07 SAFETY ANALYSIS AND RESEARCH

AERB recognises the importance of Safety Analysis & Research in support of its regulatory function. In-house safety related R&D helps in obtaining deeper insights into the issues concerning nuclear and radiation safety to arrive at scientifically sound and risk informed regulatory decisions. Safety analysis and research activities are carried out by AERB as a part of its regulatory activities. Several important developmental studies were taken up by AERB and completed during this year. A brief overview of these activities is presented in the following sections.

7.1 THERMAL HYDRAULICS SAFETY STUDY

7.1.1 Heat Transfer Studies on FBTR Fast Flux Experimental Facility

As part of regulatory review of the proposed Fast Breeder Test Reactor (FBTR) fast flux experimental facility for irradiation of radioactive samples at 400 kW, an independent verification study was carried out. During the irradiation procedure, it is essential to ensure that the temperatures do not exceed the approved safe limit during its operation of six hours. A detailed 3-D transient heat transfer analysis was carried out using COMSOL (Multi-physics software). The temperature transient at the centre of the Pu coated gold foil during irradiation is shown in Fig. 7.1(a). The temperature contour of the entire domain after six hours of operation is given in Fig. 7.1(b). It is seen that the maximum steady state temperature is well within specified limits.

7.1.2 Thermal Transient Analysis of CORAL High-level Waste Raffinate

To support the regulatory review related to replacement of cooling towers of CORAL high-level waste raffinate, an independent verification calculation for estimating the maximum temperature of raffinate waste solution, during unavailability of primary cooling water, was carried out. A transient heat transfer model was developed in-house and used to estimate the raffinate level and temperature transients for two separate scenarios of raffinate loading for 5 year period;

Case-1: The entire raffinate generated by reprocessing of 72 fuel assemblies (after a two-year cooling period, subsequent to attaining a maximum burn-up) is assumed to be filled in the storage tank in one single campaign.

Case-2: A more realistic scenario, wherein, raffinate generated by reprocessing the specified quantity of fuel assemblies each year is progressively filled and stored in the tank.

The temperature transients for Case-1 and Case-2 are depicted in Fig. 7.2 (a) and (b) respectively. The results are in agreement with results in the submission.

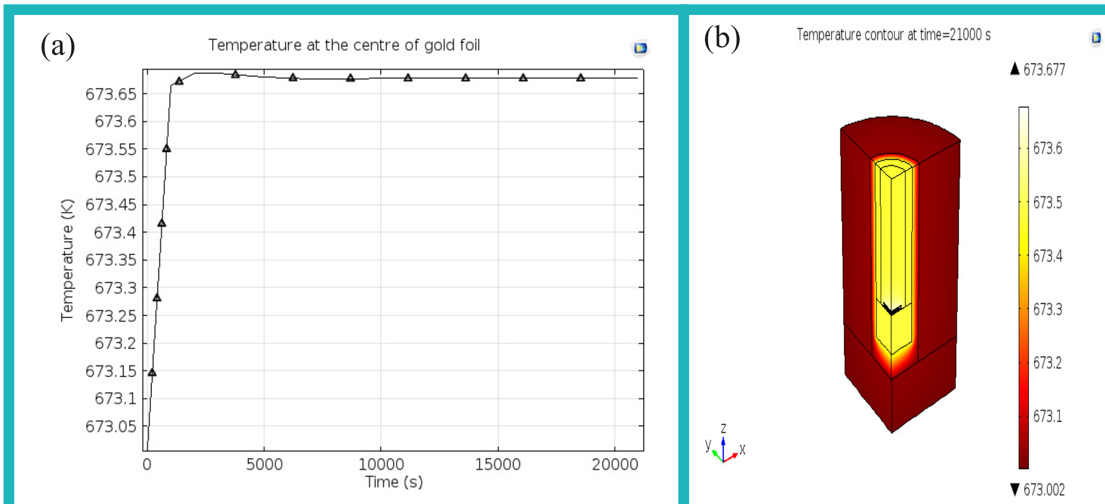
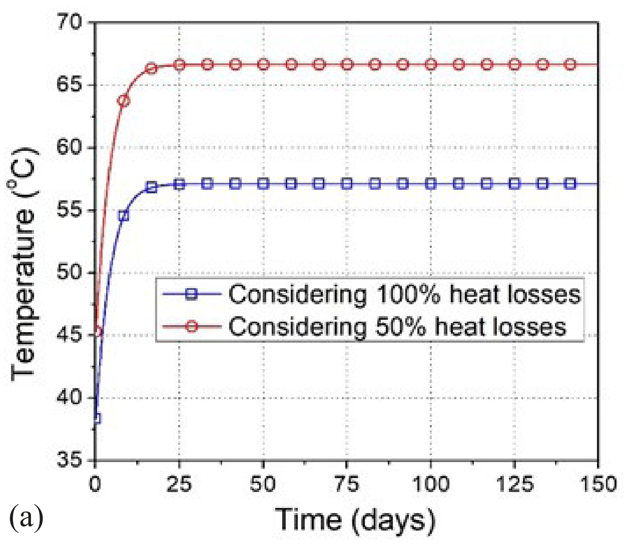


Fig. 7.1: (a) Foil Temperature Transient and (b) Contour Plot



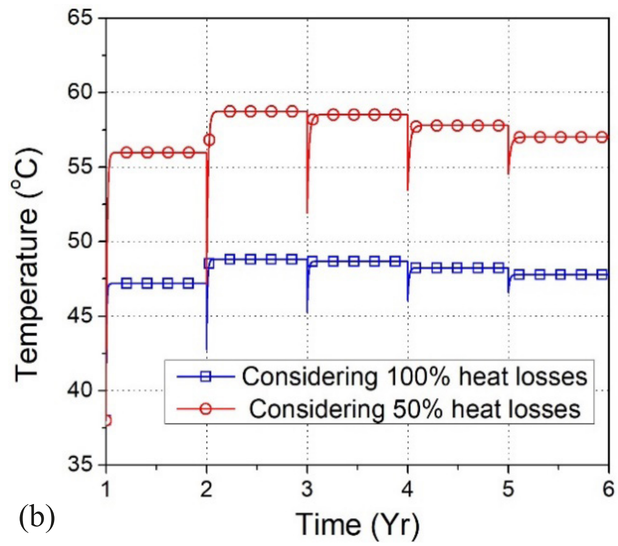


Fig. 7.2: Raffinate Temperature Transients for (a) Case-1 and (b) Case-2

7.1.3 Inadvertent Opening of PDHRS Valves in 700 MWe PHWRs

For removal of core decay heat following a severe accident scenario, nuclear reactors are equipped with passive heat removal systems that depend on the natural circulation principle. Studies were conducted to evaluate the effect of inadvertent opening of the Passive Decay Heat Removal System (PDHRS) valve during normal operation. The study revealed that reactor trip occurs on the low SG pressure signal and the changes in various parameters attain stabilized conditions within 30 minutes and all the thermal hydraulics parameters were within the prescribed operating limits.

7.1.4 Studies on Passive Containment Cooling System (PCCS)

A conceptual model of PCCS for 700 MWe PHWR containment consisting of parallel heat removal loops was developed earlier. A scoping study was initiated for setting up small scale PCCS experimental facility. A generalized model to simulate two-phase natural circulation in parallel channels was developed to study the performance of the proposed facility. This model can simulate various conditions of heat sink (infinite, open to atmosphere and external cooling etc.). To arrive at the design features of the PCCS facility, several parametric studies were conducted. Variation of flow rate is depicted in Fig. 7.3.

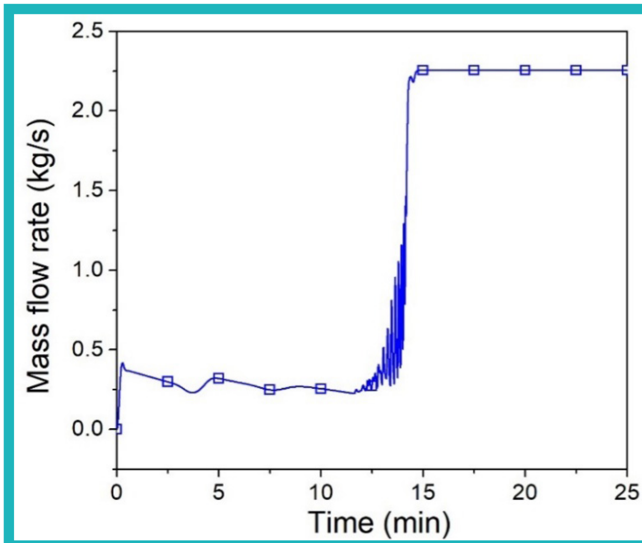


Fig. 7.3: Typical Flow Transient in the Scaled-down PCCS Loop

7.1.5 CFD Simulations of HR 49 Test of THAI Facility

Computational Fluid Dynamics (CFD) simulations were carried out as part of OECD/NEA Project with an objective to study the onset of recombination of Hydrogen in presence of counter current flow and saturated steam environment at elevated pressure using FLUENT. The analysis methodology and models were validated against experimental data of HR 49 test of THAI-3 facility. Analyses have been carried out for various counter-current velocities ranging from

0.1 m/s to 5 m/s. It was observed that time of onset of recombination is inversely proportional to forced circulation (counter current) velocity.

7.1.6 CFD Simulation of Containment Hydrogen Distribution for 700 MWe PHWR

The main objective of this work is to study the distribution of hydrogen in containment for accident scenario LOCA+LECCS+LMODC. The CFD simulation up to 17000 s was completed and there is uniform hydrogen volumetric concentration of 5.4% observed in containment above 120 m elevation (top of FM vault plane). Diagonally opposite to FM vault, two distinct layers of hydrogen concentration are observed. Predictions show that hydrogen concentration of 3.5% and 2.5% by volume in upper and lower portions respectively.

7.1.7 Coupled Neutronics Thermal-Hydraulics Safety Analysis for 540 MWe PHWR

AERB under the auspices of DAE Steering Committee to coordinate Safety Research (DAE-SCSR) has initiated a reference benchmark for the analysis of an initiating event involving strong neutron kinetics / thermal-hydraulic interaction for Indian reactors. The benchmark problem involves analysis of postulated LOCA in one of two loops. The exercise is planned in two phases. Phase 1 (1A, 1B and 1C) exercises are focused on stand-alone core Neutronics and system thermal hydraulics modelling, whereas phase 2 involves integrated core 3D neutronics and system thermal hydraulics modelling. The results of Phase 1A have been inter-compared and compiled. AERB, BARC, NPCIL and IGCAR are participating in this exercise. Specification for Phase 1C has been prepared and distributed to participants.

7.1.8 ISOMED Source Storage Cooling Analysis and Cooling Coil Design

ISOMED, the first irradiation plant for sterilization of medical products, was set up in India at Trombay with the assistance of the United Nations Development Programme (UNDP). In this facility, ^{60}Co is used as the source of radiation and the plant had been in operation for more than four decades (since 1974). The radiation source is housed in a concrete cell and this concrete is cooled by water circulating through the cooling coils embedded in the concrete shield. For the assessment of the cooling coil capability, a 3-D model of the concrete shield along with the steel liner was developed in multi-physics software COMSOL. Further, as the facility is undergoing renovation, as desired by BRIT, AERB performed the thermal analysis to arrive at the new coil design so that the concrete temperatures remain lower than 65°C. Several configurations of cooling coil were studied for low and high density concrete and an optimum pitch and location of the cooling coil for high density concrete was arrived at. Fig. 7.4 (a) and (b) show the temperature distribution in the concrete with existing and modified design respectively. Further, simulations were performed to arrive at the natural circulation flow of the loop using RELAP5 computer code also.

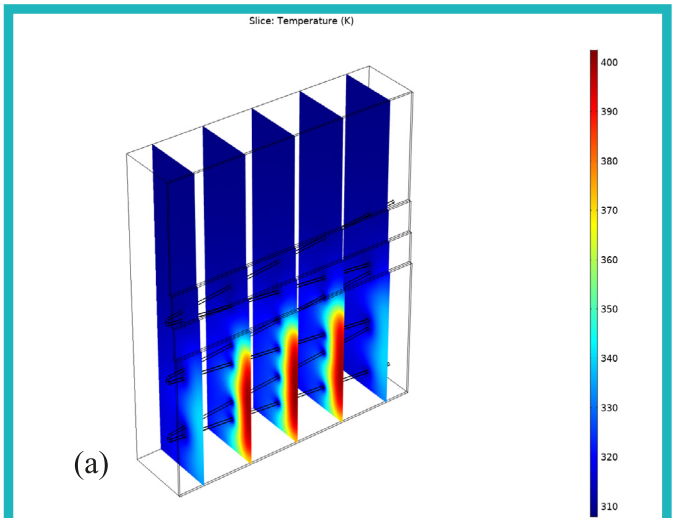


Fig. 7.4 :(a) Temperature Field with Existing Coil & Water Cooling

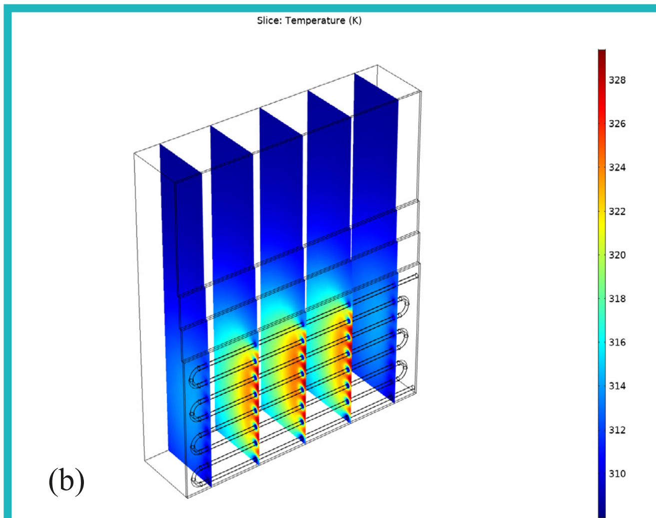


Fig. 7.4: (b) Temperature Field Slices with Modified Coil & Water Cooling

7.1.9 Containment Thermal Hydraulics response of KAPS-1 Small Leak Event

Containment thermal hydraulics analysis of KAPS-1 following small leak has been carried out by Lumped Parameter (LP) code. The predictions are compared with the recorded plant data and NPCIL predictions and are in reasonable agreement for the event.

