CHAPTER

7

SAFETY ANALYSIS AND RESEARCH

SAFETY ANALYSIS AND RESEARCH 7

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 REACTOR THERMAL HYDRAULICS SAFETY STUDIES

7.1.1. In-Core LOCA Analysis for the 540 MWe PHWRs

The accident management guideline (AMG) entry criteria in PHWRs for in-core Loss of Coolant Accident (LOCA) is based on "Sub-cooling margin in both Reactor Inlet Headers (RIH) or both Reactor Outlet Headers (ROH) of the affected loop remaining greater than a few degree centigrade for more than the specified time period" and "change in the area accident radiation monitors' readings". In the regulatory review of the AMG technical basis document, it was identified that the AMG entry criteria signal of radiation level change may get delayed during the in-core LOCA, due to the presence of the moderator in the calandria and the release via over pressure rupture disc (OPRD) line being the only release path available. An independent verification analysis for the 540 MWe PHWR was carried out to assess the effectiveness of the AMG entry criteria for the in-core LOCA scenario.

In the analysis, the Primary Heat Transport (PHT) pressure, clad temperature and moderator level were estimated. It was observed that at the time of clad failure, the PHT pressure was sufficient enough for steam along with the noble gases to flow through the feeders and bubble through the moderator. It is concluded that the AMG entry criteria is appropriate.

The variation in moderator inventory, PHT pressure and clad temperature of the affected loop is shown in the Fig. 7.1 (a), (b) and (c) respectively.

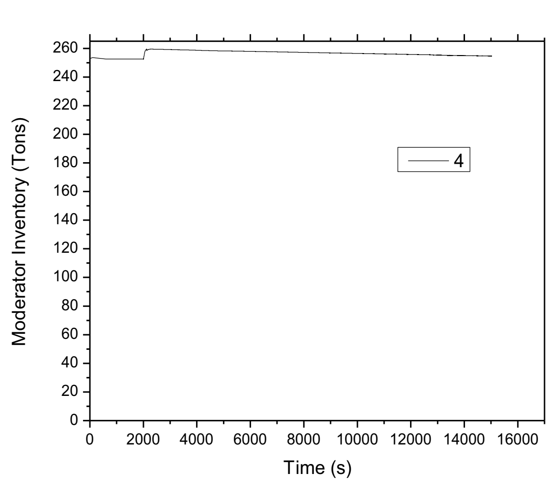


Fig. 7.1(a): Moderator Inventory

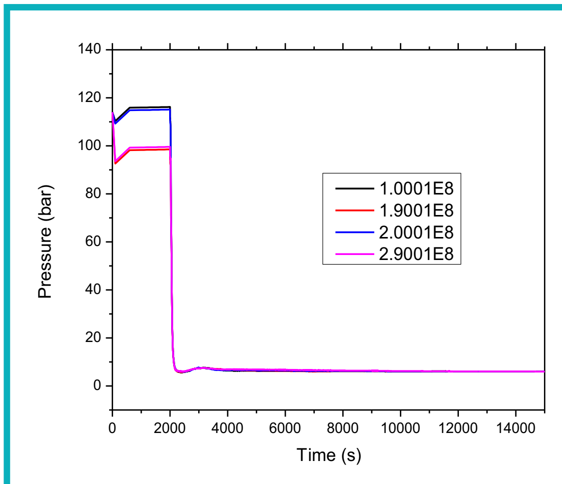


Fig. 7.1(b): PHT Pressure

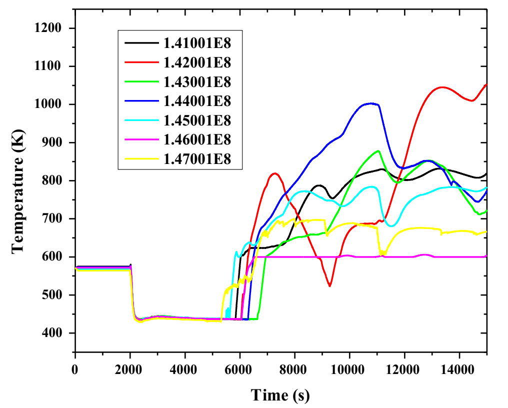


Fig. 7.1(c): Affected pass clad Temperature

7.1.2 Development of Monte Carlo Model for View Factor Calculation of the IPHWR Fuel Bundles

Fuel failure criteria and phenomenon like hydrogen generation and release of fission products are strong functions of the fuel temperature during the severe accident progression. In order to obtain realistic estimates of the fuel temperatures for safety assessment, the radiation heat transfer need to be modelled accurately with sufficient details. Radiation heat emitted by a body is proportional to fourth power of the temperature, emissivity, surface area, etc. However, when two bodies are involved, radiation heat transfer between any two bodies is also directly proportional to the view factor. The fraction of heat emitted by one body that is intercepted by other body needs to be calculated precisely. Among the various methods available for computation of view factor, Monte-Carlo method has been adopted to estimate view factors for Indian PHWR (IPHWR) fuel bundles because of its accuracy and flexibility. The method involves estimation of fraction of rays emitted by pin that are intercepted by the other pins and the Pressure Tube (PT). As number of ray's increase, this factor approaches view factor (see Fig.7.2).

The developed code was validated against analytical solution of view factor for infinitely long parallel cylinder and effect of number of rays on view factor was also studied. With 1E+06 rays; view factor could be estimated with reasonable accuracy i.e. error $\leq$ 0.1629 %.

The code was further validated with cases available in the open literature. An array of tubes arranged in square matrix was selected from the literature for the validation. The results obtained from this code are in close agreement with the results given in the literature. Radiation view factor between each pin with every other pins and PT of 220 MWe and 540/700 MWe IPHWR fuel bundle were estimated with good accuracy to avoid over/under estimation of radiation heat transfer. Further for use in safety analysis code, fuel-pin-ring wise view factors of each pin were also estimated.

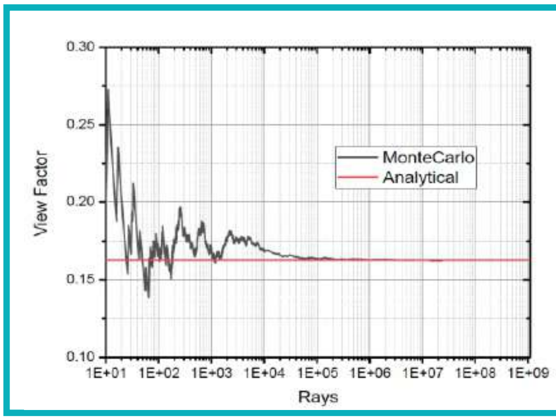


Fig. 7.2: Effect of number of rays on Computed View Factor between two Parallel Pins

7.1.3. ISOMED Source Storage Cooling Analysis: Effect of the Air Gap

ISOMED (Irradiation Sterilization of Medical Products), the first irradiation plant for sterilization of medical products, was set up in India by the Department of Atomic Energy (DAE) 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 circulation through cooling coils embedded in the concrete shild. Towards estimating the cooling capability of the embedded coil, a 3-D model of the concrete shield along with the steel liner was developed using multi-physics software COMSOL and the cooling of the concrete through this coil was also assessed. While reviewing the cooling analysis results, it was suggested that the effect of air gap between the coil & concrete, and steel liner and concrete also needs to be studied. Further, the 3D model was modified to include the air gap at various locations. It was observed that even if there is air gap in the last leg of the cooling coil (Fig. 7.3 (a) and (b)), the peak concrete temperature estimated was within the acceptable limits.

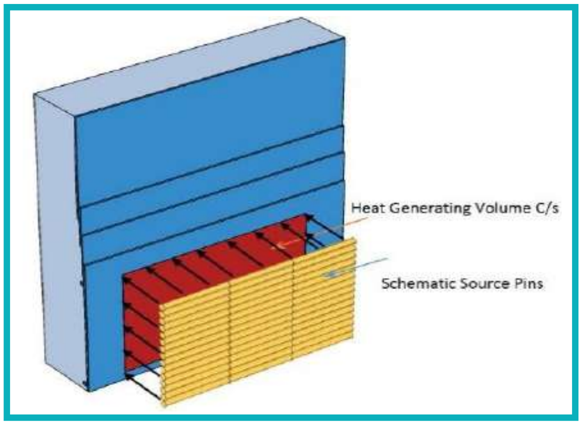


Fig.7.3 (a): Heat Generating Volume and Schematic of Source Pins

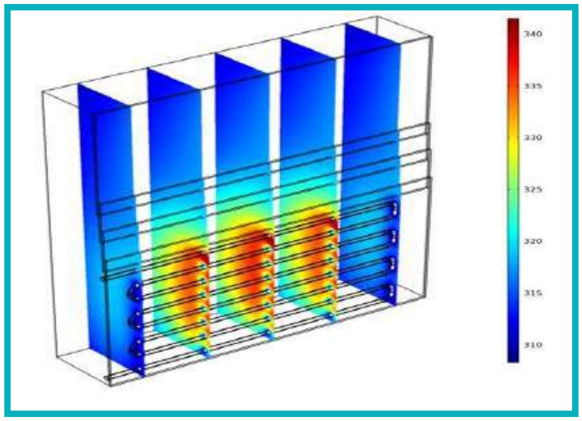


Fig. 7.3 (b): Temperature Field with Air Gap of 5mm at the Coil & Concrete Interface in last leg (NDC)

7.2 SEVERE ACCIDENT STUDIES

7.2.1 CFD Analysis for Estimation of Calandria temperature under Severe Accident Conditions

During postulated severe accident scenario, the radiation heat transfer plays a very important role. It is to be noted that radiation heat transfer cannot be modelled effectively in system thermal hydraulics codes like RELAP5 used to simulate the postulated accident sequence. To overcome this limitation, Computational Fluid Dynamics (CFD) analysis using ANSYS-Work Bench was performed to estimate the temperature distribution in PT-CT and calandria. In the analysis, the moderator inside the Calandria was not considered. A detailed 3D unstructured mesh geometric model considering all the reactor channels and calandria was developed (as shown in Fig. 7.4 (a), (b) and (c)). The temperature evolution of calandria with time was estimated along with the time available for operator intervention to take corrective actions. This analysis provided more realistic picture in case of Severe Accident where radiation heat transfer plays significant role and the time durations available for operator to take Severe Accident Analysis & Management Guidelines (SAMG) actions are estimated. The insights obtained are useful with respect to SAMG actions also.

