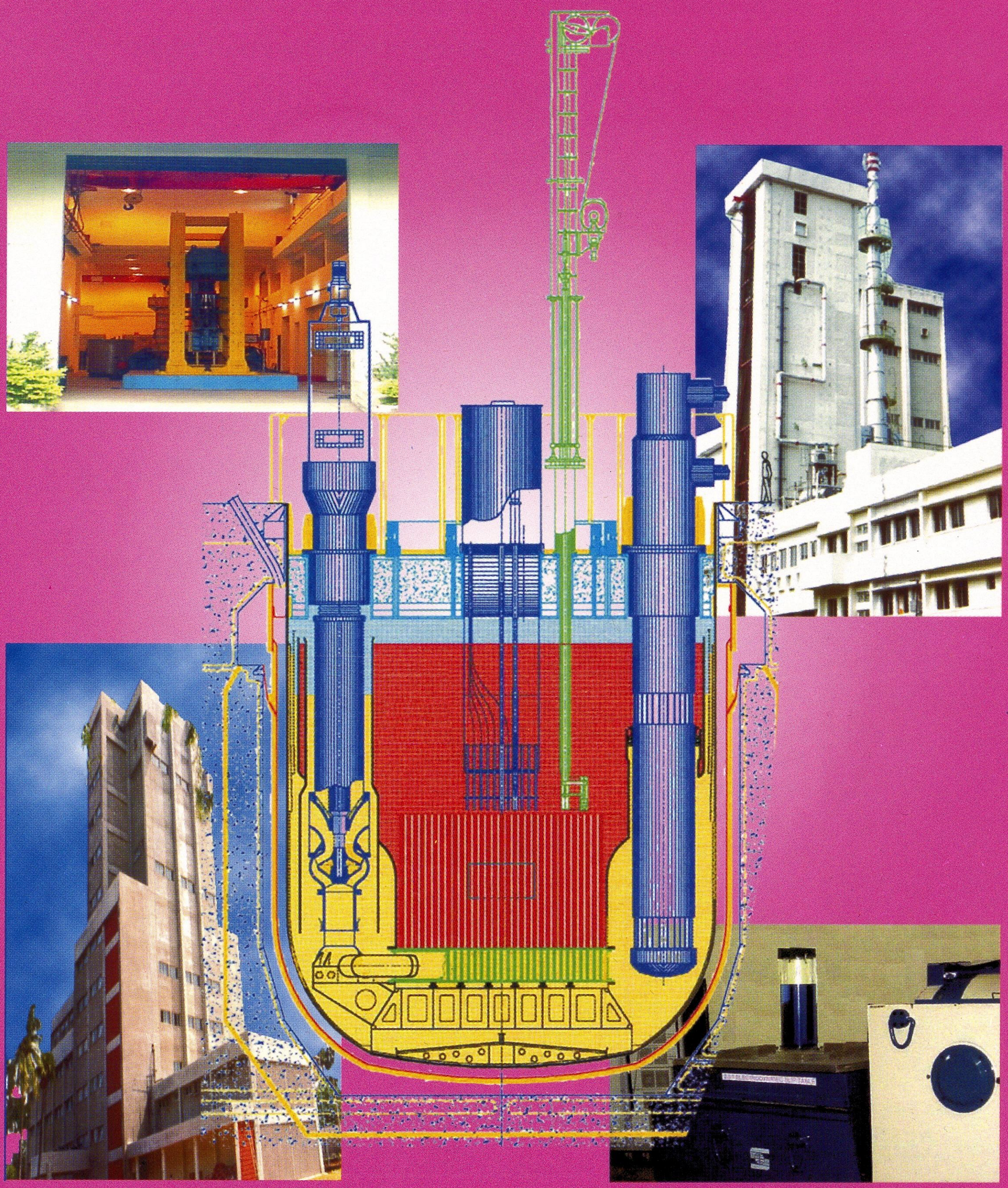
PROTOTYPE FAST BREEDER REACTOR

Engineering Research & Development



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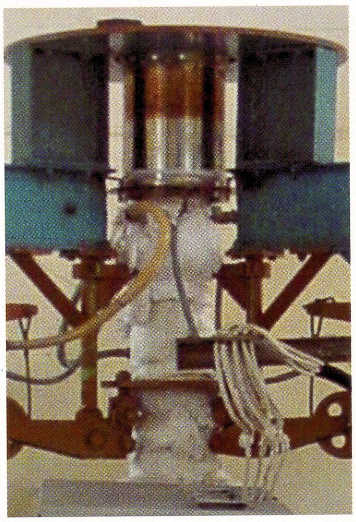
परमाणु ऊर्जा विभाग - स्वर्ण जयंती वर्ष
अगस्त २००३ - अगस्त २००४
Department of Atomic Energy
Golden Jubilee Year
August 2003 - August 2004

