Shri A.R. Sundararajan, Prof. Rama Rao, Dr. R. Chidambaram and Dr. P. Rodriguez during Laying of Foundation Stone of SRI on February 20, 1999.

SRI Guest House

Prof.Sukhatme, Shri A.R. Sundararajan and Shri S.K. Chande during SRI Guest House Inauguration on January 30, 2003

SRI

Highlights 2009

Safety Research Institute

Atomic Energy Regulatory Board, Government of India

Kalpakkam 603 102

SRI Highlights

February 2009

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FOREWORD

S.K. Sharma
Chairman, AERB

AERB has the mandate to ensure that use of ionizing radiations and nuclear energy in India does not cause undue risk or harm to the workers, members of the public and the environment. AERB carries out this responsibility through a detailed safety review before issuing licence for siting, construction and operation of nuclear and radiation facilities and thereafter through regulatory oversight during their operation. This is done using national and international safety documents and operating experience. This requires state-of-art expertise that needs to be supported by relevant research and development activities. Although the services of technical support organizations like BARC and IGCAR are available to AERB, it was considered appropriate to have an in-house research unit also for conducting regulatory research that would be beneficial in the long term. The Safety Research Institute (SRI) at Kalpakkam was established in February, 1999 with this objective.

In its ten years of existence SRI has taken up work related to radiation shielding and dosimetry, probabilistic safety assessment for nuclear power plants, structural and seismic analysis, fire modeling, thermal hydraulics and 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. Another significant activity has been to create a depository of safety related computer codes. A large number of validated codes have been installed and user friendly interfaces have been developed. SRI has also been conducting seminars, workshops and discussion meetings for designers, research groups and regulators to come together for formulation of action programmes aimed at resolving safety related issues. In recent times, work on reactor physics studies for pressurized light water reactors has also been taken up at SRI.

SRI has made good progress within the short span of ten years and this booklet 'SRI Highlights' provides a summary of the work done during this period. I hope that in the years to come this Institute will grow in strength and stature and will be able to give substantial support to AERB by way of conduct of high level regulatory research.

CONTENTS

1. About SRI

Atomic Energy Regulatory Board (AERB) has been mandated to review, enforce standards and authorize from safety angle siting, design, construction, commissioning, operation and decommissioning of nuclear and radiation facilities. To carry out this function effectively, AERB is required to equip itself with sound technical infrastructure and to build wide knowledge base. AERB's technical independence from the operating units would be best achieved by AERB's own strong research capability in safety related fields. Prof. P.Rama Rao, the then Chairman AERB, took the initiative and decided to set up the Safety Research Institute of AERB at IGCAR campus at Kalpakkam so that with the readily available infrastructure the activities of the institute could pick momentum in a short time.

The foundation stone for the Institute was laid on February 20, 1999 by Dr.R.Chidambaram, the then Chairman, AEC during the IXth Five Year Plan period with Dr.P.Rodriguez as the first director of the Institute. It moved to its own building in 2003 which has an office area of 2300 square metres. A full fledged SRI Guesthouse (that can accommodate about sixty guests) has been established in Anupuram Township. The Guesthouse has a Seminar Hall, a Conference Room and a Multimedia Room which facilitate organizing workshops/seminars/theme meetings, etc.

Research areas for SRI were chosen keeping in view their importance to safety assessment carried out by AERB and also to complement the ongoing research and development work done in DAE units. Availability of guidance from senior researchers of IGCAR was another guiding element in the selection of work areas to be pursued at SRI.

Directors of SRI since inception:

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 to July 2005

Shri S. E. Kannan
August 2005 onwards

2. AREAS OF RESEARCH

The primary objective of SRI is to build a unique research base and to develop expertise in areas that are pertinent to safety functions of AERB in a way complementary to the ongoing R&D activities of the DAE units. 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 plants and fuel cycle facilities;
- to generate/collect data needed for the 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.

Broad research areas of SRI and some of the work that has been done or is being done are given below:

NUCLEAR SAFETY STUDIES

Reactor Physics and Radiological Safety

- Development of computational models and expertise in safety analysis of Light Water Reactors.
  Light Water Reactors are expected to play a significant role in nuclear power generation in India in near future. There is a need to develop computational models, for their steady state and transient analyses. This will facilitate development of expertise needed for safety analysis of the reactors to aid AERB in regulatory decision making.
- Criticality safety studies of fuel storage and reprocessing facilities.
  To ensure safety and to validate the design of fuel storage and reprocessing facilities by AERB, these studies are being taken up.
- Validation of shielding codes through experimental studies.
  Computer codes are used for design of radiation shielding in nuclear plants and facilities. To validate such codes, experimental studies are being taken up.
- Source term estimation for and safety assessment of medical accelerators.
  A large number of particle accelerators (proton cyclotrons) to produce tracers for diagnostic applications and LINACs for cancer therapy purposes are being increasingly used in India. This work is initiated towards source term estimation, dosimetry and safety assessment of such accelerators.
- Safety code depository: SRI is organizing a depository of all safety related computer codes often used by the facility designer and safety analysts.

Probabilistic safety analysis and reliability studies:

PSA provides important safety insights in the regulatory decision making process - referred to as risk informed regulation. In order to develop expertise in this area, work has been initiated to carry out system reliability studies and PSA of nuclear plants and facilities; to develop PSA methodologies for regulatory safety assessment in fuel cycle facilities and for external events such as fire or earthquake.

Structural and Seismic Studies:

- Studies on development of safety assessment methodologies for NPP components
  This work is taken up towards qualification of various safety related systems and components of a NPP that are not covered by established codes/standards.
- Studies on estimation of component-specific

reliability. These studies are taken up to generate data related to component-specific reliability/ probability of failure that are needed for PSA studies of the plant/facility.

Fire and Thermal Hydraulics Studies

- Development and validation of a fire modeling tool for fire hazard assessment of nuclear fuel cycle facilities including nuclear power plants.
  As of now, a well validated computational model is not available for hazard assessment of nuclear plants and fuel cycle facilities. Hence the work on identification/ development of a fire modeling tool and its validation based on experimentalstudies has been taken up

- Accident analysis of Light Water Reactor systems including hydrogen transport and distribution.
  There is a need to get an in-depth understanding of the phenomena associated with the Light Water Reactor accident scenarios, in order to develop suitable computational models. Initial plans are to take up work related to hydrogen transport and distribution.

ENVIRONMENTAL SAFETY STUDIES:

SRI carries out R&D activities in the following areas:

- State-of-the-art (RS-GIS) application in: Site-Selection and EIA of NPPs; Environmental assessment of uranium mining and development of geospatial based Decision Support Systems in Disaster Management.

- Studies on safety assessment of long-term storage of radioactive wastes.

- Studies on hydrogeological modelling towards development of Site-Specific Aquifer Models.

- Formulation of approaches to model transport of radionuclides in the saturated and unsaturated zone in the ground environment including colloidal assisted transport of radionuclides in the ground environment.

- Development of geochemical models to understand the origin of minerals and role of groundwater.

3. CURRENT RESEARCH ACTIVITIES

3.1 NUCLEAR SAFETY STUDIES

3. 1 (a) REACTOR PHYSICS AND RADIOLOGICAL SAFETY STUDIES

INTRODUCTION

An important task of AERB is to perform the safety review, enforce the relevant standards and authorize operation of nuclear power stations, fuel cycle facilities and other plants involving radiation producing machines or processes. On the coming years, substantial growth is planned due to setting up reactors of new types like VVER, FBR and AHWR as well as new types of particle accelerators and LINACS.

To carry out the effective regulation of these new systems AERB needs to equip itself with the knowledge base and the technical expertise to evaluate the safety of these systems. In this context SRI has taken up research in topics of reactor physics and radiological safety that is of relevance to regulatory decision making. This research and development work involves procurement of relevant computer codes, their use for safety related studies and in some cases the development of user friendly graphical interfaces for the efficient utilization of the codes.

The current research activities include the following:

Reactor Physics Studies

- LWR physics studies with collision probability and diffusion theory codes;
- VVER-1000 LEU and MOX assembly computational benchmark analysis;
- LWR physics studies with Monte Carlo codes;
- External neutron source calculations for fast reactor startup;
- Studies on criticality safety of stacked fast reactor fuel sub assemblies

Radiological Safety Studies

- Experimental verification of safety factor used in reactor shielding design.
- Photo neutron flux and dose rate estimations for ThO_2 bundle immersed in D_2O.
- Estimation of Nb-94 activity in irradiated Zr-Nb alloys;
- New algorithm for design of beam flattening filter for high energy LINAC;
- Estimation of neutron dose contamination in LINACs.

Development of safety related computer codes and user-friendly graphical interfaces

REACTOR PHYSICS STUDIES

Light Water Reactor Studies

In view of the number of Light Water Reactors (LWRs) to be set up in the country, there is a need to develop the capability in AERB for regulatory review of LWRs. Accordingly work was initiated towards LWR physics studies at SRI. To start with, the lattice burn up computer code EXCEL, developed at BARC, with the 172 group IAEAGX cross section library in WIMS-D format was installed and tested at SRI. The code was used to analyse the VVER core of Kudunkulam Nuclear Power Plant (KKNPP) and the database of few group homogenized cross sections was obtained for all the different fuel assemblies at various uranium enrichments, control absorber configurations, burn ups, fuel temperatures, coolant temperatures, coolant densities and linear heat ratings(standard Xe, Sm loads).

Typical results obtained from EXCEL code, for

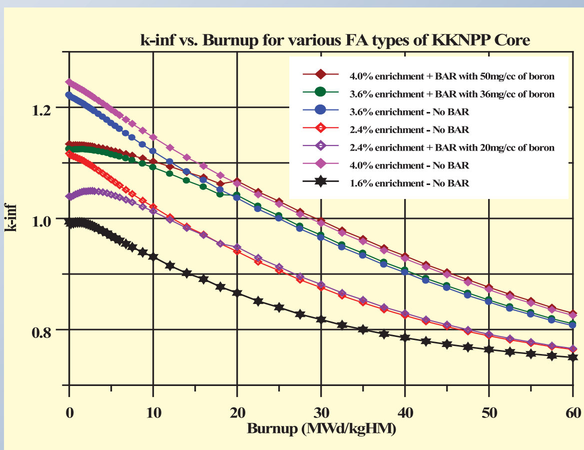

Fig.1: k-inf vs. burn up for various FAs of KKNPP core

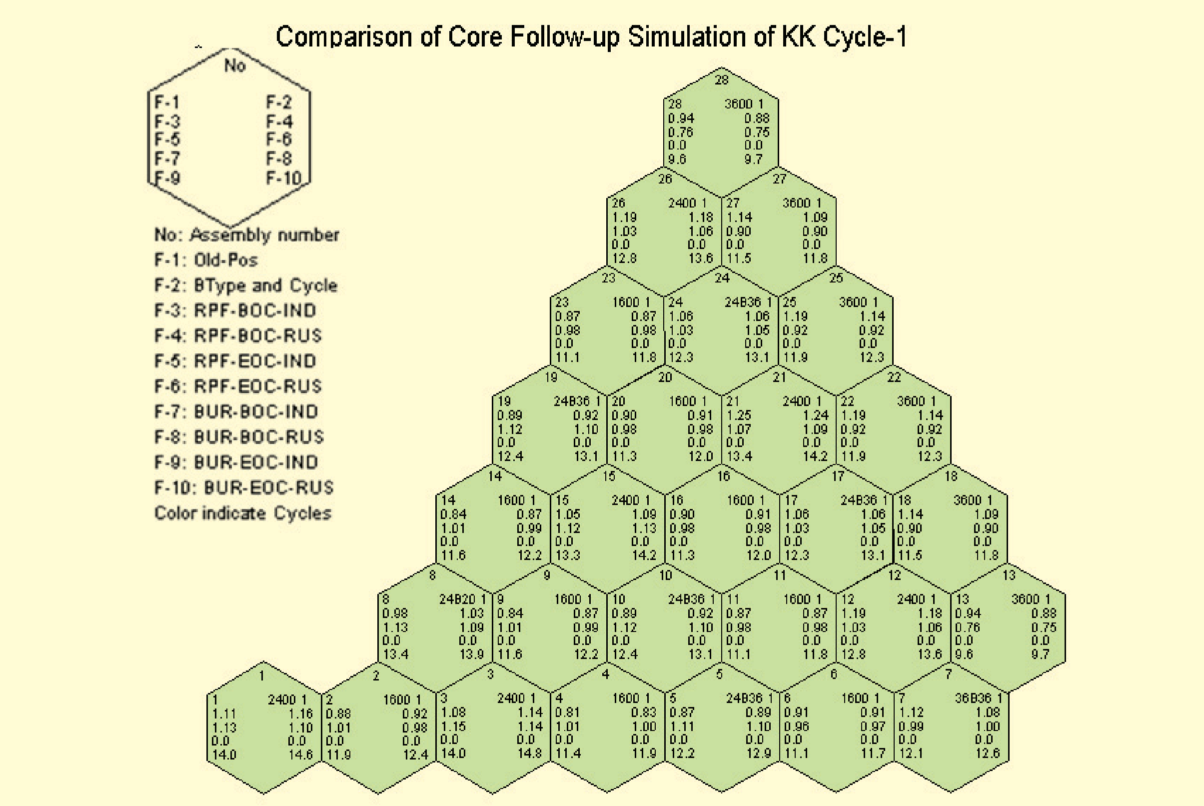

Fig.2A: Comparative study of KKNPP 1/6 symmetric fresh core

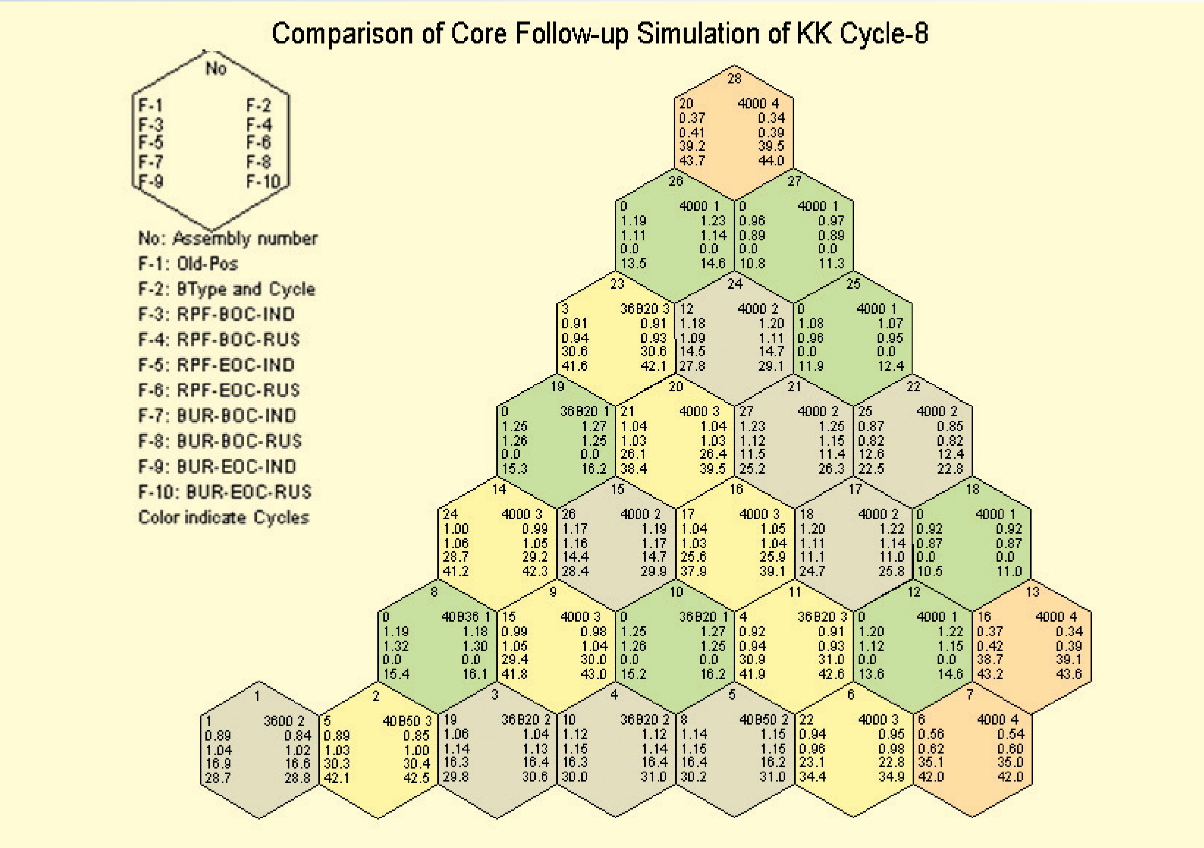

Fig.2B: Comparative study of KKNPP 1/6 symmetric core for 8th cycle results. Different colors indicate the number of cycles faced by the FAs.

k-inf as function of burn-up for various fuel assembly (FA) types of KKNPP reactor with and without burnable absorber rods (BARs) are presented in Fig.1. In general, it is observed that for all the FA types without BAR, *k-inf* decreases as a smooth function of burn-up.

In FA type with BAR there is some non-monotonic trend. The small kink near burn up of 20 MWD/kg is due to the theoretical modeling of replacement of BARs with water at the end of its first fuel cycle. The independent validation of the EXCEL was done by taking up the analysis of a series of VVER-TIC lattice benchmarks. A hexagonal geometry whole core simulator code based on 3D neutron diffusion theory, TRIHEX-FA (developed at BARC) to solve few group neutron diffusion theory in hexagonal geometry was also installed and tested at SRI. The code was successfully used to estimate reactivity of critical soluble boron, reactivity coefficients and burn up reactivity loss for cycles 1 to 8 of VVER cores of KKNPP. The radial FA power distribution in $1/6^{th}$ symmetric core of KKNPP was obtained using TRIHEX-FA code at the beginning of fuel cycle (BOC) and at the end of fuel cycle (EOC). A graphical user interface was developed to display and inter-compare the TRIHEX-FA results. Using this interface, results are presented in terms of radial power factors (RPF: ratio of FA assembly power to the average assembly power) and the burn up attained (BUR). Fig.2A and 2B show the 1st and 8th fuel cycles respectively, the comparative results and the caption "IND" stands for present Indian values and "RUS" stands for corresponding reported Russian values in these figures. All the present results are in good agreement with the Russian values and are within the acceptable limits. Detailed results of 2D and 3D peaking factors, reactivity coefficients of coolant density, coolant and fuel temperatures, power load and boron, kinetics parameters - effective delayed neutron fraction 'β', prompt neutron mean lifetime 'l' as a function of burnup for eight fuel cycles have been calculated and compared. The comparison is in good agreement. The use of these analyses for providing input for dynamic simulation studies has been taken up.

VVER-1000 LEU and MOX Benchmark Analysis

This work has been taken up by OECD members to ascertain the suitability of MOX fuel as an alternative to LEU for VVER-1000 type reactors. The Russian Federation and the United States are pursuing the deployment of mixed (UO_2 + PuO_2) fuel (MOX) in LWRs for utilization of plutonium obtained from the disassembled nuclear warheads. Recent work in Russia has focused on the certification of the calculational codes, design of MOX fuel assemblies and core configurations. The experts group established at OECD/NEA has performed several benchmarking efforts to help in the code certification process by providing experimental data and by sponsoring benchmarking exercises that provide useful verification of the calculational methods. While several computational codes and data have been certified for LEU-based fuel, the certification for MOX fuel is still required because of the essential differences in the physical characteristics of reactors fueled with LEU and MOX. In the present work an attempt is made to analyze the OECD/NEA benchmark *'A VVER-1000 LEU and MOX Assembly Computational Benchmark'* (NEA,

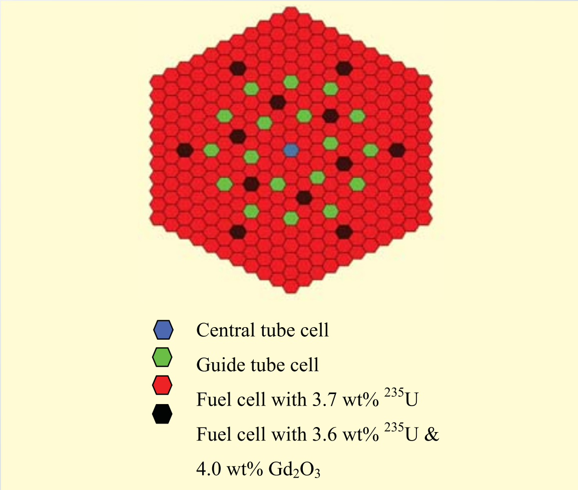

Fig. 3: Schematic diagram of a uniform LEU fuel assembly

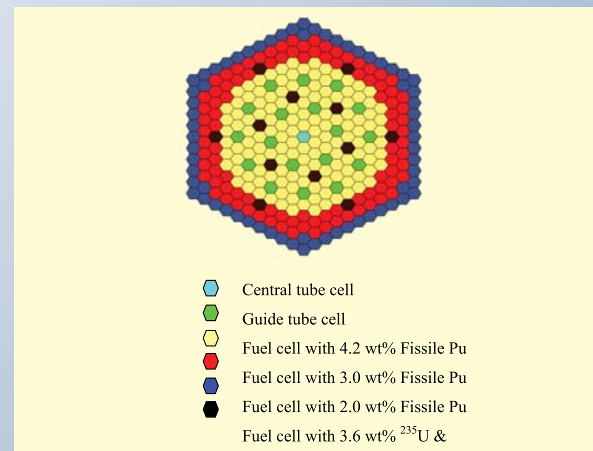

Fig. 4: Schematic diagram of a profiled MOX fuel

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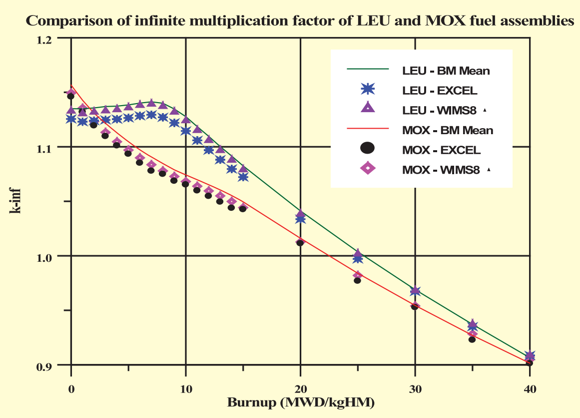


*Fig.5: Variation of **k-inf** with burn up for LEU and MOX fuel assemblies*

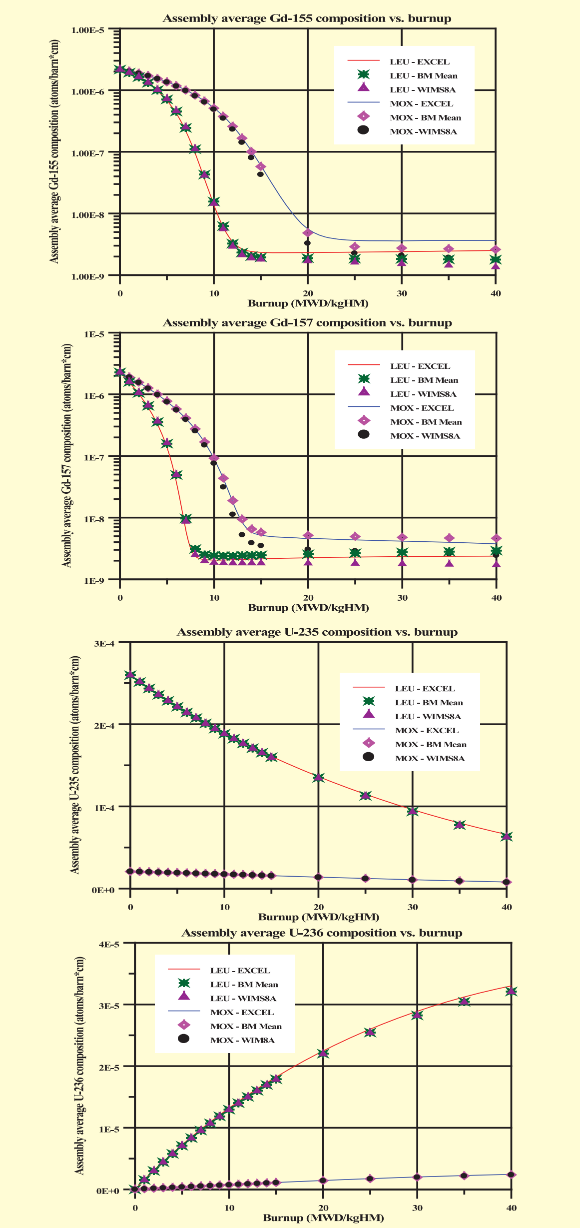


Fig.6: Assembly average isotopic composition as a function of burn up

2002). Figs.3 and 4 show schematic diagrams of profiled LEU and MOX fuel assemblies.