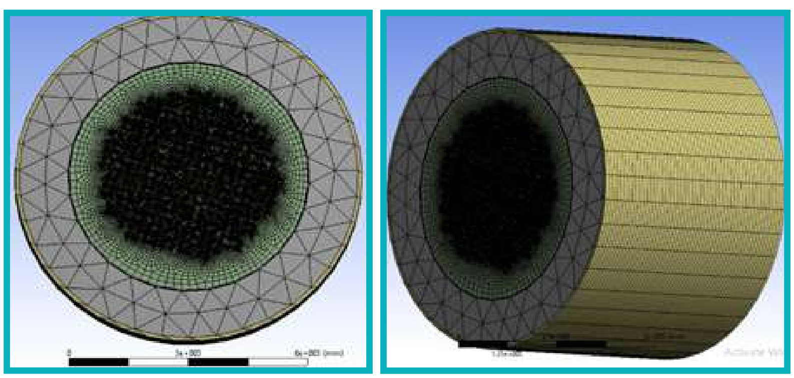


Fig. 7.4: (a) Front and (b) Side view of Mesh of Channels, Calandria & Calandria Vault

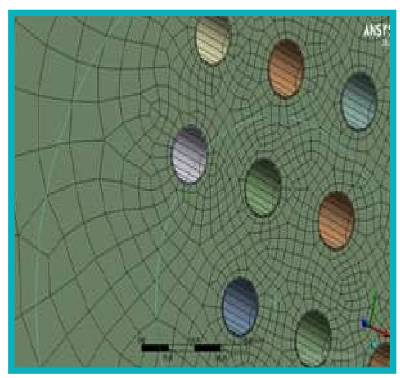


Fig.7.4 (c): Closer view of Mesh

7.2.2 3D CFD Analysis for Hydrogen Distribution in 700 MWe PHWR Containment

The objective is to determine the distribution of hydrogen in the containment in the presence of condensing steam environment without crediting hydrogen recombiners. A full 3D structured mesh for the entire containment was generated. The internal structures and wall thickness were included in this mesh. The in-house developed wall steam condensation model was utilized through the user-defined functions (UDF). The simulation has been divided into three phases, LOCA phase (0-100 s), Hydrogen release phase before OPRD rupture (100-800 s) and Moderator boil-off phase (800-36000 s). The simulation was performed up to 40,000 s (11.1 hrs).

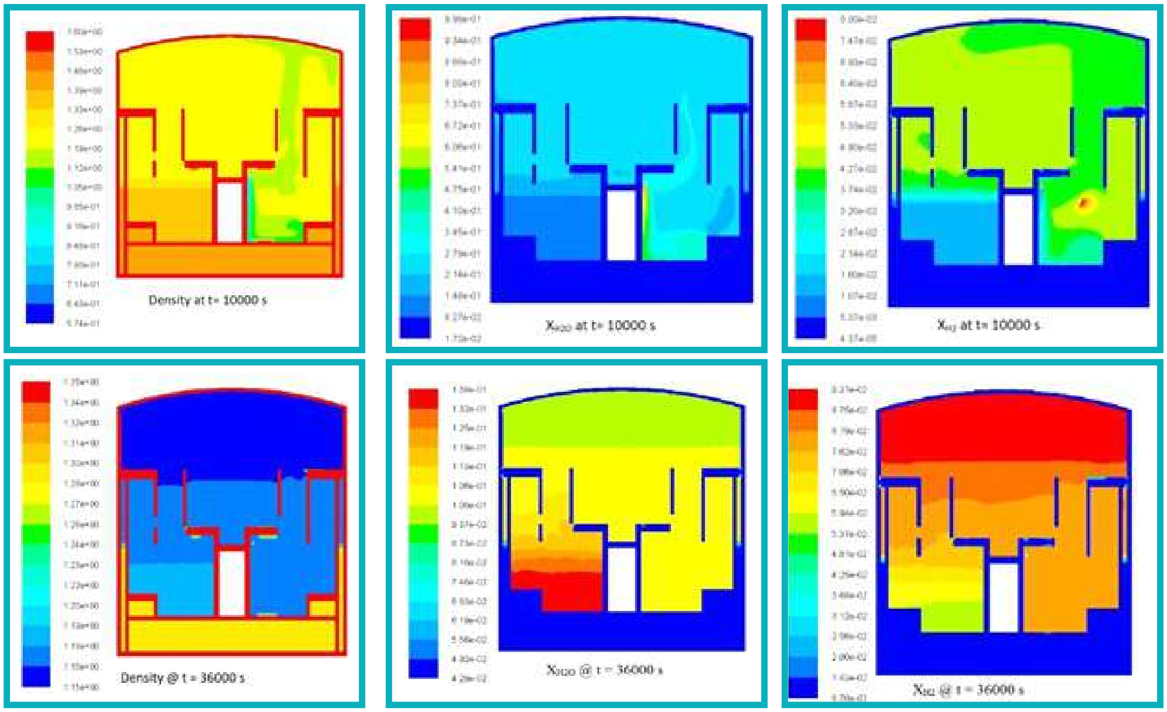


Fig.7.5: Steam, Hydrogen and Density variation in containment at the end of 10,000 s and 36,000 s

The following inferences can be drawn from the simulation results obtained:

❖ In 10,000 s, uniform hydrogen concentration of 4.5 % has been observed throughout the domain except in the injection location.

❖ At the end of 36,000 s, three layers of hydrogen containing fluid mixture with different volumetric concentration of hydrogen are observed. In the dome region, hydrogen concentration of 9.2% is observed which is followed by 8.1 % and 7 % by volume of hydrogen concentration layer.

❖ In the early phase of the hydrogen injection, all three forces viz. Buoyancy, Momentum and Diffusion are important in hydrogen transport. In the later phase only diffusion is the dominant phenomenon in hydrogen transport.

7.3 SAFETY ANALYSIS CODE DEVELOPMENT

7.3.1 Development of Model to Estimate Decay Heat from Fission Product Inventory

Decay heat has been traditionally calculated from standard decay heat curves in the system thermal hydraulic computer codes. These decay heat models/curves are basically correlations for reactor/fuel type and are easy to use. It requires reactor power, time of reactor operation at specified power and time after shutdown for estimation of the overall decay heat. This type of model is suitable to predict decay heat where all Fission Products (FP) remain within the fuel assemblies and with-in the reactor core during accident progression.

Particularly during severe accidents, the fission products inventory in the core no longer remain same as they get released from the fuel depending on the accident scenario. The standard decay heat curves may no longer be valid under such scenarios where FPs (like FPNGs, Volatiles etc.) are expected to move out of the fuel assembly. The present model is useful in estimating the decay heat in containment, containment filters, containment filtered venting system (CFVS), corium etc. in a realistic manner.

A computer code where the decay heat was estimated from first principles i.e. by calculating heat produced during decay of FPs based on the distributed inventories at various locations was developed. It is built on a library of 601 isotopes and 131 isomers covering 53 elements. The in-house developed code can predict decay heat generated from initial isotope inventory and also the distributed inventories. This Code is validated through inter-code comparison by predicting decay heat for a typical pressurized water rector (PWR). The contribution of fission product noble gases, volatiles, semi-volatile and remaining elements to decay heat are shown in the Fig. 7.6.

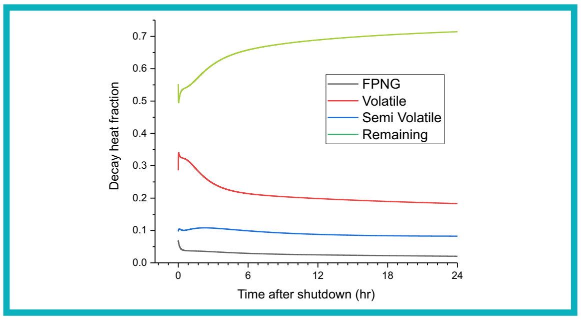


Fig. 7.6: Relative contribution of FPNGs, Volatile, Semi-volatile and remaining Isotopes in Decay Heat

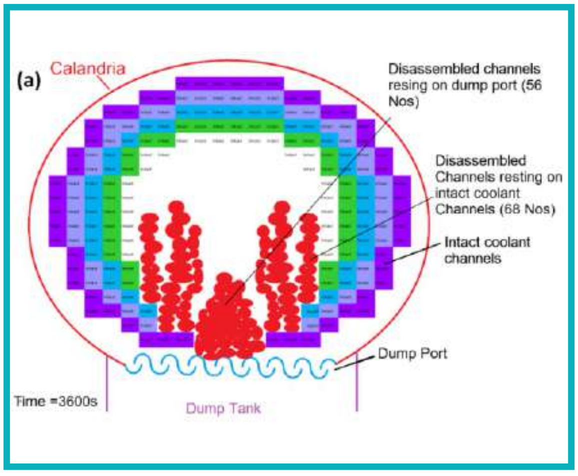
7.3.2 Effect of PCCS Operation on Containment Transients

A generalized numerical model for Passive Containment Cooling System (PCCS) that can simulate flow in a two-phase natural circulation loop containing non-uniform diameter parallel channels was developed and subsequently integrated with the in-house containment thermal hydraulics model 'THYCON'. The integrated code was applied to simulate a postulated severe accident scenario involving LOCA with failure of Emergency Core Cooling System (ECCS), moderator and calandria vault cooling system, in order to comprehend the performance of the PCCS during such events. It is observed that considerable reduction in the containment pressure is possible when PCCS comes into operation. The influence of key variables such as type of heat sink, containment volume and energy release during LOCA on performance aspects of PCCS were also quantified in the study.

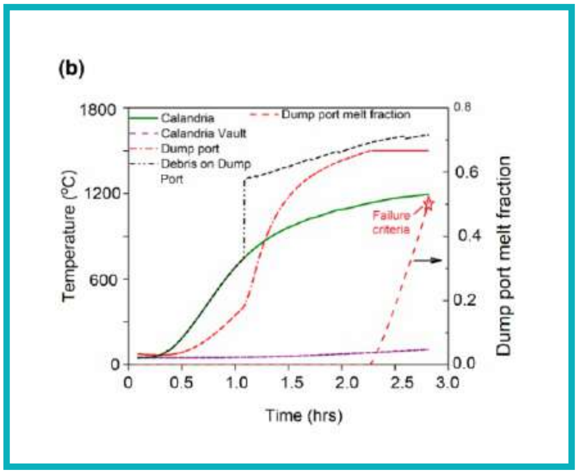
7.3.3 Development of Severe Accident Analysis Program for PHWRs

An in-house numerical computer code called Severe Accident Analysis Program (SAAP) is under development for investigating accident progression in Indian PHWRs. The code has models for simulating both in-channel and in-vessel phenomena in PHWRs. The capability of the code has been upgraded by addition of new models and refinement of existing models with inputs gained from participation in benchmark exercises under DAE Steering Committee to Coordinate Safety Research (DAE-SCSR).