7.2 SEVERE ACCIDENT STUDIES

7.2.1 Calculation of Radiation view Factor of 37 Pin Fuel Bundle

In severe accident analysis of PHWRs, as the fuel temperature rises, radiation heat transfer from fuel to Pressure Tube (PT) turns out to be significant. State of the art lumped parameter codes depend on user-defined-input view factors. Due to unavailability of standard correlation/analytical formulae to estimate view factor for PHWR fuel bundle geometry [Fig. 7.5(a)], an attempt has been made to predict it using Monte-Carlo Method. The Monte-Carlo code is validated before using it for PHWR bundle. View factors between pins of bundle and pin to PT obtained from the code is shown in Fig. 7.5(b). The predicted view factor has been used in LP codes to estimate fuel temperature during transients.

7.2.2 AERB Source Term Estimation Tool

During this period, methodology for estimating the source term for KKNPP and MAPS reactors was developed and integrated with the tool. This tool has been used for estimating source term to support the emergency exercise conducted at MAPS site. Thermal hydraulic inputs required for assessment of source term were derived from the existing analyses.

7.2.3 Analysis of Postulated Main Steam Line Break (MSLB) of 700 MWe PHWR

As part of independent verification, analysis of MSLB with crash cool down was carried out. A postulated Double-Ended Guillotine Break (DEGB) has been simulated in one of the main steam line inside primary containment. The transient analysis was carried out for a time of 490s. The results have been compared with utility submissions and found to be in agreement.

7.3 SAFETY ANALYSIS CODE DEVELOPMENT

7.3.1 Development of Models for PRABHAVINI Code

PRABHAVINI is an integral safety analysis code being developed to address Design Basis Accidents (DBA) and Design Extension Conditions (DEC) in the Indian nuclear reactors. The development work is being carried out under DAE-SCSR with contributions from BARC, NPCIL, AERB and IGCAR. Following modules were developed and contributed by AERB.

Accumulator Model: Accumulator is used in NPPs as ECCS to remove heat from the core in an event of LOCA. Inter-code comparison was carried out to validate the accumulator model.

Fission Product – Decay Heat Model: A Fission Product Decay Heat (FP-DH) computer code based on first principles has been developed at AERB and is capable

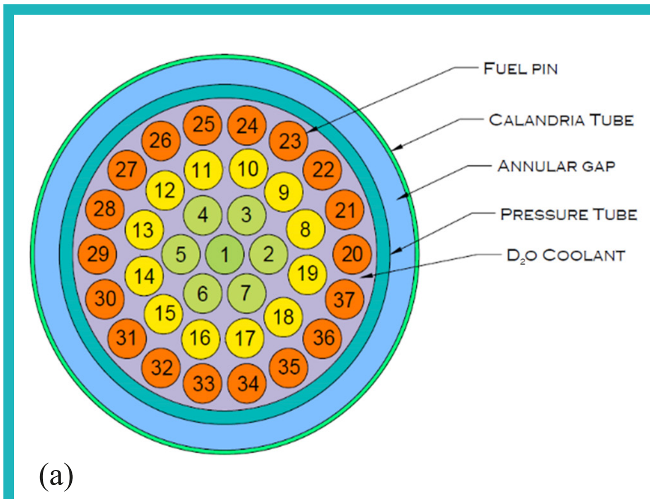


Fig.7.5(a): Schematic of 700 MWe IPHWR Fuel Bundle

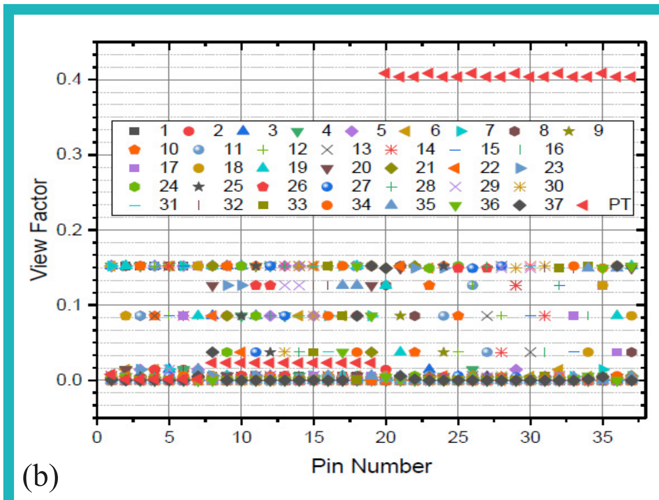


Fig.7.5(b): View Factor of Pin-to-Pin and Pin-to-PT for the Fuel Bundle

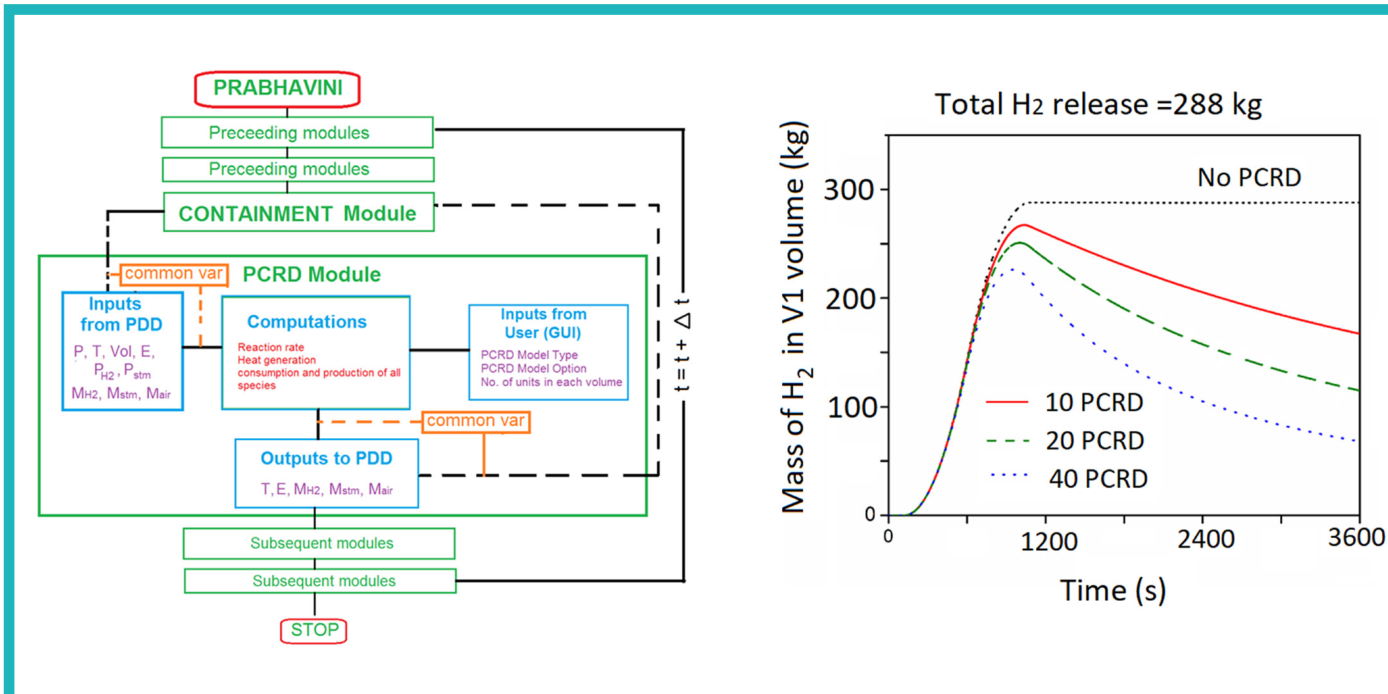


Fig. 7.6: (a) Flow Chart of Coupled Analysis of PCRD Module with Containment Module, (b) Transient of Hydrogen Mass in V1 volume due to presence of various numbers of PCRDs

to predict decay heat based on fission product inventory. It computes decay of each isotope and uses energy released from decay to compute total decay heat. FP-DH library contains half-life, parent isotope, mode of decay (α, β and γ) and energy associated with decay for 699 isotopes.

Passive Autocatalytic Recombiner Device (PCRD) Model: AERB has contributed to the development of a point model for PCRD. The PCRD model has been coupled with the containment model (PARIRODHAN) and features related to release, transport and leakage of hydrogen have also been incorporated in PARIRODHAN. Coupled simulations of hydrogen transients in the containment using PCRD model and PARIRODHAN were carried out as per the flow chart given in Fig. 7.6(a). Performance assessment of the coupled model has been completed. Typical results are shown in Fig. 7.6(b).

Containment Spray Model: A standalone containment spray system model has been developed. An empirical correlation for heat removal rate by spray system was developed through multiple regression analysis and it has been integrated with PRABHAVINI.

7.4 RADIOLOGICAL ASSESSMENT & ENVIRONMENTAL SAFETY STUDIES

7.4.1 Numerical Simulation of Atmospheric Flow Field over a Complex Terrain NPP Site

NPP site in the valley region can generate compound circulation patterns. Atmospheric flow field over such a terrain was investigated using Numerical Weather Prediction (NWP) model and Weather Research and Forecasting (WRF) for three prominent seasons; summer, winter and monsoon for predicting the atmospheric dispersion pattern of a hypothetical release of SF_6 tracer gas from a 100 m tall stack near to the NPP. Simulated plume showed spatial and diurnal variation in the concentration and extent of transport.

It was found that for summer, the most affected areas would be East, Northeast and West sectors. For the winter study case, plume is found to distribute equally on either side of the release point in the North-East, West-Southwest and Southwest sectors, whereas for the monsoon period, under strong synoptic forcing, plume is mainly found distributed in the East and Northeast sectors (Fig. 7.7).

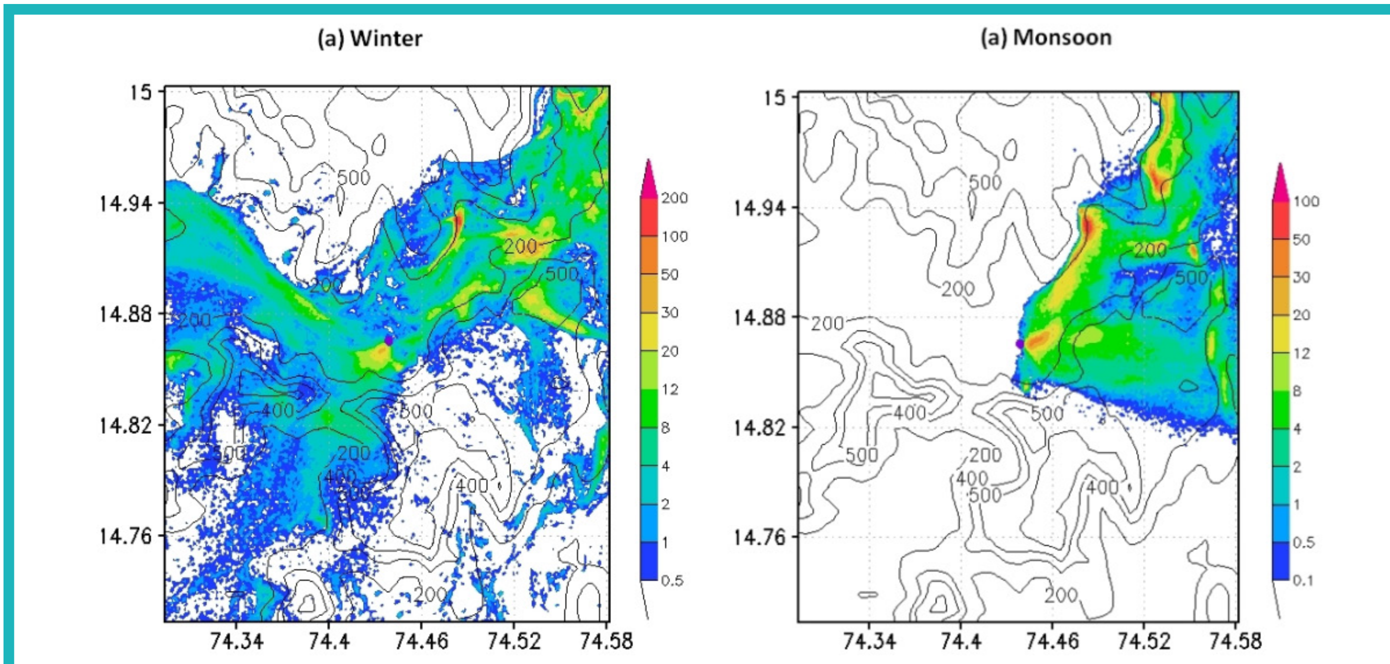


Fig.7.7: Simulated 24 hours Integrated Ground Level Concentration of SF_6 ($\mu g/m^3$) for (a) Winter and (b) Monsoon.

