breeder Test Reactor (FBTR) at Kalpakkam has been estimated as follows.

SBO Calculation For PFBR

Class IV System

The source of Class IV system is either from the offsite grid or from the turbo-generator of MAPS. Failure of this system can be due to plant-centered faults, utility grid failures or from the failure of offsite power sources induced by external events (severe weather conditions). The frequency of LOSP due to the configuration of power distribution system and the reliability of the grid is 3.8 /y. It is found that the short duration failures are well represented by exponential function and the low frequency events of long down times are represented by a two-parameter Weibull distribution function. The LOSP frequency due to severe weather (SW) events derived from the 93 years data is 4.1 x 10^{-3}/y and the frequency due to extremely severe weather (ESW) conditions for Kalpakkam site is 6 x 10^{-4}/y. Therefore, the total expected frequency of LOSP exceeding time t is expressed as:

$$\lambda_{LOSP}(t) = Max(\lambda_S e^{-\alpha_1 t}, \lambda_L e^{-\alpha_2 t^{\beta}}\ ) + \lambda_{SW}(t) + \lambda_{ESW}(t)$$

Class III System

For PFBR, there are four Class III bus sections and each fed from a Class IV supply bus and also from standby diesel generator (DG). The four DGs located in two independent DG stations and a fire barrier wall segregates the two DGs in each station. Each DG is rated to 50% of the total emergency supply and is functionally independent of the other one. The unavailability of Class III power was evaluated using the Fault Tree method and found to be 2.4 x 10^{-3} when DGs are used in 2/4:S mode (successful operation of two out of 4 DGs is sufficient to meet the station loads) and 6.8 x 10^{-4} when DGs are used in 1/4 mode. The major contributions to this arise due to common cause failure of DG to start and run. A more detailed model of the failure combinations of DGs using Markov analysis (Fig. 3) yields approximately the same results.

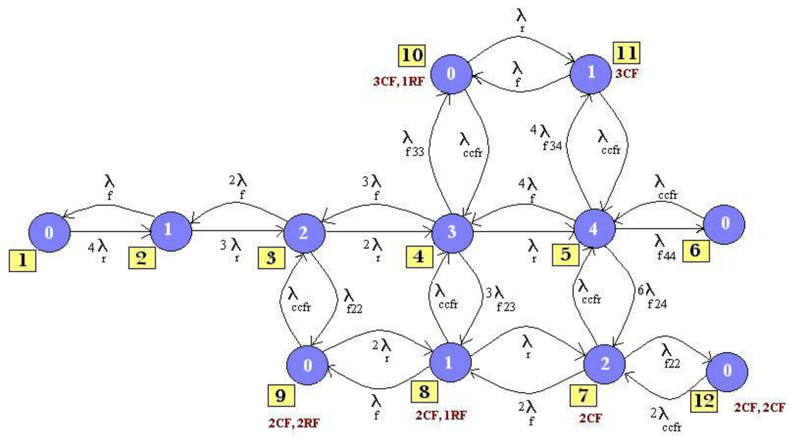


Fig. 3: Markov State Transition Diagram of 2/4 DG Configuration

The non-recovery probability of Class III power as a function of time t is given as, $e^{-t/Tr}$, where T_r is the mean time taken to repair a DG. The mean repair time is about 8 h.

Station Blackout Frequency

The SBO frequencies for various durations are evaluated (Fig. 4). It is found that when DGs are used in 2/4 mode, the frequency of SBO is ≤ 1 x 10^{-4}/r.y. for a blackout duration of 4 h or more and it is ≤ 1 x 10^{-6} /reactor year(ry) for a blackout duration of 14 h or more. The corresponding values for 1/4 DG configuration are 2 h and 11 h respectively.

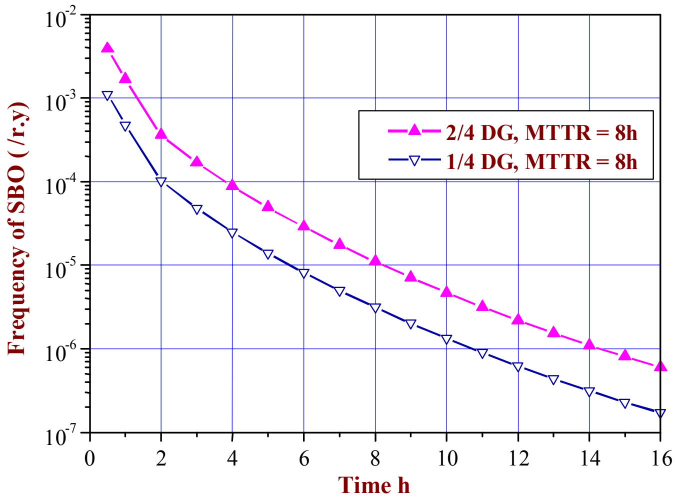


Fig. 4: Frequency of Station Blackout Events Exceeding Specified Duration

SBO Calculation For FBTR

A similar analysis of SBO frequency as a function of blackout duration was carried out for FBTR with a view to estimate the period for which one feeder out of the two linking MAPS to IGCAR can be taken out for maintenance without loss in reliability.

The frequency of LOSP at FBTR is found to be 5.2 / year. The transformer-feeder unavailability data is also included in the calculation of Class IV failure. The net frequency of LOSP as a function of down time for single feeder and double feeder cases is arrived at by taking these into account.

The unavailability of onsite emergency power supply evaluated using Markov method and Fault Tree method is 1.0 x 10^{-2} and 3.34 x 10^{-3} respectively. The frequency of SBO at FBTR with single feeder and double feeder cases, for different time durations is evaluated and given in Fig. 5. The frequency of SBO with double feeder computed using Markov method is ~ 10^{-4} /ry for 11 h and 10^{-5} /ry for 19 h duration. With one feeder out of service, it is five times higher.

Sensitivity study done with respect to DG repair time and Common Cause Failure indicates that the magnitude of SBO could be uncertain by a factor of 10, the uncertainty generally increasing with SBO duration.

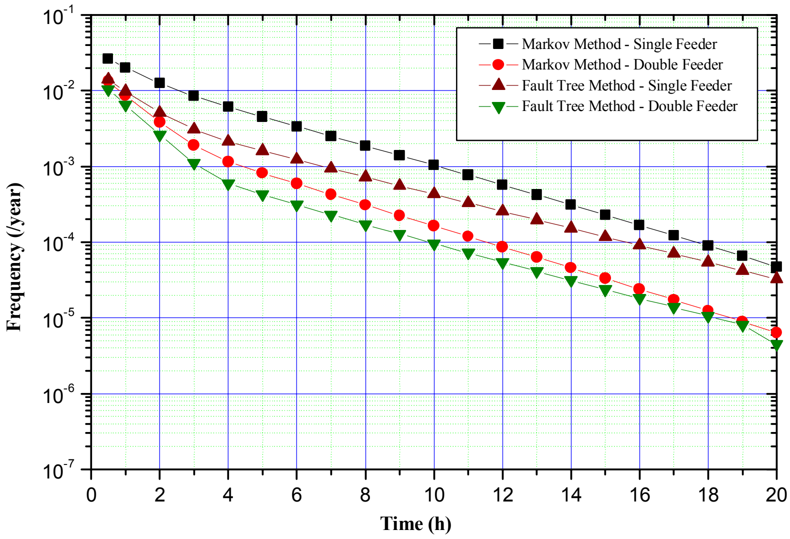


Fig. 5: Frequency of SBO for Time Exceeding Time t

RELIABILITY ANALYSIS OF SAFETY SYSTEMS OF PFBR

Safety Grade Decay Heat Removal System

There are two Decay Heat Removal System in PFBR. The first system, known as Operation Grade Decay Heat Removal System (OGDHRS) consists of primary sodium circuit, secondary sodium circuit and steam water system, which is the normal heat transport path and the same is also used for Decay Heat Removal (DHR). Whenever there is failure of OGDHR due to failure either in both the secondary sodium circuits or steam water system or loss of off site power, the second system known as Safety Grade Decay Heat Removal System (SGDHRS) is called into operation. It consists of 4 independent circuits with a heat removal capability of 8 MWt per circuit. Continuous availability of one circuit is sufficient to meet all category 4 Design Safety Limits (DSL) on fuel, clad, coolant and structural temperatures and forms the criterion for successful DHR through SGDHRS.

Decay heat is transferred to intermediate sodium by natural convection of primary sodium through the sodium-to-sodium heat exchanger (DHX). This then is passed to air through sodium-to-air heat exchanger (AHX). Hot air is released to the atmosphere through a tall stack. Appropriate elevation and design of DHX, AHX and stack ensures natural convection in the intermediate sodium and air circuit. AHX casing is provided with two dampers in the inlet and two in the outlet to enhance the deployment of the SGDHR on demand. The dampers on the same side are activated by diverse means like air pneumatic for one and class 2 electrical power for the other. Provisions are made to open them manually, if auto and remote manual means fail.

Integrity of a host of components like vessels, heat exchangers, pumps, valves, pipes, expansion tank and air stack forming the heat transport system is necessary for successful decay heat removal. As per AERB criteria, the failure frequency of DHR should be $< 1 \times 10^{-7}$ /ry. The total number of design basis event being 861 for plant lifetime of 40 year (207 demands being directly on SGDHRS and remaining ones on OGDHRS). With OGDHRS probability of 0.1/de, the number of demands on SGDHR is 7 per year.

Reliability analysis was carried out by employing Fault Tree method for the basic scheme as well as for several other options. The results are presented in Table 1. The probability of failure of SGDHRS consisting of 4 identical loops is 5.2×10^{-6}/de for a mission time of 30 d, which is much more than the requirement. The unreliability is dominated by leak rates of components like AHX, DHX, sodium dump and isolation valves.

Table 1: Results of SGDHRS Reliability Analysis for Various Options

Design Options	Failure Probability P_{DHR} (/de)	Failure Frequency λ_{DHR} (/ry) [P_{DHR} x de = 7]
Reference Case 8 MWt /loop	5.2×10^{-6}	3.6×10^{-5}
Diverse AHX	2.0×10^{-6}	1.4×10^{-5}
Diverse DHX	4.8×10^{-6}	3.4×10^{-5}
Diverse Valves	3.6×10^{-6}	2.5×10^{-5}
Diverse AHX, DHX & Valves	2.1×10^{-8}	1.5×10^{-7}
NO CCF	$< 1 \times 10^{-8}$	$< 1 \times 10^{-7}$

Note: Two Groups of Diverse Systems Assumed with No CCF Between Diverse Groups.

Diversifying DHX or dump valves alone does not give appreciable improvement, while diversifying AHX alone gives a moderate improvement. If all the three components, AHX, DHX and dump valves are diversified, the failure probability of SGDHRS is 2.1×10^{-8} /de. With 7 demands per year, the failure frequency for the diverse design is 1.5×10^{-7}/ry. This effectively meets the AERB criteria on DHR unreliability.

Shut Down System

The Shut Down System (SDS) of PFBR, shown in Fig. 6, consists of a Reactor Protection System (RPS) with functional diversity of sensors, analogue signal processing circuits, safety logics and an Actuation System (AS) consisting of absorber rods viz., Control and Safety Rod (CSR) and Diverse Safety Rod (DSR), electromagnets and drive mechanisms to drop or drive the absorber rods into the core. For achieving high reliability, two independent optically linked shutdown systems are provided. Optical inter-link enables SCRAM parameters from both RPS to trigger both the AS while maintaining electrical isolation.

The demand on the shutdown system is said to be successful if at least 8 out of the 9 CSR or 2 out of the three DSR are inserted into the core when any SCRAM parameter crosses its threshold or upon a manual signal. The SCRAM parameters have been identified based on the analysis of Design Basis Events (DBE) having potential to increase fuel, clad and coolant temperatures beyond their DSL. This may occur due to Transient Over Power (TOP) or due to Transient Under Cooling (TUC).

Reliability target: The overall non-availability of shutdown systems as stipulated by AERB shall be less than 1×10^{-6} /ry. The reliability of each shutdown system shall be such that its non-availability is less than 1×10^{-3} / ry.

The reliability evaluation of SDS was carried out using fault tree method employing the “immediate cause” approach as recommended by AERB. The top event of the fault tree is the failure of SDS to shutdown the reactor on demand. The fault tree for each SDS has to include the optical signal from the other SDS. The Fine Impulse Testing (FIT) feature used in the SDS is also included in the fault tree analysis. The Fault Trees for Global Fault like loss of flow TOP events, and Local Fault events like sub-assembly faults were developed.

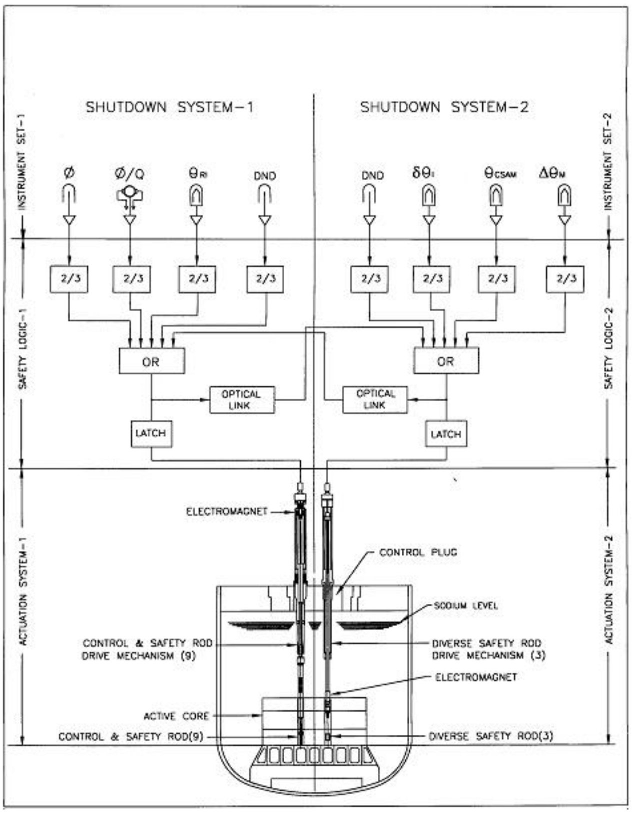


Fig. 6: Shut Down System of PFBR

Failure data: Failure rate data for SDS components have been obtained from FBTR experience and international thermal and fast reactor experience. Common Cause Failure (CCF) evaluation is done using β factor model. Appropriate \`β' is chosen for the components after taking into account the independence and diverse measures provided. Checklist method is also used to arrive at the β value for some of the components.

Reliability of SDS was computed by evaluating the minimal cut sets of the fault tree for each of the DBE using IAEA software PSAPACK 4.2.

Results: The results are shown in Table 2. The failure probability of actuation system is 3.0×10^{-8} / demand (de) and the failure probability of RPS is 1.6×10^{-8} / de for global faults and 3.5×10^{-7} /de for Local fault. The failure frequency of SDS is 9.2×10^{-7} /ry with 20 demands per year for all events except local fault, for which a demand frequency of 10^{-2}/ry is used.

Table 2: Results of SDS Reliability Analysis for Global and Local Faults

DBE	Event Frequency λ_F (/y)	Probability of Failure on Demand			Unprotected Event Frequency (/y) $P_S \lambda_F$
		Actuation System (P_A)	RPS (P_R)	Total P_S=(P_A+P_R)	
Global Fault	20	3.0×10^{-8}	1.6×10^{-8}	4.6×10^{-8}	9.2×10^{-7}
Local Fault*	1×10^{-2}	3.0×10^{-8}	3.5×10^{-7}	3.8×10^{-7}	3.8×10^{-9}
Total					**9.2×10^{-7}**

*Local Fault – Design basis blockage in a fuel SA, which is a Cat. III event [ref. PSAR]

Uncertainty and Sensitivity Analysis: Uncertainty analysis was performed to obtain the bounds for the unavailability of the top event, employing Moments Matching method and/or Monte Carlo method. The necessary computer codes have been developed and validated with the WASH-1400 results for a sample problem. The analysis indicates an error factor of 4 and 6 for the unavailability of local fault and global fault events respectively.

A sensitivity analysis of each component using measures of importance like Risk Achievement Worth and Risk Reduction Worth shows that the importance of some components is likely to be reduced when the common cause failures are reduced.

Core Temperature Monitoring System (CTMS)

Monitoring outlet temperature of individual fuel subassembly is an essential requirement for early detection of potential clad rupture / fuel melting due to overheating. CTMS caters to this need and consists of temperature sensors and Real Time Computer System (RTCS). RTCS consists of signal conditioners, real time computers in suitable configuration and switch over logic circuit. The output from CTMS initiate reactor SCRAM through pulse coded logic, power gates and reactor actuation system. CTMS is classified as safety critical system and must be highly reliable.

Reliability analysis was performed to evaluate following configurations of RTCs.

- Hot-standby system where all signals are duplicated and processed by two RTC's each generating outputs. But at any point of time output from only one system will be connected to the plant through switch over logic circuit (SOLC).

- Dual System Architecture where isolated output of each sensor is connected to 2 RTCs for signal processing. The output of RTC, subjected to 1/2 or 2/2 or 2/3 voting logic system, initiates safety action to Control Rod Drive Mechanism (CRDM).

The results for the different configuration studied along with target failure probability are given in Table 3. The hot-standby configuration meets the target for safe failures but falls short with respect to unsafe failures. It reduces the spurious trips arising from RTC, compared to 1/2 configuration, by switching over to the second system. In case of unsafe failure probability, it is comparable to the 2/2 system, as unsafe failures may not lead to a switch over to RTC-2, even if it were to work successfully. The 2/3 voting model satisfies both safe and unsafe reliability targets.

Table 3: Comparison of System Reliability for Different Options

Configuration	With CCF		Without CCF	
	Safe Failure Frequency (/y) **Target $1x10^{-1}$ /y**	Unsafe Failure Probability (/de) **Target 1×10^{-4} /de**	Safe Failure Frequency (/y) **Target $1x10^{-1}$/y**	Unsafe Failure Probability (/de) **Target 1×10^{-3} /de**
Hot-Standby	$9.1x 10^{-3}$	1.3×10^{-3}	5.6×10^{-3}	1.3×10^{-3}
1/2	1.5×10^{-1}	5.6×10^{-5}	1.5×10^{-1}	1.2×10^{-6}
2/2	3.9×10^{-3}	2.2×10^{-3}	1.5×10^{-3}	2.2×10^{-3}
2/3	**4.2×10^{-3}**	**5.9×10^{-5}**	**4.5×10^{-4}**	**3.6×10^{-6}**

RELIABILITY DEMONSTRATION TESTING OF THE COMPONENTS OF NUCLEAR POWER PLANTS

A study using Bayes Technique is carried out to demonstrate the reliability of vital components of nuclear reactor systems to validate the claim of high reliability and assure that they perform their function adequately over a specified mission time without any failure. Since test duration is inversely related to the failure probability, the test interval or number of trials needed for high reliability, long life components can be prohibitive in terms of cost and time.

Bayes Technique

Bayesian technique is applied with a view to develop a testing procedure, which can be used to demonstrate component reliability conformance to stated requirements. The procedure will help in determining the number of tests to be performed and the number of components required to be tested to conclude with a given degree of confidence that the component meets the reliability requirement based on accept and reject criteria.

It is assumed that some prior information is available about the component failure rate from the designers and reliability engineers in the form of expected failure rate and an uncertainty interval with an Upper Limit (UL) and a lower Limit (LL). Based on this, a prior distribution (gamma distribution) is assumed and is subsequently modified based on test results to form the posterior distribution. Testing continues until the posterior distribution is decisive, according to appropriate criteria.

The analysis was applied to a component having failure rate $\lambda \leq 1 \times 10^{-4}$ with UL = 10^{-3}; LL = 10^{-5} and prior confidence, $P_0 = 0.70$. The required hours of testing the component was seen to be 7728 h (equivalently 10 components can be tested for 772.8 h or 5 components for 1545.6 h); and if no failures occur, one can claim with 85 % confidence that the failure rate of the component does not exceed 1×10^{-4}. The analysis was repeated to obtain the required testing hours for various levels of confidence.