The code has been applied for detailed study of severe accident progression in old generation PHWRs caused by postulated LOCA along with failure of ECCS. In this analysis, the coolant channels are grouped and a representative channel in each group is completely discretized using finite volume method. NPP components such as calandria vessel, calandria vault, dump port and other structures outside the coolant channels are lumped entities. Energy equation is solved in all the coolant channel groups simultaneously and their interactions with structures outside the channels are computed. Sample results of the analysis are given in Fig.7.7 (a & b). Important results include time of first channel disassembly, dump port failure, mass and temperature of corium entering the dump tank and hydrogen generation.



(a) Coolant Channel Disassembly and Slumping on dump Port



(b) Temperature Transients of major Components

Fig. 7.7: Sample results from analysis for LOCA and failure of moderator pump system in older generation PHWRs

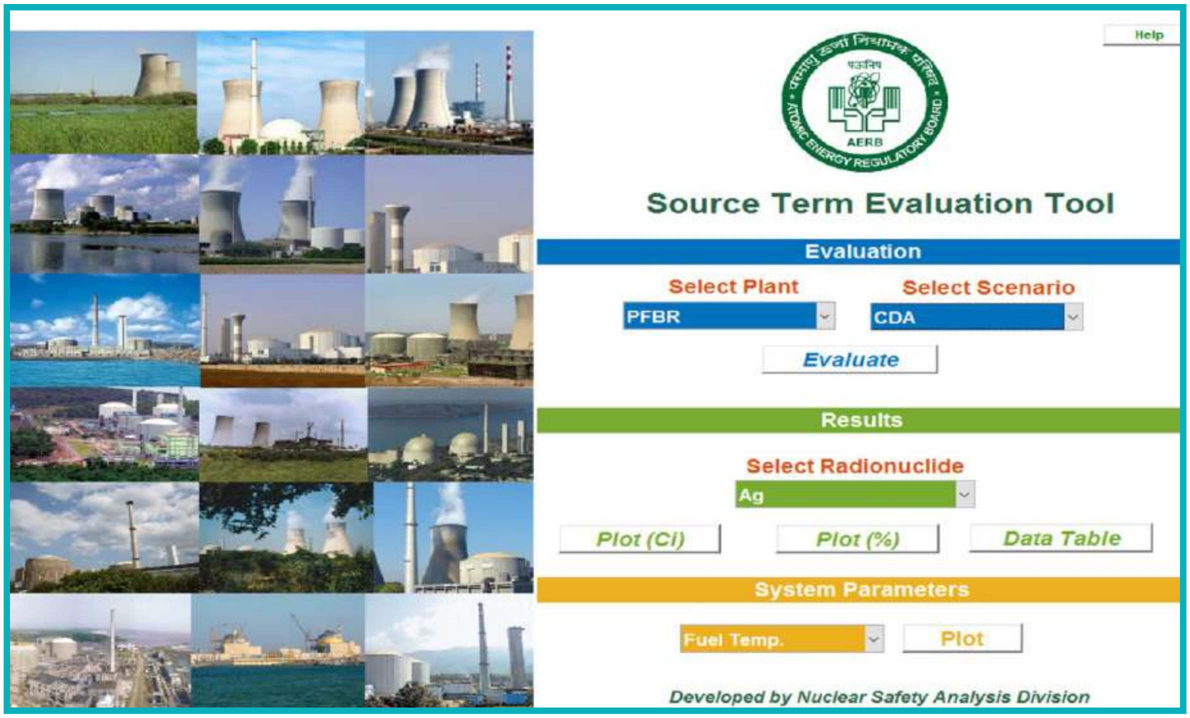


Fig. 7.8: Graphical user Interface of ASTET V3.0

7.4 RADIOLOGICAL ASSESSMENT AND ENVIRONMENTAL SAFETY STUDIES

7.4.1 Release of Version V3.0 of AERB Source Term Estimation Tool (ASTET)

Based on the requirements, an in-house software ASTET is being developed to augment computational infrastructure for Nuclear and Radiological Emergency Monitoring Centre (NREMC) at AERB. The objective of development of ASTET is to aid the informed decision-making during emergencies by quickly estimating the source term using a fast-running tool in real time. The earlier version of the AERB Source Term Estimation Tool was modified to include a module for the estimation of the source term estimation for the scenario of Core Disruptive Accident (CDA) in PFBR. The Graphical User Interface (GUI) of ASTET V3.0 is shown in Fig. 7.8. Additional input decks were also prepared for the simulation of various accident scenarios in the 220 MWe NPPs.

With these updates, now ASTET has capabilities for the estimation of source term considering the release of Fission Products from fuel to RCS, retention in RCS, release to containment and FPs behaviour in containment as per Radiological Impact Assessment (RIA) Guidelines for PHWRs (RAPS/MAPS type, standard 220MWe, 540MWe), VVERs (KKNNP) and PFBR.

There are several accident scenarios and different types of NPPs which are covered in ASTET for source term estimation. As the number of NPPs and accident scenarios are increasing with each updated version of ASTET, it was necessary to map the NPPs name with the applicable accident scenarios. ASTET, version-3, was modified and the mapping of EAL scenarios with the respective NPP in ASTET code was implemented.

7.4.2 Effect of Surface Roughness Parameters on the Wind Flow Characteristics over a Complex Terrain, Complex Site

NPP site situated in a complex terrain such as Kaiga has a special topographic setting leading to complex atmospheric circulations. Kaiga Generating Station (KGS) site is located in a valley region surrounded by hills on the three sides in the Western Ghats. The mouth of the valley region is open towards the Arabian Sea. The hills surrounding Kaiga valley region is mostly covered by tropical evergreen forest. Analysis of the meteorological data collected from the KGS site indicates high frequency of occurrence of calm winds. Numerical weather prediction model Weather Research & Forcasting (WRF) coupled with Lagrangian particle model FLEXPART is used for Radiological Impact Assessment over complex terrain sites such as Kaiga. Earlier studies carried out for the Kaiga site established that, the model overestimates the surface winds.

To resolve the deviations, studies are carried out by modifying the surface roughness parameter (z0) in the WRF model.

The high frequency of occurrence of calm winds over the Kaiga valley region is attributed to the blockage of the south westerly winds by the hills surrounding the valley. Also, the forest canopy on hills, exert a greater drag on the winds to further reduce the speed. The aerodynamic surface roughness (z0) value is the height above a surface at which the logarithmic profile of wind speed versus altitude extrapolates to zero wind speed. Surface roughness parameter influences significantly the near ground level wind velocity and a proper value of z0 needs to be used for realistic prediction of the surface winds. In this study, WRF simulations were carried out by varying z0 values. Comparison of the observed surface meteorological parameters at two observation stations with model simulated values using different surface roughness parameter values is shown in Fig.7.9.

It shows that with increasing z0 values the positive bias in the simulated wind speed is decreasing at both the observation points. Also surface meteorological parameters such as wind direction, temperature and relative humidity is found to be less sensitive towards z0. The study has indicated that surface winds are highly sensitive to surface roughness parameter (z0) and a proper value of has to be used for improved prediction of wind flow characteristics and in turn the RIA.

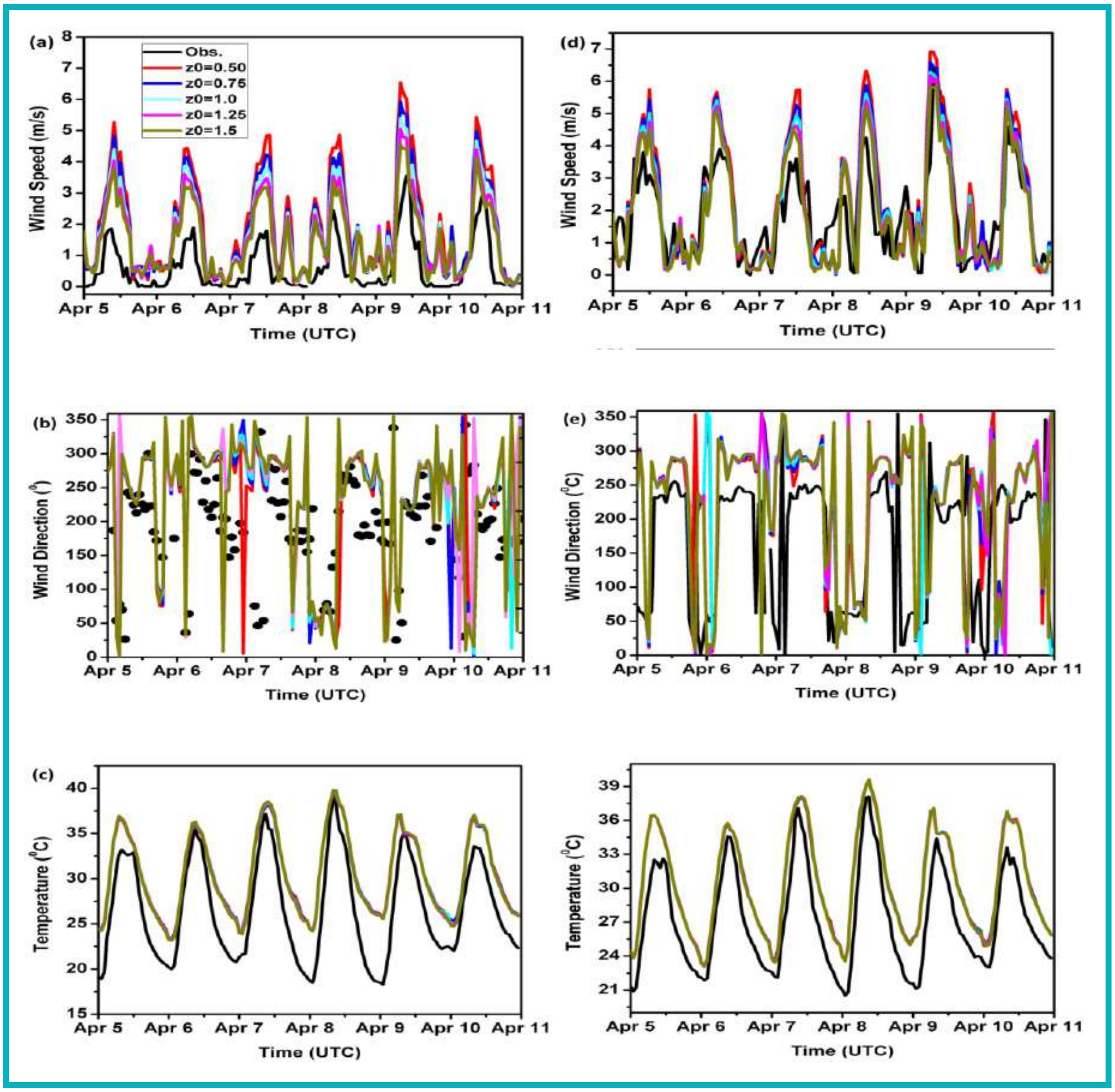


Fig. 7.9 : Time Series plot of Surface Meteorological Parameters at two typical Observation Stations for the Summer Simulation Period

7.4.3 Synthesis and Evaluation of Strippable Decontamination Gel for Decontamination Applications

For the remediation of radioactively contaminated surfaces such as glass, metal, granite etc., diluted acids or detergents are generally employed which generate secondary wastes that needs to be treated separately before it is being stored and disposed. Application of strippable polymer coating is an effective and simple technique to remove radio-nuclides from contaminated surfaces. The secondary wastes from this decontamination process exist in solid form, which is compressible and combustible. For the remediation of Cs, aqueous soluble polymer gel formulations based on citric acid and oxalic acid were synthesized at SRI Chemistry Laboratory (Fig. 7.10). Product evaluation was done on the basis of gel characteristics like decontamination ability, peel-ability, elongation break strength and solubility of spent gel in aqueous medium.

