SRI
Highlights

2010 - 2014

सत्यमेव जयते

Safety Research Institute
Atomic Energy Regulatory Board
Government of India
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SRI HIGHLIGHTS 2010-14

30 Years of Safety Regulation 1983 - 2013

परमाणु ऊर्जा नियामक परिषद

सत्यमेव जयते

भारत सरकार

GOVERNMENT OF INDIA

Atomic Energy Regulatory Board

एस.एस. बजाज, एफएनएई
S.S. BAJAJ FNAE
अध्यक्ष
CHAIRMAN

Message

The mission of the AERB is to ensure that the use of ionising radiation and nuclear energy in India does not cause unacceptable impact on the workers, members of the public and to the environment. AERB has the mandate to carry out detailed safety review for the siting, construction, commissioning, operation and decommissioning of nuclear and radiation facilities. To maintain a strong, credible and technically sound regulation, it was felt to establish an in-house safety research wing with robust technical infrastructure and wide knowledge base. Thus came into existence the Safety Research Institute (SRI) at Kalpakkam in 1999 by the initiative of former Chairman of AERB, Prof. P. Rama Rao.

I am very much delighted that Safety Research Institute Highlights is being brought out with updates on various important R&D programmes pursued during the period 2010-14. As is mandated in its formation, it is heartening to note that the R&D programs being pursued at SRI are highly relevant and focussed towards the regulatory framework of AERB. The safety studies at SRI have the objective to support and facilitate decision making by AERB at various stages of regulatory review. SRI by virtue of its multi-faceted scientific research in wider specializations, provides a platform for young researchers and engineers to fine-tune their competencies, develop domain specific expertise and in translating research results to practical regulatory decision making solutions. The endeavour has been to keep the programmes more and more oriented to the current needs of nuclear safety and regulation. The quantum and spread of areas of research at SRI have grown with time encompassing areas that include engineering safety analyses, probabilistic safety assessment, reactor & radiation physics, environment & fuel chemistry, remote sensing & GIS applications, radionuclide migration, atmospheric dispersion studies etc. The scientific activities presented in this report highlight the accomplishments by SRI and indicate its important role in regulatory safety research of AERB.

I am sure that the scope of safety research at SRI will be further enhanced and carried forward with greater passion and enthusiasm in supporting the regulatory decision making at AERB through robust R&D programmes.

(S.S.Bajaj)

Mumbai
28.11.2014

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उपाध्यक्ष
VICE CHAIRMAN

Message from Vice Chairman, AERB

It gives me great pleasure to note that SRI is bringing out an edition of SRI Highlights (2010-2014) this year. The need for developing R&D infrastructure and improving the all round competency to support its regulatory activities was recognized very early by AERB. With this aim, and in order to meet the present and future regulatory challenges on a strong technical basis, the Safety Research Institute was set up at Kalpakkam in the year 1999. Research areas are carefully chosen keeping in view their importance to safety review carried out by AERB as well as to complement ongoing R&D works at other DAE units.

SRI has grown manifolds since its inception, and today, expertise is available in several fields such as reactor physics, radiation shielding, nuclear reactor thermal hydraulics, environmental and fuel chemistry, probabilistic safety assessments and reliability studies, remote sensing and geographical information systems and several other areas of importance to AERB and the DAE.

Over the last few years, in-house R&D has resulted in the development and validation of several computer codes for safety analysis. Many of these codes are currently employed for independent verification of utility safety analyses and to support the regulatory review process. For safety issues that need experimental verification, SRI has embarked on setting up in-house experimental facilities in the area of fire safety, hydrogen mitigation and reactor thermal hydraulics. In the long run, these facilities will enable SRI to add immense value to its ongoing research activities. The existing laboratories for chemistry, physics and RS-GIS at SRI are fully equipped to cater to the experimental research activities in their respective areas.

The continued support and assistance given by IGCAR in expansion and growth of SRI is worth mentioning. It is gratifying to note that SRI is participating in collaborative research work with various R&D as well as academic institutions.

This report on SRI Highlights (2010-2014) gives an account of various R&D activities, important contributions to regulatory review and other significant achievements of SRI during the last five years. I convey my heartiest congratulations to all at SRI for bringing out this issue of SRI highlights.

(S. Duraisamy)
Vice Chairman, AERB

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ATOMIC ENERGY REGULATORY BOARD

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एवं
निदेशक, सूचना एवं तकनीकी सेवाएं प्रभाग

SECRETARY, AERB
&
Director, Industrial Plants Safety Division
&
Director, Information & Technical Services Division

Message

It is heartening to note that AERB Safety Research Institute Highlights (2010-2014) is brought out with important R&D activities pursued during the last five year period. It is necessary that dedicated in-house research is carried out in support of nuclear regulatory activities. Research programs at SRI are carefully tailored to strengthen the regulatory decision making of AERB and also complementary to the ongoing R&D activities at various other DAE units. The institute being located within IGCAR campus has provided a strong platform for collaborative research with access to various state-of-art laboratories and library facilities.

The scientific and technical contributions of SRI since its inception have been varied and significant. The domains of scientific activities at SRI ranges from studies related to reactor physics and radiation shielding aspects of thermal and fast reactors; criticality safety evaluations of fuel cycle facilities; probabilistic safety assessment of nuclear power plants; structural and seismic analysis of NPPs; fire safety and thermal hydraulics; remote sensing and GIS applications; radiological impact assessment and nuclear fuel chemistry. Over the years, the safety research at SRI has resulted in synergy, enhanced quality which helped in regulatory decision making process.

In recent times, SRI has contributed significantly towards the reactor physics analysis of the VVER reactor commissioned at Kudankulam and the Prototype Fast Breeder Reactor due for commissioning at Kalpakkam. There has also been a gradual thrust towards experimental programmes, which has resulted in setting up the state-of-art chemistry and radiation physics laboratories at SRI.

In nutshell, SRI has made excellent progress in supporting the multifarious regulatory activities of AERB during a span of fifteen years. This report highlights the important contributions made by SRI during the past five years in a crisp and lucid manner. I look forward to greater contributions from SRI and hope it will reach the higher levels in safety research in the years to come.

R Bhattacharya
(R. Bhattacharya)

Mumbai
28.11.2014

FOREWORD

The Safety Research Institute (SRI) is an institution dedicated to R&D in areas relating to regulatory research that are of concern to AERB. It is unique in terms of carrying out basic research in important safety related areas and in fostering independent applied that are relevant to AERB in its regulatory activities. By virtue of its location within the campus of Indira Gandhi Centre for Atomic Research, Kalpakkam, SRI benefits from exchange of scientific information and knowledge required for focused research besides access to various state-of-art laboratories, library facilities, etc. Since its inception in 1999, the institute has steadily grown engaging in challenging R&D activities, in areas of regulatory concerns that are nurtured to high levels of quality standards.

R&D activities, as highlighted in this report, cover significant contributions in wider range of safety related topics that include reactor & radiological safety, probabilistic risk assessment, remote sensing and GIS applications, structural and seismic safety, fire safety, hydrogen safety, thermal hydraulic analyses, environmental & fuel chemistry etc. As scientific publications are integral part of research program, emphasis is placed on timely communication of important R&D results to peer reviewed journals, national / international conferences, preparation of internal reports, scientific colloquia, etc. It is indeed gratifying to note that there is significant increase in publications at SRI. During the period 2009-14, a total of 45 papers in peer-reviewed journals, 105 papers in national / international conference proceedings etc. were published. In addition, more than 100 internal reports were also prepared on the scientific works at SRI.

Newer areas of interest to regulatory aspects, such as hydrogen management and mitigation, fast reactor physics, LWR thermal hydraulics, fire safety and hazard studies, multi-unit risk assessment, flood PSA, atmospheric dispersion modeling studies, NORM waste management, etc. were identified and significant progress has been made in the recent years. Three state-of-art laboratories are established for carrying out studies relating to radiation physics, reprocessing chemistry & isotope migration and remote sensing applications.

Further, SRI has strengthened its linkage with academic institutions by signing a MoU with Anna University, Chennai for research collaboration and increased its interactions with CLRI & IIT-M (Chennai), IISc (Bangalore), Annamalai University (Chidambaram) and Bharathidasan University (Trichy). SRI is now a recognized "Centre for Research" of Anna University for post graduate and doctoral programs. Periodic visits by various eminent personalities to the institute also provided motivation to colleagues at SRI, facilitating high quality interactions with directions for future R&D program.

Director, SRI

SRI HIGHLIGHTS 2010-14

CONTENTS

1. ABOUT SRI

The objective of Safety Research Institute is to build up a unique research and knowledge base with a strong research capability in important safety related areas to support regulatory functions of AERB. With this objective, under the initiative of Prof. P. Rama Rao, the then Chairman, AERB, Safety Research Institute at IGCAR campus at Kalpakkam was setup. The foundation stone for the institute was laid on February 20, 1999 by Dr. R. Chidambaram, the then Chairman, AEC during the IX Five Year Plan period with P. Rodriguez as first Director of the Institute.

Research areas undertaken in SRI are specifically chosen in a way to complement the ongoing R&D activities at various other DAE units. The institute located within IGCAR campus, possess an added advantage with access to various state-of-art laboratories, library facilities, interaction with large pool of scientists & engineers etc.

The domains of scientific activities at SRI ranges from Light Water and Fast Reactor Physics, Radiation Shielding & Transport and Criticality Computations, Reliability and Probabilistic Safety Assessment, Structural and Seismic Studies, Fire Safety, Thermal Hydraulic Studies, Safety Assessment of NSDF, Radionuclide migration studies, Waste management related research activities, Fuel safety studies, Remote Sensing and GIS Applications, establishment of a depository of Computer Codes etc. Structuring and prioritising of the research activities at SRI, in the above areas, are closely monitored by AERB management and their progress is periodically reviewed by AERB Executive Committee, SRI-Scientific committee (now replaced by AERB - Advisory Committee on Regulatory Safety Research – ACRSR) and AERB Board.

Directors of SRI since inception

Dr. Placid Rodriguez
November 1998 to October 2000

Shri A. R. Sundararajan
July 1999 to October 2000 (Joint Director)
November 2000 to February 2003 (Director)

Shri S.K. Chande
March 2003 to July 2005

Shri S.E. Kannan
June 2005 to July 2005 (Head, SRI)
August 2005 to May 2010 (Director, SRI)

Shri V. Balasubramaniyan
June 2010 to Feb 2012 (Head, SRI)
March 2012 onwards (Director, SRI)

ORGANISATIONAL CHART

December 2014

S. S. Bajaj, Chairman, AERB

Vice Chairman, AERB

V. Balasubramaniyan, Director-SRI

OFFICE

P.Varadarajan	UDC
Ms. Jayalaxmi T.H	Steno-II

NUCLEAR SAFETY ANALYSIS SECTION (NSAS)

Dr. D.K. Mohapatra, Head	**SO/G**
Dr. L. Thilagam	TO/D
S. S. Singh	SO/D
Mohd. Ilyas Lone	SO/D
R. Karthik	SO/C
Smt. M.R. Priya	SA/B

ACTIVITIES

- Reactor physics analyses of thermal and fast reactors
- Radiation physics and shielding studies
- Safety evaluations of nuclear fuel cycle facilities
- Radiological risk and environmental impact assessment.
- Consequence analysis

ENVIRONMENT& FUEL CHEMISTRY SECTION (EFCS)

Dr. H. Seshadri, Head	**SO/F**
Dr. C. Gurumoorthy	SO/E
Arun Aravind	SO/D
Ramakrishna Pagoti	SO/D
Krishan Kumar	SO/C
Ms. Mansi Garg	SA/B

ACTIVITIES

- NORM waste management studies
- Radionuclide migration studies in the geo-environment
- Atmospheric dispersion of radionuclides
- Fuel chemistry and safety studies
- Radionuclide sorption studies
- Geochemical studies

ENGINEERING SAFETY SECTION (ESS)

Dr. Seik Mansoor Ali, Head	**SO/F**
Jagannath Mishra	SO/F
Dr. Nilesh Agrawal	SO/E
R. Krishnachandran	SO/D
K. N. V. Adinarayana	SO/C
Tejaswi Abburi	SO/C
Gade Venkata Rajeev	SO/C
Neeraj Mohan	SO/C
Kota Sampath Bharadwaj	SO/C
Nakul Sashidharan	SO/C
L. Raj	SA/B

ACTIVITIES

- Core and containment thermal hydraulic studies for thermal reactors
- Fire safety studies
- Hydrogen safety studies
- Structural and seismic analysis
- Radionuclide migration studies
- Consequence analysis
- Setting up experimental facilities

RISKASSESSMENT AND GIS APPLICATIONS SECTION (RAGISS)

Dr. C. Senthil Kumar, Head	**SO/G**
Dr. C. Anandan	SO/D
Smt. C. Jayalakshmi	SA/D

ACTIVITIES

- PSA and reliability studies
- GIS and Remote Sensing
- Statistical analysis
- Consequence analysis
- Evolving decision support system models
- Computer networking and communication
- Infrastructure development
- Documentation, training & workshops

2. AREAS OF RESEARCH

Research areas undertaken in SRI are specifically chosen in a way to complement the ongoing R&D activities at various other DAE units. The institute, by virtue of its being located within IGCAR campus, provides added advantages to SRI personnel with access to various state-of-art laboratories, library facilities, interaction with large pool of scientists & engineers etc.

Major Areas of Research

Nuclear and Reactor Safety Studies

- Reactor Physics Studies
- Seismic Studies
- Consequence analysis and Atmospheric dispersion modeling
- Probabilistic Safety Assessment and Reliability Analysis
- Development of methodology for Passive system reliability, software reliability and human reliability

Radiation Safety Studies

- Generic Issues
- Radiation Streaming Studies
- Bulk Shielding Studies
- Radiation Dosimetry

Engineering Safety Studies

- Nuclear Reactor Thermal Hydraulics
- Hydrogen distribution and mitigation studies
- Fire safety
- Component safety and Structural reliability analysis

Environmental Safety Studies

- Remote Sensing and Geographic Information System Applications
- Design and organization of digital database and information system to carry out Environmental Impact Assessment.
- Development of digital database for sites of nuclear facilities.
- Emergency Preparedness Plans for Tsunamis and Disaster Management at Nuclear Installations
- Development of Simulation tools to predict flood inundation patterns and their validation.
- Sea Surface temperature studies around nuclear facilities

Waste Management and Radionuclide Transport Safety Studies

- NORM waste management
- Uptake of radionuclides using sorbents
- Diffusive leaching of radionuclides from waste matrices
- Degradation of toxic organic pollutants from liquid waste
- Phytoremediation of radionuclides

3. CURRENT RESEARCH ACTIVITIES

3.1 NUCLEAR SAFETY STUDIES

Introduction

During the formative days of SRI, it was envisaged that the country would expand its nuclear programme by deployment of advanced and new generation nuclear power plants in association with integral fuel cycle facilities. To carry out effective regulation of these new systems and facilities, utmost importance was given to develop technical expertise in the areas of reactor physics and radiological safety. In view of this, SRI has aptly selected research topics in the areas related to advanced light water reactors, fast breeder reactors, nuclear fuel cycle and radiological impacts arising from these systems. Recently, a radiation physics laboratory has been setup to start experimental programme for validation of codes and computational methods adopted in radiation transport studies. Going further ahead, a research programme on High Energy Dosimetry in collaboration with the Medical Physics Department of Anna University, Chennai is also initiated. The salient contributions and highlights of the important research works are delineated in the following sections.

A) REACTOR PHYSICS STUDIES

A.1) Light Water Reactor Studies

A.1.1) Analysis of Reactor Core Scenario of Kudankulam (KK) Unit-1 VVER-1000 Reactor: Initial fuel loading, First approach to criticality and Phase B & C physics experiments

Theoretical analyses of the initial fuel loading, first approach to criticality, Phase-B low power physics experiments and Phase-C physics experiments at 40% and 73% full power (FP) were carried out for VVER KK Unit-1 using indigenous deterministic code system, EXCEL-TRIHEXFA and Monte Carlo method. The analyses include partial loading of fuel assemblies in dry condition, filling of moderator with high boron, loading of remaining 103 FAs, warming up of the system, withdrawal of 10 control groups (comprising of 85 rods), approach to first criticality using boron dilution, individual, group wise and integral worth of control protection system absorber rods, power distribution at 40% and 73% FP, temperature coefficient of reactivity as a function of boron and boric acid coefficient of reactivity etc. The results of the analyses were compared with experimental values and are found to be in good agreement. KK Unit-1 was found to achieve criticality with the critical boric acid concentration, which was very close to that estimated through Monte Carlo calculations.

A.1.2) Reactivity Initiated Transient Analysis for Kudankulam VVER-1000 Reactor

VVER-1000 reactor commissioned at Kudankulam usually attains criticality at Hot Zero Power (HZP) state by removing soluble boron from moderator. At this low power, reactivity feedbacks are least effective and thus Reactivity Initiated Accident (RIA) occurring during the startup could be very severe. In view of this, reactivity insertion transients (RIT) at HZP in KK VVER-1000 reactor were studied. Reactivity insertions due to ejection and withdrawal of a single and the entire working group (Group-10) of Control Protection and System Absorber Rods (CPSARs) have been considered for RIT analysis at HZP. A 3D Space-time kinetics code, TRIKIN with thermal hydraulics feedback is used to perform the desired task. The response of core dynamics parameters like reactivity, power, maximum fuel temperature, peak fuel centerline temperature, peak fuel clad temperatures, maximum fuel enthalpy and DNBR etc. to the reactivity insertions were studied.

To highlight some of the important results, Figs. 1 and 2 demonstrate the behavior of thermal power, and maximum fuel temperature as a function of time during the transient of inadvertent withdrawal of the entire working group from the position 20% inside the core with a speed of 2 cm/s. Values reported in the Preliminary Safety Analysis Report (PSAR) are also plotted in these figures. PSAR values are presented with the consideration of trip, whereas TRIKIN results are with and without trip. It is clear from these figures that comparison during the transients generally shows good agreement with PSAR values and it renders sufficient confidence for analysis of further transients. The analyses were further extended to predict the response of core dynamics parameters to RIA occurring at higher power. Results of the analyses at higher power levels are also found to be in good agreement with PSAR values. Behavior of reactivity vs. time during inadvertent withdrawal of working group is given in Fig. 3 for various power levels.

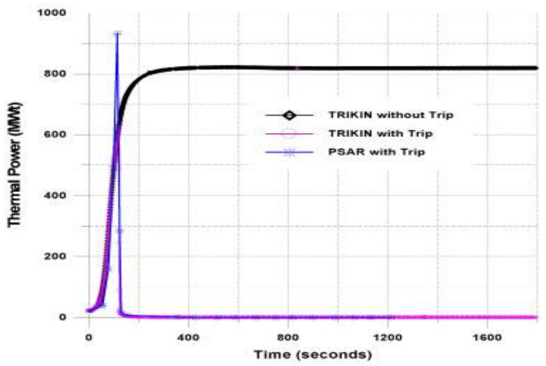


Fig. 1 : Thermal power vs. time

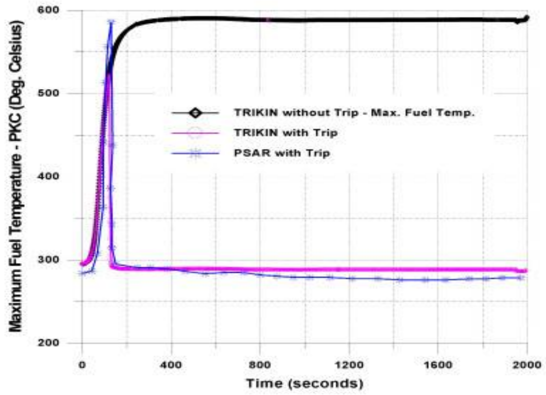


Fig. 2 : Maximum fuel temperature vs. time

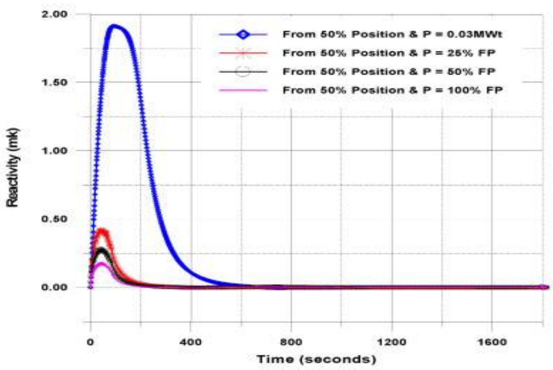


Fig. 3 : Reactivity vs. time – RWA Event

A.1.3) Benchmark Analysis of Next Generation LWR Fuels

Validation of computational tools for the design analysis of modern light water reactors (LWR) has become important in view of the renewed interest in deployment of a variety of modern LWRs in India. As a validation of lattice burnup code DRAGON, benchmark problem suite for reactor physics of LWR next generation fuels was analyzed. This benchmark problem was suggested by working party on Reactors Physics for LWR next generation fuels which was organized in Japan Atomic Energy Research Institute (JAERI). It should be noted that the next generation fuel enrichment was increased above existing design limitations for LWR fuels. Next generation

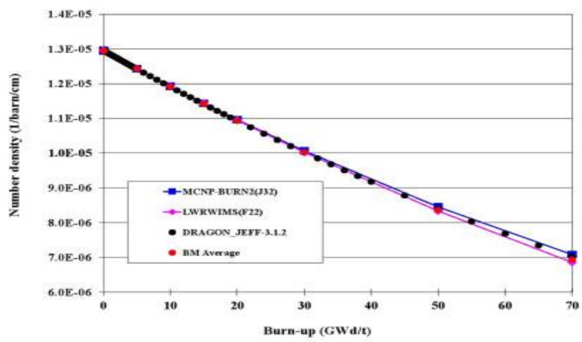


Fig. 4 : U-235 isotopic concentration vs. burnup

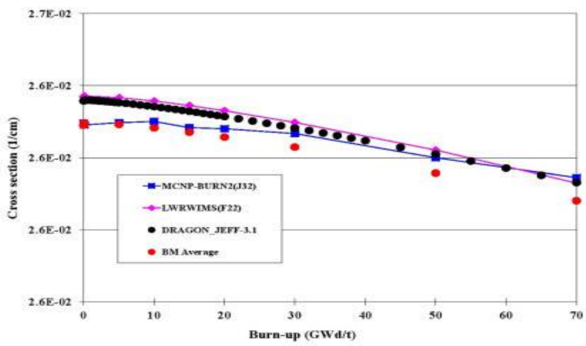


Fig. 5 : Macroscopic absorption cross-section vs. burnup

LWR fuels' discharge burnup target value was set to reach 70 GWd/t. PWR UO_2 and MOX fuel pin cell models were simulated to study burnup dependent behaviour of infinite neutron multiplication factors (k_{inf}), isotopic composition and 1-Group (1G) neutron production and absorption cross-sections in different reactor state descriptions. PWR (17X17 array) fuel assembly (FA) model of UO_2 with Gd_2O_3 mixed with UO_2 as integral burnable absorber and MOX fuel assembly model were also simulated to test predictive capability of kinf and fission rate distributions as a function of burnup. Comparison of DRAGON results with the mean values of sixteen other international codes was made and it shows good agreement among them for all physics parameters studied. Sample plots showing comparison of DRAGON simulated U-235 number density, macroscopic 1-Group (1G) neutron absorption and production cross-sections are shown in Figs. 4 to 6 for MOX pin cell model. A plot demonstrating k_{inf} variation as a function of burnup for different reactor state descriptions is shown in Fig. 7 for PWR UO_2 FA model.

A.1.4) Preliminary Steady-state Core Neutronics Analysis of European Pressurized Water Reactor

As per the national nuclear energy policy, new generation light water reactors (LWRs) of higher

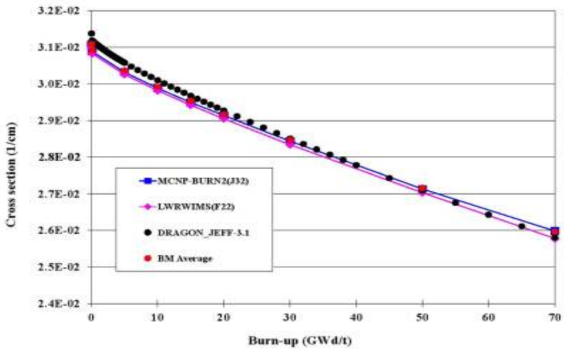


Fig. 6 : Macroscopic production cross-section vs. burnup

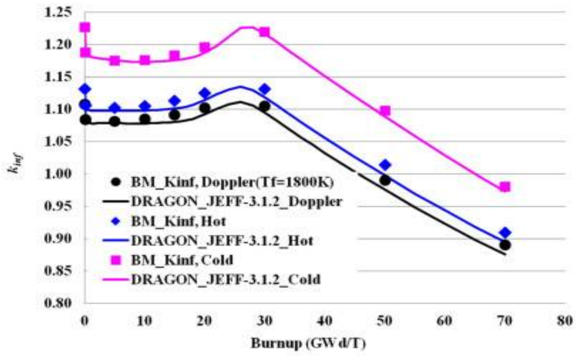


Fig. 7 : kinf vs. burnup – PWR UO_2 fuel assembly model

power capacity are being planned to be deployed in India to meet its growing electricity demand. These large LWRs need thorough technical evaluation prior to regulatory acceptance. To establish the required expertise for advanced LWR reactor physics safety evaluation, steady state core neutronics analysis of European pressurized Water Reactor (EPR) has been taken up at SRI. Unlike Russian VVER, the fuel assemblies (FAs) in EPR core and the fuel pins within an assembly are arranged in regular square arrays. An attempt was made to carry out lattice level calculations of EPR using a modern and advanced lattice burnup code, DRAGON while the core level calculations are carried out by Monte Carlo method.

Fuel assemblies of EPR are of seven basic design, which are further differentiated into ten types based on U-235 enrichments of FAs and radial and axial distributions of Gd_2O_3 in UGD (UO_2+Gd_2O_3) rods. First four types are FAs with uniform enrichments of ^{235}U like 2.0 wt% (Blanket), 2.25 wt%, 2.70 wt% and 3.25 wt%. Next six types are perturbed FAs with variable wt% of Gd_2O_3 in UGD (UO_2+Gd_2O_3) rods. A typical initial core loading map is shown in Fig. 8.