2.b RADIOLOGICAL SAFETY STUDIES

INTRODUCTION

One of the important tasks in regulatory review of nuclear and radiological facilities is the assessment of adequacy of shielding provided to protect radiation workers. The radiological safety studies group at SRI is carrying out the following tasks.

- Radiation shielding design, research, review and evaluation of various nuclear facilities, particle accelerators and LINACs

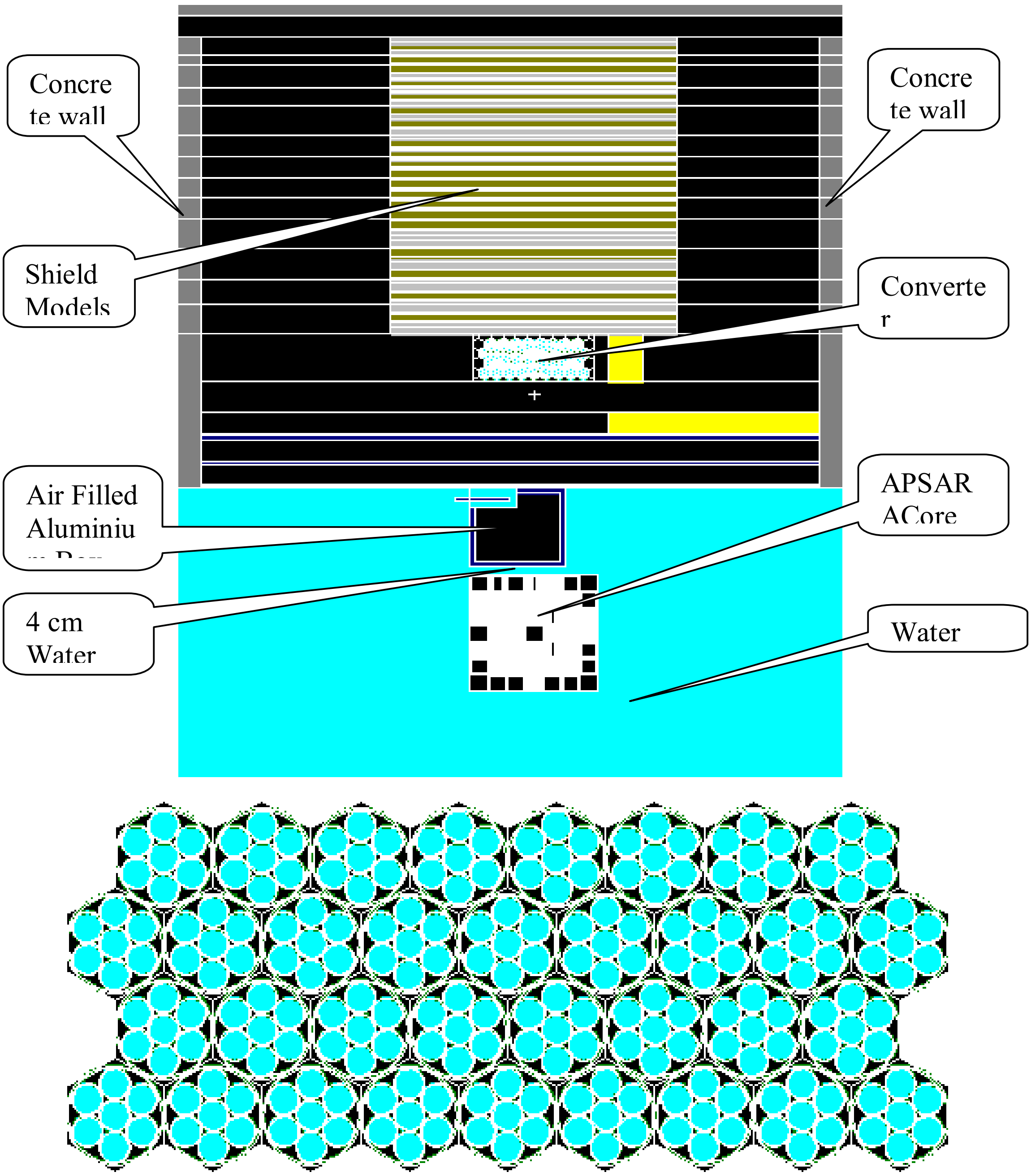
- Development of radiation transport codes to handle special applications.

- Development user friendly graphical interfaces for frequently employed radiation transport codes

- Arrangement of periodic workshops on radiation transport codes to promote young professionals to take up research work and experts to keep abreast of latest information in the concerned areas.

STUDIES ON FAST REACTOR BULK SHIELDING EXPERIMENTS AT APSARA

A series of experiments were carried out in the shielding corner facility of APSARA reactor at BARC as joint effort of several groups. The experiments were aimed at obtaining bias factors to be used with calculated values of shield design parameters of PFBR. Two important changes were made in the shielding corner facility to make the conditions representative of fast reactor environment. A depleted uranium converter assembly was introduced at the start of the beam port to convert the thermal neutron spectrum to match the anticipated leakage neutron spectrum from PFBR blanket. To make up for the loss in flux, the APSARA pool water towards shielding side was replaced by air filled hollow aluminium box. Neutron attenuation through mockups of PFBR radial and axial shields was measured using appropriate set of activation foils and SSNTDs.

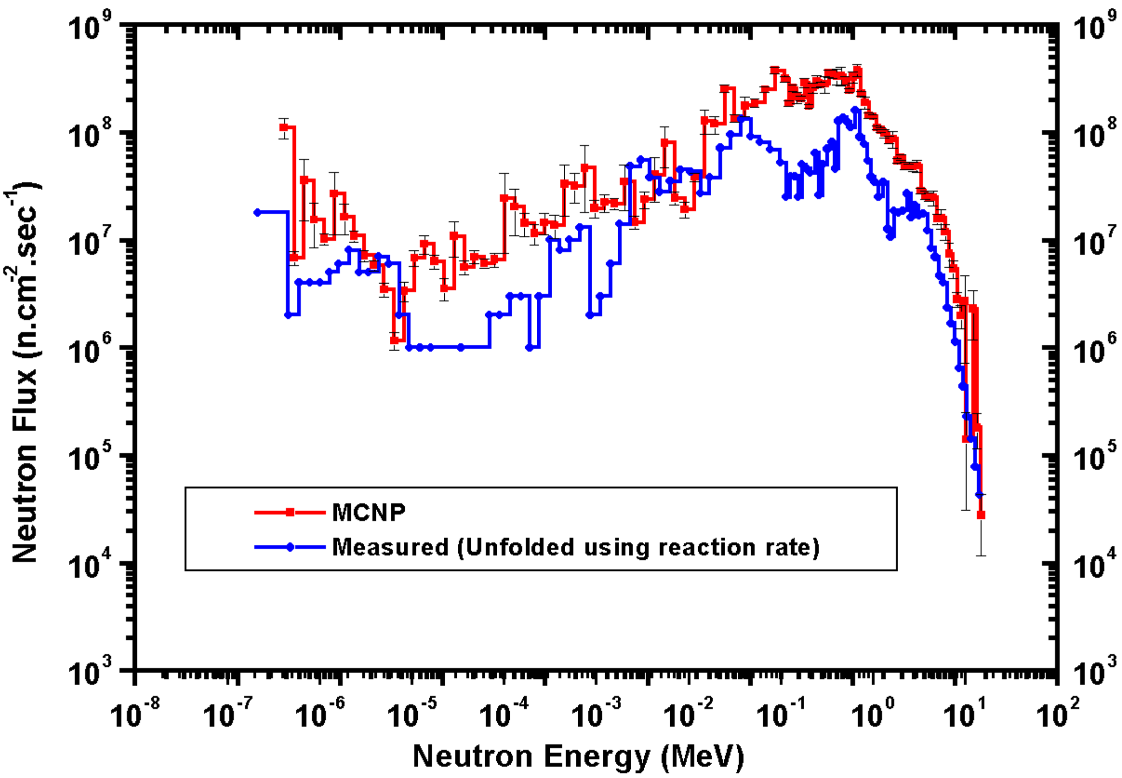


Enlarged 2D MCNP PLOT of Converter Sub-Assemblies

Fig.1 2D MCNP Plot of the Geometry Model of a Typical Experimental SET

Computations were made in parallel by different teams, SRI participating in the program with respect to calculations using MCNP code. Some geometrical details of the simulations are shown in Fig.1. By way of illustration of the results obtained, the calculated and measured neutron spectra at incident face of model shields are shown in Fig. 2. Comparison of reaction rates indicates the calculated values to be uncertain by a factor of 4 to 5 for an attenuation of 10^4 or 10^5 in boron carbide/borated graphite shield models.

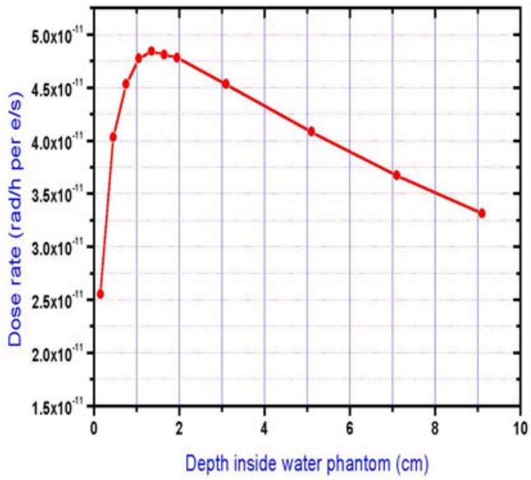
Fig.2. Comparison of MCNP calculated and Measured Neutron Group Flux at the Incident Face of the Shield Model along Core Centre Line



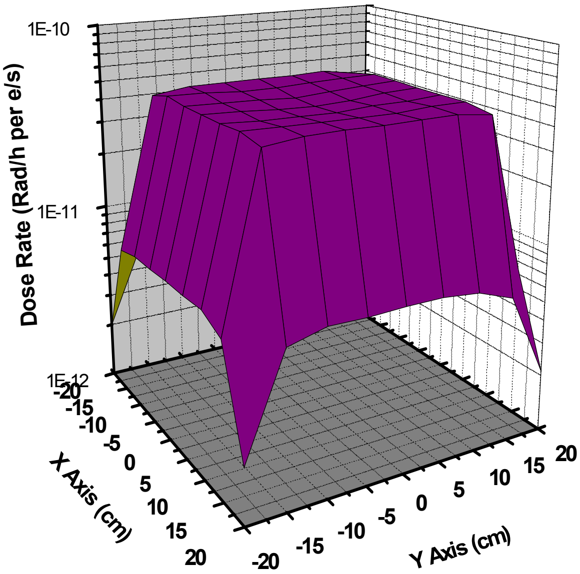
STUDIES ON BEAM CHARACTERISTICS OF A MEDICAL LINAC

High-energy electron linear accelerators are being widely used in industry and in medicine. Target used for electron impingement and the flattening filter are the main components that decide the quality of X-rays generated and dose profile of the beam, which are parameters of practical significance in therapy and treatment planning. Considerable efforts go towards their design. Obviously, the ability to calculate radiation doses accurately forms an essential requirement in support of the design. One such design evaluation task taken up was with respect to a 6 MV LINAC. Dose characteristics inside a water phantom due to 6 MV beam was studied using MCNP transport code.

The simulated performance characteristics such as dose build-up in water phantom as a function of depth, 3D plot of dose rates at build up region and percent depth dose contours are presented in Fig. 3. The dose rates are shown in rad/h normalized to per electron striking the target per second. The maximum in the dose rate buildup occurring at a depth of 1.5cm as well as the pattern of buildup agree well with values reported in literature. The simulation shows a square field of uniform dose rate having a marginal variation of +6% to –4% with respect to the mean. The simulation results also showed that the flattening filter reduces the field by a factor of 2. Dose range of 90-100% extends up to 6cm depth inside the water phantom, which is a desirable characteristic of higher energy beam for radiation therapy. The calculated dose rate of 312 rad/min for a target current of 67.27mA agrees remarkably well with the measured valued of 307 rad/min.

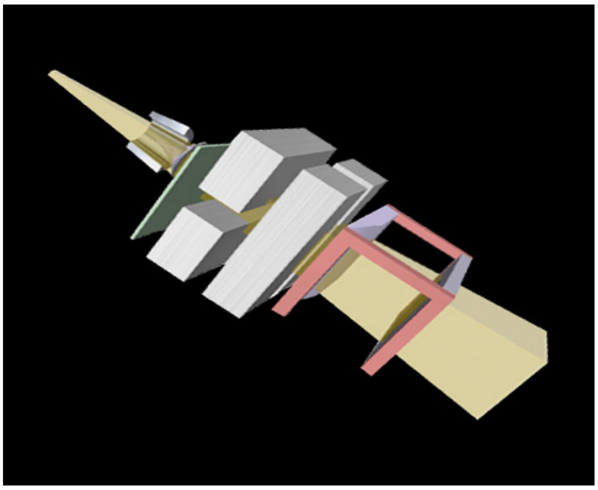


Computed Dose Build up inside
Water Phantom for 6MV Beam

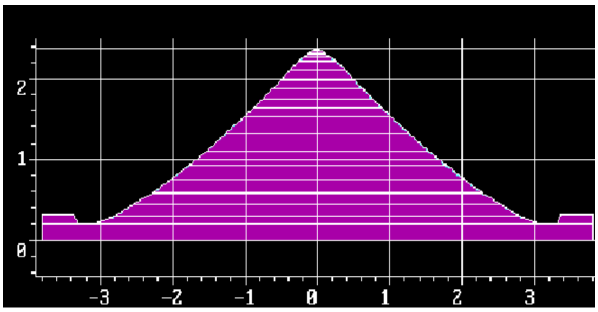


3D Surface Plot of Dose Rates at
Maximum Depth Dose Plane

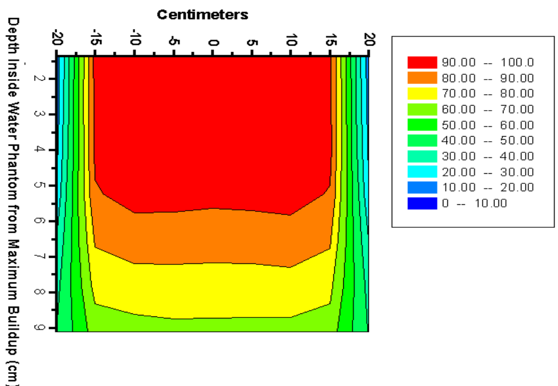
A typical Linear Accelerator based Treatment System



Model Showing Different Parts in the Beam Path of a Linear Accelerator



MCNP 2D Plot of Flattening Filter Model
for 6MV Electron Beam



Percent Depth Dose Contours
Inside Water Phantom

Fig. 3. Simulation of Beam Characteristics of LINAC: Model used and Results

SHIELDING EVALUATIONS

Shield Design Evaluation For Regualtory Clearnce

The number of installations using particle accelerators for radiation processing is on the increase. Safety evaluation for licensing of such units calls for checking the adequacy of shield design. Three such units that came to AERB for clearance have been reviewed at SRI. The table-1 gives the summary information of each of these accelerators.

Table1: Summary Information of Accelerators Reviewed for Shielding Adequacy

Accelerator Type	Beam Energy & Current	Company Commissioning	USE/APPLICATION
Proton Accelerator	16.5 MeV 80 μA	RMC, Mumbai	Production of Short Lived Positron Emitting Isotopes
Electron Accelerator	1.5 MeV 15 mA	Radiant Cables India, Limited	Production of Long Lasting Cables
Electron Accelerator	3.0 MeV 50 mA	NICCO Corporation Limited, Kolkatta	Production of Long Lasting Cables

Shielding calculations were carried out by the standard method suggested in NCRP Report-51 for the parameters of the accelerator in consideration. The required shielding thickness was computed and compared with the proposed shield design of the accelerator. The shield design was checked to see the adherence with design dose criteria of AERB. Necessary recommendations were made in cases where the design dose criteria are not met.

Shielding Evaluation Of Panbit Irradiator

A typical application involving shielding of gamma radiation is that of a PANBIT blood irradiator. A unit that was being exported to Bangladesh consisted of 100 Kilo Curies of Co-60 in 24 cylindrical pencils arranged in the circumference of a source cage. The source cage has a dimension of radius 6.3 cm and a height of 39 cm. For computation of the dose rates, the source was modeled in different ways as (a) point source (b) cylindrical source and (c) cylindrical shell source, shielded by lead and iron. The dose rates at 5 cm and 100 cm along the axial and radial directions were computed. Point source approximation calculations have been performed by MATHCAD and verified by hand calculations. Cylindrical source approximation computations have been performed by MATHCAD as given by Jaeger Shielding Compendium and by point kernel integration shielding code QAD-CGGP. Cylindrical shell computation has been carried out using QAD-CGGP.

The results obtained using different ways of modeling the source were compared to validate the method used. The cylindrical shell source model closely resembles the source description given in the PANBIT irradiator. The typical values of dose rates estimated are about 43 mR/h in the radial direction a distance of 5 cm from the surface (assuming a shield consisting of 26 cm of lead and 1 cm of iron), and about 35 mR/h in the end position (bottom) at 5 cm from the surface (assuming a shield consisting of 26 cm of lead and 3 cm of iron). The evaluation pointed out the need for additional shielding of 5 cm lead to reduce the dose rates to the stipulated 2 mR/h at 5 cm from the surface and 1 mR/h at 1 m from the surface. However the results showed that the shielding provided is adequate for the transport purposes.

SHIELDING DESIGN

Kamini Beam Port Shield Optimization

KAMINI is a U-233 fueled research reactor, which is used as a neutron source for radiography and activation analysis. It has three beam ports (south, west and north). The space available for carrying out experiments is very limited. There was a need to optimize the shield structure. Such a study was carried out for south and west beam ports using MCNP, which is briefly described below.

Shielding Optimization of South Beam Port

Source Term

There are three main components of radiation emerging from the KAMINI reactor south beam: neutrons, prompt gamma rays from the core, and fission product gamma rays from the irradiated fuel assembly kept for neutron radiography. Of these prompt gamma rays from the reactor core do not contribute significantly to the dose at the points of interest. The measured neutron flux levels at 30kW of operation are 0.46×10^{8} $cm^{-2}.sec^{-1}$ ($0 < E < 0.4eV$) and 1.39×10^{8} $cm^{-2}.sec^{-1}$ ($0.4eV < E < 10MeV$). The spectral distribution of neutrons is also available for carrying out the shielding computations. The size of the irradiated fuel pin contained in the assembly undergoing neutron radiography is small (0.5cm diameter, 1cm long of strength of 3.7×10^{15} photons/s with average energy 0.8MeV) and may be assumed as a point source for computations. The analysis began with a scoping study based on the point kernel method. The thickness of lead shield required to attenuate these photons to acceptable levels was found to be 25 cm, which is to be kept on top of the proposed shield structure.

Assessing the source term is very important for shield design. For this purpose, the dose rates at several locations due to the existing shield structure were estimated and compared with measured values (see Table 2) The computed values agree within a factor of 2-3 for capture gamma dose rates and even better for neutron dose. The differences are partly due to input uncertainties and partly due to statistical uncertainty of Monte Carlo method.