In the present benchmark analysis, the aforementioned calculations are performed for S1 state (operating poisoned state) with a power density of 108 MWt/m³ up to a burnup of 40 MWD/kgHM with sufficiently fine burnup steps to obtain accurate results, particularly till the burn out of the Gd absorber. The *k-inf* variation of LEU and MOX assemblies with respect to burn-up is shown in Fig.5 for the operating poisoned state (S1). The EXCEL values are compared with international benchmark mean values. The results of the code WIMS8A are also included in the plots of *k-inf* for ready comparison. For LEU assembly, as the burnup increases the reactivity increases initially till Gd burns out and then monotonically decreases. But such an initial increase in *k-inf* with burnup is not seen in MOX assembly. This can be attributed to the harder neutron spectrum due to higher absorption cross section of plutonium isotopes. The gadolinium isotopes burn slower in the MOX fuel assembly and hence the reactivity decreases slowly with burnup, up to the point at which the burnable absorber gadolinium is nearly depleted, and then, decreases faster with burnup in a linear manner. It is also seen that *k-inf* of MOX assembly decreases slower with burnup than that of uranium fuel. This is mostly due to capture in Pu-240 which gets converted to Pu-241, which is a fissile nuclide with higher absorption and η value than even Pu-239. The changes in isotopic composition for some of the important nuclides are shown in Fig. 6.

LWR Studies with Monte Carlo Codes

KENO-VI Monte Carlo module of the SCALE V.5 Computer Code System of ORNL was commissioned at SRI and run for the KKNPP core as a means of alternate validation of the BARC diffusion theory codes. The ORIGEN module of SCALE system was also used to do independent burn up verification calculations. Work has been also taken up for use of the codes for analysis of alternate fuel cycles like MOX. With the above activities the capability at SRI for LWR steady state reactor physics analysis computations has been established.

External Neutron Source Calculations for Prototype Fast Breeder Reactor (PFBR) Start Up

This work using Monte Carlo method was carried out to cross check calculations made earlier at IGCAR by diffusion theory. External neutron source subassemblies (Sb-Be) are used to obtain an enhanced count rate at the detector location for the efficient start up of PFBR. The external sources considered in the present calculation are of 35 cm length, located at the first blanket ring of the PFBR core. The length of external source is arrived at by parametric analysis, in order to handle it safely by the existing IFTM (Inclined Fuel Transfer Machine of PFBR) and replace it after 5 cycles of operation and storage. It is found that with 3 such external source subassemblies, it is possible to get a maximum of 45.5 cps as shutdown count rate (with detector sensitivity of 1 cps/nv) on the control plug detectors after 2 months of reactor operation. This count rates reduces to 9.1 cps if the detector efficiency reduces from 1 cps/nv to 0.2 cps/nv. During the source subassembly replacement only 2 active external source subassemblies will be present in the core, which can give rise to count rates of 30.5 and 6.1 cps with detector sensitivities of 1 and 0.2 cps/nv respectively.

Criticality Safety of Stacked Fast Reactor Fuel Subassemblies

The fuel subassemblies (FSA) of a fast reactor are normally either assembled in controlled critical configurations in the reactor core or stored under sub-critical configurations in fuel storage facilities. A new study was made of the criticality safety of inadvertent stacking of fresh FSA during the fabrication process on the shop floor. The study was used to provide input data for the safety review of the Fuel Assembly and Storage Building of the Prototype Fast Breeder Reactor.

While the workshop assembly procedure administratively precludes stacking of finished FSA, the present new study establishes the safety margins available in the case of mal-operation resulting in the inadvertent stacking of a few FSA.

The finite lattice criticality calculations were made using the 3D Monte-Carlo code MCNP with the inbuilt Evaluated Nuclear Data File (ENDF)/B-VI. The following finite lattice cases were

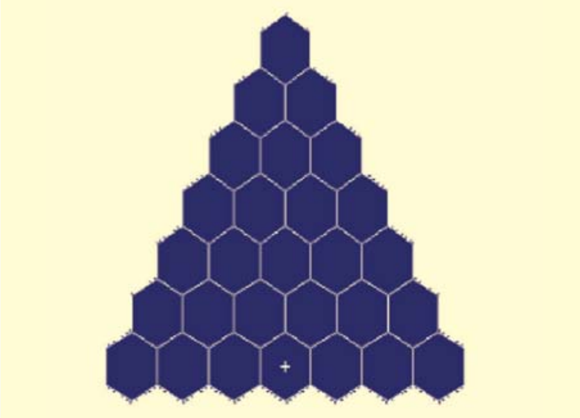


Fig.7: Closely stacked lattice of 28 FSA (Triangular shape)

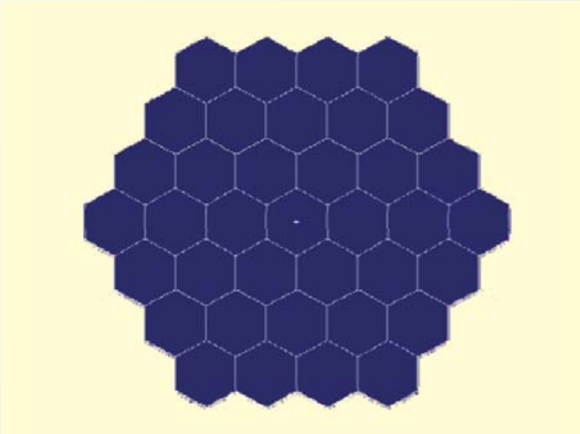


Fig. 8: HCP lattice of 37 FSA

considered. First a finite lattice of FSA closely stacked in an overall jagged triangular shape as would be natural when stacking on the floor (Fig. 7) and secondly a Hexagonal Closely Packed (HCP) lattice of FSA (Fig. 8).

The results obtained showed that for inadvertent stacking in air the k_{eff} remained $\leq$ 0.9 for up to 57 FSA in the overall triangular shaped stacking and up to 46 FSA in HCP stacking. However, in case of water flooded conditions the safe ($k_{eff} \leq 0.9$) number of FSA reduced to 6 for the case of the HCP stacking.

RADIOLOGICAL SAFETY STUDIES

Experimental Verification of Safety Factor used in Reactor Shield Design

In the design of PFBR, penetrations are provided in the top shield to carry out some essential operations. Radiation streaming is envisaged through such penetrations. To avoid radiation streaming, complementary shielding is provided by incorporating a suitable safety factor to

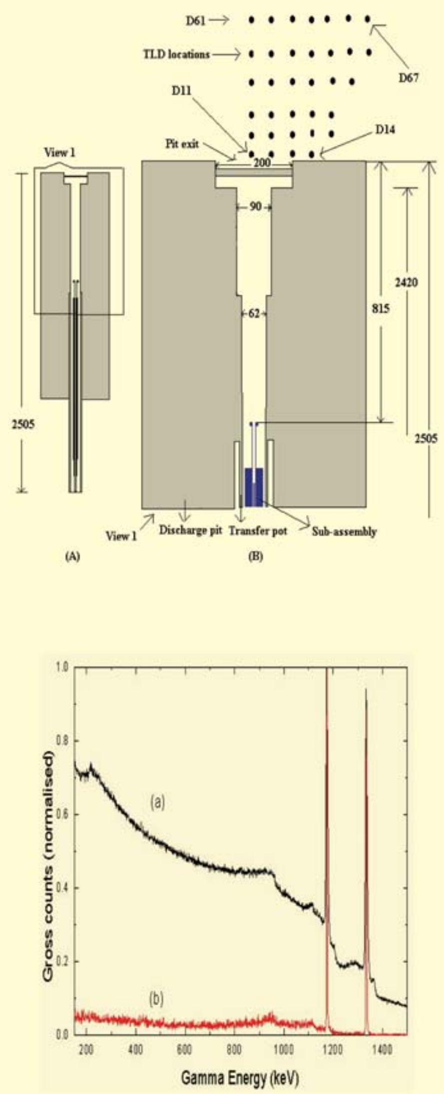


Fig.9 (A): Schematic diagram of subassembly storage pit of FBTR
Fig.9 (B): Gamma spectral distribution by Germanium detector.
a) At 1000 mm from the mid point of the pit exit and at an angle 10^o away from the axial line.
b) Standard ^{60}Co spectrum

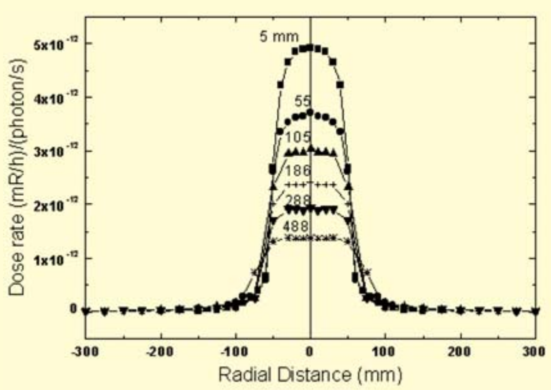


Fig. 10: Calculated dose rates above the exit of the pit. Height from the top of fit is indicated in mm(work carried out in collaboration with IGCAR)

account for uncertainties involved in the calculations. The magnitude of the safety factor, while designing the reactor shielding, has been validated by undertaking experimental measurements on a similar geometry vis-à-vis the computed values obtained using MCNP code. The work was carried out in collaboration with IGCAR.

An experimental set-up in the form of discharge pits wherein irradiated (SA) is being stored was identified in FBTR. The schematic of the pit with TLDs locations is shown in Fig. 9 (A) and the Count spectrum obtained with HPGe spectrometer is shown Fig.9 (B) which indicates the presence of induced activity of Co-60. The discharge pit geometry at FBTR is proved to be the most suitable experimental facility for analysing the radiation streaming through a cylindrical duct geometry. The magnitude of the dose rates obtained by calculation and measurement varies within a factor of two for most of the locations which is considered acceptable. The present work clearly concurs with the general assumption of the safety factor of two incorporated in shield designs using Monte Carlo code for gamma ray shielding. Typical computed results are given in Fig.10.

Photo Neutron Flux and Dose Rate Estimations for ThO_2 Bundle Immersed In D_2O

This assignment was undertaken in collaboration with RPDD, BARC to quantify the photo neutron flux as well as dose rate for the situation when thoria bundle gets completely immersed in a cylindrical heavy water pool of radius 165cm and height 200cm. The assignment has application in AHWR neutron flux calculations. Computational model of thoria (ThO_2) bundle consisting of nineteen pins with Zircaloy clad was made using MCNP code. The photon emitted by $_{81}Tl^{208}$ isotope formed in the decay chain of $_{90}Th^{232}$ causes the production of photo-neutrons in heavy water. The source term of $_{81}Tl^{208}$ isotope formed in the Thoria bundle under study (year of manufacture 1984) has been estimated using ORIGEN-S code and is found to be 1.57 x 10^7 photons/sec. Photo-neutron flux and dose rate on contact at the centre of the Thoria bundle estimated are 25.86n.cm^{-2}.sec^{-1} and 2.02µSv/h

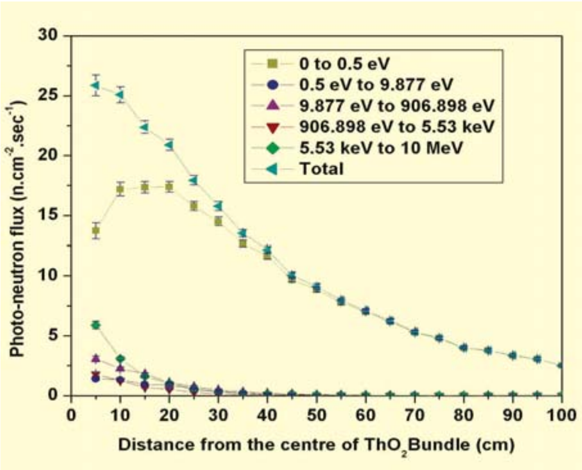


Fig.11: Radial Variation of Photo Neutron Flux in D_2O Pool

respectively. Photo-neutron flux and dose rate estimated at the top of the thoria bundle on contact are 18.2 n.cm^{-2}.sec^{-1} and 1.54 µSv/h respectively. The reaction through which photo-neutron is produced with deuterium and natural decay of thorium is shown below.

${}_1H^2$ $+ \rightarrow \gamma$ ${}_1H^1$ $+$ ${}_0n^1$ (threshold energy of the reaction = 2.226 MeV)

Photo-neutron flux (Fig.11) as well as dose rates at different radial and axial distances from the centre of the thoria bundle immersed in heavy water pool was computed using coupled photon neutron particle transport code. Photo-neutron flux and dose rate at 5cm radially away (on contact) from the centre of the thoria bundle are 25.86 n.cm-2.sec-1 and 2.02µSv/h respectively. Subsequently, total gamma dose rate on the surface has been found to be 580 µGy/h.

Estimation of Niobium-94 Activity in Irradiated Zr-Nb Alloys

Estimation of magnitudes of long lived isotopes produced in the reactor structures is an important parameter for deciding proper waste disposal. This work was carried out in collaboration with KARP, BARCF. Two garter springs are used to support the pressure tubes (in PHWRs) to prevent contact between the coolant tube and calandria. Opportunity to measure the induced radioactivity in garter spring (Fig.12) was utilized using HPGe gamma ray spectrometry (during the Enmasse Coolant Channel Replacement (EMCCR) work of Madras Atomic Power Station UNIT-1:MAPS-I). The -spectrum indicated the

Fig.12: Photograph of Garter spring
Composition and dimensions : GS wire - (Zircaloy-2).
GS (Zr-Nb-Cu, with cobalt impurity)
Weight - 43.5 gms
I.D = 91 mm O.D = 105 mm
Thickness = 7mm

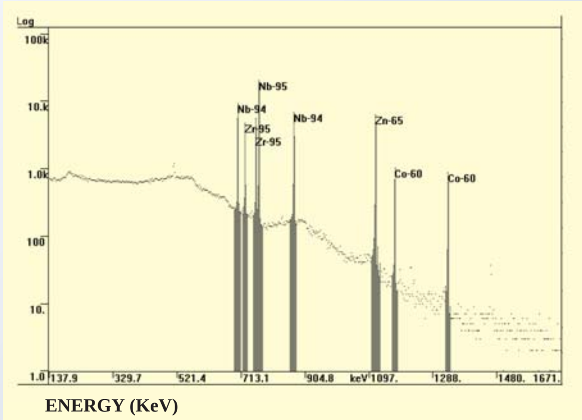


Fig.13: HPGe gamma ray spectrum of irradiated and cooled Garter spring. Prominent isotopes are indicated

presence of Zr-95, Nb-95, Nb-94, Zn 65 and Co-60 as shown in Fig.13. The measured specific activities of the radionuclides are compared with those calculated using ORIGEN2 code and found to be in good agreement. The experimental data is used for estimating the dose rates on the garter spring and also for validating the gamma ray shielding code GUI2QAD. The dose rates obtained from the experimental data are found to be in good agreement with those calculated using the GUI2QAD code when the cobalt impurity is assumed to be about 0.2 ppm. The significance of Nb-94, with 0.70, 0.87 MeV γ-photons and ~20300 years half life present in the irradiated garter springs has been highlighted.

New Algorithm for Design of Beam Flattening Filter for High Energy LINAC

High energy electron linear accelerators (LINAC) are being widely employed for treatment of

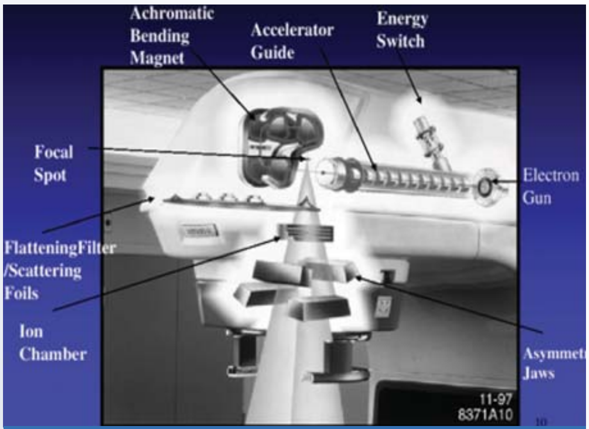


Fig.14A: Schematic of LINAC Head with Beam Flatteners

Fig.14B: Expanded View of Beam Flatteners

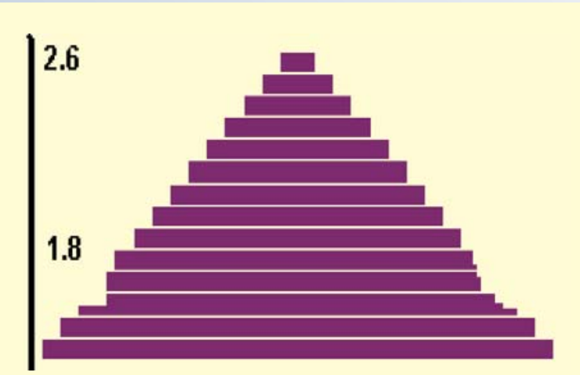


Fig.15: Approximate shape of beam flattener

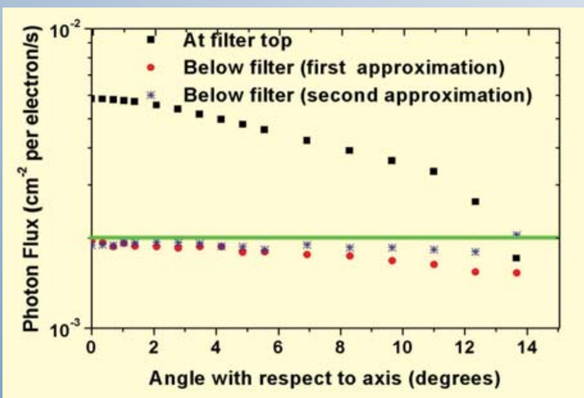


Fig.16: Monte Carlo simulated intensity levels with iterations. Green line is the desired level

tumors in the country. High energy electrons on impinging the target, produce Bremsstrahlung photons that are forward peaked. However, medical applications demand uniform nearly flat radiation level. To achieve desired level, beam flatteners are used. Fig.14A. shows the schematic of LINAC Head with Beam Flatteners and Fig.14B Expanded View of Beam Flatteners in conical shape.

A new approach for the beam flattening filter design has been proposed based on the iterative algorithm. The steps involved are estimation of photon flux and mean energy distribution at the flattener mounting plane at various angles from the central axis by Monte Carlo simulation and computation of thickness $X(\theta)$ of the material at each angle (θ) as a guess estimate using

$X_0(\theta) = \log(I(\theta)/I_0)/\mu$

Where I_0 is the desired intensity level

$I(\theta)$ is the Monte Carlo simulated intensity at an angle θ

μ is the linear attenuation coefficient of the beam flattener material.

Approximate shape of the beam flattener obtained as initial guess is shown in Fig.15. with y-axis showing the scale in cm of tungsten.

With this guess values of thicknesses, Monte Carlo simulation is repeated for obtaining $I(\theta)$ values and the next best estimate of thicknesses is obtained using

$X_{n+1}(\theta) = X_0(\theta) * \log(I(\theta)/I_0$

The above expression is used iteratively till desired convergence limit is obtained. Fig.16 shows the convergence of radiation level quickly with 2 iterations for 6 MeV LINAC case. Results generated for 10 and 18 MV cases agree with the published values and thus validate the algorithm proposed.

Estimation of Neutron Dose Contamination in LINACs

AERB constituted a committee to ascertain the magnitudes of neutron dose rates in relation to photon dose rates produced in the LINACs. As it is well known that LINACs operating at higher voltages (> 10 MV) can also produce photo-neutrons, which would expose the entire body of the patient. The magnitudes of such doses in relation to the photon doses are of importance. The magnitude of neutron doses depends on the operating voltage, target, beam flatteners and collimators provided in the machine. Neutron production and doses for 18 MV beam used for photon radiotherapy have been investigated. The results of the studies show that the neutron doses are marginal and the magnitude is 200 µSv per Gray of photon dose and the magnitudes are confirmed with measurements using bubble dosimeters. This work was performed in collaboration with RSD of IGCAR.

