10. **NUCLEAR SAFETY**

**INTRODUCTION**

This chapter deals with work done in connection with safety of reactor components. The areas addressed in this connection are postulated pressure tube failure accidents inside the calandria, analysis of aircraft impact on reactor building, fire modeling, thermal analysis of transportation cask for transportation of nuclear materials etc. Several computational codes like FLUSOL & FLUSHELL, IMPACT, PFIRE-M, FAIR, risk monitor etc., have been developed and used. Methodology and testing for qualification approval and ageing studies on hardware systems/components/materials used in nuclear instrumentation have also been dealt with.
10.1 FLUID-STRUCTURE INTERACTION STUDIES FOR PHWR CALANDRIA AND IN-CORE COMPONENTS

Postulated pressure tube failure accident inside the calandria is one of the important design basis accidents that needed to be addressed for ensuring the integrity of calandria and in core safety related components. Earlier studies were carried out to obtain the limiting pressure on the calandria shell and the influence of pressurization of calandria shell due to volume addition of the flashing fluid, bubble growth, shock wave propagation, bubble condensation and bubble collapse were investigated with a one dimensional model. This pioneer study helped to identify the influence of above mentioned various parameters.

For detailed investigation of the problem, transient finite element two-dimensional code FLUSOL and three-dimensional code FLUSHEL were developed to analyze this class of reactor safety problems. In these studies the influence of shock wave propagation on a local six channels model was studied with coupled fluid and shell model. This model was used to analyze the shock wave loading of the neighboring channels due to postulated failure of channel C1 shown in the figure. Fish mouth opening and double ended rupture of the channel leading to generation of loadings due to the surface wave front and line wave front in case of axial cracking of the pressure tube were simulated in this study for TAPS 3 & 4 540 MWe PHWRs. This local model included the wave reflection effects from the neighboring channels while radiation boundary was used at the moderator boundaries to avoid any spurious wave reflections.

It was concluded that the neighboring channels would meet the design requirements though the local shock pressures were higher than the calandria shell pressure as shown in the table. The predictions made from these studies have been verified with the results of the simulated pressure tube failure accident experiments carried out for PHWRs in Whiteshell laboratory.

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10.2 DAMAGE EVALUATION IN 540 MWE INDIAN PHWR NUCLEAR CONTAINMENT FOR AIR CRAFT IMPACT

The loading time history for Boeing and Airbus categories of aircrafts has been generated with in-house code “IMPACT” developed for soft and hard missile impacting on rigid and deformable targets for penetration and perforation simulation. Similitude relations were developed using Riera’s model as reference aircraft Boeing 707-320 (with known crushing strength and mass distribution along the aircraft length) and load time histories were developed for the modern aircrafts. The transient analysis assumes the impact due to smaller domestic planes of Boeing and Airbus families with high velocity which could be possibly maneuvered at lower heights (typically ~50m for a nuclear containment) easily compared to heavier aircrafts of larger sizes, which is a realistic postulation.

Nonlinear transient dynamic analysis of 540 MWe PHWR containment structure has been carried out for Boeing 707-320 and Airbus 300B4-200 impact for cracking, crushing and rebar yielding evaluation. Computational simulation of scabbing, spalling, penetration, perforation and damage evaluation of safety structures for internal / external hard / soft missiles with multiple barriers has been illustrated. The conclusions made from the analysis are:

- Outer Containment Wall [OWC] would suffer local perforation with a peak local deformation of 117mm at 0.19 sec.

- The stress and strain values at impact location are within the limits till 0.19 seconds subsequently the load gets transferred to inner containment wall due to local perforation of outer containment wall.
- Overall integrity of OCW structure would be maintained as the global displacement at points away from impact is of the order of ~ 2 to 5 mm Inner Containment Wall (ICW).
- Local cracking and rebar yielding in ICW observed with maximum displacement of 115 mm.
- The global displacements at points away from impact is of the order of ~ 5 to 10 mm.
- No perforation of inner containment wall is observed.

The conclusions made in this analysis are (with total available wall thickness of 1.36 m for combined outer and inner containment wall) similar to the recently published report by United States Nuclear Regulatory Commission (USNRC) for US NPP stations.