(a) Gel in as applied form

(b) Peeled Gel from the planchette as a film

(c) Planchette after removal of Gel

Fig.7.10: Aqueous soluble Gel

In order to evaluate the synthesized gel, decontamination studies were performed in planchettes made of aluminium and SS with low level liquid waste at CWMF, Kalpakkam. Known amount of the gel was applied on the planchette using a brush and the gel was allowed to dry for 24 hours. The gel was then peeled off and the planchette was counted again for gross activity. It was observed that the entire gel got dried and could be peeled off like a film. The planchette was clean without any sample stain or dirt. Measurements indicated that more than 92% of the activity could be removed using the citric acid-based gel formulation while it was more than 99% for the oxalic acid-based gel.

Fig.7.11: Decontamination of SS Equipment by Gel

In order to find out the efficacy of the gel on SS surfaces, it was applied on an SS equipment surface where known amount of simulated Cs liquid waste was already distributed. The gel was allowed to dry and peeled off subsequently (see Fig.7.11). It can be clearly seen that the gel was peeled as a single sheet. Negligible concentration of Cs was observed in the decontaminated surface.

The present study has demonstrated that for the decontamination of glass, metal and other surfaces, aqueous soluble decontamination gel is very effective. It not only gives excellent decontamination factors but also helps in the recovery of the useful metal ions. The methodology adopted is non-toxic, cost effective and does not produce any secondary wastes which would also help in waste minimization and man-rem reduction.

7.4.4 Synthesis, Characterization and Evaluation of Novel Extractants for the separation of Am (III) from Trivalent Lanthanides

The selective partitioning of trivalent actinides (An (III)) from high level-liquid waste (HLLW) is considered as an essential step in the management of spent nuclear fuel. The presence of Ln (III) in actinide product reduces the efficiency of transmutation, since the lanthanides are efficient neutron poisons that absorb neutrons preferentially during transmutation. Since it is quite difficult to selectively extract trivalent actinides alone from the HLLW containing significant amounts of chemically similar lanthanides, it is essential to extract lanthanides and actinides together as group into organic phase followed by selective stripping of actinides alone from the loaded organic phase.

Soft base N-donor ligands based on Bis-1,2,4-Triazinylpyridines (BTPs), Bis-1,2,4-Triazinylbipyridines (BTBPs) and Bis-1,2,4-Triazinylphenanthrolines (BTPhen) are ideal candidates for the selective separation of trivalent actinides from trivalent lanthanides and other fission products in high level liquid waste. This method involves the group separation of trivalent lanthanides and actinides together by a suitable solvent followed by selective stripping of actinides by an aqueous phase containing the Bis-1,2,4-Triazine ligand. Through AERB-CSRP project, tetra-sulphonated derivatives of Bis-1,2,4-Triazines (BTP) were synthesized, characterized and evaluated for the separation of Am(III) from Eu(III) for demonstrating the feasibility of lanthanide-actinide separation.

Excellent Separation Factors (SF) were obtained for the separation of Am (III) from Eu(III). Radiation stability was established by irradiating the synthesized extractants in the Gamma Irradiation Chamber and carrying out the separation studies.

7.4.5 Synthesis, Characterization and Evaluation of Zero-valent Iron Nano-particles for removal of Nitrate from an aqueous stream

Nitrate containing waste streams are being generated at various stages in nuclear fuel cycle operations. These aqueous streams are generally neutralized prior to either storage or biological treatment. Among the different treatment techniques for nitrate bearing liquid waste, denitrification using nano-catalysts such as zero-valent iron has drawn a lot of attention in the recent times. The present work deals with the synthesis, characterization and evaluation of zero-valent iron nano-particles for nitrate removal from aqueous media. Nano-sized zero-valent iron (nZVI) particles were synthesized by chemical reduction method. The structural, morphological and compositional analysis of synthesised nZVI particles were carried out using a Scanning Electron Microscope (SEM) (see Fig.7.12(a)). The extent of nitrate removal from the aqueous stream was followed using ion chromatographic (IC) procedure. The removal efficiency of nZVI particles was found to have a significant role by varying pH, catalyst loading, effluent concentration, reaction time etc. It was observed that nearly 90% of nitrate removal could be achieved in 60 minutes by using this methodology at pH 3 (Fig.7.12(b)). Nano sized zero-valent particles have proved to be potential candidates for the effective removal of nitrate from aqueous streams.

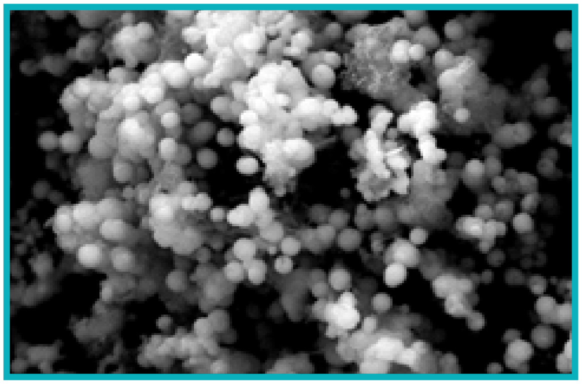


Fig. 7.12 (a) : SEM Micrograph of zero-valent iron nano-particles

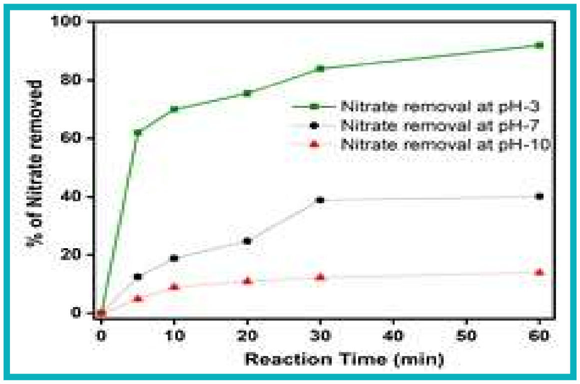


Fig. 7.12 (b) : Comparison of nitrate removal using zero-valent iron nano-particles at different pH

7.4.6 Synthesis and Characterization of Neodymium and Uranium loaded Strontium Borophosphate (SBP) Glasses for Waste Immobilization applications

Various material types have been considered as candidates for immobilization of specific radioactive wastes such as glass, ceramics, cement, bitumen and metals. Glass is widely considered to be the benchmark material for long-term immobilization of complex mixed radioactive wastes. Borophosphate glass has much attention due to their high chemical durability as well as mechanical stability. Such materials are finding applications 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. Neodymium and uranium are doped in SBP to study the solubility and thermal stability. The formation of SBP glass was observed by X-ray Diffraction (XRD) examination of the sample (Fig.7.13(a)).

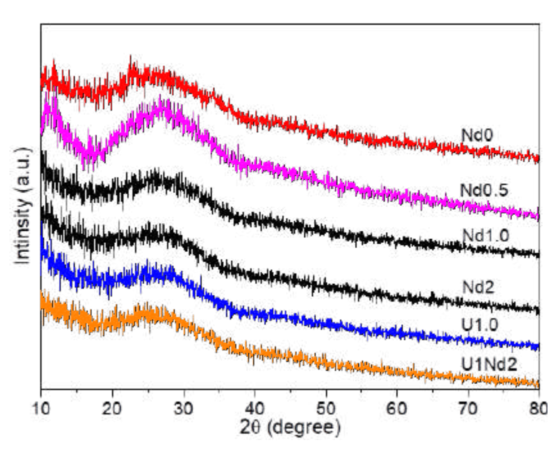


Fig.7.13(a): XRD of neodymium and Uranium Samples

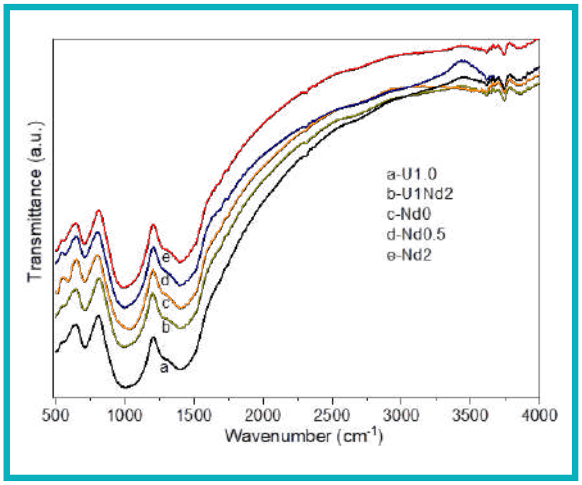


Fig.7.13 (b) : FTIR spectra of neodymium and Uranium Samples

The thermal analysis of the prepared glass materials was carried out by simultaneous Thermo-Gravimetry and differential thermal analysis (TG-DTA). Structural characterization was done by FTIR and RAMAN. Fourier Transform Infrared (FTIR) spectra reveal different network groups in borophosphate glasses (Fig.7.13(b)). Studies have shown that neodymium and uranium loading do not affect borophosphate glass formation.

7.4.7 Generation of GIS based Geo-spatial database on Soil Characteristics around NPP Sites

The knowledge of soil resource of an area is a basic necessity for planners and researchers. To ensure safety of public and environment from the radiological impact due to the operation of NPPs, continuous monitoring of the soils around the site is necessary. With this objective, a digital geo-spatial database on soils of Indian NPP sites and their surroundings up to 30 km radial zone are generated. The soil database was prepared with an aim to provide basic information such as the soil type, distribution, characterization and classification. The database is stored in digital form and can be retrieved for future analysis. For example, soil map of Kaiga Generation Station (KGS) with soil taxonomy as legend is shown in the Fig.7.14(a) & (b).