Table 2: Comparison of measured and calculated dose rates at south beam port

Detector Location	Neutron Dose Rate (μSv/h)		Gamma Dose Rate (μGy/h)	
	Measured	Calculated	Measured	Calculated
(At door)	45	39	25	72
(20cm from door)	20	34	15	37

With the source term as above, the dose rate levels in the radiography pit is of the order of 185 Sv/h, which calls for an attenuation factor of 10^{7}. It is found from scoping studies that a concrete thickness greater than 90 cm can provide the required neutron attenuation factor. On account of floor loading considerations, the concrete thickness (50 cm till floor level) and lead shield thickness (25 cm to attenuate photons from the irradiated fuel assembly kept for neutron radiography) have been fixed. Therefore, a parametric analysis was done to arrive at a paraffin shield thickness required to reduce the dose rate levels to desired values. With the rough estimate of shielding thickness MCNP computations were performed for the proposed shield structure (Fig. 4) and following conclusions were made.

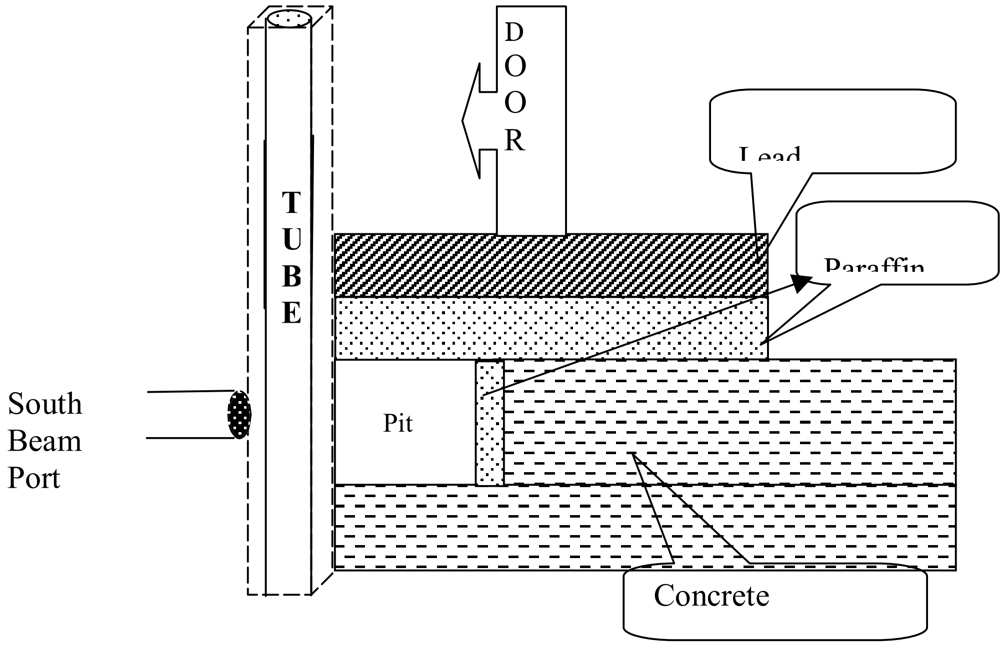


Fig.4: Elevation view of proposed shielding structure at the south beam

Based on the analysis, the existing shield structure in the radiography pit was modified by using a movable shield on top of the pit, with dimensions of height 50 cm (25 cm of paraffin and 25 cm of lead), width 75 cm, and length of 172 cm extending about 95 cm into the adjoining room. Since, the width of the movable shield is not adequate, a fixed shield of width 50 cm on either side of the movable shield is necessary to reduce the dose levels.

Dose rates have been measured after placing the recommended shields and the maximum dose rates have been observed to lie below 100 μSv/h, complying quite well with the design intent.

Shielding Optimization of West Beam Port

Shield design for the West beam port proceeded along similar lines. Neutron and capture gamma dose rates at top and side of the existing shield configuration around the west beam port were computed using MCNP and compared with the measured values (see Table 3) to have an independent assessment of the neutron source term used for the subsequent transport calculations.

Table 3: MCNP COMPUTED DOSE RATES ON TOP AND SIDE OF THE EXISTING SHIELD AROUND THE WEST BEAM PORT OF THE KAMINI REACTOR (30KW)

Location	Neutron Dose Rate (μSv/h)		Photon Dose Rate (μSv/h)	
	Calculated	Measured	Calculated	Measured
Top and at the centre of movable concrete block	0.23	BKG*	0.36	BKG*
Radiation Physics Area (80 cm above the floor level)	0.43	BKG*	1.1	BKG*

* BKG is 0.1μSv/h for neutrons and 0.3μSv/h for photons.

Scoping studies based on NCRP-51 report was made and found to be approximately 7 tenth value layers are required for neutron dose rate reduction to acceptable values, including the capture gamma dose rates. Hardly about 1m space is available for providing the shield.

Hence, a composite shield of 50 cm paraffin and 50 cm concrete (density 3.6 g/cm^3) is proposed in the forward direction beyond the experimental cavity of length 1m. This would correspond to 4 and 2.5 tenth value layers respectively for neutron and capture photon.

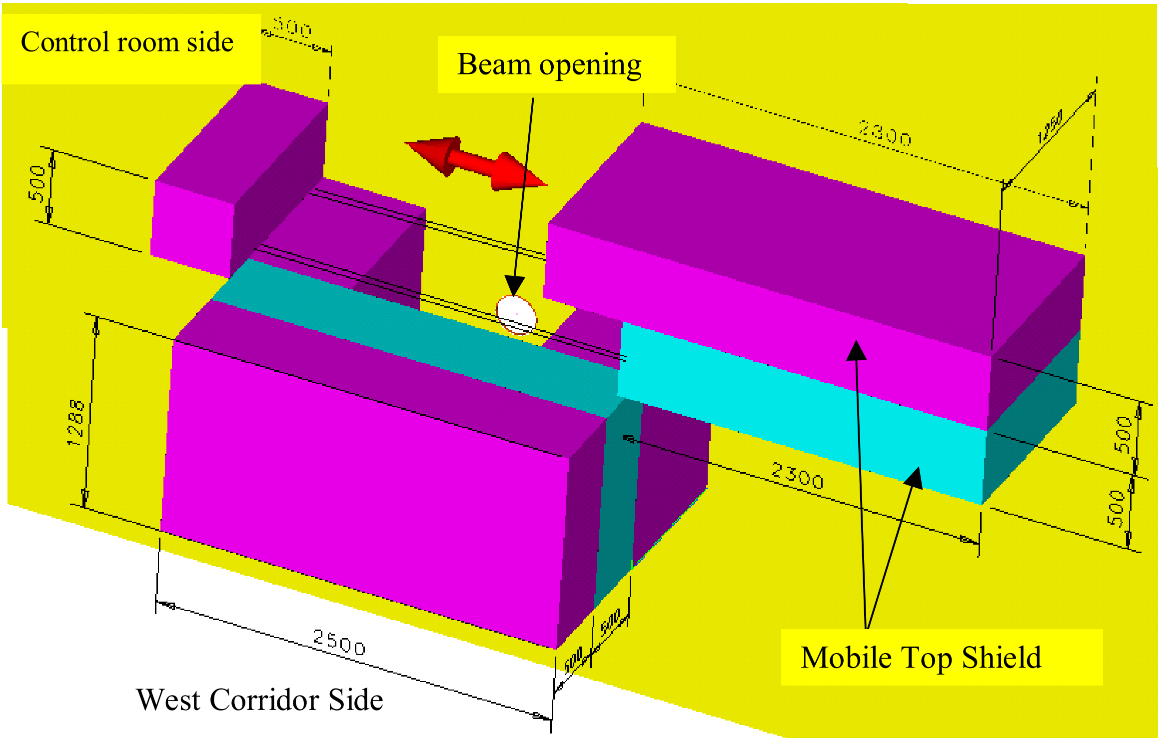
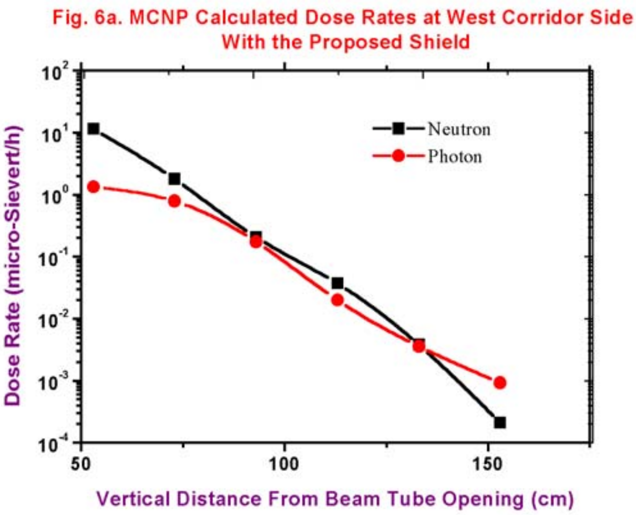
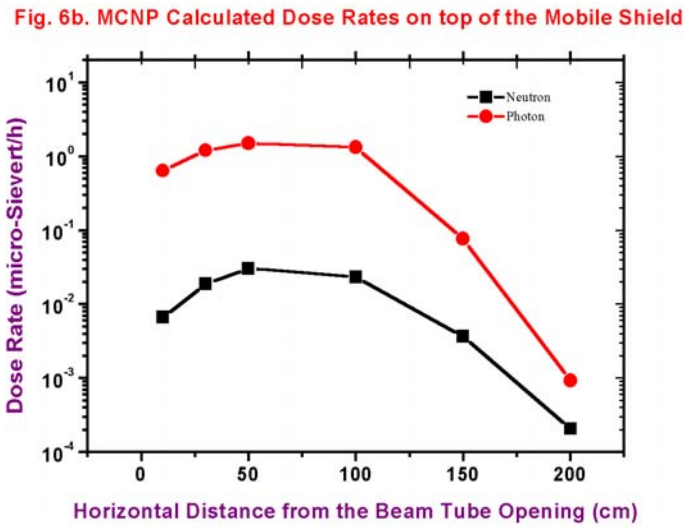


Fig.5: Isometric View of the Proposed Shield Structure around the West Beam Port

top shield is proposed (see Fig. 5). The dose rates computed along the central line at different locations on top of the proposed shield using MCNP code are presented in the Fig.6a and 6b.





Shielding Design For Radiation Streaming In PFBR

Radiation streaming studies through ducts and voids is different from radiation transport through bulk materials. All radiation transport codes are not suitable for such streaming studies. Special computer codes are available to treat the radiation transport through the irregularities such as ducts and voids present inside the bulk shield. MCNP is a suitable code for such radiation streaming studies. Suitability of MCNP for streaming calculations has been verified using experimental data available in literature. It is seen that the dose rates computed by MCNP code for all the experimental cases are in good agreement with the measured data and are within a factor of 2.

An application of MCNP code was made to assess the streaming radiation through Top Shield (TS) of PFBR. The Top Shield of PFBR consists of Roof Slab (RS), Large Rotatable Plug (LRP), Small Rotatable Plug (SRP) and Control Plug (CP) and it forms the closure of the reactor vessel in the axial direction. It provides shielding, isolation of reactor atmosphere, approach to any of the subassembly position and provide auxiliary penetrations to the reactor from outside. By rotating the plugs to a specific angle the subassembly position is approached. To facilitate rotation, the annular gaps are provided around the plugs. The loss of shielding because of the annular gaps results in radiation streaming at the accessible areas above TS. Also many other streaming gaps exist in the top shield, such as the annular gaps in transfer arm, inside in-vessel fuel transfer machine, the intermediate heat exchanger, primary sodium pump, delayed neutron detecting system etc. Adequate shielding should be provided for the radiation streaming paths so that the dose rates at accessible locations are as per the design dose criteria. The results of shield design computations for (i) transfer arm and (ii) RS-LRP-SRP-CP are presented below.

Transfer Arm Shield Design

Transfer arm is used for transferring fuel, blanket and control subassemblies from the reactor core to the storage location and from the storage location to the in-vessel transfer post (IVTP). Transfer arm is also used to transfer fresh fuel subassemblies from IVTP to their locations in the core. The annular gaps provided for the vertical movements of the transfer arm penetrates the small rotating plug (SRP) and result in radiation streaming from the reactor core to the top of the SRP. The annular gap between the outer gripper tube and guide tube (41 mm) forms the largest streaming gap. Shield plugs are provided inside the streaming gap to stop the direct streaming paths but the reflected radiation coming to the lateral side of the guide tube results in appreciable dose rates at accessible locations on top of SRP. Another situation also needs to be considered. Due to the movement of internals of transfer arm machine during fuel handling, the shield plugs provided in the straight streaming path gets disarrayed.

The sources of radiation are (1) primary sodium gamma rays [~1.0 x 10^{10} γ-cm^{-2}-s^{-1}], (2) core gamma rays [2.2 x 10^{6} γ-cm^{-2}-s^{-1} with an average energy of 1 MeV], (3) core neutrons [1.08 x 10^{7} n-cm^{-2}-s^{-1} with the mean energy 70eV at 80cm below argon cover gas], and (4) fission product gamma rays [2.27 x 10^{17} gammas per subassembly with energy ranging from 0.01 MeV to 7.5 MeV].

Three-dimensional modeling of the gripper subassembly, guide tube and the annular gaps between them is made along with SRP through which the transfer arm mechanism passes.

A 2D view of the geometry as modeled in MCNP code for the reactor operating condition is shown in Fig.7.

Dose rates within the annular gap and on the outer surface of the guide tube on SRP top are presented in the Table 4. It can be seen that the neutron dose rates outside the guide tube is negligible but $_{11}Na^{24}$ gamma dose rates should be brought down by a factor of 10 to meet the shield design criteria. Hence, one-tenth value layer of Steel (80 mm for 2.76 MeV gamma ray) around the guide tube from the SRP top to a height of 2600 mm is proposed, to bring down the dose to about 10 µSv/h or less. Neutron dose rate at any of the locations above SRP is negligible compared to gamma dose rates.

Table 4: Dose Rates at SRP Top

Dose Receptor Location (See Fig. 6)	**Dose Rates ((µSv/h)**	
	Neutron (includes capture gamma)	**$_{11}Na^{24}$ Gamma**
(D_{01}) 41 mm annular gap	1.22E+01	1.64E+05
(D_{02}) Outer edge of guide tube	6.66E-03	1.38E+02
(D_{03}) Outer edge of 80 mm Steel	1.24E-05	1.30E+01

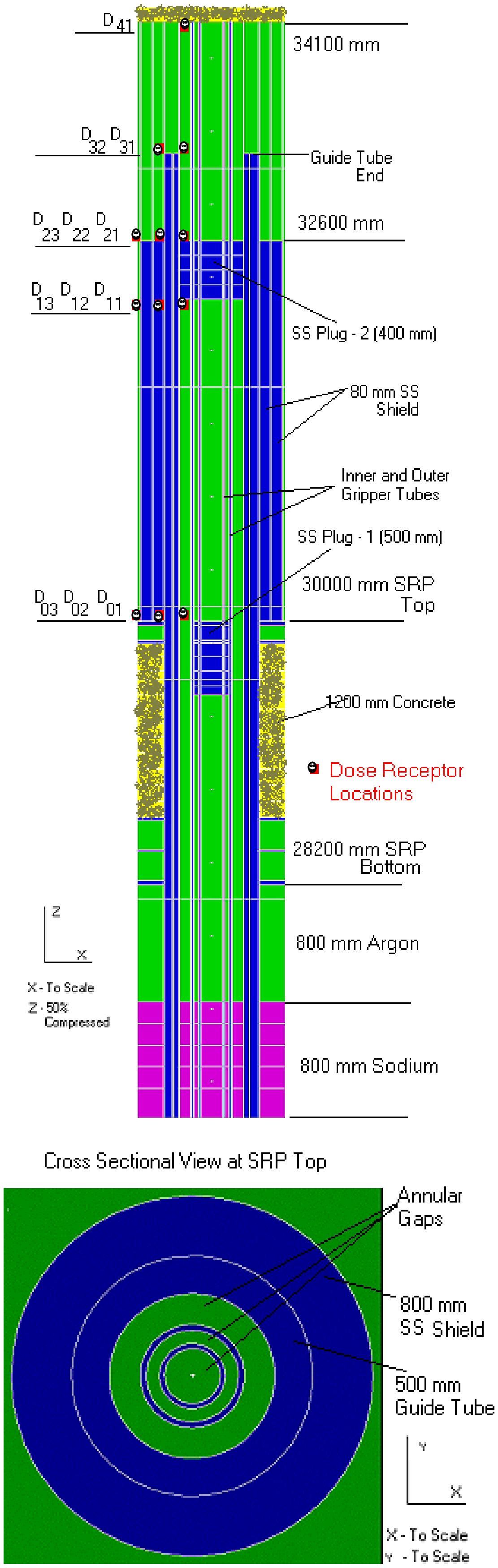


Fig.7: 2D View of Transfer Arm as Modeled in MCNP Code for Reactor Operation

During fuel-handling operation, the gamma dose rates computed at all the accessible locations around the transfer arm are about 2 μSv/h or less.

Some pertinent observations during the study are:
The steel plug-2 of 400 mm thickness reduces the streaming dose rates by 5 orders of magnitude The steel plug-1 of thickness 500 mm is found to be ineffective in reducing the dose rates at the accessible areas both during reactor operation and during fuel handling condition.

The computed dose rates could have a maximum uncertainty of a factor of 2. Since the dose rates computed at controlled access areas are only about 10 μSv/h, the proposed shield meets the shield design criterion of 25 μSv/h adequately.

Reduction in annular gap by a factor of 2 can reduce the required shield thickness by a factor of 4.

Shield Design for Streaming Paths in Top Shield

The radiation source term for the dose rate computations for the streaming paths in this case is same as in the case of transfer arm. But, there are three streaming paths, that is the annular gaps between (1) roof slab (RS) and large rotatable plug (LRP) (2) large rotatable plug (LRP) and small rotatable plug (SRP) and (3) small rotatable plug (SRP) and control plug (CP). Three-dimensional modeling of the top shield is made using MCNP. The largest annular gap among the three is 25 mm. Since neutron dose rate was found to be negligible from the scoping studies, only gamma ray dose computation is made.

The gamma rays from ${}_{11}Na^{24}$ are the most significant contributor to the dose rate at the accessible locations. The results of the computations are summarized below.

The dose rates at the accessible locations outside the RS/LRP, LRP/SRP and SRP/CP top structures were in the ranges of 5 to 70, 5 to 30 and 5 to 40 μSv/h for the existing design. The acceptable dose rate above TS is 25 μSv/h, which is further reduced to 10 μSv/h by applying a safety factor of 2.5. Therefore, complimentary shielding made up of CS with density 7.8 gm/cc was considered. With this, the dose rates were found to be about 10 μSv/h or less at all the accessible locations after studying the combined effect from all the annular gaps. During fuel handling operation i.e. one day after shutdown, with 33% of nominal power saturation activity of ${}_{11}Na^{24}$, the gamma dose rate computed at all the accessible locations above TS are about 3.5 μSv/h.

Shielding Design For PFBR RCB During CDA

One of the primary objectives of the reactor containment building (RCB) is to contain the radioactive material release during a core disruptive accident (CDA), thereby minimizing the radiation exposure of site personnel and members of public. The thickness of RCB walls is decided so as to minimize the external radiation exposure as well as to withstand the over pressures generated during the burning of sodium under the CDA.

The external gamma dose rates at various distances from RCB at different elevations from the ground level due to bottled up fission products inside RCB during a hypothetical CDA in Proto-type Fast Breeder Reactor (PFBR) was calculated. The dose rate computations were made for (i) external gamma dose rate inside control room and backup control room to assess the habitability and (ii) the dose rates on the inner walls of the RCB to assess the amount of dose rates seen by the instruments inside RCB.

The estimated total core inventory of radioactive fission products is 4.64 x 10^7 TBq, consisting of different radioisotopes of krypton, xenon, iodine, bromine, cesium, rubidium, cerium, barium, tellurium, ruthenium. A fraction of the radioisotopes from the core is released to the RCB. The release fractions suggested by three different groups namely (1) OECD expert committee release fractions (2) SNR release fractions and (3) European Fast Reactor release fractions were considered in the computations which were made using the QAD-CGGP point kernel code.