DEVELOPMENT OF SAFETY RELATED COMPUTER CODES

IGSHIELD

IGSHIELD is an Interactive Gamma-ray Shielding code developed in Visual Basic for WINDOWS operating system. The computational methodology is based on the point kernel technique and is capable of handling complex configuration of sources and shields. Fig.17 shows the opening screen of the code with insets showing various capabilities of the code. The salient features of the code are i) provision to declare point, line, plane, sphere, slab, cylinder, hexagonal cylinder and cone as sources ii) handling of multiple source shapes with accompanying energy distributions, iii) arbitrary orientation of sources and iv) provision to declare array of sources. The code is validated by comparing the results for the standard shielding problems with analytical equations and MCNP code.

BRACHYTPS

Most of the commercially available brachytherapy treatment planning systems (TPS) software estimate the absorbed dose at a point, taking care only of the contributions of individual sources and the source distribution, neglecting the dose perturbations arising from the applicator design and construction. For precise

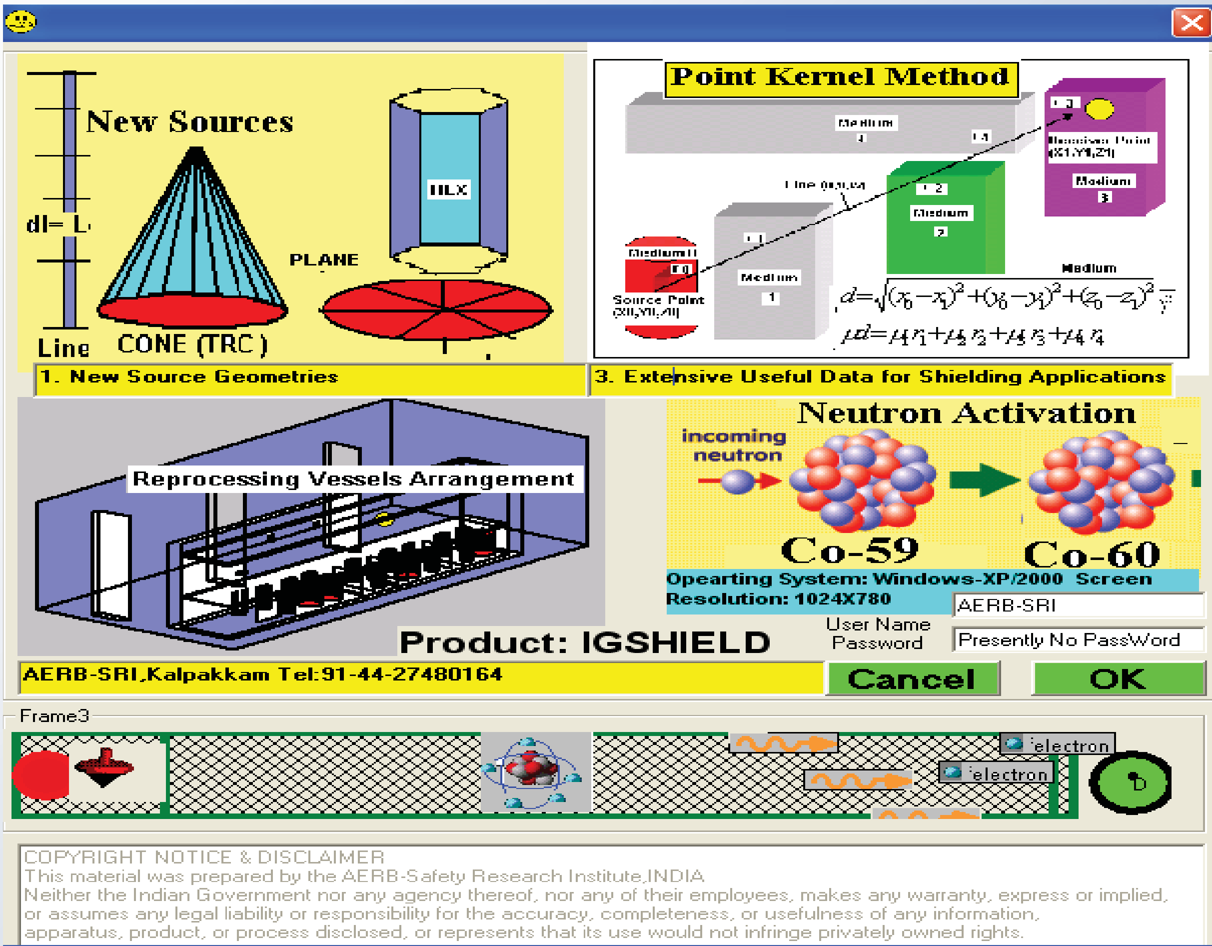


Fig.17: Opening screen of the IGSHIELD code with insets showing some features of it.

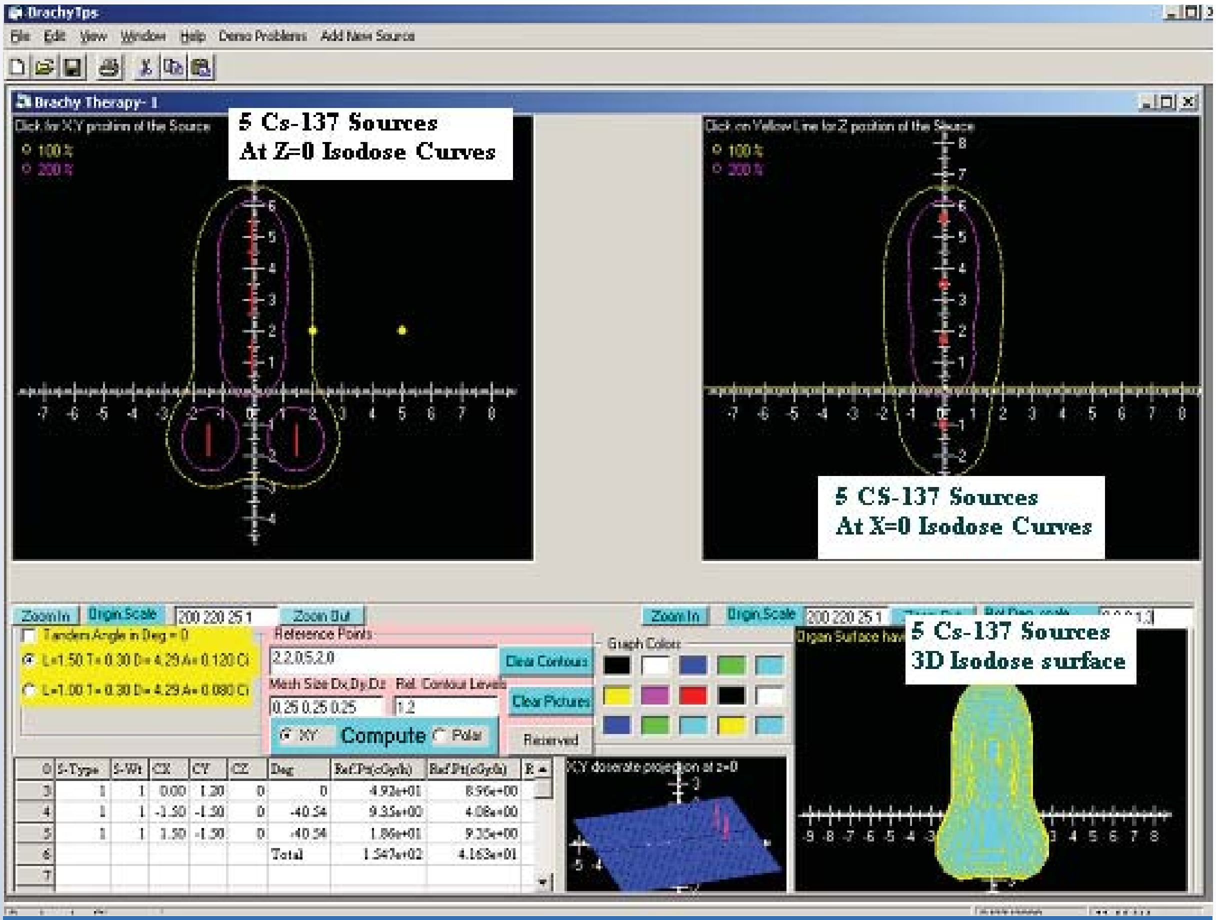


Fig.18: BrachyTPS code interactive window with isodose curves of BRIT applicator

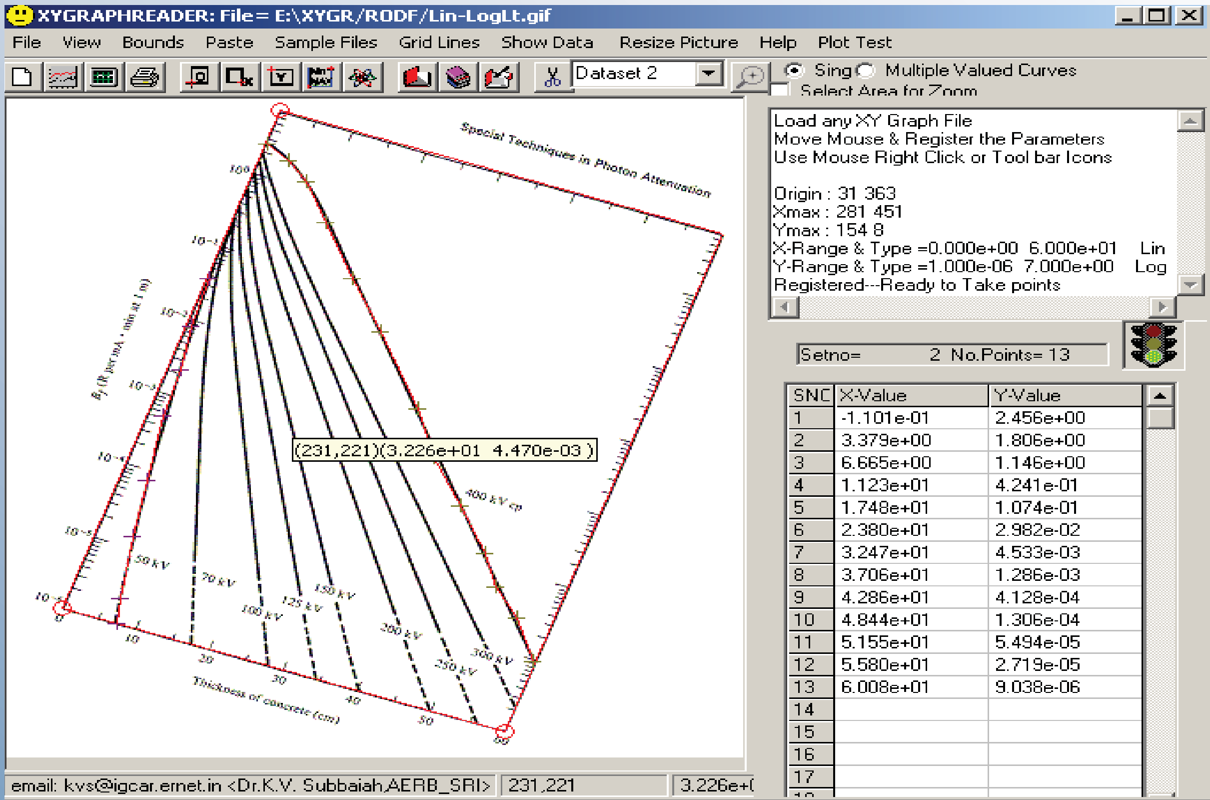


Fig.19: Sample of Graph Reader Utility showing x and y values on the right side grid box

dose estimation, it is required to account for the effect of tissue, applicator and shielding material heterogeneities that exist in intracavitary brachytherapy applicators. In this regard, interactive point kernel code (BrachyTPS) has been developed to perform independent dose calculations by taking into account source distribution and attenuation of source, applicator and tissue materials. Comparision of the present results with those obtained by Monte Carlo method showed an excellent agreement and it is found to be <5% in all the cases studied. Fig.18 shows the main window of BrachyTPS code with inset drawings of isodose distribution lines for a BRIT applicator loaded with 5 Cs-137 sources.

XYGRAPHREADER

This is an utility to digitize X and Y values from scanned graphs. This is a program in VB-6 written for WINDOWS-2000/WINDOWS-XP operating systems. It is based on the principle of converting x,y pixel position into corresponding values from the input values provided by the user, a need often felt by researchers. Fig.19 shows the form layout design with a sample of graph reader utility. This program is in use at IGCAR, AERB and BARC.

Web Interface for Source Shield Calculation

A web interface was developed using JAVA for calculation of dose rate from a source in arbitrary complex geometry of shield configurations. The calculation is based on point kernel technique in which original source volume is divided into small pieces of voxels (small volume). The voxel points are generated using a Monte Carlo algorithm. From each voxel the optical path traversed in the material media to the detector point is estimated, then a suitable buildup factor is applied to account for scattered contribution and the dose rate is computed based on point source formula. Finally, the dose rates are summed over all voxels to estimate the dose rate due to entire volume of the source. The application is made available on the intranet server and is being used by health physicists.

The Active Server Page (ASP) of the application is shown in Fig.20.

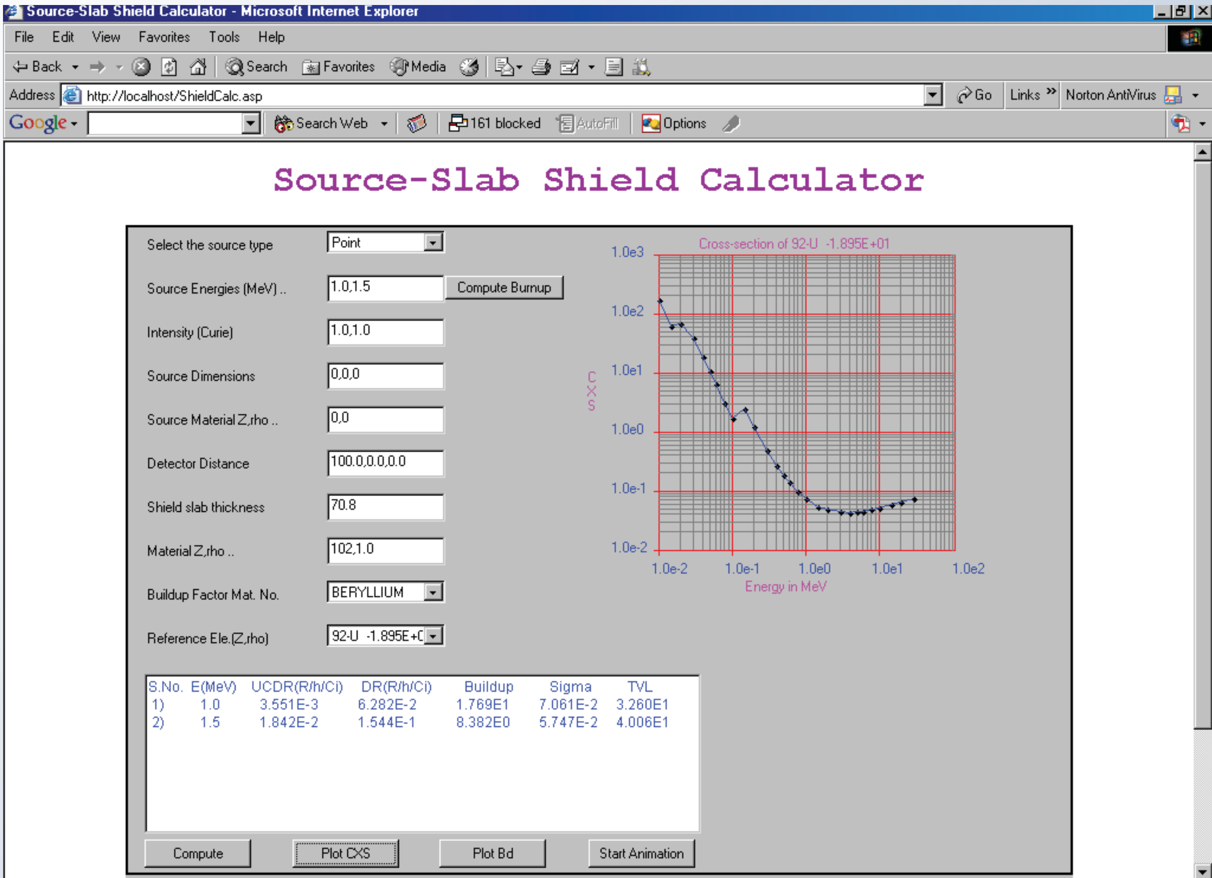


Fig.20: ASP page layout of source-slab shield calculator. On right side display of log-log graph of U-235 photon cross section

3.1 (b) PROBABILISTIC SAFETY ASSESSMENT AND RELIABILITY STUDIES

INTRODUCTION

Probabilistic Safety Assessment (PSA) is a systematic and comprehensive methodology to evaluate risks associated with a complex engineered technological entity such as a facility, a spacecraft or a nuclear power plant. PSA of an NPP is an integrated assessment of the safety of the plant by considering the probability, progression and consequences of equipment failures or transient conditions. It provides an approach to determine whether the safety systems are adequate, the plant design is balanced, the defence in depth requirement has been realized and the risk is as low as reasonable achievable. It identifies the failure scenarios and derives numerical estimates that provide a consistent measure of the safety of the plant and the risks to workers and members of the public. Internationally, PSA is increasingly being used as a part of the decision making process to assess the level of safety of nuclear power plants. The methodologies have matured and the insights gained from the PSAs are being used with those from deterministic analysis in the regulatory decision making process - referred to as risk informed regulation.

AERB has identified PSA as one of the main areas of research and development activity in SRI. The following are some of the works that have been completed or are currently in progress:

1. Level-1 PSA of PFBR
2. Reliability analysis of Decay Heat Removal Systems of PFBR
   - Failure Modes and Effects Analysis of Safety Grade Decay Heat Removal System (SGDHRS)
   - Reliability of SGDHRS and OGDHRS
   - Functional Reliability Analysis of SGDHRS
3. Neutronic Channels Reliability for Shutdown system
4. Reliability Analysis for Seismic Re-evaluation of Fast Breeder Test Reactor
5. Database on Fast Reactor Component Failure
6. Estimation of Optimum Test Interval for Maintenance of Standby Systems

LEVEL - 1 PSA OF PFBR

Level-1 PSA is performed to identify the potential accident sequence of events that can lead to core damage and estimate the core damage frequency in a plant. It provides insights into the strengths and weaknesses of the safety systems and procedures provided in the plant to prevent core damage. SRI is actively participating in carrying out the Level-1 PSA of Prototype Fast Breeder Reactor (PFBR). The reliability analysis of important safety systems viz, shutdown system, decay heat removal system, power supply system have been completed. Presently, grouping of potential initiating events, development of event trees and fault trees are in progress.

RELIABILITY ANALYSIS OF DECAY HEAT REMOVAL SYSTEMS (DHRS) OF PFBR

In PFBR, there are two systems to remove the decay heat. The first system, known as Operating 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 in steam water system or there is a loss of off site power, the second system known as Safety Grade Decay Heat Removal System (SGDHRS) is called into operation. Reliability analysis of decay heat removal systems is carried out by identifying the potential failure modes of the components to be included in the analysis using a technique called Failure Modes and Effects Analysis (FMEA) and then a detailed estimation of system failure using fault trees.

Failure Modes and Effects Analysis (FMEA) of SGDHRS

FMEA is a systematic, analytical bottom-up approach, to identify and focus on those areas in the design and manufacturing process for the prevention, reduction, and elimination of non-conformances in a system. Each potential defect or failure mode for all the components in a system is analyzed, to determine the severity of the failure effects, with the purpose of

Table-1: Sensitive Components and Recommended Actions

Components	Expected Impact on Reliability	Recommended Actions
Main System		
DHX, AHX	High	Diverse components
Dampers	High	Diverse components.
Intermediate Circuit Piping	Moderate	NIL
Sensors and associated instrumentation for Damper auto closing	Moderate	Diverse PLC for the two types of loops suggested
Temperature indicators at the AHX outlet	Low	NIL
Support and Interfacing Systems		
Dump valves	Moderate	Diversity, separation barriers, interlocks to prevent manual dumping of all loops. Separation barriers to ensure the availability of other dump valves in case of fire or leak in one valve.
Filling line to DHX	Moderate	NIL
Check valve in argon lines of Expansion tank	Low	NIL
Common argon header/lines	Low	Separate Lines/capacity for each group.
Class I & II Power supplies	Low	Independent supply for each group
Plugging indicator and purification circuit	No Impact	NIL
Nitrogen Injection	No Impact	NIL

eliminating them. Thus, FMEA is a defect prevention tool that can formalize system reliability planning. The final result of the analysis is the generation of a Risk Priority Number (RPN) for each component of the system and remedial action for those components which has unacceptably high RPN.

A detailed FMEA was carried out for SGDHRS and RPN was estimated for all the components of the system. Table-1 presents the results of some important system components and support / auxiliary systems and the recommended actions.

Reliability of SGDHRS and OGDHRS

Reliability analysis of both SGDHRS and OGDHRS was carried out by employing Fault Tree method. As per AERB criteria, the failure frequency of DHR should be < 1 x 10^{-7} /ry. For SGDHRS, analysis of different options of design indicates that diversifying the sodium heat exchangers or dump valves alone do not give appreciable

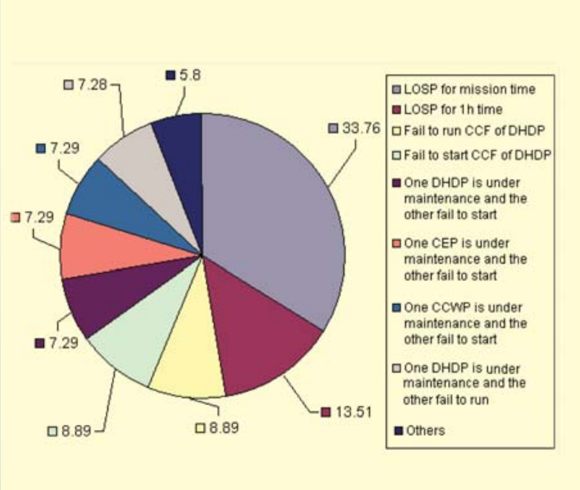


Fig. 1 Contribution of Events for OGDHRS Unavailability

improvement. Diversifying air heat exchangers alone gives a moderate improvement. However, if all the three components are diversified, very good improvement is observed. The unavailability of OGDHRS on demand for different types of initiating events and their contributions were found out. Sensitivity analysis was carried out to study the system response to variations in basic input data. The pie chart (Fig. 1) shows the system unavailability contribution from various events.