Topography is shown as contour and the release point is represented by a violet dot. Simulations have indicated relatively high ground level concentration around the release point mainly along the valley indicating the flow channelling effect.

7.4.2 Synthesis, Characterization and Evaluation of Novel Extractants for the separation of Lanthanides and Actinides

To minimize long-term radiotoxicity of High Level Liquid Waste (HLLW), the separation of minor actinides and their subsequent transmutation into shorter-lived nuclides is essential. The bis (triazinyl) pyridines (BTP) have been identified as superior extractants for the separation of highly radiotoxic trivalent actinides. The BTP compounds having phenyl rings fitted with sulphonic acid groups are found to be aqueous soluble and they have sufficient hydrolytic and radiolytic stability. In this study, extraction of Americium (Am (III)) and Europium (Eu (III)) together from 3M nitric acid solution is carried out as a first step using glycosamides.

For the separation of Eu (III) from Am (III), the synthesized BTP extractants were successfully tested for the separation of Europium from Americium. A separation factor of more than 200 has been obtained for the BTP extractant under 0.5M acidity (Fig. 7.8). Further to the evaluation, scaling up of 30g of bis-

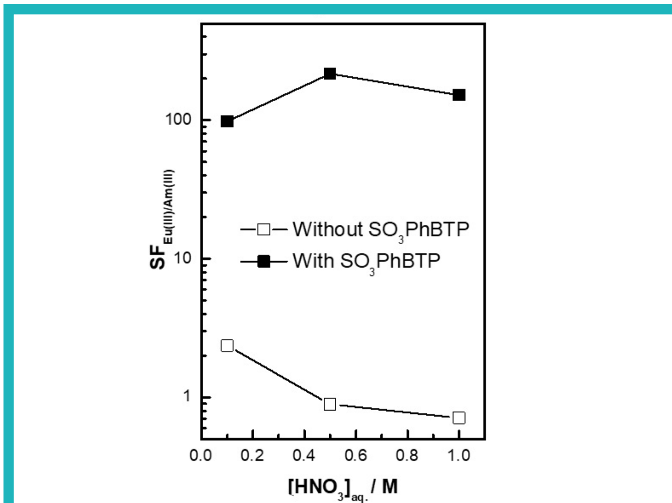


Fig.7.8: Effect of SO_3-Ph-BTP on the Separation Factor (SF) of Eu(III) over Am(III)

triazinyl pyridine extractant has been completed for evaluation of their extraction studies using mixer-settler. Five different aqueous soluble extractants were synthesized and characterized as a part of AERB-CSRP project.

7.4.3 Studies aimed at Man-Rem Reduction due to Liquid Waste Containing Oxalate

Oxalate precipitation route is generally adopted for the reconversion of Plutonium Nitrate solutions to Plutonium Oxide for the tail-end purification process.

Large volumes of oxalic acid supernatant solution were generated in this process and the destruction of oxalic acid is essential for the recovery of plutonium in the supernatant as well as for the safe disposal of the oxalate waste.

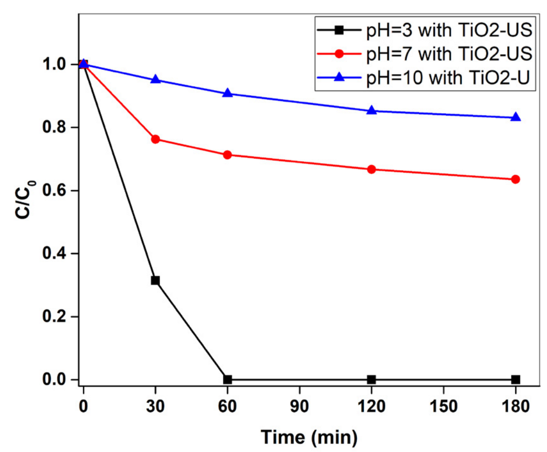


Fig. 7.9: Photocatalytic Degradation of Potassium Oxalate

Complete degradation of oxalic acid from simulated liquid waste containing oxalate ion was successfully carried out using photocatalytic route using photoreactor at SRI Chemistry laboratory. Degradation was followed using ion chromatographic system and complete degradation could be obtained for the simulated waste containing oxalate. Degradation of oxalic acid follow the order, Acidic > Neutral >Alkaline pH (Fig. 7.9). The methodology would help in reduction of man-rem and secondary waste generation.

7.4.4 Retention of Iodine in Aqueous Streams using Novel Adsorbents

Radioiodine could be released from nuclear reactor systems during low probable postulated accidental scenarios. Iodine is expected to be released into the containment as aerosols containing metal iodides such as CsI, AgI, InI, FeI_2, etc. The unique feature is that if iodine can dissolve in water, it can undergo chemical transformations to volatile chemical form which can partition back into the containment atmosphere. For the purpose of retaining iodine in the sump itself, sorbents which can adsorb and retain iodine in the aqueous phase are being employed. For this, silver coated alumina was prepared in-house using chemical impregnation method and characterized using techniques such as XRD, SEM etc. Iodine removal from aqueous phase was successfully demonstrated at different pH ranges. The iodine removal was found to follow the order, Neutral > Acidic > Alkaline.

7.4.5 Synthesis and Thermal Characterization of Cerium loaded Strontium Borophosphate Glasses

Borophosphate glasses have high chemical durability as well as the mechanical stability. They are finding application in vitrification of radioactive waste and glass to metal seals of biomaterials. Strontium borophosphate (SBP) glasses were synthesized by adding lithium as a flux and zinc as a modifier. Cerium has been used as a surrogate for plutonium bearing waste. It is observed that up to 4 mol % of Ce loading does not affect the formation of borophosphate glass. The uniformity of the waste loading was determined by SEM-EDX (Fig. 7.10(a) and (b)). The uniform distribution of Ce could be observed in the glass host by EDX elemental mapping.

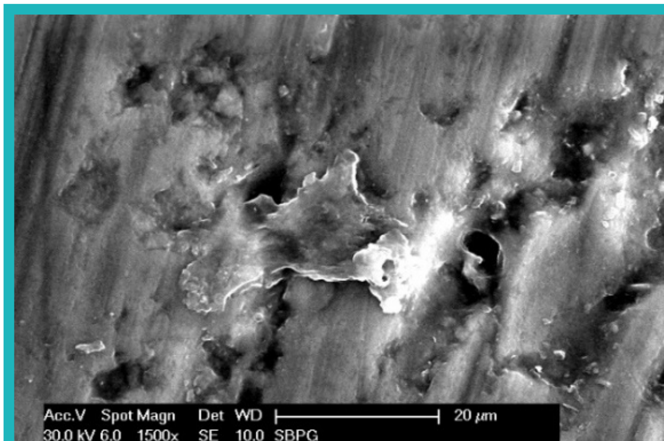
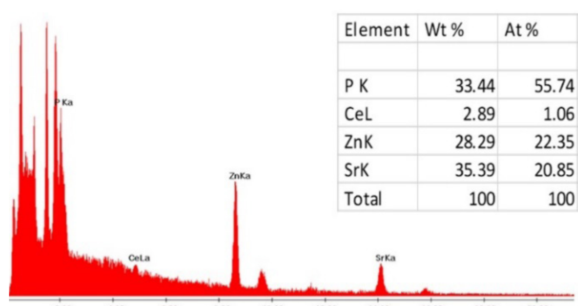


Fig. 7.10(a): SEM Image of C0.5 Sample



Element	Wt %	At %
P K	33.44	55.74
CeL	2.89	1.06
ZnK	28.29	22.35
SrK	35.39	20.85
Total	100	100

Fig. 7.10(b): Elemental Analysis of C0.5 Sample

7.4.6 Generation of Site-specific Data and Creation of Information System

As a part of NREMC activity, a geo-spatial database has been generated for NAPS emergency planning zone on villages with latest population data (Fig. 7.11(a)), emergency zones, sectors, rallying posts. The emergency planning zone constitutes of 316 villages. A GUI based information system for off-site emergency management is developed to assimilate all the site-specific information and integrate with atmospheric dispersion models for query-based data retrieval and display. The system can also simulate atmospheric dispersion to display the release during low probable postulated accidental conditions. An overlay of the plume on the village map provides information for quick assessment of emergency situation (Fig. 7.11(b)). This system can be utilized for verification of decision support system during a postulated accident condition.

7.4.7 Preparation of Land use/Land cover Map for all Sites (Emergency Planning Zone)

Land use/land cover map is prepared for all NPP sites by employing Resourcesat-2 satellite imagery and updated with Landsat-8 data of 2019 based on National Remote Sensing Centre (NRSC) classification manual. Unsupervised classification is performed by using image processing software and then manual classification is performed by following visual interpretation techniques. The land use/ land cover types present in the study area are forest plantation, crop land, agricultural plantation, coastal plantation, build-up area, wastelands including sand dunes and water bodies. For interpretation and classification, False colour composite of Landsast-8 data (Band 3 assigned as Red, Band 2 assigned as Green, Band 1 assigned as Blue) (Fig. 7.12(a)) for Kakrapar site has been created. The major land use categories are forest with plantation, land with scrub, build up area, barren lands, wet sands, agricultural lands and surface water-bodies include rivers, reservoirs, dams, tanks, ponds, backwater etc. (Fig. 7.12(b)).

7.4.8 Geo-spatial baseline Database Generation for Jaitapur Site

Geo-spatial database such as geology, soil, drainage network with surface water bodies, hydro-geomorphology, land use/ land cove, village map with population details and road network is prepared for Jaitapur site (Refer Fig. 7.13(a) and Fig. 7.13(b)). The data can be effectively used in NREMC decision support system.

7.4.9 Preparation of Geo-spatial Database on Soil types around Power Plant Sites

Soil map with physio-chemical property are prepared for Kalpakkam, Kudankulam and Kaiga site. The map contains the physical boundary of each soil type in map form and respective physio-chemical properties in tabular form.

7.4.10 Spatio-temporal Surveillance on Urbanization in the No Growth Zone of MAPS

Medium resolution satellite data of LANDSAT Band 8 having spatial resolution of 15m is compared for the year 2000 and 2005 within 5 km radial zone. The changes in land usage are identified as cultivated lands converted into residential plots (Fig. 7.14).

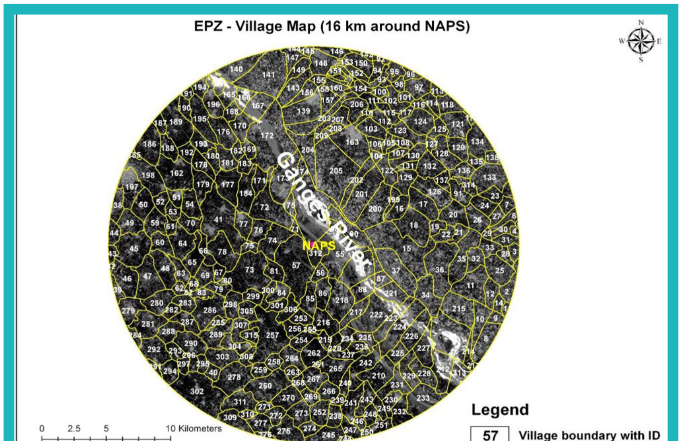


Fig. 7.11(a): Village Map of NAPS Emergency Planning Zone (EPZ)

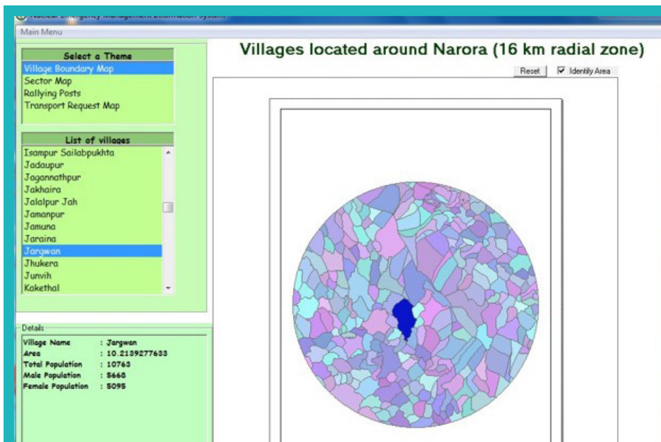


Fig. 7.11(b): NEMIS-Decision Support System for NAPS showing the Village based Query