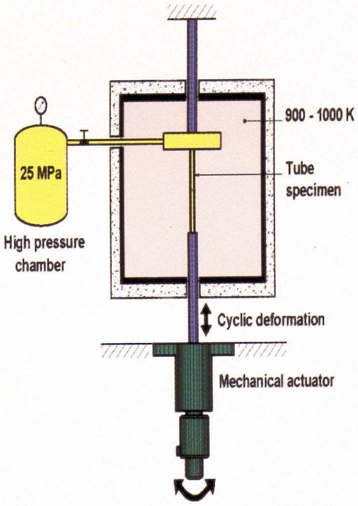
Indira Gandhi Centre for Atomic Research
Kalpakkam - 603 102
December 2003

Design and research & development for 500 MWe Prototype Fast Breeder Reactor (PFBR) is the mandate of Indira Gandhi Centre for Atomic Research. PFBR is a sodium cooled, pool type, mixed oxide fuelled reactor. The design of Nuclear Steam Supply System components is completed with considerable R&D backup. The R&D is oriented towards the generation of design data, design validation and optimization of PFBR. The main R&D activities in areas of reactor physics and shielding, thermal-hydraulics, structural mechanics, component development, fuel cycle, sodium chemistry, materials development, instrumentation & control and safety, are being carried out at IGCAR. Further, certain R&D activities are being carried out in collaboration with national research institutes. Important activities completed in the domain of thermal hydraulics, structural mechanics, component development and safety are highlighted.

HIGH TEMPERATURE COMPONENTS TESTING



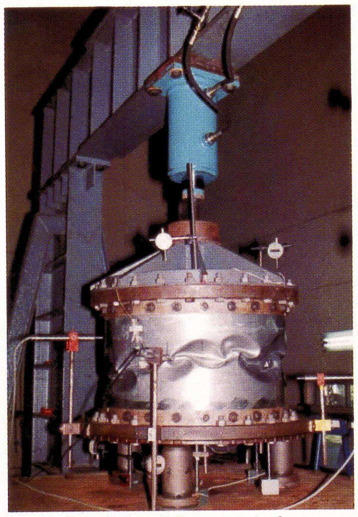
Simulation of ratching



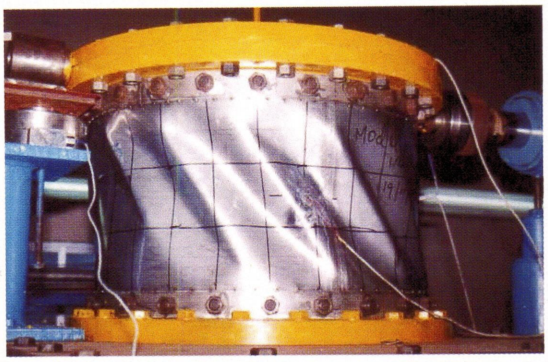
SG/IHX tube tests

PFBR components operate at high temperature (820 K for out of core components and 973 K for core) with large temperature gradients. High temperature failure modes such as ratcheting, creep and fatigue damage, mainly under thermal loadings, decide the plant life. Tests are conducted on models of shells subjecting them to axially varying temperature gradients to simulate thermal ratcheting which affects the vessels facing sodium free level variations. Creep-fatigue damage assessment methodology is validated based on tests on specimens having component features. The rolled and welded tube to tubesheet joints of intermediate heat exchangers (IHX) were qualified by carrying out pull out strength tests after simulating creep relaxation. The steam generator (SG) tubes, made of modified 9Cr-1Mo (G91) are tested at high temperature with internal pressure and axial displacement to simulate the creep rupture. All these tests have raised the confidence in the design of high temperature components, thereby confirming the design life of 40 y.