Functional Reliability of SGDHRS

Passive systems are increasingly deployed in nuclear industry with the objective of increasing reliability and safety of operations. Methods for assessing the functional reliability of thermal hydraulic passive systems, viz., systems with moving working fluid, address the issues in natural buoyancy driven flow that could result in a failure to meet the design safety limits under accident scenarios. Evaluation of functional reliability of SGDHRS was done. A simple functional diagram of DHR is shown in Fig.2. Various parameters that affect the performance of SGDHRS were identified and analysed. Using response surface method and with a large number of Monte Carlo simulations, functional failure probability of SGDHRS was estimated. The evolution of hot pool temperature is shown in Fig. 3. It is found from the analysis that the probability of functional failure of SGDHRS (on natural convection) to limit temperatures of critical structures to their design safety limit is dependent on the number and duration of loop availability during the initial few hours of mission. A procedure was developed to integrate the functional failure probability with classical reliability to estimate the total DHR failure probability in PFBR.

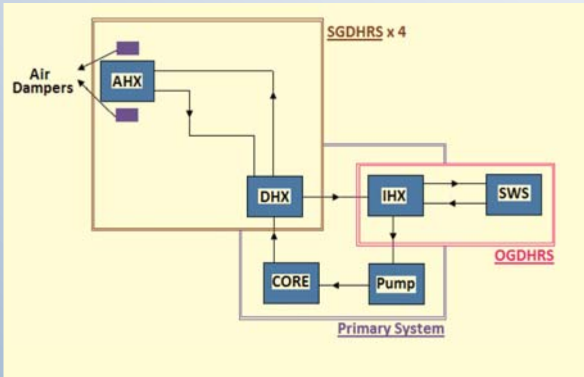


Fig.2 Reliability block diagram of DHR of PFBR.

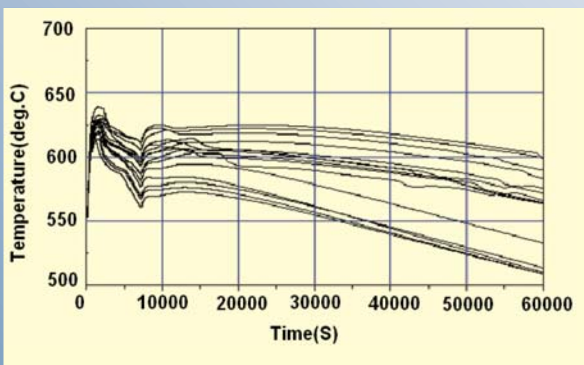


Fig. 3 Evolution of Hot Pool Temperature as function of time when 3 loops failed at 2h.

NEUTRONIC CHANNELS RELIABILITY FOR SHUTDOWN SYSTEM

PFBR has two Shutdown Systems (SDS) and each SDS consists of reactor protection system, actuation system and safety support systems. To compute and compare individual SDS reliability, reliability of SDS1 that receives signals from neutronic channels was evaluated by considering signals from neutronic and flow parameters. The failure probability of individual channel was estimated and it was found that if there are redundant channels the failure probability increases marginally and is dominated by common cause failures of flow sensors, flux sensors and P/Q computing element.

RELIABILITY ANALYSIS FOR SEISMIC RE-EVALUATION OF FAST BREEDER TEST REACTOR (FBTR)

Seismic re-evaluation of FBTR is carried out to review the seismic capacity of the safety related Structures, Systems and Components (SSCs) of the plant required to achieve four safety functions viz., safe shutdown of the plant, maintain the plant in safe shutdown condition, achieve decay heat removal and confine radioactive materials. This review exercise is conducted with respect to the ground motion, termed as Review Basis Ground Motion (RBGM). The activity of re-evaluation of FBTR is divided into ten tasks as shown in the flow sheet (Fig.4).

SSCs needed for above mentioned functions are selected based on safety analysis by developing Event Trees and Fault Trees for potential seismic initiating events. The objective of this task is to identify the frontline and support systems that perform the required safety functions. A list of around 450 Seismic Structures, Systems and Components was arrived at based on the analysis.

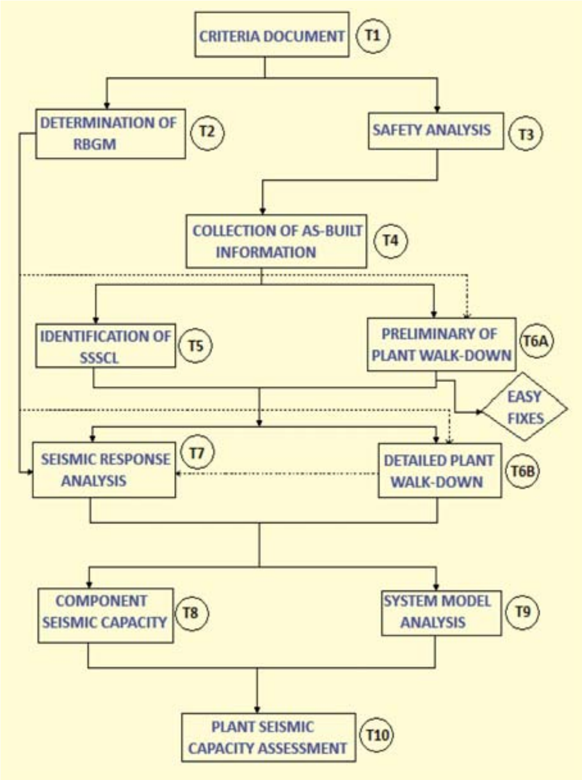


Fig.4 Task flow for seismic re-evaluation of FBTR

The SSCs are grouped into three categories based on their structural and functional features for their qualification either by plant walkdown as per guidelines given in 'Generic Implementation Procedure" of Department of Energy USA (DOEGIP) or by analysis or testing.

Two seismic plant walkdowns were performed to evaluate the key physical attributes of the SSCs and to evaluate their seismic capacity. Items that pass the screening are considered to possess adequate seismic capacity and those that do not pass the screening are called outliers and a detailed review / upgradation is necessary for these items depending on the potential risk.

A procedure was developed to assess the capacity of each component in terms of a single parameter called High Confidence of Low Probability of Failure (HCLPF) capacity. The HCLPF capacity is represented in term of acceleration due to gravity (g).

After the review and finalization of system fault trees and event trees and identification of the accident sequences for all the initiating events, system HCLPF is computed by using the component HCLPF capacity. By deterministic analysis, the seismic margin of FBTR was obtained with the dominant contributors identified viz., LOR due to pump seizure and LOR due to sodium leak.

By probabilistic approach, seismic core damage frequency was estimated using the component fragility. In view of the large number of components in the structures, systems and component list, grouping of components based on location is done for component fragility estimation.

DATABASE ON FAST REACTOR COMPONENT FAILURE

A visual basic application was developed with MS-SQL as the backend to maintain a database of failure rate data on fast reactor components. The component failure data is stored and retrieved from a relational database system. The application provides a user friendly interface to add, modify and obtain a suitable failure rate for a given component specification viz., safety system, category, group. Apart from graphical representation of the collected data and calculation of basic statistical information such as mean, median, standard deviation, the application also provides a facility to combine the operating experience with the stored data and estimate the posterior failure data using Bayesian technique (Fig.5). The application is made available on IGCAR intranet for the users to retrieve component reliability information.

ESTIMATION OF OPTIMUM TEST INTERVAL FOR MAINTENANCE OF STANDBY SYSTEMS

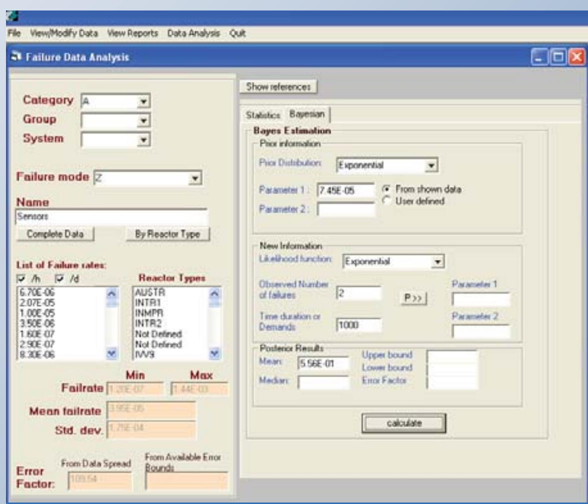


Fig. 5 Snapshot for Bayes Application

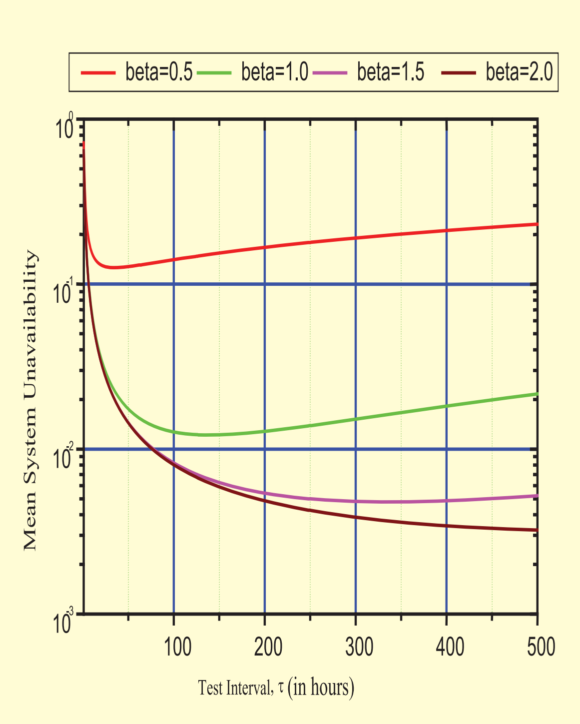


Fig.6 Mean System Unavailability with Weibull Failure Time distribution for 1/2 DG configuration

A statistical model for an optimum maintenance scheduling for periodically tested standby Diesel Generators (DGs) in nuclear power plants which minimizes the system failure rate based on reliability has been developed. A simple analytical expression was obtained for the mean unavailability of such systems from which an optimum maintenance period can be arrived. When a Weibull distribution is assumed for the DG failure times, the mean system unavailability (Fig. 6) is expressed as

$$U_T = \left[\frac{\tau_r}{\tau} + \delta + q_0 + (1 - q_0)\frac{(\lambda\tau)^\beta}{\beta + 1} \right]$$

and the optimum test interval is obtained as

as $$\tau^* = \sqrt[(\beta+1)]{\frac{(\beta+1)\tau_r}{(1 - q_0)\beta\lambda^\beta}}$$

where is the standby failure rate, r is the test duration, is the test interval, is the probability of human error during testing, q0 is the failure probability on demand and is the shape parameter.

3.1 (c) STRUCTURAL AND SEISMIC STUDIES

INTRODUCTION

The safety of a nuclear power plant (NPP) depends upon a number of factors - internal and external to the plant. Earthquake is one of the external events considered during the assessment of design safety features. Design Criteria of Nuclear Power Plant (NPP) require that all Structures, Systems and Components (SSCs) required for the safety functions must be qualified for Operating Basis Earthquakes (OBE) and Safe Shutdown Earthquakes (SSE). Some of the studies performed here towards the seismic qualification of structures, systems and components are as follows:

- Structural analysis of systems for seismic re-evaluation of Fast Breeder Test Reactor (FBTR)
- Seismic qualification of control room panel by a Novel FEA approach
- Development of a procedure for accounting for multiple support excitations

STRUCTURAL ANALYSIS OF SYSTEMS FOR SEISMIC RE-EVALUATION OF FBTR

Analytical methods were used for the seismic qualification of some of the safety related SSCs of FBTR, based on the structural and functional features. Analysis of the system was performed with respect to the ground motion generated for FBTR seismic re-evaluation, termed as Review Basis Ground Motion (RBGM).

A procedure was developed to calculate the capacity of each component in terms of High Confidence of Low Probability of Failure (HCLPF) capacity. Since RBGM is given in terms of acceleration due to gravity (g), HCLPF capacity is also calculated in the same units. The HCLPF capacity is based on the various failure modes of components and their corresponding limits given in the codes like ASME or RCC-MR.

Secondary Sodium Systems

The secondary sodium system is important for the decay heat (heat generated in core after reactor shutdown) removal from the core. This system needs to maintain its pressure boundary integrity for successful removal of decay heat.

Secondary sodium system was analyzed using the Floor Response Spectra (FRS) generated for steam generator ground floor level. Except IHX, all major equipment like pumps, expansion tanks, surge tanks, steam generators (SGs) etc. are mounted on steel structures in the steam generator building. Therefore, the steel structures were also modeled along with other equipment for seismic qualification. The integrated model is shown in the Fig. 1. Finite element analysis (FEA) code ANSYS was used for modeling and analysis.

The SGs of FBTR is a highly flexible system. Analysis of SG module was performed in two steps. First, it was modeled as a pipe element and

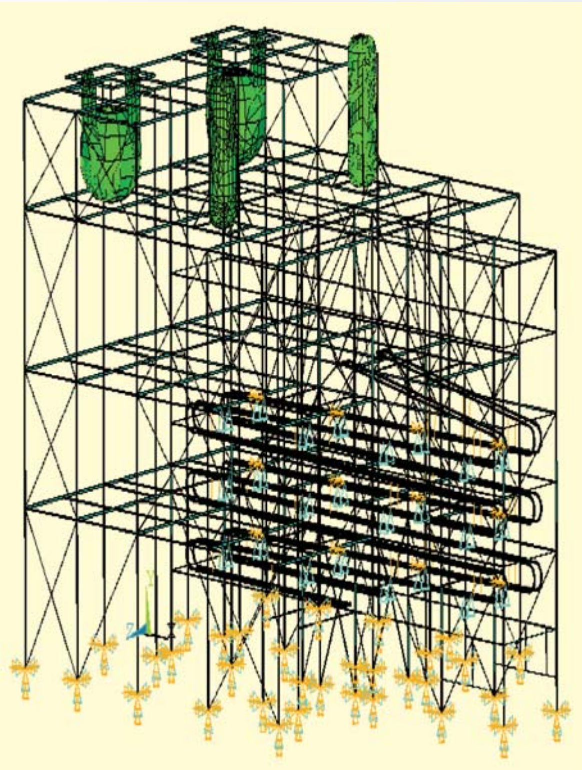


Fig.1 FEM model of SG, Expn.Tank and Surge Tank along with the steel structure

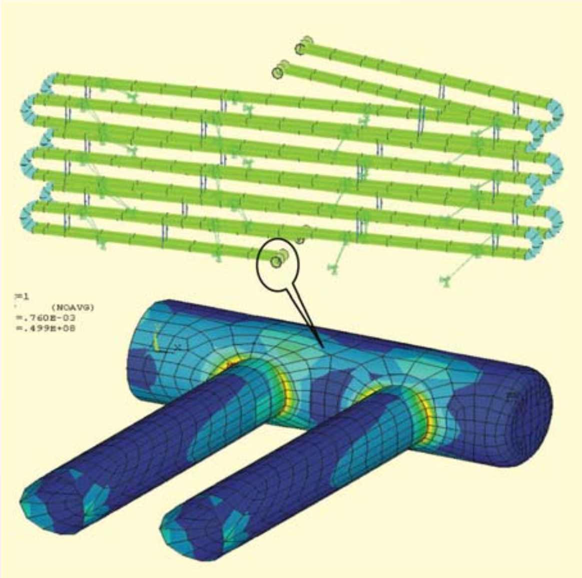


Fig.2 Membrane stress in SG Header of FBTR

the displacement boundary condition was extracted for critical locations like headers. Subsequently it was modeled in detail with shell element with displacement conditions obtained from the previous step, as shown in the Fig.-2.

Surge tanks, expansion tanks etc. were checked for structural integrity. Nozzles were assessed as per the guideline given in the Welding Research Council bulletin (WRC-297). Whenever the criteria of applicability of the bulletin were not met, detailed analysis was performed for the stress evaluation.

Pre-heating and Emergency Cooling System and Service Water System

Apart from the primary and secondary sodium systems which form the main heat transport path, one backup system was also analyzed. The most important backup system for prolonged cooling is the Pre-Heating and Emergency Cooling (PHEC) system and the Service Water System (SWS).

PHEC circulates cooled nitrogen between the reactor vessel and its double envelope to remove the decay heat. This system is classified as Seismic Category-I and Safety Class-II. Qualification of this system was performed with ASME Section-III, Subsection-NC. It contains a large no. of bellows and a procedure has been developed for the calculation of axial and lateral duty of the bellows.

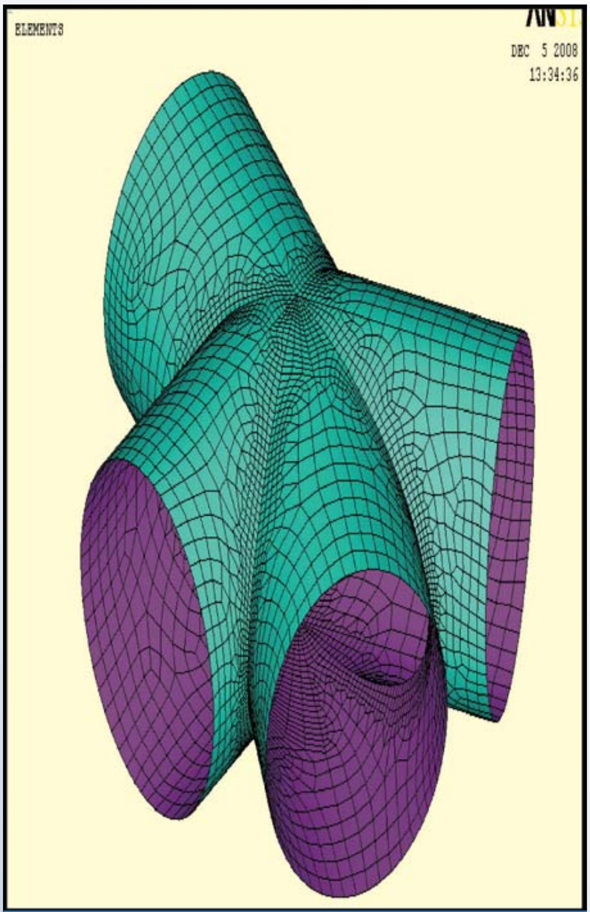


Fig.3 FEM model of a branch connection

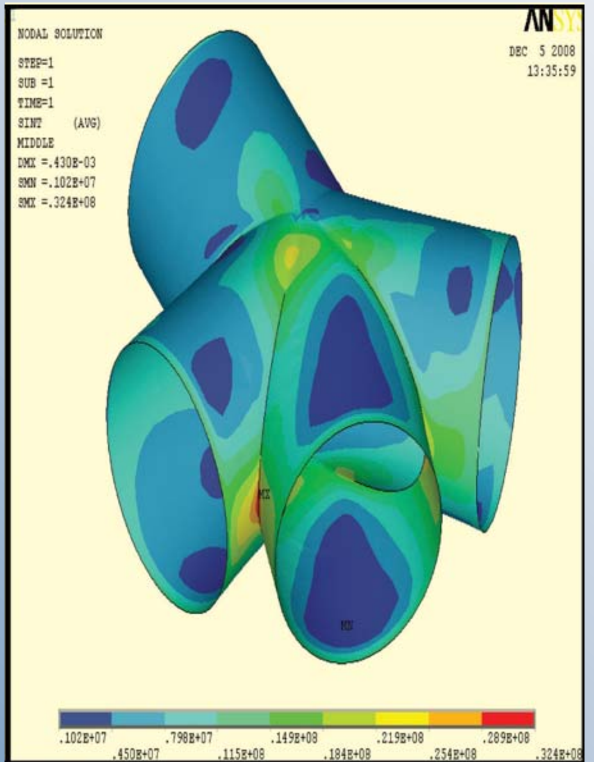


Fig.4 Membrane stress intensity in the branch

PHEC system inlet and outlet header have high diameter/thickness (d/t) ratio as well as many non-standard pipe fittings as shown in the Fig. 3 & 4. Codes have not specified stress indices for this type of component. Detailed FEM analysis was performed for the qualification of non-standard pipe fittings. Seismic analysis of this system has demonstrated that the system has sufficient capacity for RBGM level earthquake.

SWS is a low temperature system and adequately designed for the seismic load. At a few places, it was found that supports need some strengthening for capacity enhancement.

Although most of the systems of FBTR are not formally designed for RBGM level earthquake, it was found that many systems have satisfactory strength to withstand such loads. Some of the components like PHEC heat exchangers; expansion tank etc. have HCLPF capacity less than peak ground acceleration (PGA) of RBGM. Also few branches and nozzles in the secondary sodium system have lesser capacity and hence need retrofitting for capacity up-gradation.

SEISMIC QUALIFICATION OF CONTROL ROOM PANEL BY A NOVEL FEA APPROACH

Control room panel is a critical safety related equipment that needs to be qualified for OBE and SSE. A Finite Element Analysis (FEA) procedure has been developed to suitably supplement the costly method of shake table test for equipment like control panel, which are made of thin plates and minor attachments.

Control panels, as shown in the Fig. 5, having minor sections, thin plates, grids, bolted joints etc., are complex to model for FEA. The modeling of thin plates and tie rods using plate and bar elements overestimate the stiffness, as it does not properly account for the local vibrating mode under dynamic excitations. To overcome these difficulties, thin sections were modeled using beam and brace members. The inspiration for using brace members came from the behavior of a shear wall under dynamic loads in the building. The dimension of the brace member is obtained by fine-tuning the natural frequency obtained from the test.