The RCB in PFBR is modeled as a rectangular parallelepiped having graded concrete walls of different thickness on its sides. The control room on one side of the RCB shares a common graded concrete wall with RCB. This common concrete wall is having a thickness of 1.5 m till the control room height. The backup control room with a wall thickness of 60 cm concrete is 5.5 m away from one of the outer surface of the RCB concrete wall.

The dose rates on the inner walls (mid-big refers to centre of the length of the wall, mid-small refers to centre of the breadth of the wall, and corner refers to the meeting point of the walls) of the RCB are shown in Fig.8.

Dose rate and the cumulative dose expected in the control room and the back up control room are shown in Fig. 9 to Fig.12.

Fig.8: Variation of cumulative dose with time inside RCB (EFR)

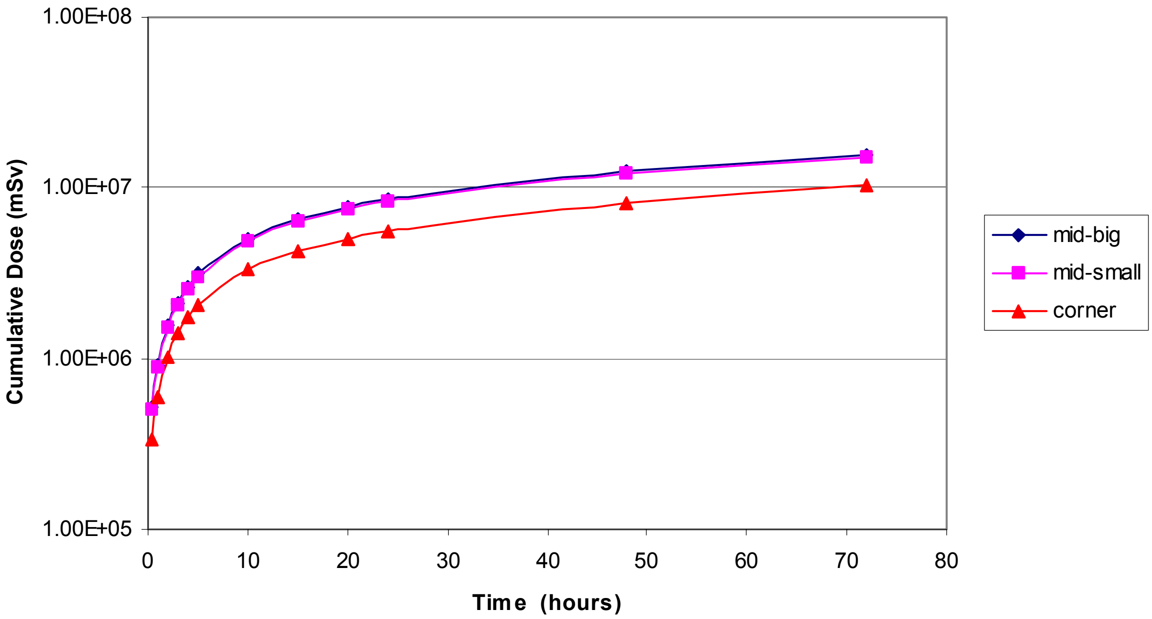


Fig.9: Variation of Dose Rate in Control Room at Different Distances from the RCB Wall after Hypothetical Core Disruptive Accident (EFR Release Fraction)

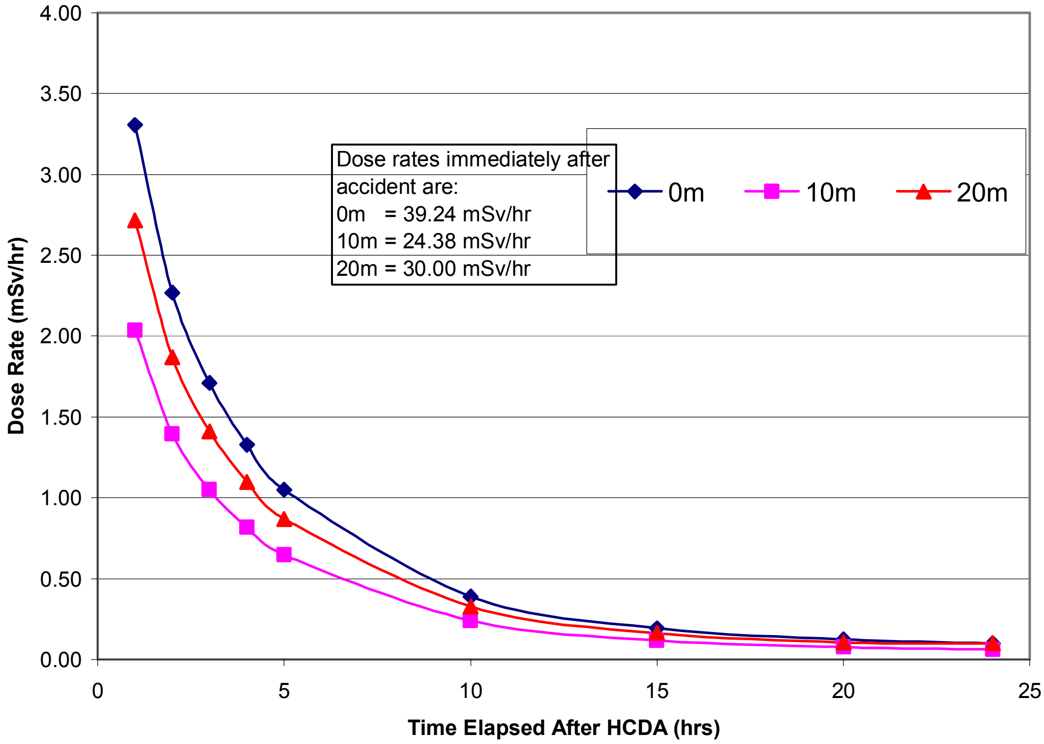


Fig.10: Variation of Cumulative Dose in Control Room at Different Distances from the RCB Wall after Hypothetical Core Disruptive Accident
(EFR Release Fraction)

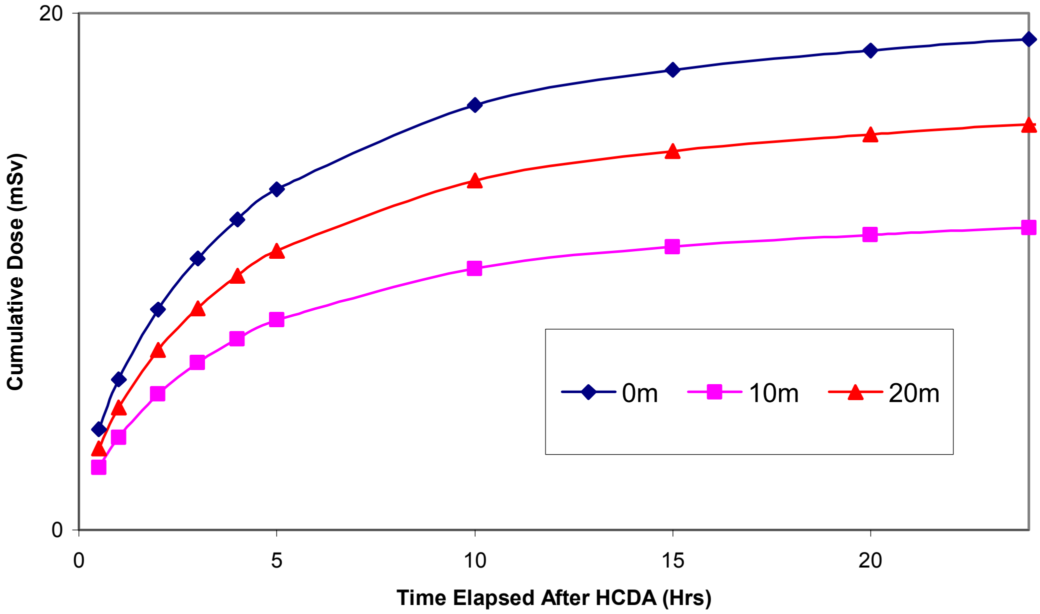


Fig.11: Variation of Dose Rate in Backup Control Room (maximum dose location) after CDA
(OECD Release Fractions)

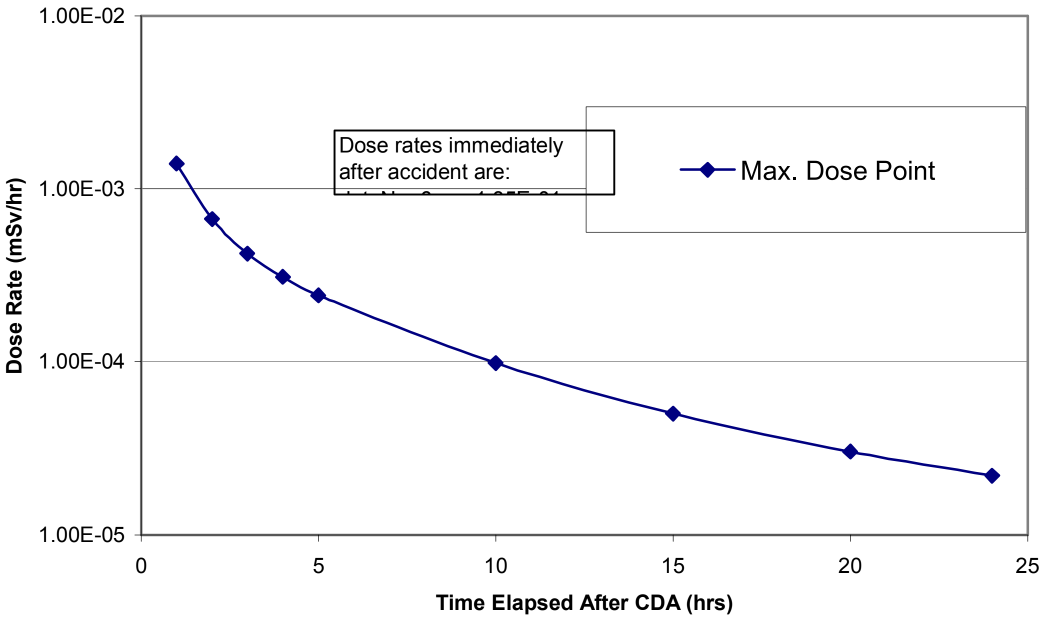
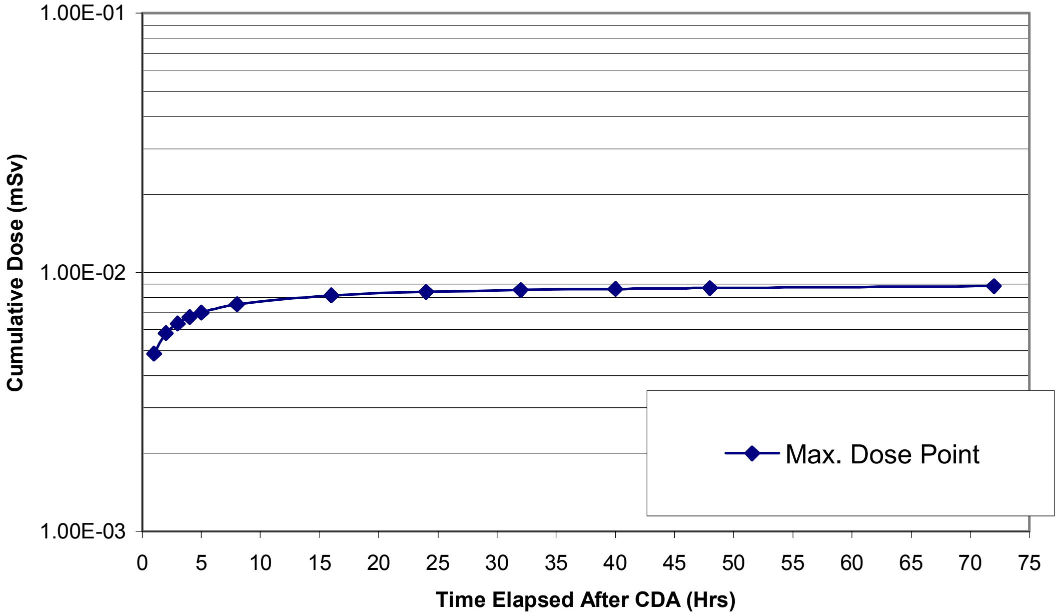


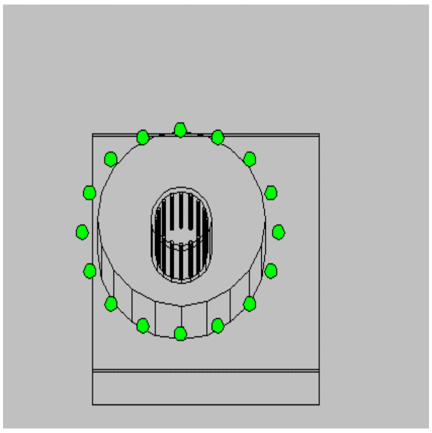
Fig.12: Variation of Cumulative Dose in Backup Control Room after CDA
(OECD Release Fractions)



SOFTWARE DEVELOPMENT
GUI2QAD-3D

GUI2QAD-3D is an interface program developed at SRI to aid the user in the preparation of input for QAD-CGGPIC, a point kernel shielding code, written in FORTRAN, for fast neutron and gamma-shielding calculations. The shielding code uses combinatorial geometry for describing the shield. The interface program enables interactive input and viewing the same in 3D geometry with arbitrary rotations along X, Y and Z- axes (see Fig. 13). The salient features of the package include

1. Handles off centered multiple identical sources
2. Axis of cylindrical sources can be parallel to any of the axis
3. Provides plots of buildup factors (ANSI-1990) and material cross sections
4. Estimates dose rates for point source slab shield configuration
5. Interactive input of CG geometry with prompt 3D view and rotation
6. Fission product decay power computation and plots for source term calculations
7. Provision to read and graphically display picture input file



This code has been validated at ORNL, USA and is available to users internationally at RSICC, USA (CCC-697/GUI2QAD-3D).

View-CXS: Neutron and Photon Cross-Sections Viewer

It is often informative and quite useful to view the radiation interaction data on which the results of radiation transport calculations mostly would depend upon. The data depend on the type of radiation, the number of materials (isotopes) and the details of interaction processes dealt in the particular cross section evaluation. For neutrons and photons, the amount of data required is voluminous owing to a large variety of interaction processes and also due to considerable variation in the magnitude of cross sections with the energy of the particle. Therefore, the data are usually condensed by splitting the range into number of intervals or

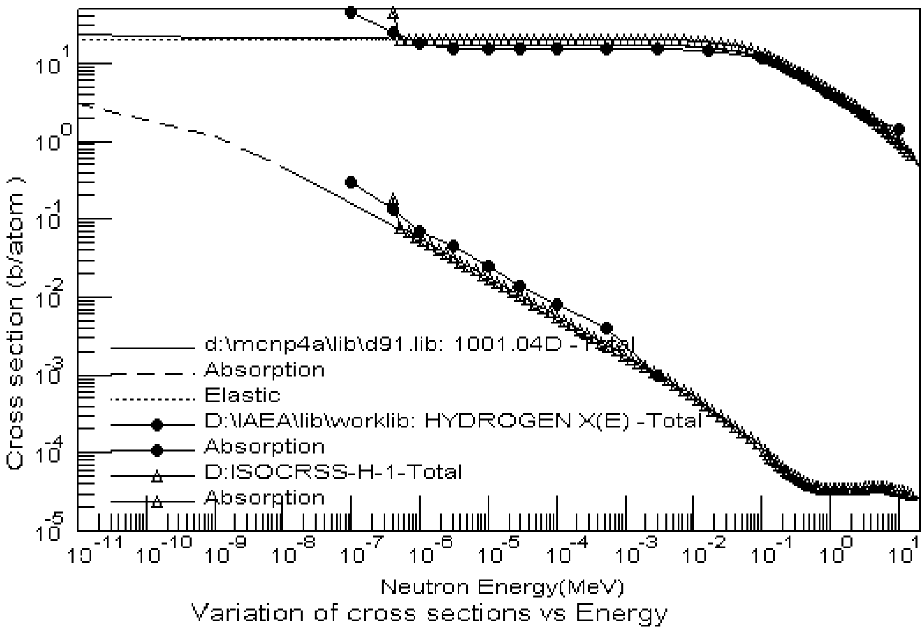


Fig.14: Elemental Comparison of Neutron Cross-Section for Hydrogen from Different Libraries

groups and fitting the original data to suitable functions to a desired degree of accuracy and then storing the coefficients of the fitted data. In order to facilitate the users to have feel for the magnitudes of cross sections and to visualize the trends, a graphical user-friendly interactive interface has been developed in Visual Basic-6. It can process neutron and photon interaction cross sections in ACE (MCNP), DLC (ANISN), AMPX (KENO) format. This code has been validated at ORNL, USA and is available to users internationally at RSICC, USA (PSR-514/VIEW-CXS). Typical demonstrations of plots using VIEW-CXS are shown in Fig. 14-16.

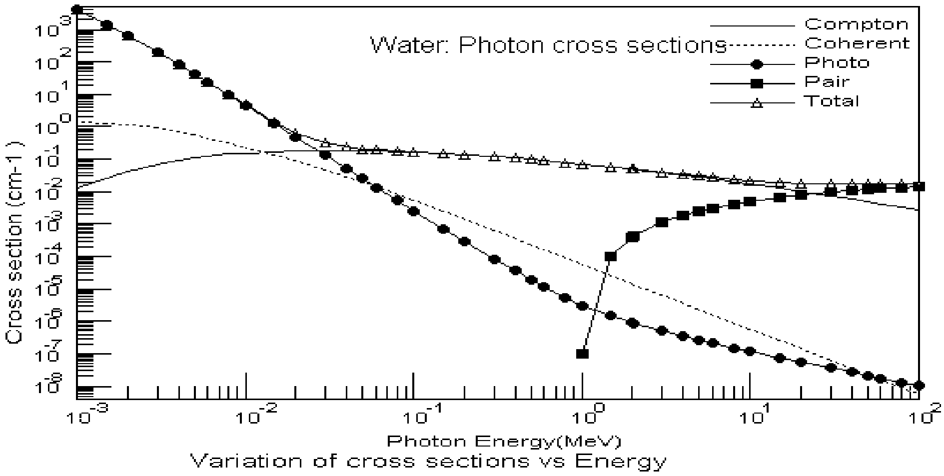


Fig.15: Differential and total Photon cross-sections of water [Atomic conc. In atoms/ (barn-cm) units are H (0.067) and O (0.0335)]

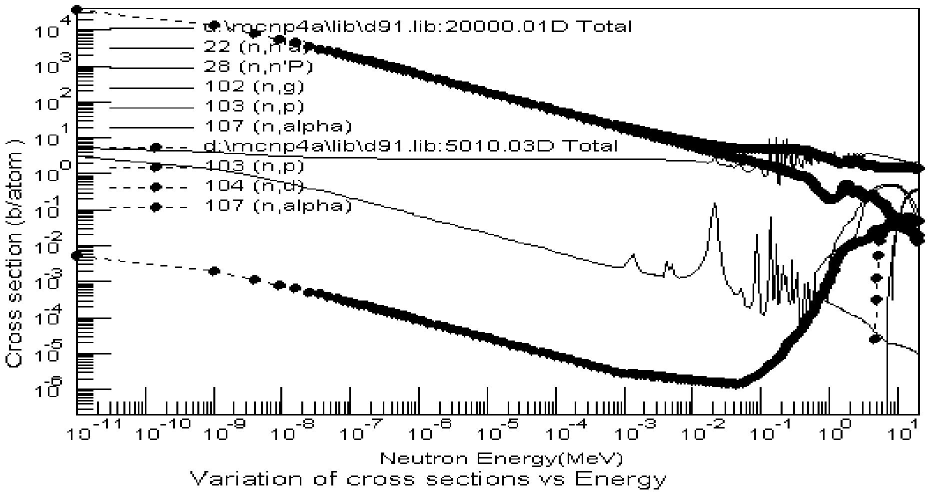


Fig.16: Differential neutron Nuclear reaction cross-sections of Cadmium and Boron-10 isotopes

2. C Remote Sensing and GIS Facility

Introduction

One of the aims of AERB-SRI is to support environmental decision-making within AERB and DAE as a whole, by generating and disseminating information about the state of environment in and around DAE installations. The effort is towards providing relevant inputs to Environmental Impact Assessment (EIA) studies of Nuclear Power Plants (NPPs). To this end, A Remote Sensing- Geographic Information System Facility (RS-GIS Facility) is being created at SRI, Kalpakkam. The facility would maintain a digital database on all the existing nuclear facilities using the past RS data and other collateral information on population, ground water, land use/land cover, radiation level etc. Table 1 and 2 show the organization of the proposed database. This program executed in collaboration with the Space Applications Centre (SAC), Department of Space, Ahmedabad is intended, to start with, to demonstrate the design and organization of a digital database and information system for carrying out EIA. Presently, the infrastructure required for the RS-GIS data processing has been established (see fig. 1). A pilot study, taking Kalpakkam as an example, is underway with the following objectives.