7.4.8 Impact of Construction of Combined BHAVINI-MAPS Coolant Discharge Canal on Coastal Morphology

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

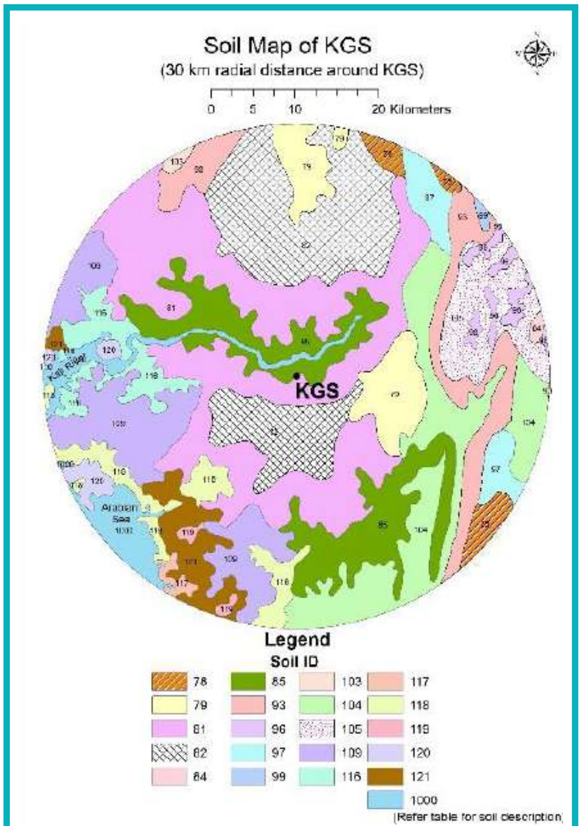


Fig.7.14 (a) : Soil map of Kaiga Generation Station (KGS)

Soil ID	Major Taxonomic Classification
78	Typic Kandiustalfs
79	Typic Kanhaplustlafs
81	Kandic Paleustalfs
82	Kandic Paleustalfs
84	Kandic Paleustalfs
85	Kandic Paleustalfs
93	Kanhaplic Haplustalfs
96	Aquic Ustropepts
97	Ustoxic Dystropept
99	Ustoxic Dystropepts
103	Ustic Kandihumults
104	Ustic Kandihumults
105	Ustic Palehumults
109	Ustic Haplohumults
116	Aquic Ustifluvents
117	Aquic Ustifluvents
118	Typic Kandiustults
119	Haplustults
120	Haplustults
121	Haplustalfs
1000	Waterbody

Fig.7.14(b): Soil Taxonomy of Kaiga Generation Station (KGS)

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 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 at both corners of the wall. Due to this structure there is no impact observed in the nearby environment. Further investigations are in progress. Fig. 7.15 shows the channel discharge.

Fig.7.15: BHAVINI-MAPS Discharge Channel

7.5 PROBABILISTIC SAFETY STUDIES

7.5.1 Shutdown Probabilistic Safety Assessment (PSA)

The review of utility report on shutdown PSA (Rev-0) of TAPS-1&2 has been completed. The shutdown probabilistic safety assessment (SDPSA) was carried out for refuelling outage (cold shutdown for maintenance) with reactor coolant temperature at 55°C; pressure at atmospheric and decay heat 2% of full power in the beginning. In this study, typical refuelling shutdown plan of TAPS-2 has been used to derive the plant operating states. The results of the shutdown PSA indicate that a high level of defence-in depth exists in TAPS-1&2 design during shutdown mode of operation. This is evident from the final CDF value, as well as from the predominantly higher order minimal cut-sets (MCS) observed in the core damage sequences. It was observed that no human action is dominating in top 100 minimal cut-sets. Station has incorporated several hook-up points to inject water as a part of post Fukushima modifications. However, the credit for the above hook-ups, were not considered in the present study to have a confidence that the existing design will comply with the general safety objectives. Moreover, considering the hook-ups in the analysis will mask the plant possible vulnerabilities, important human actions and system weaknesses. Taking the credit of these hook-ups, the CDF value will reduce substantially.

7.6 EXPERIMENTAL STUDIES

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

The COre Melt REtention Facility (COMREF) is established in SRI Engineering hall at Kalpakkam, to investigate the in-vessel retention capability of calandria vessel during postulated severe accident conditions. The decay heat flux due to corium on the calandria vessel inner surface is simulated using an induction heating machine. The heat flux across the plate is measured using an array of thermocouples mounted on the both side of plates. Several experiments have been carried out in COMREF to ascertain the possibility of sustained film boiling on the outer surface of the calandria vessel. Experiments were carried out for both saturated and sub-cooled boiling by varying the applied heat flux to study the in-vessel retention capability of calandria vessel. For sub-cooled pool boiling without any obstructions, test vessel was able to dissipate the heat into the vault water up to a heat flux of 270 kW/m2. However, for sub-cooled pool boiling with obstructions at outer surface like calandria outlet pipes, lower heat removal was observed.

Numerical studies were also carried out to understand the effect of calandria vault water temperature in the presence of flow obstructions. Based on these studies, a correlation has emerged between the Critical Heat Flux (CHF) value and the degree of sub-cooling of the bulk boiling liquid.

7.6.2 Performance of Fire-Retardant Paint on Cables

As part of confirmatory research and competence development, experimental studies in the field of cable fires were continued in Compartment Fire Test Facility (CFTF) at SRI, Kalpakkam. Fire performance of cables coated with different thicknesses of intumescent paint was investigated. Test samples were prepared and experiments were carried out based on available standards for cable tests (IEC 60332-3), with some modifications. Cable samples were prepared using aged cables and also using fresh cables used for power, control and signal transmission. Parameters such as electrical continuity after fire exposure, spontaneous ignition, swelling, decomposition, flame propagation length, mass loss percentage (Fig. 7.16(a)), effect of fire-retardant coating on cable performance etc. were obtained. It was observed from that electrical continuity of uncoated cables was lost within certain cores. However, a coating thickness of 1.5 mm ensured continuity in all the cable samples tested. A snapshot of cable samples before and after exposure to fire is shown in Fig.7.16(b).

Fig. 7.16 (a) Mass Loss Percentage of several Cables with varying number of coatings

Fig. 7.16 (b) Snapshot of Cable samples before and after exposure to fire

7.6.3 Water and Steam Interaction Facility (WASIF)

The Water and Steam Interaction Facility (WASIF) Phase-1 has been commissioned within the high bay of the SRI Engineering hall at Kalpakkam to investigate Condensation Induced Water Hammer (CIWH) phenomena. It has been set up in collaboration with BARC. As part of commissioning of the facility, several activities were undertaken such as performance test of steam boiler in stand-alone mode, preparation of test matrix, experimental procedure, safety precautions to be taken etc. Sixteen CIWH experiments were conducted using test pipe sizes of 40 mm and 65 mm to understand the influence of important parameters. All experiments considered steam in boxed up condition initially within the test pipe & subsequent injection of sub-cooled water into the pipe. Synchronised functioning of various equipment, components and instrumentation system was demonstrated. The repeatability and response of systems to various input conditions, and the healthiness of the facility for conducting further experimental campaigns was also ascertained. Based on preliminary experiments, key process and geometrical variables have been identified for further investigations. A photograph of the WASIF facility and control panel is shown in Fig. 7.17(a) and (b) respectively.

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 related safety issues. There are nine PT-CT assemblies and each is provided with tubular heater of 3.6 kW. The heating rate varies linearly from one end to the other, so as that the PT inner surface temperature varies from around 250 to 300 °C. The direction of heating is reversed in alternate channels. Thermocouples have been installed to measure temperatures at PT inner surface, CT outer surface, annulus gas region etc. (see Fig. 7.18(a)). Experiments were conducted with annulus gas flow rates of 0, 2 and 4 L/min., to study the heat transfer characteristics. Fig. 7.18(b) shows the temperature transients in the annular gas region for a typical channel for 2 L/min flow rate. On the outer surface of the tubes, cooling by natural as well as forced convective conditions were ensured in separate experiments. A full three-dimensional model of the annulus region was developed using a commercial CFD software to simulate the experiments. It was observed that the temperature profile within the annulus region is independent of flow velocities under laminar conditions.

Fig. 7.17 (a): WASIF Phase-I Setup

Fig. 7.18 (a) AGMS and coolant channel heat-up facility

Fig. 7.17(b): Control panel

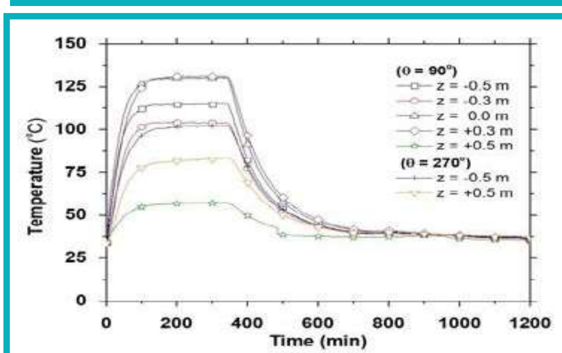


Fig.7.18 (b) Temperature variation within the annular space of channel (1, 2) with time for 2 L/min

7.7 REACTOR PHYSICS STUDIES

7.7.1 Coupled Neutronics-Thermal Hydraulics Benchmark

One of the important benefits of coupled neutronics and thermal hydraulics modelling of reactor cores is relaxation of safety margins without compromising the NPP safety, allowing higher operating power and extended fuel cycles. However, demonstrating credibility of such code system requires comprehensive validation exercise. Towards this, AERB has floated a benchmark exercise on coupled code systems. The benchmark exercise will be carried out in two phases and it has been mooted under the ambit of DAE Steering Committee to coordinate Safety Research (DAE-SCSR). First phase (1A, 1B and 1C) exercises of this benchmark are focused on stand-alone core neutronics and system thermal hydraulics modelling. Phase 1A of the benchmark exercise is focused on assessment of steady state and transient core neutronics estimation capability of the employed models, whereas Phase 1B and 1C are focused on assessment of system modelling. Compilation of specifications and analysis of Phase 1A by AERB participants is completed and Phase-2 will be initiated following conclusion of first three exercises of Phase-1.