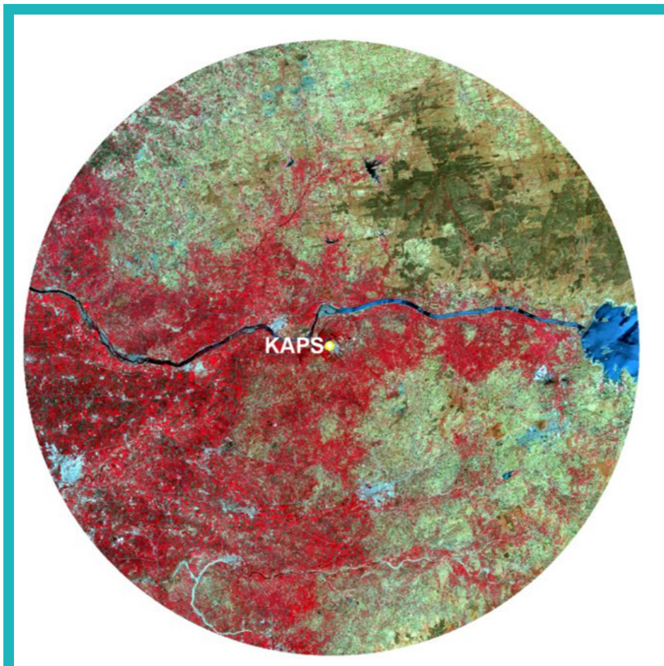


Fig. 7.12(a): Satellite Image of Kakrapar Site

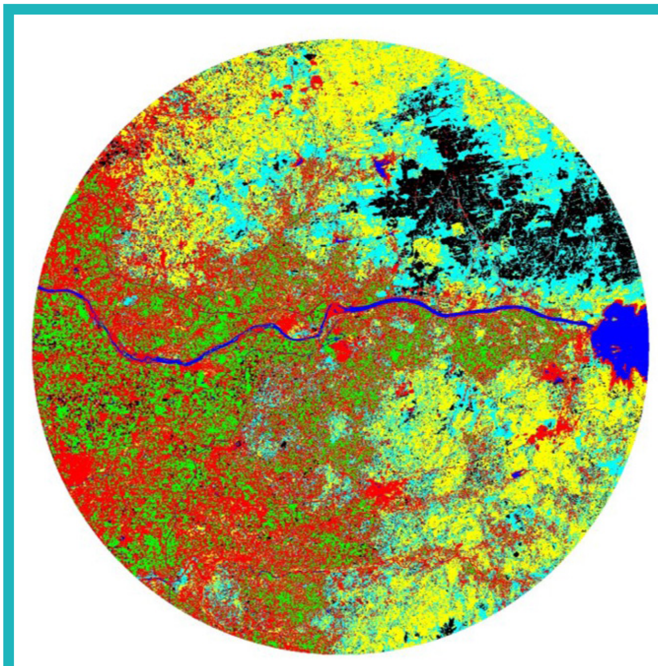


Fig. 7.12(b): Classified Land use/land Cover Map of Kakrapar Site

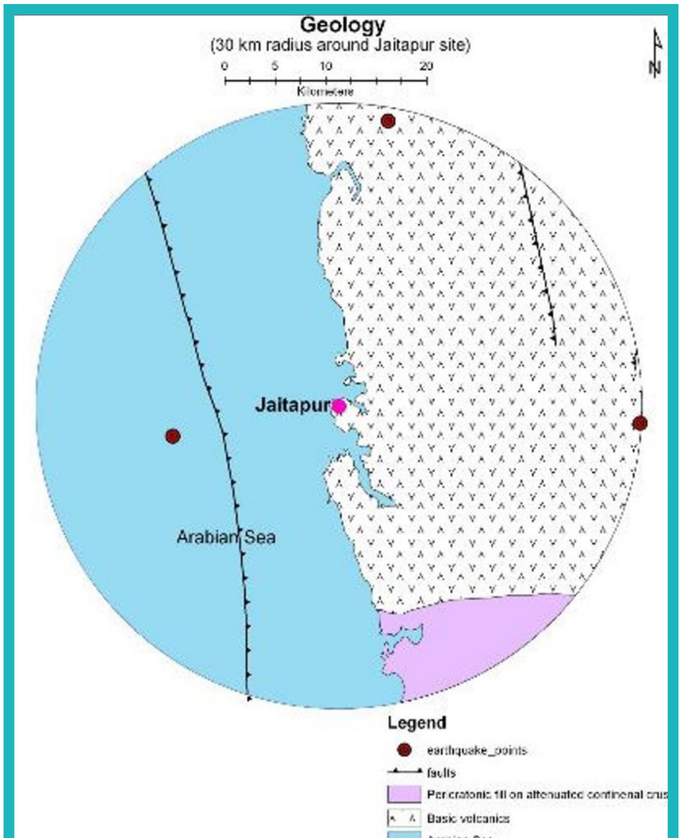


Fig. 7.13(a): Geology Map

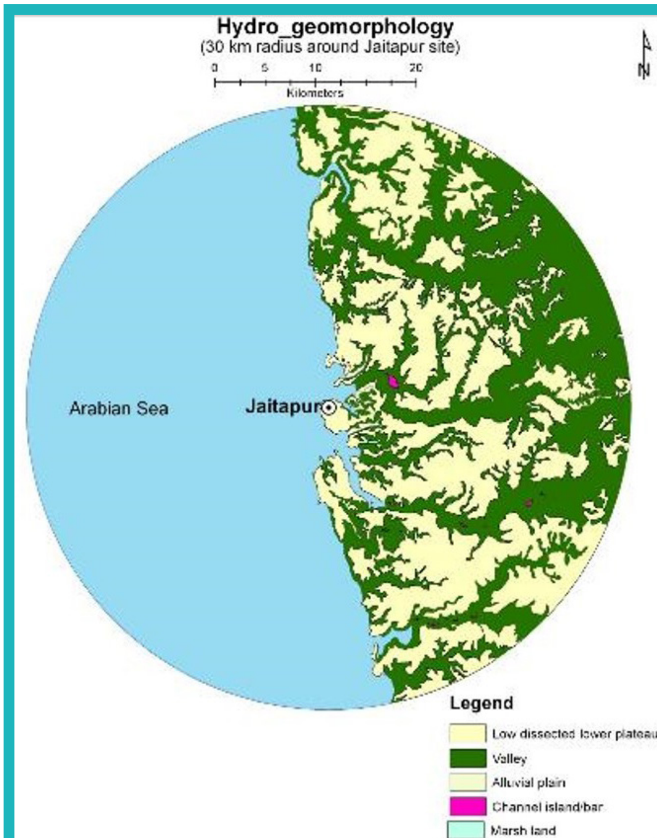


Fig. 7.13(b): Hydro-Geomorphology Map

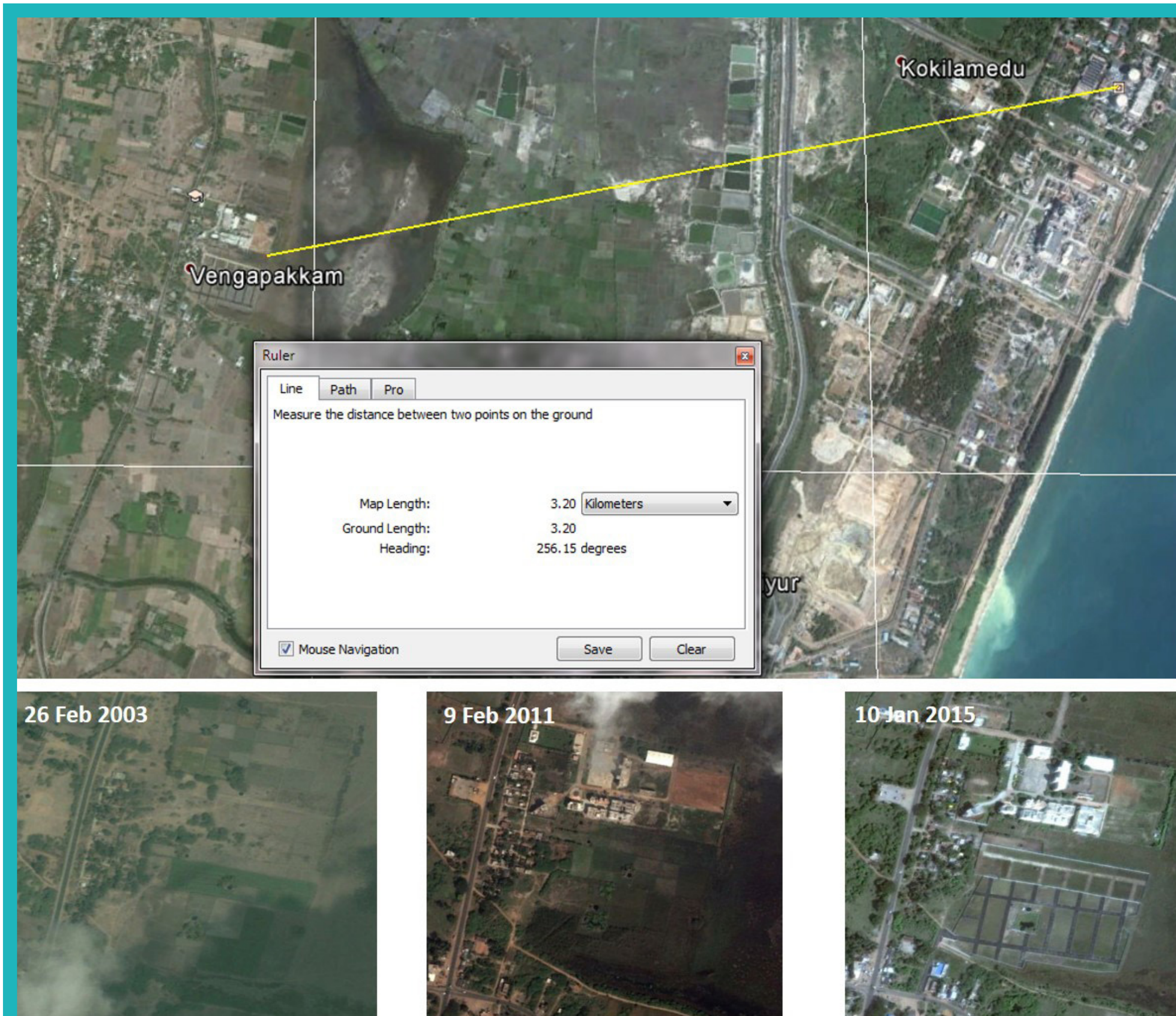


Fig.7.14: Change in the Land use/ Land cover in the No Growth Zone of Kalpakkam Site

Agricultural land converted as build-up area and residential plot near Venkampakkam in the No Growth Zone. (source: Google Earth Pro)

7.5 PROBABILISTIC SAFETY STUDIES

7.5.1 Shutdown and Low Power Probabilistic Safety Analysis (PSA)

The review of Shutdown and Low Power PSA report (Rev-0) of MAPS-1&2 has been completed. The PSA has been carried out for one typical BSD of MAPS-1. The BSD duration is divided into Plant Operating States (POS) with each POS of 2-5 days duration depending on the plant configuration. These POSs are modelled to cover the changing plant configuration due to maintenance. The Core Damage Frequency (CDF) of Shutdown and Low Power PSA has been found to be two orders of magnitude lower than internal events PSA-CDF.

7.6 EXPERIMENTAL STUDIES

7.6.1 Cable Fire Studies in CFTF

Experiments on power and control cable fires were continued within Compartment Fire Test Facility (CFTF). The new set of experiments focused on investigating the effect of hydrocarbon / oil spill fires on power and instrumentation cables laid in horizontal configuration. Diesel pool fires having Heat Release Rate (HRR) of approximately 150-250 kW were used. Both bare as well as cables coated with fire retardant paint, placed in horizontal cable trays were used as target combustibles. The parameters of interest were electrical continuity, sheath and insulation damage, effect of Fire-Retardant Paint (FRP) etc. upon exposure to hot gases and flames in the CFTF enclosure. This activity was carried out in collaboration with NPCIL and salient observations were consolidated in a technical note. A snapshot of cables before and after exposure to fire is shown in Fig. 7.15.

Fig. 7.15: Snapshot of Cables subjected to Diesel Pool Fire in an Enclosure

7.6.2 Fire Hazard Assessment of Electrical Cabinet

Studies on fires in electrical cabinets containing components critical to reactor safety are being carried out in collaboration with IGCAR and BHAVINI to support regulatory safety review. A 3-D model of local electrical control cabinet has been developed as per inputs provided by BHAVINI. Several numerical simulations have been carried out by varying fire parameters. A snapshot of cable fire development within the cabinet is shown in Fig. 7.16.

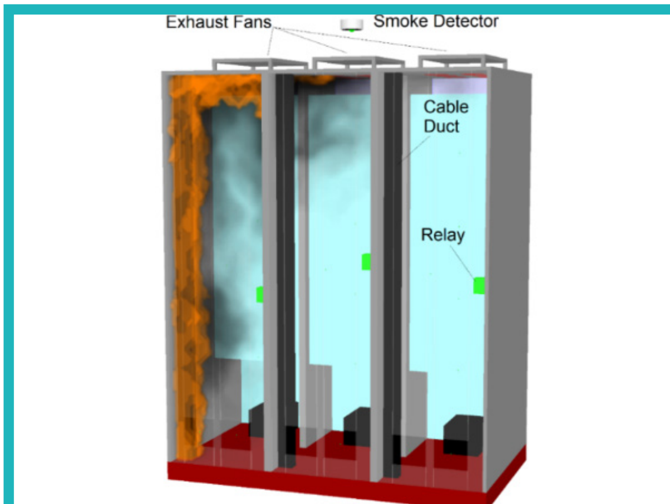


Fig. 7.16: Development of Cable Fire within an Electrical Cabinet

7.6.3 HYdrogen MItigation Facility (HYMIF)

Experimental studies on passive hydrogen removal were continued mainly on (a) hydrogen removal rate and (b) surface temperature of catalyst. Based on earlier experiments in HYMIF, it was observed that coupons with Pt coated on ceramic substrate gives high reaction rate but high catalyst surface temperature. It is found that coupons with Pt deposited on SS mesh results in lower temperature, but the reaction rate per unit area is also low. As only one of the required objectives is met, further experiments were carried out with both types of coupons used in combination as shown in Fig. 7.17(a). It was found that this resulted in synergetic performance as depicted in Fig. 7.17(b). The cordierite plate temperatures are lower and the average reaction rate is higher than when only cordierite plates are used.