1) To prepare thematic maps related to land use/cover, hydro-geomorphology, ground water prospects, surface water bodies, transportation network, natural hazards such as flood, erosion etc. (Table 2) on 1:12,500 scale using IRS-1C/1D satellite data.
2) To create a digital database of all natural resources as well as socio-economic conditions.
3) To develop integrated analysis, modeling procedures and information retrieval with query system for EIA study.

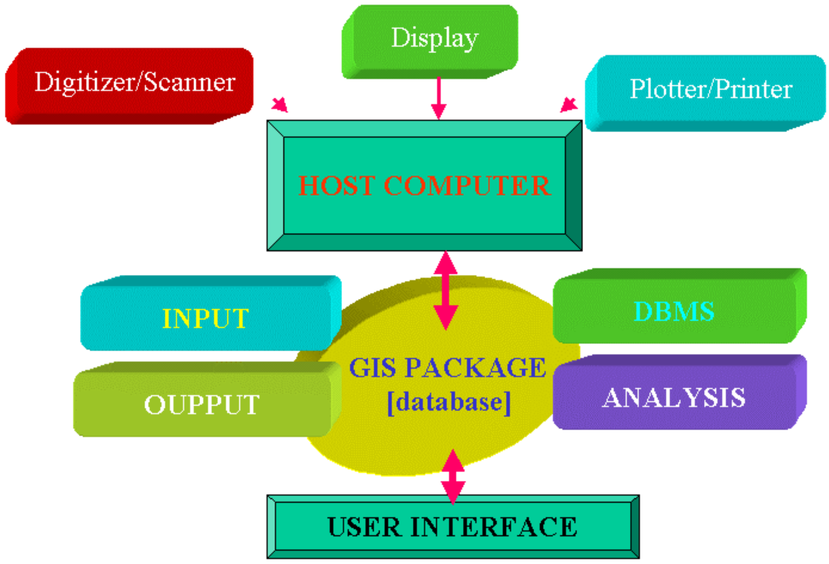


Figure 1: Schematic of GIS studies

Table-1: Standards for digital database

SPATIAL DATA BASE		
i)	Input	Thematic maps with latitude and longitude system
ii)	Scale	1:12,500
iii)	Accuracy standards Location	12.5 meters (1mm on 1:12,500 scale)
	MMU*	37.5 m x 37.5 m (3 mm x 3 mm on 1:12,500 scale)
iv)	Co-ordinate	Polyconic projection (Central longitude of the Development Authority)/UTM
v)	Registration	Four corners of SOI maps, Intersection of roads, road and stream intersection, bench marks etc.
vi)	Map standards	SOI topographic maps on 1:25,000 scale
vii)	Query units	Point, Regular and Irregular poly, Village poly and Municipalities poly.
viii)	Organisation	Tile structure with reference to SOI topographic maps
NON-SPATIAL DATABASE		
i)	Database unit	
	a) Census data	Village/Municipal ward
	b) Thematic data	Feature basis (Poly, line and point)
ii)	Link to spatial data	
	a) Census data	Village code/Municipal ward
	b) Themtatic data	Feature code
iii)	Query unit	Village and district as a whole
iv)	Organisation	Relational tables using indexed keys

*MMU = Minimum Mapping Unit (Note: The units less than this area are dissolved)

Table 2: Thematic Maps

Sr. No.	Theme	Feature type	Data Source
1	Land use* (1971, 1981,1991,2000)	Polygon	SOI map, RS data
2	Hydrogeomorphology	Polygon	RS data
3	Slope map	Polygon	SOI maps
4	Geology	Polygon	GSI
5	Ground water prospects	Polygon	RS data
6	Soil	Polygon	NBSS and LUP
7	Transportation network*	Line	SOI map, RS data
8	Drainage*	Line	SOI map
9	Natural Hazard	Polygon	RS data, SOI maps
10	Administrative boundary*	Polygon	District collectorate and census records
11	Climate		IMD
12	Demography		Census
13	Infrastructure		District collectorate and census records
14	Utilities and Services*		District collectorate and census records
15	Others (Pollution etc.)		IMD and other relevant institutions.

EIA of Kalpakkam Site: A Case Study

The Kalpakkam site is bounded by 12°25’ to 12°50’ North latitude and 79°50’ to 80°10’ East longitudes having seawater on the east (Bay of Bengal) and land on three sides.

Methodology

IRS LISS-III and Panchromatic satellite data (2001) along with Survey of India (SOI) maps (1972) has been primarily used for the generation of land use maps and change detection around the NPP site at Kalpakkam. The collateral data on dietary pattern around the NPP site and radionuclides in foodstuffs has been collected from report on “EIA/EMP for the 500MWe PFBR (2000), IGCAR, Kalpakkam” for the comparative evaluation.

The satellite data (LISS-III) has been registered with SOI map (**Figure-1**) using ground control points spread across the study area. A reconnaissance fieldwork has been carried out in the study area to collect the ground data related to various land use/cover categories. This data has been used as training sets for carrying out supervised classification in GIS environment. Finally a land use/cover map of the NPP site and its environs has been prepared using supervised classification employing maximum likelihood classifier.

The first phase of the “Environmental Impact Assessment” of Kalpakkam has been carried out with respect to the release of radionuclides into the environment.

The proposed EIA information system consists of both spatial and non-spatial data for different levels of EIA planning. The steps involved in creating the digital database using RS and GIS techniques are given below.

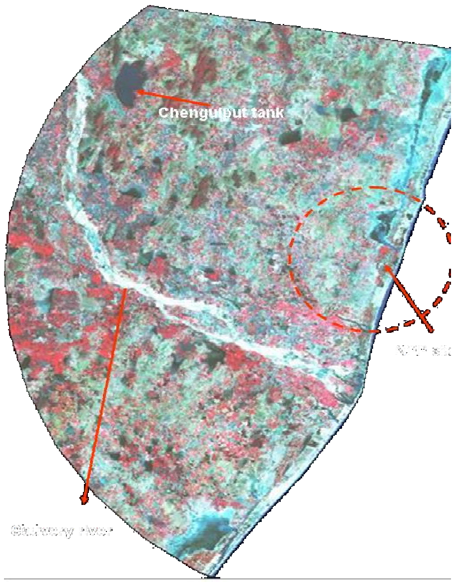


Figure 1: IRS-1C LISS-III Imagery covering Kalpakkam site.

- Generation of spatial framework in GIS environment on the basis 15' x 15' and of 5' x 5' graticule grids for the entire nuclear power plant region.
- Creation of a digital database as per the standards for each of the thematic layers.
- Integration of thematic maps using Union/Intersection techniques in GIS environment.
- EIA analysis using multi-parametric weighted index approach. The parameters to be considered for the purpose are: natural characteristics of the terrain, Geology, slope, Natural hazard, Ground Water Prospect, Transportation, Soil, Drainage etc. Problem specific modeling and sensitivity will also be attempted.

Land use transformation around NPP site

The area of about 30 km radius around the NPP has been selected to study the land transformations over a period of about 20 years using SOI map and latest satellite data (2001). At first, SOI map on 1:250,000 scale surveyed in the year 1981 has been evaluated and a land use/land cover map prepared which is presented in **Figure-2.** The extent of area under different land use/cover categories is given in **Table-3.** Then the land use/cover map using LISS-III data was prepared employing supervised classification based upon maximum likelihood classifier. The map is presented in **Figure-3** and the area statistics in **Table-4.**

The area under built-up land has increased from 1.64 % to 6.40 % by encroaching upon agricultural land, wastelands surrounding around the settlements. As a result, the area under agriculture land has reduced from 82.87% to 74.67 %. Apart from the marginal agricultural lands, some of the productive agriculture land around Chengleput settlement has been lost for urbanization. The forest cover has been reduced from 3.54 % to 3.21 %. The area under forest cover includes not only the normal vegetation but also plantations. Some of the forest cover has been converted into scrubland. It has also been observed that some of the scrubland also got converted into plantation, may be due to afforestation measures taken in the study area. The area under wasteland has drastically increased from 0.36 % to 4.82 %. Similarly the area under sand category also increased from a meager 0.73 % to 7.59 % due to the drying of water bodies and drying up of grasses etc.

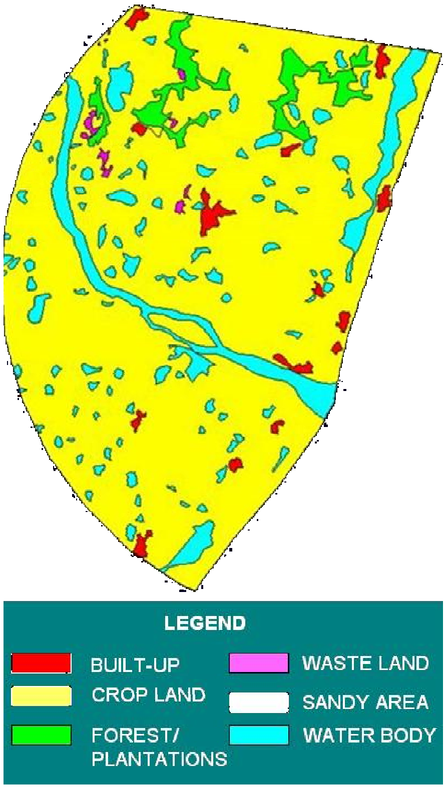


Figure-2: Land use/cover map prepared using SOI

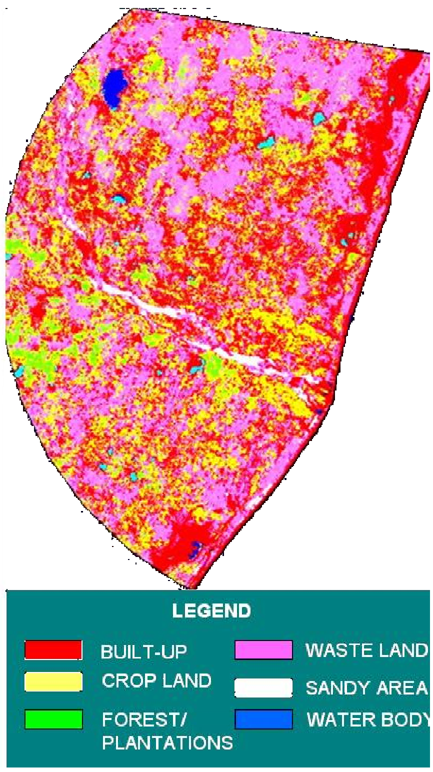


Figure-3: Land use/cover map prepared using IRS LISS-III

Table-3: Area under different land use categories (1981)

S.No.	Land use cate gory	Area in ha	Area in per cent
1	Built-up	1964	1.64
2	Agriculture	98998	82.87
3	Forest	4223	3.54
4	Wasteland	435	0.36
5	Sand	872	0.73
6	Water bodies	12968	10.86
Total		119460	100.00

Table-4: Area under different land use categories (2001)

S.No.	Land use cate gory	Area in ha	Area in per cent
1	Built-up	7650	6.40
2	Agriculture	89200	74.67
3	Forest	3836	3.21
4	Wasteland	5756	4.82
5	Sand	9062	7.59
6	Water bodies	3956	3.31
Total		119460	100.00

Summary

The preliminary change analysis carried out in this study suggests that the land use transformation has been due to various developmental and other natural activities but not due to setting up of NPP. It has also been observed that the level of radioactivity in and around the NPP is either within the limits or negligible to make any impact on the environment.

The present study is the result of only a preliminary analysis. However, a detailed study is planned to create a comprehensive information system to carry out environment impact assessment at a large scale. It has been noticed that detailing the level – III land use categories is must, if one wants to understand the radiological impact of nuclear installations due to changes in climate flora and fauna, cropping patterns soil and demography. Further work is in progress.

2.d Radionuclide Migration Studies

INTRODUCTION

Disposal of High Level Radioactive Waste in deep geological formations is considered worldwide as the suitable option to safeguard the man and environment for extended period. Before a site is selected for disposal, its suitability has to be investigated by detailed study of the interaction of the waste forms with surrounding geo environment. Given the geological, hydrological, geochemical, geophysical and geotechnical characteristics of the site, one has to study the long-term behaviour of radionuclides essentially by mathematical models. The migration behaviour of radionuclides through the rock mass is a very slow process, which is governed by mechanisms like advection, and/or diffusion, particularly the latter. One of the key inputs to these models is the so-called diffusion constant. The slowness of the diffusion process makes direct experimental measurement of this parameter quite time consuming. Moreover, laboratory scale experiments also suffer from the limitations due to the material complexity, difficulty associated with reproduction of the boundary conditions controlling the governing mechanism(s), etc. Simulation of accelerated ageing is a convenient way out of this situation. Centrifuge modelling, which is emerging as a useful technique to study and model various geo-environmental problems, offers promise in this respect.

Principles of Centrifuge Modelling

In a centrifuge, the rock sample (model) experiences the same magnitude and distribution of self-weight stresses as those of its prototype. The main difference between the model and its prototype is that the linear dimensions of prototype are scaled down by a factor of *N*, at centrifugal acceleration *N* times greater than g, the acceleration due to Earth's gravity.

Details of Experiments

An attempt has been made to simulate migration of radionuclides through the rock mass in a small geo-technical centrifuge. Simulation was done for the cases of non-sorbing (Chloride and Iodide) and sorbing (Cesium and Strontium) ions. Transport through the intact and fractured rock masses was considered. To achieve this conventional 1-g and centrifuge (*N*-g) tests have been performed.

The Centrifuge facility, depicted in Figure 1 available at Indian Institute of Technology, Bombay has been employed. The details of the centrifuge are presented in Table 1. Diffusion experiments were carried out using specially designed diffusion cells as shown in Figures 2 & 3 for testing intact (Type-I) and fractured rock (Type-II) samples respectively. These experimental setups were calibrated prior to conducting experiments. Concentration of diffused chloride ion was evaluated by monitoring the conductivity changes with the help of conductivity meter developed by Ultra Sensitivity Devices and Technology Section (USDTS), IGCAR, Kalpakkam. The concentration of iodide ion was determined with the help of a UV spectrophotometer and the concentration of Cs and Sr ions were determined with the help of Inductively Coupled Plasma-Mass Spectrophotometer (ICP-MS) respectively.

Figure 1 Geotechnical centrifuge

Table 1: Centrifuge details

Type	Swinging buckets on both sides of the arm
Arm radius	200 mm
Max. outer radius	315 mm
rpm range	250-1000
Max. acceleration	300 g
Capacity	0.72 g tons
Spin-up time	20 s
Spin-down time	80 s

Computation of Diffusion Coefficient (D_i) and Results

The concentration of the solution in the measuring compartment, C_t, with Ultra-pure water (conductivity <1µS) is monitored, with varying time and centrifugation efforts. This concentration expressed relative to the source concentration (i.e. C_t/C_0) versus time t is plotted as the diffusion curve. Diffusion coefficient of the ion is evaluated from the slope of the diffusion curve and by solving the 1-D diffusion equation. Some of the measured and processed data are summarized in Table 2.

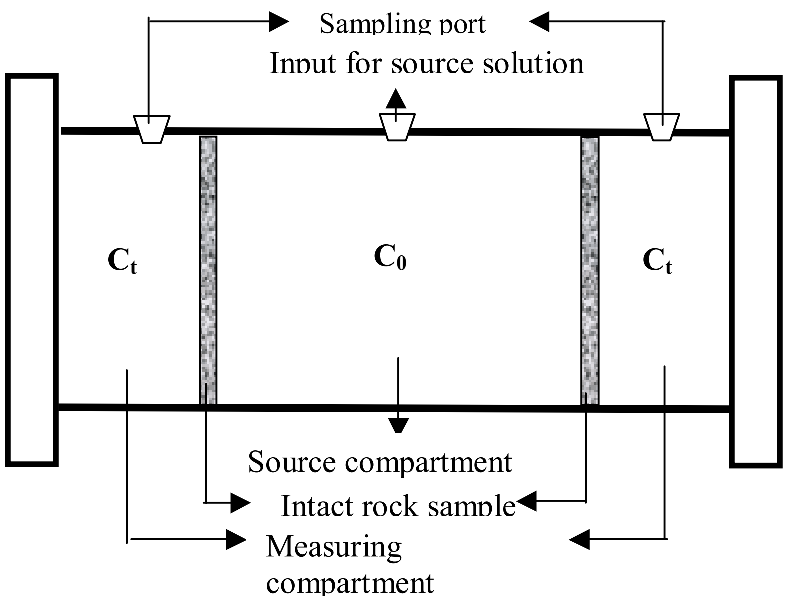


Fig. 2 Type-I Diffusion cell for testing intact rock samples

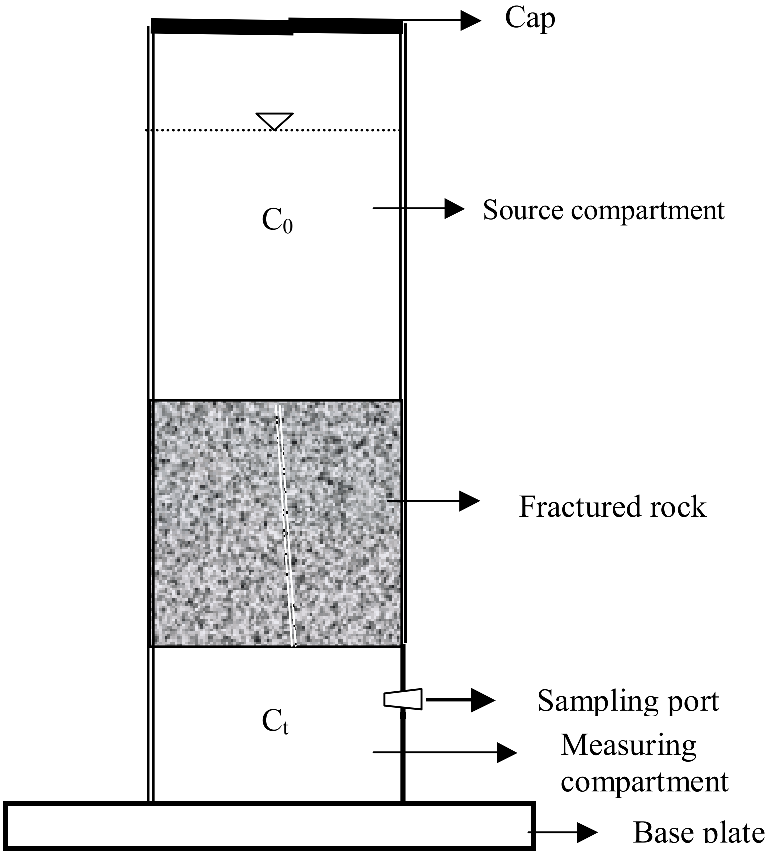


Fig. 3 Type-II Diffusion cell for testing fractured rock sample

Table 2: Diffusion Coefficient for Cl, I, Cs and Sr ions
(for 3 mm thick fractured rock sample)

Source Solution	Ion	D_i ($\times 10^{-14}$ m^2/s)			
		N			
		1-g	50	75	100
NaCl	**Cl**	**2.1**	**3.30**	**7.59**	**11.6**
CsI	**I**	**0.31**	**2.03**	**2.63**	**4.89**
	Cs	**1.29**	**8.34**	**1.78**	**2.62**
$SrCl_2$	**Sr**	**2.51**	**9.35**	**1.59**	**3.30**

Conclusions

The present set of experiments was done with a view to examining the potential of geotechnical centrifuge for measuring diffusion characteristics of slow processes spanning a very long time scale.