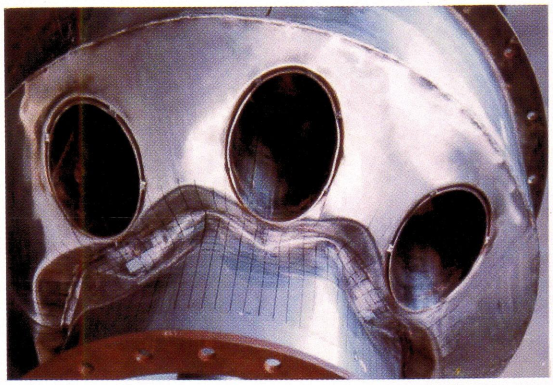
BUCKLING INVESTIGATIONS



Shell buckling test rig



Shear buckling of main vessel model

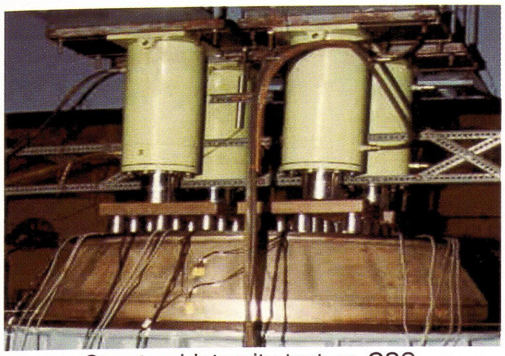


Buckling of inner vessel model

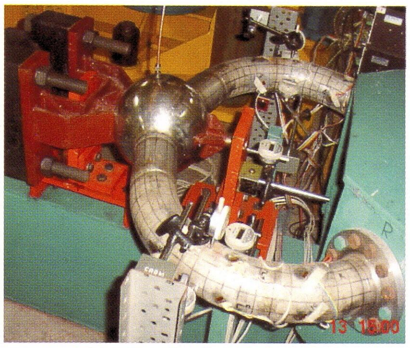
Reactor assembly vessels are basically large diameter, thin walled shells which are prone to buckling. Geometrical imperfections imposed during manufacturing and subsequently by progressive deformations, during operation, plastic deformations, seismic loadings and their dynamic effects, reduce the buckling strength significantly. Buckling of vessels have been investigated extensively with the help of numerical analysis, which are validated thoroughly based on tests on scaled down models. Tests are conducted on 1/30th and 1/18th models of main vessel under bending moment and shear forces and on 1/13th models of inner vessel under internal pressure and concentrated loads.

STRUCTURAL INTEGRITY ASSESSMENT

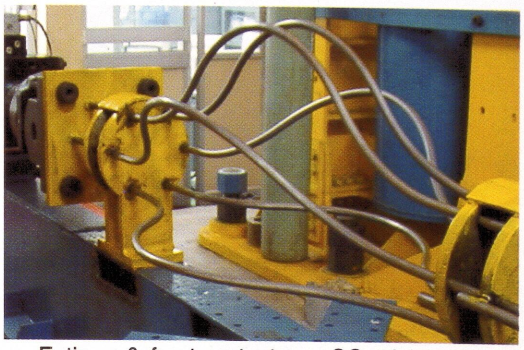
The structural reliability of core support structure (CSS) is investigated by tests on 1/5th models with conservative manufacturing deviations, such as presence of a long through wall crack at the critical load transmitting path and absence of critical stiffeners. It has been demonstrated that the gross deformations and reduced load carrying capacity are acceptable.



Structural integrity test on CSS



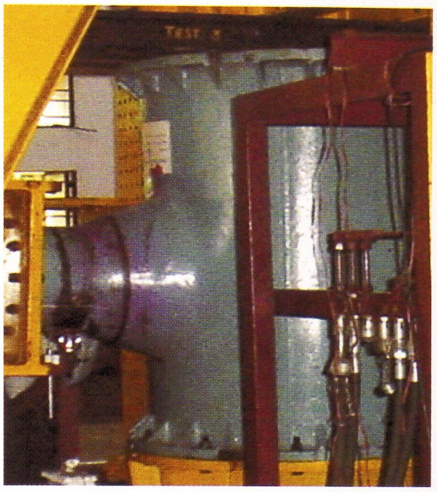
Tests of primary pipe



Fatigue & fracture tests on SG tube bends

RCC-MR: A16 (2002) is used for fracture assessment including the analysis for leak before break (LBB) justification. The crack propagation behaviour is studied in 316LN and G91 steel plates subjected to bending loads. Besides, a series of fatigue and fracture tests conducted on $1/5^{th}$ models of primary sodium pipe and SG tube bends, subjected to internal pressure and bending moment, confirmed the application of A16 rules as well as validated the fatigue design of SG tube bend.

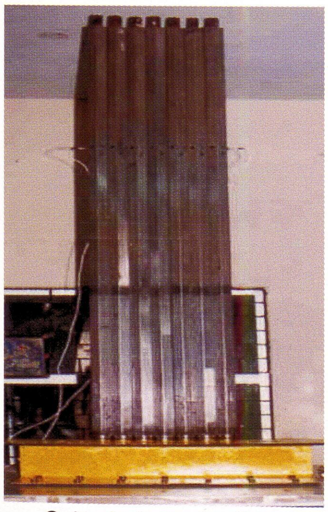
LBB is justified for main vessel, secondary sodium piping and steam generator shell nozzle junctions and validated with experimental data. Fatigue and fracture tests were conducted on typical large size tees, bends and SG shell nozzle junctions of sodium circuit, with the objective of demonstrating LBB.



