CHAPTER 1

INTRODUCTION

1.1 INDIA’S THREE STAGE NUCLEAR POWER PROGRAMME

The present electricity generation capacity in India is 100 GWe. With the growing energy demand, India needs about 400 GWe capacity for a modest living standard for its people. In India, the primary energy resource for electricity generation is coal and the coal resources in the country is estimated to be 200 billion tonnes; which is adequate for energy demand for about 50 – 70 years. The other resources like gas, oil, wind, solar and bio-mass are very limited. The only viable energy resource is nuclear energy (Raghupathy et al 2004). In India, Nuclear power programme is being implemented in three stages considering limited uranium resources (61,000t) and vast thorium resources (5,00,000t) found in the monazite sands of coastal regions of South India in the country. India's three-stage nuclear power programme was formulated by Dr. Homi Bhabha in the 1950s to secure the country's long term energy independence. The ultimate focus of the programme is on enabling the thorium reserves of India to be utilized in meeting the country's energy requirements. Thorium is particularly attractive for India, as it has only around 1–2% of the global uranium reserves, but one of the largest shares of global thorium reserves at about 30% of the total world thorium reserves (Jain 2000).

The Indian nuclear programme was conceived based on, unique sequential three-stages and associated technologies essentially to aim at
optimum utilization of the indigenous nuclear resource profile of modest Uranium and abundant Thorium resources. This sequential three-stage program (Figure 1.1) is based on a closed fuel cycle, where the spent fuel of one stage is reprocessed to produce fuel for the next stage.

![Diagram of India's three stage nuclear power programme](image)

**Figure 1.1  India’s three stage nuclear power programme (Jain 2000)**

The first stage comprises Pressurized Heavy Water Reactors (PHWRs) which make use of natural uranium. Natural uranium contains 0.7% Uranium (Ur)235 which is fissile and 99.3 % Uranium 238 which is not fissile; however it is converted to fissile element Plutonium 239 (Pu239) in the reactor. The second stage, comprising of Fast Breeder Reactors (FBRs) are fuelled by mixed oxide of Ur238 and Pu239, recovered by reprocessing the spent fuel of the first stage. In FBRs, Pu239 undergoes fission producing energy, and producing Pu239 by transmutation of Ur238. Thus the FBRs produce energy and fuel, hence termed Breeders. FBRs produce more fuel than consume. Over a period of time, Pu inventory can be built up feeding Ur238.
Thorium 232, which constitutes world’s third largest reserves in India, is not fissile and therefore needs to be converted to a fissile material, Ur233, by transmutation in a fast breeder reactor for the third stage. This is to be achieved through second stage of the program, consisting of commercial operation of FBRs. In the second stage, once sufficient inventory of Pu239 is built up, Thorium232 will be introduced as a blanket material to be converted to Ur233. In the third stage, the reactor uses Thorium 232 directly and produces Uranium 233. Considering the sequential nature of the indigenous nuclear power program, and the lead time involved at each stage, it is expected that appreciable time will be taken for direct thorium utilization.

FBRs utilize the natural Ur fuel very effectively (~75%) through breeding and thus provide a rapid growth potential (300 GWe for about 30 years). The use of thorium in FBRs in the third stage will make it a much larger resource (1500 billion tonnes of coal equivalent) than the coal, oil and gas resources. Thus the three stage nuclear energy programme will provide long term energy security utilizing the indigenous uranium and thorium reserves.

1.1.1 Current Status

The first stage consisting of PHWR has reached a state of commercial maturity and the second stage of FBRs has been commercially launched with the construction of 500MWe FBR at Indra Gandhi Centre for Atomic Research (IGCAR), Kalpakkam. The third stage systems have been developed at pilot scale. The development of commercial technology of third stage is under way currently (Gajapathy et al 2008).

1.2 MOTIVATION FOR THE RESEARCH WORK

The design of breeder reactors was originally proposed in India in 1950s as a part of India’s three stage nuclear program. A 10 MWt fast reactor
was established in 1965 at the Bhabha Atomic Research Center (BARC) and has not become a success (Thomas et al 2002). And in 1969, India’s Department of Atomic Energy (DAE) obtained the design of test reactor from French Atomic Energy Commission (CEA) which was India’s first Fast Breeder Test Reactor (FBTR).

The first one and a half decades witnessed several accidents of varying intensity. A major accident happened in May 1987 due to the failure of a protective circuit in the rotation of large rotatable plug and the event was described as the result of ‘a complex mechanical interaction’. The reactor was then commissioned only in May 1989 (Srinivasan et al 2006).

Another incident which led to sodium leak occurred in September 2002 due to defective manufacturing process of sodium valves. The leak led to spilling of 75 kg of sodium over the cabin floor. It took three months for bringing the cabin back to normal condition after removing the radioactive sodium. The FBTR has also faced many accidents with an availability factor of approximately 20% in the first twenty years of its life (IGCAR 2003).