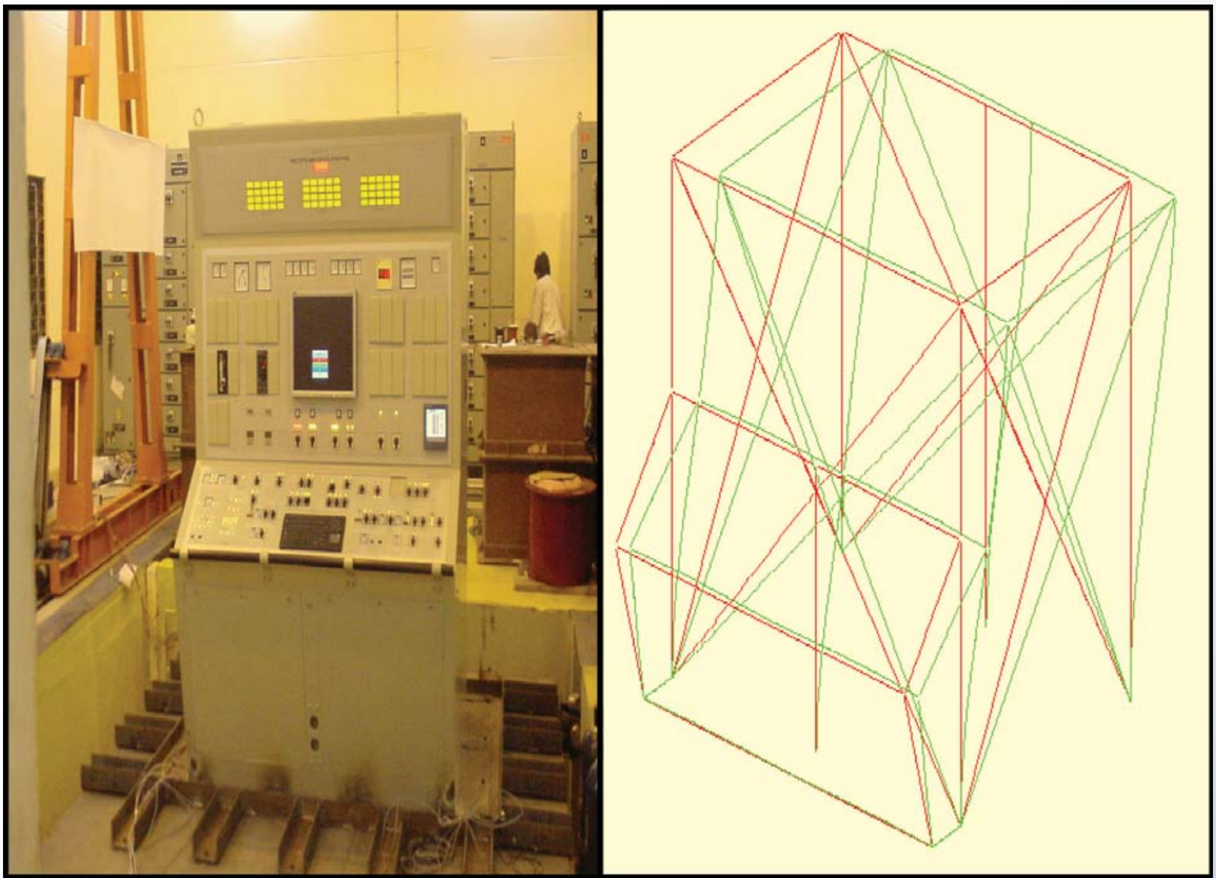


Fig.5 (a) Control Panel (b) Deformed shape of idealized model the under tri-axial excitation

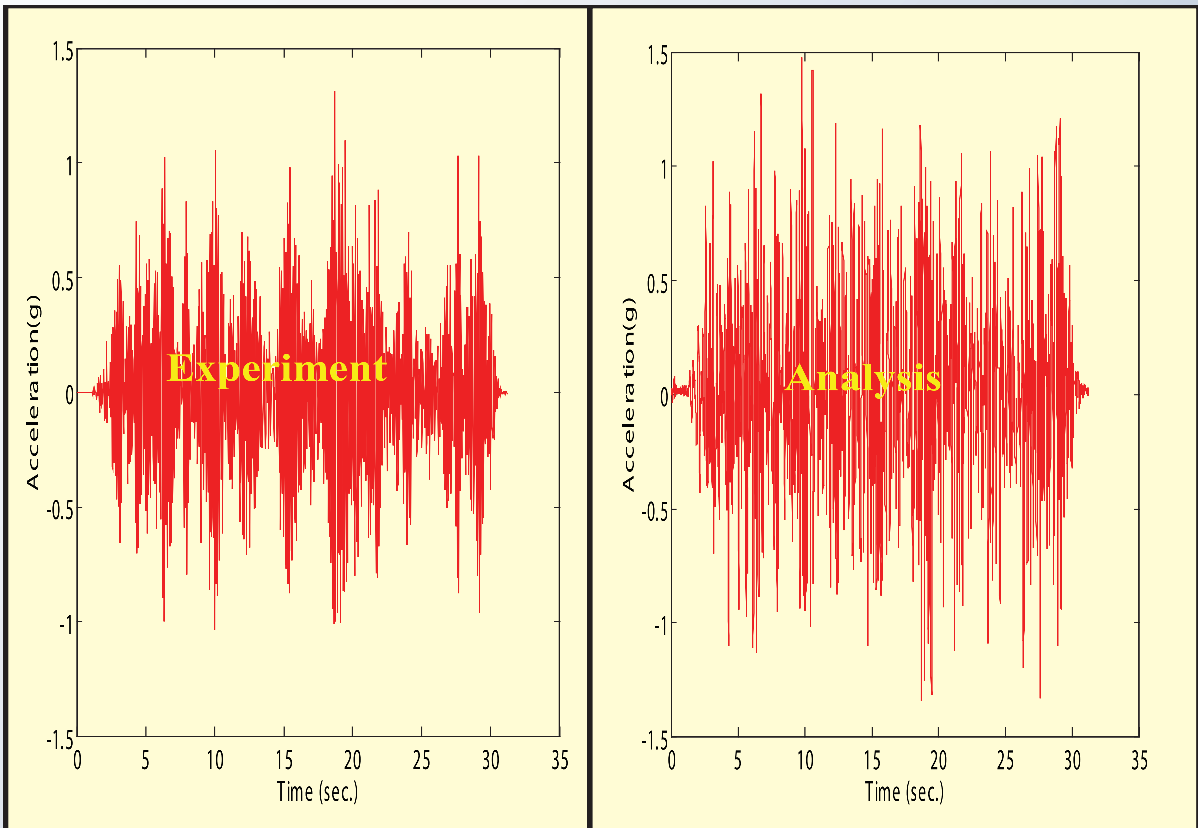


Fig.6 Time-histories of the response of panel

In this procedure of modeling, a natural frequency of the panel is needed for finding the brace dimensions. Natural frequencies can be obtained from modal testing methods like normal-mode or transfer function method. It is seen that the shear area of minor components contribute very little towards the stiffness of panel in the lateral direction. Brace cross-section area obtained by comparing with the test result is only 2.5% of the total shear area of minor components. Hence if one can take the brace area equal to 2-3% of the shear area offered by thin plates or minor sections, it should give a reasonable approximation.

FEA code CASTEM was used for the analysis of a control panel. Seismic analysis of the control panel was performed using Response Spectrum Analysis (RSA) as well as Time-History Analysis (THA) for OBE and SSE excitations. The time-history of the response from experiment and analysis is shown in Fig.6.

Model with brace and beam member not only satisfactorily predicted the maximum responses like displacement, acceleration, and stresses developed in the panel, but also Power Spectral Density (PSD) of the response is in good agreement with the test results as shown in Fig. 7.

DEVELOPMENT OF A PROCEDURE FOR ACCOUNTING MULTIPLE SUPPORT EXCITATIONS

In nuclear power plants, many components are supported at multiple locations of the building structures with non-uniform seismic excitations applied at each of these locations and anchor points. For example, a typical piping system may span different buildings and be anchored at several different elevations as shown in Fig. 8

One way of accounting for the multiple support excitations (MSE) is by enveloped response spectrum, which usually gives highly conservative results. Another approach of accounting MSE is based on various combination of response from different supports excitations.

To implement the procedure based on combinations of responses, an FEM module was written in MATLAB, which can perform static as well as seismic analysis using response spectrum method. A few pipeline configurations were analyzed for MSE with the FEM module and results for different combination rules like SRSS, algebraic, absolute sum etc. were validated with the piping analysis code CAESER, matching with which was found to be satisfactory .

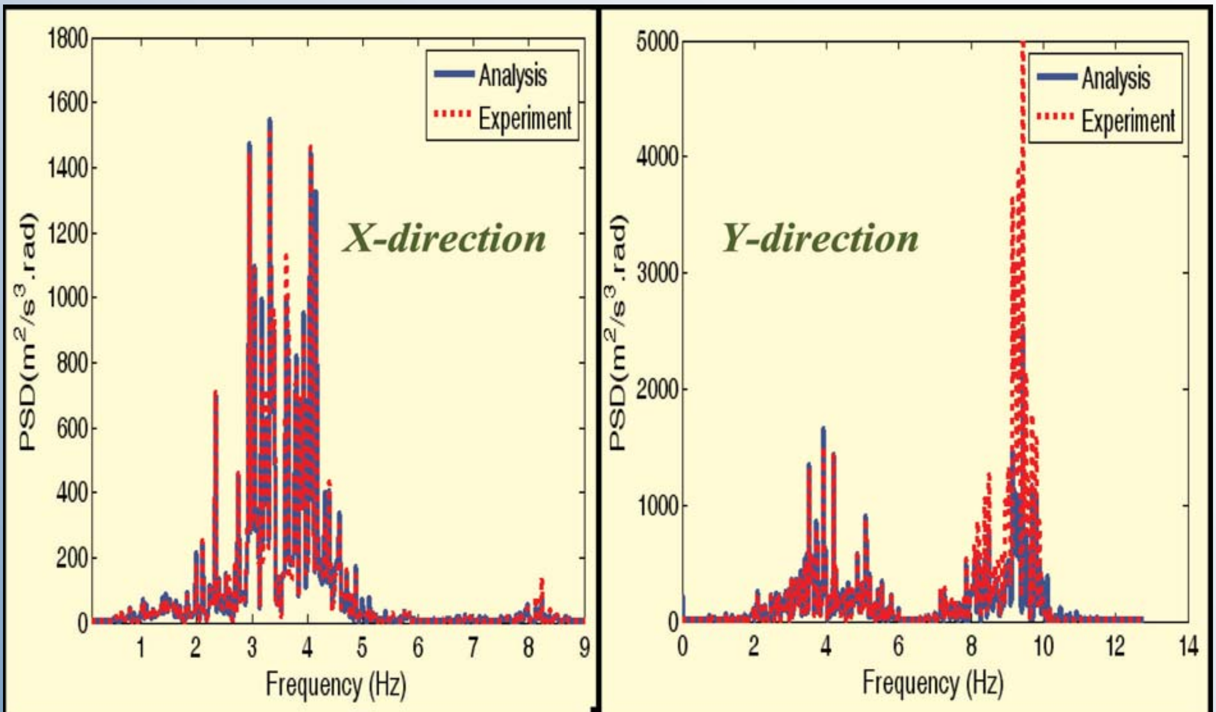


Fig.7 PSD of Response (acceleration) of panel from Experiment and Analysis

The above mentioned method was further used for the configurations shown in Fig. 9, which represent typical primary and secondary structures in a plant. MSE gives lower values of forces and moments in comparison to enveloped response spectrum in almost all the cases as expected. But the degree of reduction of forces and moments varies with the geometric configuration of the system.

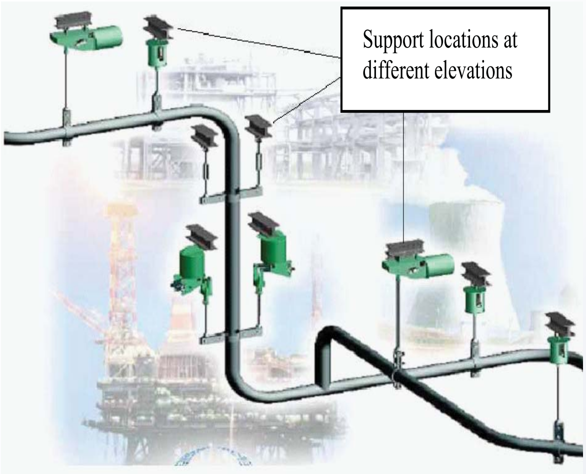


Fig. 8 A typical pipeline supported at different elevation

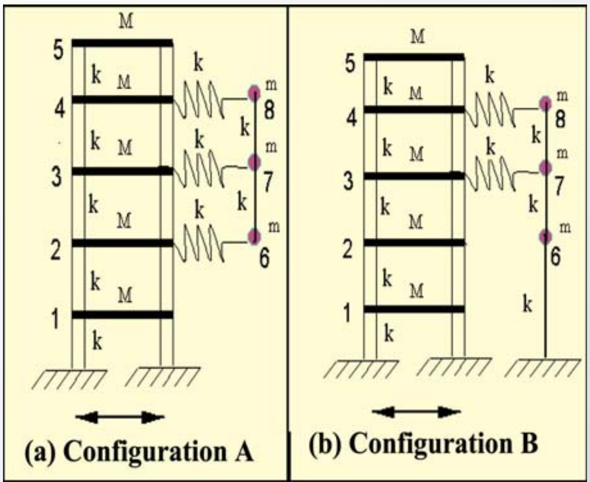


Fig.9 Configuration of system analysed for Multiple Support Excitation

3.1(d) FIRE MODELING AND THERMAL HYDRAULICS STUDIES

FIRE MODELING STUDIES

Introduction

Fire safety in nuclear installations is an important safety issue and is accorded a high priority by AERB. In order to develop strong in-house research capabilities that can aid regulatory process, fire safety is identified as one of the key areas of research in SRI. The activities initiated under fire safety studies are as follows:

- Development of a comprehensive and well validated computational tool for fire hazard analysis of nuclear facilities.
- Design of experimental Enclosure Fire Test Facility (EFTF) for validation of the developed computational model.

Numerical Studies in Fire Modeling

The requirement of a well-validated fire-modeling tool for fire hazard assessment of nuclear power plant / facility is a long felt need. At SRI, a CFD code called 'Fire Dynamics Simulator (FDS)' is being assessed for its suitability to model various enclosure fire scenarios that can occur in nuclear facilities. The focus is on the following two aspects: (1) Its capability to predict over-ventilated and under-ventilated fire scenarios and (2) Its ability to assess the effectiveness of heat and smoke detectors, sprinkler as well as suppression systems.

A few numerical studies have been completed to address conventional fire scenarios, like oil/solvent fires in enclosures for well ventilated as well as under-ventilated conditions. Fig.1 shows the enclosure configuration and dimensions used for one such study. The enclosure is initially kept at a negative pressure of 3 inches of water column. A pool of solvent (n-Dodecane) spread on the floor within a rectangular area of 0.8 m x 0.5 m^2, against one sidewall, is the combustible material. Fire is initiated at time t=10 minutes when steady operating conditions are attained.

Quantitative estimates of the enclosure fire behavior were obtained for various sequence of operation of the ventilation system components as shown in Table.1 (times indicated are subsequent to occurrence of the fire). Fig.2 shows the typical enclosure pressure transients. Apart from providing insight into the FDS code,

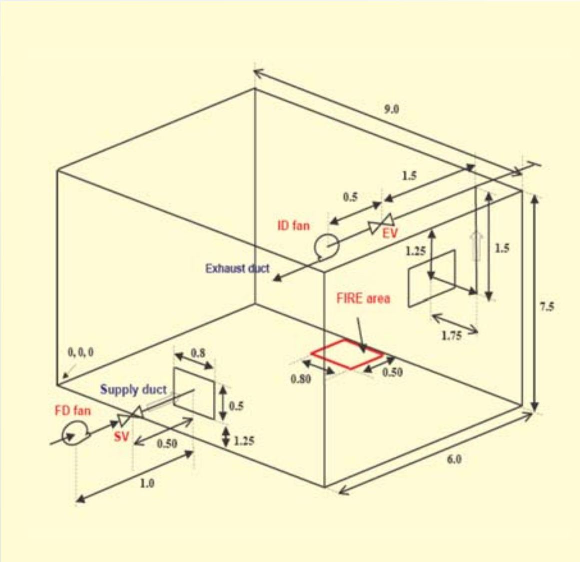


Fig.1 Schematic of the enclosure (dimensions in metres)

Table.1 Control schemes

Scheme. No.	Supply valve (SV) status	Exhaust valve (SV) status
V1	OPEN	OPEN
V2	CLOSED: 2 min OPEN: 22 min	OPEN
V3	CLOSED: 2 min OPEN: 22 min	CLOSED: 4 min OPEN: 15 min
V4	CLOSED: 2 min OPEN: 22 min	CLOSED: 2 min OPEN: 15 min
V5	CLOSED: Instantaneously OPEN: 22 min	CLOSED: Instantaneously OPEN: 15 min

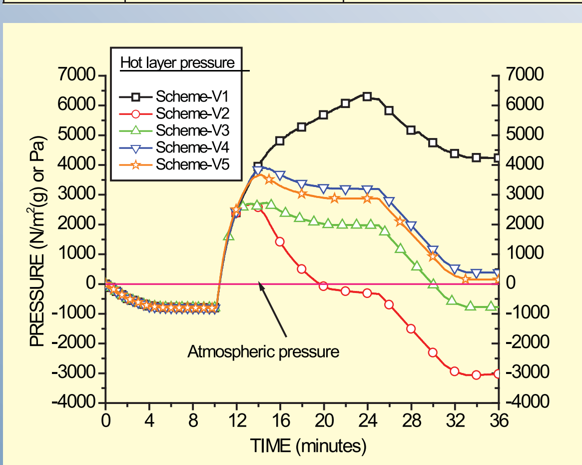


Fig.2 Pressure transients

this study highlighted the features of ventilation based fire control strategy and the flexibility of operation that might be required to minimize the consequences of a fire.

Efforts were also made to evaluate the predictive capabilities of FDS code pertaining to detection and application of fire extinguishing media like water sprays and injection of CO_2.

These studies revealed a number of weaknesses of FDS mainly in the area of modeling of the fire spread process and extinction. With a view to overcome some of these limitations and to enhance the predictive capability, suitable modifications to this code are being identified. For the purpose of validation of the code, an experimental programme is being taken up in collaboration with IGCAR, Kalpakkam.

Design of Enclosure Fire Test Facility

An experimental facility called Enclosure Fire Test Facility (EFTF) is being planned. A series of enclosure fire experiments for different types of conventional fires like oil/solvent fires and cable fires will be conducted to generate scenario specific database.

The test enclosure and its exhaust hood conform to ASTM E-603 (2006) standard configuration. The minimum number and type of instruments will also be in accordance with that specified by above standard. The facility will be constructed of fire barrier panels with adequate fire resistance rating. It will have provision for horizontal and vertical partitioning of the enclosure to obtain multi-compartment configurations. Instrumentation for measuring pressure, temperature, heat flux, gas flow, species concentration, fuel mass loss rate, flue gas analysis etc will be provided. A dedicated data acquisition system (DAS) will record the above parameters during the course of the experiment. The limiting fire size is being assessed. The basic approach for designing the enclosure boundaries for desirable performance is to predict enclosure fire transients for various fuel pool sizes, pool location, quantity and type of fuel using a computational tool and subsequently using this information to establish the critical conditions for design.

THERMAL HYDRAULICS STUDIES ON HYDROGEN GENERATION TRANSPORT & DISTRIBUTION

Introduction

Generation and distribution of hydrogen during a severe accident in a reactor is an issue of prime concern to AERB and hence an area of research at SRI. The emphasis is on numerical studies backed up by suitable experiments. Numerical studies are being conducted related to hydrogen generation in the reactor core, its transportation in the primary heat transport (PHT) system and subsequent release and distribution in the containment. To achieve these objectives, Lumped Parameter (LP) code ASTEC is being used. For CFD simulations, codes available at IGCAR and IIT Madras are being used. Efforts are on to develop an in-house CFD code to simulate hydrogen distribution in enclosures in the presence of steam condensation and engineered safety features.

Hydrogen generation during accidents in LWRs

Computational tool ASTEC (Accident Source Term Evaluation Code) that has been developed by IRSN of France and GRS of Germany is presently being used for accident analysis in LWR. The study includes analysis for accident situations such as Station Black Out (SBO), Small and Large Break LOCA events and many other accident situations. The study aims at extending our understanding of the outcome of these events by using the additional models available in this code. Hydrogen generation and transport in PHT is included in these studies.

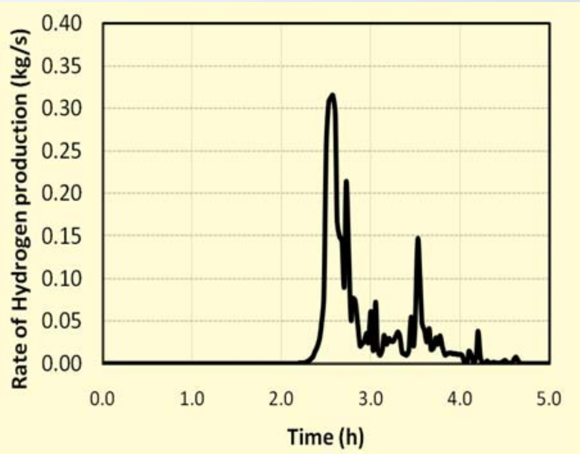


Fig.3 Hydrogen generation with time (ASTEC result)

Fig.3 shows the hydrogen generated due to metal water reaction at elevated temperature for the case of SBO and unavailability of decay heat removal system for a PWR900 reactor.

Numerical studies on hydrogen distribution

A good CFD code for containment thermal hydraulics and hydrogen distribution studies should adequately address phenomena like natural convection, thermal and density stratification, condensation, turbulent dispersion, buoyancy/inertia driven jets and entrainment of ambient gases in the jet/plume.

In order to assess the capability of a commercially available CFD code to model these phenomena, simulations were done for hydrogen distribution following release as a high velocity jet from a circular nozzle as shown in Fig.4. The above geometry is part of an international standard benchmark problem under the 6th framework research programme of the European commission for hydrogen safety (HySafe programme).

The results were compared with those from nine different codes and several issues related to numerical modeling of the associated phenomena were identified. During the course of the study, it was learnt that fine grid near the nozzle and very small time steps are required to adequately resolve the jet during the release phase. Fig.5 shows the difference in stratification levels for release of the same quantity of hydrogen from nozzles of different diameters.

It was found that there is a highly convective region near the hydrogen jet and ambient air gets entrained and mixed with hydrogen. Impingement of the jet on the wall causes wall jets and enhances the entrainment. Natural convection sets in due to non-uniform density and the mass diffusion is initially turbulent and perhaps becomes laminar over a long period of time.