The important point to note is the time required for the experiment. In the case of Cl ion the actual process takes 50 days to attain $C_t/C_0=20\times10^{-4}$ through a 6 m thick fractured rock mass, whereas the same could be modeled in the geotechnical centrifuge in 7 min. The corresponding values in the case of a 0.3 m thick intact rock mass are 520 days and 75 min, respectively. Likewise, for the sorbing ions time taken to achieve $C_t/C_0=20\times10^{-4}$, corresponding to 1-g and 100-g, would be of the order of 10^8 s (=1157 days) and 10^5 s ($\cong$1 day). The study demonstrates usefulness of the geotechnical centrifuge for modeling radionuclide migration through the rock mass, in a short duration.

3. Collaborative Research Programme

Introduction

One of the objectives of SRI is to promote inter institutional coordinated research programme in areas of interest to regulatory functions of AERB. One such collaborative research programme initiated by SRI was aimed at development of a realistic atmospheric dispersion model suitable to coastal sites. This research project is successfully completed in collaboration with four premiere institutions outside DAE, clearly demonstrated the benefits resulting from a planned coordinated programme with a well defined and time bound research goals.

An outline of the research theme

In the design and operation of nuclear facilities, the radiological consequence of the effluents released through the stack would have to satisfy the safety limits. A thorough knowledge of the atmospheric mean wind and turbulence conditions over the specific site should be assimilated as they determine the path ways for effluent particles. These conditions are governed by the processes in and the characteristics of the Atmospheric Boundary Layer (ABL) and change continuously with time as well as space. The nature of the terrain introduces further complexities in ABL.

For normal operating conditions, one needs an estimate of the average annual exposure likely at different locations in and around the site for distances up to several kilometres. AERB follows a traditionally well established guideline of international stature for estimating the radiological safety parameters. A widely used method for this purpose is based on Gaussian Plume Model (GPM) coupled with site specific meteorological data such as annual wind rose, stability class and the eddy diffusion parameters for the pollutants. The GPM approach is grossly simple and yet fairly reliable over short distances up to ten kilometres and over plain terrain. It is to be noted that even for this simplified model the eddy diffusion parameters are not estimated for Indian climate conditions even for plain terrain. Moreover, GPM does not account for any fluctuation in the

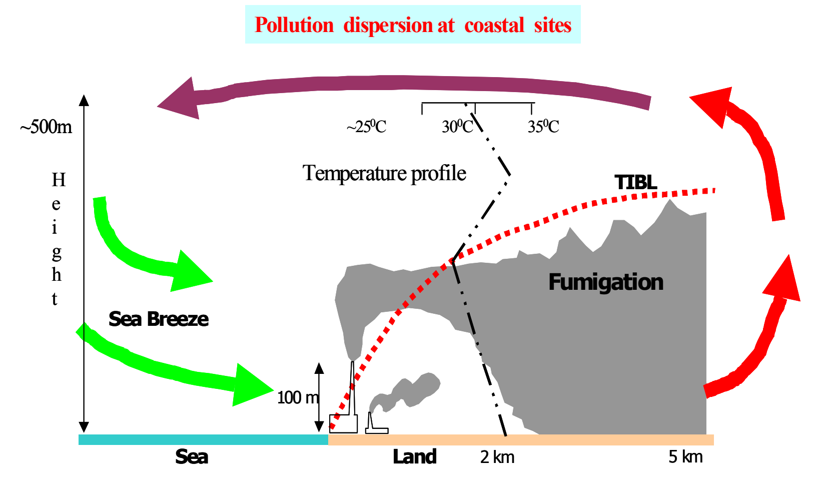
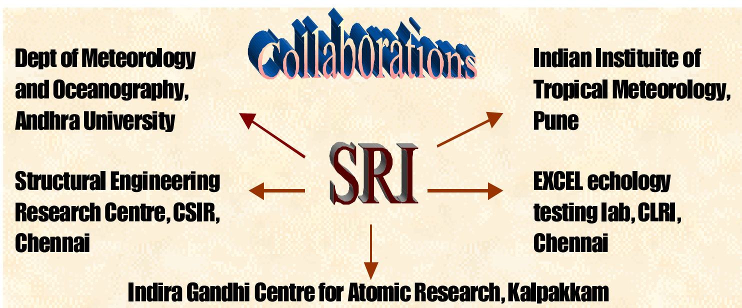


Fig.1 Atmospheric dispersion near a coastal site

environmental dose for the same release rate that may occur purely due to the specific atmospheric condition at a given site say, for example, fumigation at a coastal site. Thus, the model is of limited utility in coastal site or sites near a large water body and in hilly terrain where the ABL gets modified drastically.

In a coastal site, the differential temperature between land and sea introduces a Thermal Internal Boundary Layer (TIBL) with different turbulence below and above the layer, and a convective circulation known as the sea breeze develops which extends up to 50 to 100km inland (Fig.1). Even the vertical extent of the sea breeze is known to vary with time from the onset of the sea breeze. In a hilly terrain, differential heating and cooling of the mountain side and the plains result in valley and drainage winds. A method to incorporate the effects of these in dispersion calculation is therefore imperative.



Box-1

The more detailed meso-scale models are, of course, computationally intensive and would be directly applicable for short term/emergency situations where the GPM does not hold. By back projecting the data from a limited number of meteorological and radiation level observations, these models could also be employed to estimate the source term during an off-normal stack discharge. Besides, by studying the deviation between the predictions of these models and GPM, a hybrid approach for long term assessment can be established where by GPM results with suitable correction factors would be acceptable. Establishment of correction factors for GPM to undertake long term assessment as well as development of numerical models applicable for short term / emergency situations are thrust areas for theoretical work. Validation studies aimed

OVERVIEW OF THE COLLABORATION WORK

Theoretical Studies

- Development of more realistic numerical models specific to coastal site.
- Particulate dispersion model for trajectory / source term estimation for complex terrain.

Experimental Work

- Comprehensive meteorological measurement campaign to generate bench mark data for models validation.
- Simultaneous tracer release experiments for dispersion parameters estimation

Box-2

at comparing the theoretically predicted concentration values with the experimental values obtained during the large scale field tracer release experiment would be of significance for realistic development / application of dispersion models either for routine releases or during emergency situations.

There is thus a need for,

1. Developing mesoscale models to simulate local wind field near coastal/mountainous site,

2. Developing transport and dispersion model for concentration estimation,

3. Theoretical validation studies on the model using actual met measurements,

4. Conducting validation studies using tracer technique under different atmospheric conditions, and

5. Evolving an environmental impact assessment criteria for regulatory applications.

Identifying experts from in-house laboratories and other national research centres and universities (Box-1), SRI embarked upon a broad based research programme under the theme of Coastal Atmospheric Dispersion Studies with a view to meet the above said requirements.

Module-I

Structural Engineering Research Centre, CSIR, Chennai

Investigation on wind characteristics and other site specific parameters at Kalpakkam

Project overview

- Design, fabrication and installation of meteorological towers
- Estimation of site specific wind parameters
- Studies on wind load of structures under extreme wind conditions
- Compilation of the data base for regulatory models

Deliverables

- 50m met. towers designed and installed at three locations
- Site specific surface parameters and wind load analysis results

Module-II

Dept. of Meteorology and Oceanography, Andhra University

Characterization of TIBL along the eastern coast of India

Project overview

- Sea breeze frontal characteristics
- Extensive profile measurements in the boundary layer
- Parameterization of TIBL turbulence
- Validation of mesoscale flow and dispersion model
- Cyclone model validation

Deliverables

- Tethered balloon system and mini SODAR
- Boundary layer data at coastal sites for model validation
- TIBL characteristics at Kalpakkam
- A suitable model for cyclone simulation

Module-III

Central Leather Research Institute, Chennai

Development of tracer release, sampling and analysis technique

Project overview

- Characterization of tracer release, sampling and measurement system
- SF_6 tracer gas chromatographic analyzer
- Standardisation of calibration procedure
- Establishment of tracer release protocol
- Consolidation of data for modelling studies

Deliverables

- Gas Chromatograph analyser for SF_6 tracer gas.
- Manual for calibration and release protocol
- SF_6 concentration distribution data from field experiments

Box-3

An overview of the Collaborative programme

As the study programme involves installation of towers, extensive meteorological measurements, tracer release experiments and theoretical works and model developments, Institutes with expertise in specific fields were identified and the project was split into different modules. The institutes are, Department of Meteorology and Oceanography, Andhra University, Visakhapatnam, Structural Engineering Research Centre (SERC), Chennai, Central Leather Research Institute (CLRI), Chennai, Indian Institute of Tropical Meteorology, Pune and Indira

Gandhi Centre for Atomic Research, Kalpakkam. IGCAR was entrusted with the role of coordinating the research works. The over all study programme is shown in the box-2. Each module with specific tasks and deliverables are listed in box-3.

Highlights of sub-modules and their achievements:

SERC Project

Inauguration of the Instrument control cabin (Oct.'2000)

Boundary layer wind characteristics under coastal terrain conditions, during normal and extreme wind climate are highly important for the safe design of buildings and structural components. With the primary goal of understanding the wind flow in various seasons over Kalpakkam coast, the SERC project had a defined, time bound scope and activities. Project activity began on October 1999 and was completed with in the stipulated period.

One of the 3 towers at Anupuram

Design of the met tower

A 50 m tall guyed tower (square in plan) with hinged base and supported by guys at four levels was designed at SERC Wind Engineering Laboratory. For easy dismantling and portability of the tower, it was designed for shallow foundations for hinged base and guy anchors. The mast is made up of built up sections using angles and MS rods. It has a ladder arrangement in two faces to enable climbing up the tower for fixing sensors. Fabrication and erection were completed at all the three sites. The tensions in guy-wires were maintained by pre-tensioning with tension meters to ensure desired structural performance.

The tower was instrumented with 3cup anemometers and direction sensors at 10m,17m, 29m and 50m respectively to study the wind characteristics. For measurement of wind induced tower

Prof.Ramarao, former Chairman, AERB, inaugurating the instrument cabin of the tower. On the right is the control electronics r and datalogger

STUDY PROGRAMME

Design study of 50m guyed tower; Three towers installed at 0.5km, 5.5km & 16km inland from the coast; Instrumentation involves cup and sonic anemometers, strain gauge and accelerometers and a PC based datalogger and processing unit

vibrations, two tri-axial accelerometers were fixed at 29m and at 50m levels.

Analysis of the data

All the sensors were powered with DC supply. The data acquisition, online scrutiny and archival was done using a data logger and the software was developed at IGCAR. Sampling was done with a rate of 0.016Hz for the normal data collection and at 20Hz for extreme wind data. At Anupuram site, total 361 records of high frequency turbulence data were collected for a period from 8 months and at IGCAR, over 500 records of data for 5 months were collected.

A cyclonic event was recorded at Anupuram tower during the period of measurements. All data

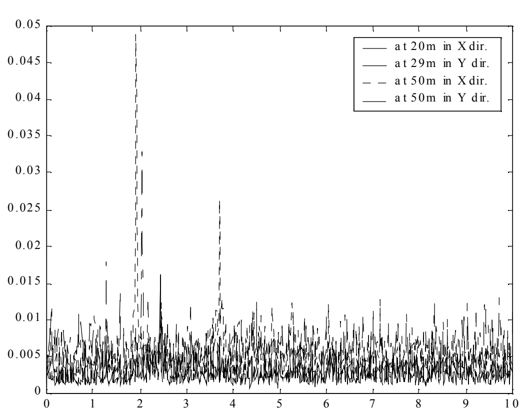


Fig-3 Measured acceleration spectrums on the tower

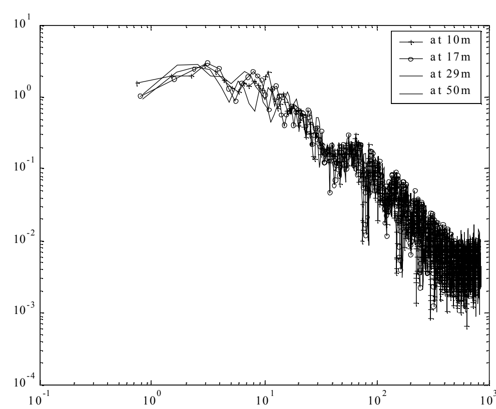


Fig- Turbulence spectrum of normal wind at Anupuram collected during normal wind conditions were found to be stationary and were considered for detailed analysis. Analysis of this data includes determination of wind and terrain characteristics such as the mean wind velocity and its variance with height, power law coefficient, turbulence

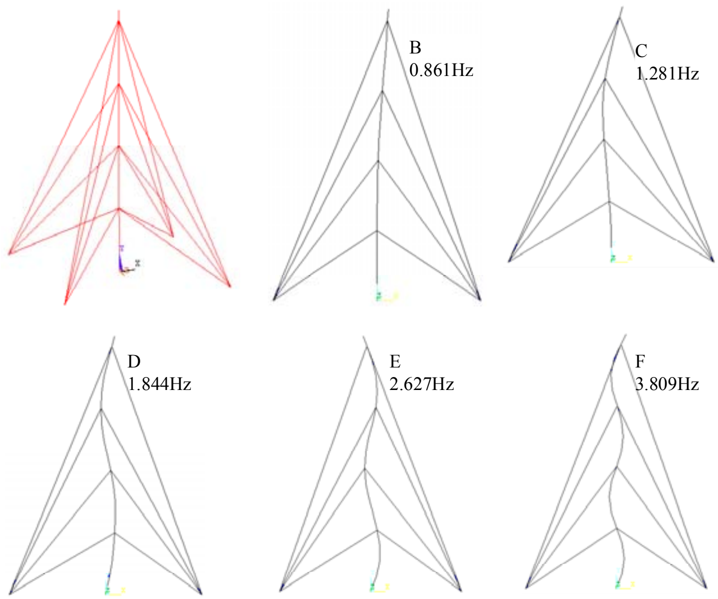


Fig.4

FEM analysis of (a) 3D FEM model, (b) to (f) Mode (bending) shapes for 5 modes

intensity and its variation with height and roughness length.

For dynamic analysis of structures representation of wind against individual frequency components is preferable since this permits relatively easy identification of the frequency component of the load which may create resonance condition of the structure. A turbulence spectrum shows the variation in the energy of the wind at different frequencies. The power spectral density values S (n) corresponding to each frequency (n) for a typical data is shown in Figure 2.

The Cyclone wind turbulence spectrum is also obtained during this period. Based on statistical analysis of the measured acceleration data, maximum, minimum and standard deviation of accelerations were evaluated at 29m and 50m for the X and Y direction. The spectrum of acceleration response measured is highly helpful not only to detect the vibration modes of the structure, but also helps identifying the spurt of gust wind flow, in any built-up area. Fig.-3 shows the acceleration spectrum of a typical record. The various peak frequencies identified are 1.01 Hz, 1.26 Hz, 1.9 Hz, 2.0 Hz, 2.11 Hz, 2.47 Hz, 3.08 Hz, 3.66 Hz respectively in the ascending order. These values are also evaluated using an FEM –Model and some fundamental modes are given in figure-4. It is more than obvioius that a guyed tower can get excited in many modes. This also substantiates that at times there is considerable energy in wind specially in cyclonic wind to excite structures having modes with frequency higher than 1 Hz. Further analysis of the data is being continued.

Andhra University Project

One of the important phenomena recurring at the coastal sites during sea breeze is the development of TIBL. This layer forms with in the usual Atmospheric boundary layer over the land due to the in-land advection of cold marine air. This air suppresses the developing day time mixing layer over land to an intermittent level close to the surface, say about 100m to 200m with in which intense turbulence mixing takes place. Any plume emitted from tall stacks above this level near the coast would intersect the TIBL and fumigate with in this as shown in Fig.1. However no direct observation of this phenomena has been made in India. While analysing environmental radioactive dose data due to Ar-41 plume released from MAPS, it was established

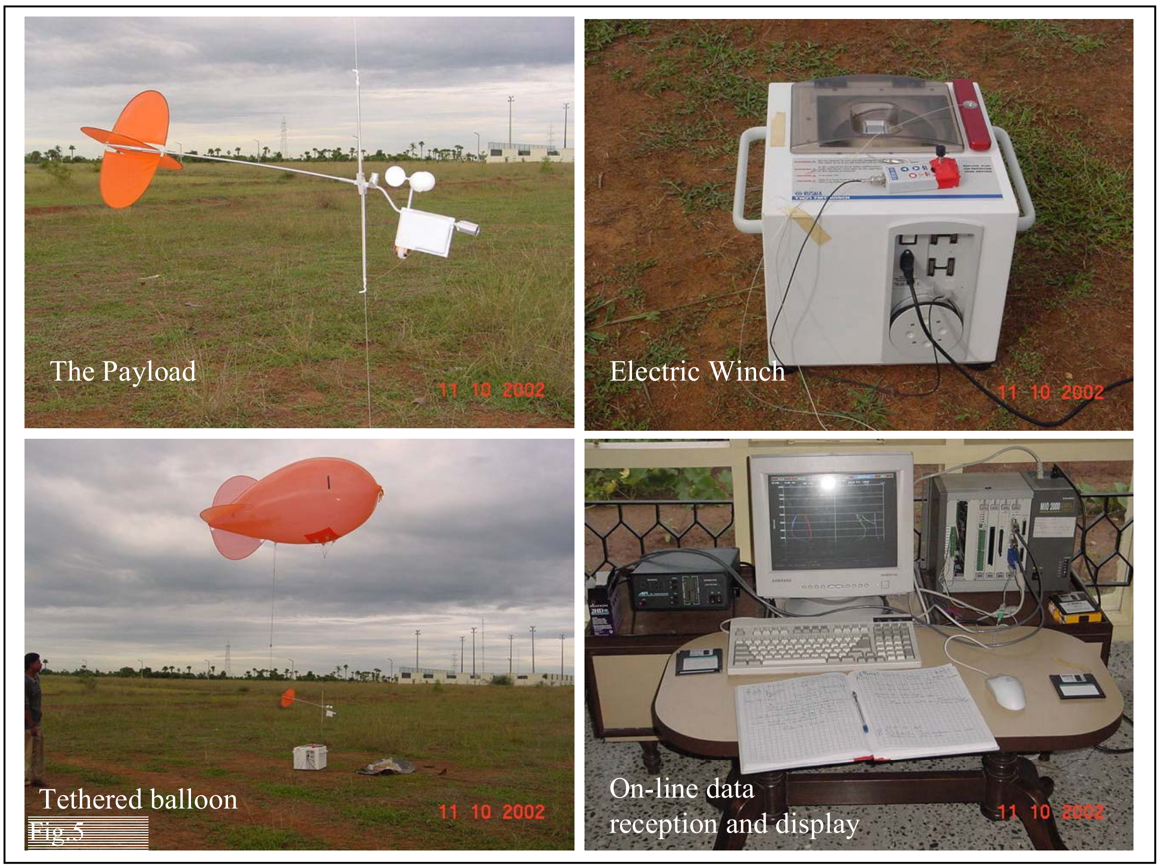


Fig.5

that during sea breeze times the doses were as high as 2.5 times the values that would be computed for the worst condition using GPM. Modelling fumigation by modifying the GPM and as well as by using Monte Carlo particle modelling technique it was shown that the fumigation dose showed enhancement by a similar factor shown in the measurements. The height of the

TIBL being one of the important input, needs to be measured and its spatial and temporal variations are to be studied. In order to measure the vertical profile of temperature, wind and other parameters, a tethered balloon system and a Mini-Sodar were procured under the collaborative projects and field experiments were carried out by Andhra University with the help of experts from Indian Institute of Tropical Meteorology, Pune and Radialogical Impact Assessment Section, IGCAR.