The DAE requested budget support from government of India in 1983 to start a larger PFBR and the approval for design and construction permit was awarded in 1990. The construction of the reactor began in October 2004 and now it is under commissioning. Three levels of reactor safety measure is adopted for the design of PFBR, viz., design with adequate safety margin, early detection of abnormal events to prevent accidents and mitigation of consequences of accidents, if any. The safety features also include two diverse reactor shut down systems and a decay heat removal system. The safety measure also involves in-service inspection of main and safety vessels, sodium piping and steam generators. A provision for early detection of leak in sodium circuits is also made in the design (Kumar et al 2008).
Based on the operating experience of FBTR, Design Basic Events (DBE) are categorized with adequate margins. Events with probability of occurrence > $10^{-6}$/Reactor Year (RY) are called as DBE and those with probability < $10^{-6}$/RY are termed as Beyond Design Basis Events (BDBE). The DBE are further categorized based on the probability of occurrence as given in Table 1.1 and design limits are mentioned in Table 1.2.

The roof slab considered in the present work plays a vital role in the safe operation of the reactor. The roof slab forms the top shield of the reactor assembly and is of box type construction. Being a part of the primary leak-tight boundary of the reactor, the roof slab is classified under safety class-1 components (Mitra et al 2012). Roof slab is complicated due to the presence of various penetrations for IHX, PSP and DHX and supports all the components of the PFBR including the main vessel with fuel and liquid sodium. The integrity of roof slab is very important even under the most unlikely events, like, hypothetical core disruptive accident, fall of heavy objects on it etc., (Jalaldeen et al 1991).

Commercial Fast Breeder Reactor (CFBR) is planned to be commissioned in 2023 after PFBR with thick plate concept for large rotatable plugs instead of the present box type structure (Mitra et al 2012). The conceptual design for CFBR has been started with the experience gained during the construction of PFBR.

Fast breeder reactor is envisaged to play a key role in the assertion of energy security in future for India. Studies on PFBR technology helps in the scaling of the Indian fast reactor program (Maity et al 2010). As the roof slab supports the entire load of the reactor, it plays a major role on the structural health and safe operation of the nuclear reactor. This necessitates the assessment of structural integrity of the roof slab and optimizing its performance under static and dynamic loading conditions.
1.3 PROTOTYPE FAST BREEDER REACTOR

Indira Gandhi Centre for Atomic Research, Kalpakkam is responsible for the establishment of fast breeder technology in India. The commissioning of 40 MWt / 13 MWe FBTR at Kalpakkam in 1985 marked the beginning of FBR programme in India. Considerable operating experience has been gained and has given the confidence to commence the next phase of FBR programme, i.e., construction of Prototype Fast Breeder Reactor (PFBR).

PFBR is a 500 MWe, sodium cooled, pool type, mixed oxide fuelled reactor with two secondary loops. The reactor is located at Kalpakkam, close to the 2 x 220 MWe PHWR units of Madras Atomic Power Station (MAPS). PFBR used plutonium as fuel and sodium as coolant. High quality stainless steels are the structural materials chosen because of their good compatibility with sodium and high temperature strength.

The reactor assembly (Figure 1.2) refers to all the structural components, which help to support the core and the allied mechanisms for heat removal, power control and shut down. It consists of grid plate, core support structure, inner vessel, main vessel, safety vessel, roof slab (top shield), rotatable plugs, control plug and absorber rod drive mechanisms. A large stainless steel vessel, 12.9 m in diameter, called ‘main vessel’, houses the internals of the reactor. The vessel holds the large pool of liquid sodium weighing about 1,100 tonnes. The main vessel has no penetration and the bottom closure shape is designed to enhance its structural rigidity under core load and sodium pressure. It is supported at the top by welding to the outer shell of the roof slab and is free to expand downward to accommodate thermal expansion. The main function of the inner vessel is to separate the hot and cold pools of sodium. The lower part of the inner vessel surrounds the
core. It has penetrations for the Intermediate Heat Exchangers (IHXs), Decay Heat Exchangers (DHXs) and the Primary Sodium Pumps (PSPs).

The grid plate locates the core subassemblies and is supported by the core support structure, which is welded to the bottom of the main vessel. This structure, together with the inner vessel, acts also as a barrier between the hot and cold pools of sodium. The hot and cold pools of sodium are represented with ‘orange’ and ‘yellow’ colour respectively in the reactor assembly (Figure 1.2) and the flow diagram (Figure 1.3).