<table>
<thead>
<tr>
<th>Aircraft model</th>
<th>Length of the aircraft (m)</th>
<th>Total Weight (Kg)</th>
<th>Engine Weight (Kg)</th>
<th>Peak Load Including Engine (MN)</th>
<th>Duration Of Impact (sec)</th>
<th>Crushed Length (m)</th>
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</table>

Summary of Loading Time History Generated for Various Aircrafts

Displacement at Point A of outer containment wall (Impact Load Point)

Displacement at Point B of outer containment wall
10.3 COMPUTATIONAL FLUID DYNAMIC FIRE MODELLING FOR NUCLEAR POWER PLANTS

Conventionally, the fire propagation and consequence analysis is carried out by simplified methods, flow models and zone model based codes. Availability of advanced computational machines has now facilitated the use of CFD codes for such studies. In recent years a framework has been established for modeling fire in NPPs and allied facilities which uses the CFD codes apart from the conventional codes. The Cost-Effective Simple Fire Modelling techniques (CESFM), which involved the use of simple, empirical equation have been developed and utilized for solving the international benchmark problem of emergency switch gear room of a PWR and a practical problem of diesel generator room of Madras Atomic Power Station power plant. Various zone model based codes like CFAST, BRANZFIRE and OZONE have been acquired and utilized in the studies on ventilated fires, visibility studies apart from applications in NPPs. An in-house two layer single room zone model based code PFIRE-M has also been developed. Neural network based methodology has been used to predict the sprinkler actuation time and time to flashover occurrence in event of fire. The data required for training the network was generated using the zone model based code BRANZFIRE.

An acquired CFD code FDS (Fire Dynamics Simulator), which is an advanced and dedicated large eddy simulation based code has been extensively validated by modelling natural convection, forced convection, mixed convection, tunnel fire,
shopping complex fire, ventilated fire, Steckler experiment, buoyant plume, plume in presence of a cross wind and other reported experiments and benchmarked against international cable fire benchmark problem for emergency switchgear room of a PWR.

Apart from the applied validation studies the FDS has been used for basic research and separate effects studies on plume entrainment, pulsating pool fire, plume puffing, plume flow structures, flame exhaust, bi-directional flow in presence of a large opening and oscillatory flow behaviour in ceiling opening.

CFD Modelling is being extensively used in solving the applied problems from nuclear industry i.e., evaluation of various possible fire fighting strategies for NPPs, a hypothetical fire near Indian PHWR containment building due to a plane crash and fire risk assessment for the Cobalt Teletherapy System.
An innovative procedure has been established wherein the input from the deterministic fire modeling in terms of fire propagation, fire detection parameter and equipment survival goes to the Probabilistic Fire modeling which in turn estimates conditional probability of damage to the safe-shutdown system (target) during a postulated fire. The fire growth time has been determined using the CFD based code for various rooms with various situations like ventilated/ unventilated/ Door open/ Door Close/ Trash fire/ Power cable Fire etc. These fire growth times have been used to calculate the non suppression frequency by probabilistic modelling. Significant improvements were achieved in the PSA conclusions when aided by the CFD based deterministic analysis as compared to conventional zone model analysis.

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10.4 THERMAL ANALYSIS OF TRANSPORTATION CASK

The transportation cask/packages are used to transport radioactive material from one place to another. Different types of packages are used for the transport of radioactive material. Packages are classified to various categories based on activity and physical form of material contained in the package which are specified in the regulatory guides. The package should demonstrate its compliance to various tests under normal and accident conditions and to general design requirement as required by regulatory authority. The overall objectives of all these tests are to demonstrate that the loss of radioactive contents or dose to the public do not exceed the respective limits specified in the regulatory guides.

For thermal test, package should be tested on fully engulf fire for at least 30 minutes. A fire should be controlled to an extent that it sufficiently engulfs a test package and develops at least the required minimum heat flux (based on the temperature of 800°C to ambient temperature) to the package. A package may be thermally tested in a furnace if acceptable conditions in the furnace can be achieved. Alternately, the same can be achieved by carrying out requisite thermal analysis. The thermal analyses of a number of casks such as cask for transportation of Thorium fuel, marine product irradiator, exposure device etc. have been carried out.

- Spent Fuel Transportation Cask

Fire test analysis of spent fuel cask has been carried out. This cask is 2026 mm long 1176 mm width and 1346 mm in height. Due to symmetry 1/8th model has been modeled. Finite element mesh considered for this geometry is shown below. Design basis: Type-B(M), Shape: Cuboidal, Shielding thickness (lead): 150 mm, Cavity size: 1600 mm x 750 mm x 920 mm

Analysis for normal condition

For this case maximum surface temperature was obtained for the cask with ambient temperature of 42°C with a specified insulation to account for solar heat flux incident on outer surface and heat transfer coefficient. The temperature obtained was within the limit.

Fire test analysis

During fire

Fire test has been simulated by specifying boundary condition of 800°C for 30 minutes with no solar heat flux, surface absorptivity of 0.8 and flame emissivity of 1.0. Initial condition for temperature distribution for fire test has been taken from normal condition. Suitable heat transfer coefficient on outer surface with temperature dependent material properties has been considered. Melting is modeled as enthalpy formulation. The contours plot at the end of the 30 minutes are shown in Figure. It is found that maximum temperature occurs at the corners and melt front penetrates up to about 63% thickness at corner of lead at 30 minute fire test.