An analysis of the steady state reactor core physics and safety parameters of US-EPR for initial fuel loading was performed using Monte Carlo method

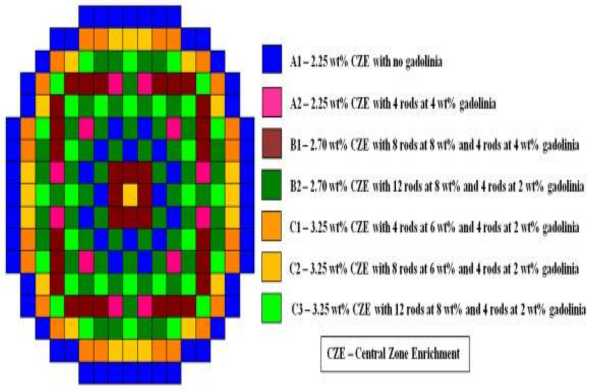


Fig. 8 : Typical initial core loading map of EPR

employing various nuclear data libraries like ENDF/B-VI, ENDF/B-VII, JEFF-3.1 and JENDL-3.2. Effective neutron multiplication factors (K_{eff}) for various reactor states, worth of control and shutdown banks and delayed neutron fraction have been calculated and compared with US-EPR Final Safety Analysis Report (FSAR) and are observed to be in good agreement. A Comparison of the calculated worth of control and shutdown banks with the US-FSAR values is shown in Fig. 9. Worth difference with respect to FSAR values varies within ±130 pcm. Variation of infinite neutron multiplication factor (k_{inf}) and isotopic composition of individual nuclides as function of burnup were studied. A result of k_{inf} variation as a function of burnup is shown in Fig. 10 for different FAs of EPR.

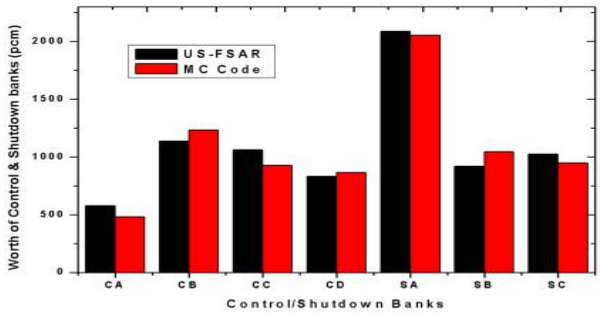


Fig. 9 : Comparison of calculated worth of control and shutdown banks with US-FSAR values

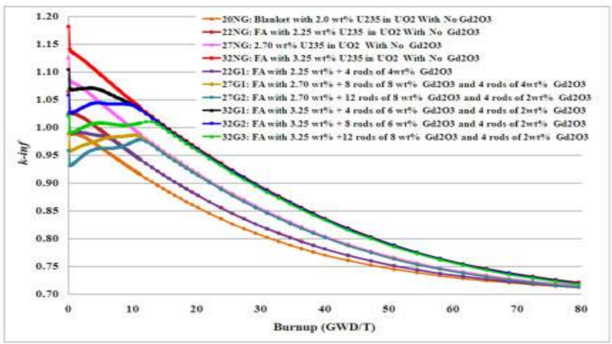


Fig. 10 : k-infinity variation as a function of burnup for different FAs of EPR

A.2) Fuel Cycle Studies

A.2.1) Transmutation Characteristics of Minor Actinides in different Nuclear Reactor Spectra

During nuclear power generation highly radiotoxic by-products such as the minor actinides are formed, which accumulate in spent fuel storage or reprocessing plants on a larger scale. To minimize the long term radio toxicity risk from these minor actinides, theoretical and experimental studies are going on to transmute the long lived minor actinides to short lived or stable species. In this regard, transmutation characteristic of Am-241, one of the major contributors to the radio toxicity of the spent nuclear reactor fuel, was analyzed in thermal and fast reactor neutron spectra by fuel depletion code, ORIGEN. The evolution of the actinides masses (normalized to the initial ^{241}Am mass of 1 ton) with time is shown in Fig. 11 for various Indian power reactors along with the standard fast and thermal reactor spectra of ORIGEN code with their respective flux levels. The study indicates that even though the transmutation rates are higher in thermal spectrum than in fast spectrum for the same level of neutron flux, ultimately fast reactors turn out to be superior for transmutation due to their high flux level of ~10^{16} n/cm^2/s as compared to thermal reactors with the low flux level of ~10^{14} n/cm^2/s.

A.2.2) External Coupling of Monte Carlo Neutronics and ORIGEN Codes for Fuel Depletion Analysis

Fuel depletion calculations coupled with Monte Carlo neutronics codes have gained attention due to the ability to accurately model complex geometries with better estimation of various nuclide inventories. In view of this, an effort has been made to develop a linkage program that performs time-dependent burnup calculations by combining Monte Carlo (MC) neutronics code and nuclear fuel depletion code, ORIGEN as a necessary and required advancement over existing diffusion theory codes.

The MC code is capable of providing the neutron flux and effective one-group cross sections for different regions to be use in ORIGEN calculations. ORIGEN in turn performs multi-nuclide depletion calculations for each region with MC code generated one-group cross-section and provides material compositions for next MC simulations. By iterative coupling of these two codes, burnup studies can be carried out for any generalized, user-defined geometry. Fortran90 based coupling program has been developed and validated against deterministic lattice code calculations for a PWR pin cell benchmark. Burnup dependent infinite multiplication factor (k_{inf}), one-group microscopic cross-section of actinides and isotopic compositions are calculated for the PWR pin-cell and compared with the calculations by deterministic lattice code DRAGON. As a demonstration, the variation of U-235 composition with burnup is illustrated in Fig. 12.

Figure 13 shows the variation of kinf as a function of burnup for three different cases: (1) without updating burnup dependent one-group cross-sections in ORIGEN (i.e.) by making use of ORIGEN library, PWRUE.LIB (4.2% U-235 fuel, 3 Cycle PWR to achieve 50MWd/kg, Thermal Spectrum), which comprises of 1307 different nuclides, including 78 actinides and 825 fission products as such for all burnup steps, (2) updating burnup dependent one-group cross-sections in ORIGEN with MC code generated values for actinides, and (3) updating with MC code generated cross sections for actinides as well as for fission products. It is also clearly demonstrated in Fig. 13 that the burnup dependent one-group cross sections of both actinides and fission products are important for fuel depletion calculation.

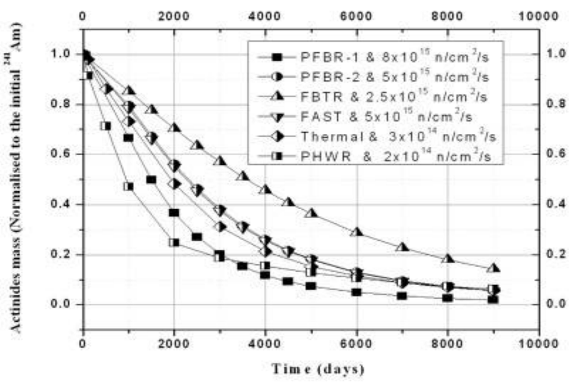


Fig. 11 : Actinides mass vs. time for various reactor spectra with their respective flux levels

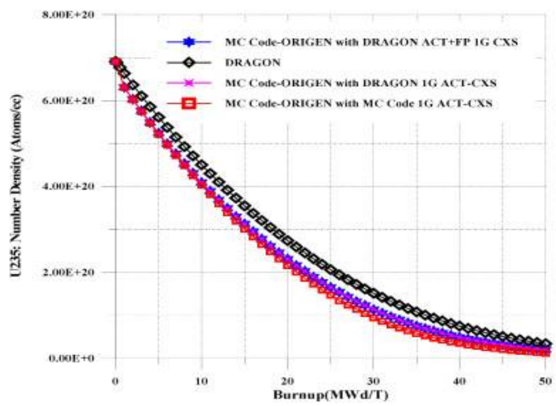


Fig. 12: U-235 isotopic composition vs. burnup

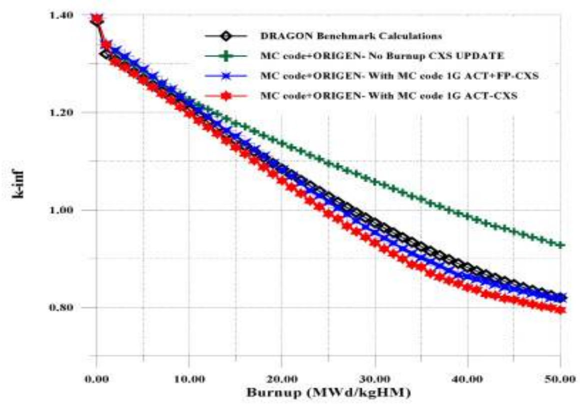


Fig.13: k-inf vs. burnup for different cases studied

B) RADIOLOGICAL SAFETY STUDIES

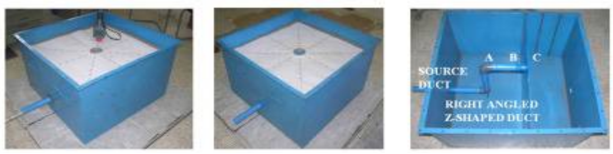
B.1) Radiation Streaming Through In-homogeneities in Shield Structures

In-homogeneities in shield structures lead to considerable amount of leakage radiation (streaming) giving rise to increased radiation levels in accessible areas. It is very difficult to evaluate and quantify the streaming radiation doses due to radiation propagation through the in-homogeneities like ducts and voids present in the shields, as the shield structures are complex and there is no general approach or empirical formula found suitable for solving all kinds of shielding problems. Usually, Monte Carlo based radiation transport codes which are applicable for complex configuration of geometry and materials are employed to analyze such problems. In order to validate the methodologies and calculation strategies adopted in radiation transport codes, it is essential to carry out experimental measurements for typical radiation streaming cases. Towards these objectives, prototype gamma and neutron streaming experiments have been carried out simulating various in-homogeneities present in bulk shields. The data thus generated were analyzed by simulating the experimental set up with Monte Carlo code and thereby validating the input parameters for the code for similar radiation streaming problems. Since penetrations and ducts through bulk shields are unavoidable in a nuclear facility, the results on radiation streaming simulations and experiments will be very useful for the shield structure optimization and the regulatory review of this optimization.

B.1.1) Gamma Streaming Experiments in Bulk Shields with In-homogeneities

Gamma radiation streaming experiments were carried out by employing a point Co-60 source of activity ~ 0.07mCi, NaI (Tl) based hand held gamma spectrometer and ducts of various sizes and shapes. Experimental set up consists of a stainless steel box of dimensions 80 cm X 80 cm X 50 cm filled with sand as shield material. Straight hollow cylindrical, annular and right angular z-shaped ducts were used as in-homogeneities in sand shielding material. These duct shapes were arrived based on past experience in shield design optimization calculations for Prototype Fast Breeder Reactor. Schematic of the experimental set up containing various ducts as heterogeneity in shield material are shown in Fig. 14. A PVC pipe of 2 cm thickness was employed as shield duct as shown in the figure. Height of this shield duct above the source duct is 25 cm. In case of straight annular duct, it consists of two straight hollow cylinders with the inner one filled with sand shield material. Inner and outer radii of the outer cylindrical duct are 4.3 cm and 4.5 cm respectively. Inner cylinders with various diameters were employed to construct ducts of different annular dimensions. In both the cases, dose rate measurements were made for 13 dwell positions of the source and at 40 detector locations for each dwell position. The gamma source is introduced through a bottom pipe as shown in the figure and the dose rate measuring device is shown on the top of the shield with measurement points marked on a sheet of paper.

The right angular z-shaped duct set up is provided with the slots to create shield material with variable thickness. Dose rates were measured at three different locations A, B and C as shown in Fig. 14. They are chosen such that the location A is after one bend, C is after the second bend and at the exit of the duct and the location B is at the mid-way between A and C. The source is kept inside the shield at the entrance of the bottom duct. Sample plots of contours depicting the gamma streaming through the straight hollow cylindrical duct and straight annular ducts of two different dimensions are shown in Fig. 15. These contours have been generated from the experimental dose rate measurements for various source locations. Monte Carlo simulated theoretical gamma dose rate distributions were found to agree well with the experimental measurements, which give confidence for future analysis of gamma streaming problems.



Hollow cylindrical duct | Annular cylindrical duct | Z-shaped duct

Fig. 14 : Schematic of experimental setup

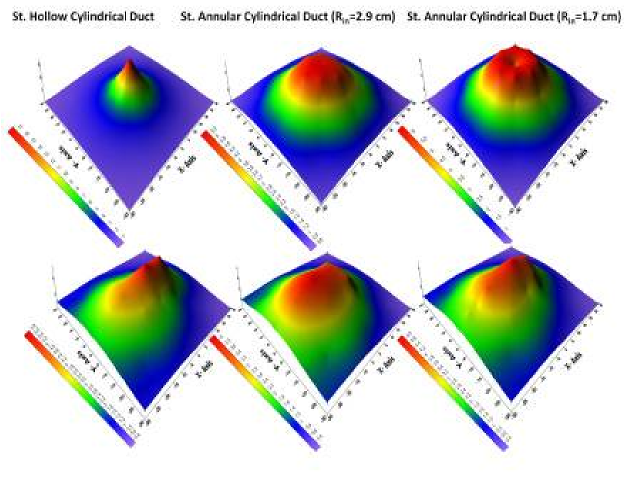


Fig. 15 : Contours depicting gamma streaming - Source position at centre (top) and source 10 cm away from centre (bottom)

B.1.2) Neutron Streaming Experiments

Neutron streaming experiments were carried out in a Linear Accelerator (LINAC) facility of IGCAR. There are three trenches in the LINAC hall for laying the cooling channels, power and control cables to the neutron generator from the control room. Neutron streaming experiments were conducted in one of the trenches by keeping a standard Am-Be (5Ci) neutron source at the entrance point of the trench. A pictorial view of the experimental arrangement is shown in Fig. 16. Dose rates were measured at five locations (D1-D5) along the trench including its entrance and exit end. Neutron monitor, REM counter and neutron spectrometer were used to measure neutron dose rates. The theoretical modeling of the measurements by Monte Carlo method and comparison of results are shown in Fig. 17. The results of the experiments were compared with theoretical Monte Carlo simulations to validate the neutron streaming analysis methodology.

B.2) Development of Computer Code for Optimization of Gamma Ray Shield Design

Computer codes for radiation shielding analysis need to be developed with the modern numerical algorithms for fast computations. A new computer code named as "SHIELD-DESIGN" was developed in Visual Basic Version 6.0 for precise and quick evaluation of gamma ray shield designs. In the conventional forward calculations using point kernel method, the sources and geometry are specified along with the shield thickness to obtain dose rates at desired locations. But in this code, a new computational technique based on inverse calculation and iterative procedures were employed to estimate the shield thickness using the dose rate criteria as input information. This new technique enables quicker and easier way of shield design optimization by computing shield thickness required in a single run whereas in the forward shield design optimization method, dose rate computations need

Fig. 16: Pictorial view of the neutron streaming experiment

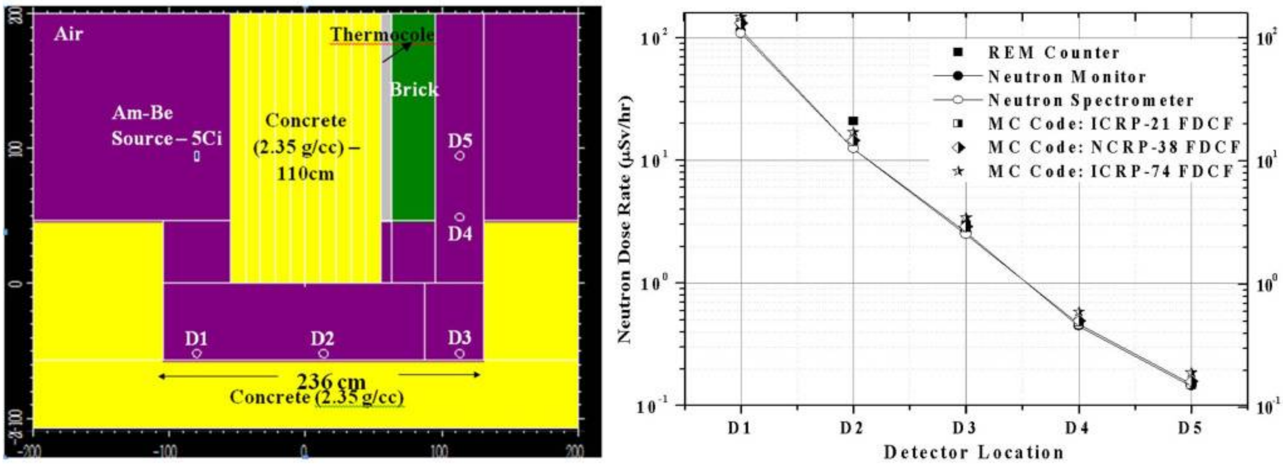


Fig. 17 : Theoretical modeling of the neutron streaming experiment and comparison of results

to be carried out for various shield thicknesses. The shielding thickness arrived with this code provides a first hand information to the designer to carry out forward calculations for accurate optimization. Also, the code is very useful to regulators for quicker verification of shield calculations submitted by the designers for approval. The validation of the developed code was carried out by performing analyses of the gamma shielding design for some standard problems. The results were compared with those of renowned forward calculations, experimental measurements, NCRP-51 benchmarks and other gamma radiation shielding utilities like RAD-PRO and satisfactory agreement was observed.

B.3) Monte Carlo Modeling of Gamma Radiation Detectors

Monte Carlo (MC) simulations were carried out in support of experiments to determine the photo peak efficiency of a 2” x 2” NaI (Tl) scintillation based gamma radiation detector as a continuous function of energy. MC simulations give the exact behavior of efficiency as a function of energy whereas the experiments are limited due to non-availability of sources for continuous energy. Gamma spectra of various standard radionuclides like Co-57, Co-60, Ba-133 and Cs-137 have been simulated and compared with the experimental spectra to validate MC methodology. A comparison of experimental and simulated spectra for Co-60 gamma source is shown in Fig. 18. The percentage deviations observed between the measurements and MC simulations for photo peak efficiencies range from -12 % to +7 %.

C) FAST REACTOR STUDIES

C.1) Steady State Reactor Physics Analysis of 500 MWe Prototype Fast Breeder Reactor

The Prototype Fast Breeder Reactor (PFBR) is in advanced stage of construction and expected to attain criticality in near future. PFBR is a 500 MWe sodium cooled, pool type, mixed oxide fuelled reactor. Owing to the significance of this reactor being the first of the kind designed indigenously in the country, SRI has taken keen interest in the evaluation of the reactor physics aspects of PFBR in order to support the regulatory decision making.

As a first step, the fuel loading scheme and first approach to criticality of PFBR is analyzed independently and compared with that of the designers. Initial fuel loading of PFBR is envisaged to be carried out in several batches, each batch containing a fixed number of fuel assemblies.

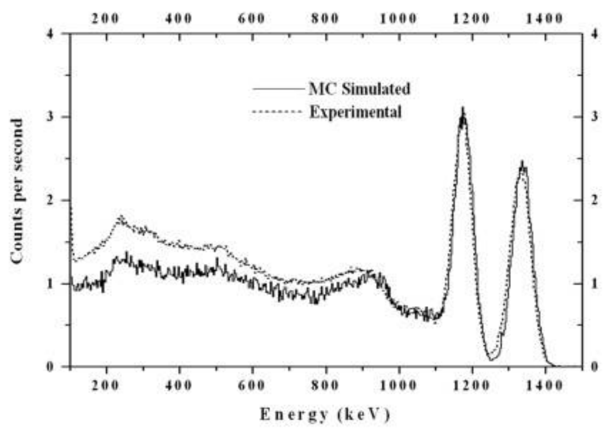


Fig. 18 : Comparison of experimental and simulated spectra of Co-60 gamma source

Criticality calculation is carried out by Monte Carlo modeling during each batch of fuel loading to verify the neutron multiplication factor (k-eff) for ensuring safety. Consequently, the absorber rod worth and shutdown margin during each batch is also estimated. Further, the individual control and shutdown system worth, i.e., the Control and Safety Rods (CSR) and the Diverse Safety Rods (DSR) worth at the beginning of life core are estimated by pin-wise modeling of the absorber elements for better accuracy. Prior to the PFBR calculations, the computational scheme and cross-section data set used were validated by analyzing the Russian BN-600 fast reactor benchmark and FBTR experimental data. A map of the nominal core of PFBR is shown in Fig. 19.

The important reactor safety parameters like the prompt neutron life time, effective delayed neutron fraction and reactivity coefficients are estimated for the nominal core of PFBR. However, in the event of partial power operation of the reactor, these parameters will be estimated once again.

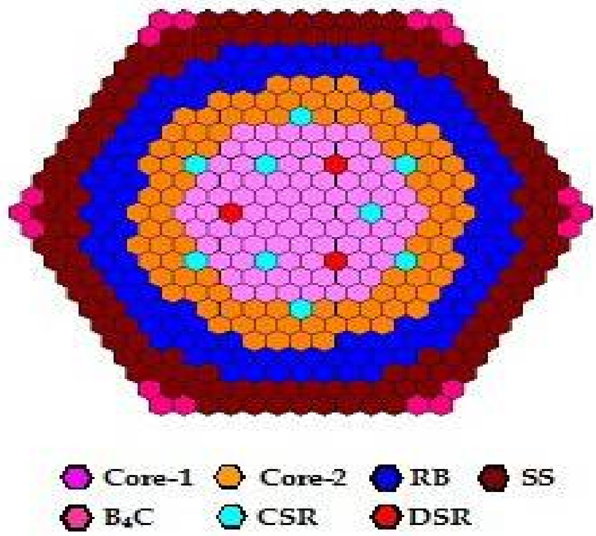


Fig. 19 : Nominal core map of PFBR

C.2) Reactor Physics Analyses of Metal Fuelled Fast Reactors

The overall nuclear power growth in India will mostly depend on the faster growth of Fast Breeder Reactors (FBRs). Metal fuel cycle is the appropriate choice for achieving a faster growth of FBRs, which offers higher breeding gain and lower fuel doubling time. In view of this, several metal fuelled fast breeder reactor (MFBR) designs with varied core compositions have been conceptualized. The MFBR designs being proposed in the Indian context are of 500 MWe and 1000 MWe capacities. The core design of the 500 MWe MFBR is similar to that of the Prototype Fast Breeder Reactor (PFBR) while the 1000 MWe design is slightly different. The metal fuels considered in the MFBR design are alloys of Uranium-Plutonium-Zirconium with varied concentration of zirconium. The selection of U-Pu-Zr alloy was the fundamental reason for the superior safety characteristics of experimental breeder reactor EBR-II.

As a research work, different metal-fuelled cores have been analyzed and the reactor physics parameters are compared with that of MOX fuelled PFBR. The effect of the number of rows of radial blankets and reactor size on breeding ratio is also studied. Normally the zirconium fraction in the metal fuel is 6-10%. However from better breeding consideration the MFBRs analyzed in this study contain U-Pu-Zr(6%) fuel. In this study the burnup analysis of a MFBR has also been carried out. The core configurations of a typical 1000 MWe MFBR is shown in Fig. 20.

The study indicates that metal fuelled FBRs offer high breeding ratio which can be around 1.5 for a 1000 MWe design with U-Pu-Zr(6%) fuel. Even though it is well known that higher Zr content in the fuel alloy gives better structural integrity, it found that by reducing the Zr content the breeding gain improves. The important safety concern in metal fuelled FBRs is the high positive sodium void worth. Despite the inherent safety being established through transient analyses in MFBRs, methods to reduce the higher sodium worth should be worked out in future for ensuring safety in such reactor systems. Another important finding of the study is that, the Pu vector is more stable with burnup in metal core compared to that in oxide core. Since multiple fuel recycling is interlinked with FBR growth, this behavior indicates that it is perhaps easier to maintain constant Plutonium isotopic composition (Pu vector) with multiple recycling in metal-fuelled FBR as compared MOX fuelled ones.

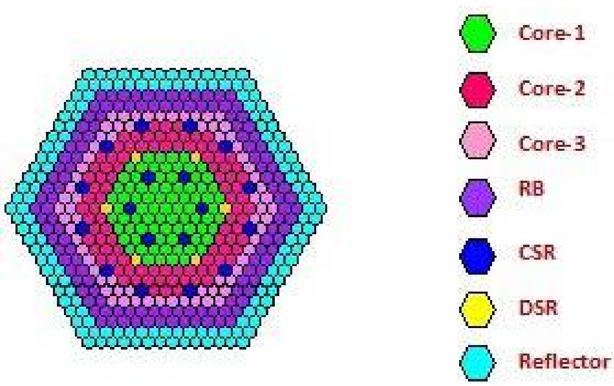


Fig. 20 : Typical core configuration of a 1000 MWe MFBR

C.3) Thorium Utilization in Fast Reactor

Utilization of the vast thorium reserves in India has been pursued vigorously for establishing energy sustainability through nuclear power. It is well known that the FBR stage is crucial for breeding U-233 from Th-232, which is required for launching the 3rd stage of Indian nuclear power programme. However studies have revealed that introduction of thorium in MOX fuelled fast reactors would not be beneficial as the breeding gains would be low. For effective utilization of thorium, the best choice is a fast reactor fuelled with metal fuel which can attain higher breeding gain. As a R&D topic, thorium was introduced in the blankets of the previously described 1000 MWe MFBR core (with 2 rows of radial blanket) in various configurations and the effects on the core parameters were determined. In the first case, thorium metal was introduced in both the radial and axial blankets of the reference core replacing the depleted uranium metal. In the second case, the two rows of radial blanket were replaced with thorium while the axial blanket was uranium. In the last case, the outer row of radial blanket was replaced with thorium while the inner row of radial blanket and the axial blanket were uranium. For estimating the breeding ratio, in each case the core excess reactivity was maintained at the same level, which was obtained by adjusting the core enrichments. The results of the study indicate that by introducing thorium in the blankets, the breeding ratio of the MFBR does not change drastically and remains in the range of 1.4 to 1.5. The Pu inventory in the core also remains almost same as that of the reference core because the core enrichments are only slightly changed. It is also observed that when the entire uranium blanket is replaced with thorium, the sodium void worth decreases which is an advantage from the safety point of view.

C.4) Fast Reactor Core Dynamics and Stability Analysis

As an integral part of safety evaluation, it is necessary to analyze the dynamic behavior of fast reactors and establish their stability range under different reactivity insertion conditions. Attempts are made to develop indigenous computational tools in MATLAB environment for studying the transient and dynamic behavior of fast reactors. For characterization of stable dynamic regimes of fast reactors, a mathematical model is developed and applied to FBTR and PFBR. The model based on control system theory incorporates linearized neutron kinetics and point thermal hydraulics in state-space. From the state-space equation, time and Laplace domain analysis can be carried out easily.