![Diagram of reactor assembly](image)

**Figure 1.2 Reactor assembly (Parthasarathy et al 2012)**
The reactor assembly has two other vessels besides the main vessel, namely, the safety vessel and the inner vessel. The main function of the safety vessel is to contain sodium in the event of a leakage from the main vessel, limiting the fall in the sodium level and thus assuring cooling of the core. The gap between the vessels is 300 mm which permits robotic visual and ultrasonic inspection of the vessels. The roof slab which is a box type structure filled with concrete, forms the top shield for thermal and biological shielding in the top axial direction. The biological shielding in the radial and bottom axial directions outside the main vessel is provided by the reactor vault concrete. Figure 1.3 explains the heat transport system of PFBR. It mainly consists of primary sodium circuit, secondary sodium circuit and steam-water system.

![Diagram of the heat transport system of PFBR](image)

**Figure 1.3 Heat transport system of PFBR (Gajapathy 2008)**
Liquid sodium at 397\(^{\circ}\)C is circulated by two primary sodium pumps through the core and in turn gets heated to 547\(^{\circ}\)C. The hot primary sodium is radioactive and is not used directly to produce steam, but transfers the heat to the secondary sodium through four IHXs. The non radioactive secondary sodium is circulated through two independent secondary loops, each having a sodium pump, two intermediate heat exchangers and four steam generators and produces steam. Re-fuelling is done at an interval of 185 effective full power days. The fuel is handled by transfer arm machines, positioned with the help of two rotatable plugs namely Large Rotatable Plug (LRP) and small rotatable plug. The LRP is mounted on the roof slab and the small rotatable plug is in turn mounted on the LRP.

1.4  DESCRIPTION OF ROOF SLAB

The roof slab (Figure 1.4), which forms the top shield, supports the main vessel, rotatable plugs, control plug, in-vessel fuel handling machine, primary sodium pumps, IHX and heat exchangers of the safety grade decay heat removal system. Air is used for cooling the roof slab. A ‘warm top shield’ concept has been chosen to avoid deposition of sodium in the annular gaps. Rotatable plugs provide access to all the core subassemblies, which require fuel handling. It provides biological and thermal shielding in the top axial direction of the reactor. Concrete is used as the shielding material.

Roof slab is essentially a box structure with top and bottom plates interconnected by vertical shells and radial stiffeners welded to them. The gap between top and bottom plates excluding the space occupied by its cooling system is filled with concrete. The diameter and height of the roof slab in the 500 MWe PFBR are 12.9m and 1.8m respectively. It is also made in sectors in shop floor and then integrated at site workshop before final assembly. The material of construction is special carbon steel (A48P2) to avoid lamellar tearing. The selection of roof slab material involves the following
considerations; good mechanical strength in the temperature range of 298 K – 493 K, better weldability, no need for post weld heat treatment, compatible with sodium mist laden argon gas and possibility to weld with main vessel material (Mannan et al 2003). Roof slab has a number of radial stiffeners and T welds. These welds are highly restrained and also heavy welding is involved. The thickness of plates is limited to 30 mm in order to avoid post weld heat treatment. The roof slab is exposed to hot sodium pool and argon cover gas on its bottom side and reactor containment building air on its top. Heat transfer takes place between the sodium pool and the area of bottom plate / vertical shells exposed to the sodium pool through the cover gas. A single thermal shield is provided below the bottom plate of roof slab to reduce the heat transferred to the bottom plate from the sodium pool.

Figure 1.4 Model of roof slab

A separate cooling system is provided in each compartment within the roof slab to remove the heat transferred to the roof slab and to maintain the temperature difference between top and bottom plates. Air is used as the
coolant in a closed loop and there is only one barrier between the cooling air and cover gas. The cooling system pressure is kept slightly negative with respect to cover gas thereby avoiding entry of air into the argon cover gas space. The top surface is covered with 150 mm thick mineral wool insulation to reduce heat losses and to maintain the temperature of top plate at 383 K (110°C). The temperature of top thermal insulation is around 333 K (60°C). Similar cooling arrangements are provided for both large and small rotatable plugs for maintaining the above temperature conditions.

It is evident that the roof slab is subjected to static and dynamic (including seismic) loads and hence it is essential to assess the safety margin available for the roof slab within the design and functional limits.

1.5 DESIGN CRITERIA FOR ROOF SLAB

The roof slab is a class 1 component designed to comply with the French design code, RCC-MR. There is a stringent requirement on deflection and slope under normal operating condition based on functional considerations. The analysis of the roof slab is aimed at predicting the deflection, slope, stress and to check its compliance with the design limit based on RCC-MR code.

RCC-MR is a French code for liquid metal fast reactor issued by AFCEN (French Society for Design and Construction Rules for Nuclear Island Components). The RCC-MR consists of a set of technical rules to be followed for the design and construction of the mechanical equipments of a nuclear installation, and mainly the equipments related to the installation safety. It was developed initially for sodium fast reactors, and this code is applicable to high temperature structures but can also be used for components of other types of nuclear installations. This code has been adopted from the
rules of ASME-Boiler and Pressure Vessel Code, Section VIII, Pressure Vessels, Division 2.