Post fire

Analysis has been continued after 30 minute fire until all temperature start dropping. The boundary conditions of 42°C as ambient temperature, 0.8 as surface emissivity and suitable heat transfer coefficient were applied for post fire analysis. The melt front continues to penetrate post fire due to the stored heat. It was found that corner region completely melted after ~48 minutes approximately. The temperature variation at

![Sectional view FE Model](image)
the corner location of various materials across thickness is shown in figure.

Similarly, analysis has been carried out for the cask used for transportation of exposure device and marine product irradiator. A finite element mesh and contour plot for the exposure device are shown in the figures.

These analyses help in design improvement and reduce the number of tests needed for demonstration of the test compliance.

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10.5 FUEL MODELLING UNDER NORMAL OPERATION AND ACCIDENTAL CONDITIONS

The prediction of fuel pin behavior under different reactor transients is an important requirement for safety analysis. The fuel pin including clad forms the first barrier against release of radioactive material to public domain. Hence, the assurance of integrity of clad under all reactor scenarios will improve reactor safety considerably.

A finite element based code FAIR has been developed over the years for this purpose. This code has modeling capability of complex thermo-mechanical and chemical processes occurring in a nuclear reactor fuel pin.

Code addresses following issues related to high burn-up of fuel:
- Conductivity Degradation
- Radial Flux Redistribution
- Fission Gas Release from HBS
- Pellet-Clad Mechanical Interaction
- High Burnup Grain Structure
- Burnup Dependent Grain Boundary Saturation Limit
- Burnup Dependent Mechanical Properties
- Burnup Dependent Failure Mechanism
- Fuel Matrix Saturation Limit
- He Adsorption and Release

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Results computed by code FAIR for a high burnup fuel
10.6 RISK MONITOR – AN APPLICATION OF PSA

In nuclear power plants, safety is an important issue. Probabilistic Safety Analysis study provides insights into plant processes, mechanisms and possible interaction between plant systems, both for existing plants with operating histories and for plants still in the design stage. In view of this, on line safety has received lot of attention from operation and maintenance personnel.

Risk Monitoring can be defined as being the process whereby a complex technical facility is continuously monitored as regards the functioning or non-functioning of its different subsystems and the risk emanating from the facility is evaluated on the basis of this information. In the widest sense it can be regarded as being part of the risk management of a plant. Risk Monitoring provides safety status information for a plant and thus aids in decision making about whether continued plant operation is tolerable under certain system function outages. It may also support operations and be of help in deciding on maintenance strategies allowing immediate assessment of different plant configurations.

Risk Monitor, a PC based software which can assess the risk profile has been developed. This software can be used to optimise the operation in Nuclear Power Plants with respect to risk over the operating time. Risk Monitor is user friendly and can re-evaluate Core Damage Frequency (CDF) for changes in component status, test interval, initiating event frequency etc. Plant restoration advice, when the plant is in high risk configuration, current status of all plant equipment and equipment maintenance prioritization are also provided in the package. Using this software, ‘What-If’ analysis can also be done.

Software Developmental Aspects of Risk Monitor

The Software has been developed in Visual Basic. The various modules developed in the package are as follows and main screen is shown in the following figure.

a) System Modelling Options
b) Main Summary & On-Line Risk
c) Component data base
d) Component Out-of-Service & Restore
e) What-If Analysis

Applications

Various applications of Risk Monitor software are given below:

Decision-making in operations

If CDF value exceeds the prescribed probabilistic safety criterion, efforts should always be made to lower the CDF through different tests and maintenance policies.

Maintenance strategies

Risk Achievement Worth (RAW), which is the ratio of risk when a component is down to the nominal risk, is the well suited indicator for deciding maintenance policies. Maintenance actions need to be planned according to the order in which RAWs of components are ranked, i.e. components having higher RAWs need to be maintained with higher priority. Similarly, component having higher Risk Reduction Worth (RRW), which is defined as ratio of nominal risk to the risk when a component is completely available, should be given attention from the design point of view, since these can enhance the reliability of the system. This type of decision is less sensitive to the absolute values of the component failure parameters; however, the relative values of failure parameters influence the values of RAW and RRW.

Risk Informed In-Service Inspection

The Risk Informed In-Service Inspection (RI-ISI) program aims at integrating traditional engineering evaluations with insights gained from PSA. The prime use of PSA is to obtain an estimate
of risk and relegate it to various systems and down to components to obtain an idea of their importance in terms of contribution to the risk. Risk Monitor can be effectively employed for analysing the change in CDF whenever there is a change in Inspection plans and thereby analyse for an optimum scheduling plan. Risk importance measures like Fussel-Vesely and Birnbaum Importance are evaluated for various components and systems in the Risk Monitor for risk informed inspection planning.