The implemented procedure was validated by reproducing the results reported for a lead cooled fast reactor design. Subsequently, the model was applied for simplified Unprotected Transient Over Power (step reactivity insertion) and Unprotected Loss of Flow (step flow reduction) incidents in FBTR and PFBR. Further, stability analysis was carried out by root-locus method for different power levels, flow levels and reactivity coefficients. The limiting values of the reactivity coefficients for stability in both FBTR and PFBR were also evaluated.

C.5) Study of Uncontrolled Withdrawal of CSR Incident in PFBR

As a part of regulatory evaluation, the incident of uncontrolled withdrawal of the most reactive Control and Safety Rod (CSR) in the 500 MWe Prototype Fast Breeder Reactor (PFBR) is analyzed. This important enveloping DBE has been analyzed at low and high powers to demonstrate the availability of SCRAM parameters to trigger the dropping of absorber rods in the two shutdown systems. During the incident, reactivity is continuously added to the reactor. Modeling of the incident is done by solving the reactor kinetics equation with instantaneous reactivity feedback by considering static power coefficient. For solving the stiff coupled differential equations, the algorithm of numerical differentiation formulas is used from MATLAB ODE solver. The algorithm and computer program used for the analysis have been validated against standard benchmark results. The incident is analyzed under three categories of reactivity insertions, (a) for constant reactivity ramp rate of 4 pcm/sec during low and high power, (b) for different reactivity ramp rates (2, 4, 8 and 15 pcm/sec) at a constant power, (c) for different reactivity ramp rates during reactor startup. The power and reactivity profiles for reactivity insertion rate of 4 pcm/sec at different initial powers, which is anticipated during the incident are shown in Figs. 21 and 22 respectively. Results obtained from the transient study indicate that this incident can be terminated safely by appropriate trip settings provided in the design of PFBR.

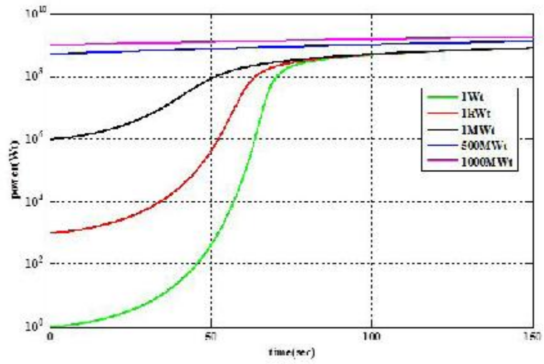


Fig. 21 : Power profile for constant reactivity insertion rate with different initial powers

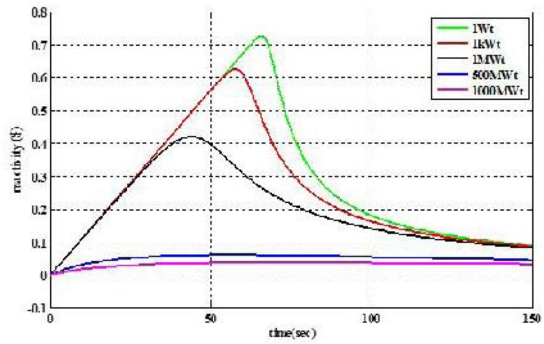


Fig. 22 : Reactivity variation for constant reactivity insertion rate with different Initial powers

3.2 RISK ASSESSMENT AND GIS APPLICATIONS

Introduction

In addition to the deterministic analysis, Probabilistic safety assessment (PSA) is being used as part of the decision making process to assess the level of safety in Indian nuclear power plants. To support AERB activities in this area, PSA and reliability has been identified as one of the research activities at SRI. Some of the important activities in PSA include regulatory review of PSA documents, passive system reliability analysis, software reliability, external events PSA studies, multi-unit risk assessment, etc.

Remote sensing and GIS (RS-GIS) based activities are also undertaken to support AERB in environmental assessment and in deriving guidelines for emergency management during accidental conditions. Towards this, a state-of-art RS-GIS laboratory is established in SRI with necessary infrastructure facilities. Development of baseline data to independently carry out environmental impact assessment in Nuclear Power Plant sites, validation of satellite derived sea surface temperatures, studies related to application of RS-GIS in atmospheric dispersion and tsunami inundation modelling, development of decision support systems for emergency management, etc. are some of the important research activities being carried out. Further, collaborative projects with premier academic institutions such as Anna University, Bharathidasan University are in progress.

The details of activities carried out in above areas are provided in subsequent sections.

A) RISK ASSESSMENT STUDIES

A.1) Accomplishment of Seismic Re-evaluation of FBTR

Seismic evaluation of a nuclear installation is a major safety concern. With the changing seismic design and safety requirements, it is also important to re-evaluate existing nuclear plants. In this context, a seismic re-evaluation exercise of a fast breeder test reactor (FBTR) at Kalpakkam is carried out.

The safety objectives identified for re-evaluation are (i) Safe shutdown of the plant (ii) maintaining in safe shutdown condition, (iii) long term decay heat removal, and (iv) containment of radioactivity. The methodology adopted for seismic PSA of FBTR is depicted in Fig.1, which has primarily five stages. First stage is the site specific seismic hazard assessment. Second stage covers the safety analysis, which includes characterization of initiating events, accident sequence analysis by means of logic tree for each seismic induced initiating event, development of fault trees for each primary and support system identified in the logic tree of accident sequences. Impact of human errors is modeled in this stage. One of the outcomes of this stage is identification of seismic structures, systems and components whose seismic capacity is to be determined. As-built information along with operation history and plant walk down provide necessary input for the activities of this stage. Next stage involves with determination of components' seismic fragility and capacity either

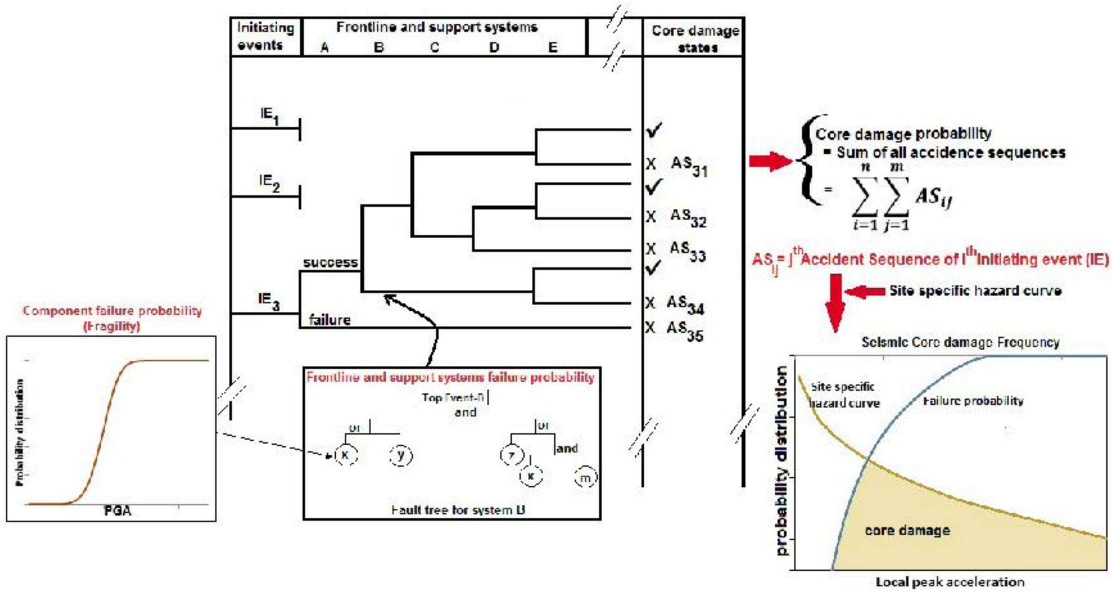


Fig. 1 : Schematic of SPSA methodology

by direct method or by indirect method. The direct method applies analytical approach and testing; while, the indirect method is an experienced based method of in-plant evaluation with the help of plant walk down and from which the easy fixes are established. In the fourth stage, the system model analysis is carried out to calculate the plant seismic fragility and/or capacity from the component fragility or capacity. Finally, the plant seismic risk assessment is done in the fifth stage to derive the seismic core damage frequency of the reactor by convoluting the plant seismic fragility over the seismic hazard curve of the site.

While undertaking this exercise, some deviations were taken from and modifications incorporated in the present practice of SPSA of an NPP. The events caused by a seismic event are generally clustered into the bins of different acceleration. In the SPSA of FBTR, the entire seismic event was considered as a single initiating event. The review basis ground motion parameters were derived by rational maximization of the site specific seismic hazards. As there is not much experience in SPSA of a fast reactor, the SSSCL was derived from the accident sequences. System fragilities are estimated using the cut sets of fault trees followed by estimation of event tree fragility. Fragility of the components was derived adopting a generic approach established by summarizing the existing methods. Finally, plant fragility is evaluated for different values of acceleration to get a fragility curve of the plant and by convoluting it over the seismic hazard curve seismic core damage frequency of plant is obtained. A sensitivity analysis was conducted to identify the key contributors to the core damage frequency, which provides the insight in phasing of the retrofitting activity.

A.2) External Flooding Probabilistic Safety Analysis of PFBR

Nuclear power plants are designed to possess a high level of reliability against a gamut of internal and external events, generally called as design basis events. Redundancy is one of the various principles adopted to achieve high level of reliability. However, external events pose a definitive challenge to redundancy, solely due to its ability to induce common cause failures. In comparison to other external event PSA like earthquake and fire, flood PSA received less attention over the years. This emerged from the perception that floods are less likely than fires and earthquake to induce accidents that contribute significantly to the overall risk of nuclear power plant. Conducting a full-fledged external flood PSA for Indian NPP was under consideration by AERB for some time. Subsequent to the Fukushima accidents AERB, in consultation with IGCAR, mooted a proposal to conduct a demonstration flood PSA for PFBR site. The scope of this activity includes accomplishing external flood PSA for PFBR by carrying out the necessary activities like probabilistic external flood hazard analysis of PFBR site, system modeling and quantification, external flood fragility assessment and necessary plant walkdowns. The outcome of the exercise would be a methodology of external flood PSA with pilot application to PFBR.

Probabilistic safety analysis for external flooding is performed along similar lines as that for other external events. The major activities include flood-hazard analysis, fragility and vulnerability evaluation, plant and system analysis and a release-frequency analysis.

As part of the flood-hazard analysis, evaluation of the frequency of occurrence of different external flood events such as storm surge, rainfall and tsunami based on site specific probabilistic evaluation are carried out. Tsunami data collected from the east coast of India for a period of 273 years is analysed to estimate the probability of exceedance of different run up heights (Fig.2). Plant walkdowns followed by system analysis are carried out to identify list of structures, systems and components required for safe shutdown and for long term decay heat removal. The next task is to compute component fragility to estimate system flood fragility and estimate the core damage frequency due to external flooding thus accomplishing the task of external flood PSA. The work is in progress.

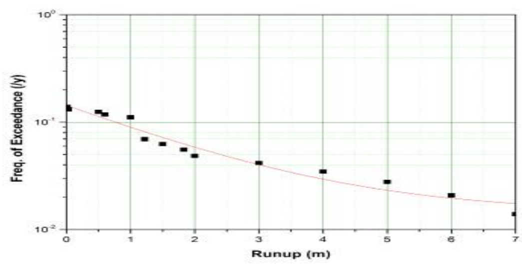


Fig. 2 : Run-up height based on tsunami events in the region for a period of 273 years (1737-2010)

A.3) Hybrid approach for estimation of Software Reliability in Nuclear Safety Systems

The increasing use of computer based systems for safety critical operations in nuclear applications demands a systematic way of estimating software reliability. The high reliability requirements of safety critical software systems make this task imperative as well. Compared to general purpose systems, software designed for safety critical applications are smaller and focussed, robust and have in-built fault tolerant features, designed with defense in depth, meant to fail in fail-safe mode and are expected to have low failure rates. Software systems in nuclear reactors are classified into three categories based on their importance to safety, viz., safety critical, safety related and non-nuclear safety systems. For each category, AERB issues guidelines on best practices in software requirement analysis, defense in depth design, safe programming practices, verification and validation processes etc. in line with international practices.

The best way to ensure that the software used in a safety critical system meets a required reliability is through formal verification, a process of proving certain properties in the designed algorithm, with respect to its requirement specification written in mathematical language/notation. Unfortunately, exhaustive formal verification is not always feasible due to the difficulties involved such as state space explosion and difficulties involved in practical application of formal methods. Also, a major assumption in formal verification is that the requirements specification captures all the desired properties correctly. If this assumption is violated, the formal verification becomes invalid. Moreover, software failures are mainly caused due to design faults and not due to wear off. Such design faults are often difficult to visualize, classify, detect and debug. Unlike hardware reliability, the software reliability is not a pure function of time and hence the definition of software reliability with respect to time is arguable. In addition, the probabilistic nature of software reliability is due to its operational profile and the difficulty in detecting infeasible paths in the software. Typical characteristics of software demand an approach different than in hardware systems. For a reliability estimation of the safety critical software, software testing seems to be the most suitable method. However, in this approach, the amount of time required in testing or demonstrating ultra-high reliability is in-feasible. Software testing with large number of test cases without analyzing the quality/ effectiveness of test cases, cannot give confidence on the reliability estimate.

Although considerable research has been performed, standard methods for software reliability estimation are not reported. Most of the problems appear mainly due to uncertainty involved in reliability parameters such as time to failure, time between failures, number of faults identified, etc. and in identifying the factors such as software complexity, difficulty in identifying suitable metrics, difficulty in exhaustive testing and difficulty in quantifying effectiveness of test cases, that contribute to software reliability estimation. The widely used black box models (also called reliability growth models) are influenced by hardware reliability modelling techniques and have assumptions that are not suitable for safety and mission critical systems. For example,

1. There are fixed number of faults in the software being tested.
2. No additional faults are introduced when a bug found is eliminated.
3. Each fault has the same contribution to the unreliability of the software; and software with fewer faults is more reliable than one with more faults.
4. The probability of two or more software failures occurring simultaneously is negligible.
5. Enough and accurate software failure data is available for analysis.

In the present study, a theoretical approach that combines results of software verification and testing to quantify the software reliability in nuclear safety systems is proposed. In this approach, a method for generating efficient test cases, ensuring adequacy of software testing using appropriate software metrics such as Modified Condition Decision Coverage (MC/ DC) and Linear Code Sequence and Jump (LCSAJ) coverage and mutation testing are suggested.

The test cases are generated through techniques such as model based testing, controlled random number generation, equivalence partitioning and boundary value analysis. The generated test cases are verified by checking against functional specification, invariants and safety properties. The test cases which satisfy these conditions are termed as verified test cases. Redundant test cases which follow the same path of executions are removed. Test coverage is calculated as a weighted average, to provide importance to large, complex, and frequently called functions:

$$Test\ Coverage = \frac{\sum_i w_i t_i}{\sum_i w_i}$$

Where t_i, the conservative test coverage achieved for each function during system testing = minimum(LCSAJ, MC/DC. Statement coverage)

And w_i, the weight assigned to each function = No. of statements x cyclomatic complexity x frequency of function call

Mutation testing is a fault injection technique, where realistic faults are induced intentionally into the source code. The fault induced program is known as a mutant. The proposed approach requires a set of single fault (first order) mutants. The result of mutation testing is the mutation score, defined as the ratio of number of mutants killed by the test cases to the total number of mutants generated. In this process, a simplified method for automatic detection and elimination of equivalent mutants is proposed. Test adequacy is measured as the product of mutation score and test coverage.

By generating large number of mutants, and ignoring all the unkilled mutants, the reliability is estimated as:

$$Reliability = Test\ adequacy \times \frac{No.\ of\ times\ at\ least\ one\ of\ the\ verified\ test\ cases\ failed}{Total\ no.\ of\ mutants\ killed}$$

The advantage of this approach is its simplicity, but its results could be biased when estimating reliability for a highly verified software, i.e.: If the mutation testing is not effective enough, then large number of verified test cases may incorrectly lead to a higher reliability estimate. Also, it is difficult to integrate operational profile into the approach. This approach is more suitable for non-safety applications, but may also be used for systems important to safety to get an initial/quick approximate estimate of the reliability.

Another similar approach based on the principle that, if in a given program, reliability of an execution path p is known, then other paths in the program sharing code with the path p also share the reliability of path p. For example: in Fig. 3, a program has four paths p_1, p_2, p_3 and p_4; and the paths p_3,p_4 share reliability of p_2. If the reliability of path p_2 (i.e.: R_2) is known, then the reliability of any path pi (i.e.: R_i) can be estimated by:

$R_i = R_2 \times$ (fraction of code shared between pi and p_2)

The fraction of code shared between paths is

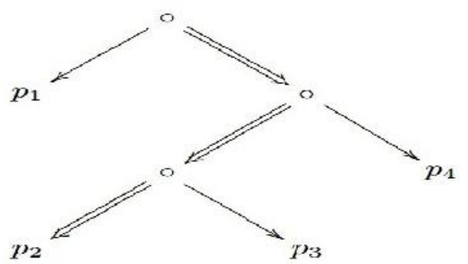


Fig. 3 : Paths in a program (⇒ indicates a path whose reliability is known)

estimated statistically through mutation testing, by injecting faults in paths for which reliability is unknown (e.g.: path p_3). For example: in (Fig. 4) the first injected fault causes the test cases running through paths p_2, p_3, and p_4 to fail; whereas the second injected fault fails test case running through path p_3. If several such single fault (first order) mutants are generated, and are tested against the

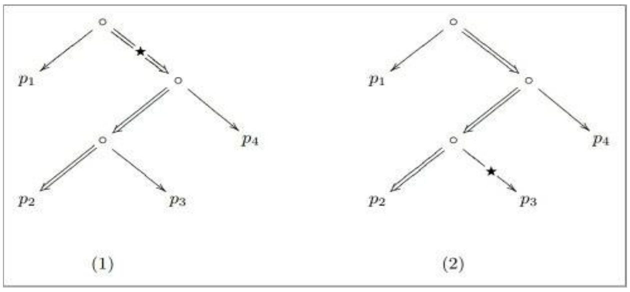


Fig. 4 : Faults induced in path p3 (★ indicates a induced fault)

test cases, then the fraction of code shared between paths pi and p_2 may be estimated by:

$$Fraction\ of\ code\ shared\ between\ p_i\ and\ p_2 = \frac{F_{i2}}{F_{22}}$$

where F_{i2} is number of times test cases running through path p_i has failed, given that a fault was induced in path p_2; and F_{22} is number of times test cases running through path p_2 has failed, given that a fault was induced in path p_2.

In real life applications though, an un-verified path may share code with several other verified paths, and may even form cycles. To address such issues, a systematic way to estimate the fraction of code shared among paths and the software reliability is established through an indigenous tool. Unlike the first approach, here integration of the operational profile in the reliability estimate is possible and it ensures that un-verified test cases fail during mutation testing; thus eliminating any bias present due to large number of verified test cases. This property makes the reliability estimate realistic and more suitable for systems important to safety.

Traditional reliability models assume availability of accurate and adequate software failure data, which is often difficult to collect. Also, for a newly built plant with no failure history, the software reliability estimation methods do not apply and in such situations, the proposed approach can be adopted for an initial estimation of software reliability which is represented as a function of three variables: test adequacy, the amount of software verification carried out and the reusability of verified code in the software. The proposed framework has shown how software verification can be combined with software testing to assess a realistic estimate of the software reliability and is expected to aid in the quantitative risk assessment and in the licensing process. Considering the fact that all safety–critical software undergo rigorous testing and verification to ensure correctness; the proposed approach is expected to aid regulators in licensing computer based safety systems in nuclear applications.

A.4) Common Cause Failure Analysis for Engineered Safety Systems Using Alpha Factors Obtained by Mapping Technique

For ensuring adequate safety in nuclear power plants (NPPs), the engineered safety systems are deployed with redundancy which is considered as a fundamental technique for fault tolerance. However, in redundant systems, common cause failures (CCFs) are considered to be the major contributor to risk and therefore quantifying CCF is essential to demonstrate the reliability of a system. Over the years, various methods such as Beta factor, Multiple Greek Letter, Binomial Failure Rate, Alpha factor have been developed for probabilistic assessment of common cause failures in NPPs.

In the present study, alpha factor model is applied for the assessment of CCF of safety systems deployed at Indian nuclear power plants. A method to overcome the difficulties in estimation of the coefficients viz., alpha factors in the model, importance of deriving plant specific alpha factors and sensitivity of common cause contribution to the total system failure probability with respect to hazard imposed by various CCF events is highlighted. An attempt is made to extend the approach as described in NUREG/CR-5500 to provide more explicit guidance for a statistical approach to derive plant specific coefficients for CCF analysis especially for high redundant systems. A comparison of Alpha factor method and Beta factor method is also presented taking insights from the case studies of safety systems of the Indian Nuclear Power Plants.

The alpha factor model estimates the CCF frequencies from a set of ratios of failures and the total component failure rate. The parameters of the model are

Q_T ≡ total failure probability of each component (includes independent and common-cause events)

$\alpha^{(m)}_k$ ≡ fraction of the total probability of failure events that occur in the system involving the failure of k components in a system of m components due to a common-cause.

The CCF basic event equation for any k out of m components failing in case of staggered testing is given by [2]:

$$Q_{CCF} = Q_T \sum_{i=k}^{m} \begin{pmatrix} m \\ \frac{i}{m-1} \\ i-1 \end{pmatrix} \alpha_i^{(m)} = Q_T \sum_{i=k}^{m} \binom{m}{i} \alpha_i^{(m)}$$

where:

$\alpha^{(m)}_i$ = the ratio of i and only i CCF failures to total failures in a system of m components

m = the number of total components in the component group

k = the failure criteria for a number of component failures in the component group

Q_T = the random failure probability (total)

Q_{CCF} = the failure probability of k and greater than k components due to CCF

A technique has been proposed by NUREG/CR-5485 technique for CCF analysis using 'event impact vector'. An impact vector is a numerical representation of a CCF event and is classified according to the level of impact of common cause events. In this technique, the impact vectors are modified to reflect the likelihood of the occurrence of the event in the specific system of interest. This method is also known as mapping. The mapped impact vectors are finally used to arrive at alpha factors.

The number of events in each impact category (nk) is calculated by adding the corresponding elements of the impact vectors.

$$n_k = \sum_{j=1}^{n} P_k(j)$$

where: $P_k(j)$ = the k^{th} element of the impact vector for event j, and n is the number of CCF events.

Finally, the alpha factors are estimated using the $\alpha_k^{(m)} = \frac{n_k}{\sum_{k=1}^{m} n_k}$ expression:

The application of alpha factor technique in CCF analysis is further demonstrated with two varied real applications for Indian nuclear power plants in the following section. A MATLAB code has been developed to estimate the alpha factors and then compute CCF contribution to total failure probability. The case studies are carried out with the help of same code as it is capable of handling various redundant configurations.

Some of the conclusions made from the study are

1. The alpha factor model realistically assesses the contribution of each of the CCF event based upon subjective assessment of a constant ρ, conditional probability of each component failure given a shock.
2. Alpha factors are found to be less sensitive to change in the value of mapping up beta and this sensitivity further reduces as more number of components are added to the system.
3. Contribution of CCF events to total failure probability is found to be less sensitive to the value of mapping up beta but it is found to be highly sensitive to the change in success criterion for the system.

4. The use of alpha factors is found to be highly suitable, especially for large redundant configuration and with stringent success criteria. In such cases, the use of beta factor model yields highly repressed estimates, thereby underestimating the risks imposed by common cause events.

A.5) Analysis of Damper Control Systems and its Impact on SGDHRS Reliability

Safety Grade Decay Heat Removal (SGDHR) system of Prototype Fast Breeder Reactor consists of four independent but identical loops each having 8 MW heat removal capacity at a hot pool temperature of 820 K. Out of this, two loops are essential for first 12 hrs and subsequently one loop is enough to provide the required decay heat removal to ensure proper cooling of the core and limit the main vessel, internals and bulk sodium temperatures within safe limits. SGDHR system is a completely passive system except for the dampers on the air side.

The objective of this study is to model the relay logic, actuators, valves and power supply for both the electrical and pneumatic dampers of SGDHR system to compute the probability of failure on demand in opening of dampers and subsequently its impact on SGDHRS reliability.

The damper opening logic system of SGDHR is modeled using Fault Tree Analysis from the ladder logic wherein the immediate failures that could lead to the system or subsystem component failure are logically developed using logic gates. Common cause failure analysis is carried out using beta factor model. Two cases were analysed viz., i) Failure probability of electrical and pneumatic dampers in a loop to assess their individual performance in a loop ii) Failure probability of the complete SGDHR damper system with and without common cause failures of motors, class II system, valves and relays. Significant contribution of CCF of redundant components such as motors, class II power supply and valves is seen in the failure probability of SGDHR dampers. Pneumatic damper is found to be more reliable as compared to electric damper. The probability of failure of SGDHR dampers when CCF is modelled is arrived at based on the generic component failure data and is found to be 2.45e-11/de and without accounting for CCF, SGDHR damper failure probability is found to be 1.85e-22/de.

A.6) Risk Assessment of Multi-Unit Nuclear Power Plant Sites against External Hazards

Most of the nuclear power producing sites in the world houses multiple units/plants. Such sites are faced with hazards generated from external events: earthquake, tsunami, flood etc, and can threaten the safety of nuclear power plants. Further, risk from a multiple unit site and its impact on the public and environment was evident during the Fukushima nuclear disaster that took place in March 2011. For a single unit site, probabilistic risk assessment technique identifies the potential accident scenarios, their consequences, and estimates the core damage frequency that arise due to external events. This challenging task becomes even more complex when the safety assessment of a multiple unit site with external events that has the potential to generate one or more correlated hazards or a combination of non-correlated hazards are to be modeled. As of now, there exists no established approach or methodology to estimate the risk from a multi-unit nuclear power plant site due to external hazards.

The study attempts to develop a methodology for estimating risk contribution from a multi-unit nuclear power plant site arising from external hazards. It also highlights the importance of unique risks present for multi-unit sites arising from shared system, common cause failures, failure correlations, cliff-edge effects, etc. attributable to different external hazards. For the estimation of risk, accident likelihood is to be measured in ‘events per multi-unit year’ instead of ‘events per reactor year’. The term site core damage frequency (SCDF) is used which is described as the frequency of at least single core damage per site per year. SCDF is overall risk associated with the site obtained by means of integrating the risk from core damage in one or more units at the site.

The methodology is described as follows: Firstly, all possible site specific external hazards that can affect the multiple units of nuclear plant site are identified. The hazards can also be a result of correlated failures. They are further categorized as either definite or conditional. Definite external hazards are the hazards that will always affect multiple units and conditional external hazards are those which only under certain circumstances will affect multiple units. Initiating events are then identified and categorized for all external hazards and subsequently event trees and fault tree models are developed for further analysis. The integrated approach explained in earlier sections for multi-unit safety assessment considering all categories of hazards is depicted in Fig. 5.