7.7.2 Stability Analysis of 700 MWe PHWR

Soft neutron spectrum and separate coolant moderator circuit leads to weak reactivity feedbacks in PHWRs. This deficiency is overcome through reactor regulating system (RRS). Partial coolant voiding near channel exit tends to give positive reactivity feedback beyond certain level of power. However, core stability has been demonstrated by taking the credit of RRS. An independent verification analysis of core stability has been taken up for 700 MWe PHWR. A generic model was developed for stability analysis of bulk power control loop of 700 MWe PHWR which was improved by incorporating zonal models of reactor power, poison dynamics, LZCS, in-core detectors & FMS detectors. Analysis has been 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 RRS. Obtained results of gain values for bulk and zonal control loop have been found to be in agreement with design reported results.

7.7.3 Estimation of Critical Channel Power in 700 MWe PHWR

Regional Overpower Protection System (ROPS) is one of many FOAK design features incorporated in the Indian 700 MWe PHWR that enables reactor trip on local overpower. Critical Channel Power (CCP) is an important aspect of the ROPS design, which brings out the levels of local overpower to be considered in deciding the trip set points. CCP for a coolant channel is the power at which the coolant dry-out or fuel melting begins. Determination of minimum CCP for the core involves solution of fuel and coolant heat transport equations to determine the Critical Heat Flux Ratios (CHFR) and fuel temperatures throughout the core for different reactivity device configurations. It is expected that the reactor will operate in nominal configuration during most of its design life. CCPs for nominal core configuration were independently verified at AERB using the in-house core thermal hydraulics model. It was found that for nominal configuration, CCP occurs due to coolant dry-out (i.e. due to reduction in mCHFR below 1.1) in all the channels. The minimum CCP (mCCP) value obtained in the present analysis has been found to be in agreement with the design reported value.

7.7.4 Independent Verification Analyses of KAPS-3 Start-up Commissioning

Towards independent verification of commissioning reports on start-up of KAPP-3, reactor physics calculations were carried out using DRAGON-DONJON code system. Important parameters like neutron multiplication factor, reactivity device worth, calibration of reactivity devices, moderator and primary heat transport (PHT) temperatures reactivity load etc., were estimated for the test conditions. Analytical findings show good agreement between design estimates and independent analysis results for most of the cases.

7.7.5 Sensitivity and Uncertainty Studies in Core Physics

Various parameters govern the analytical uncertainties in core physics. A case study on the same has been performed using IAEA-CRP lattice benchmark on 37-pin fuel bundle as reference problem using DRAGON code. The code has a collection of models for simulating the neutronics behaviour of a unit cell or a fuel cluster in a nuclear reactor. The variation of the neutron multiplication factors (k and keff), isotopic composition change of U-235 and Pu-239 with burn-up, various reactivity effects due to change in fuel temperature, coolant void percentage, moderator purity and boron in moderator are studied and compared with the IAEA-CRP benchmark. Sensitivity and Uncertainty analysis of infinite multiplication factor (k) with respect to individual input parameter(s) have been addressed for 37-element fuel cluster of PHWR with UO2 (NU). Sensitivity analyses have shown that sensitivity coefficient of k is highest for fuel enrichment and lowest for fuel temperature. The present study also brings out an estimate of upper bound in lattice code computed maximum uncertainty in multiplication factor.

7.7.6 Validation of EXCEL-TRIHEX-FA Code System for VVER-1000 Reactor

EXCEL-TRIHEX FA code system is used at AERB for core physics studies of KK VVERs. Towards validation of these codes, reactivity worth of Control and Protection System Absorber Rods (CPSARs) was estimated for a standard VVER-1000 and compared with other international state-of-the-art code results. For a given core and fuel assembly configuration, few group homogenized cross-sections prepared by a lattice burn-up code EXCEL were used in a three dimensional (3-D) diffusion theory based core physics analysis code TRIHEX-FA. Starting with the calculations of critical boric acid concentrations for various configurations of control groups (8, 9 and 10) in a fresh clean core condition at hot zero power (HZP), steady state reactor physics simulations were carried out for insertion of individual control group in steps one group at a time to estimate the differential worth and hence, the integral worth. Furthermore, for some specific configurations of control groups (8, 9 and 10) as reported in the literature, the worth of individual CPSAR of each group of control and protection system were calculated and compared with the available results in the literature. Comparisons show better agreement of the results of the present EXCEL-TRIHEX-FA simulations with the design calculations. This establishes the credibility of EXCEL-TRIHEX-FA code system for VVER core physics estimations.

7.7.7 Nuclear Design Analysis for KK-VVERs

Twin units of VVER-1000 reactors located at Kudankulam employ 300 EFPD batch refuelling where, Reload Safety Evaluation Reports (RSER) comprising of nuclear design estimates are generated for each fuel cycle. In order to support safety review of fuel loading pattern for each of the above refuelling cycles, independent verification of nuclear design has been carried out using in-house code system. In addition, nuclear design estimates reported in PSAR for units 3-6 employing UO2-Gd2O3 integral fuel burnable absorber (IFBA) based advanced fuel assembly designs with cycle length of ~340 EFPDs have also been independently verified through in-house analysis. Important core safety parameters were evaluated and verified against the design intent and respective acceptance criteria.

7.7.8 Analysis of Uncontrolled Withdrawal of Control Rods in VVER

As part of Russian-Indian bilateral benchmark exercise of VVER Regulators Forum's Working Group (RPWG) of Reactor Physics Analysis, postulated uncontrolled withdrawal of Control and Protection System Absorber Rods in a VVER-1000 has been analyzed using in-house dynamics code TRIKIN. Inadvertent withdrawal of a single or group of CPSARs have been simulated without Safety and Control Rod Accelerated Movement (SCRAM) at minimum controllable level (MCL) of power. Core transient response in terms of reactivity, normalized power, 2-D and 3-D power peaking factors (Kqmax and Kvmax), average linear heat generation rate, maximum fuel temperature, peak fuel center-line temperature, coolant exit temperature, peak fuel clad temperatures, maximum fuel enthalpy etc., have been studied. Analysis finding reveal that core safety parameters do not exceed their prescribed limits for cases considered.

7.7.9 CDA Pre-disassembly Modelling for FBR

A computer code TRAN-SCORE has been developed for analyzing flow and power transient for pre-disassembly phase of core disruptive accident (CDA) in fast breeder reactors (FBR). TRAN-SCORE models include neutron kinetics and associated thermal hydraulics for real time reactivity feedbacks. CDA progression depends on the reactivity feedback which is governed by void evolution and its distribution in the core. To account these two different models, slip flow and slug expulsion models have been employed in the code. A comparative exercise for pre-disassembly phase using TRAN-SCORE code with results from another international code SAS-1 has been done for a reference problem. TRAN-SCORE evaluated core response following loss of flow (LOF) and transient over power (TOP) events has been found in agreement with SAS-1 results.

7.7.10 Development of Fast Reactor Core Dynamics Tool

An in-house and independent computational tool has been developed to analyse various operational transients which occur in a Sodium Cooled Fast Reactor. The developed tool employs coupled neutronics-thermal hydraulics formalism with pertinent reactivity feedback models. The integrated code has been validated against international benchmark results for the event of Transient Over Power Accident (TOPA) and Loss of Flow Accident (LOFA) in a medium sized SFR. Further, the code has been applied to analyse various postulated events in the PFBR at full power. Evolution of SCRAM parameters in those events has been calculated to estimate arrival time for different trip settings.

7.7.11 Reactor Physics Review of Future FBTR Core

Review analysis has been carried out for the conceptual design of future FBTR core that is planned to attain 40 MWth power. Monte Carlo simulations are carried out independently to calculate various physics parameters of the complex core. The study evaluates the core averaged Danger coefficient, reactivity effect of small variation in core parameters, Subassembly replacement worth at different locations, flux spectrum, breeding ratio and subassembly power distribution. In addition, a computer program has been developed by coupling reactor kinetics with decay heat module used to evaluate the power decay profile following SCRAM and LOR. Further, review analysis has been carried out for the postulated events of one primary sodium pump trip, one primary sodium pump seizure, power failure and inadvertent withdrawal of a control rod at full power and low power in FBTR. The analyses were carried out by in-house developed core dynamics tool to estimate the evolution of power, reactivity and core temperatures during these events.

7.7.12 Analysis of Disassembly Phase during CDA in Fast Reactor by Multi-group Neutronics Model

Core Disruptive Accident (CDA), which has a very low probability of occurrence, is designated as a Design Extension Condition (DEC) in fast reactors. In the disassembly phase of this accident, the reactor undergoes a prompt-critical excursion that is terminated by core disassembly process. Towards analysing this phase, a computer code was developed indigenously to simulate the core energetics. The mathematical model consists of coupled point kinetics and 2D hydrodynamics in Eulerian framework. However, international literature indicates that in a super-prompt critical transient the point kinetics model may not be fully applicable. Therefore, for further refinement and better prediction, an attempt has been made to couple the earlier developed hydrodynamics module with space-time kinetics. In the study a multi neutron energy group space-time kinetics solver has been developed and validated by utilising the fully implicit numerical scheme. The coupled code has been tested against a severe transient benchmark in Fast Flux Test Facility (FFTF) reactor model, which was originally evaluated by the VENUS-2 code.

7.8 STRUCTURAL ANALYSIS AND MATERIAL STUDIES

7.8.1 Thermo-Mechanical Diffusion of Hydrogen in Metal (Zircaloy, Steel)

Diffusion behaviour of hydrogen in PT material under different thermo-mechanical loading plays significant role in its safety assessment. To obtain insight into hydrogen diffusion behaviour in metals and alloys under varying stress and temperature distribution, a numerical study is initiated. Simulation was carried out in three steps in an idealised cantilever beam. In the **first step** the concentration at the free end of the beam is ramped to 100 ppm over the step and the steady-state distribution is determined. During the **second step** a bending moment is applied at the end of the beam to achieve a maximum tensile stress of 7.5 MPa and maintained until steady-state mass diffusion conditions are reached. In the **third step** the temperature field is applied to the mass diffusion analysis. A temperature gradient of 3×10^4 K/m is applied until steady-state mass diffusion conditions are reached. Evaluated Hydrogen concentrations were compared with analytical results and were in agreement. Fig. 7.19(a) & (b) shows Hydrogen Concentrations.