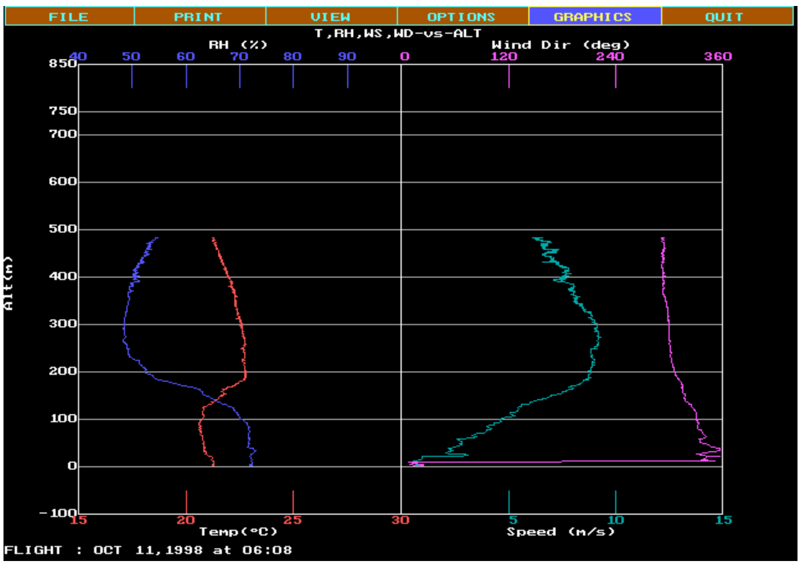


Fig.6. Morning profile of an elevated inversion layer (100m-200m).

A few field campaigns (Coastal Atmospheric Boundary Layer Experiments – CABLE) were carried out at Kalpakkam by using the tethered balloon system and mini-sodar along with conventional tower based equipments. The tethered balloon system consists of an aerodynamically designed balloon, an electric winch for rolling the tether, payloads with embedded transmitters and a receiver and data acquisition system (Fig.5). About 100 flights were made using the balloon covering different seasons and different times of the day and a good number of data have been collected. Analysis of the data revealed realistic boundary layer characteristics like the morning inversion layers, day time mixed layers, the TIBL during sea breeze hours etc (Fig.6). The mini-sodar operates using a phased array acoustic sensor and is a portable system (Fig.7). The system gives data up to 500m on the wind, turbulence and mixing heights. Diurnal variations of the wind profile and mixing height (red patches) are shown in Figure 8. Analysis of the wind data showed the directional shear, low level jets, wide fluctuations in the direction during transition times etc. In the initial phase of the experiments, inter-comparison

Fig.7 Portable mini-sodar system

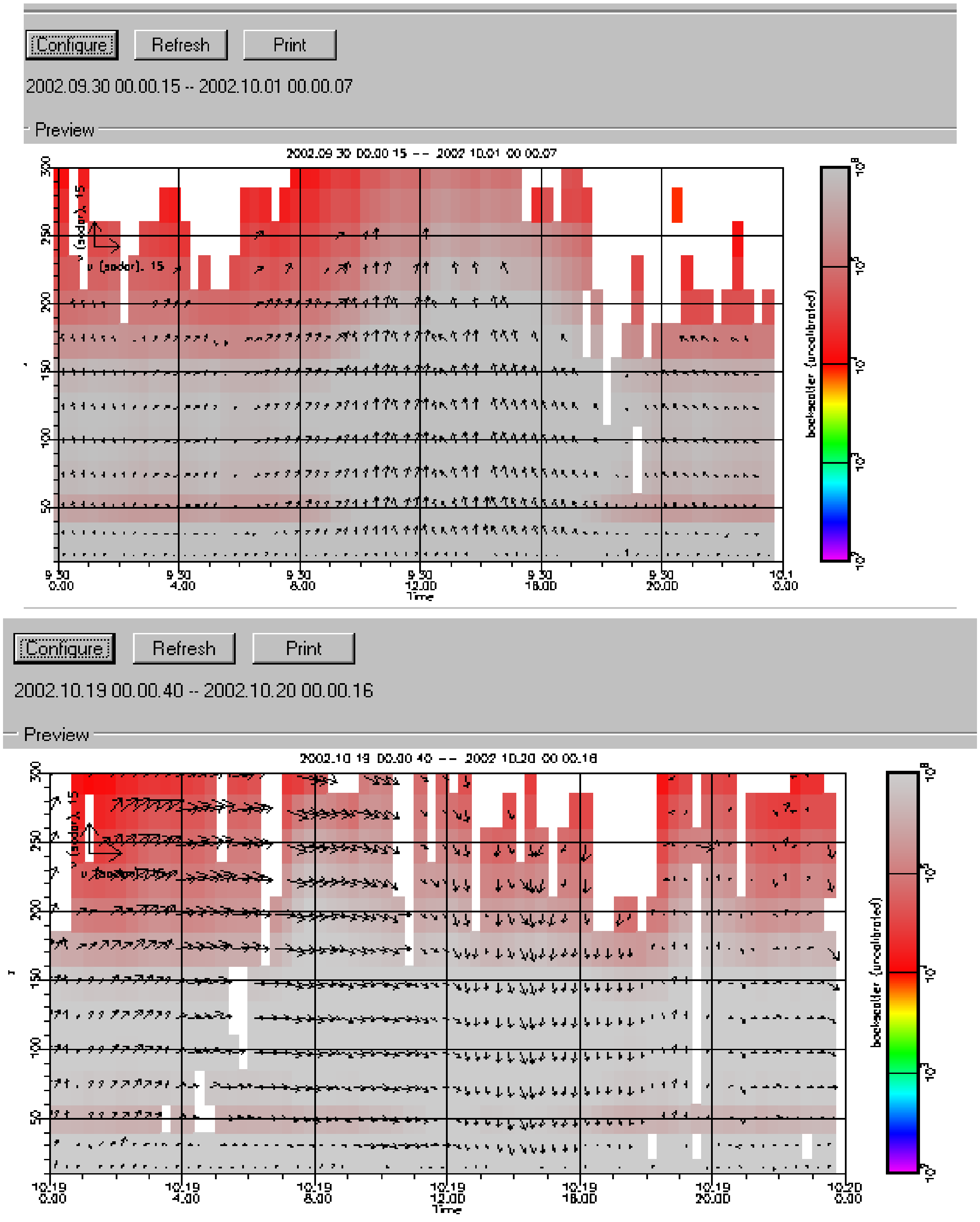


Fig.8 Wind profile for two days . Red patches indicate the mixing layer height

of similar data from all the sensors were made to ensure the reliability of the measurements. During each experiments, attempts to measure specific characteristics features of the site were made and the profiles under sea breeze conditions were extensively collected. A few dozen measurements using balloon balloon system were made during night times as well. The diurnal mixing height shown by the sodar is valid only during morning and evening hours as the height would be beyond the range of sodar during day time conditions.The TIBL formation around 200m is seen during the onset of sea breeze at 14:00 Hrs on both days the Horizontal direction fluctuationcollected collected from mini-sodar for 15 days on these two months reveal that the

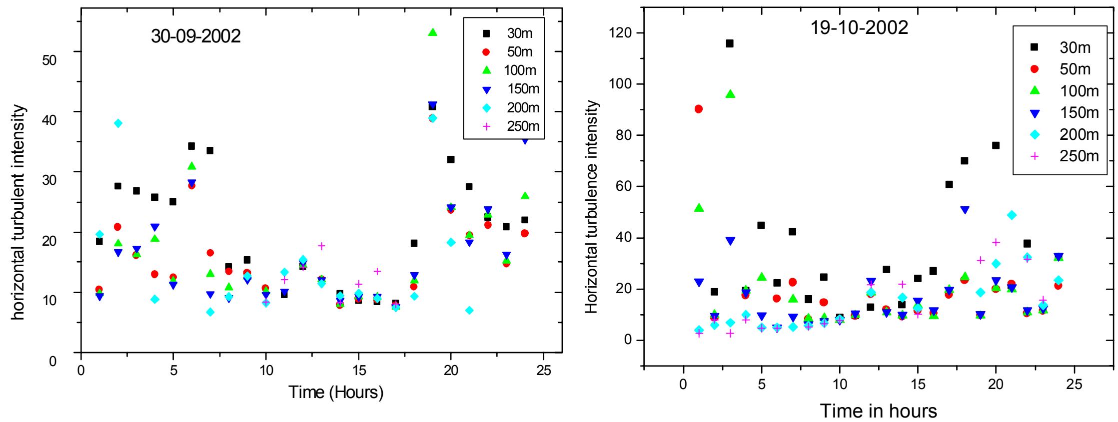


Fig.9. Diurnal variation of directional fluctuation; conditions favourable for plume meandering

fluctuation is large during transition times when the wind speed is very low. The σ_θ is values are shown in Fig.9.

Model developments and validation study

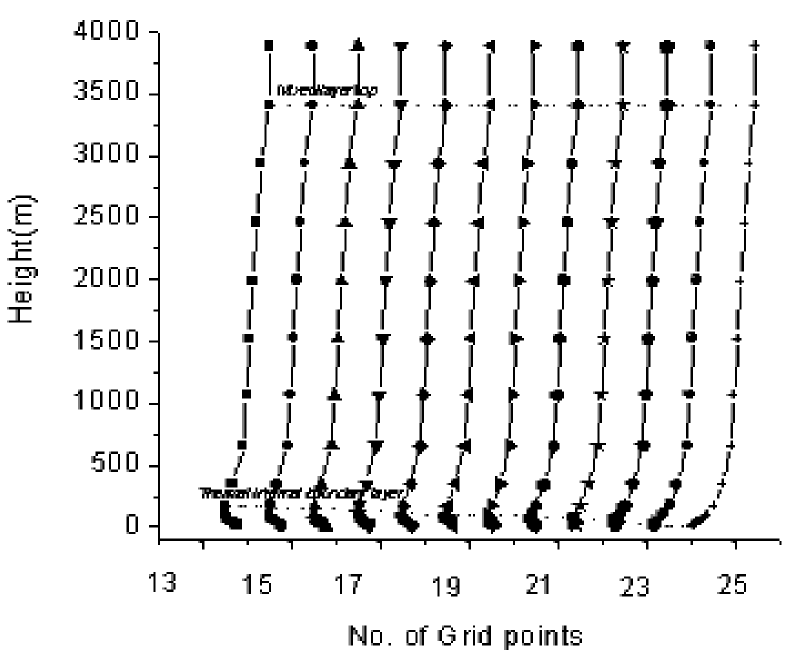


Fig.10. Vertical potential temperature profiles across the coast. Grids 13 to 22 are over land. TIBL height is identified as the low level inversion

Development and validation of of a Numerical model for simulating the meteorological fields over Kalpakkam is one of the objectives of the collaborative programme. A first version of a 3D mesoscale atmospheric model (MAM-I) was developed for simulating the sea breeze and TIBL over Kalpakkam region covering about 100km around the site. The model solves the hydrodynamic equations describing the atmospheric flow with suitable approximations and realistic assumptions for a coastal site. The model can be run in

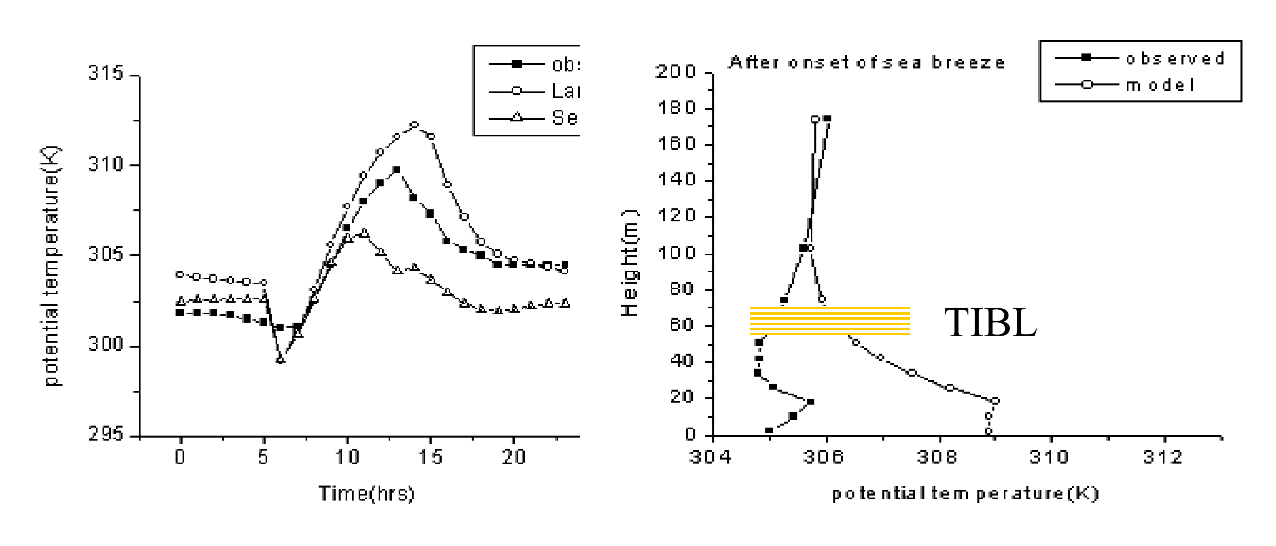


Fig.11 Comparison of model values with measurements.

a PC with Pentium –II processor and takes 40minutes for 24 hrs forecast of the wind field. Comprehensive data gathered during experiments were used for validating MAM-I for different conditions of the atmosphere. The simulated potential temperature profiles during sea breeze time are shown in figure 10 over the land and sea. The lower inversion layer identifiable from this figure corresponds to the TIBL. The simulated TIBL and surface temperature are compared with measurements using tower based and balloon based sensors and are shown in figure 11. The validated model has been used for estimating the site boundary doses for PFBR site at IGCAR. Further improvements on numerical modelling work has been initiated for generic applications including regulatory purposes.

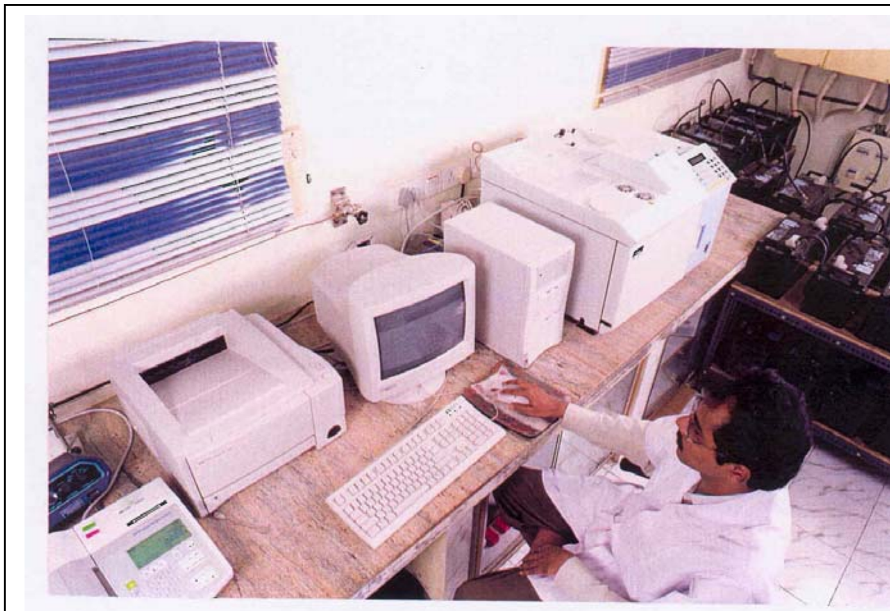
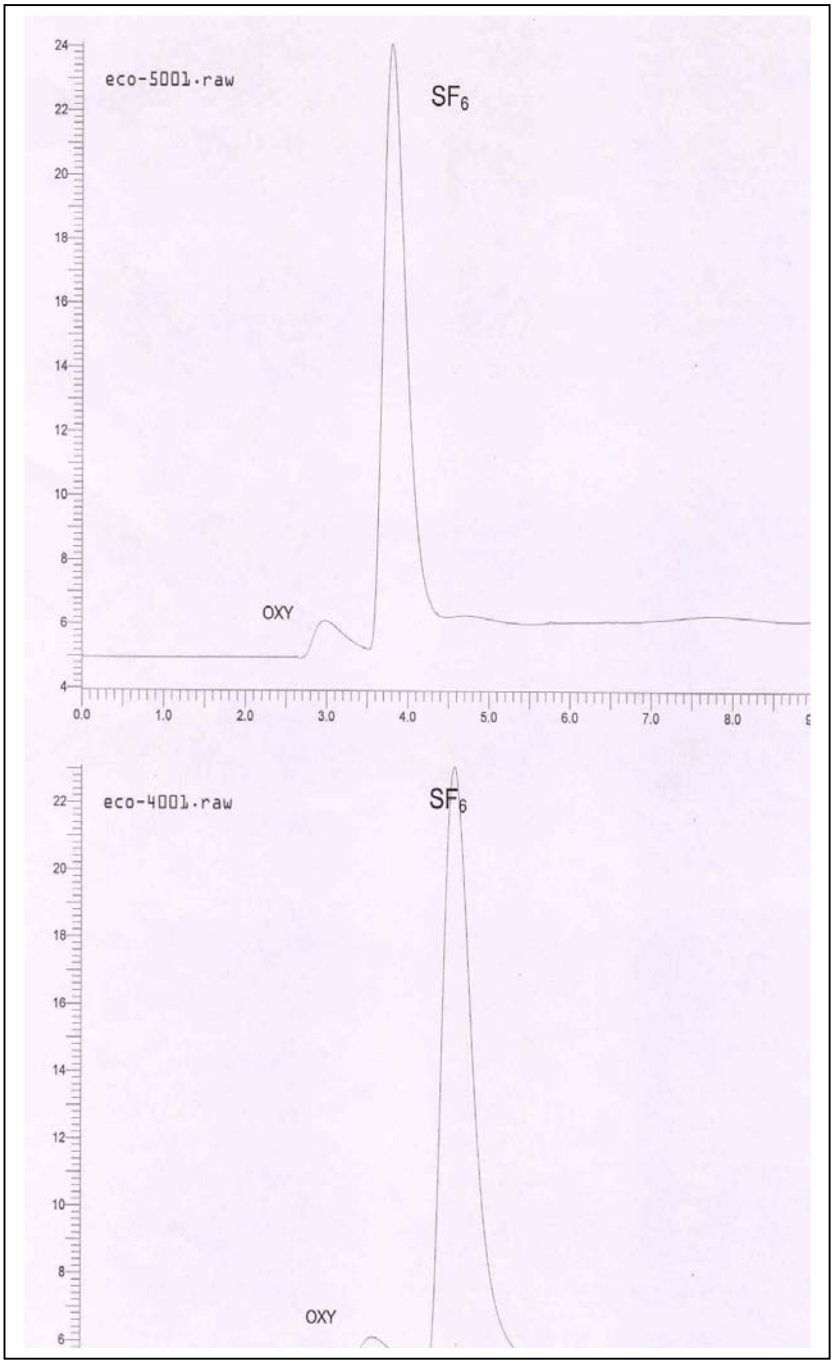


Fig.12 Gas Chromatograph for SF_6 tracer gas detection

CLRI Project

Tracer release experiment

As discussed in the in the beginning, the eddy diffusion parameters used in regulatory models are those obtained from simple experiments conducted in mid-latitude environments in other countries for plain terrain. Similar experiments need to be conducted in Indian tropical climatic conditions and preferably around the application site. A tracer release experiment was envisaged for Kalpakkam coastal terrain which involves release, sampling and analysis of a tracer gas under different atmospheric conditions. SF_6 gas has been found to be used for this purpose due to its non-toxic inert chemical properties and easy detectability using a Gas Chromatograph (GC).