Although currently, CFD codes are able to model basic flow of gases in a confinement, they do not have models for steam condensation. At best, a user coding can be used to implement simple models for condensation.

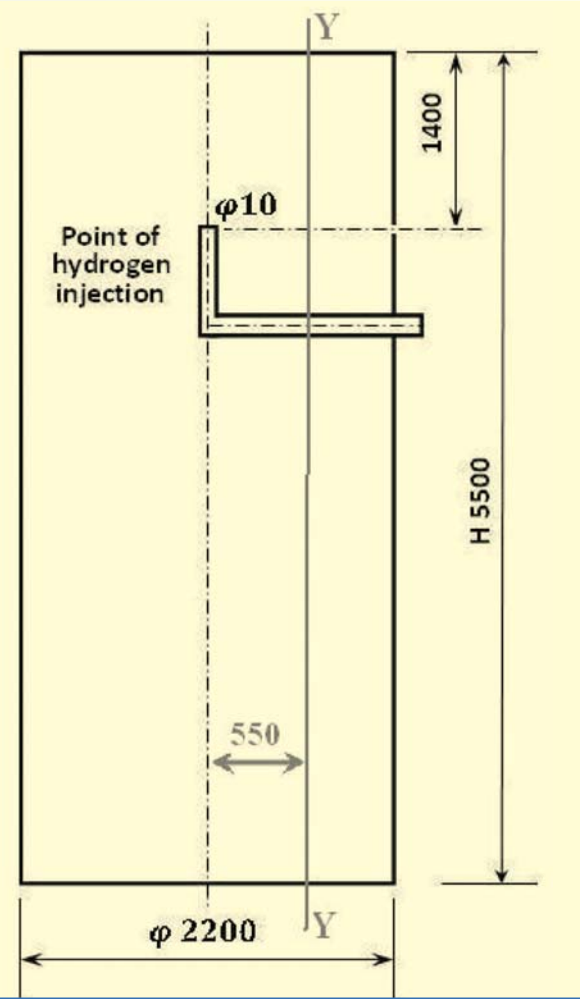


Fig.4 Geometry of setup for hydrogen distribution (dimensions in mm)

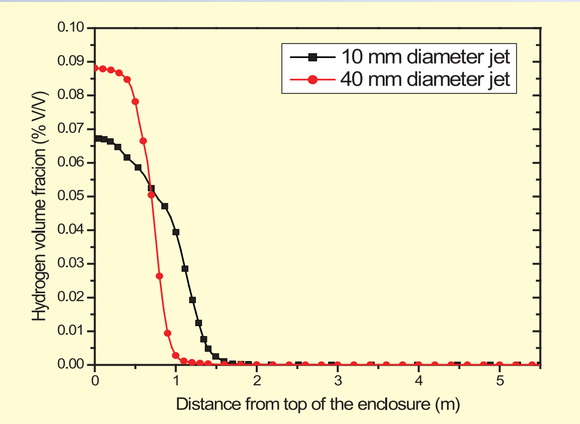


Fig.5 Hydrogen volume fraction vs distance from top of the enclosure

Numerical study of jets

Owing to the limitation of CFD packages in addressing the effect of steam condensation on hydrogen distribution, a finite volume code is being developed at SRI. This code will attempt to solve hydrogen distribution problems from first principles and will comprise of necessary models for steam condensation and interaction of jets/plumes with structures. In its present form, the code is capable of simulating the release of laminar vertical jets in a compartment. Fig. 6 shows a contour plot of vertical velocity for a plane laminar jet, obtained using the code.

For the purpose of validation, the dimensionless velocity profiles and centerline velocity at different axial locations have been compared with the experimental results. The numerical results agree reasonably well with experimental data.

Turbulence models are being incorporated in the code. This code will then be upgraded to solve a set of equations for multi-species and multi-phase jet problems and to give the mass and energy transfer associated with interaction of a steam hydrogen jet with structures in an enclosure.

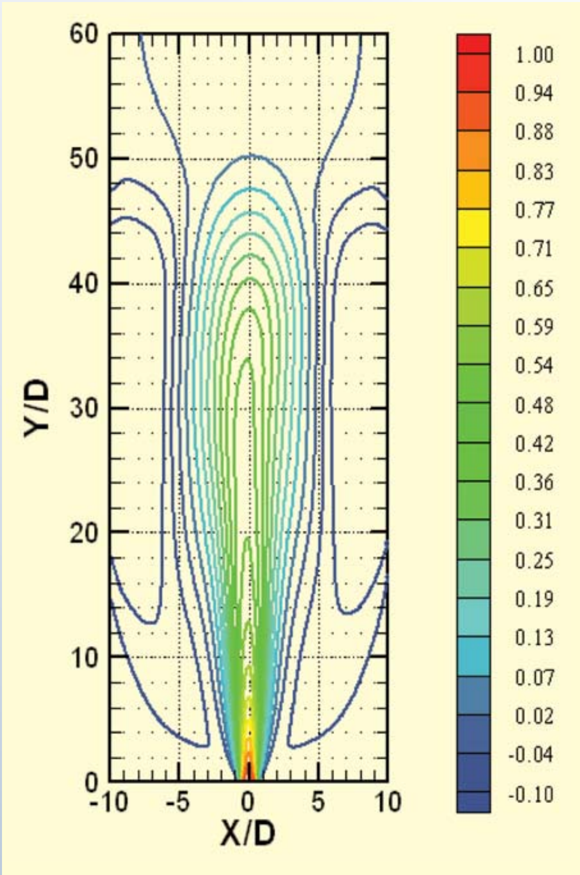


Fig.6 Contour plot for jet of air (Reynolds number of 42).

3.2 ENVIRONMENTAL SAFETY STUDIES

REMOTE SENSING AND GIS STUDIES

Introduction

AERB has setup a Remote Sensing-Geographical Information System (RS-GIS) facility at Safety Research Institute, Kalpakkam whose main objective is to demonstrate the design and organization of a digital database and information system for carrying out Environmental Impact Assessment (EIA) for all the nuclear power plants in the country and to carry out and promote safety related research and analysis in selected areas of relevance to the regulatory function. This programme was initiated as a collaborative effort between SRI and Space Applications Centre (SAC), ISRO, Ahmedabad.

Site Selection and EIA of Nuclear Installations Using RS-GIS Techniques

Methodology for the site selection of NPPs using RS-GIS techniques was established with Kalpakkam as a candidate site (Fig.1). Spatial database on various thematic maps such as drainage, soil, geology, land use/ land cover, geomorphology, etc. were prepared and each theme was assigned weightages according to Saaty's scale of relative importance and a model was generated in GIS environment to do integration analysis using the Eigen Matrix model to select a suitable site and assess its environmental impact for the construction of the Nuclear Power Plants.

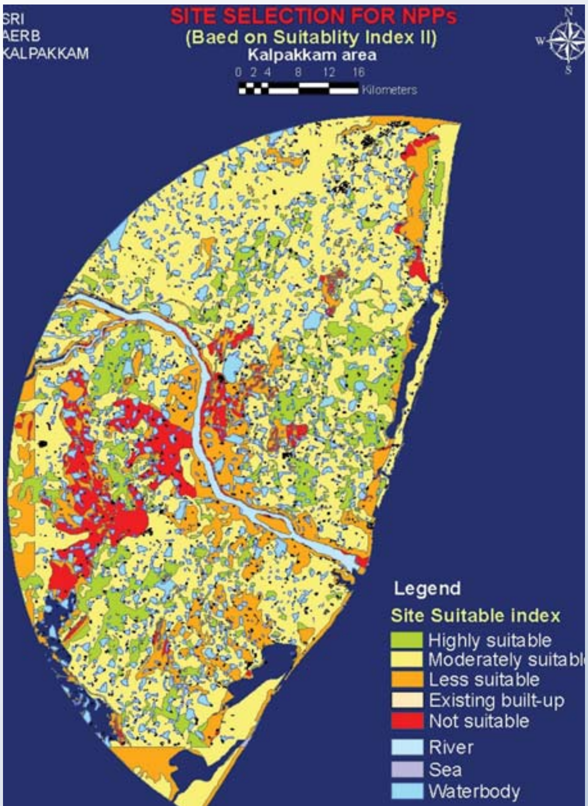


Fig.1 Selection of suitable site for a typical NPP

RS-GIS Inputs to run the MM5 Code and Atmospheric Dispersion Code

Online Emergency Response System (ONERS) has been set up by DAE for Kalpakkam site, in which SRI, BARC, IGCAR and Regional Remote Sensing Centre (RRSSC), Nagpur have participated. SRI provided the RS-GIS inputs in the pre-determined format that are required to create query shells by RRSSC, Nagpur.

An atmospheric dispersion code had been developed by IGCAR to understand the plume dispersion pattern of effluents released from MAPS stack. The required RS-GIS inputs such as village boundary map, satellite image etc. necessary to run the MM5 code and atmospheric dispersion code, have been generated and provided by SRI.

Spatial DSS for Nuclear Emergency

Study on "Spatial Decision Support System (DSS) for Nuclear Emergency" has been taken up. The DSS is capable of displaying the probable path of plume direction, villages to be evacuated and alternative routes for evacuation etc. A Graphical User Interface (GUI) tool for Emergency Response System was created to display the probable path of plume direction and road network data with the background of satellite imageries. Also, a user-friendly query has been built on Plume dispersion pattern using AML (arc macro language) programming in GIS platform.

GIS Based Environmental Radiation Impact Mapping

Study related to the application of GIS techniques to map the radiation impact in Kalpakkam environment (30 km radial zone from the MAPS) was taken up. The environmental radiation data of both external and internal doses collected for the years 1974, 1984, 1994 and 2004 have been interpreted to understand the radiation impact spatially (Fig.2). It was observed that the total dose received by the persons in the zone of interest did not exceed 6 microSv/y as against the allowed limit of 1000 microSv/y.

Impact Assessment of 26th December 2004 Tsunami Surge along Kalpakkam Coastal Area

The tsunami, which struck the east cost of India on 26th December 2004, impacted life and property of Kalpakkam's coastal environment. To study the effect of tsunami on the coastal environment, high-resolution satellite images of pre-dated (19.11.2004) and post-dated (29.12.2004) data were used. Also, aerial images of Kalpakkam Coast of 28th December were employed to demarcate the inundation zones, morphological changes, loss of vegetation and assess the impact in detail on land use/land cover features (Fig.3). Pre and post tsunami bathymetry data was employed to understand the impact of tsunami in near-shore environment.

Simulated (Sea) Water Inundation Model (SWIM)

A SWIM model has been developed to identify different land areas under inundation due to the rise in water levels. Using GRID module in Arc/Info GIS, a query-based model has been developed. In the background, map or imagery of the study area is being displayed and query can be performed over it for each 1m of water level rise using the derived Digital Elevation Model (DEM) based on the logical condition applied on to the grid layer. This DEM was used to evaluate the flood inundation patterns for every 1m level water rise in the vicinity of Kalpakkam Nuclear Power Plant site (50 km radius from NPP). The

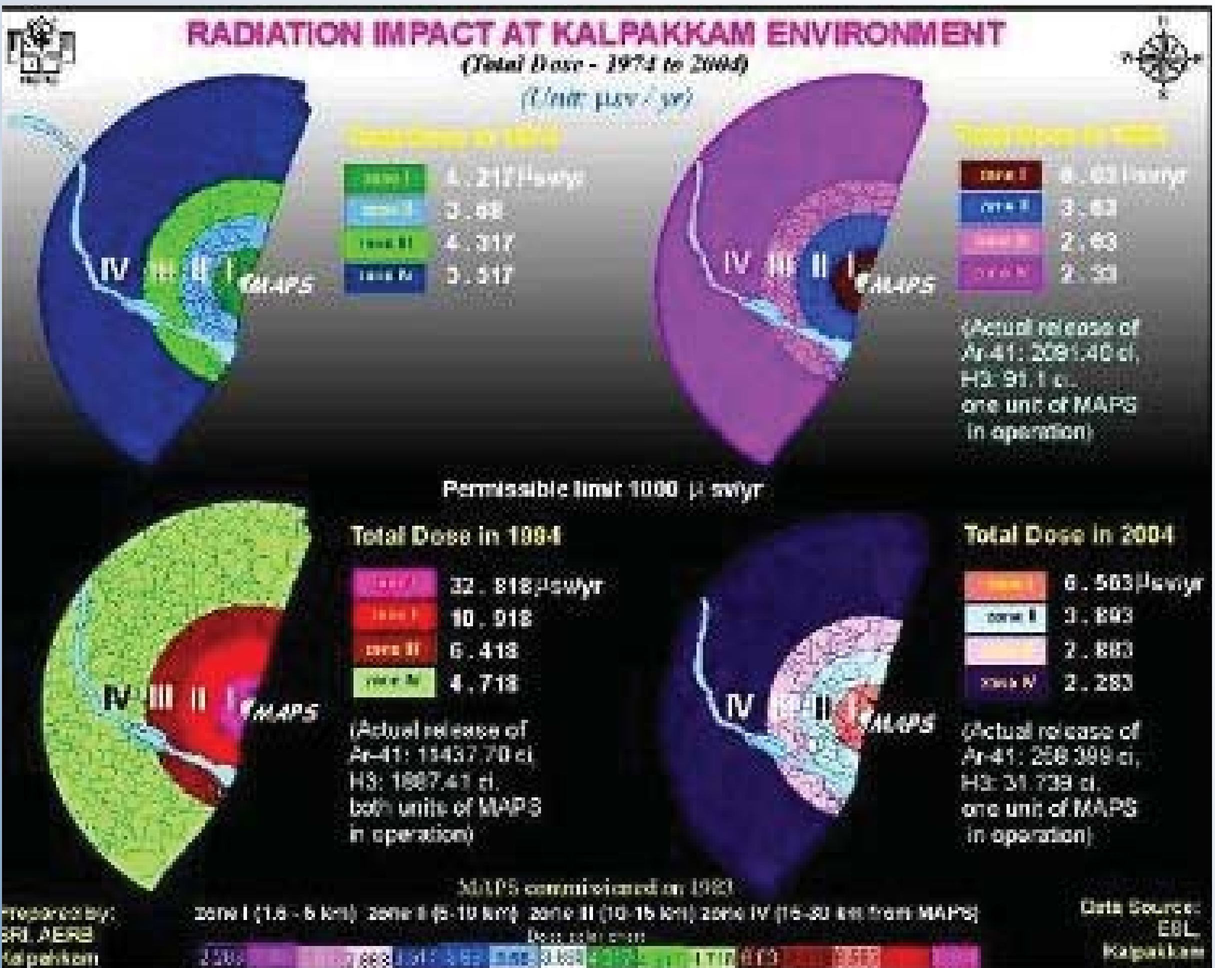


Fig.2 Radiation Impact at Kalpakkam Environment

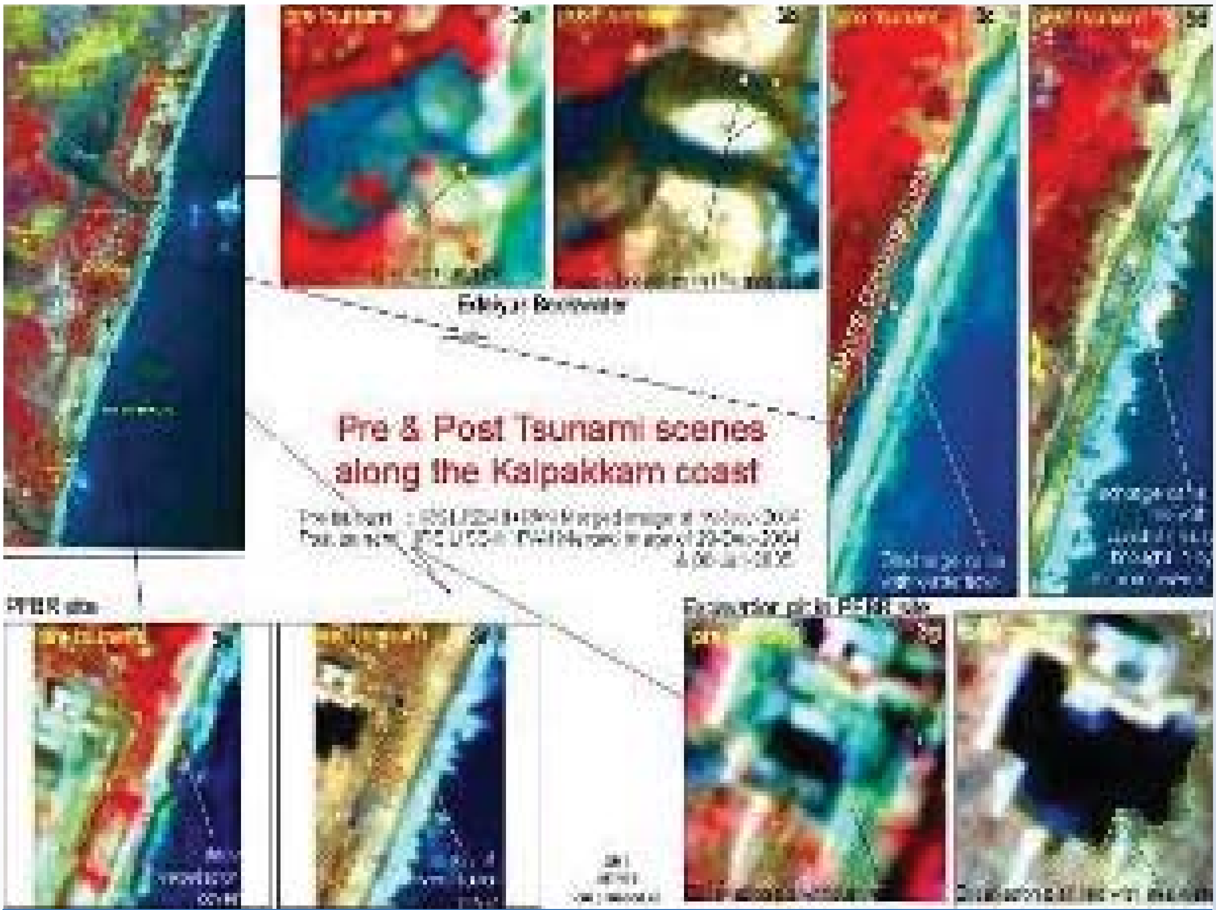


Fig.3 Pre- and post- Tsunami impact scenes along the Kalpakkam Coast

DEM was employed for generating inundation map with 4m, 5m, 6m and 7m water level rise in the study area keeping the satellite imagery in the background.

It was quite evident that the inundation pattern and its impact are not constant along the coast. It varies with respect to different undulating patterns of the terrain, depressions and land cover type and bathymetry. These results are validated with the help of a detailed field survey carried out in the study area.

The experience from this study reveals that it is a cost-effective approach, which can be employed to generate required information fast for suggesting mitigation measures at a gross level. GIS techniques help in integrating multi-parameter spatial formation for generating locale-specific plans.

As SRTM data provides elevation information of buildings and treetops, the inundation simulation using SWIM predicted for 500 m stretch of Casuarinas plantation in the power plant site with rise of wave height by 5m (Fig.4). The simulation showed agreement with ground truth data. From the simulated patterns one could also effectively plan emergency evacuation with available infrastructure i.e. road and rail network.

Mapping of Surface Water Bodies around Kalpakkam Area

A study was carried out to arrive at accurate information on area under surface water bodies around 30 km radial zone from MAPS (Fig.5). Changes in the land use/ land cover of the surface water bodies were also studied. The likely causes have been identified using multi-dated satellite data.

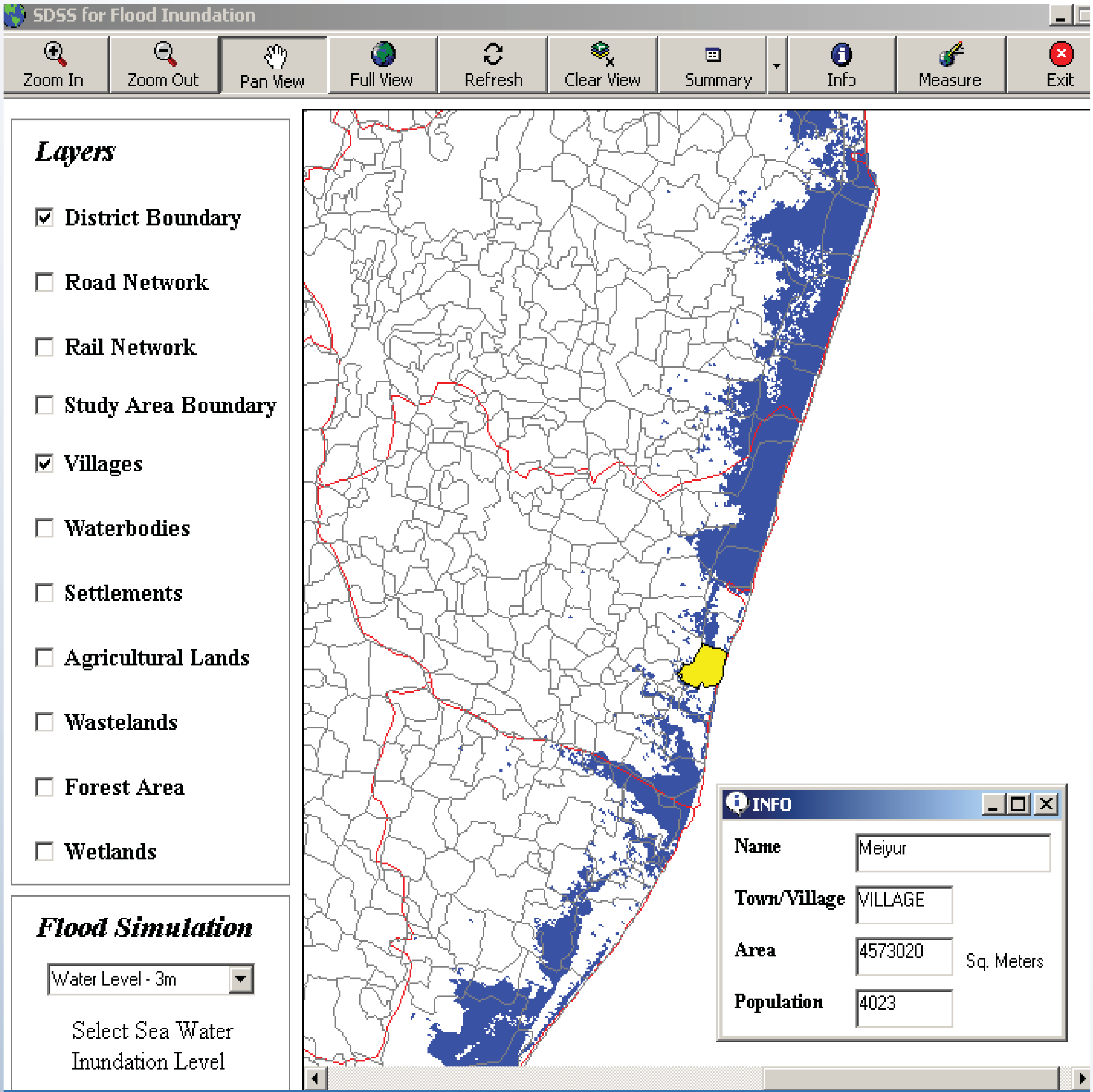


Fig.4 Simulated Sea Water inundation pattern

Fig.5 Satellite data showing land use/ land cover changes in surface water bodies

Spatial Database Creation on DAE Townships for Emergency Preparedness

Updation of spatial data base on the buildings, roads, power lines etc. for Kalpakkam as well as Anupuram townships have been carried out with high resolution satellite imagery such as CARTOSAT-1 and QUICKBIRD-2 by employing GIS software. This is attempted to generate up-to-date database for planning emergency preparedness steps.