Key issues which need to be addressed while modeling event trees and fault trees are given below:

1. Shared systems or connections: Shared systems or connections are those links that physically connect structure, systems and components

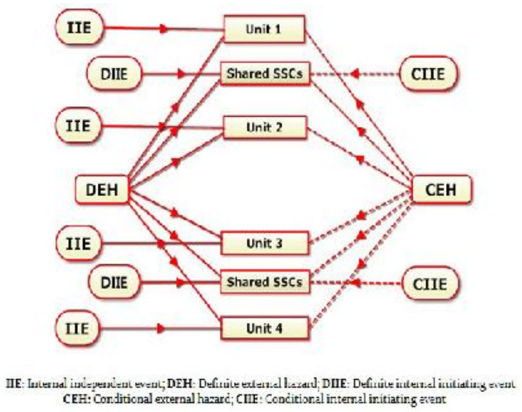


Fig. 5: Risk assessment in a multi-unit NPP site

(SSCs) of multiple units. These connections are further categorized in three different sub-classes. The first subclass is a single SSC. This occurs when multiple units are dependent on a single SSC for simultaneous support. The second subclass is time sequential sharing or cross-connected SSCs. This is a SSC that is able to fully support any single unit but it is not capable of simultaneously supporting multiple units. The third subclass is standby sharing. Standby sharing occurs when multiple units share a standby or spare SSC which can only be used to support a single unit.

2. Identical components: These are those components that have the same design, operation, and operating environment in multiple units. It means that the components are designed, installed, and maintained nearly identically and are operated in same manner making them susceptible to traditional common cause failures that are considered for single units.

3. Human dependencies: Human dependencies arise when human interaction with a machine affects multiple units. This can be an operator, a maintenance team member, a member of an installation crew, or the like.

   Human dependencies are classified under two subclasses: pre-initiating event and post-initiating event actions. Pre initiators refer to those human actions that occur before an event and have a potential to create latent conditions.

4. Proximity Dependencies: Proximity dependencies arise when a single environment has the potential to affect multiple units of the site. This common environment may be either intentionally or unintentionally get created.

SCDF for each hazard is ev aluated from the Boolean expression obtained from all the event tree accident sequences of the corresponding external hazard. Finally, the total core damage frequency of the multi-unit site is obtained as the sum of all SCDF from external hazards.

Site core damage frequency, SCDF = $\sum_{i=1}^{nunits}\sum_{j=1}^{5}\sum_{k=1}^{m} CDF(i,j,k)$

where,

- i denote the number of simultaneous core damages
- j denote the category of hazard or event
- k denote the type of hazard in jth category
- m denote the total number of types of hazard in jth category.

Therefore, CDF(i,j,k) denotes the frequency of i number of simultaneous core damages due to j^{th} category of hazard type k;

j=1 refers to definite external hazards for the site
j=2 refers to conditional external hazards for the site
j=3 refers to definite internal events for the site
j=4 refers to conditional internal events for the site
j=5 refers to internal independent events considering for all units

B) REMOTE SENSING AND GIS STUDIES

AERB has set up a Remote Sensing and Geographic Information System (RS-GIS) facility at Safety Research Institute, Kalpakkam with the objective of generation of digital database on nuclear installations for carrying out environmental impact assessment, creation of information system for emergency management and to promote safety research related to regulatory function. The following studies are carried out as part of RS-GIS activity.

B.1) Environmental Assessment of Kudankulam Nuclear Power Plant site using Satellite Remote Sensing and GIS

A study on the environmental assessment of Kudankulam Nuclear Plant site (KKNPP), Tirunelveli District, Tamil Nadu by employing RS-GIS techniques has been taken up. Under this study, environmental baseline data for 50 km radius around KKNPP on Surface Water bodies, Lithology, Soil, drainage network, Road network, Digital elevation model (DEM) (Fig. 1), land use/ land cover and village map has been generated. These data can be employed

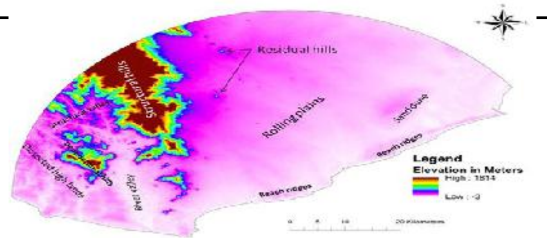


Fig. 1 : DEM showing various morphological features around KKNNP area

to evaluate various environmental scenarios with regard to environmental assessment and emergency preparedness.

B.2) Mapping of Sea Surface Temperature Measurement (SST) around MAPS

Sea surface temperature studies, employing multi-dated satellite thermal infra-red imagery was carried out. The aim of the study is to identify the temporal characteristics of the thermal plume signature around a coastal NPP site due to condenser coolant discharges. The task includes,

a. Baseline study on the prevailing sea surface temperature pattern,

b. Processing and conversion of Satellite data to derive Sea Surface Temperature by employing suitable algorithms and

c. Estimation of difference between actual and measured temperature.

B.2.1) Baseline Study

Prior to the study on satellite based mapping of Sea Surface Temperature pattern, a preliminary baseline survey to understand the existing thermal discharge pattern has been taken up at Kalpakkam site. This work involves in situ measurement of sea surface temperature (skin temperature) at intake and mixing zone i.e. the zone where the heated water/effluent meets the receiving body (i.e. final outlet) and calculation of ΔT. The in situ sea surface temperature at condenser coolant intake and mixing zone are periodically measured by employing YSI multi parameter device during the study period October 2009 to December 2012. Totally, 29 measurements were made at the above locations. The surface temperature is measured at site by dipping the probe into the water body directly. From the difference between mixing zone and intake temperature, ΔT is calculated.

The study indicate that ΔT was low during northerly current than the southerly current as latter was comparatively weaker than the former. It attributed the influence of the long shore current speed in the cooling and dispersion pattern of heated water discharge.

B.2.2) Mapping of Sea surface Temperature using satellite Thermal Infra Red data for the period (2000-2013)

Subsequently to the baseline study, multi-dated satellite Thermal Infra Red (TIR) image assessment has been done to demonstrate the spatial and temporal characteristics of the thermal plume signature and

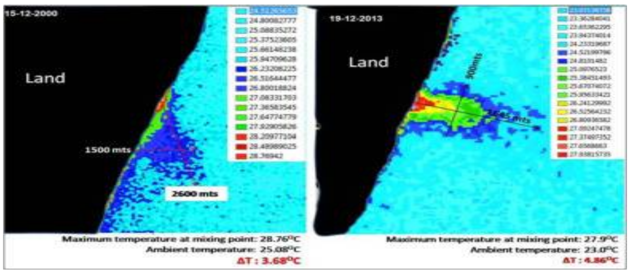


Fig. 2 : Processed satellite thermal infra red data shows the SST pattern around Kalpakkam

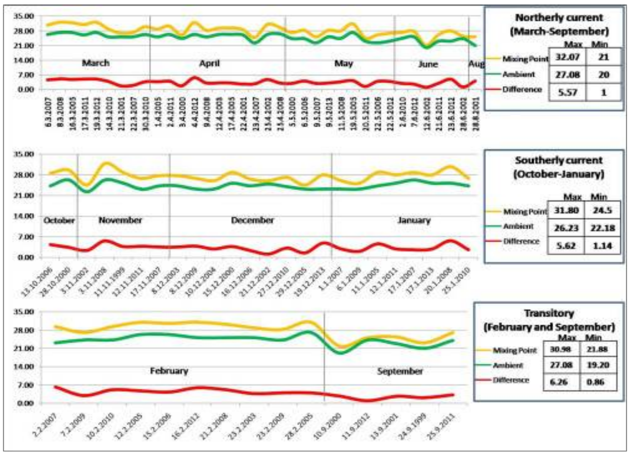


Fig. 3 : Seasonal behavior of the thermal plume

to determine what extent the MAPS condenser discharge influence the temperature distribution of the coastal water around the Kalpakkam coast.

A total of 95 satellite Thermal Infra Red data (2000 to 2013) from Landsat were analyzed to know the thermal plume character with respect to seasons. The downloaded satellite data were processed to convert the Digital Numbers into brightness temperature by using standard formula given in the Landsat user's handbook (Fig. 2). Also, the shape, dimensions and spreading pattern of the thermal plume were observed. The data were classified into three seasons, via. northern, southern and transitory (Fig. 3). The seasonal behavior of the thermal plumes was also observed. From this study, it was observed that the ΔT is in the range of 0.86°C to 6.26°C.

B.2.3) Estimation of difference between satellite derived and insitu measured SST

To calculate the difference between actual and satellite derived temperature (i.e. to validate the satellite derived temperature by comparing 'in situ' measured temperature) along Kalpakkam Sea, a boat expedition corresponding to the day of satellite pass on Kalpakkam has been conducted (Fig. 4). Satellite derived SST was compared with 'in situ' SST and it was observed that the satellite derived SST shows lower than the in situ SST (1.5°C -4.5°C).

Fig. 4 : 'insitu' measurement of sea surface temperature at Kalpakkam Sea

B.3) Mapping of Sea Surface Temperature Measurement (SST) around KKNPP

A satellite derived sea surface temperature data (post commissioning period) for the period July 2013 to February 2014 are generated by processing Landsat 7 and Landsat 8 thermal infra red data. The satellite digital numbers are converted into temperature data by employing image processing software. One such processed data is shown in the Fig. 5. The analysis based on the data observed during a short period on the post commission sea surface temperature shows that the plume spreads over 2 sq.km area and the temperature difference between intake and mixing zone at receiving body ranges from 0.3°C to 3.1°C.

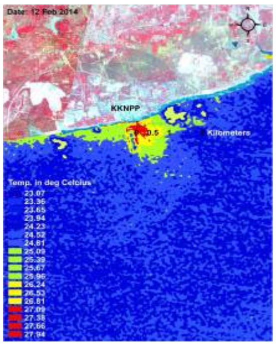


Fig. 5 : Processed LANDSAT ETM+ TIR B62 data showing Sea surface temperature distribution around KKNPP. (Land portion overlapped with visible band imagery)

B.4) Investigation of changes in beach morphology due to Tsunami Protection Wall

The Tsunami, which struck the east coast of India on 26th December 2004, caused huge damage to life, property and environment. Kalpakkam located in the south east coast of India is one of the areas affected by the tsunami. A slew of remedial measures were initiated at Kalpakkam in 2006 and construction of coastal armoring in the form of Tsunami Protection Wall (TPW) of 3.2 km length was one of them. A study was undertaken to assess the impact of this TPW on the surroundings based on periodic measurements of High Water Line (HWL) before and after construction of the wall. As the residential area at Kalpakkam is located between fishing hamlets at northern and southern side, it is necessary to understand the impact of TPW, if any, in the surrounding area and

Fig. 6 : Typical elements of Tsunami Protection Wall constructed along Kalpakkam Township

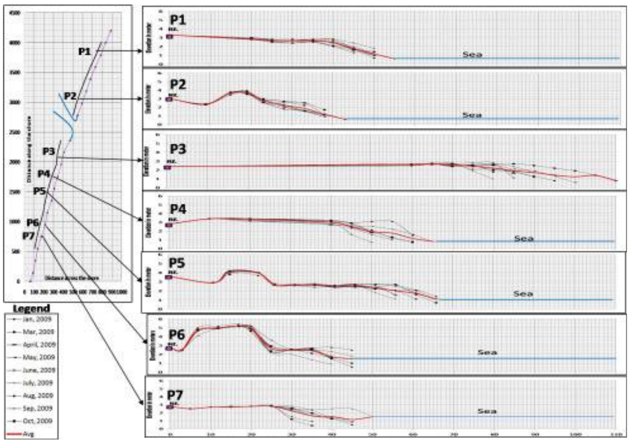


Fig. 7 : Sedimentation pattern (accretion/erosion) observed along Kalpakkam coast

on the fishing hamlets. Towards this assessment, high resolution satellite data such as Quickbird and IKONOS were employed (for the years 2002, 2003, 2009 and 2011) to measure the HWL (Fig. 6). In addition, monthly beach profiles were carried out to measure the sedimentation pattern at selected transects for the year 2009 (Fig. 7).

The average variation in the position of HWL between pre- and post- construction period clearly indicates that the response of beach is uniform along armored and unarmored beach as the TPW was constructed ≈40 meters away from the HWL. The detailed investigations and analysis revealed no significant impact on the beach morphology and sedimentation patterns due to the construction of TPW, within the residential areas as well as at fishing hamlets. There is no significant impact in front, behind and at both ends of the armoring.

B.5) Environmental assessment of combined outfall channel for the PFBR-MAPS on the marine environment

A combined outfall channel (with guided bund and tsunami protection and shore protection wall) was constructed to discharge the thermal effluent of PFBR and MAPS. To assess the environmental impact, high resolution satellite data for the pre and post construction period was analysed. The High Water Line (HWL) digitized from the data was compared and image analysis is performed to identify the changes in the morphology along the Kalpakkam coast due

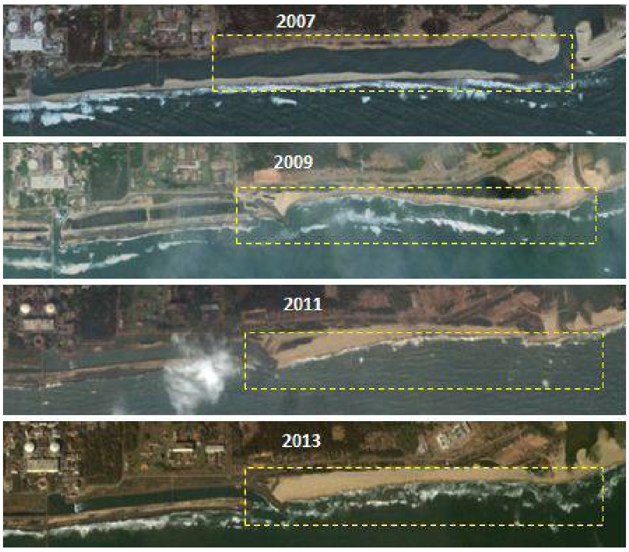


Fig. 8 : Disappearance/disuse of old canal due to construction of new canal at the northern end of the canal

to impact of the wall. The initial observation shows that there is no significant change in the high water line. However, minor changes are observed in the sedimentation pattern both the corner of the wall (Fig. 8). Due to this structure there is no impact is observed in the nearby environment.

B.6) Nuclear Emergency Management Information System (NEMIS)

AERB constituted a committee on nuclear and radiological emergency monitoring to serve as centralized agency to disseminate real time data during an accident at nuclear site. As a first step, the committee recommended development of a user friendly information system for off-site emergency management during nuclear accidents. SRI with the available RS-GIS facility has taken the initiative and has developed the system called NEMIS (Nuclear Emergency Management Information System) for Kalpakkam site. After a detail review of the system, the same will be extended further to all sites.

The information system has been created for the Emergency Planning Zone of the MAPS identified in the Manual on Emergency Preparedness published by Kalpakkam DAE Centre. This includes exclusive zone, sterile zone and emergency planning zone. This decision support system will provide the site specific information and plume details, such as plume direction, areas affected, population to be evacuated, shelter details, etc. The system contain information on population density, village boundary map with socio-economic pattern, sector map (Fig. 9), rallying post, etc. collected from various authorized sources such as census department, environmental survey laboratory, etc. Information is provided under three broad categories, viz., Village information, plume information and emergency planning.

The village information will give basic information such as village name, population and area of the village in square km for 30 km around MAPS. The plume information menu will give information on direction of plume movement and detailed information on the area of impact. The third, emergency planning will give information on the shortest route to reach the rallying posts, the buses required for evacuation, the facility available in the rally points etc. This information system is a Graphical User Interface based customization using VB code integrated with GIS software.

By using this information system, one can visualize the site information, plan the preparedness for off-site emergency management during real emergency as well as drills. The required site specific GIS maps were generated and included in the system.

B.7) Generation of Database on Volcanoes in the vicinity of Indian Subcontinent

The database on volcanoes located in the vicinity of Indian subcontinent (between 400 to 1200 longitudes and between -600 to 400 latitudes) is developed based on a survey of past records and a review of the scientific literature. The data on volcanic events are gathered from scientific and scholarly sources, regional and worldwide databases, individual event reports, and unpublished works. This database is intended for use as ready reference for regulatory purposes.

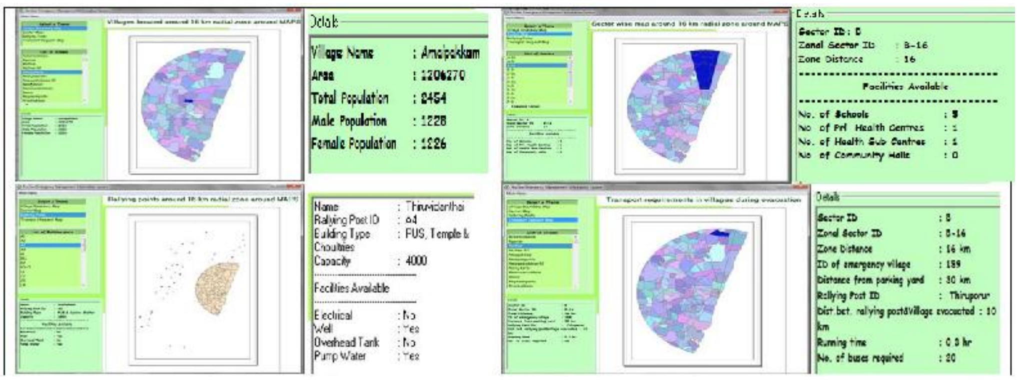


Fig. 9 : Different type of information available at NEMIS

3.3 ENGINEERING SAFETY STUDIES

Introduction

The Engineering Safety Studies at SRI include safety review, analysis, and research in the area of structural and seismic analysis, reactor thermal hydraulics, fire safety, hydrogen safety and radiological impact assessment. The focus in structural mechanics has been mainly on safety assessment of fast reactor components. Core as well as containment thermal hydraulic safety analysis is being carried out as part of safety review work for existing as well as for upcoming PHWRs. Recently, review of TAPS-1&2 safety systems have also been taken up. For these analyses, wherever necessary, appropriate in-house computational tools are also developed and used. In the area of fire and hydrogen safety, both theoretical as well as experimental research programs are underway. Studies in the area of radionuclide migration in geo-sphere and external hazard are pursued using numerical modeling. Further, collaborative projects with leading R&D institutions such as BARC and IITs are also underway. The details of activities carried out in above areas are provided in subsequent sections.

A) STRUCTURAL AND SEISMIC ANALYSIS

A.1) Seismic Margin Assessment for NPP Components

A methodology for seismic margin assessment of NPP components, based on comprehensive and transparent risk and safety assessment (stress test) has been developed for Beyond Design Basis Event (BDBE) earthquake. The developed methodology mainly consists of following steps:

- Development of an integrated model of component in the parametric form.
- Seismic analysis against Safe Shutdown Earthquake (SSE) load.
- Identification of critical locations and margin assessment against predominant failure modes.
- Postulation of circumferential crack at the most critical location.
- Crack sensitivity analysis against important variables.
- Seismic margin assessment against BDBE earthquake using R6 method.

The applicability of the methodology to seismic margin assessment was carried out for Safety Grade Decay Heat Removal (SGDHR) system of PFBR. The objective was to evaluate the seismic margin for BDBE earthquake. It was observed that Sodium-Air Heat Exchanger (AHX) of this system is most critical component due to its unique design features. The integrated finite element analysis of SGDHR system reveals that header-tube junction at the end of inlet header of AHX is the most stressed location in the system during SSE. The analysis result is shown in the Fig.1. The seismic capacity of header-tube junction of AHX was assessed against the fatigue-ratchet, which is the most predominant failure mode and is many times higher than PGA of SSE. For the purpose of analysis, a through-wall circumferential crack is postulated at the most stressed location and capacity assessment of the cracked tube is carried out using the R6 method. The R6 method is an assessment procedure for evaluation of structural integrity of defected structures by accounting two basic failure modes- plastic collapse and fracture. This method requires estimation of two parameters L_r and K_r, where L_r denotes the nearness to the plastic collapse and K_r denotes the nearness to crack growth

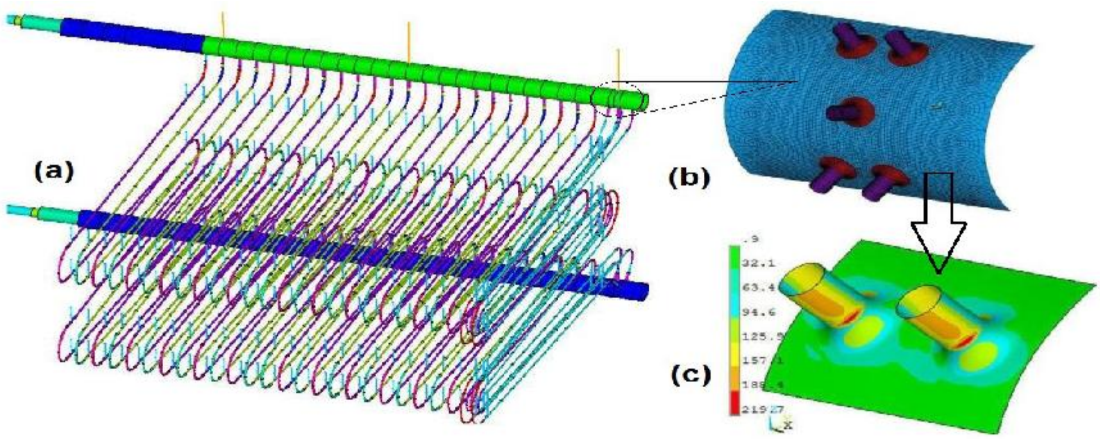


Fig. 1 : (a) FE model of AHX (b) Shell model of header and tube junctions (c) Von Mises stress during SSE

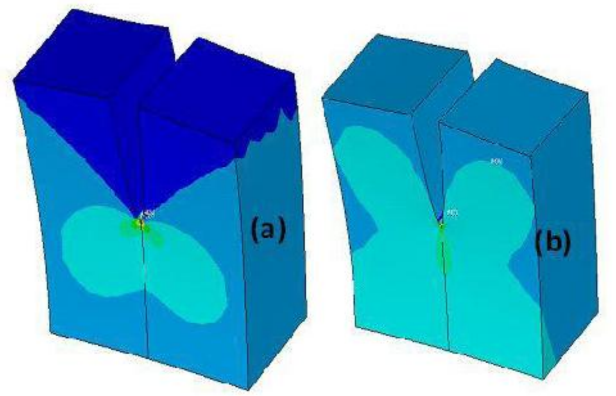


Fig. 2 : Stress profile in the AHX tube under SSE load (a) along the crack growth direction (b) normal to crack growth direction

initiation. The stress intensity factor for evaluation of K_r is estimated from the stress analysis of cracked tube. The stress distribution near crack is shown in Fig.2. The seismic margin is represented in term of load factor (F^L), which is derived from failure assessment diagram developed using the option-2 curve.

A parametric study is carried out by varying the crack angle of the postulated crack. The crack angle is varied from 10° to 110° and F_L is estimated against SSE. As the crack angle increases the assessment point moves closer to limiting load line and F_L decreases. Fig.3 shows that when crack angle reach 105°, then crack will become unstable for SSE load. Therefore, even for large crack of 100°, which is highly improbable as it will get detected at earlier stage, AHX tube will not fail due to SSE load. For the BDBE earthquake the margin is estimated by increasing the Peak Ground Acceleration (PGA) of earthquake. A through wall circumferential crack of 45°, which is a conservative estimate, is taken for the margin assessment. The estimated for the 45° circumferentially cracked tube under SSE load condition is 1.72. Also for fixed crack length L_r and K_r

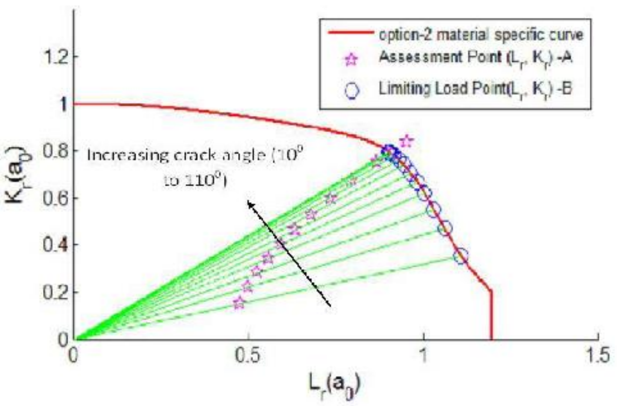


Fig. 3 : Variation of assessment points and limit load points with respect to crack angle.

will increase linearly with respect to PGA. Therefore, it is concluded that even an earthquake with PGA equal to 0.268g, which is 1.72 times the PGA of SSE of PFBR, will not be able to cause instability in the AHX tube for postulated crack of 45°.

A.2) Failure Probability Evaluation of NPP Components

The probabilistic safety analysis of NPP is performed to quantify the risk posed by both intrinsic and extrinsic factors. The basic ingredient of quantitative risk assessment is the estimation of failure probability of safety related components under these factors. Generally, these components are designed based on working stress and resistance factor design methods, which does not readily provide failure probability under various loading states. This necessitates the development of a methodology based on structural reliability theory for evaluation of their failure probability, considering uncertainties in loading states, material strengths and dimensions etc.

In this direction, a method based on Higher Order Response Surface Method (HORSM) is developed for evaluation of failure probability of components. This method is based on response surface approach, which is computationally efficient and gives reasonably accurate estimate of failure probability. It uses the orthogonal Chebyshev polynomial for order estimation of random variables, mixed-order selection and higher order response surface generation. It has been validated against the well known technique of Monte Carlo Simulation (MCS), which is the most accurate of all the available methods but is computationally too intensive for components with low failure probabilities.

Fig.4 shows the comparison of the failure probability estimated from HORSM and other methods like First Order Reliability Method (FORM) and Higher Order Stochastic Response Surface Method (HORSM) for

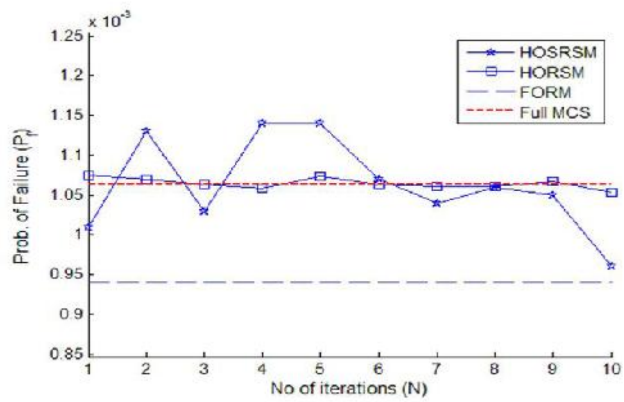


Fig. 4 : Failure probability of rotating disk using 10 different set of random numbers

the limit state based on burst margin of the rotating annular disk. The failure probabilities predicted by HORSM and MCS for ten iterations are fairly close and show the robustness of the method. For wider application, the present method is integrated with a finite element code. Component failure probability can now be assessed under failure modes like plastic collapse, fatigue, as well as instability of a postulated crack.