Review of Technical Specifications

Technical Specifications are usually based on deterministic assessment and engineering judgement. Technical specifications based on probabilistic considerations can be evolved to optimise the Allowable Outage Times (AOT) and Surveillance Test Intervals (STI) for various Systems and components.

Emergency Operating Procedures and Risk Management

The Emergency Operating Procedures (EOPs) are generally based on the considerations of failures in process systems. From the event tree relevant to a particular process system failure, safety systems can be identified and their availability can be ensured so as to maintain the plant in a safe domain. EOPs for the operation of certain safety systems (e.g. fire water injection to steam generator in station blackout situation) based on dominating accident sequences as identified in PSA can be effectively used in risk management.

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10.7 LOCA QUALIFICATION AND THERMAL AND RADIATION AGEING STUDIES OF THE COMPONENTS USED IN NUCLEAR INSTRUMENTATION

Introduction

Components in various systems in Nuclear Power Plants may be subjected to harsh environmental conditions like high humidity, temperature and radiation during the normal operation as well as during the accident condition such as LOCA. Hence, it is essential to ensure reliable operation of these components during the above conditions. Towards this objective, qualification approval and ageing studies on hardware systems/components/materials, to provide reasonable assurance regarding their survival capability under simulated environment even at the end of specified service life, is needed. Facilities like PANBIT and LOCA simulator have been set up within BARC. A similar facility to test bigger components like pump motors, motorised valves, etc has been setup at Electrical Research and Development Association (ERDA), Vadodara.

Technical services are being regularly provided to upcoming and operating Nuclear Power Stations which have significantly helped in taking appropriate decisions in the areas such as (i) standardisation of new engineering hardware and their procurement, (ii) estimation of residual life, (iii) failure analysis and reliability improvement and (iv) import substitution.

Ageing research studies are being carried out in respect of electronic, electrical and process instruments. These include studies on cables and elastomeric materials etc. for various NPPs and other nuclear facilities.

Test and Measurement Facilities

Thermal ageing

Thermal chambers of various ranges and different dimensions are available for carrying out thermal ageing studies of components and equipment. Temperature range varies from room ambient to 300°C as desired. Provision for on-line performance monitoring of the items being tested has also been provided in all thermal chambers. Process air connection can also be made for the testing of process instruments.
LOCA environment simulator

This is a cylindrical vessel made of 6 mm thick stainless steel sheet. Internal diameter is 100 cm and straight length is 120 cm. Maximum steam temperature and pressure achievable are 150°C and 3.4 kg(g) (50 psig) respectively. Oil free air compressor has been connected to the simulator for performance evaluation of pneumatic devices during LOCA test. Provision has been made for recording and scanning of steam temperature, monitoring of pressure, water spray and on-line measurement of performance parameters. Two safety devices, a pressure relief valve set at 35 psig and a rupture disc with 50 psig rupture pressure, have been mounted on the simulator. Provision has also been made for manual release of steam and draining of condensate.

Synergism Simulator

In order to study interaction effects of combined environments, prevailing simultaneously in NPP containment, synergism simulator has been set up in BARC in collaboration with BRIT. This facility consists of temperature humidity chamber, gamma radiation source along with a provision for applying electrical stresses. Internal dimensions of the chamber are 84 x 84 x 90 cm³. It is possible to vary magnitudes of these stresses as per design of experiment. Temperature can be varied from room ambient to 80°C with relative humidity up to 95±5%. However, temperature can be varied from room ambient to 150°C when used as temperature chamber alone. Dose rates can be varied from 2 to 30 krad/hr by using 3 lead shields for the attenuation of gamma field. It is also possible to study dose rate effects. Provision has been made for on-line measurement of performance parameters. Dose rate outside the synergism simulator (original existing PANBIT facility) can be varied from 1 to 900 krad/hr depending upon the distance of test items with respect to source.
### Effect of LOCA test (120°C steam) on insulation resistance (IR) of unaged cables

<table>
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<tr>
<th>Sr No.</th>
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<th>After LOCA</th>
</tr>
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<tr>
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<td>10^9</td>
<td>&gt;10^{12}</td>
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<tr>
<td>3</td>
<td>EPR/Neoprene rubber</td>
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<tr>
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### Effect of LOCA test (120°C steam) on insulation resistance (IR) of thermally aged (100°C for 60 days) cables

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### Effect of LOCA test (120°C steam) on insulation resistance (IR) of radiation aged (50 Mrad) cables

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