A GC with Electron Capture detector has been procured, installed and calibrated for SF_6 detection. The system shown in Fig12 is installed at CLRI. A minimum detectability of 5ppt level has been achieved. A sample out put of SF_6 detection in the output spectrum is shown in Fig.13.

Benefits from the collaborative effort

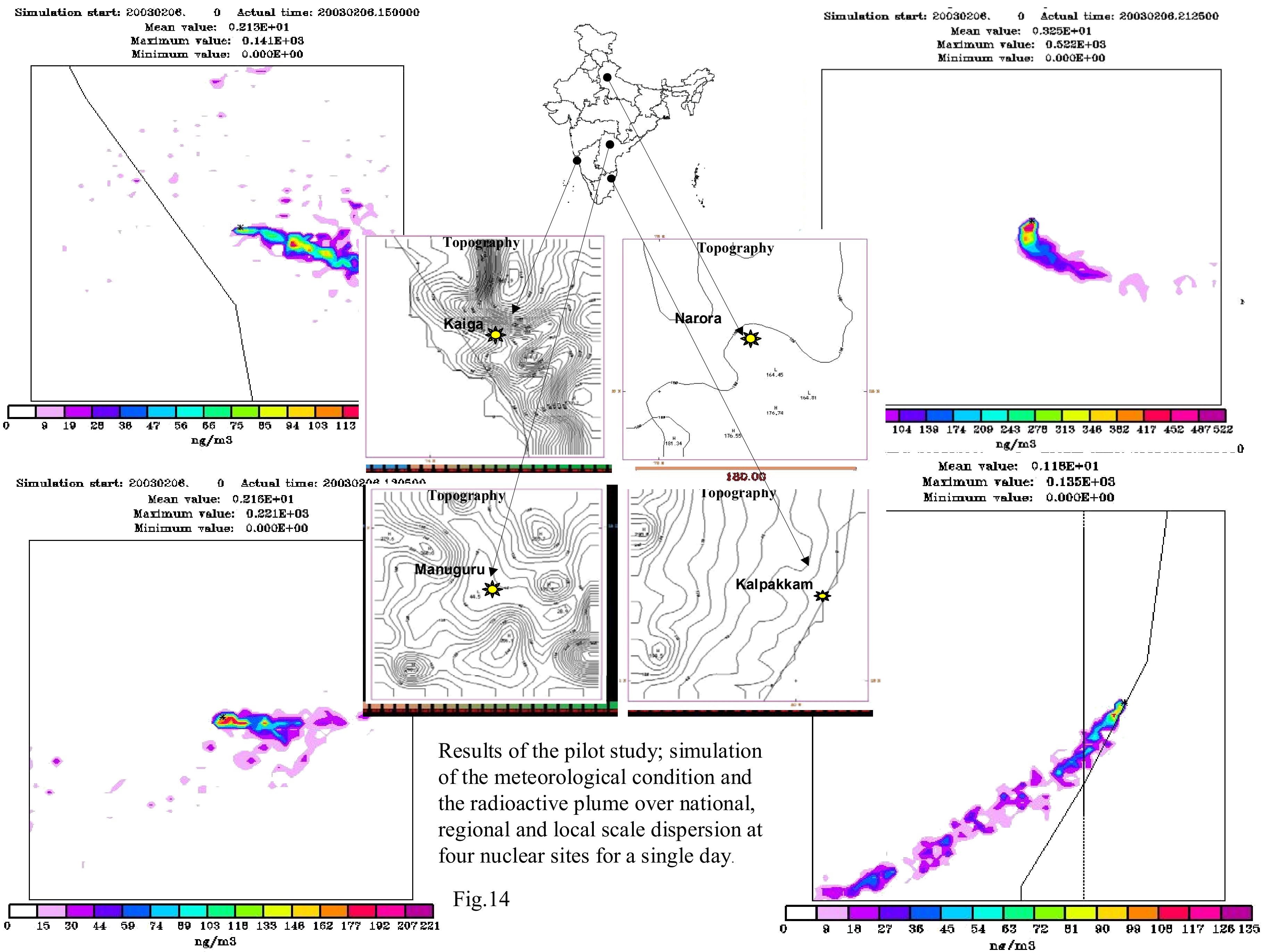
The overall objectives of the collaborative projects have been fulfilled at large and the benefits are highlighted in the Box-4. A good amount of comprehensive site specific data have been gathered and used for model validation study. The towers, instruments and equipments are being used in on-going field experimental programs. The tracer release experiments are being conducted and the dispersion parameters would be derived at the earliest for use in regulatory applications. An important out put of the collaborative programme has been the development of a more generic multi-range dispersion model for all nuclear power stations in India that could be operated from a central control facility which is very useful to forecast and analyse the environmental consequences of an inadvertent release. An example is shown in figure 14. The atmospheric forecast model MM5 developed by Pennsylvania State University and the dispersion model FLEXPART developed by BOKU, Vienna have been implemented for Indian terrain condition and tuned to adopt the national weather data that could be taken as on-line input. The nested modelling system simulates the wind field on a coarse grid mesh covering the subcontinent and the same over a fine mesh around the region of interest. The input data is available from IMD for the larger area of the nation and the site specific data from the nuclear power plant of interest shall have to be made available on line to the central regulatory control facility. The forecast is for the entire nations in a coarse mesh and region of specific interest in a fine grid mesh of 1km resolution. The topography, land-use and nuclear isotopic composition data are almost constant inputs to the model and needs only time of release and release quantity. Efforts are also initiated in source term estimation in case of inadvertent unknown releases based on inverse modelling technique using environmental measurements.

BENEFITS FROM THE PROJECTS

- ***Correction factor for short term GPM for coastal sites***
- ***Meteorological field lab as a facility for future studies***
- ***In-house expertise for conducting field measurements with state of art instruments***
- ***A validated, comprehensive model system for regulatory Recommendation***

Box-4

All these developments are in tune with the international approach being followed in this area of research and the collaborative research programme has led to achieve a high standard in improving the regulatory recommendations and monitoring.



Results of the pilot study; simulation of the meteorological condition and the radioactive plume over national, regional and local scale dispersion at four nuclear sites for a single day.

Fig.14

4. Discussion Meets/Seminars/ Workshops

One of the primary objectives of SRI is to provide a forum for designers, operators, research groups and regulators to come together for exchange of information and expertise. As part of these efforts SRI conducted following programmes in the past four years.

1. Workshop on Gamma Ray Shielding (February 20-26, 1999)

A week long workshop on gamma ray shielding was inaugurated on February 20, 1999 by Dr. R. Chidambaram, Chairman, Department of Atomic Energy along with the foundation stone laying ceremony of Safety Research Institute (SRI) of Atomic Energy Regulatory Board. This workshop was jointly organised by the Kalpakkam chapter of ISRP and IGCAR. The main purpose of the workshop was to teach, train and motivate young and talented professionals to pursue research and to help the experts to keep abreast of latest trends and methods in the area of gamma ray shielding. The workshop covered the subject starting from the fundamentals such as interaction of radiation with matter to the complicated design aspects of reactor shielding including estimation of external gamma dose rates from reactor containment buildings. The participants obtained hands-on training on realistic shielding problems using a system of shielding computer codes. After the training, diskettes containing the computer codes ASFIT, QADCG-GP, NUCHART and XCOM along with the corresponding Graphical User Interfaces codes were distributed to the participants.

About 45 professionals both from within and outside DAE facilities attended the workshop. - From the feed back session, it was learnt that the participants have been greatly benefited and wished to have more of this kind in the future. AERB intends to utilize the services of such trained experts for its peer review purposes.

2. Discussion Meet on Probabilistic Safety Assessment (August 6-7, 1999)

Use of Probabilistic Safety Assessment for regulatory purposes is gaining ground in several countries. SRI in collaboration with The Institution of Engineers (I), Kalpakkam Local Chapter and IGCAR organized a two-day discussion meet on Probabilistic Safety Assessment. The main focus of the meet was on the state of the art and international scenario of PSA practiced for nuclear power plants. The meet also discussed PSA activities at BARC, NPCIL and at IGCAR.

About 50 participants mainly from major DAE units and NPCIL participated in the discussion meet. The meet produced an excellent forum for exchange of ideas and led to identification of areas for further work and mechanism to integrate the efforts of different groups.

3. Discussion Meet on Feedback Experiences on Safety Related Unusual Occurrences and Adherence to Technical Specifications for Nuclear Plants (December 9-10, 1999)

A two day discussion meet on Feedback Experiences on Safety Related Unusual Occurrences and Adherence to Technical Specifications for Nuclear Plants was jointly organized by SRI-AERB and Indian Nuclear Society (Kalpakkam Branch). This meet, first of its kind, was organized to consolidate the feedback experience on safety related unusual occurrences during the operation of nuclear plants over the past few decades. A need for certain standardization in the method and interpretation and reporting of safety related unusual occurrences (SROURs) and also simplification in the technical specifications has often been expressed. The discussion meet was to provide a forum to address this important need. The meeting deliberated on the following aspects:

- Feed back experience on SROURs had focused on repeated incidents of interest in nuclear plants
- Incidents caused due to human errors and difficulties adhering to reporting criteria
- Importance of analyzing trends and precursor leading to major events
- Violation of technical specification: causes and difficulties faced in observing certain surveillance requirements

Participants numbering about 65 were invited mainly from all the Nuclear Power stations in the level of station superintendents/ technical superintendents or any senior personnel of the plant. AERB personnel presented their impressions and feedback on the review and reporting methodology. Following this discussion meet AERB and NPPs took efforts to simplify Technical Specifications for the plants without compromise on the safety requirements.

4. Workshop on Monte Carlo: Radiation Transport (February 7-18, 2000)

A two Weeks workshop on Monte Carlo methods in Radiation Transport organsied by SRI was intended to introduce the researchers the basic ideas of the Monte Carlo Method as well as to provide training in Monte Carlo codes with special emphasis on the use of MCNP Code. Practical exercises designed in this training had enabled the participants to gain a working knowledge of the subject, to write and develop/modify computer programs on their own to suit their needs. The morning sessions were fully dedicated to series of lectures on theoretical aspects involved in Monte Carlo methods and the hands on training with thought provoking/demonstrative examples covering practical applications of radiation transport problems in the after noon sessions. To aid the users in the preparation input, a graphical user interface GUI2MCNP, developed at SRI was introduced and the same was distributed to all the participants for subsequent use.

About 30 researchers drawn from various national institutes attended this training program. An unique feature of this workshop was the forum it provided for the specialists to discuss the current trends of research and methods being developed in this area. This enabled the faculty members of the workshop to keep abreast of information on latest trends.

5. Discussion Meeting on Fire Hazard Analysis and Modeling (August 28-29,2000)

SRI-AERB in association with the Indian Institute of Chemical Engineers, Kalpakkam Regional Centre organized a two day discussion meeting on Fire Hazard Analysis and Modeling at Kalpakkam during August 28-29,2000. Fire Hazard analysis is a key issue affecting the safety of nuclear installations. The objectives of the meeting were

- To identify the strengths and weaknesses of the prevailing system
- To forge a path for future coordinated work amongst units of DAE, NPCIL and AERB
- To establish Fire Safety Analysis tool as part of the National Repository of Computer Codes

A number of specialists working on Fire Safety and Fire Protection Engineers of operating power plants covered wide range of topics. The participants got benefited from the vast experience available in the country on details of modeling and the hazard potentials.

6. ESL Professional Meet-2001 (January 23-24, 2001)

SRI-AERB and Association for Waste Management and Remediation of Environment (AWARE) organized a meeting at Kalpakkam for the professionals of ESLs from all DAE facilities during January 23-24, 2001. Since its very inception, the DAE has laid a strong emphasis on environmental surveillance program. It has been the practice at every site to commission an Environmental Survey Laboratory (ESL) well before the commencement of operations of the plant.

The systematic studies carried out by the ESLs have enabled the establishment of baseline environmental data, transfer coefficients for radionuclides in the environmental matrices, identification of indicator organism and assessment of atmospheric dispersion models. The ESL meet provided an excellent forum for dissemination of knowledge and experience gained so far in the important areas of environmental monitoring. Officer-in-charge from each of the ESLs made a presentation of their monitoring program highlighting the site specific aspects and important findings from their R&D work.. The main focus of this meet was to bring to the attention of the delegates about the technological development in environmental monitoring, Environment surveillance practices in and around fuel mining areas, disposal grounds, reprocessing facilities and nuclear power stations of India.

7. Discussion Meet on Computer Based Safety Systems in Nuclear Power Plants (November 28-29, 2001)

With the increased use of computers in reactor instrumentation and control systems, the safety concerns have increased many folds. The major problem associated with the safety of computerbased system in a nuclear power plant is the non-availability of qualified safety assessment of associated hardware and software. The standard principles of redundancy and diversity practiced in conventional systems are not applicable in software controlled systems. It is in this context SRI- AERB, Indian Nuclear Society (Kalpakkam Branch) and IGCAR, Kalpakkam organised a two-day discussion meet on Computer Based Safety System (CBSS) in Nuclear Power Plants during November 28-29, 2001 at Kalpakkam. The discussion meet focused on design experiences, diagnostic skills, surveillance and also on formal methods for software verification, validation and qualification process. The meet also covered several other pertinent topics such as the configuration management or back fitting of real time computer systems, operation and maintenance experiences in C&I systems of commercial power plants, research reactors, light combat air craft, satellite launch vehicles etc.

The discussion meet was attended by over 60 participants from all major DAE units and non-DAE units such as ISRO, ADA-LCA and IIT Bombay. There were 14 invited lectures by experts covering various aspects of design, feedback, licensing and retrofitting. The panel discussion addressed various issues such as regulatory aspects, communication methodology, standardization of programming language, failure analysis of hardware and software etc.

8. Discussion Meeting on Inter-Institutional Collaborative Research (January 30, 2003)

DAE through BRNS and AERB promotes activities of research in nuclear science, engineering and safety in universities, academic institutions and national laboratories. In addition DAE units like BARC, IGCAR and NPCIL interact with universities and the institutes to avail of their expertise. Many of the research contracts awarded to institutions are related to nuclear or radiological safety. SRI organized a discussion meeting to review and share the experience accumulated over the years in the award of such research contracts. The participants included scientists from BARC, DAE-BRNS, IGCAR, IITM etc. The detailed presentations on the progress of the coordinated research activities carried out by BARC in the area of thermal ecology and by SRI in the area of atmospheric dispersion studies clearly brought out the enormous benefits resulting from well planned coordinated research programmes. They have well defined and time bound research goals, are cost effective and widen the knowledge base.

5. COMPUTER CODE DEPOSITORY

Introduction

One of the important activities SRI has taken up is the commissioning of a national repository of computer codes pertaining to safety analysis and hazard evaluation. As part of this activity SRI is carrying out the following tasks.

- Identification of resource personnel and codes within the country.
- Development of standard benchmark data for testing the codes.
- Development of graphical user interfaces for promoting faster and easy way of learning.
- Establishing contacts with outside agencies such as NEA data bank and IAEA etc for information interchange.
- Periodic training of personnel in the use of computer codes.
- Users forum – Creation of WEBSITE.
- SRI had also organized a few workshops to train the participants in handling certain codes. Copies of the codes have also been distributed to the participants.

CURRENTLY AVAILABLE SAFETY ANALYSIS CODES	
REACTOR PHYSICS CODES	Shielding and Criticality Codes TORT, WOTRAN, ANISN, ASFIT, MORSE, KENO, MCNP Detector Response construction and Unfolding codes MARTHA, OSR, EGS4, FORIST, DUST, GAUSSV, SANDII, SAMPO
ENVIRONMENTAL ASSESSMENT CODES	Atmospheric Models MAM-1, MM5 Dispersion Codes SPEEDI
PROBABILITY SAFETY ASSESSMENT	RiskSpectrum, PSA PACK
ACCIDENT ANALYSIS CODES	Design cross checking DB3 Safety Engineering Codes SOFIRE, REXCO, VENUS, NACOM
RS – GIS SOFTWARE PACKAGE USED IN EIA STUDIES	ARCINFO 8.1, ArcView 8.1 ERDAS Imagine Professional 8.4 ERDAS Imagine Virtual Digital Image Processing Software-ENVI Ver 3.4
GRAPHICAL USER INTERFACES	GUI2MCNP, GUI2KENO, GUI2QAD-3D, View-CXS

6. SRI Personnel Profile

Shri S. K. Chande	**Director** **Scientific Officer (OS)**

Name	**Designation**	**Area of work**
Dr. K.V.Subbaiah	Head, Radiation Safety Analysis Section Scientific Officer (G) (Ph.D Guide in Madras University)	Radiation transport and shielding computations with S_N and Monte Carlo methods, Accelerator Safety, Radiation Dosimetry.
Dr. P.Sasidhar	Scientific Officer (F) (Ph.D Guide in Anna University and Madras University)	Safety assessment of radioactive waste disposal facilities.
Shri G.Janakiraman	Scientific Officer (F)	Radiological safety in medical and industrial applications of radiation.
Shri C. Senthil Kumar	Scientific Officer (E)	Probabilistic Safety Assessment
Shri C. Gurumoorthy	Scientific Officer (C)	Safety assessment of radioactive waste disposal facilities
Shri C.Sunil Sunny	Scientific Officer (C)	Health Physics and Radiation Shielding
Shri Yoshihiro Hirao	Visiting Scientist from National Maritime Research Institute, Japan	Radiation Shielding
Shri Manoj Kukreja	Senior Research Fellow	Remote Sensing Applications

Deputation abroad:

1. Dr. K.V. Subbaiah: Participated in the First Asian and Oceanic Congress on Radiation Protection (AOCRP-1), Seoul, Korea during October 20-24, 2002.

2. Dr. P. Sasidhar: Nominated to IAEA-CRP on "Application of Safety Assessment Methodologies (ASAM)" for near-surface disposal facilities and has participated in two RCMs held at IAEA, Vienna in 2002 and 2003.

3. C. Senthil Kumar: Participated in the 11th International Conference on Nuclear Engineering at Tokyo, Japan during April 10-23, 2003.

7. Project Profile

The project for Safety Research Institute was sanctioned in October 1998 as part of IX Five Year plan scheme under AERB. The foundation stone for the Institute was laid by Dr.R. Chidamabaram, Chairman AEC on February, 20, 1999.

Salient Features of the Project

Financial outlay for the project	: Rs. 9.98 crores
SRI Building at Kalpakkam Campus Floor space	: 2200 Sq.m
Proposed Manpower (I Phase)	: 35
SRI Guest House at Anupuram Floor Space	: 2800 Sq.m (60 beds)

Facility for Researchers
Computer code depository, RS and GIS facility, Environmental Chemistry Laboratory, Radiological Instrumentation, Access to R&D laboratories in IGCAR and BARC facilities and Library and Information Services at IGCAR.

Project Leaders

Chairman AERB

Prof. P. Rama Rao	From beginning of SRI Project to November 1999
Prof. S.P. Sukhatme	January 2000 onwards

Director SRI

Dr. Placid Rodriguez	November 1998 to October 2000
Shri A.R. Sundararajan	July 1999 to October 2000 (Joint Director)
	October 2000 to February 2003
Shri S.K. Chande	March 2003 onwards

8. PUBLICATIONS BY SRI STAFF MEMBERS

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