The feasibility of using this method for actual NPP components was demonstrated by application to the expansion bellow at PFBR RCB penetration. Generally, these bellows have lower design margins than that for other mechanical components and in the event of failure it provides a potential source of leakage path of radioactivity to atmosphere. An analysis of such a bellow, under the operating load, was carried out using a parametric finite element model. The FE model and Von Mises stress range in the bellow inner surface due to thermal expansion is shown in Fig.5. The predominant failure mode observed from the analysis is the low cycle fatigue. The failure probability estimated by HORSM, which required nearly 100 number of deterministic evaluation, is 7.95e-07. The low failure probability

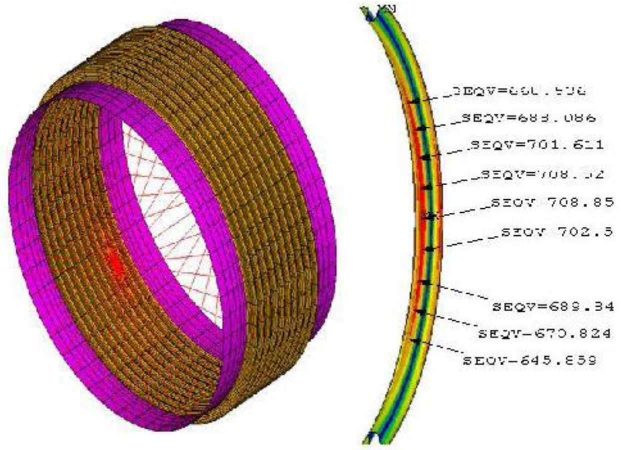


Fig. 5 : FE model and Von Mises stress range of expansion bellow at RCB

indicates that the expansion bellow will be able to maintain its structural integrity with high reliability at the normal operating condition. The present method is computationally efficient and robust, and will be useful for assessing the reliability of important safety systems of NPP during regulatory review process.

B) FIRE SAFETY STUDIES

B.1) Compartment Fire Test Facility

A Compartment Fire Test Facility (CFTF) is being setup with the objective of having a comprehensive experimental program on fire safety, with emphasis on investigating compartment fires and mitigation aspects. CFTF is located within a large experimental hall provided by IGCAR near the Mini-Sodium Fire facility (MINA). The focus is on conducting experiments using combustible fuels/solvents/solvent-mixtures that are employed predominantly in NPPs/fuel reprocessing units. Specific experiments that are of interest from regulatory perspective will also be carried out in this facility. The experimental database will provide inputs for regulatory activities in the area of fire safety and would also be useful for preparation of a monograph on enclosure fires. Further, it will be used for validation of computational fire modeling tools that are being developed and used at SRI. It is planned to initiate round robin exercises with other DAE/research units in this area.

As part of thermal design of the facility, the limiting fire size (Design Basis Fire) and its duration were estimated. Several pool fire scenarios involving various quantities liquid fuels/combustible solvents such as n-dodecane and ethanol were simulated to investigate the effect fuel quantity, ventilation (forced/natural draught) and multi-compartmentalization of the fire room etc and arrive at the design basis fire. Fig. 6(a) shows the snapshot of fire temperature at the central plane of the room at various instants for one case involving n-dodecane fire. Fig. 6(b) shows the comparison of hot gas temperature transients with the standard fire curve (ISO 834 fire curve) for two cases; one involving 28 kg ethanol occupying 4% of the floor area (case-2) and the other involving 7 kg of n-dodecane occupying 1% of the floor area (case-6). Such investigations were also useful for estimating the required fire rating and selection of suitable fire barrier lining for the compartment.

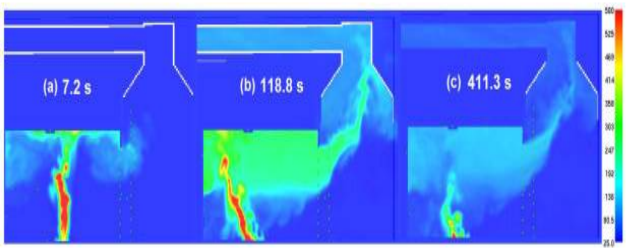


Fig. 6 (a): Snapshots of fire at various instants

The mechanical design, fabrication and erection of all components of CFTF have subsequently been completed. Fig. 6(c) shows the CFTF and its control room.

B.2) Development of Fire Modelling Tools

SRI has been pursuing fundamental research in the area of combustion/fire modeling, with particular emphasis on developing numerical models for fire spread process. As part of this effort, a CFD code

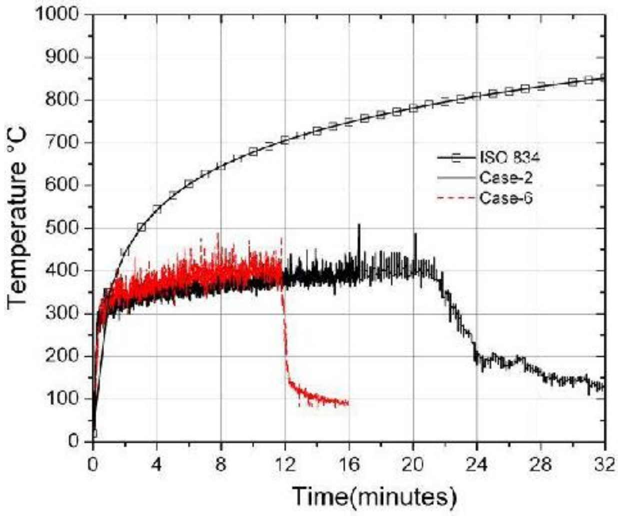


Fig. 6(b): Comparison with standard fire curve

Fig. 6(c): CFTF and control room

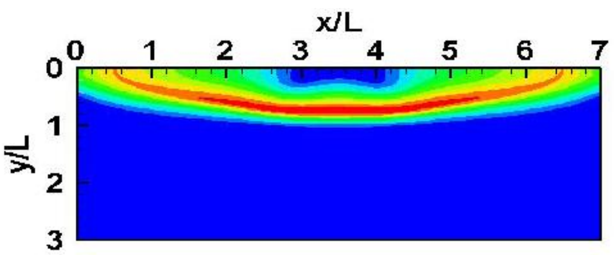


Fig. 6(d): θ = +90° (Roof flame; fuel surface (x/L=3 to x/L=4)

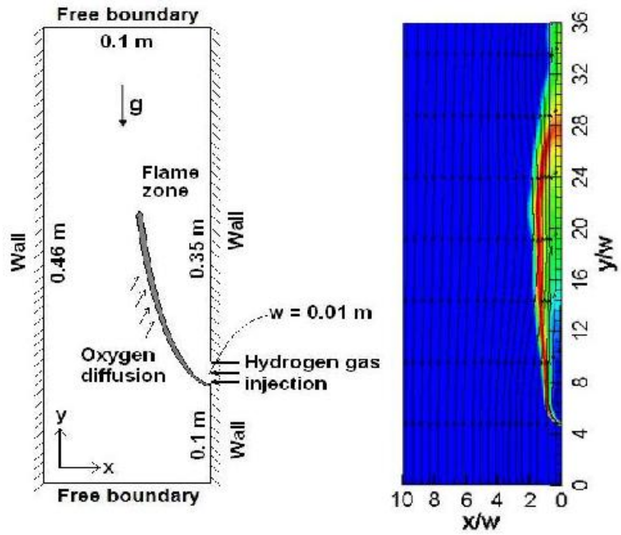


Fig.6(e): Computational domain, hydrogen flame shape and streamlines for u∞=2.0 m/s.

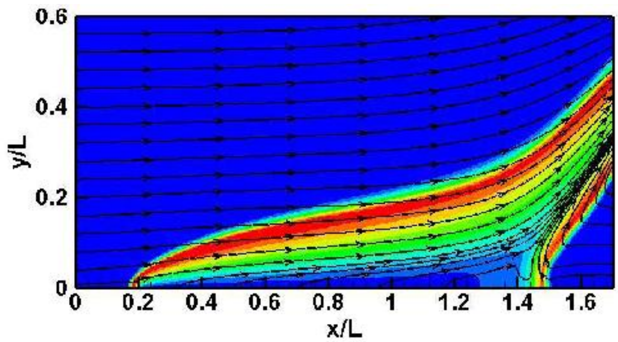


Fig. 6(f): Flame for u∞=0.75 m/s; fuel pool x/L=0.2 to 1.2

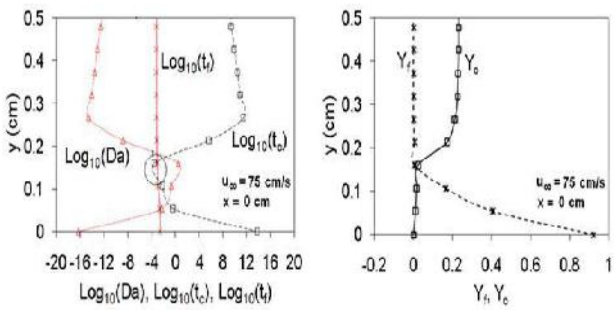


Fig. 6(g): Variation of local Da (left); fuel and oxygen mass fraction(right), normal to fuel surface at x/L=0.2 (leading edge)

based on Finite Volume method with a range of capabilities has been developed in two stages. The first stage development focused on steady diffusion flames under forced as well as natural convective conditions. This model was thoroughly validated against experimental results available in literature. It was then applied to simulate co-flow, cross and opposed flow flames above liquid fuel surfaces. Later, it was also used to simulate hydrogen plume combustion. Fig. 6(d) shows the predicted flame shape (indicated by temperature contours) for fuel burning on the ceiling under natural convective conditions. Fig. 6(e) shows the burning of a hydrogen plume predicted by the numerical model.

This CFD code was used to investigate and subsequently develop a method to characterize the flame stability and extinction. The presence or absence of flame at a particular location was quantified in terms of 'local' Damköhler number (Da), as opposed to the traditional method of using 'global' Damköhler number. It was shown that irrespective of the flow configuration, a flame is sustained only at a location where the local Damköhler number is equal

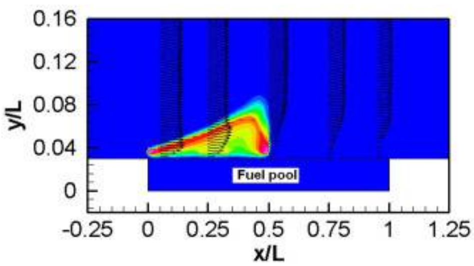


Fig. 6(h): Flame location at t=0.15s for u∞=1.3 m/s

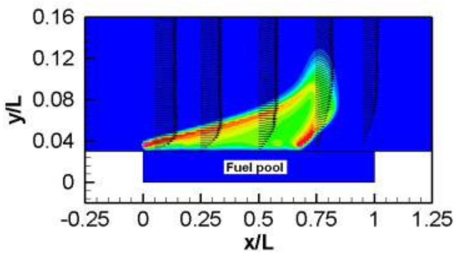


Fig. 6(i): Flame location at t=0.25s for u∞=1.3 m/s

to or slightly greater than unity. To demonstrate the applicability of this concept, numerical studies are performed for various flame configurations like co-flow, cross and opposed flow flames and the idea was applied to these flames. Fig. 6(f) shows a boundary layer type flame over ethanol pool surface. Fig. 6(g) provides the analysis at the fuel leading edge; the flame is sustained where Da≈1 (or $\log_{10}$Da≈0) and where fuel is completely consumed.

These initial works served as a convenient starting point for development of a transient CFD code for predicting flame spread over liquid surfaces. The physics involved in transient flame spread was captured by developing a multi-phase CFD model wherein the gas and liquid phases were coupled using appropriate interfacial boundary conditions. Its capability was demonstrated by means of a case study involving concurrent flame spread on methanol pool 100 cm long, 3 cm deep and of infinite width. A computational domain in the gas and liquid phases were taken as 150 cm × 150 cm and 100 cm × 3 cm respectively. The free stream air velocity was varied from 0.5 m/s to 5.2 m/s and the flow is from left to right. The fuel is ignited at the leading edge. The results were validated against experimental data available in literature. Figs.6 (h) and (i) show the flame propagation by means of temperature contours and velocity vector plots at two instants i.e., t=0.15 s and 0.25 s. The average flame speed is found to be approximately 1.68 m/s, which is in good agreement with the experimental value of 1.76 m/s obtained for similar conditions.

B.3) Hot Cell Fire Hazard Assessment

It is proposed to add a sodium cleaning system inside a hot cell at RML, IGCAR for removal of sodium from

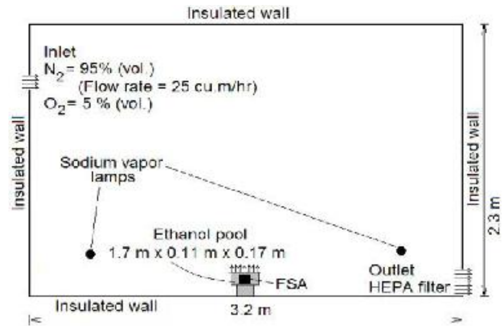


Fig. 6(j): Hot cell with FSA (side view)

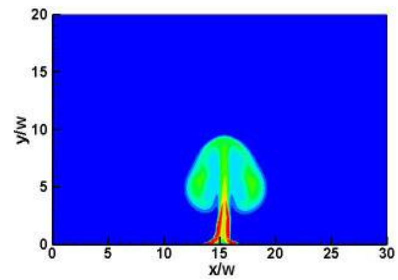


Fig. 6(k): Fire development [21% O2 in inlet flow and in cell atmosphere]; Pool width (w) = 11cm; corresponding non-dimensional distances are indicated in x and y axis

a Failed Fuel Subassembly (FSA). The FSA with fuel pin bundle is immersed in 15 L of 99.9% pure ethyl alcohol held in a container. The decay heat of FSA is around 160 W, which is likely to generate ethyl alcohol vapours, with potential to form a flammable mixture within the cell. About 100 g of sodium is expected to be present in the FSA, and therefore, the possibility of generation of hydrogen due to reaction of ethyl alcohol and sodium also exists. Fig. 6(j) shows a schematic of the cross sectional view of hot cell with FSA. A systematic study of the fire hazard potential within the hot cell was undertaken using computational tools developed in-house at SRI. The investigations involved transient heat and mass transfer calculation to determine ethyl alcohol bulk temperature, vapor generation rate and assessing the possibility of its ignition for various hot cell operating conditions. Parametric studies were conducted with respect to FSA decay heat, oxygen volume fraction in the incoming flow, strength & location of ignition sources etc. Fig. 6(k) shows the fire development for a case, wherein the oxygen level corresponds to ambient conditions. The difficulty in igniting and sustaining a flame at low oxygen levels (< 9.9 % v/v) was demonstrated by the numerical studies.

Based on the above studies, several recommendations and suggestions were provided for safe operating conditions to preclude the occurrence of fire within the hot cell. Some of them are as follows: (a) The main parameter to be controlled is the oxygen concentration, which needs to be kept well below the

MOC level (9.9% vol.) at all times during operation (b) It appears highly unlikely that the sodium vapor lamps (1000 W) would be able to act as an ignition source to cause a fire, (c) Auto ignition temperature of ethyl alcohol vapor is around 365°C. In the absence of any major heat source to cause the temperature rise to this level, this mode of ignition is not possible, (d) Even if the entire amount of sodium present in FSA reacts with ethyl alcohol, it may not produce sufficient amount of hydrogen to form a flammable mixture and cause hydrogen deflagration, (e) The flammability diagram of ethyl alcohol and hydrogen can be employed for developing a strategy to avoid formation of flammable mixture while diluting fuel rich mixtures. A detailed report was submitted to RML, IGCAR for further action.

C) HYDROGEN safety Studies

C.1) Development of in-house CFD Code

A CFD code called Hydrogen Distribution Simulator (HDS) has been developed at SRI to study hydrogen distribution and mitigation processes. This code solves the governing equations for flow, heat and mass transfer and turbulence in Cartesian geometries. Buoyancy enhanced k-ε model for turbulence has been incorporated and the code has been validated against standard benchmark problems like lid driven cavity, laminar and turbulent natural convection, laminar and turbulent jets, double diffusion convection, buoyant plumes etc. Models for steam condensation and reaction kinetics for hydrogen removal by Passive Autocatalytic Recombiners (PAR) are available. A falling film steam condensation model has been implemented in the HDS code that takes into account the effect of turbulence, shear on film surface and presence of non-condensable gases in the vicinity of the film. It has been validated against the experimental results from CONAN facility. The rate of reaction on the catalyst coated surface is estimated based on a point model for reaction kinetics (PMRK) that contains a 12 step model for surface reactions. This model has also been validated against the experimental data from the REKO-3 facility.

C.2) Hydrogen Distribution Studies

Hydrogen distribution in single enclosure, for jet release of hydrogen / hydrogen-steam mixture under laminar and turbulent flow conditions with and without the presence of steam has been investigated using HDS. Fig. 7(a) shows the evolution of a jet and the resultant hydrogen distribution with and without steam condensation. The influence of steam condensation on the distribution is clearly

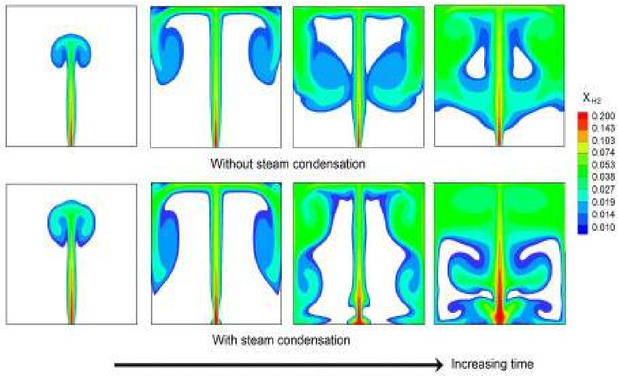


Fig. 7(a): Contour plots of hydrogen mole fraction without and with steam condensation at same instants of time.

evident. Using HDS, four numerical indices have also been developed to quantify the state of mixing and deflagration potential of hydrogen distribution within a containment. The first index, namely, average mole fraction, depicts the average quantity of hydrogen in the containment at any time. The second index, namely, the non-uniformity index, is used to quantify the non-uniformity or stratification in the distribution. The third and fourth indices, namely, deflagration volume fraction and deflagration pressure ratio, are used to depict the size of flammable cloud formed and the possible pressure rise in case of deflagration in the containment. These four indices can be used to study the effect of mixing and mitigation measures on the resultant distribution of hydrogen in air and steam environment.

C.3) Hydrogen Mitigation Studies

A combination of CFD simulations using HDS code and parametric studies using the stand alone PMRK code has been used to test the appropriateness of single and multi-step models for hydrogen recombination within the PAR. The geometry of the REKO-3 facility, the computational domain and the results obtained using various models are shown in Figs. 7(b) and (c). It was found that the existing multi-step models lead to high temperature at the leading edge of the catalyst plates. Hence, an improved model with temperature dependent sticking coefficients was developed in-house and validated.

The gas concentration near the plates, plate temperature and active catalyst sites are parameters and the reaction rate is the outcome in a reaction chemistry model. A comparison of temperature profile along the length of a centrally located catalyst plate obtained using various models is shown in Fig. 7(d). The PMRK has been used to develop the improved multi-step model as well as study the reaction rates at different input conditions. The dependence of rate of reaction on selected input parameters (using the

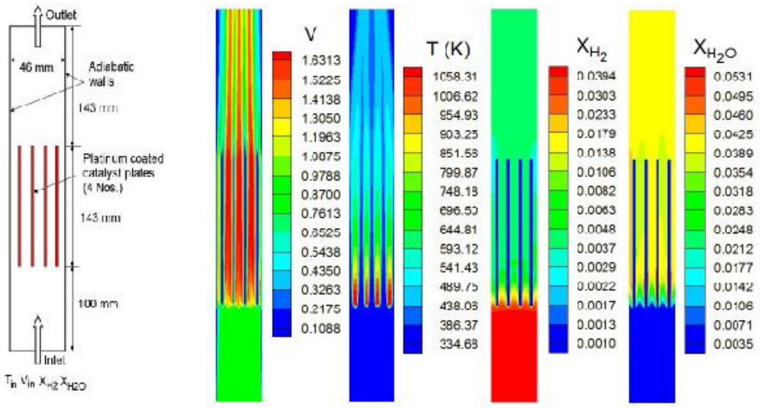


Fig.7 : Numerical study of hydrogen mitigation within the REKO-3 facility: (b) Computational domain; (c) Contour plots of velocity, temperature, and mole fractions of hydrogen and steam

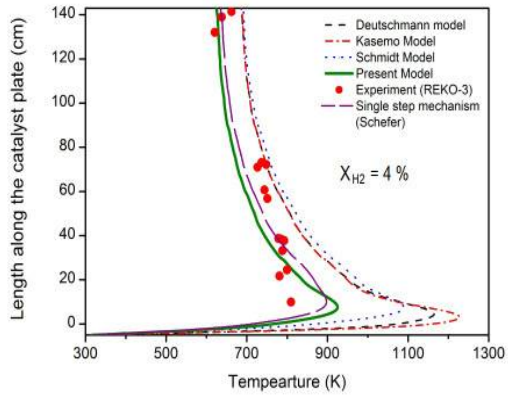


Fig. 7(d): Temperature profile along the length of a middle catalyst plate.

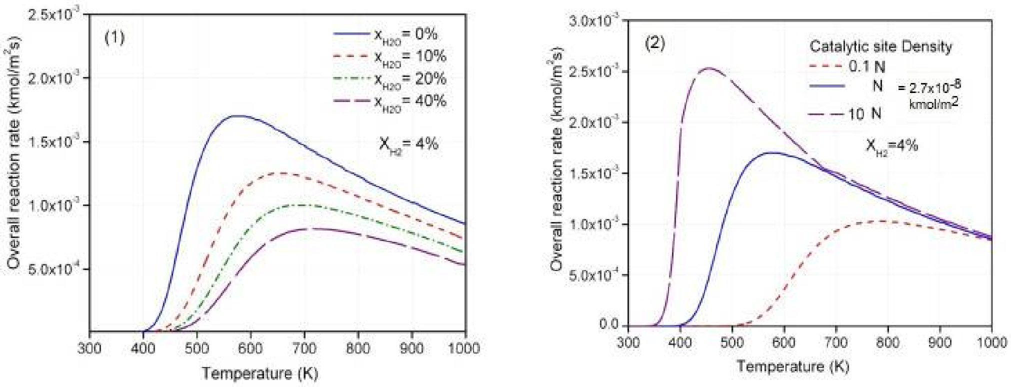


Fig. 7(e): Dependence of rate of hydrogen recombination on temperature: effect of change in (1) steam mole fraction; (2) catalytic site density.

improved multi-step) model was also investigated. It was found that the reactions start above a certain threshold temperature and peaks at certain optimal temperature. At higher temperatures, the reaction rates are lower. As hydrogen mole fraction increases, the peak in reaction rate shifts to higher temperature. Further, the number of active catalyst sites per unit area has a significant influence on the reaction rates [See Fig. 7 (e)]

D) THERMAL HYDRAULIC ANALYSIS AND SAFETY STUDIES

D.1) Core Thermal Hydraulics

D.1.1) Numerical Simulation of Thermal Striping Phenomena

Thermal striping, which occurs due to random temperature fluctuations induced by improper mixing of different temperature fluid streams, is a complex phenomenon. Since it is an important safety issue in FBRs, experimental and numerical studies were initiated in collaboration with FRTG, IGCAR. Several numerical simulations were carried out to validate the available numerical models with experimental data reported in literature. In one such simulation, a three-dimensional geometrical configuration was considered, wherein, two streams of water, one hot and one cold impinges on a steel plate kept inside a cylindrical chamber. The major parameters of interest include the mean and RMS temperatures and velocities, particularly near the plate surface. The computational domain was discretized using approximately 2 million meshes. Appropriate turbulence model and time steps were adopted to capture the thermal striping phenomena. Fig. 8(a) shows the plot of mean temperature at and near the plate surface. The sharp rise in the temperature fluctuations in the central region can be observed from the plot of RMS temperatures shown in Fig. 8(b). These results are found to agree well with those reported in literature and are successful in simulating the thermal striping phenomena. This analysis has been useful in arriving at a numerical scheme that will be capable of predicting the thermal striping phenomena in PFBR.

D.1.2) Safety Studies on SGDHR System of PFBR

The Prototype Fast Breeder Reactor (PFBR) has a

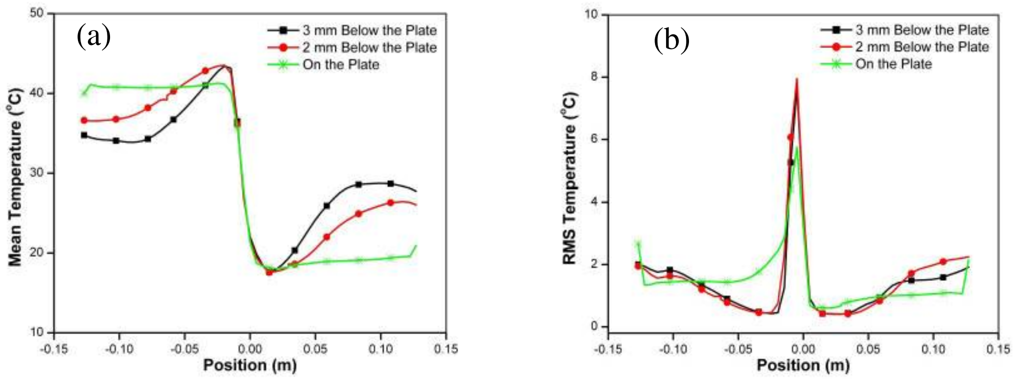


Fig. 8 : (a) Mean temperature near the plate, (b) RMS Temperature near the Plate

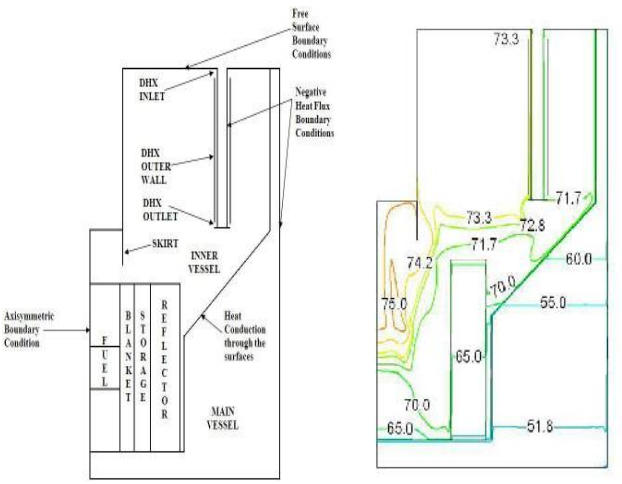


Fig. 9 : (a) Model Geometry, (b) Temperature (°C) contours for SST k-ω Model

Safety Grade Decay Heat Removal System (SGDHRS) to ensure the removal of decay heat after reactor shutdown under station blackout conditions. The SGDHRS works completely under natural circulation phenomena. Experiments have been conducted in SAMRAT facility of IGCAR using water as a simulant to establish the adequacy of the design of SGDHRS. Complementary numerical studies have also been performed with the overall objective of extend the analysis to PFBR.