7.6.4 AGMS and Coolant Channel Heat-up Facility

An experimental facility has been set-up at SRI, Kalpakkam for investigating coolant channel heat-up and annulus gas monitoring system. Tests were carried out under no annulus gas flow conditions to demonstrate the synchronized functioning of all heaters, heater control modules, thermocouples and data acquisition system. Fig. 7.18 shows the temperature transients in the annular gas region for channel (1, 2), located at middle of bottom row in the 3×3 channel matrix.

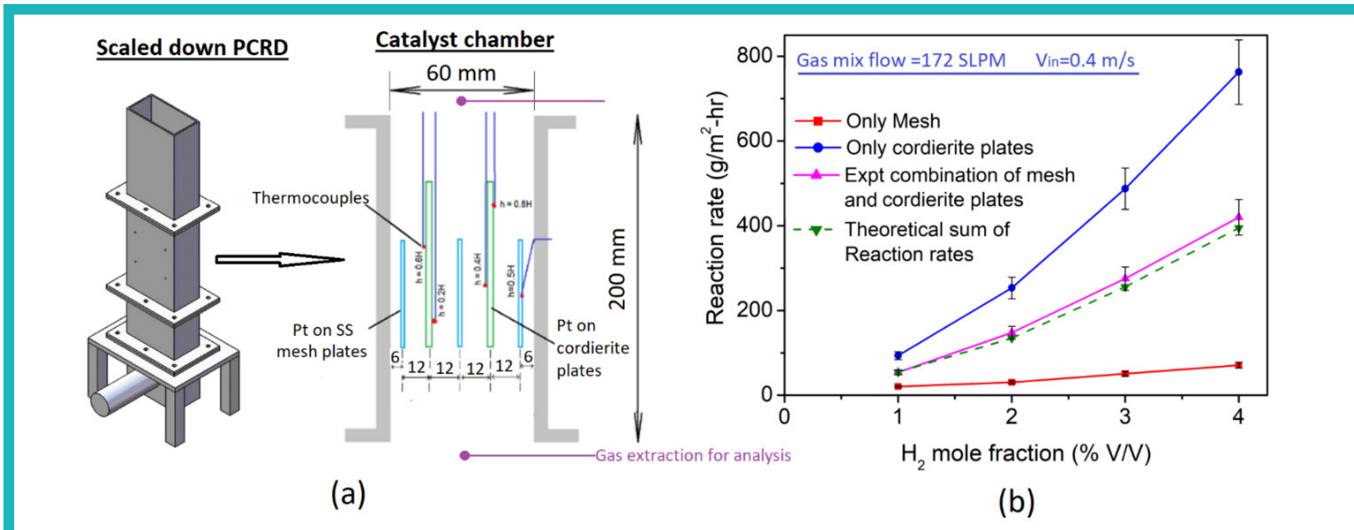


Fig. 7.17: Details of Experimental Studies on Hydrogen Mitigation: (a) Arrangement of Catalyst Coupon within the PAR Chamber, (b) Reaction Rate Data from Experiments

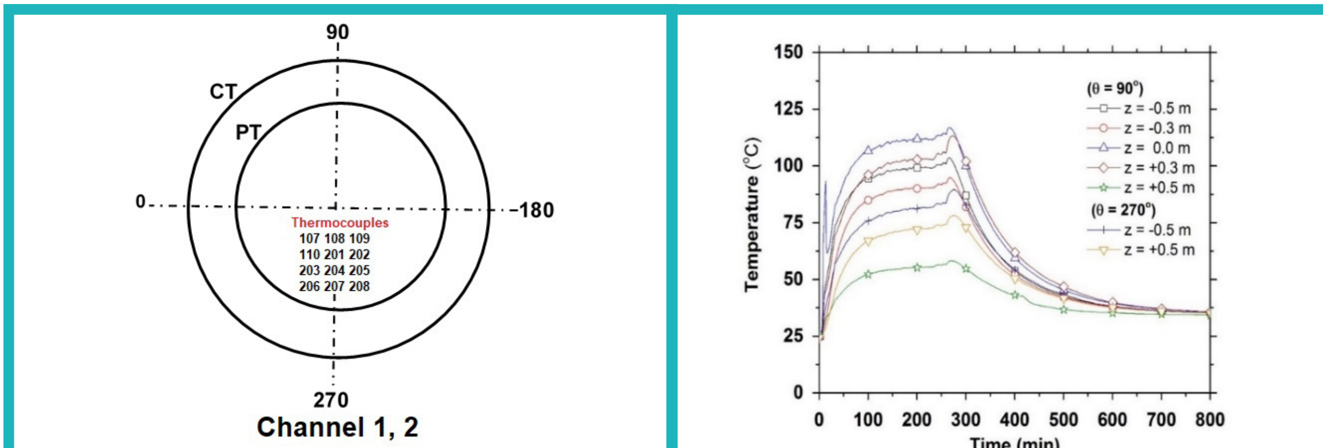


Fig. 7.18: Temperature Transients in the Annular Gas Region; z=0 represents centre of the channel; Negative values of 'z' are towards the panel end

7.6.5 Water and Steam Interaction Facility (WASIF)

An experimental facility has been setup within the high bay of SRI engineering hall in collaboration with BARC to investigate various forms of Direct Contact Condensation (DCC) phenomena. The operating procedure for conducting the experiments has been fine-tuned. Several Condensation Induced Water Hammer (CIWH) experiments for the case when water is injected at various flow rates into stagnant steam within the test pipe have been completed. The steam pressure in the test pipe was varied in the range of 1 to 4 bar. The instrumentation system was able to capture pressure pulses of different magnitude in these cases.

7.6.6 Molten Corium Concrete Interaction (MCCI) Experimental Studies

Effect of MCCI phenomena needs to be investigated for the Indian specific concrete used in NPPs. Experiment was conducted at IIT Bombay in collaboration with BARC and NPCIL with simulant material where the melt was poured in the concrete cavity at ~ 2000 °C.

The concrete ablation phenomenon was observed and studied. Fig. 7.19 (a) and (b) show the cut view of the concrete block after experiment and the variation of temperature. The data so generated will be used for the validation of the computer models.

Fig. 7.19(a): Cut View of the Concrete Block after Experiment

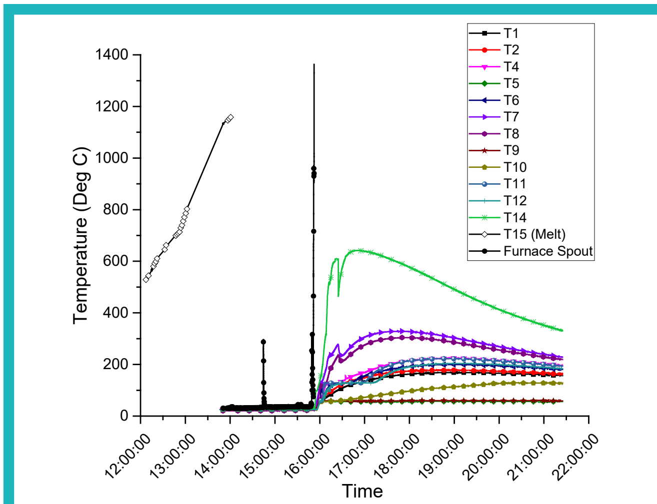


Fig. 7.19(b): Variation of Temperature as Recorded by Different Thermocouples

7.6.7 In-vessel Retention of Corium in Calandria Vessel of PHWRs

Subsequent to commissioning of COre Melt REtention Facility (COMREF), experiments were carried out to study the in-vessel retention capability of calandria vessel in PHWRs. It was observed that the curved geometry like calandria will be able to dissipate the decay heat from corium with high margin if there are no obstructions on its outer surface. It was also observed that sub-cooling of water plays a significant role in the minimum heat flux required for nucleate boiling.

7.6.8 High Temperature Properties of Calandria Material

To evaluate the in-vessel retention capability of the calandria against thermal-mechanical loads generated from corium, the experimental studies were initiated for determination of high temperature tensile and creep properties and damage behaviour of calandria material (SS304L).

Tensile tests were carried out over the temperatures ranging from 300 K to 1123 K for the nominal strain rates of $3\times10^{-3}s^{-1}$, $3\times10^{-4}s^{-1}$ and $3\times10^{-5}s^{-1}$. Influence of temperature on the engineering stress-strain curves is shown in Fig. 7.20 (a) for the strain rate $3\times10^{-5}s^{-1}$. The systematic decrease in stress values with increase in temperature is noticed. As compared to 300 K and 773 K, the strong influence of strain rate on stress values is observed at 973 K in this material (Fig. 7.20 (b)). The variations in tensile properties with temperature exhibited the three distinct temperature regimes i.e. Regime-I (up to 473 K), Regime-II (473-873 K) and Regime-III (>873 K).

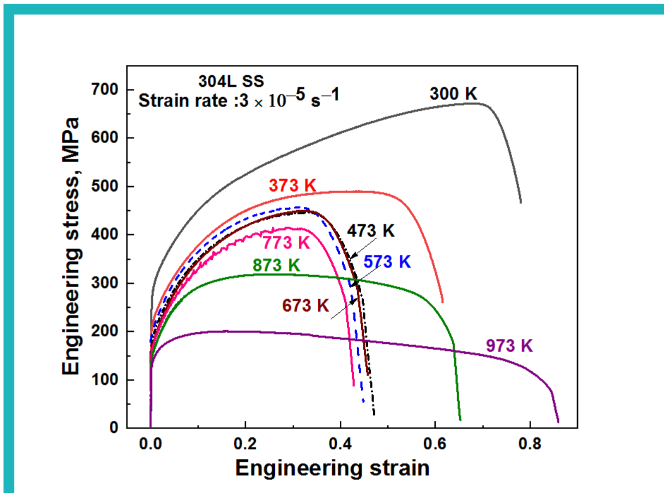


Fig. 7.20(a): Engineering Stress-Strain Curves of SS304L at $3\times10^{-5}s^{-1}$ Strain Rate at different Temperatures

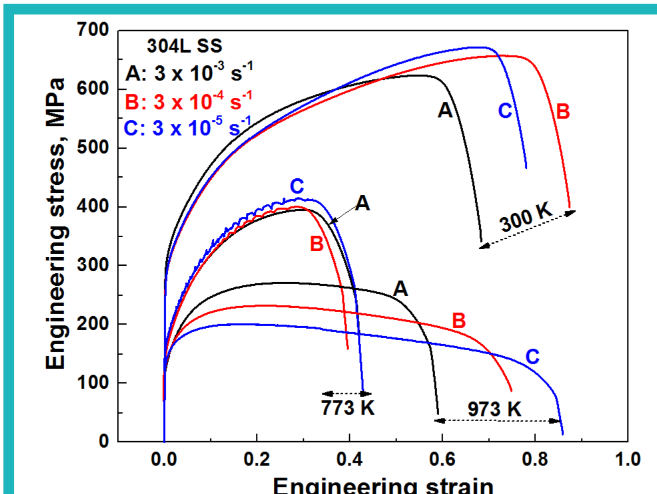


Fig.7.20(b): Effect of Strain Rate on Engineering Stress-Strain Curves of SS304L at Temperatures 300, 773 and 973 K

7.6.9 Experimental Studies of Iodine Behaviour in Reactor Containment

In order to minimize the release of radioactivity to environment in the low probable postulated LOCA initiated severe accident conditions, it is essential to study the fission products behaviour. Iodine is a major contributor to the potential source term to the environment. Understanding its behaviour inside the containment and also in the environment is an essential prerequisite for arriving safety margins.

To study the iodine behaviour in containment, lab scale experimental setup was designed at chemistry laboratory at SRI Kalpakkam. By this experimental setup iodine interaction with paint and iodine adsorption properties on various adsorbents is being studied. The experimental set up was installed and experimental parameters were optimized. Iodine vapours are generated using re-sublimized iodine and vapours were passed through the reaction chamber by using argon as a carrier gas and scrubbed through potassium hydroxide solution.

7.6.10 Experimental Studies Pertaining to removal of Nitrate from Aqueous Stream

Waste water streams containing nitrate are being generated at various stages in nuclear fuel cycle operations. These aqueous streams are neutralized prior to either storage or biological treatment. Among the different treatment techniques, denitrification of waste stream using nano catalysts such as zero valent iron is a promising alternative technique. In-house synthesis, characterization and evaluation of zero-valent iron nanoparticles have been carried out. Nano sized zero-valent iron particles (nZVI) were synthesized by chemical reduction method. Influence of experimental conditions on nitrate removal including pH, catalyst loading, effluent concentration, reaction time etc. was systematically studied (Fig. 7.21). Using the catalyst, the nitrate present in the liquid waste is converted to innocuous products which can be easily disposed into the environment. More than 90 % nitrate removal could be achieved in 60 min using the synthesized nanoparticles.