7.8.2 High Temperature Properties of Calandria Vessel Material

Since high temperature tensile and creep properties of calandria vessel material plays

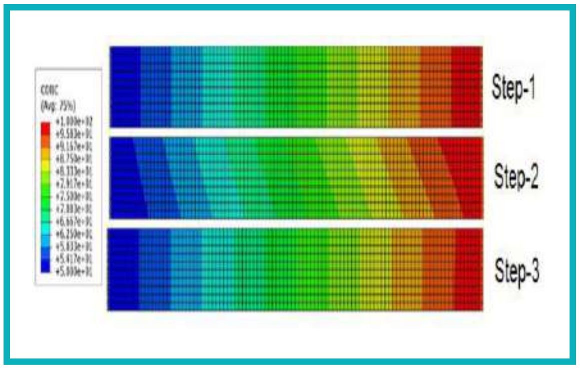


Fig. 7.19 (a) : Contour plot of Hydrogen Concentration at the end of each step

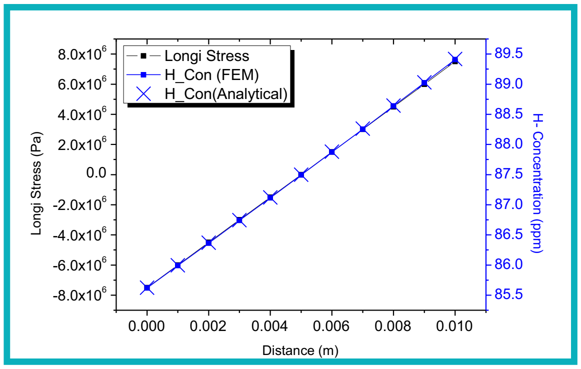


Fig. 7.19 (b): Steady-state Hydrogen Concentration at the end of Step 2 (x=0.075 m)

significant role in evaluating the time to failure of calandria vessel during a hypothetical postulated severe accident scenario in PHWRs, an experimental study was initiated at SRI, Kalpakkam in collaboration with IGCAR to estimate high temperature tensile and creep data that can be used in the finite element framework. Tensile tests were performed over a temperature range of RT to 1123 K using the nominal strain rates of 3E(-3)/s, 3E(-4)/s & 3E(-5)/s. The specimens were heated in a high temperature three-zone temperature control furnace and test was performed under a monotonic displacement controlled condition with different cross head speed.

In order to visualise the influence of temperature on the engineering stress(s)-engineering strain (e) data, s-e curves obtained for the steel is shown in Fig. 7.20(a) for the strain 3 104 s1. At the intermediate temperatures (473 K-873 K), stress-strain curves of the steel displayed the serrated flow behaviour. In addition to that (Fig.7.20(b)), peaks/plateaus in flow stress/strength and work hardening rate, negative strain rate sensitivity and ductility minima were noticed. The observed anomalies are attributed to the occurrence of dynamic strain ageing at intermediate temperatures. The rapid decrease (Fig. 7.20(c)) in flow stress/strength values and work hardening rate, and increase in ductility with increase in temperature and decrease in strain rate suggested the dominance of strain softening at high temperatures above 873K. Fractographic examinations were performed on the tested specimens using scanning electron microscope. On the fracture surface at low magnifications displayed a typical cup and cone fracture for all the strain rates and temperatures examined in this study. Determination of creep and damage properties are in progress.

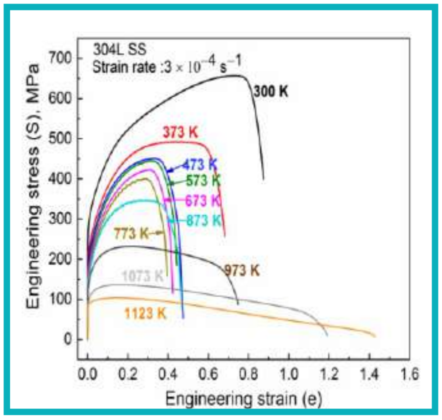


Fig. 7.20(a): Engineering Stress-Strain Curves

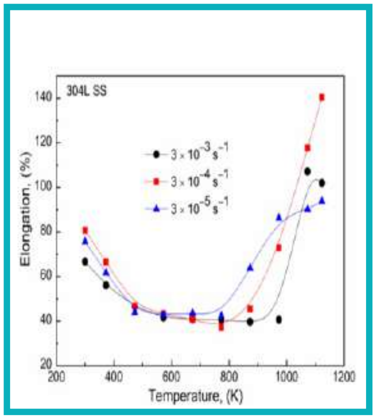


Fig. 7.20(b): Variations in Total Elongation (%) to Fracture

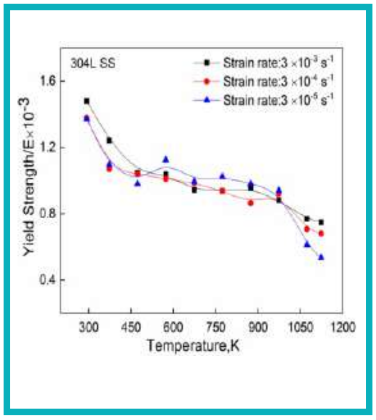


Fig. 7.20(c): Variations in Normalized yield strength with Temperature at different Strain Rates for 304L Stainless Steel

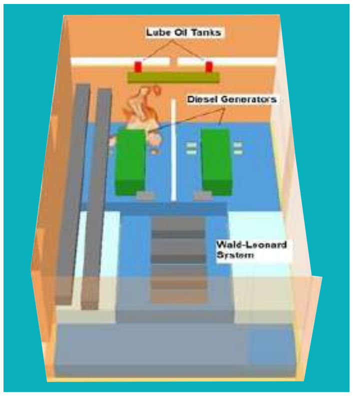


Fig.7.21(a) FDS model of a typical DG room

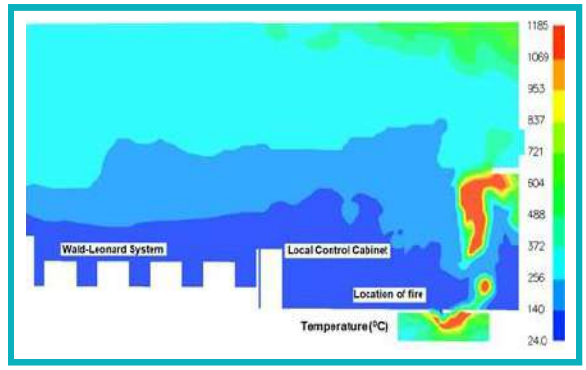


Fig. 7.21(b) Temperature Contour during postulated Fire (Side view)

7.9 FIRE SAFETY STUDIES

7.9.1 Fire Hazard in Emergency DG Building

Fire hazard due to diesel spill and its subsequent ignition within an emergency DG building was assessed by means of numerical studies. A 3D model of a large DG room geometry was developed using Fire Dynamics Simulator (FDS) with inputs from IGCAR (See Fig. 7.21(a)). The objective was to assess whether a diesel spill fire could result in failure of safety critical components such as control cabinets present in the vicinity. The assumed fire scenario involved rupture of a pipe carrying diesel from a day-tank to the DG sets inside the building. The spilled diesel is collected in a dyke and is assumed to get ignited due to a heat source. Apart from the Computational Fluid Dynamics simulations, data from previous experiments in Compartment Fire Test Facility (CFTF) at SRI were also used to estimate fire size and duration. Hot gas and surface temperature, heat flux etc., at target locations critical to safety (i.e., Local control cabinet, Wald-Leonard etc.) were obtained. A contour plot of hot gas temperature is shown in Fig. 7.21(b). Further simulations are in progress to understand the effect of ventilation area, diesel spill quantity etc

7.9.2 Numerical Studies on Fire Rating of Fire-Resistant Panels/Doors

Preliminary numerical studies were undertaken to assess the fire resistance rating of the fire doors in the battery bank room of MAPS. A 2D model of the geometry was developed using a commercial CFD code, COMSOL. Fig. 7.22(a) shows the computational mesh. Heat conduction through the solid materials of panel, convection and radiation on the inner and outer surfaces were modelled. Suitable model for contact resistance at the insulation-metal surface was used. Gas temperature on the exposed side of panel was varied as per the standard Time-Temperature Curve given as per available Indian Standard for fire resistance rating for fire doors (IS 3614). An estimate of the Fire Resistance Rating (FRR) of the fire barrier was obtained using thermal criteria of the outer surface temperature as described in the standard. A temperature contour plot at the end of 1 hr (3600 s) of simulation is shown in Fig. 7.22(b).

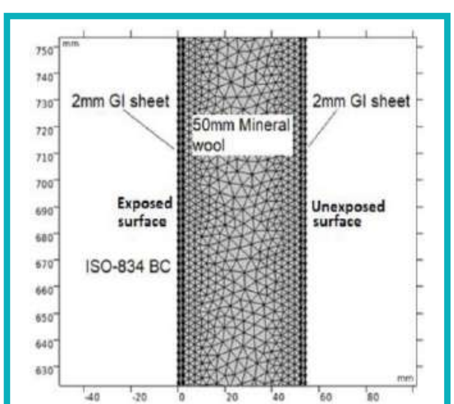


Fig. 7.22(a) : Computational Mesh

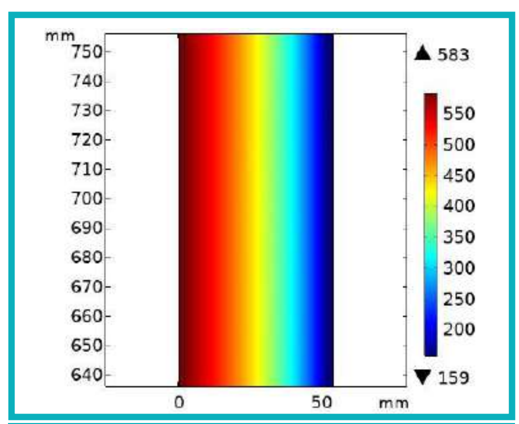
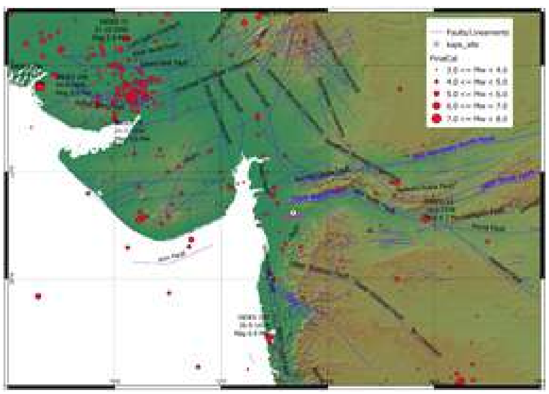


Fig. 7.22(b): Temperature Contour at the end of 3600 s

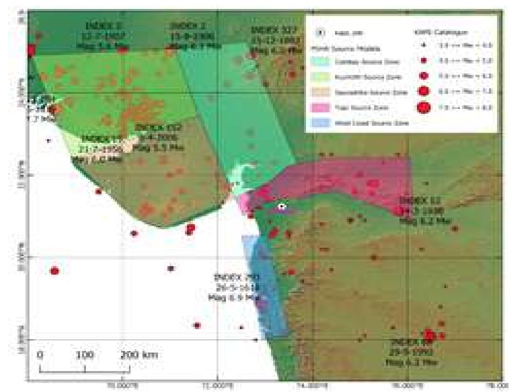
7.10 SAFETY STUDIES TO SUPPORT REVIEW AND ASSESSMENT

7.10.1 Probabilistic Seismic Hazard Analysis of KAPS site

KAPS is situated in Surat District of Gujarat. The Probabilistic Seismic Hazard Analysis (PSHA) Method is introduced in latest draft version of AERB Safety Guide on 'Seismic Studies and Design basis Ground Motion for NPP sites (No. AERB/SG/S-11), as a method of estimating design basis ground motion (DBGM). The DBGM for KAPS site was derived based on deterministic approaches. As part of independent verification and to support safety review of submission from NPCIL, AERB initiated analysis of seismic hazard at KAPS site using PSHA. An in-house source model was developed based on the earthquake catalogue and seismotectonics of the region. The seismotectonic map and seismic source models are shown in Fig. 7.23. The study results will be used in the safety review of seismic re-evaluation of KAPS.