The two-dimensional axi-symmetric model geometry with boundary conditions is shown in Fig. 9(a). The porous media approach for modeling the core subassemblies was found to be adequate for modeling the flow pattern and heat transfer within and outside the core. At the outset, the capability of different turbulence models (Standard k-ε and Shear Stress Transport k-ω models) in predicting the natural circulation phenomena was assessed. Various thermal hydraulic parameters within the hot and cold pool have been analyzed and compared for their qualitative behavior with the data reported in literature. Fig 9(b) shows the temperature contour plots for the SST k-ω model case. The computed results in SST k-ω model case are within 15% variation from the experimental results. The effectiveness and superiority of SST k-ω model in predicting the natural circulation phenomena as compared to the standard k-ε model was demonstrated. Subsequent parametric studies were useful in arriving at a suitable numerical scheme for simulating SGDHR functionality in PFBR.

D.2) Containment Thermal Hydraulics

D.2.1) Safety Review Analysis for KGS-1&2

As part of safety review work, detailed thermal hydraulic calculations were performed for KGS-1&2 containment system. In-house containment thermal hydraulics code 'THYCON' was used for this purpose. For a LBLOCA scenario involving 200% RIH break, with loss of ECCS, an independent verification showed that primary containment peak pressure and temperature values are 0.89 kg/cm^2 (g) and 119.76°C respectively. These are fairly close to the values of 0.85 kg/cm^2 (g) and 118.4°C reported in SAR provided by the utility. Fig. 10 (a) and (b) show the pressure and temperature transients for one case, where BOPs are available and the V1-V2 leakage area is taken as 1 ft^2. Such studies and comparisons were made for several RIH and ROH break cases and the influence of various parameters was clearly brought out.

D.2.2) Short term containment system response studies

A detailed investigation was carried out using 'THYCON' to quantify the influence of geometrical and thermal hydraulic parameters on the KGS-1&2 containment peak pressure. Factors such as initial blow down conditions, initial relative humidity and temperature within the containment building, heat transfer to structures, availability of suppression pool, downcomer submergence depth, V1/V2 volume ratio, total containment volume, vent flow area, downcomer suppression pool bypass etc. were considered. A base case was first considered, for which the peak pressure was obtained. Thereafter, the parameters listed above were suitably altered over appropriate range to assess their influence on the peak pressure. These were then compared with the base case value to quantify the effect. For example, it was found that higher initial RH/building temperature results in lower containment peak pressure. The effect is more pronounced for small break LOCA [Figs. 11(a)].The efficacy of suppression pool in limiting the peak pressure is higher for large break LOCA, when compared to small break LOCA. The drywell peak pressure increases with increase in downcomer submergence [Fig. 11(b)]. These investigations give useful insight into short term containment system response that can be utilized during regulatory review process.

D.2.3) Safety Review of TAPS-1&2 Containment Filtered Venting System (CFVS)

Venting of the containment through appropriate filtration systems is now being considered as a severe accident management strategy for maintaining the integrity of containment building and also as a means to reduce the radiological consequences to the public and environment. The option of filtered containment venting appears to have gained more importance in the post Fukushima accident backdrop.

The utility has proposed to install a Containment Filtered Venting System (CFVS) system for TAPS-1&2. The DBR is under review at SRI and preliminary comments and observations have been provided. For some of the safety issues, independent verification studies have been initiated. Among the many safety issues to be addressed prior to installing a CFVS, one pertains to the adequacy of the water inventory in the CFVS scrubber tank to absorb the energy content of steam and decaying radioactive isotopes that are likely to be discharged into it following a severe accident. For this, a simple mathematical model for heat transport was developed and the response of scrubber tank to heat load under single and two units SBO conditions in TAPS-1&2 was investigated. The technical details and information provided in design report on CFVS forms the starting point for the present investigation and some of these data are used as input for the present model. Fig. 12(a) and (b) shows the calculated cumulative radioactive decay heat and the ensuing water temperature rise in the scrubber tank, respectively. Further analytical

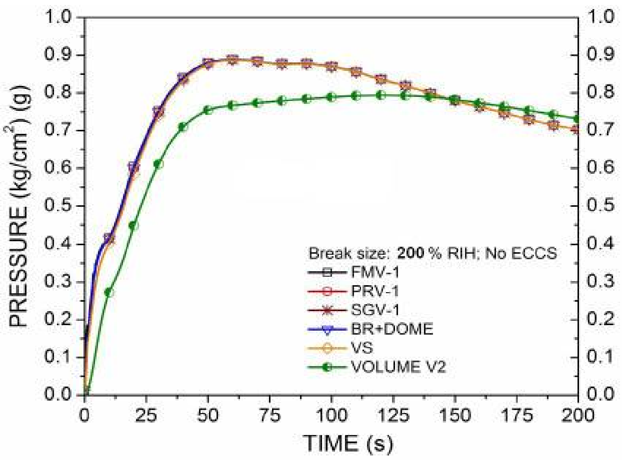


Fig. 10(a): Pressure transients for 200% RIH break

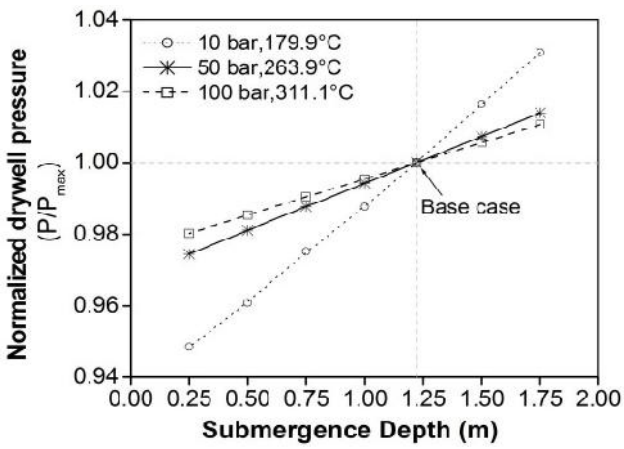


Fig.11(a): Effect of submergence depth

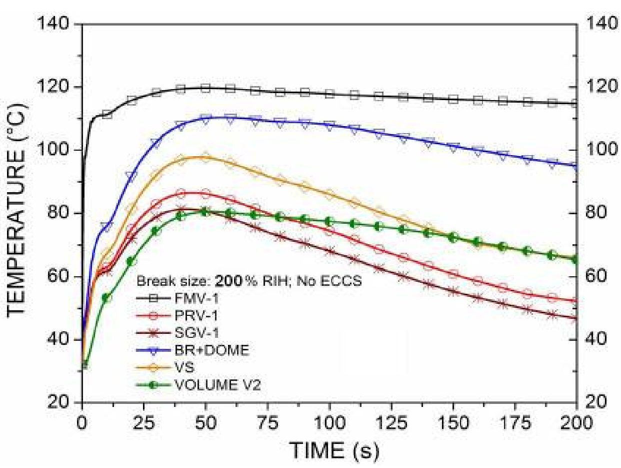


Fig. 10(b): Temperature transients for 200% RIH break

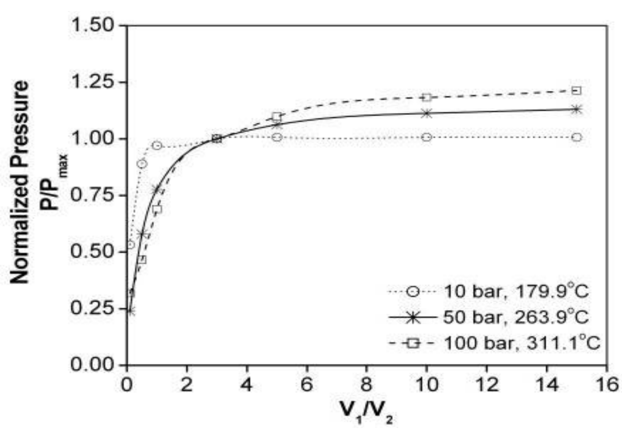


Fig.11(b): Influence of V_1/V_2 Ratio on Containment Peak Pressure work is underway to conduct a thorough review of all aspects of this important safety system.

D.2.4) Capability enhancement of THYCON code

Further developmental work was undertaken to enhance the capability of 'THYCON by incorporating modules for building coolers. Some of the earlier studies were repeated to obtain the long term containment pressure and temperature transients. Fig. 13 (a) and (b) show the effect of cooler operation on the long term transients.

E) RADIONUCLIDE MIGRATION STUDIES

E.1) Modelling of Ca leaching from Concrete Engineered Barrier in a NSDF

In any Near Surface Disposal Facility (NSDF) for radioactive wastes, wastes are immobilized in matrices such as cement, and are disposed in various engineering modules depending upon its activity level in the waste form. The engineering module isolates the radioactive waste from the surrounding geo-environment. As long as the concrete engineered barrier is intact, the release of the radioactive waste from the disposal facility does not occur. However, every concrete structure has a finite life time which degrades with time, depending on the type of concrete and external / internal environmental factors. Owing to the safety implications of such a breach in the concrete barrier and the possible consequences, this study was undertaken. The idea was to investigate the effect of properties of the concrete barrier (porosity and calcium content) on the radio nuclide migration within the barrier.

A numerical model was developed to investigate the degradation of a wall of an engineered barrier in a typical NSDF. The concrete wall thickness is taken as 25 cm with initial porosity of 0.14. The conceptual model assumes rain water seepage into the engineered barrier through the top cover and dissolving of radio nuclides from the waste matrix and subsequent degradation of concrete wall by the infiltrated water. The spatial and temporal variation of concrete porosity and calcium has been brought out in this study. Fig.14 (a) shows the variation in concrete porosity along concrete thickness at various times. Fig.14 (b) shows the temporal variation of

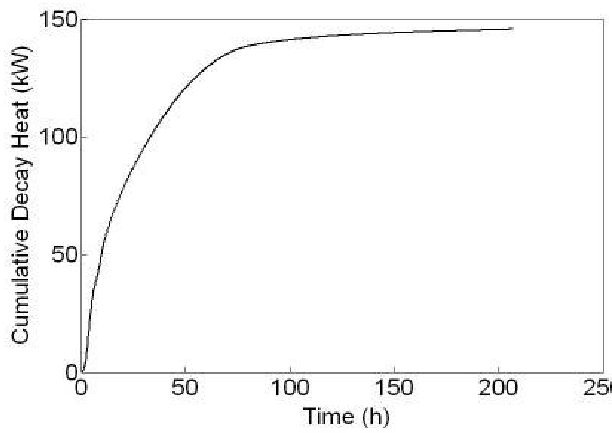


Fig.12(a): Cumulative radioactive decay heat in ST vs. time for two unit SBO

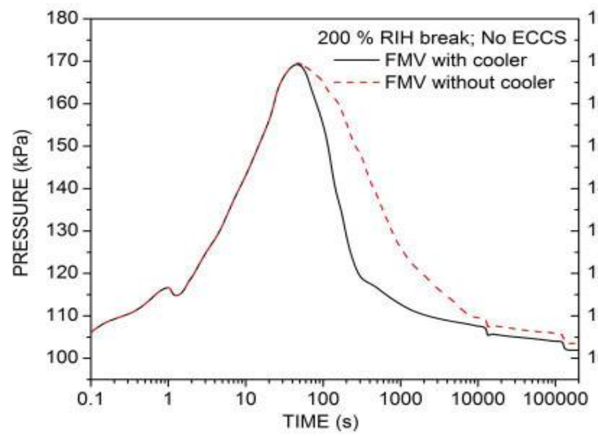


Fig.13(a): FMV pressure transients with and without operation of building coolers

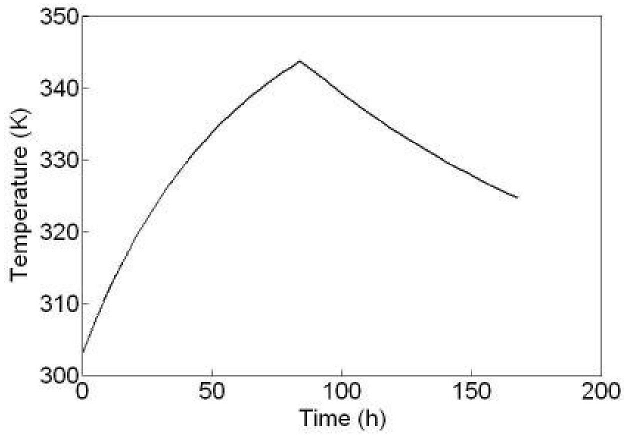


Fig.12(b): ST water temperature transient for two units SBO

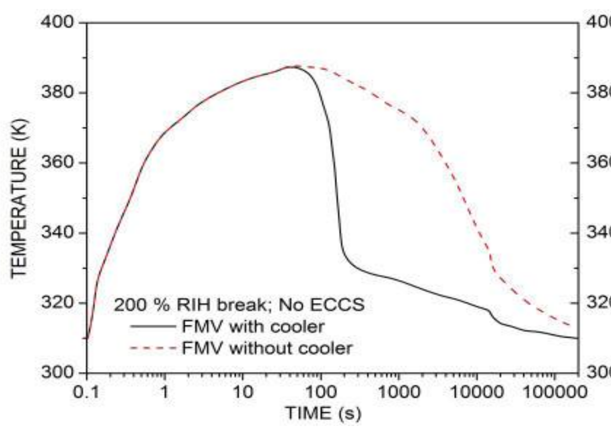


Fig.13(b): FMV temperature transients with and without operation of building coolers calcium concentration and porosity at a particular depth.

E.2) Radionuclide migration studies using PORFLOW software

Studies in the area of radionuclide transport in geo-environment have assumed much importance following the recent Fukushima accidents. The CFD code PORFLOW is now being used for conducting studies in this area. A BRNS Round-Robin exercise problem was first solved to benchmark the results from this code with those from other models. Fairly good agreement could be obtained. Subsequently, studies on transport of Sr-90 in an unconfined aquifer under specified conditions were taken up. It is well known that during the migration, daughter products (Y-90 and Zr-90) will be generated due to radioactive decay. The studies are being conducted assuming an unconfined aquifer of dimensions 2000m × 500m. A constant source for Sr-90 was assumed at the top of the aquifer, at an appropriate distance from the lateral sides. A constant infiltration of 0.001m/day and a ground water velocity of 0.02m/day have been given as initial conditions. In the process of radionuclides migration, ground water movement also plays important role and therefore, a bore well was also modeled to investigate its effect on the migration process. The effect of parameters such as radionuclide type, decay products migration, infiltration rate and the bore wells location on the migration of Sr-90 and its daughter products were quantified. Typical results are presented in Fig.15 (a) through (c).

F) EXTERNAL HAZARD STUDIES

F.1) Investigations on submarine volcanic activity on coastal NPPs

As part of capability development for hazard assessment of NPPs located along coastal sites, due to submarine volcanic activity, a detailed mathematical model was developed. The model consists of differential equations of motion of ballistic projectile generated by volcanic eruption from within the sea bed. A computer program was developed to solve these equations. Parametric studies were then conducted with respect to the magnitude of the ejecta pressure, caprock fragment material properties, its size, duration of volcanic impulsive force etc. Very conservative input data were assumed for these parameters, with the

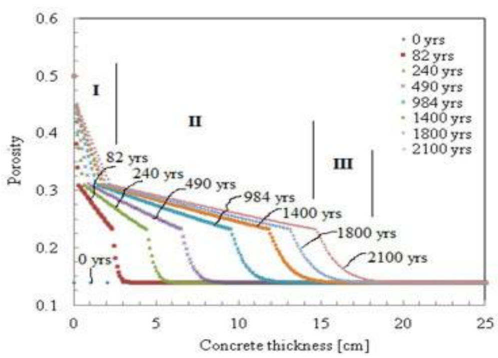


Fig.14(a): Variation of porosity across concrete thickness with time

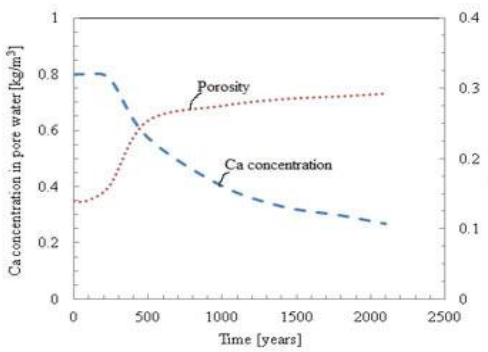


Fig.14(b): Variation of calcium concentration and porosity at a depth of 5 cm from the inner side of wall

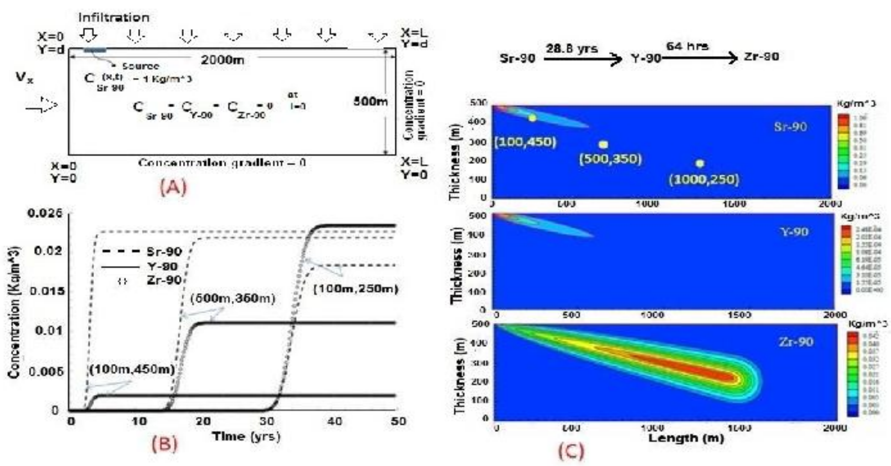


Fig.15(a): Conceptual aquifer domain, (b) Concentration profile of parent (Sr-90) & daughter products (Y-90 & Zr-90) and (c) Contour plots of radionuclides concentration at the end of 50 years objective to calculate the range of Volcanic Ballistic Projectile (VBP) under various conditions. It was shown that under realistic conditions, the hazard from the postulated submarine eruption would be negligible. The findings of this study are brought out in the form of a detailed report. This model can be utilized in future for undertaking plant specific case studies.

G) COLLABORATIVE PROJECTS

G.1) Hydrogen Mixing Studies (AIHMS)

AERB-IIT Madras Hydrogen Mixing Studies (AIHMS) facility has been erected within the IIT Madras premises to carry out studies related to hydrogen distribution in steam condensing environment. A schematic diagram of the facility is shown in Fig. 16(a). The main test vessel is a 2 m^3 multi-compartment enclosure [see Fig. 16(b)] with temperature controlled walls. Helium and steam mixture will be injected at the center at different elevations and the distribution in presence of steam condensation will be studied. This experimental study is also backed by elaborate numerical program at both SRI and IIT Madras.

G.2) Water and Steam Interaction Facility (WASIF)

As a part of comprehensive research and development program in the area of reactor safety, an experimental facility is planned in collaboration with BARC to identify and investigate key parameters that influence DCC in various modes, and to quantify their impact on associated reactor components and systems. These studies are expected to provide a better understanding of the several related, complex physical phenomena involved in DCC, and would help in undertaking the development of new theoretical models. The focus of this experimental program is to comprehensively investigate safety issues relevant to DCC that have a direct bearing on the design and safety of reactor systems and components.

H) EXPERIMENTAL PROGRAMMES

H.1) Fire mitigation studies within CFTF

With minor modifications and by using additional instrumentation within the Compartmental Fire Test Facility (CFTF), fire mitigation studies will be conducted for oil/solvent and cable fires. The activity will include refurbishing of the existing facility with suitable fire barrier material and fabrication, installation and commissioning of additional components. The focus of these studies will be on investigating and quantifying the effectiveness of fire mitigation using agents such as water mist and inert gases.

H.2) Hydrogen Mitigation Experiments

Studies on hydrogen recombination using catalytic oxidation and controlled combustion are planned to be conducted during the XII plan (2012-2017). An experimental facility called the Hydrogen Mitigation facility (HYMIF) is planned to be fabricated, erected and commissioned within the SRI Engineering hall. This facility will consist of a bench setup and an enclosure setup. In the bench setup, experiments will be conducted in forced flow conditions and the reaction parameters will be measured under controlled conditions. The recombiner devices will be erected within a large enclosure and then the hydrogen recombination will be studied in passive mode. The detailed project report, test matrix, design basis and detailed designs have been completed. The project will provide valuable insights into hydrogen mitigation and help in regulatory decisions.

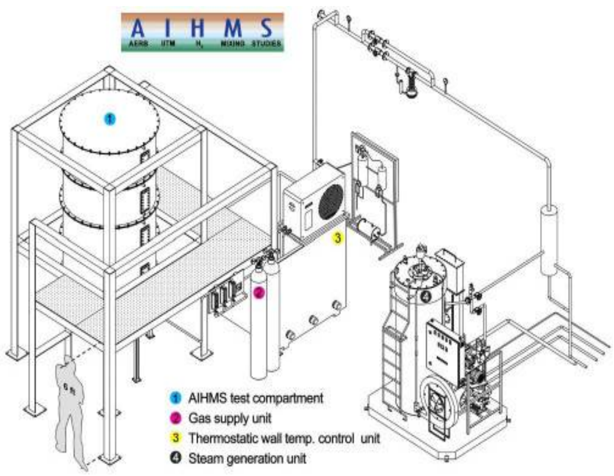


Fig. 16 : The AIHMS facility: (a) Schematic of the entire setup; (b) As built test chamber

3.4 ENVIRONMENT SAFETY & FUEL CHEMISTRY STUDIES

Introduction

R&D activities in the areas of fuel chemistry, waste management and environmental chemistry such as radionuclide transport studies in ground and air, nuclear fuel safety studies including metallic fuel safety studies, pyroprocessing related safety studies, waste management related safety studies, NORM waste related safety studies, Environmental geochemistry are carried out to support the regulatory activities pertaining to reactor and nuclear fuel cycle facilities.

Environmental Chemistry Laboratory (ECL) has been setup with sophisticated analytical instrumentation and has facilities to handle low level radioactive works such as fume hoods, once through ventilation system, low level waste collection tank, survey monitors to cater the needs of Type-I radiochemical laboratory of AERB. Analytical services are also rendered to other divisions of IGCAR and BARCF periodically.

A) THERMOKINETIC BEHAVIOR OF TRI-N-BUTYL PHOSPHATE UNDER ADIABATIC CONDITIONS

For the recovery of uranium and plutonium from the spent fuel, a solvent extraction process, Plutonium Uranium EXtraction (PUREX) process is adopted worldwide where Tri-butyl phosphate (TBP) is used as an extracting agent. Though TBP has high selectivity for Uranium and Plutonium, it has few disadvantages including its high viscosity, density and its degradation to dibutyl phosphate (DBP) and monobutyl phosphate (MBP) besides the possibility of formation of reactive red oil. Hence, it is important to investigate the thermokinetic behavior of TBP and its degradation products and understand the parameters that influence the formation of reactive red oil.

Through coordinated research programme with CISRA, CLRI, Chennai, the thermokinetic behavior was investigated using the world's benchmark calorimeter, the Accelerating Rate Calorimeter (ARC). TBP exhibits multiple exothermic behavior patterns with onset temperatures 250 and 375°C. The self heat rate studies carried out TBP when subjected to heat wait search cycle undergoes an exothermic cleavage leading to its breakdown as smaller products. The multiple exotherms recorded beyond 375°C may be attributed to the decomposition of primary degradation products. 30% TBP in n-dodecane show only single exotherm, that extends up to 450°C. The time temperature ARC profiles of 100% and 30% TBP are given in Fig.1. The Fourier Transform Infra Red (FTIR) characterization for the TBP sample collected after the ARC experiment indicates the reduction in the number of peaks and formation of new peaks confirming the degradation of TBP (Fig.2).

The time temperature ARC output for reactive thermal behaviour of 30 wt % of TBP in dodecane with varying concentrations of HNO_3 (Fig. 3) showed that with increase in acid concentration the onset temperature and the time for onset of exothermic activity preceded. The changes in the pattern of temperature rise indicated the degree of thermal reactivity. The self heat rate profiles for varying concentrations of acid showed a clear trend in onset and peak heat release rate with increase in acid concentrations. The straight line observed on plotting ln k* Vs 1/T confirmed the assumption that the thermal decomposition of TBP followed first order kinetics and the slope of the plots is equal to ΔE/R. The activation energy calculated was 32.691 kCal mol-1, 27.123 kCal mol-1, 23.02 The straight line observed on plotting ln k* Vs 1/T confirmed the assumption that the thermal decomposition of TBP followed first order kinetics and the slope of the plots is equal to ΔE/R. The activation energy calculated was 32.691 kCal mol^{-1}, 27.123 kCal mol^{-1}, 23.024 kCal mol^{-1}, 22.170 kCal mol^{-1} & 19.793 kCal mol^{-1} for 1:1 mixture of 30 TBP with 4N, 8N, 10N, 12N, 16N HNO_3 respectively. The oxidation of butanol formed from the hydrolysis of TBP to carboxylic acid in the presence of nitric acid is the self-heating reaction for red oil formation and energy release.

ARC experiments were carried out with irradiated TBP which confirmed the change of TBP structure due to irradiation and Gas Chromatography Mass Spectrometric (GC-MS) studies confirmed the presence of dibutyl phosphate. It is expected that the irradiation leads to the free radical mediated cleavage of TBP into daughter products which is also evident in the changes in the exotherm patterns and FT-IR patterns of irradiated TBP (Fig.4) compared to non-irradiated TBP. The above thermal hazard studies of TBP using accelerating Rate Calorimeter (ARC) is extremely reliable and carried out for the first time.