PERFORMANCE/SAFETY ASSESSMENT OF NEAR SURFACE DISPOSAL FACILITIES OF KALPAKKAM SITE BASED ON IAEA-ISAM APPROACH

Introduction

The study involves the performance and safety assessment of Near Surface Disposal (NSDF) facilities at Kalpakkam by generating site specific databases, developing release scenarios, constructing radioactivity release conceptual models and analyzing the results to draw conclusions to improve the integrity of disposal facilities. The Safety Assessment of Kalpakkam Near Surface Disposal Facility (NSDF) follows the IAEA-ISAM (Improvement of Safety Assessment Methodology) methodology.

ISAM Approach

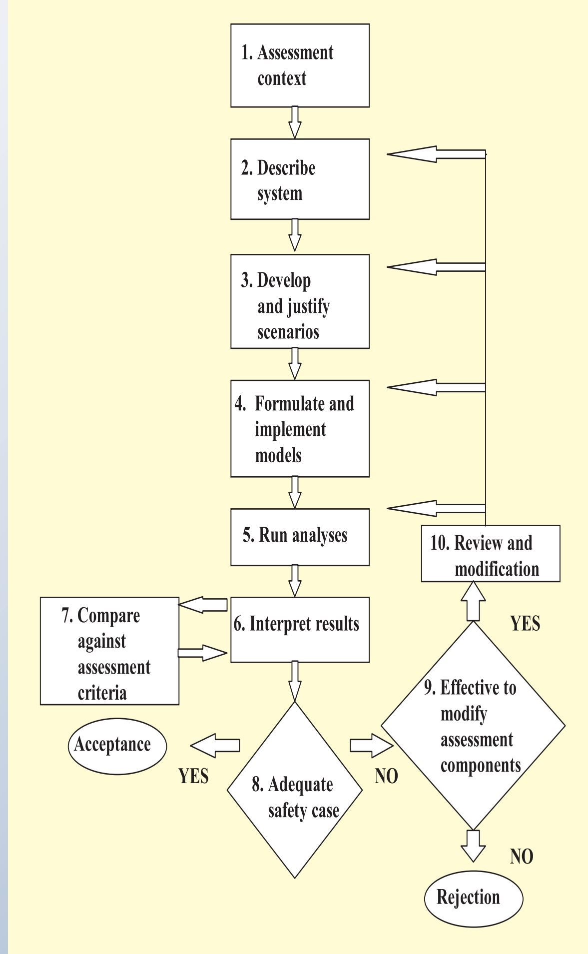
The flow chart for the process steps is given in the flow chart

ISAM methodology steps are:

- specification of the assessment context
- description of the radioactive waste disposal system
- development and justification of scenarios
- formulation and implementation of conceptual and mathematical models and
- Analysis of results.

A common element in many scenario generation methodologies is the initial construction of a list of all Features, Events and Processes (FEPs) that could directly or indirectly influences the disposal system and the migration and fate of radio nuclides within it. These FEPs for normal evolution scenario (design basis scenarios) have been identified after applying certain exclusion criteria for the Kalpakkam disposal system and documented in a systematic way. The exclusion criteria followed for screening ISAM-FEPs (Table.1) The relative importance of each FEP has been reviewed, using expert judgments. The resultant list of FEPs is employed with the system description to formulate scenarios. The scenarios that are identified and quantitatively assessed are mathematically represented.

The chosen FEPs provided the basic guidelines to evolve a site-specific conceptual model. The conceptual model can be employed to formulate and implement numerical / analytical models. The results of the screening of FEPs of Kalpakkam site show that only ground water pathway is the major pathway causing radiological dose to general public. Further studies are in progress.



Flow chart

Table-1: Typical Features, Events and Processes (FEPs) for Radionuclide Transport

RADIONUCLIDE/CONTAMINANT FACTORS	
Features, Events and Process (FEPs)	Remarks
Dissolution, precipitation and crystallisation, contaminant.	Chemical reaction of vault material with pore water.
Speciation and solubility, contaminant.	Solubility change caused by chemical interaction between waste and pore water has to be taken into account.
Sorption/desorption processes, contaminant	Chemical interaction of waste with pore water has to be taken into account.
Colloids, contaminant interactions and transport	Colloid facilitated transport of radio nuclides through ground water has to be taken into account.
Chemical/Complexing agents, effects on contaminant speciation/transport	Effect of chelating agents has to be taken into account.
Transport of contaminants through Water pathway	Ground water transport has to taken into account.
Food chains, uptake of contaminants in	Agricultural lands west of disposal facility.
EXPOSURE FACTORS	
Intake of contaminants through drinking water, foodstuffs and drugs,	Contaminants enter into human body through drinking water, foodstuffs and drugs only, as atmospheric transport of contaminants is considered to be very negligible.
Exposure modes	Only ingestion route.

HYDRO-GEOLOGICAL INVESTIGATIONS AT KALPAKKAM

Introduction

Radioactive wastes after appropriate immobilization are disposed of in underground environment. In view of this, radionuclide release scenarios from shallow ground disposal facilities must have adequate knowledge of hydrogeology of the site to predict the groundwater flow, direction and contaminant migration. Towards this, efforts are underway to develop an aquifer model based on the hydrogeological and geochemical investigations.

Groundwater Modeling

The periodical (monthly) monitoring of watertable fluctuations, groundwater sample collection and analysis for ionic concentration were carried out. A steady increase in watertable was noted due to the recharge by rainfall during the months of August-September. The annual watertable fluctuation chart showed gradual rise in watertable due to the recharge during the months of July to November and the water table was declining during the period December to June.

Geophysical Survey

The resistivity survey is a geophysical technique to interpret the lithological boundaries (aquifer

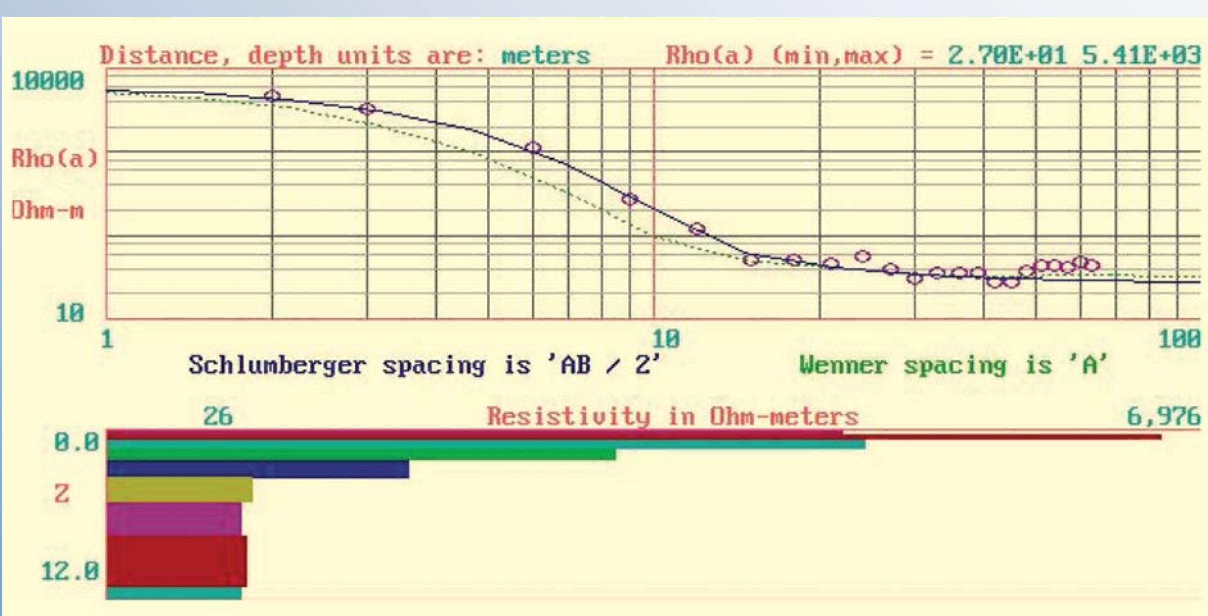


Fig.1 Subsurface lithological classification using Resistivity survey

characterisation) from the resistivity variations with respect to depth. The survey was carried out in five different stations. The resistivity survey indicates that the area is characterized by three distinct litho units viz. sandy layer with clay pockets (4 m thickness) overlying weathered and fractured rock of thickness 11 m (Fig.1). This weathered and fractured layer is underlined by massive charnockite beyond 15 m.

A groundwater flow model was generated using MODFLOW (MODular three dimensional finite difference groundwater FLOW model, developed by USGS (1988)) with Visual MODFLOW Pro graphic user interface. The contouring package, SURFER version 8 was used to generate the grid data for topography and initial head.

A single layer groundwater flow model for April 2006 for steady state (Fig.2) and transient state for April 2006- December 2006 have been generated for Kalpakkam plant site, considering the sandy and weathered formations as a single homogenous unit layer. The topography varies between 6 to 12.5 m above MSL

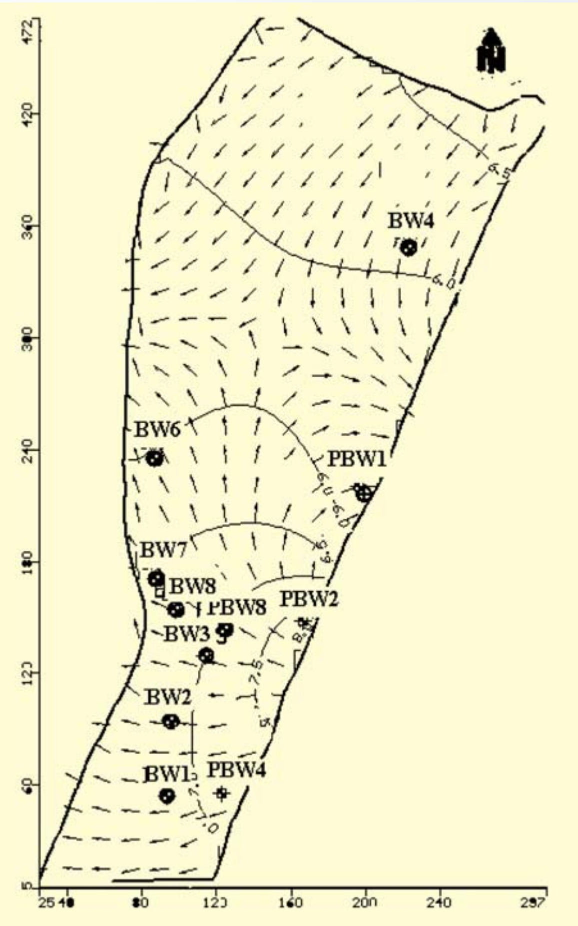


Fig.2 Groundwater flow direction (April 2006) (Single Layer)

The hydraulic conductivity values assigned were Kx=3.2E-8 m/s, Ky=3.2E-8 m/s and Kz=3.2E-9 m/s. The specific storage of the aquifer (Ss) = 2.38E-4. The April 2006 watertable was assigned as the initial head condition. The constant head boundary was assigned along east, west and north direction of the model domain. The model was calibrated by varying the hydraulic properties and boundary conditions within the site-specific conditions to attain a good match between the calculated and real field head values. Watertable heads at 12 out of 14 bore-wells fall within 95% confidence level. A good correlation between the simulated and field measured values was obtained with a correlation coefficient of 0.942

Groundwater Chemistry

The pH, temperature, electrical conductance (EC), salinity and total dissolved solids (TDS) were measured in situ using portable bore well logger (Multi probe system, YSI 556 MPS).

The samples collected were analyzed in the laboratory for Ca^{2+}, Mg^{2+}, CO_3^{2-}, HCO_3^{2-} (titration); Na+ (flame photometry); Cl^-, NO_3- and SO_4^{2-} (spectrophotometry and ion-chromatography). The groundwater chemistry

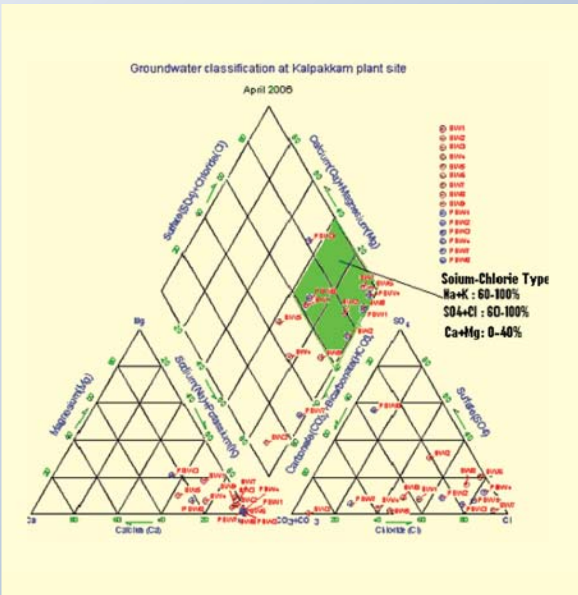


Fig.3 Piper trilinear diagram showing groundwater types at Kalpakkam site

Table:2 Groundwater Characterization

Parameter	pH	TDS	EC µs/cm	Ca^{2+} ppm	Mg^{2+} ppm	Na^{+} ppm	CO_3^{2-} ppm	HCO_3^{2-} ppm	Cl^{-} ppm	NO_3^{-} ppm	SO_4^{2-} ppm
Minimum	7.1	116	200	0.99	0.5	20	0.5	25	20	0.4	0.5
Maximum	8.7	11850	20533	35	99	5844	33	606	5720	286	1854
Mean	7.7	2012	3489	14	13	1026	11	259	751	113	219
Std. Dev.	0.3	3060	5322	11.03	24	1604	14	197	1479	72	475

data are presented in Table-2. The chemical analysis of the water samples has indicated that the most dominant ions include magnesium (Mg^{2+}), sodium (Na^{+}), chloride (Cl^{-}), sulfate ($SO4^{2-}$) and nitrate (NO_{3-}).

The Piper trilinear diagram, plotted to classify the groundwater revealed that the groundwater of Kalpakkam nuclear plant site is of Na-Cl +$SO4^{2-}$ type and Na - $CO3^{2-}$+$HCO3^{2-}$ type. Bore well samples BW3, BW4, BW5, BW9 and PBW9 have been categorized under carbonate-bicarbonate type. The BW6, BW7& BW8 which lie closer to Buckingham canal and the PBW1, PBW2, PBW3 & PBW4 lying closer to Bay of Bengal fall under chloride type (Fig.3). The presence of higher concentration of magnesium than calcium in the areas of high chloride levels present in the close vicinity of Buckingham canal and Bay of Bengal suggests the saline water incursion.

Groundwater Corrosiveness

For better understanding of the corrosive nature of the ground water in Kalpakkam region, the corrosivity indices like Langeliar Saturation Index

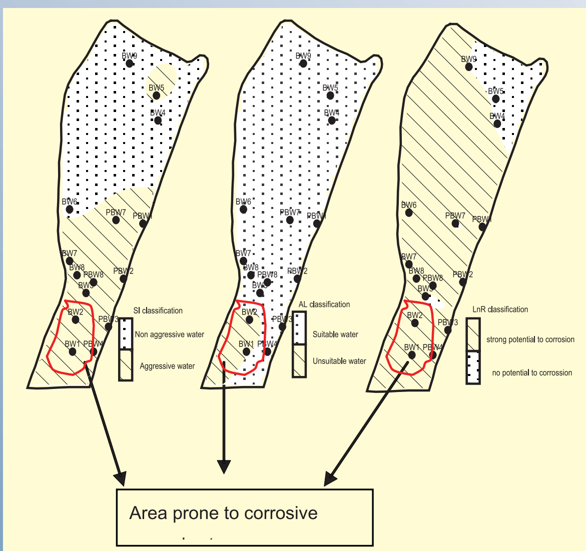


Fig. 4 Corrosive indices of groundwaters at Kalpakkam

(SI), Aggressivity Index (AI) and Larson Ratio (LnR) were employed. According to the above corrosive indices, the groundwaters of southern part of the study area are highly corrosive in nature. Presence of higher concentrations of chloride (Cl-) and sulphate ($SO4^{2-}$) makes the groundwater more corrosive. The saline water incursion along the seacoast and Buckingham canal could be source of supply of these ions into the groundwater system (Fig.4). Proper precautionary steps have to be taken during sub-surface civil constructions particularly in the southern part of the study area.

COLLOIDAL TRANSPORT OF RADIO NUCLIDES IN GROUNDWATER

Introduction

Study of colloid-facilitated transport of radionuclides in sub-surface is being taken up at SRI. The objective of the study is (1) to understand the role of colloids on the transport of fission products cesium and strontium by carrying out detailed characterization of colloidal particles for its size, size distribution, morphology, elemental composition, and zeta potential (2) to understand the geochemical processes occurring during sorption.

Groundwater samples were collected from bore wells in the study area and their characterization with respect to physicochemical parameters were carried out. Turbidity was monitored in the field using a Hach portable turbidity meter. Two groundwater types were identified in the study area (1) freshwater type and (2) brackish type based on the salinity levels in the groundwater samples.

Turbidity Variation in the Study Area

Turbidity measurement provides a qualitative insight into colloid concentration. The turbidity of groundwater samples varied between 0.38

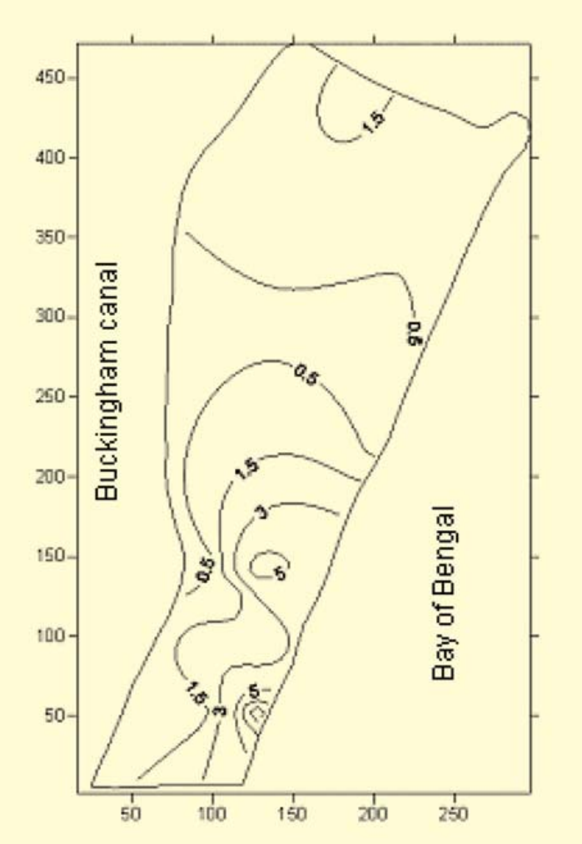


Fig. 1 Groundwater turbidity contours at Kalpakkam

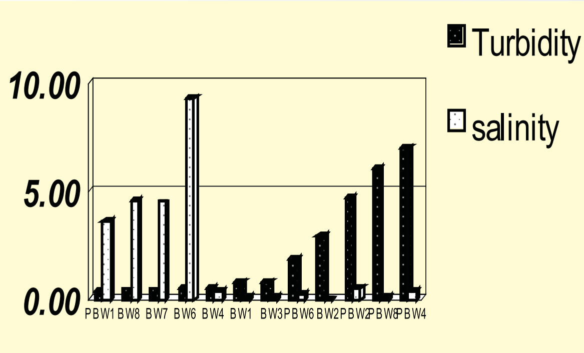


Fig.2 Variation of Turbidity with Salinity

NTU and 7.0 NTU.(Fig.1) The turbidity values are less in the region near Buckingham canal where the groundwater is saline in nature (Fig.2).

Groundwater Colloid Characterisation

The groundwater colloids were characterized for its size, size distribution, morphology, elemental composition and zeta potential using techniques such as photon correlation spectroscopy, particle size analyzer, scanning electron microscopy, energy dispersive X-ray analysis, X-Ray diffraction technique (Fig.3) and zetasizer. The colloid concentration varied between 0.05mg/L and

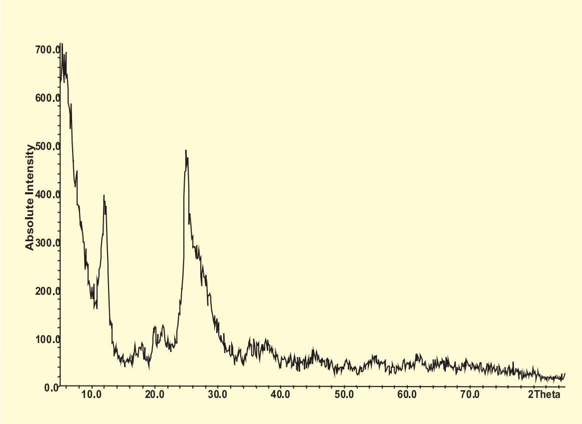


Fig.3 XRD analysis

6mg/L. The number of particles was found to vary between 3 X 10^9 to 3 X 10^{11} particles/L. Colloid concentration was less in bore wells that are near to saline water bodies. The average colloid size varied between 200nm and 350nm for various samples. The zeta potential of the colloidal particles varied in the range - 25.5 mV to - 34.0 mV. SEM analysis of colloidal particles revealed the presence of clays particularly kaolinite (Fig.4).