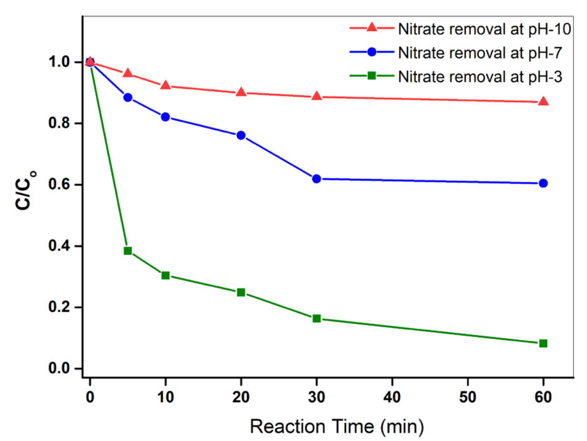


Fig. 7.21: pH Influence on Nitrate removal from Aqueous Streams

7.7 REACTOR PHYSICS STUDIES

7.7.1 Safety Review and Analysis of First Approach to Criticality of KAPP-3&4

The KAPP-3&4 PHWR-700 MWe design utilizes various safety systems and features to meet the requisite safety requirements as brought out in AERB regulatory documents. Calculations were carried out to study First Approach to Criticality (FAC) of KAPP-3 using independent core neutronics code system. The results like variation of effective neutron multiplication factor and reactivity due to draining of ZCCs, withdrawal of various reactivity devices and boron dilution during FAC were calculated and compared with the design calculations as part of independent verification.

7.7.2 Safety Analysis of TAPS BWR Core using In-house Code

As a part of independent verification of safety analysis, reactor physics studies have been initiated using in-house computer code VISWAM. Independent lattice physics calculations are carried out for the reload pattern 2 fuel. The lattice burn-up code has been used and salient results like the variation of the neutron multiplication factors (K_∞ and K_{eff}) with burn-up, different fuel temperature are calculated for the lattice. Further, various reactivity effects due to change in coolant temperature, fuel temperature and void are also calculated. The core calculations have also been performed to find out the multiplication factor, burn-up distribution, power peaking factors and power distribution throughout the Cycle-26 reload. The void distribution, MCHFR and control rod inventory have also been estimated.

7.7.3 Analysis of Reactivity Initiated Transients in a VVER-1000 Reactor

Reactivity Initiated Transient (RIT) analysis due to ejection/withdrawal of Control Protection and System Absorber Rods (CPSARs) in a VVER-1000 reactor was carried out using TRIKIN as part of Indo-Russian RPWG bilateral benchmark. Simulating the above transients at hot zero power and other power operating conditions (25%, 50% and 100% Full Power), the response of core dynamics parameters to the reactivity insertions were studied to demonstrate the ability of the design for terminating this kind of transients.

7.7.4 Analysis of Benchmark on Coupled Neutronics-Thermal Hydraulics Code System

As an intra-DAE Benchmark on the Coupled neutronics and System thermal hydraulics (ABCS), LOCA in a 540 MWe PHWR was identified for inter-code comparison exercise. In the first phase, standalone core neutronics (static) calculations were performed using AERB in-house code REDAC (REactor Dynamics Analysis Code). Parameters like reactivity device worths in different configurations and reactivity coefficients for changes in fuel and coolant states were determined. The estimations were found to be in good agreement with other codes.

The transient phase of the exercise was analysed to evaluate the prediction capability of TRIVAC module of DRAGON code system. The problem involves localized perturbation in terms of defined changes in coolant density in one half of the core leading to power rise which will be arrested through reactor SCRAM. Results like SCRAM worth vs. time curve, time of occurrence of trip signals due to period (less than 10 s) and power (110% of FP), reactor power and core reactivity vs. time, and 3D power distributions were calculated. The results of TRIVAC were found to show the expected trend.

7.7.5 OECD-NEA THAI-3 Project and Analysis of THAI Hydrogen Deflagration Tests

As a part of the international collaborative OECD-NEA-CSNI THAI-3 project, analysis of the database from a number of hydrogen deflagration experiments performed in the THAI facility was carried out within a theoretical framework. The peak pressures and temperatures obtained during the experiments were compared with the theoretical AICC (Adiabatic Isochoric Complete Combustion) estimates. As an illustration, Fig. 7.22 shows the variation of calculated pressure ratio (burnt gas pressure / initial pressure) for different initial hydrogen and steam mole fractions. The experimental trends were consistent with the theoretical estimates and brought out the influence of important parameters like heat losses, combustion completeness etc. Dynamic combustion behaviour was analysed within the framework of calculated laminar burning velocities and experimentally measured flame speeds. The influence of direction of propagation, initial temperature and non-uniformity on flame propagation was investigated. This analysis provided useful insights into the assessment of the static and dynamic effects of slow deflagrations and development of a methodology for modelling of slow deflagrations in hydrogen-air-steam mixtures.

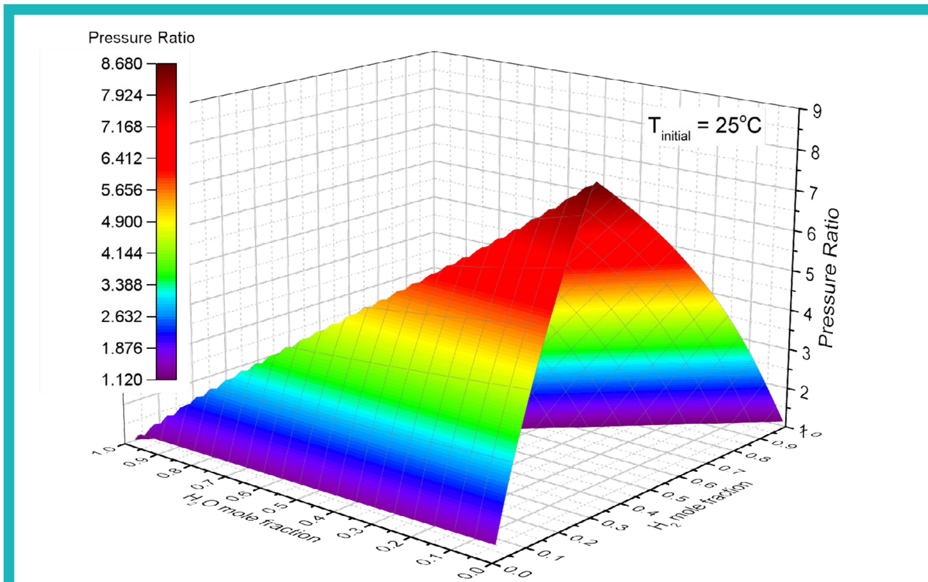


Fig. 7.22: Calculated (AICC) Pressure Ratio for different initial Mixture Compositions

7.7.6 Development of In-house Code for Transient Analysis of Sodium Cooled Fast Reactor (SCFR)

An in-house computer code has been developed for analysing flow and power transients in fast breeder reactors. Capability of the code has been improved by incorporating slug-expulsion model for analysing void progression and flow inside the core with significant voiding. Comparison of pre-disassembly phase of an un-protected transient over-power accident (UTOPA) scenario was carried out with FBR accident analysis code, SAS-1A. Pressures, temperature, void fraction, vapour quality, saturation temperature of coolant and temperature distributions of fuel, clad and structures were estimated at different times and compared with SAS-1A results.

7.7.7 Independent Verification Analysis of PHWR-700 Stability

As a part of independent verification, stability analysis of the total power control loop of 700 MWe PHWR was carried out. Analysis was carried out in discrete-time domain by linearizing the system around its equilibrium points and identifying Eigen values of the closed-loop system. The dynamics of the system vary widely depending on operating power levels, core–fuelling states and cycle time of reactor regulating system. The input to the stability analysis are dynamics governing system parameters and the output is a stability characterization in terms of gain of total power control loop at any power level. The results independently verified the stability analysis and respective gain values of total power control loop.

7.7.8 ATF Coating Material for PHWR based on Neutronics Evaluation

Coating around Zircalloy based cladding material enhances accident tolerance capability due to its various favourable properties such as high melting point, low oxidation rate, low hydrogen generation and remain stable even in nuclear accident scenarios. However loading of coating material in the reactor will impact on neutronic properties. Based on the exhaustive literature survey, in total 24 materials have been identified and considered for their reactivity load assessment for PHWR-700 reactor. The reactivity load for both side coating thickness of 10 μm, 20 μm, 30 μm and 40 μm are used for estimating the reactivity load for potential 24 number of Zircaloy based ATF clad coating materials for PHWR700. Based on the analysis, the coating materials are classified into three categories with respect to reactivity loads: best suitable (with small reactivity load 0 to -5 mk), may be acceptable (with medium reactivity load -5 mk to -10 mk) and not suitable (with large reactivity load more than -10 mk). Full power days penalty due to the ATF material coating are also assessed.

7.7.9 Effect of Americium Build-up on Core Physics Parameters of Fast Reactor

In order to evaluate effect of Americium build up on core neutronics parameters, ^{241}Am build-up due to decay of ^{241}Pu isotope in fresh fuel subassemblies has been estimated for residence periods of two and five years. By using the evaluated number densities of the isotopes after considering the residence period, neutronics calculations have been performed to evaluate core excess reactivity in case of PFBR. It is found that the delay of fuel loading in core, could lead to considerable reduction in core excess reactivity due to decay of Pu isotope.

7.7.10 Investigation of Attenuation Characteristics of Advanced Shield Materials

Investigation of neutron attenuation characteristics of advanced shielding materials like Portland cement mixed with 10wt% Samarium Oxide (Sm_2O_3) and ordinary concrete admixture with 10wt% Borated Polyethylene was carried out for typical Light Water Reactor (LWR) neutron spectrum. Neutron spectra simulated with and without shielding materials of interests given in Fig. 7.23 clearly demonstrate the effect of each shielding material on neutron spectrum.

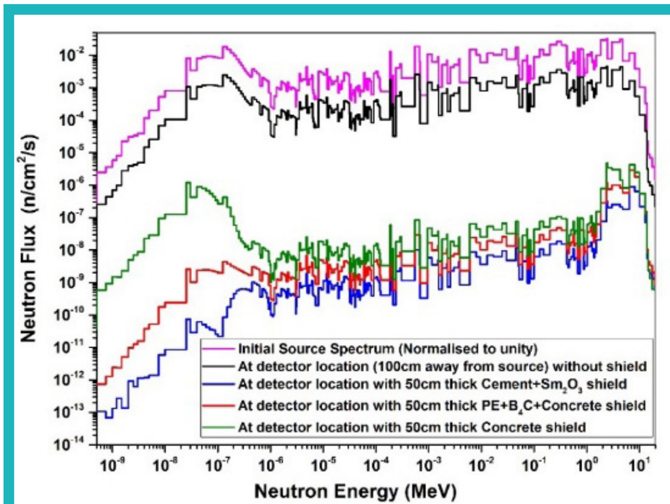


Fig. 7.23: Comparison of Neutron Dose Rate Attenuation Behaviour

The shielding efficacy of both the advanced materials of interest are observed to be better than ordinary concrete for LWR spectrum. Samarium, which has a higher thermal neutron absorption cross-section of 40800 barns for ^{149}Sm, shows predominant attenuation behaviour compared to Boron with thermal neutron absorption cross-section of 3840 barns for ^{10}B. The difference in weight fractions of Samarium and Boron in respective shield materials also partly contribute to the effect.

7.7.11 Thermal Neutron Dosimetry using ^{6}LiF:Mg,Ti and ^{7}LiF:Mg,Ti based Thermoluminescence Detectors

Owing to the significantly different cross sections of ^{6}Li and ^{7}Li for interaction with thermal neutrons, a combined use of ^{6}Li and ^{7}Li enriched thermoluminescence (TL) detectors gives clear estimation of gamma and neutron doses in the mixed field of gamma and neutron. Based on this fact, suitability of ^{6}LiF:Mg,Ti (MTS-6: 95.62% ^{6}Li and 4.38% ^{7}Li) and ^{7}LiF:Mg,Ti (MTS-7: 99.993% ^{7}LiF and 0.007% ^{6}LiF) thermoluminescence detectors was studied for thermal neutron dose mapping around paraffin wax moderated 5Ci ^{241}Am-Be neutron source. The calibration source and facilities (Thermal neutron flux standard facility and gamma standard source facility) of IGCAR, Kalpakkam were used for irradiation purpose. Unlike the traditional methods which make use of calibration curve obtained with gamma field to provide information on gamma equivalent neutron dose, the calibration curve obtained with the gamma and neutron fields has been used (for MTS-6) in the present study to give clear estimation of neutron dose. The thermal neutron dose values measured using MTS-6 and MTS-7 TL detectors were found to show the expected trends and good agreement with the theoretical Monte Carlo simulations.