The basic design limits in the code are based on \( S_m \), the design stress intensity. For standard materials (36K, 45K, 60K, and 75K), the design stress intensity is \( 2/3 \) of the minimum specified yield strength \( S_y \). For non-standard materials, the design stress intensity is the lower of \( 2/3 \) of \( S_y \) and half of the ultimate tensile strength \( S_u \). When the bending stress ‘\( P_b \)’ component is combined with membrane stress ‘\( P_m \)’ component at each surface, the resulting stress intensity, \( P_m + P_b \) is limited to 1.5 \( S_m \). The average primary shear stress across a section loaded under design conditions in pure shear shall be limited to 0.6 \( S_m \). The maximum primary shear under design conditions, exclusive of stress concentration at the periphery of a solid section in torsion, shall be limited to 0.8 \( S_m \).

As the roof slab is made of thin plates and shells, two stress components involved are membrane stress and combined bending and membrane stress. The stress intensity mentioned in the design code has been evaluated based on the membrane and combined bending and membrane stress components.

The code defines the following four levels of loading categories (Selvaraj et al 2004):

- **Service Level A** loading – Normal operation (Operating basis)
- **Service Level B** – Includes any abnormal incidence like operator error, control malfunction, loss of power etc., and are anticipated to occur at moderate frequency
- Service Level C (Emergency Conditions)- loadings that have a low probability of occurrence and needs shutdown for repair of damage

- Service Level D (Faulty conditions)- Extremely low probable loadings and operation of reactor is impaired (Safe shutdown)

Based on the operating experience of reactors, the following probabilistic safety assessment approach is formulated and Table 1.1 describes the various design basis events and their probability of occurrence. PFBR comes under category 1 mentioned in Table 1.1 and have to be designed for Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) and should satisfy level A and level D loadings respectively as per RCC-MR.

**Table 1.1 Classification of design basis events**

<table>
<thead>
<tr>
<th>Category</th>
<th>Frequency of occurrence</th>
<th>Examples</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>&gt; 1</td>
<td>Normal operation, planned start up and shut down</td>
</tr>
<tr>
<td>2</td>
<td>$10^{-2}$ to 1</td>
<td>Off-site power failure, pump trip</td>
</tr>
<tr>
<td>3</td>
<td>$10^{-4}$ to $10^{-2}$</td>
<td>Station blackout, pump seizure</td>
</tr>
<tr>
<td>4</td>
<td>$10^{-6}$ to $10^{-4}$</td>
<td>Primary pipe rupture</td>
</tr>
</tbody>
</table>

The primary stress intensities under normal plus OBE and normal plus SSE should respect the criteria given in Table 1.2. In addition to the stress limits, the vertical deflection and slope at the inner edge of the roof slab should be restricted within 5 mm and $43.6 \times 10^{-4}$ radians.
Table 1.2 Limits for primary stress intensity

<table>
<thead>
<tr>
<th>Stress intensity component, MPa</th>
<th>Static (Level A)</th>
<th>Seismic (Level D)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Limit</td>
<td>Value</td>
</tr>
<tr>
<td>Membrane $P_m$</td>
<td>$S_m$</td>
<td>150</td>
</tr>
<tr>
<td>Combined membrane and bending, $P_m + P_h$</td>
<td>1.5 * $S_m$</td>
<td>225</td>
</tr>
</tbody>
</table>

The limit for deflection and slope is specified to avoid the peak in the floor response spectra and in-turn to lower the vertical seismic response of the components mounted on roof slab as any excessive vertical movement will cause reactivity oscillation due to the relative movement of control rods with respect to active core of reactor. The deflection has been measured at the inner shell location of the roof slab.

1.6 ORGANIZATION OF THE THESIS

The thesis is arranged into nine chapters as follows. Chapter 1 presents an introduction to the various nuclear reactor programmes in India. The chapter also discusses the features of Prototype Fast Breeder Reactor and the design details of roof slab. The motivation, scope and objective of the present investigation are also outlined in this chapter. The literature survey pertaining to the objective of the research is presented in Chapter 2. The research approach is highlighted in Chapter 3.

Chapter 4 deals with the development of numerical model of the roof slab. A description of various loads acting on the roof slab and the methods of applying the same are discussed. The development of similitude relations required for establishing the scaled down model of roof slab is
presented in Chapter 5. The chapter also discusses the methodology adopted in the fabrication of roof slab. Experimental investigations carried out on the roof slab and a comparison of the results with that of numerical model are given in Chapter 6. Chapter 7 describes the methodology of developing metamodels and the application of metamodels for design optimization of roof slab. The optimum roof slab configuration has been tested for design adequacy as presented in Chapter 8. Chapter 9 presents the conclusion and suggestions for future work.