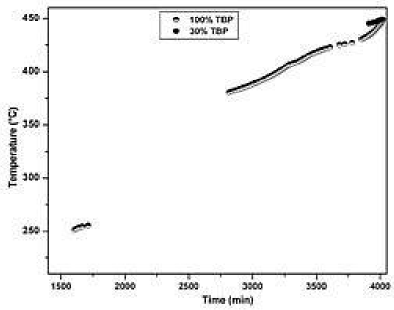


Fig. 1 : Comparative time-temperature profile of 100% TBP and 30% TBP

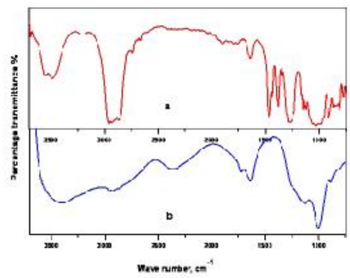


Fig. 2 : FTIR spectra of Tributyl Phosphate (a) before ARC experiment (b) after ARC experiment

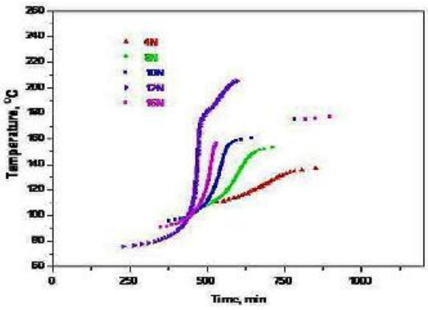


Fig. 3 : Temperature rise profiles for 30% TBP in varying nitric acid concentrations in 1:1 ratio

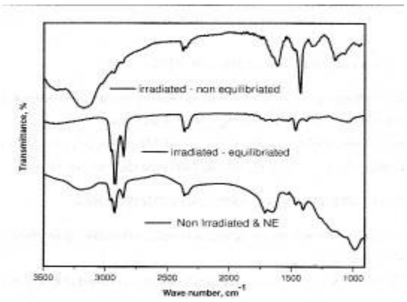


Fig. 4 : Comparison of FTIR spectra of irradiated equilibrated, non-irradiated non-equilibrated and irradiated non-equilibrated TBP-HNO_3 mixture

B) DEGRADATION OF ORGANIC POLLUTANTS IN AQUEOUS STREAMS USING ADVANCED OXIDATION NANOTECHNOLOGY

The environmental pollution due to rapid industrialization all around the world includes water pollution, air pollution and generation of hazardous organic wastes. The organic waste treatment pertaining to nuclear industries is complex owing to the presence of radioactivity. Recent research has highlighted the importance of mineralization of the organic pollutants in aqueous solutions involving Advanced Oxidation Processes (AOPs). Advanced Oxidation Processes (AOP's) are defined as near ambient temperature and pressure which are based on generation of ^{-}OH to initiate oxidative destruction of organics. ^{-}OH is generated either by the effect of photolysis on oxidants such as O_3, H_2O_2 or photocatalyst suspension in water or by non-photochemical methods such as ultrasonication or Fenton process depending on the type of AOP adopted. It is able to oxidize and mineralize almost every organic molecule, yielding CO_2 and water.

For the destruction of organic pollutants encountered in the nuclear industry, the possibility of adopting AOPs was studied. For the study, nanoparticles of titania and Gallia photocatalysts were synthesized in-house and were characterized for establishing their nanosize, catalytic activity and surface morphology. Nanocrystalline anatase titania photocatalyst was successfully synthesized using sol-gel route wherein titanium ethoxide was used as the precursor. The sol-gel method was coupled with conventional stirring and ultrasonication and two types of nano TiO_2 photocatalysts, TiO_2-US and TiO_2-S were synthesized. Similarly, nano sized polymorphs of gallium oxide, β-Ga_2O_3 and γ-Ga_2O_3 were synthesized using solution combustion route. Pure Gallium metal was used as the starting material and urea was used as the fuel for the combustion process. The as-combusted product was found to be the metastable β-Ga_2O_3 which on annealing at 600°C for nearly 200 hours produces γ-Ga_2O_3.

Characterizational techniques such as X-ray diffraction (XRD), particle size analysis (BET), Thermogravimentric analysis (TGA), Transmission Electron Microscopy (TEM), Scanning Electron Microscopy (SEM), Diffuse Reflectance Spectroscopy (DRS), Raman Spectroscopy and Fourier Transform Infrared Spectroscopy (FT-IR) were employed for establishing the parameters such as nanosize, surface morphology and catalytic activity (Fig.5). A cylindrical photoreactor (HEBER Scientific) was suitably fabricated and tailored to suit our photocatalytic experiments (Fig.6). The photoreactor had a quartz reaction vessel with a capacity of 1000 ± 200mL and has a lamp housing consisting of four low pressure mercury lamps with UV irradiation wavelength

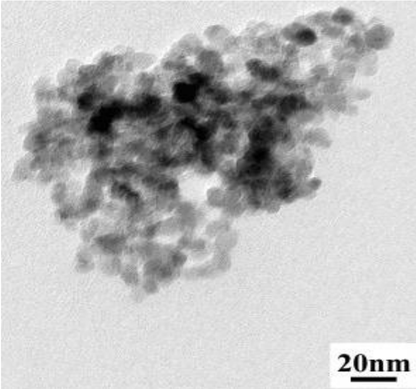


Fig. 5 : "TEM micrograph TiO_2-S photocatalyst"

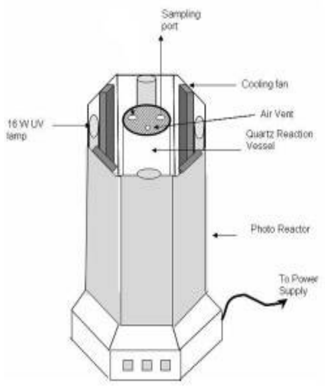


Fig. 6 : Cylindrical Photoreactor

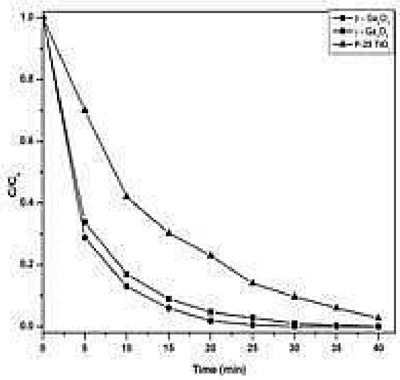


Fig. 7 : Comparision of degradation of TBP using β-Ga_2O_3, γ-Ga_2O_3 and P-25 TiO_2

of 254nm. The synthesized nanocatalysts were employed for the degradation of selected organic pollutants which were chosen by considering their interference in nuclear waste treatment methods and also their toxicity.

Complete degradation of 1000mg/l EDTA was demonstrated using synthesized nanoparticles of titania and gallia phototcatalysts and the liquid waste remained after photocatlytic experiment containing the degraded EDTA gave tenfold increase in the decontamination factors for the chemical precipitation step. Similarly, complete degradation of 1000mg/l hydrazine could be achieved in 210 minutes using the photocatalysts studied. Toxic hydrazine was converted into innocuous products such as nitrogen and water.

Using the nanosized catalysts, degradation of dissolved Tri-n-butyl phosphate (TBP) was successfully demonstrated and the degradation could be achieved in less than 40 minutes using the four nanosized photocatalysts employed in the study (Fig.7). The photocatalytic efficiency of the synthesized photocatalysts were compared with benchmark photocatalyst, P-25 TiO_2 (Degussa), Germany and the photocatalytic performance of the synthesized titania and Gallia were found to be far superior to P-25 for the degradation of all pollutants taken for the present study. The quantity of the nano catalysts employed was less (10-50 mg/l), reusable and the methodology adopted is environmentally benign.

C) HAZARD OPERABILITY STUDIES FOR NUCLEAR FUEL CYCLE FACILITIES

Hazard Operability (HAZOP) is a structured and systematic examination of a planned or existing process or operation in order to identify and evaluate hazards that may represent risks to personnel or equipment, or prevent efficient operation. The concerns with respect to fuel cycle facilities include criticality, radiation protection to workers, chemical hazards, fire and explosion hazards. The design approaches should ensure that the structures, systems and components important to safety have appropriate characteristics, specifications and material composition to perform the required safety functions. HAZOP of the head end cycle pertaining to the fuel reprocessing plant (FRP) of fast reactor fuel recycle facilities was taken up as a case study. The study can be extended to the other cycles subsequently leading to HAZOP for the entire facility. The study also involves the identification of process risks, safety measures already in place and specific recommendations for the improvement in process safety.

After a careful study of the Head end process the possible hazards are postulated. For example, in dissolver, reaction between HNO_3 and fuel material is highly exothermic that would lead to the build-up of off gas which needs to be controlled. Also, in Dissolver Off Gas System (DOGS), deposition of ruthenium in the ducts would lead to the build-up of radioactivity. The other possible hazards include leakage/spillage of hydrazine containing tanks that may lead to fire hazard. Some of the recommendations based on the HAZOP study were made. Ruthenium traps at appropriate location in the ducts of the dissolver-off gas system (DOGS) to monitor and avoid build-up of alpha activity due to RuO_2 deposition. To maintain proper ventilation during laser based dismantling and fuel pin chopping to avoid any fire and radioactivity getting airborne. Also, control of temperature during the acid killing operation to avoid the runaway reaction leading to the build up of temperature and pressure.

The study has provided an overview of the possible deviations in process parameters during operation and their consequences. It is necessary that this study is extended to quantify the probability of events and estimate the risk. Although a standard practice exists for conducting HAZOP in industries, it is the first time that HAZOP is applied to a nuclear fuel reprocessing plant.

D) INFLUENCE OF SALINE AQUIFERS ON HYDROGEOCHEMICAL CHARACTERISTICS AT COASTAL SITE

Geochemical parameters pertaining to the nuclear plant sites attach significance owing to the different types of applications at the site such as construction of underground waste disposal facilities, buildings, tall stack etc .The site specific information of a nuclear site includes the influence of groundwater geochemistry such as interaction of radionuclides with soil or rock. A study has been taken up to evaluate the parameters that influence the geochemistry of a coastal aquifer where the site is covered by saline water bodies. The data obtained from the chemical analyses were used for graphical plots and geochemical calculations. Aqueous speciation model, PHREEQC was employed to find out the Saturation Indices (SI) of possible minerals in the study area.

For the study, water samples were collected from fourteen bore wells for four seasons covering one hydrological cycle. Chemical characterization was carried out by using wet chemical analyses as well instrumental chemical analyses including Dual Channel Ion Chromatography system (IC850, METROHM, Switzerland) and double beam Atomic Absorption Spectrometer (Sensaa, GBC, Australia).

The study of Ca/Mg ratio of groundwater supports the dissolution of calcite and dolomite present. If the ratio of Ca/Mg=1, dissolution of dolomite should occur whereas, higher ratio is indicative of greater calcite contribution. For the study area, most of the points on the Ca/Mg plot lie closer to the line (Ca/Mg=1) indicating the dissolution of dolomite. Borewells away from the saline water bodies however show higher ratio of Ca/Mg (>2) especially in the post monsoon due to the dissolution of silicate minerals. Borewells closer to the saline water bodies show Ca/Mg ratio less than 1 irrespective of the seasons indicating the conversion of groundwater to saline (Fig.8). The processes of ion exchange and reverse ion exchange taking place in the study area can be understood by two chloro alkaline indices (CAI-I and CAI-II). Scholler indices for Kalpakkam site show that the locations away from the saline water bodies have fresh water wherein ion exchange takes place and the indices are negative. On the other hand, borewells closer to the saline water bodies which already have rich deposits of Mg and Ca in the aquifer gets exchanged with Na in the groundwater. Such borewell locations show positive values indicative of reverse ion exchange (Fig 9a&9b).

In the present study, to determine the chemical equilibrium between minerals and water, saturation indices (SI) of all possible minerals were evaluated for four seasons. For Kalpakkam site, minerals such as Anhydrite, Gypsum, halite were undersaturated irrespective of the seasons. Minerals such as dolomite and calcite get precipitated especially in borewells closer to saline water bodies irrespective of the seasons. The study demonstrated that Kalpakkam study area has been dominated by processes including weathering, ion exchange and dissolution and precipitation of minerals. The findings also reveal the possibility of ion exchange and reverse ion exchange of groundwater that has important implications on the radionuclide migration through geo-environment.

E) PREDICTION OF GROUNDWATER TABLE FLUCTUATIONS DUE TO PRECIPITATION

Water table fluctuations are influenced by precipitation during monsoon period for coastal aquifers. For the accurate prediction of groundwater head for such coastal aquifer, analytical model developed by Park and Parker was modified to suit the aquifer conditions based on precipitation. Estimation of parameters such as rate co-efficient (k) per day was carried out which was subsequently employed for the estimation of another variable, recharge precipitation ratio (α). The ‘k’ and ‘α’ derived from the five year data period (Jan-2005 to Feb 2010) was applied to predict the groundwater table for the next two year period (March 2010-December 2011). In all the three test cases studied, systematic time lag of

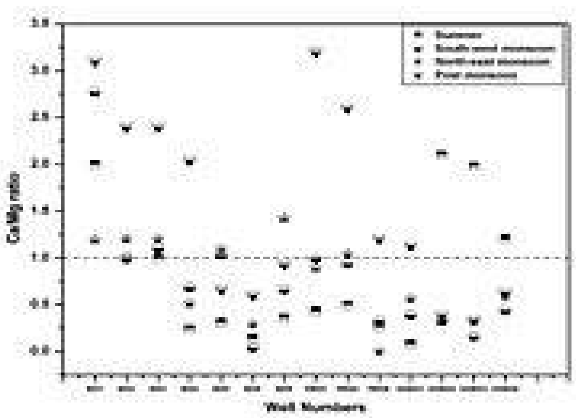


Fig. 8 : Plot of Ca/Mg molar ratio for groundwater samples

the estimated head with respect to observed head was observed which was more prominent during the monsoon period. The recharge uncertainty was found to be responsible for the same. Factors such as number of rainfall events, thickness of the unsaturated zone, soil texture, the type and size of the vegetation and the geology of the aquifer were found to be responsible for the time lag. As the time lag was systematic, a specific correction factor for precipitation term (P) in the Park and Parker equation was introduced to improve the prediction capabilities of semi-analytical equation ($P^* = P + xP$ where P = preceding 90 day cumulative precipitations, x = Correlation factor 0.5). By introducing the correction factor, a perceptible reduction in the time lag was observed which in turn has improved the correlation coefficient for the prediction of groundwater head (Fig.10). The site specific semi-analytical equation developed for coastal aquifer would help a great deal in groundwater prospecting and management and site selection of important installations besides prediction of the performance of a disposal facility over a period of time.

F) NUMERICAL MODELING OF CONTAMINANT TRANSPORT PHENOMENA

Under the activity, a Round Robin exercise on ground water flow and contaminant transport modeling from a typical uranium tailings pond was carried out by using the flow and contaminant transport modeling software Visual MODFLOW. The tailings contain naturally occurring long-lived radionuclides such as ^{238}U, ^{234}U, ^{230}Th and ^{226}Ra. Due to infiltration of water, there is a possibility of leaching of these radionuclides from the tailings pond to the unconfined aquifer. Upon reaching onto the unconfined aquifer, the contaminants will be subjected to advection, hydrodynamic dispersion, retardation and radioactive decay in the subsurface environment. Further, ingrowths of progenies will occur in the tailings pond as well as in the aquifer during transit. For assessing the radiological safety of the tailings pond, it is essential to evaluate the concentrations of ^{238}U, ^{234}U, ^{230}Th, ^{226}Ra and its progenies in the aquifer on a spatial and temporal scale.

Numerical simulation for a period of 104 years revealed that ground water flow reaches a steady state condition well within 5.5 x 10^3 years. ^{238}U and ^{234}U plumes were found to be migrating to a distance of approximately 950 m from the edge of the tailings pond, while ^{230}Th and ^{226}Ra plumes were migrating to only 170 m and 250 m respectively, due to their higher K_d values.

G) ATMOSPHERIC DISPERSION STUDIES

Under the study, a Gaussian plume model for dispersion of pollutants for short range releases (< 10-20 km) was developed. The model can be employed for ground as well as elevated releases such as from a stack. The model gives the steady state concentrations of pollutants in three dimensional spatial co-ordinates. Time integrated Gaussian plume equation for continuous releases of pollutants from a point source was used in the model. Here the concentrations of pollutants downwind are characterized by Gaussian plume parameters σ_y and σ_z. Gaussian plume parameters σ_y and σ_z for various atmospheric stability classes were calculated by using Pasquill-Gifford dispersion parameters for ground releases and by using Briggs dispersion parameters for elevated releases. Various sub-models to compute buoyant plume rise, momentum plume rise and source depletion model for dry and wet depositions were added. A decay correction term was also added to extent the use of the model to radioactive releases.

Dry as well as wet depositions are two processes which depletes the contaminant plume. Most widely used model for dry deposition from a Gaussian plume is the source depletion model originally proposed by Chamberlain for elevated sources. The dry deposition is caused by a combination of mechanical processes

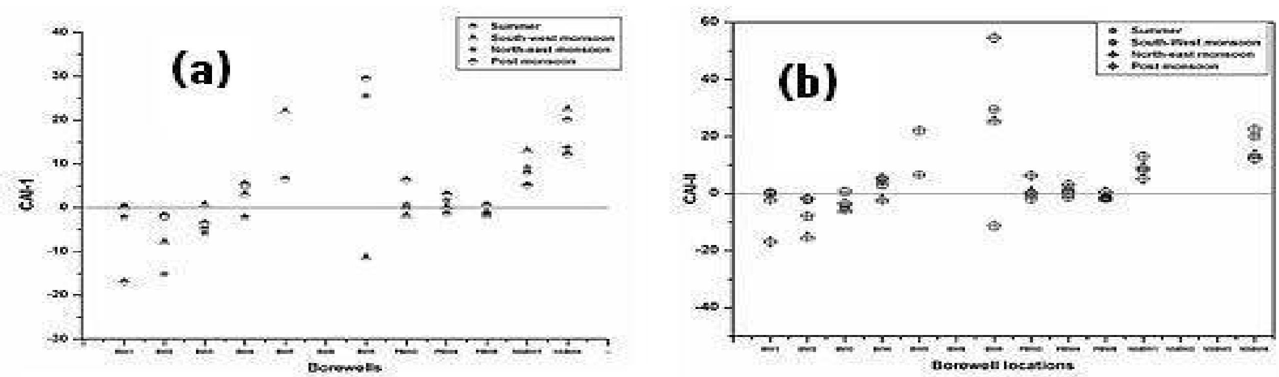


Fig. 9 : Chloro-alkaline Indices CAI-1(a) and CAI-II (b) for Kalpakkam site

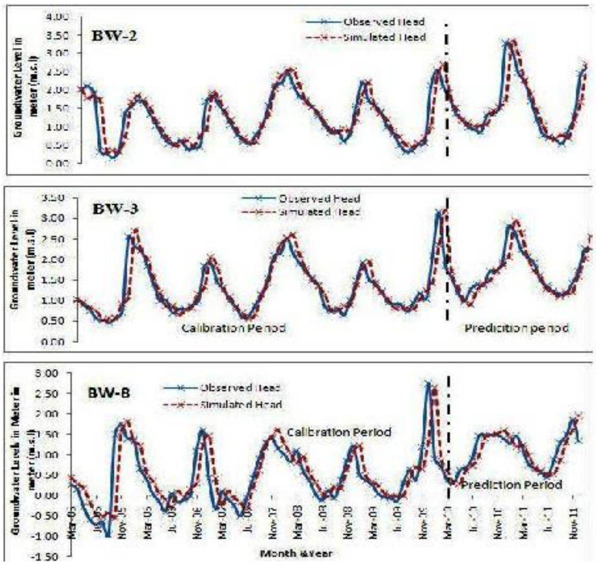


Fig. 10 : Comparison of observed and estimated groundwater head for borewells for calibration as well as prediction period

such as gravitational setting, turbulent and molecular diffusion and inertial impaction. In the absence of detailed microphysical information, dry deposition is assumed to be directly proportional to the air concentration at the immediate ground level. Source depletion fraction, (Q'_x/Q) for dry deposition was estimated as a function of distance from the source for a pollutant with deposition velocity 10^{-2} cm/s, for stability class F. The wind velocity is assumed to be 1 m/s. (Q'_x/Q) estimated were compared against the corresponding literature data for different release heights 10 m, 20 m, 30 m, 50 m and 100 m and are in good agreement (Fig. 11).

Wet deposition is the removal of radionuclides from an effluent plume by rain or snow. In wet deposition, the deposition rate dependent on the total amount of radioactivity contained in the plume and also the rain or snow fall rate. Under precipitation or snow fall, the depleted source strength becomes a function of both dry and wet deposition. By assuming a dry deposition velocity of 10^{-2} m/s, wind speed of 1 m/s and a precipitation rate of 1 mm/hr, source depletion fraction, (Q'_x/Q) under wet condition was determined as a function of distance from the source for stability class F, for source release heights 10 m, 20 m, 30 m, 50 m and 100 m (Fig. 12).

Gaussian plume model developed was benchmarked against health physics code Hot Spot developed by National Atmospheric Release Advisory Centre (NARAC), USA. For Benchmarking a sample problem of ^{131}I release through a stack from a nuclear facility over a plain terrain was formulated. Downwind ground level ^{131}I concentration estimated over a rural population zone by using the developed

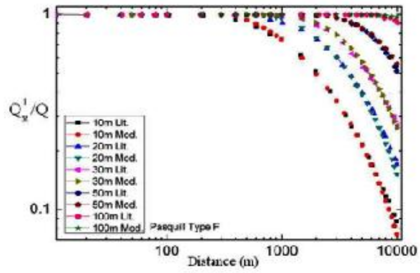


Fig. 11 : Source depletion fraction for dry deposition, as a function of distance from source for Pasquill type F

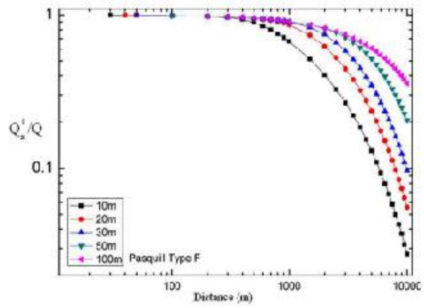
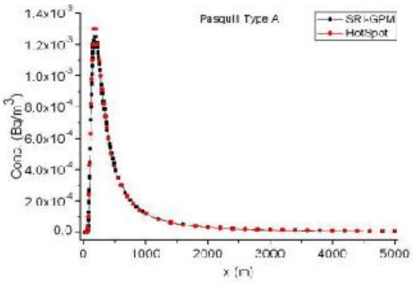


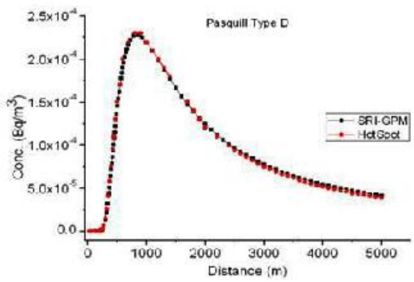
Fig. 12 : Source depletion fraction under wet condition, as a function of distance from source for Pasquill type F

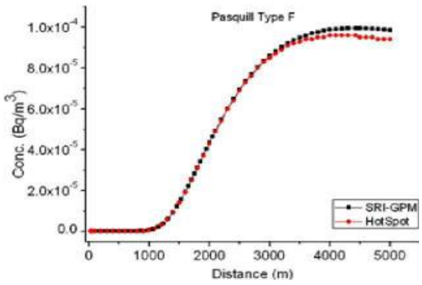
Gaussian plume was compared against the plume concentration estimated by using the health physics code Hot Spot, for all the six categories of stability classes and are in good agreement. Comparison of downwind ground level ^{131}I concentration for stability classes A (Unstable), D (Neutral) and F (Stable) are shown in Fig. 13, 14 and 15.

H) DIFFUSIVE LEACHING OF RADIONUCLIDES FROM IMMOBILIZED WASTE MATRICES

Low and intermediate level wastes generated from various nuclear installations are disposed off in Near Surface Disposal Facilities (NSDFs). The radionuclides are immobilized in solid matrices before placing it in metal containers, which in turn are buried in various disposal modules, depending upon its activity levels. The primary objective of a radioactive waste disposal facility is to isolate radioactive waste from the general population and the environment until the radionuclides in the waste have decayed to acceptable levels. Isolation of the radioactive waste is achieved by providing multiple barriers between the waste and the biosphere. Disposal module, metal container, backfill materials, disposal cover and the waste form itself act as containment and or retardation barrier against radionuclide transport. However, under adverse environmental circumstances these barriers can fail, allowing the infiltrating rain water to come in contact with the waste form. Once water comes in contact with the







Downwind ground level 131I concentration

Fig.13: Stability class A | Fig.14: Stability class D | Fig.15: Stability class F

waste form, radionuclides can be leached onto the surrounding media by various processes. Leach rates of various radionuclides constitute the source term to hydrogeological transport models.

A Numerical model has been developed to study the leach rates of radionuclides from a single solid waste matrix. The governing equation was formulated in the form of a two-dimensional diffusion equation in cylindrical co-ordinate system. This equation was solved numerically to obtain the concentration distribution in the waste matrix at different time periods as shown in Fig. 16. From the concentration distribution, leach rate from the surface of the waste matrix was evaluated in Bq/day. For a sample cement specimen of given diameter and height, containing ^{137}Cs radionuclide, leach rates obtained numerically were compared against the analytical values, for specific boundary conditions. Comparison shows that they are in good agreement.

I) LYSIMETER BASED TRANSPORT OF RADIONUCLIDES IN UNSATURATED ZONE

Waste disposal facilities for radionuclides are generally located in the unsaturated soil zone. In case of breach in the waste disposal facility, radionuclides have to migrate through the unsaturated zone to reach the ground water table. The soil water characteristic curve (SWCC) and unsaturated permeability coefficients are the two important parameters in developing transport models for moisture and contaminant transport in the unsaturated zone.

In this study by using the lysimeter installed at different depths, soil pore-water was collected and analysed for major cations and anions. A synthetic water sample was prepared by adding suitable amounts of major cations and anions to simulate the pore water condition. By using simulated pore water, distribution coefficient and diffusion coefficient values of the soil were determined for ^{90}Sr. From the undisturbed soil specimen collected from the waste disposal facility, total suction measurements were carried out by using the filter paper technique.