Seismotectonic Map of KAPS Region

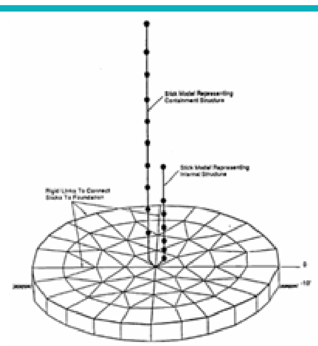


In-house developed Source Model for KAPS Site

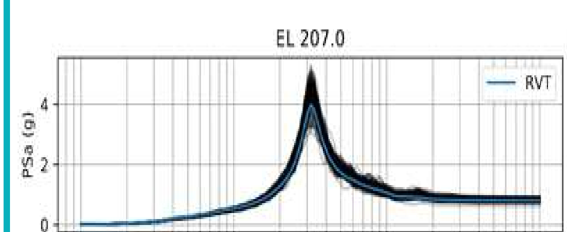
Fig. 7.23: Source Models for KAPS Probabilistic Seismic Hazard Analysis Study

7.10.2 Soil Structure Interaction Analysis based on Random Vibration Theory

Probabilistic seismic Soil Structure Interaction (SSI) Analysis technique is one of the latest and acceptable approaches for seismic analysis of nuclear structure. To develop an in-house expertise in this new emerging area of seismic analysis, AERB initiated a study on application of Random Vibration Theory (RVT) in Seismic SSI analysis. As part of the study, computer codes were generated around existing SSI framework. The developed codes were validated for a SSI analysis of representative reactor building (RB) of PWR wherein the RB was modelled as a beam-mass model (Stick model). The PWR RB stick model and comparison of RVT results with Monte-Carlo simulation results are shown in Fig. 7.24.



Stick Model of PWR Reactor Building



Stick Model of PWR Reactor Building

Fig. 7.24: Comparison of RVT Results with Monte-Carlo Simulation Results

7.10.3 Investigation of certain modelling aspects for Dynamic SSI Analysis of Combined Pile Raft Foundation using SASSI

Soil structure interaction is a critical aspect for structures founded on soil. For GHAVP, a soil site, detailed dynamic soil structure interaction is being undertaken using specialized software, SASSI for major structures (NB, CB, SAB and WMP). NB and CB have combined pile raft foundations, whereas SAB and WMP are normal raft foundations. Combined Pile Raft Foundation (CPRF) is being considered for Indian NPPs for the first time. Towards supporting the safety review of CPRF analysis in SASSI, independent studies were carried out by AERB to investigate two aspects: (a) modelling approach for piles in SASSI, and (b) effect of excavated volume (EV)/NF soil.

In SASSI, for foundations embedded in soil, excavated volume (EV) along with near field (NF) soil needs to be modelled to appropriately capture the interaction between the foundation and the soil. Initially approach adopted by NPCIL did not include the EV. To understand the effect of both EV and NF soil on the responses, independent analysis of structure-foundation system in SASSI was undertaken with and without excavated volume. Results indicated significant increase in the super structure and raft forces when near field soil is modeled (see Fig. 7.25), this resulted in revision of analysis approach for structures, where these effects are significant.

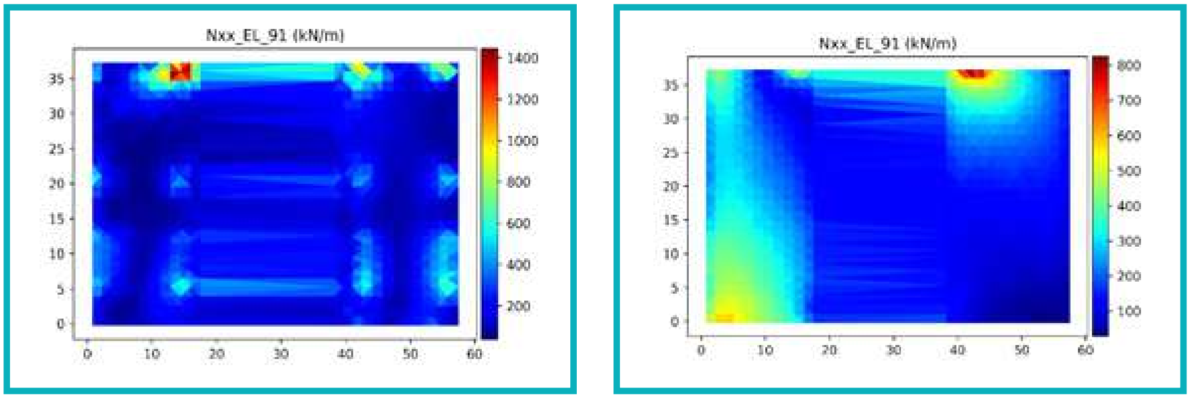


Fig. 7.25: Nxx at Raft for (a) With EV and NF Soil and (b) Without EV and NF Soil

Another aspect which was investigated was the approach for modelling piles in CPRF. Considering the computational constraints, initial studies considered piles as beam elements without NF soil and EV. To understand the effect of modelling pile as solid elements and also the impact of near field soil, AERB undertook a study on a representative structure consisting of 196 nos. of piles. The pile density and super structure dynamic characteristics were tuned to match that of nuclear building. Analysis was carried out for three cases, pile modelled using solid elements, beam elements and without piles. It was observed that for this configuration of the building, the nodal response spectra and stress resultants are in the same ballpark across the different approaches (see figure 7.26). However, to capture the pile-pile interaction the need for modelling NF soil was found necessary. Based on the findings, it was recommended that final design analysis should consider soil elements for pile, and modelling of near field soil and excavated volume.

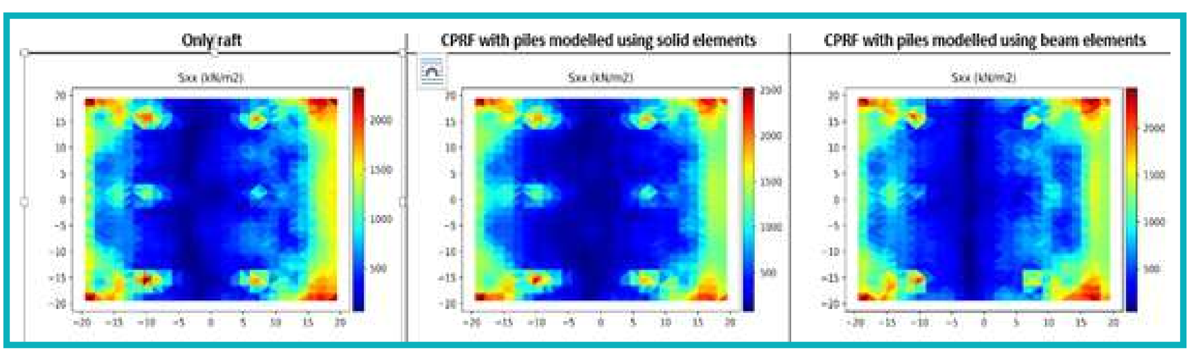


Fig. 7.26: Stress in raft level for case (a) Only Raft without Piles, (b) Piles Modelled using Solid Elements and (c) Piles Modelled using Beam Elements

7.11 AERB FUNDED SAFETY RESEARCH PROGRAMME

AERB promotes and funds research in radiation safety and industrial safety as part of its programme. AERB Committee for Safety Research Programmes (CSRP) frames guidelines for the same, evaluates safety research proposals, recommends grants for research projects and monitor their progress periodically. During this period, CSRP recommended two new projects. It also approved the renewal of five 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, AERB had received about 52 applications requesting financial assistance for conducting Seminars, Symposium and Conferences. But due to COVID-19 pandemic, financial assistance was provided to only five programs.

Table 7.1: New Research Projects Approved

Sr. No	Project Title	Principal Investigator	Organisation
1.	Establishment and capacity building for rapid radiation triage and dose estimation using gene expression biomarker	Dr. P. Venkatachalam	Sri Ramachandra Institute of Higher Education and Research, Perur, Chennai
2.	Development of an on-line Measurement System for Hydrogen Concentration in Steam Environment	Dr.U.Ramachandraiah	Hindustan Institute of Technology and Science, Chennai

Table 7.2: Research Projects Renewed

S. No.	Project Title	Principal Investigator	Organisation
1.	Assessment of Liquefaction Potential through Analytical Methods	Dr. S. D. Anitha Kumari	M. S. Ramaiah University of Applied Sciences, Bengaluru
2.	Performance Evaluation of Generic Compounds with Effective Functional Group as Corrosion Inhibitors for RC Structures	Dr. Shweta Goyal	Thapar Institute of Engineering & Technology, Patiala
3.	Phytoremediation of Radioactive elements (Cesium and Strontium) from contaminated soil and water	Dr. N. K. Dhal	CSIR - IMMT Bhubaneswar
4.	Studies on levels of Natural Radiation in the Environment of hill Districts of Manipur	Dr. S. Nabadwip Singh	Oriental College, Takyel, Imphal
5.	Determination of Anisotropic elastic constants and Anisotropic yield parameters for Zr-2.5% Nb Pressure Tubes	Dr. Avijit Kumar Metya	CSIR -NML, Jamshedpur

Note: For any further information on 'Safety Analysis and Research' related studies, may be contacted at *head.nsad@aerb.gov.in* and head.sri@aerb.gov.in