7.8 STRUCTURAL ANALYSIS AND MATERIAL STUDIES

7.8.1 Studies for Mechanical Properties Evaluation of Zr-2.5%Nb PT Specimen

Zr-2.5%Nb pressure tube material is orthotropic in nature because of its crystal structure and the mechanical processing it undergoes during the manufacturing. Hence, for investigating realistic failure simulation the consideration of realistic anisotropic material behaviour is very important. The numerical studies conducted to study the effect of Zr-2.5% Nb anisotropy on notch stress triaxiality and stress intensity factor in a tension test specimen showed that stress triaxiality for anisotropic model is higher than the isotropic material (Fig. 7.24 (a) and (b)). This indicates early prediction of crack initiation with anisotropic material model compared to isotropic material model. Another numerical study conducted on the development of residual stress in a cylinder made from Zr-2.5%Nb observed that during the auto-frettage, the residual stress obtained for the two-materials models were different (Fig. 7.24(c) and (d)).

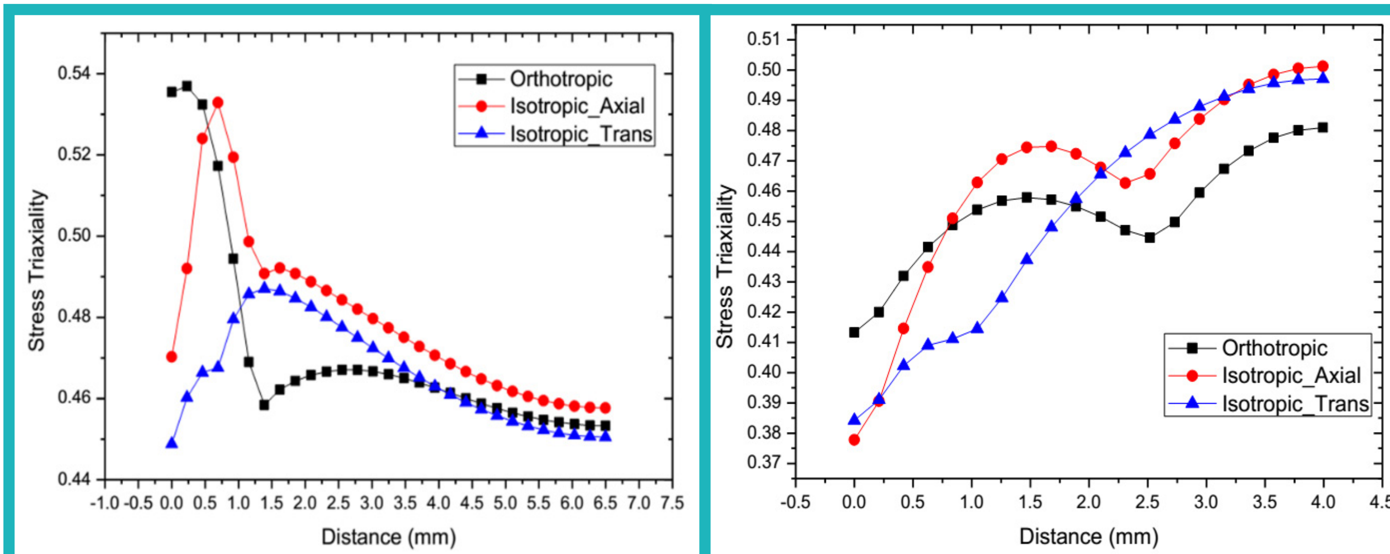


Fig. 7.24(a) &(b) : Plot of Stress Triaxiality along remaining Ligament for (a) 2.5 mm Notch Radius and (b) 5 mm Notch Radius

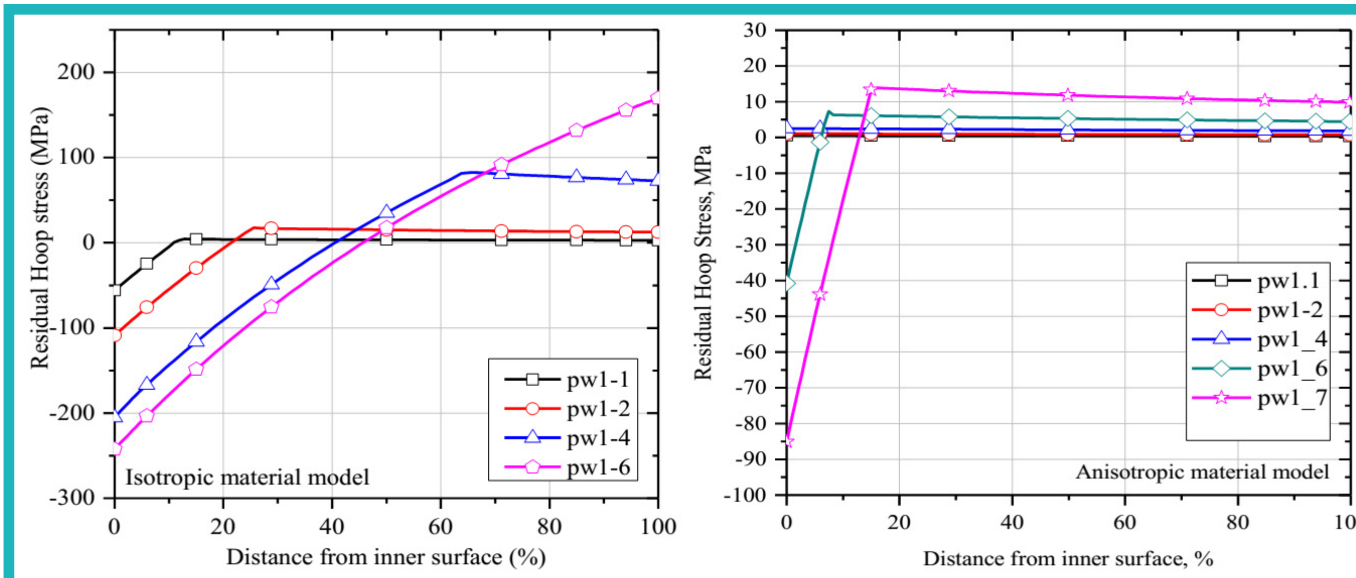


Fig. 7.24 (c)&(d) : Residual Hoop Stress across the Thickness for (c) Isotropic Material Model and (d) Anisotropic Material Model for different internal Pressure

Therefore, an experimental program to evaluate the parameters for material model which accounts for anisotropy and cyclic behaviour for the Indian Pressure Tube specimen is initiated. The gap areas are identified through in-depth study of available published data. This program is in collaboration with NML, Jamshedpur.

7.8.2 Evaluation of Fracture Mechanics Parameters using XFEM

A benchmark was proposed by OECD/NEA to enable a comparison of XFEM capability of different codes used in the nuclear industry. AERB participated in the benchmark. The principal objectives of the project were to compare Stress Intensity Factor (SIF) obtained by classical FE method and analytical formulae to SIF obtained by X-FEM.

Simulation of benchmark problems for fracture mechanics parameters K_I, K_{II}, K_{III} for different type of loadings (mechanical, thermal) was carried out using XFEM tool of ABAQUS.

The FE model of semi elliptical underclad crack in the shell of the reactor pressure vessel is shown in (Fig. 7.25(a)). Sequential heat transfer and structural analyses were performed to evaluate SIF values for thermal loading which corresponds to LOCA conditions in reactor pressure vessel. Fracture mechanics parameters like SIF, J-integral etc. were evaluated and these were compared with analytical solutions (RSE-M code) as shown in (Fig. 7.25(b)). Overall capabilities and advantages of XFEM (ABAQUS) for fitness assessment of NPP components were brought out.

7.9 SAFETY STUDIES TO SUPPORT REVIEW AND ASSESSMENT

7.9.1 Benchmarking SSI Analysis of Lotung Quarter Scaled Containment Model

Soil-Structure Interaction (SSI) is an important aspect of structural engineering, especially for massive constructions, such as nuclear power plants, concrete and earth dams, on soft soils. To ensure safety of NPP structures of deep soil sites, detailed analysis to capture the SSI phenomena realistically is needed. To have a detailed understanding of the analytical approach for such analysis, a Standard Problem Exercise (SPE) on SSI was conceived with USNRC, using the seismic response data from SSI experiments conducted on the scaled containment model at Lotung, Taiwan. As the first task of the SPE, benchmarking SSI analysis of the Lotung quarter scale containment model was carried out in ACS SASSI, considering one case of horizontal East-West shaking corresponding to the earthquake event recorded on May 20, 1986. The response spectra at top and bottom of the containment, top and bottom of the steam generator has been compared with the recorded responses (Fig. 7.26). It was observed that responses from the SSI analysis are generally in good agreement with the recorded motions.