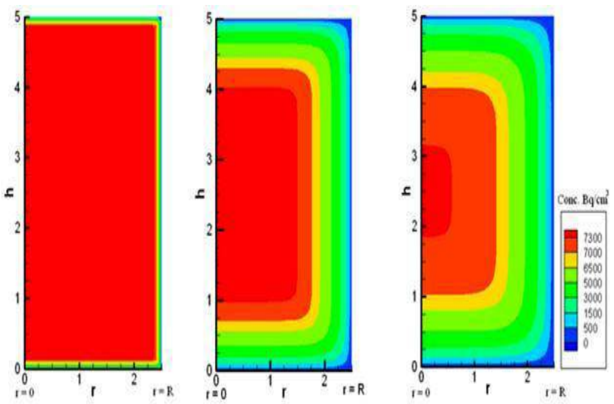


Fig. 16 : Two dimensional plot of Cs-137 concentration distribution in one symmetrical half of the waste matrix after 1day, 100 days and 200 days of leaching

Saturated permeability coefficients were determined after saturating the soil specimens collected. Unsaturated permeability coefficient values were determined from the SWCC generated for Kalpakkam soil and saturated permeability coefficient values, by using the Brooks and Corey equation. From the estimated parameters, strontium migration through the unsaturated zone for different volumetric water content was estimated by using the Ogata-Banks equation for one dimensional contaminant migration through homogeneous soil layer. Estimated ^{90}Sr concentration in the ground water at different water contents of the soil is shown in Fig. 17.

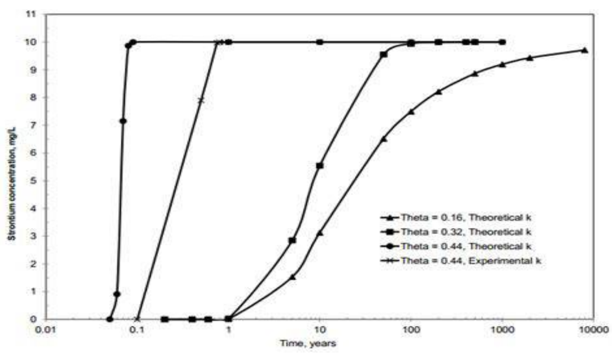


Fig.17 : ^{90}Sr concentration in ground water at different volumetric water content

SRI HIGHLIGHTS 2010-14

4. RESEARCH LABORATORIES

Radiation Physics Laboratory

A well-equipped Radiation Physics laboratory has been established at SRI to carry out radiation shielding and environmental radioactivity related experiments. This laboratory has come into existence in 2011 during the XI plan project to validate the computational tools used for radiation shielding and streaming studies against experiments. The laboratory houses a scintillation based NaI detector and an advanced High Purity Germanium Detector (HPGe) for gamma spectrometric studies; a hand held neutron spectrometer for neutron dose and spectrum measurements; a TLD reader and a portable teletector for gamma radiation survey and monitoring. Apart from the instruments, several activation foils are also procured for carrying out neutron shielding experiments.

In the recent past, the notable experiments carried out by using the laboratory facilities are as follows:

- Lab-scale gamma streaming experiments in ducts and voids of various shapes
- Neutron streaming experiments in bent ducts
- Shield design optimization for Am-Be neutron source
- Efficiency measurement of gamma spectrometric devices and validation through Monte Carlo simulations
- Environmental radioactivity measurements of soils from various parts of south India

In addition to the above experiments, the gamma spectrometric devices are regularly used to measure the radioactivity present in liquid samples collected from various nuclear facilities of IGCAR.

Chemistry Laboratory

SRI chemical laboratory was inaugurated on 21.3.2005 by Shri S.K.Sharma, former Chairman, AERB. The laboratory is equipped with analytical instrumentation such as Ion Chromatography, Atomic Absorption spectrometer, UV-Visible spectrophotometer, Photoreactor, Nephelo-turbidity meter, Water purification system, High precision electronic balance etc. The lab also has facilities to handle low level radioactive works such as fume hoods, once through ventilation system, low level waste collection tank, Hand and cloth monitors and shoe change barrier to cater the needs of Type-I radiochemical laboratory of AERB. Radiation survey and counting instruments such as Area gamma monitors, beta and gamma counting instruments and hand monitors are in place. Analytical services are also rendered to other divisions of IGCAR and BARCF periodically.

The following are some of the analytical activities being carried out at SRI chemistry laboratory

- Characterization of bore water sample using wet chemical and instrumental techniques
- Degradation of organic pollutants using in-house synthesized nanoparticles of Titania and Gallia by employing photo reactor

Radiation Physics Laboratory

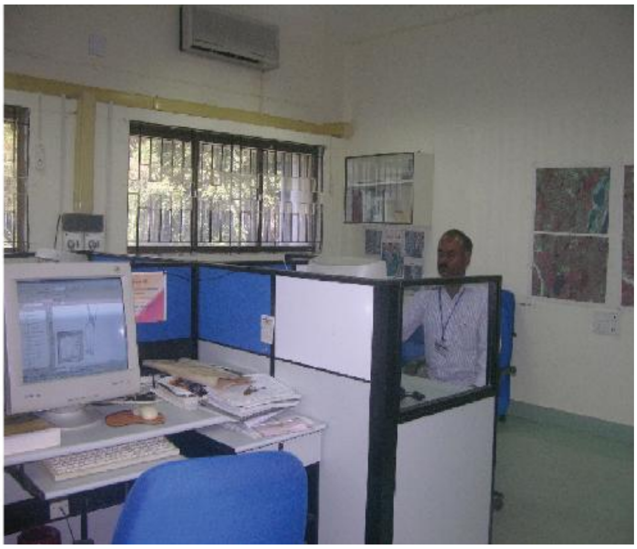
- Uptake studies of radionuclides using inorganic sorbents
- Dip coating of Alumina and Titania prepared by sol-gel techniques
- Geochemical analysis and geophysical analysis of environmental samples

Remote Sensing and GIS Laboratory

A state-of-art laboratory for RS-GIS based activities is established in SRI. The lab consists of

- High resolution topographic data for select areas of interest to AERB
- High resolution satellite data for DAE sites (Digital aerial photo, world view, Quickbird, IKONOS, Cartosat, IRS-PS)
- Geology maps published by GIS for important sites
- Soil maps
- Image processing, GIS and photogrammetric software


Chemistry Laboratory

RS-GIS Laboratory

SRI HIGHIGHTS 2010-14

5. HRD, ACADEMIC INTERACTIONS, SIGNIFICANT EVENTS AND OTHER ACTIVITIES

SRI maintained academic interactions with several institutions through collaborative research. A memorandum of understanding with Anna University was signed to undertake projects for mutual benefits. Theme meetings, training courses and workshops were organized with wide participation both from within and outside DAE. Lectures and invited talks were delivered at premier academic institutions such as IITs. Colloquiums were arranged periodically to share and exchange new research ideas and visits by eminent scientists provided encouragement to take up challenging activities. The groundbreaking ceremony for the construction of SRI Engineering Hall to conduct safety significant experiments was performed on 28th Aug, 2014.

5.1 ATOMIC ENERGY REGULATORY BOARD SIGNS MEMORANDUM OF UNDERSTANDING WITH ANNA UNIVERSITY, CHENNAI FOR RESEARCH COLLABORATION

To promote and develop cooperation and synergy in mutually beneficial areas of research related to regulatory aspects of nuclear facilities and to enhance collaborative research with academic institutions, Atomic Energy Regulatory Board (AERB) signed a Memorandum of Understanding (MoU) with Anna University (AU), Chennai. The MoU was signed by Secretary, AERB and Registrar, AU on May 21, 2013 in Chennai in the presence of Chairman, AERB and Vice-Chancellor, AU.

The MoU is aimed to promote joint activities between AERB and AU with a view to accelerate the pace of research in advanced and challenging areas of nuclear science and technology. Mutual benefits that accrue from the interactions include time bound applied research, enhanced professional skills, and opportunities for students to work in cutting areas of research in science and technology.

(L to R sitting) Shri R. Bhattacharya, Secretary, AERB, Shri S. S. Bajaj, Chairman, AERB, Prof. Dr. P. Kaliraj, Vice-Chancellor, AU and Dr. S. Sivanesan, Registrar, AU MoU with senior officials of AU and Safety Research Institute, Kalpakkam

The generic areas of collaborative research include, but are not limited to, Reliability Engineering, Physics, Environmental and Geographical Information Sciences, Remote Sensing, Thermal hydraulics, Structural Mechanics, Disaster Mitigation and Management etc. The MoU facilitates sharing of infrastructural resources, promote scholarly activities and provide opportunity for researchers / doctoral students of AU leading to strengthening of research activities at Safety Research Institute, the regulatory research wing of AERB located at Kalpakkam. The MoU will enhance academic interactions bringing the vision of establishing academic and institutional collaboration closer to fruition in regulatory research activities.

5.2 GUIDANCE TO TSOs AND RESEARCH SCHOLARS

SRI-AERB has been providing project work and guidance to students and research scholars in science and engineering in the area of reactor physics, structural analysis, PSA and reliability, computational fluid dynamics, fire modeling, ground water flow and radionuclide migration etc. During the period, SRI Officers have guided research scholars and students on the following projects:

- Guidance was provided to Trainee officers from the BARC training school at IGCAR, for the following project works;
  - Numerical Studies of Hydrogen Distribution in BWR Mark–I Containment.
  - Numerical Studies on Hydraulic Transients during Pump Startup and coast down in an Adiabatic Closed Loop System
  - Degradation of Hydrazine in aqueous streams by using Advance Oxidation Methods
  - Treatment of degradation products of TBP by using Advanced Oxidation Processes
  - Uncontrolled withdrawal of a CSR in PFBR
  - Analysis of critical experimental benchmarks by deterministic method
  - Application of Chernick model for estimation of reactivity due to load transients in a thermal reactor

- Guidance was provided to MSc (Physics) students of Calicut University, Kerala for the following dissertation project works;
  - Efficiency calibration of High Pure Germanium Detector
  - Efficiency calibration of Sodium Iodide Scintillation Detector
- Expert guidance and technical advice was provided to HBNI Research Scholar, IGCAR for Ph.D. research work on “Development & application of PSA methodologies for estimating risk associated with nuclear systems & facilities”.

5.3 REGULATORY INSPECTIONS

SRI Officers participate in the regulatory inspections periodically as part of the AERB team.

5.4 LECTURES / INVITED TALKS

- “Enclosure Fires” at MAPS, Kalpakkam, during the fire safety week, April 2014.
- Dr.H.Seshadri, SRI delivered two invited talks on “Challenges in the chemistry aspects of nuclear fuel cycle” and “Synthesis, Characterization and evaluation of nanocatalysts for waste treatment applications” in the National Conference on Advances in Applied Chemical Sciences and Materials (ACSMT2014) held at Bharathidasan Institute of Technology, Anna University, Tiruchirappalli on Oct. 17-18, 2014.
- “Experimental program and numerical studies on enclosure fires at SRI” at the International Workshop on Fire Research, at IIT Kanpur, August 2013.
- Dr. C. Senthil Kumar, SRI delivered an invited lecture on reliability and probabilistic safety assessment (PSA) methodology and its application in nuclear industry during Feb. 14-15, 2013 at SSN College of engineering, Chennai.
- Dr. D. K. Mohapatra, SRI delivered a lecture on “FBR Physics” in the Reactor Physics refresher course conducted at AERB, Mumbai on 26th March 2013.
- Shri R.Kaviyarasan, delivered an invited lecture on “Hydro-geochemical modeling using PHREEQC and MODFLOW during 21-31 Jan 2013 at Annamalai University, Chidambaram.
- Dr. D. K. Mohapatra, SRI delivered an invited lecture on “Theoretical and experimental investigations in KAMINI” in the Theme Meeting on Experiences and Challenges in the Physics Aspects of Design and Operation of Experimental Reactors and Mockup Facilities held at IGCAR, Kalpakkam on 7th Sept. 2012.
- “Design of an experimental facility for compartment fire studies” at the Workshop on Evolving Trends and Technologies in Fire Safety” at AERB Mumbai, February 09, 2012.
- Dr. D. K. Mohapatra, SRI delivered a lecture on “Physics aspects of metal fuelled fast reactors” in the IAEA Technical Meeting on Fast Reactor Physics and Technology held at IGCAR, Kalpakkam during Nov.14-18, 2011
- “Experimental and numerical modeling of enclosure fires” at the Theme Meeting on Challenges in Thermal Hydraulics of Nuclear Reactors, at IGCAR Kalpakkam, February 18-19, 2010.
- Lectures at BARC training school at IGCAR
  SRI officials continued to serve as faculty members and contributed to the training of science and engineering students of BARC training school at IGCAR.
- Lectures at IIT Madras / IIT Kanpur
  SRI officials delivered lectures for M. Tech/ M.S/ Ph.D. students on fire safety and FBR heat transport systems.

5.5 SRI COLLOQUIA

Sl. No.	*Name of speaker*	*Topic*
1	Dr. C. Senthil Kumar	Software reliability estimation methods
2	Shri M. Shankar Ram	Application of Index-based methods for mapping land use categories
3	Shri. Arun Aravind	Transmutation Characteristics of Minor Actinides in thermal and fast neutron spectra
4	Ms. L. Thilagam	Reactor Physics Analyses of LEU and MOX Fuelled Light Water Reactor (LWR) Cores Using Indigenous Code Systems
5.	Mr. J. Christopher	Tensile Flow and Work-hardening behaviour of P9 Steel
6	Smt. R. Deepthi Rani	Role of Colloids in the Transport of Radionuclides in Groundwater
7	Shri Rahul Shukla	Estimtion of neutron flux by principle component analysis(pca)
8	Shri Pranav Paliwal	Computational fluid dynamic investigations on cellular convection
9	Shri Anuj Kumar Deo	Behaviour of MOX fuel & characterization of Fuel-clad gap during irradiation
10	Shri Sudhanshu Shekhar Singh	Benchmark problems for three-dimensional neutron transport code

11	Shri Sajith Mathews	Safety Analysis of Passive Systems
12	Shri Nilesh Agrawal	Hydrogen safety studies using in-house codes
13	Shri Suresh Kumar	Application of ultrasonic nondestructive evaluation for microstructural characterization and defect detection in PFBR components
14	Dr. S.K. Gupta	Application of Conduction Heat Transfer in Nuclear Power Plants: Some Advanced Topics
15	Shri Krishna Chandran R.	Thermal hydraulic studies on 'Safety Grade Decay Heat Removal System'
16	Dr. C. Anandan	Spatial mapping of Sea Surface Temperature of MAPS Condenser Coolant Discharges Using RS – GIS
17	Shri Sudhanshu Shekhar Singh	Effect of Thorium introduction in Metal fuelled fast reactor
18	Shri K. K. Vaze	Overview of Activities of Reactor Design & Development Group, BARC
19	Dr. S.K. Gupta	Certain aspects of Thermal Hydraulics of Nuclear Reactors.
20	Shri Prashant Sharma	Development of Plant Dynamics Code for Steam Water System of PFBR
21	Dr. H. Seshadri	Degradation of organic pollutants encountered in nuclear industry using environmentally benign technologies
22	Shri Seik Mansoor Ali	Numerical Modeling of Flames/Fire over Condensed Fuel Surfaces
23	Shri Jagannath Mishra	Evaluation of failure probability of components
24	Shri Adinarayana Karibandhi	Modelling studies related to Near Surface Disposal Facility (NSDF)
25	Dr. D.K. Mohapatra	Physics Aspects of Metal Fuelled Fast Reactors
26	Shri Ramakrishna Pagoti	Safety studies pertaining to the metallic nuclear fuel materials
27	Shri Krishan Kumar	Safety Studies related to Pyrochemical reprocessing of nuclear fuels
28	Shri Kaviyarasan	Application of Groundwater Model for Scientific and Regulatory Purpose: Kalpakkam – A Case Study
29	Shri Venkata Rajeev Gade	Numerical investigations of hydrogen recombination within passive autocatalytic re-combiners

5.6 SEMINARS / CONFERENCES / WORKSHOPS ORGANISED

S.no	*Title*	*Period*
1.	International Workshop on External Flooding Hazards at Nuclear Power Plant Sites in Commemoration of the 5 Years of Indian Ocean Tsunami Event	January 11-15, 2010
2.	Training Program on RELAP was organized by Safety Research Institute (SRI) under the auspices of the Indian Society of Radiation Physics, Kalpakkam Chapter	Apr 9 – May 13 2011
3.	Orientation Course for Regulatory Process (OCRP-SRI-2011) at SRI, Kalpakkam	June 2011
4.	International workshop on "New Horizons in Nuclear Reactor Thermal Hydraulics & Safety"	Jan. 02-03, 2012
5.	Theme Meeting on Severe Accident Analysis and Experiments	Apr. 26-27, 2012

1. The 108th board meeting was held at SRI conference Room during February 13-14, 2012.
2. Orientation Course on Environmental Impact Assessment (EIA) of Nuclear Fuel Cycle Facilities was organised by IGCAR in association with IARP (K) at SRI Guest House, Anupuram during 8-12 July 2013.

Inauguration of RELAP Training Programme at SRI Guest House

Inauguration of OCRP-2011

Theme Meeting on Severe Accident Analysis and Experiments

5.7 LAYING FOUNDATION FOR SRI ENGINEERING HALL

To support the regulatory decision making by AERB and to carry out dedicated in-house experiments, an engineering hall is being set up under the XII plan. This building will house a high bay (160 sq. m, height 8.5 m), a low bay (160 sq. m, height 4.25 m), an office space (160 sq. m) along with an entrance lobby. In the first phase, three experimental facilities, namely Hydrogen Mitigation Facility (HYMIF), Water and Steam Interaction Facility (WASIF) and Calandria Vessel under Core Collapse (CVCC) facility are envisaged to come up within the high bay of the hall and radiation shielding experiments are planned within the low bay.

Shri S.S. Bajaj, Chairman, AERB, being briefed by Shri V. Balasubramaniyan, Dir., SRI, on the details of the SRI Engineering hall followed by laying of foundation for the building by Chairman

The groundbreaking ceremony for the construction of SRI Engineering Hall was performed on 28th Aug, 2014. Chairman, AERB kindly graced the occasion and laid the foundation for the building. Several senior officials of IGCAR and SRI including Dr. T.Jayakumar (Dir., MMG & Officiating Dir., IGCAR), Shri K.K.Rajan (Dir., ESG, IGCAR), Shri. C.Sivathanu Pillai (AD, CEG) and Shri V.Balasubramaniyan (Dir., SRI) were present on the occasion. Chairman, AERB evinced keen interest in the details and addressed the gathering highlighting the benefits of having a dedicated in-house experimental facility.

5.8 VISIT OF DIGNITARIES

1) The Director General of IAEA Mr. Yukiya Amano, Chief De Cabinet Mr. Rafael Crossi and Director, Office of External Relations & Policy Co-ordination Mr. S. Akbaruddin of International Atomic Energy Agency accompanied by Dr. Baldev Raj, Director, ICAR and other senior DAE officials visited Safety Research Institute, Kalpakkam on January 18th, 2011. Director General evinced keen interest in the activities of SRI and made specific enquiries with respect to the research on seismic evaluation of nuclear plants.
2) Prof. Balraj Seghal, Professor of Nuclear Safety, Royal Institute of Technology (KTH), Sweden visited during January 30-31, 2012.

Visit of Director General of IAEA Mr. Yukiya Amano

Visit of Dr. M. R. Srinivasan to SRI, Kalpakkam

3) Dr. M. R. Srinivasan, Former Chairman, Atomic Energy Commission and currently Member, National Security Advisory Board visited Safety Research Institute, Kalpakkam on February 27, 2012.

5.9 AWARDS AND ACHIEVEMENTS

1. In 2011, Dr. C. Senthil Kumar and Shri Jagannath Mishra received the group achievement award along with 8 other scientists/engineers for their individual contributions and excellent team work for successfully accomplishing the activity titled “Seismic Re-evaluation of FBTR”.

2. In 2012, Dr. L. Thilagam, SRI received the group achievement award along with 6 other scientists/ engineers for their individual contributions and excellent team work for successfully accomplishing the activities related to commissioning of KK-NPP-1&2 at Kundakulam.

5.10 HIGHER STUDIES

Ph.D.

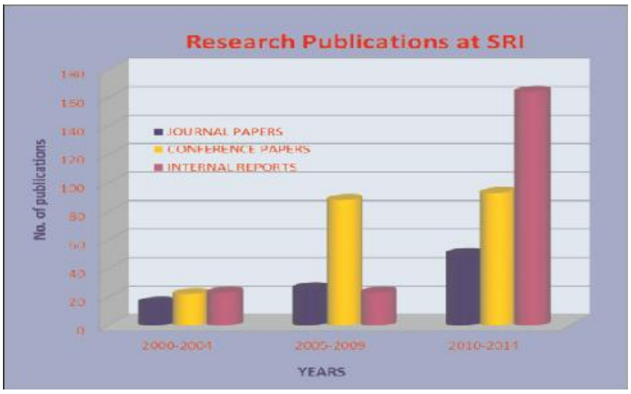
- L.Thilagam, “Reactor Physics Analysis of LEU & MOX Fuelled Light Water Reactor Cores”, University of Madras, Chennai (November 2011).
- Seik Mansoor Ali, “Numerical Modeling of Laminar Flame Characteristics over Liquid Methanol Surface”, IIT Madras, Chennai (July 2012).
- H.Seshadri, “Degration of organic pollutants in aqueous streams using novel nano sized photo catalysts”, University of Madras, Chennai (September 2012).
- Nilesh Agrawal, “Studies on Mixing and Deflagration Potential of Hydrogen-Air and Hydrogen-Steam-Air Distributions in Enclosures”, IIT Madras, Chennai (July 2013).

M.Tech.

- Anuj Kumar Deo,” Thermal design of metallic fuel pin & core for future FBRs”, Homi Bhabha National Institute, Mumbai (2010)
- Pranav Paliwal, “Computational fluid dynamic investigations on cellular convection”, Homi Bhabha National Institute, Mumbai (2011)
- Prashant Sharma, “Plant dynamics of PFBR steam water system”, Homi Bhabha National Institute, Mumbai (2012)
- Venkata Rajeev Gade, “Numerical study of Passive Auto Catalytic Recombiner using a detailed Reaction Model”, Homi Bhabha National Institute, Mumbai. (2014).

PUBLICATIONS

Peer reviewed International Journals	:	50
National and International Conferences	:	93
Internal Reports	:	164
Databases / Compendium	:	2



JOURNAL PUBLICATIONS

1) U.Karmegam, S.Chidambaram, P.Sasidhar, R.Manivannan, S.Manikandan and P.Anandhan, “Geochemical Characterization of groundwater’s of shallow coastal aquifer in and around Kalpakkam, South India, Research Journal of Environmental and Earth Sciences, 2(4), 170-177, 2010

2) L. Thilagam, R. Karthikeyan, V. Jagannathan, K.V. Subbaiah and S.M. Lee, “Intercomparison of JEF2.2 and JEFF3.1 Evaluated Nuclear Data through MCNP Analysis of A VVER-1000 MOX Core Computational Benchmark”, Annals of Nuclear Energy, Vol. 37, pp. 144-165, 2010.

3) L. Thilagam, K.V. Subbaiah, K. Thayalan and S.E. Kannan, “Suitability of Point Kernel Dose Calculation Techniques in Brachytherapy Treatment Planning”, Indian Journal of Med Physics, 235: 88-99, 2010.

4) Seik Mansoor Ali, V. Raghavan and Shaligram Tiwari., (2010), “A study of steady laminar diffusion flame over methanol pool surface”, International Journal of Heat and Mass Transfer, Vol. 53 pp. 4696-4706.

5) Seik Mansoor Ali, V. Raghavan and Ali Rangwala., (2010), “A numerical study of quasi-steady burning

characteristics of a condensed fuel: effect of angular orientation of fuel surface", Combustion Theory and Modelling, Vol.14 (4), pp. 495-518.

6) R. K. Chandran, I. Banerjee, G. Padmakumar, K. S. Reddy., (2010),"Numerical analysis of thermal striping phenomena using a two-jet water model", Engineering Applications of Computational Fluid Mechanics, Vol. 4, pp. 209-221.

7) S.Chidambaram, U. Karmegam, M.V. Prasannna, P. Sasidhar and M.V.Vasanthavigar, "A study on hydrochemical elucidation of coastal groundwater in and around Kalpakkam region, southern India", Environ Earth Sci., Vol 64(5), 2011.

8) R.Deepthi Rani and P.Sasidhar, Physico-chemical characterization of clays at Kalpakkam nuclear plant site towards assessment of radioactive waste disposal, Nuclear Engineering and Design, doi: 10.1016/J.nuclengdes.2011.03.017

9) Jayashree E, Ramaswamy V and Seshadri H, "Synthesis, Characterization and photocatalytic activity of CdS-Montmorrillonite clay composites", Nanoscience and Nano Technology: An Indian Journal, 5(3), 2011

10) Nilesh Agrawal, Seik Mansoor Ali, K. Velusamy, S.K. Das, (2011) "A Method to characterize mixing and flammability of hydrogen air mixtures in enclosures", International Journal of Hydrogen Energy, 36, 12607-17.

11) R. Krishna Chandran, Indranil Banerjee, G. Padmakumar and K. S. Reddy (2011), "Investigation of Thermal Striping in Prototype Fast Breeder Reactor Using Ten-Jet Water Model", Heat Transfer Engineering, Vol. 32 (5), pp. 369-383.

12) Seik Mansoor Ali et al., (2011), "A numerical study of leading edge anchoring characteristics of ethanol sourced laminar boundary layer diffusion flames", Combust. Sci Technol. 183, pp. 1133-1145.

13) C. Anandan and P. Sasidhar, Geomatics, "Changes in coastal morphology at Kalpakkam, East Coast, India due to 26 December 2004 Sumatra tsunami", Natural Hazards and Risk.2: 2, 183- 192, 2011.

14) P. Priyada, M. Margret, R. Ramar, Shivaramu, M. Menaka, L. Thilagam, B. Venkataraman, and Baldev Raj, "Intercomparison of gamma scattering, gammatography, and radiography techniques for mild steel non-uniform corrosion detection", Rev. Sci. Instrum., Vol. 82, pp. 035115, 2011.

15) Vijayalakshmi Saravanan, C. Senthil Kumar, Sasikumar Punnekkat and D.P. Kothari, "A study on factors influencing power consumption in multithreaded and multicore CPUs", WSEAS Transactions on Computers, Vol.10(3), 2011.

16) T. Sajith Mathews, A. John Arul, U. Parthasarathy, C. Senthil Kumar, K.V. Subbaiah, and P. Mohanakrishnan, "Passive system reliability analysis using Response Conditioning Method with an application to failure frequency estimation of Decay Heat Removal of PFBR", Nuclear Engineering and Design, Vol. 241 (6), 2011.

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