Separation and Characterisation of Clay Colloids from Soil Samples

To determine whether the mineralogy of colloids is reflective of aquifer mineralogy, clay samples were collected from different locations within the plant site and colloidal clay fractions were separated. Characterization of clay colloids for its size, shape and XRD (Fig.5) and SEM and EDX analysis (Fig.6), and zeta potential were carried out. Analysis has shown that clay minerals composed predominantly of kaolinite. The size of clay colloid is 235 nm and zeta potential is -34.0 mV .

GROUNDWATER GEOCHEMICAL MODELING

Introduction

Kalpakkam being a coastal aquifer is being surrounded by water bodies and contains salt pans, back waters etc. Installations like FRFCF, WIP are coming up on the western side of Kalpakkam site and it is essential to understand the geochemistry of Kalpakkam site. Also, the knowledge about groundwater geochemistry is

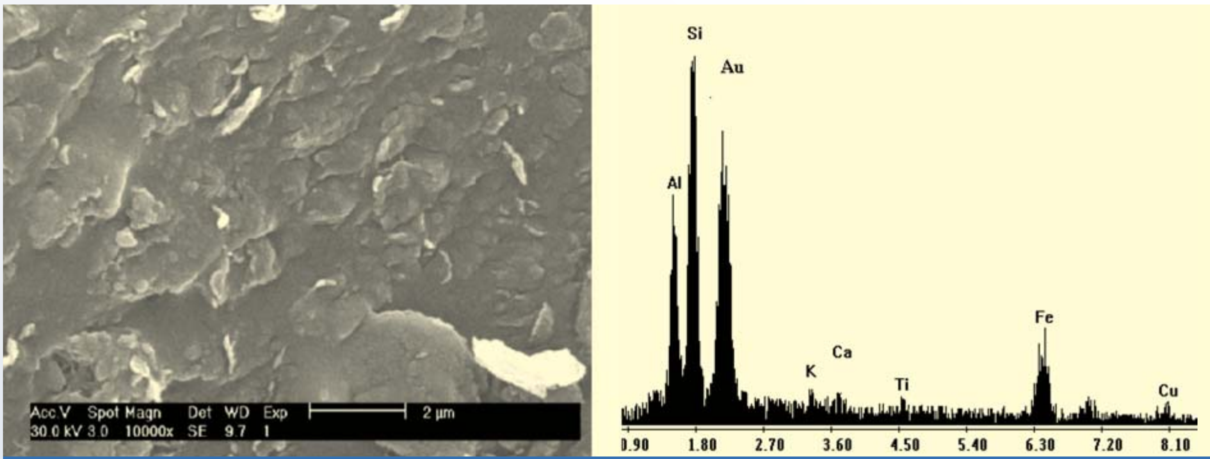


Fig.4 SEM/EDX of groundwater colloids

essential for finding out the possible minerals that can be formed in that zone.

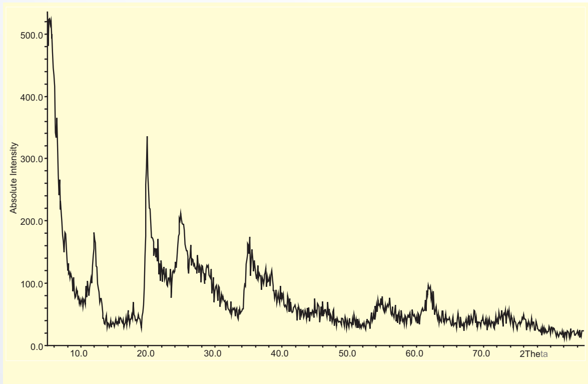


Fig.5 XRD analysis of field clay samples

Geochemical Modeling Using PHREEQC

For this study, water samples are collected from various bore wells located around Kalpakkam plant site every month covering the entire span of seasons (NW monsoon, SE monsoon, Post monsoon and Summer). Samples collected are analyzed for various anions (Cl^-, SO_4^{2-}, CO_3^{2-}, HCO_3^-, PO_4^{3-}, NO_3^-, F^-, SiO_4^{2-}) and cations (Ca^{2+}, Mg^{2+}, Al^{3+}, Fe^{3+}, Na^+, K^+ etc) at SRI laboratory.

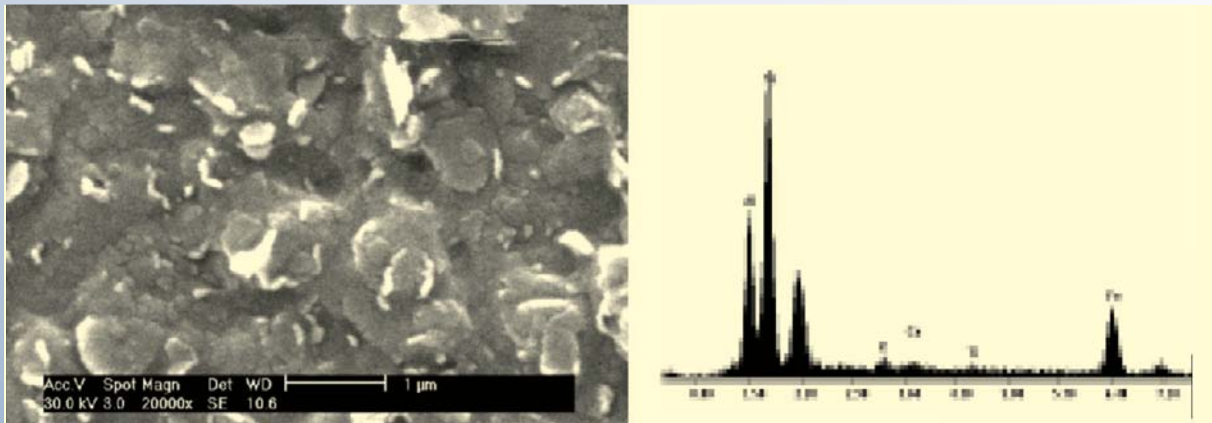


Fig.6 SEM / EDX of field clay samples

The analytical data are interpreted by mixing model , PHREEQC software that helps in the study of mixtures of waters to interpret the analytical results.

Initial studies have shown that Kalpakkam site has predominantly minerals like calcite, aragonite and dolomite. The salinity and chloride levels are found at the bore wells that are located near Buckingham canal and PFBR site. Carbonate and bicarbonate minerals are the dominant anions whereas calcium, magnesium and sodium are the dominant cations. Calcite and dolomite are the minerals found near Buckingham canal. Charnockite rock constitutes Kalpakkam site. Studies are in progress to find out the other minerals and ions that are specific for Kalpakkam site.

SRI INFRASTRUCTURE

RS-GIS Laboratory

A RS-GIS laboratory has been set up at SRI to carry out Environmental Impact Assessment (EIA) for all nuclear power plants in the country and also to promote safety related research. It was inaugurated on 6.10.2003 by Shri S.K. Sharma, the then Vice Chairman, AERB (Fig. 1).

RS-GIS Laboratory is equipped with the following software and hardware (Fig.2)

Software: ArcInfo GIS v 9.2, Arcview v9.2, ERDAS Imagine Professional v8.1, ENVI v4.3, SURFER v8.0, ORIGIN v7.5
Hardware: A0 Scanner, A0 Plotter, AO Digitizer, Two Pentium IV Servers and Two Client Pentium IV PCs

Environmental Chemistry Laboratory

A well-equipped Environmental Chemistry Laboratory has been established at SRI to cater to the analytical activities pursued. The laboratory was inaugurated on 21.3.2005 by Shri S.K.Sharma, Chairman, AERB (Fig.3).

Analytical instruments and other equipment available in the laboratory include Ion Chromatograph, UV-VIS Spectrophotometer, Turbidity meter, Moisture probe, Conductivity meter, Sonicator, ground water sampling equipment etc (Fig.4).

Fig.1 Inauguration of RS-GIS laboratory at SRI by Shri S.K. Sharma, the then Vice Chairman, AERB, on 6.10.2003

Fig.3 Inauguration of Environmental Chemistry Laboratory by Shri S.K.Sharma, Chairman, AERB on 21.3.2005

Fig.2 A view of RS-GIS Laboratory

Fig.4 A view of Environmental Chemistry Laboratory

4. OTHER ACTIVITIES

4.1 DISCUSSION MEETS/WORKSHOPS/SEMINARS

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. The following table lists the Discussion Meets/Workshops/Seminars etc. organized in the last ten years:

Sl. No.	Topic Of The Discussion Meet/Workshop/Seminar	Date
1.	Gamma Ray Shielding	Feb 20-24, 1999
2.	Probabilistic Safety Assessment	Aug 6-7, 1999
3.	Feedback Experience on Safety Related Unusual Occurrences and Adherence to Technical Specification for Nuclear Plants	Dec 9-10, 1999
4.	Monte-Carlo: Radiation Transport	Feb 7-18, 2000
5.	Fire Hazard Analysis and Modeling	Aug 28-29, 2000
6.	ESL Professionals Meet	Jan 23-24, 2001
7.	Computer Based Safety Systems in NPPs	Nov 28-29, 2001
8.	Inter-Institutional Collaborative Research	Jan 30, 2003
9.	EIA of Nuclear Facilities	Oct 6, 2003

Sl. No.	Topic Of The Discussion Meet/Workshop/Seminar	Date
10.	Certification Course on Safety Aspects in Beach Sand Minerals and NORM Industry	Jul 13-19, 2004
11.	Application of Two Phase Flow and Heat Transfer to Indian Nuclear Reactors	Sept 13-15, 2004
12.	Safe Handling of Plutonium	Mar 22-23, 2005
13.	External Flooding Hazards at NPP Sites	Aug 29-Sept 2, 2005
14.	Identification, Reporting and Analysis of Low Level and Near Miss Events	Dec 19-20, 2005
15.	Application of PSA in NPPs: Status and Future Directions	Aug 10-11, 2006
16.	Certification Course on Safety Aspects in Beach Sand Minerals and NORM Industry	Sept 18-19, 2006
17.	Internal Radiation Dosimetry	Jun 20-22, 2007
18.	GroundWater Modeling using Visual MODFLOW and PHREEQC	Aug 29-31, 2007
19.	Fire Modeling	Sept 20-21, 2007 At Mumbai
20.	Emergency Exercises by NPPs-Site and Off site: Challenges and Constraints	Jun 12-13, 2008
21.	Seismic Re-evaluation of Existing Nuclear Facilities	Sept 17-18, 2008

Safety Research Institute

4.2 COMPUTER CODE DEPOSITORY

One of the important activities SRI has taken up is the commissioning of a 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 computer codes.
- 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.
- 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 CODES		
	Acquired	**Developed**
REACTOR PHYSICS & RADIOLOGICAL SAFETY CODES	EXCEL , TRIHEX-FA , SCALE 5.0	ASFIT, DUST, IGSHIELD
PROBABILITY SAFETY ASSESSMENT CODES	RISK SPECTRUM 2.0 , RELEX 2008	--
ACCIDENT ANALYSIS CODES	SOFIRE, NACOM	--
THERMAL HYDRAULICS CODES	ASTEC	THYCON
FIRE HAZARD ANALYSIS CODES	FIRE Dynamics Simulator (FDS), CFAST	--
GRAPHICAL USER INTERFACES	--	GUI2MCNP, GUI2KENO, GUI2QAD3D, View-CXS, GUI2TRIHEX-FA
RS – GIS RELATED SOFTWARE	**GIS SOFTWARE**: ARCGIS 9.2 and ARC VIEW 9.2 **IMAGE PROCESSING SOFTWARES**: ERDAS IMAGIN PROFESSIONAL 8.4 and ENVI 4.3	--
GROUNDWATER MODELING CODE	VISUAL MODFLOW PRO 4.1	--
GEOCHEMICAL MODELING CODE	Equillibrium Chemistry Model: PHREEQC V.2	--

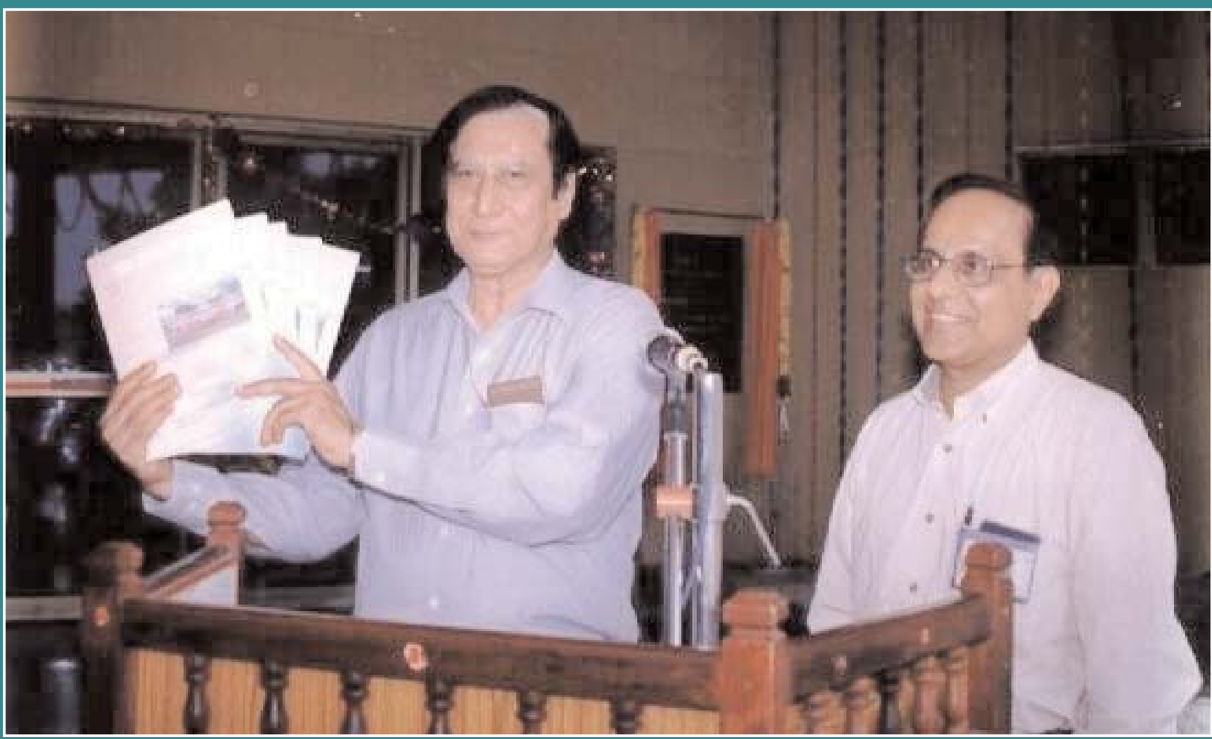
Shri P.K.Ghosh, Shri A.R.Sundararajan, Shri S.B.Bhoje, Prof.Sukhatme and Dr.S.M.Lee during the Discussion Meeting on `FIREHAM` August 28-29, 2000.

Dr.S.M.Lee, Shri G.R.Srinivasan, Shri Umesh Chandra and Shri P.Swaminathan during the Discussion Meeting on `Computer Based Safety Systems in Nuclear Power Plants`, November 28-29, 2001.

Dr.S.M.Lee, Shri S.B.Bhoje, Prof.Sukhatme, Prof.M.S.Ananth and Dr.P.Sasidhar Discussion Meeting on `Inter-Institutional Collaborative Research`, January 30, 2003.

Dr.P.Rodriguez, Prof.Sukhatme, Shri S.B.Bhoje, Shri A.R.Sundararajan and Shri S.K.Chande during SRI Guest House inauguration on January 30, 2003.

Shri S.K.Sharma, Dr.Baldev Raj, Prof.Sukhatme, Shri S.K.Chande and Shri A.R.Sundararajan during inauguaration of SRI Office Building on October 6, 2003.



Shri S.K.Sharma, Vice-Chairman, AERB releasing the SRI Highlights on October 6, 2003.

Dr.Anil Kakodkar, Chairman AEC, delivering the address in the International Workshop on External Flooding Hazards at Nuclear Power Plants, August 29 -September 2, 2005.

Shri S.K.Chande, Dr.Baldev Raj, Shri S.K.Sharma and Smt.Uma Seshadri during the Discussion Meet on `Seismic Re-Evaluation of Existing Nuclear Facilities` September 17-18, 2009

Appendix - I

HUMAN RESOURCES

SRI encourages its staff to acquire additional qualifications, with a view to facilitate qualitative improvement in their capabilities to take up the research assignments. Currently three SRI officers have registered for Ph.D.

SRI also provides opportunities for experts with adequate qualification, experience and expertise to take up assignments in areas relevant to SRI and AERB. At present one Visiting Scientist is working in SRI.

Another feature of SRI is enabling research scholars to take up research work under the guidance of SRI officers. At present six research scholars (five SRFs and one JRF) are working in SRI and five of them have registered for Ph.D.

The following table gives a profile of the SRI personnel.

Shri S. E. Kannan	**Director** **Scientific Officer (H+)**

Name	Designation	Area of work
Dr. K. V. Subbaiah	Head, Radiation Safety Analysis Section (RSAS), Scientific Officer (G) (Ph. D Guide in Madras University)	Radiation transport and shielding computations with SN and Monte Carlo methods; Accelerator safety; Radiation dosimetry
Dr. P. Sasidhar	Scientific Officer (G), (Ph. D Guide in Anna University & Madras University)	Safety assessment of radioactive waste disposal facilities; Groundwater and contaminant transport; Environmental assessment
Dr. C. Senthil Kumar	Scientific Officer (F)	Probabilistic safety assessment; Reliability and software development
Shri Hariharan Seshadri	Scientific Officer (E)	Reprocessing chemistry; Geochemical computational modeling
Shri O. S. Seik Mansoor Ali	Scientific Officer (E)	Fire modeling and containment thermal hydraulic safety studies
Shri Jagannath Mishra	Scientific Officer (E)	Structural and seismic analysis; Structural reliability
Dr. Sunil Sunny	Scientific Officer (D)	Health physics and radiation shielding **(On PDF to Texas A&M University, USA from 16.08.2007 to till date)**
Dr. C. Anandan	Scientific Officer (D)	Application of RS-GIS techniques in nuclear industry
Shri Subrata Bera	Scientific Officer (D)	Reactor physics and dynamics

Name	Designation	Area of work
Dr. C. Gurumoorthy	Scientific Officer (D)	Radionuclide migration in unsaturated zone; Geohydrological investigations **(PDF at Tokyo Institute of Technology in Japan during 12.11.2005 to 11.11.2007)**
Shri Nilesh Agrawal	Scientific Officer (D)	Thermal hydraulics and computational fluid dynamics
Shri Arun Aravind	Scientific Officer (C)	Radionuclide or contaminant modeling in saturated zone
Shri Swayam Mallick	Scientific Officer (C)	Solvent extraction chemistry in reprocessing
Shri G. Suresh Kumar	Scientific Officer (C)	In-service inspection and residual life assessment of nuclear components
Shri Krishna Chandran R.	Scientific Officer (C)	Thermal hydraulics and computational fluid dynamics

Visiting Scientist:

Dr. S. M. Lee	Raja Ramanna Fellow	Reactor physics and nuclear safety

SRF/JRF:

Shri E. Vinnitha	Senior Research Fellow	Environmental effects of power plant effluents containing chemical biocides on entrained organisms.
Shri L. Thilagam	Senior Research Fellow	Light water reactor physics
Shri M. Sankarram	Senior Research Fellow	Development of decision support systems using RS-GIS techniques
Ms. R. Deepthi Rani	Senior Research Fellow	Colloid mediated radionuclide migration
Shri Sajith T. Mathews	Senior Research Fellow	Reliability and thermal hydraulics
Shri R. Kaviyarasan	Junior research Fellow	Groundwater modeling studies

Appendix - II

PUBLICATIONS (1999-2008)

JOURNAL PUBLICATIONS:

1. *Sunil Sunny C. and Subbaiah K.V.,* "Radiation Shield Design Evaluation for GE PET Trace Proton Cyclotron", Journal of Radiation Protection and Environment, Vol. 23, No. 4, 2000.

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5. *John Arul A., Senthil Kumar C., Marimuthu S. and Om Pal Singh,* "The Power Law Character of Off-site Power Failures", Annals of Nuclear Energy, Vol. 30, pp1401-1408, 2003.

6. *Kumar, S.S., Sivaiah, M.V., Venkatesan, K.A., Krishna, R.M., Murthy, G.S. and Sasidhar, P.,* "Removal of Cesium and Strontium from Acidic Solution using a Composite of Zirconium Molybdate and Zirconium Tungstate", Journal of Radioanalytical and Nuclear Chemistry, Vol. 258, pp321, 2003.

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10. *Subbaiah K.V. and Sunil Sunny C.,* "View-CXS: Neutron and Photon Cross-sections Viewer", Annals of Nuclear Energy, Vol. 31, No. 8, pp923-931, 2004.

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15. *Sivaiah, M.V., Venkatesan, K.A., Sasidhar, P., Krishna, R.M. and Murthy, G.S.,* "Ion Exchange Studies of Cerium (III) on Uranium Antimonite", Journal of Nuclear and Radiochemical Science, Vol.5, pp7, 2004.

16. *Sivaiah, M.V., Kumar, S.S., Venkatesan, K.A., Sasidhar, P., Krishna, R.M., and Murthy, G.S.,* "Sorption Studies of Strontium on Zirconium Modified Vermiculite", Journal of Nuclear and Radiochemical sciences, Vol.5, No.2, pp33-36, 2004.

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Ph.D. THESES

1. *Gurumoorthy C.,* "Simulation of Radionuclide Migration through Rock Mass in a Geotechnical Centrifuge", Indian Institute of Technology-Bombay, Mumbai, (2002).

2. *Sunil Sunny C.,* "Monte Carlo Simulation Studies on Radiation Transport through Bulk Media", University of Madras, Chennai, (2004).

3. *Senthil Kumar C.,* "New Approaches in Reliability Analysis of Nuclear Reactor Safety Systems", Anna University, Chennai, (2005).

4. *Anandan C.* "Mapping and Mitigation of Geohazards in Mountainous Systems using Remote Sensing and GIS - A Model study from parts of Western Ghats, South India". Bharathidasan University, Tiruchirappalli, (2008).

In addition to above, SRI has brought out following publications (1999-2008):

Conferences	:	96
Reports	:	44