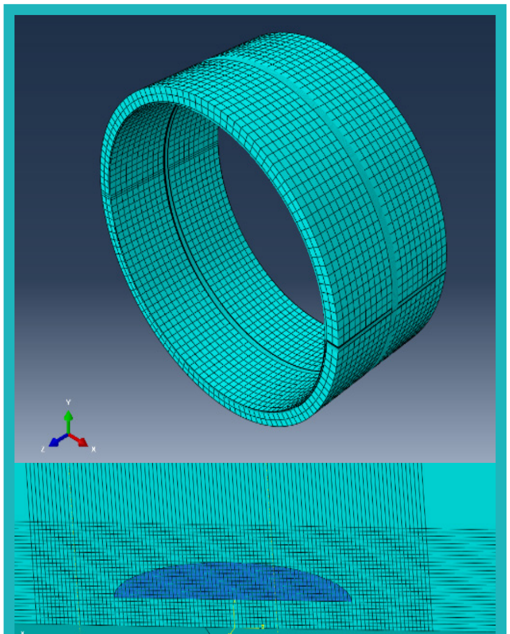


Fig. 7.25(a): FE model of Benchmark problem with Surface Flaw

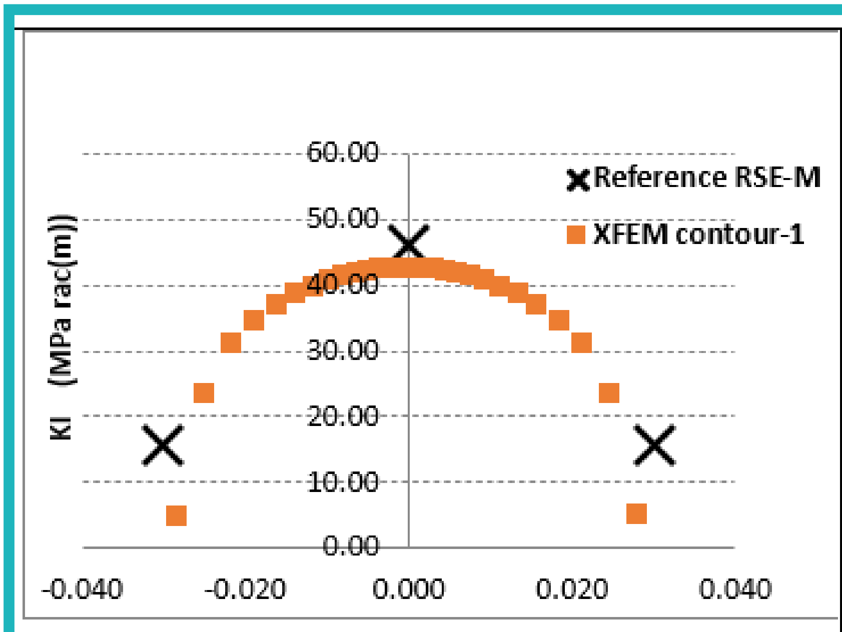


Fig. 7.25(b): Evolution of KI along the Crack Front (in Clad) for XFEM

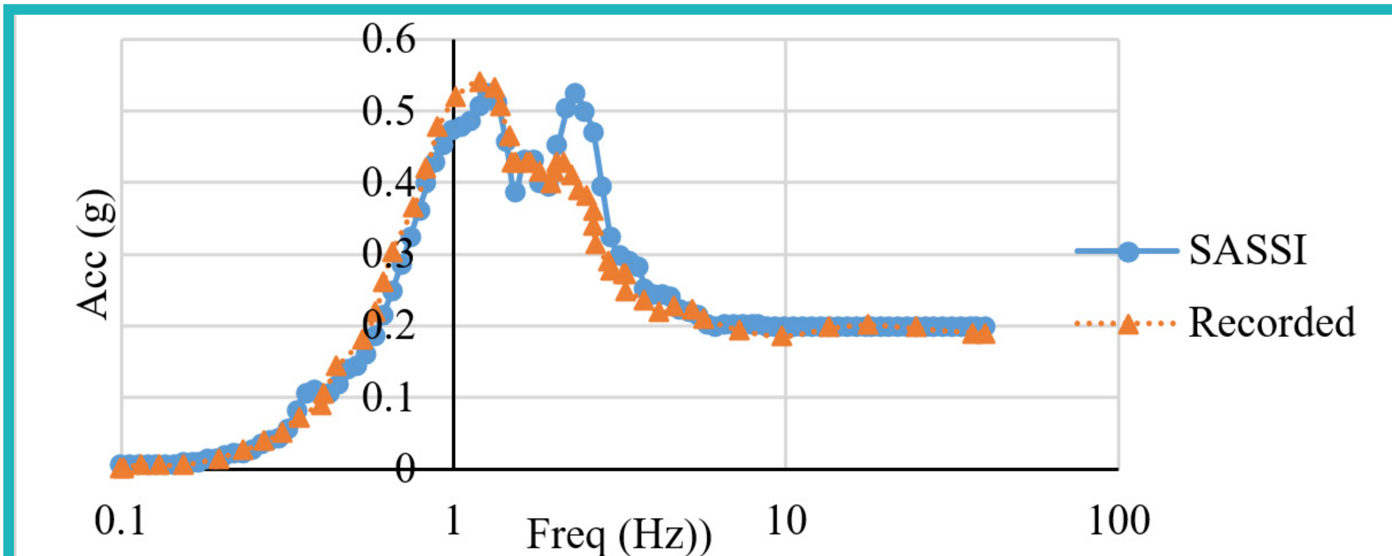
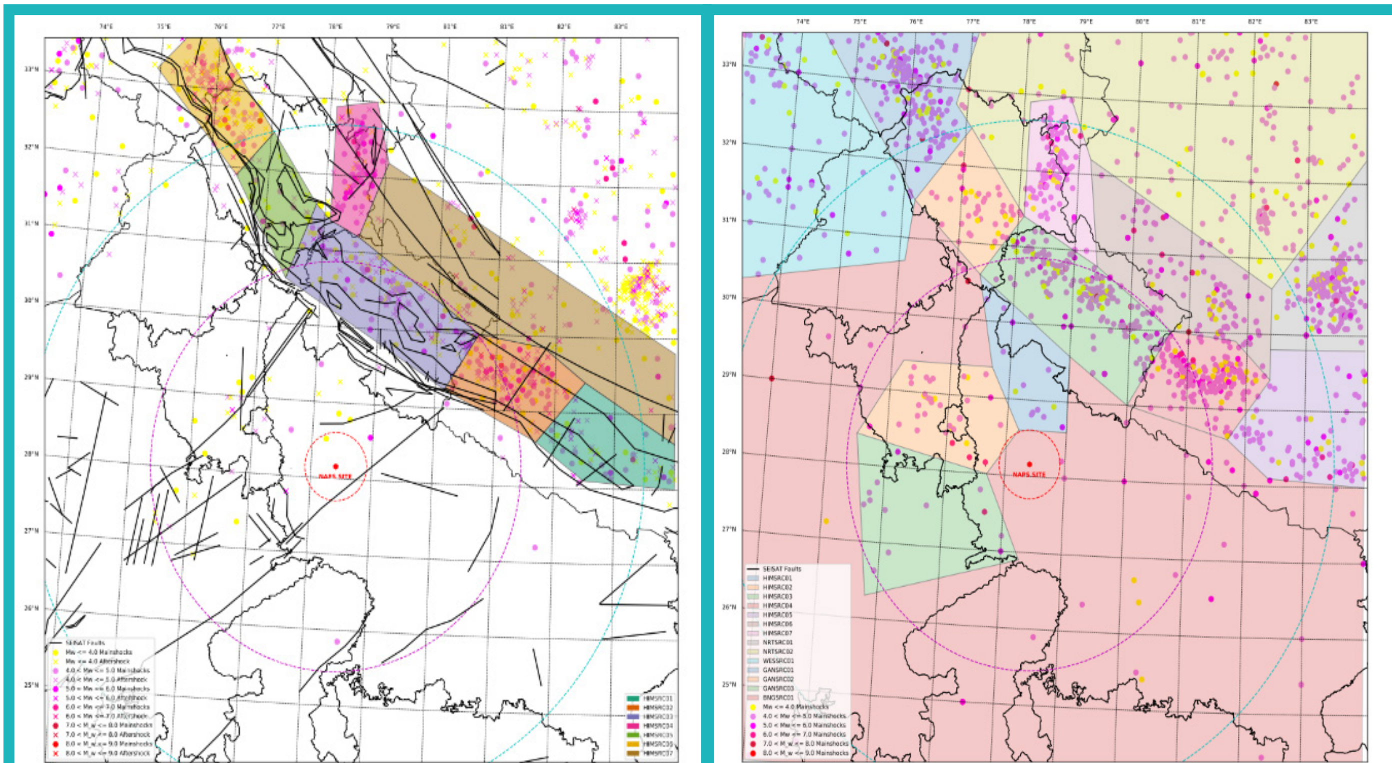


Fig. 7.26: Comparison of Responses obtained from Analysis with Recorded Responses

7.9.2 Probabilistic Seismic Hazard Analysis of NAPS Site

NAPS is situated in the Bulandshahar District of Uttar Pradesh. As part of Post-Fukushima action, NPCIL has carried out re-evaluation of seismic hazard at site using Deterministic Seismic Hazard Analysis (DSHA) methodology. AERB initiated probabilistic seismic hazard analysis of NAPS site to verify the correctness and identify key issues in assessment. Two source models namely NDMA and USGS were developed from respective earthquake catalogue for PSHA study. These source models are shown in Figure 7.27. The study results will be used in the safety review of seismic re-evaluation of NAPS.



Seven identified sources for NAPS site from USGS Catalogue

Fifteen identified sources for NAPS site from NDMA Catalogue

Fig. 7.27: Source Models for NAPS PSHA Study

7.9.3 Dynamic Soil-Structure Interaction Analysis of Control Building of GHAVP-1&2

As part of independent verification to provide inputs for design safety review of civil engineering aspects of GHAVP, dynamic SSI analysis of control building was carried out by AERB in specialised SSI software, SASSI. SSI analysis is conducted in frequency domain and includes analytical half-space model for layered soil which captures soil wave propagation effects in all frequency ranges. The analytical half-space model coupled with structural FEM provides a unique frequency dependent SSI analysis capability unlike traditional FEM softwares. Typical contour of induced internal force in control building for typical soil characteristics is shown in Fig. 7.28.

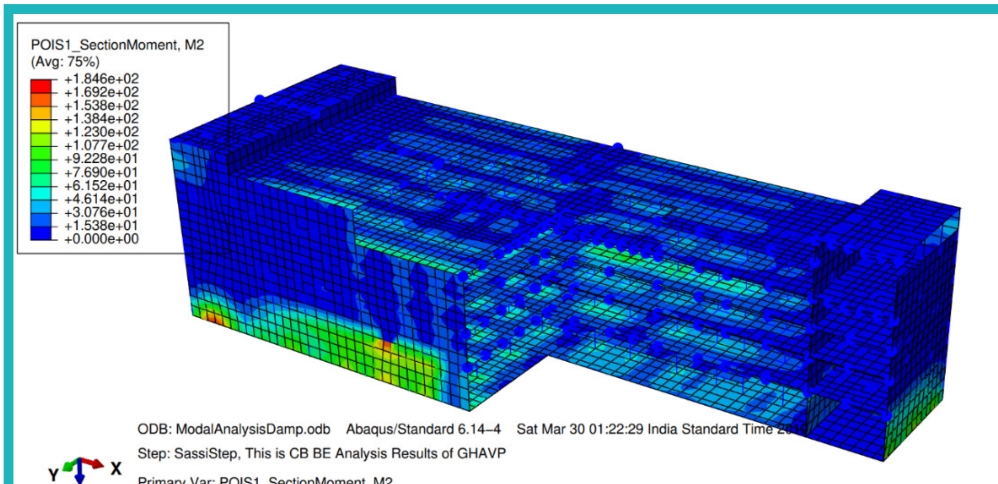


Fig. 7.28: Typical Response Contour Generated from SASSI Output (Cut view)

7.9.4 Numerical Analysis of Multi-level Excavation in Alluvial Soil and Impact of Multiple Buildings on Settlement: A Case Study of GHAVP Site

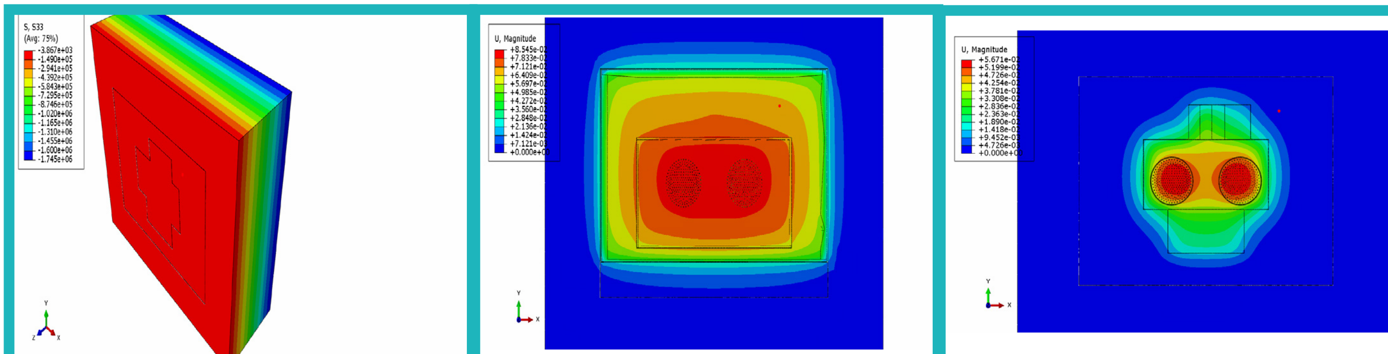
Excavation at NPP sites involves large area with multiple levels of excavated bed for founding structures at different depths. Due to removal of overburden during excavation, stress is relieved from the soil below the foundation. Consequently, to the extent that the soil behaves elastically, the soil beneath the excavation and adjacent ground tend to move upward called 'Heave'. A study of this phenomena considering the excavation for GHAVP, Haryana site is carried out.

In the analysis, a detailed FE model of the soil (~750mX650mX200m), excavation, and ground improvement (GI) layer is developed using site specific geotechnical properties. A sequential analysis involving simulation of Geostatic stress state, removal of excavated soil up to the excavation depth and soil-cement re-filling (using sub-merged density of material) up to the founding depth is undertaken. Initial geostatic state of stress, heave of the excavated bed, and reduction of the heave on account of re-filling with soil-cement mix is shown in Fig. 7.29. Detailed SSI analysis of safety related structures also taking into account the residual effect of excavation phenomena is under progress.

7.10 AERB FUNDED SAFETY RESEARCH PROGRAMME

AERB promotes and funds research in radiation safety and industrial safety. AERB Committee for Safety Research Programmes (CSRP) frames guidelines for the same and also evaluates, recommends grants for research projects and monitor their progress periodically. During this period, CSRP recommended four new projects. It also approved the renewal of ten ongoing projects. The details are given in Tables 7.1 and 7.2.

AERB also provides financial assistance to Universities, Research Institutions and Professional Associations for holding symposia and conferences on the subjects of interest to AERB. During this period, financial assistance was provided to 34 Seminars, Symposia and Conferences.



Geostatic State of Stress before Excavation (Isometric view)

Heave of the Excavated Bed (Plan)

Residual Heave after GI Filling (Plan)

Fig. 7.29: Results of Multilevel Excavation Analysis

Table 7.1: New Research Projects Approved

S. No.	Project Title	Principal Investigator	Organisation
1	Experimental and Numerical Evaluation of Double Containment Structures of Indian PHWR against Hard Missile Impact due to External Event	Dr. Mohd. Ashraf Iqbal	IIT Roorkee
2	Molten Corium Concrete Interaction Studies	Dr. Arunkumar Sridharan	IIT Bombay, Mumbai
3	Assessment of Liquefaction Potential through Analytical Methods	Dr. S. D. Anitha Kumari	M. S. Ramaiah University of Applied Sciences, Bangalore
4	Performance Evaluation of Generic Compounds with Effective Functional Group as Corrosion Inhibitors for RC Structures	Dr. Shweta Goyal	Thapar Institute of Engineering & Technology, Patiala

Table -7.2: Research Projects Renewed

S. No.	Project Title	Principal Investigator	Organisation
1	Synthesis, Characterisation of N_2 Donor Extractants for Separation of Minor Actinides from Lanthanides	Prof. N. S. Karthikeyan	Easwari Engg. College, Chennai
2	Patient specific dose calculation using Cone Beam CT with local and Global HU correction strategies and Deformable Image Registration based Adaptive Radiotherapy	Dr. B. Paul Ravindran	CMC, Vellore
3	Synthesis of Chitosan based Ploy Electrolyte Ultrafiltration Membrane for the Remediation of Cesium from Aqueous Media	Dr. M. Dharmendirakumar	Anna University, Chennai
4	Study of Fundamental Heat Transfer Characteristics in the presence of Non-condensable for designing Long Term Passive Heat Removal system for Containment	Dr. Arunkumar Sridharan	IIT Bombay, Mumbai
5	Studies on levels of Natural Radiation in the Environment of hill Districts of Manipur	Dr. S. Nabadwip Singh	Oriental College, Takyel, Imphal
6	Improving Radiation Safety Standards in Dental Practice	Dr. A. Saravana Kumar	PSG Institute of Medical Sciences Research and Hospitals, Coimbatore
7	Studies on Environmental Radioactivity Levels in and around Chitrial Uranium Mineralized areas of Nalgonda District, Telangana State	Dr. Ch. Gopal Reddy	Osmania University, Hyderabad
8	Low Pressure Nanofiltration for removal of Monovalent and Bivalent Salts from Leached Liquor during Alkaline Uranium Ore Processing	Dr. Sirshendu De	IIT, Kharagpur
9	Numerical Crack Growth Studies in Hydrided Pressure Tube of PHWR	Dr. Indra Vir Singh	IIT Roorkee
10	Determination of Anisotropic Elastic Constants and Anisotropic Yield Parameters for Zr-2.5% Nb Pressure Tubes	Dr. Avijit Kumar Metya	CSIR-NML, Jamshedpur