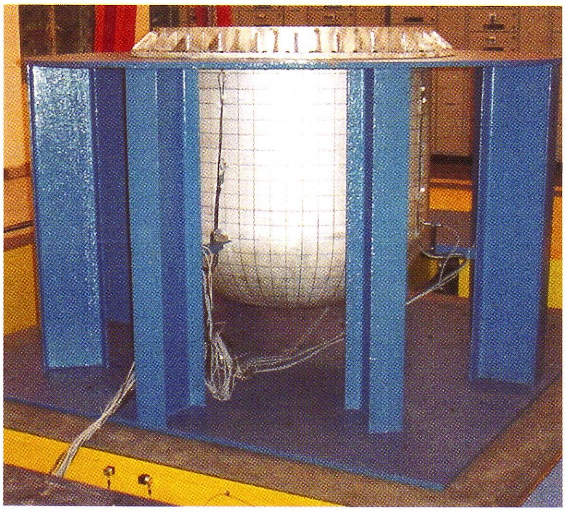
LBB tests on Tee, Bend and SG shell nozzle junction

STRUCTURAL DYNAMICS

In view of thin walled large diameter shell structures with the associated fluid effects, structural dynamics problems are very critical in FBR. Some of them are pump and flow induced vibrations, seismic excitations, pressure transients in IHX and pipings due to a large sodium-water reaction in SG and core disruptive accident (CDA). The structural dynamics problems have been analysed in detail to demonstrate structural integrity. A few critical aspects are validated.



Seismic tests on CSA

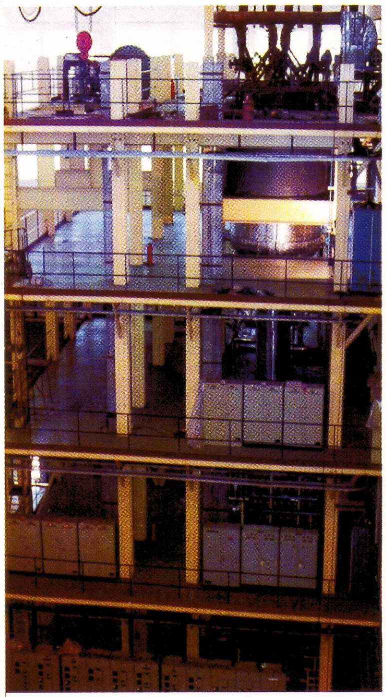


Seismic tests on RA model

A tested vessel of TRIG series

For the seismic qualification studies, a 2 t capacity electrodynamic slip table of size 1 x 1 m and a 10 t capacity hydraulic shake table of size 3 x 3 m have been built. The seismic tests on core subassembly models (CSA) and on 1/10th models of reactor assembly subjected to simulated base excitations have helped to understand the complex fluid-structure interaction dynamics and also provided important data to validate the sophisticated computer codes.

A series of tests conducted on scaled down model of main vessel (MV) with its internals helped in understanding the mechanical consequences of CDA, apart from validation of in-house code FUSTIN. The entire experimental programme, involves 3 TRIG test series to characterise explosives (TRIG-I), tests on 1/30th MV models without internals (TRIG-II) and 1/13th reactor assembly (RA) models with internals (TRIG-III). From the test series, the rupture potential of main vessel is found to be 1200 MJ (the design value is 100 MJ), structural integrity of IHX and decay heat exchangers is assured and sodium release to RCB has been quantified.



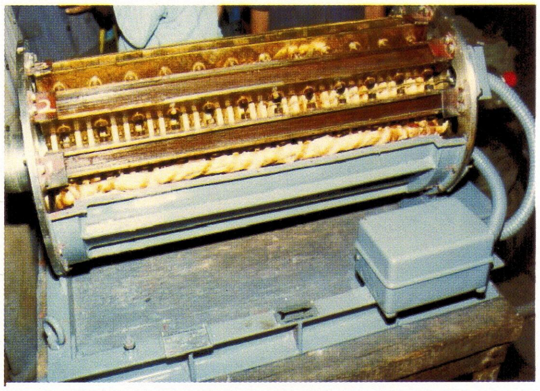
Ultrasonic under - sodium viewer in FBTR

LARGE COMPONENT TEST RIG

The Large Component Test Rig (LCTR) is a dynamic sodium loop with a holdup of 80 t of sodium. The rig is mainly used for the functional testing / design validation, data generation and model studies of components. The rig has three test vessels to test absorber rod drive mechanisms, fuel handling machines and heat & mass transfer studies and has logged 30,000 h of operation at high temperature (823 K).

ELECTROMAGNETIC PUMPS

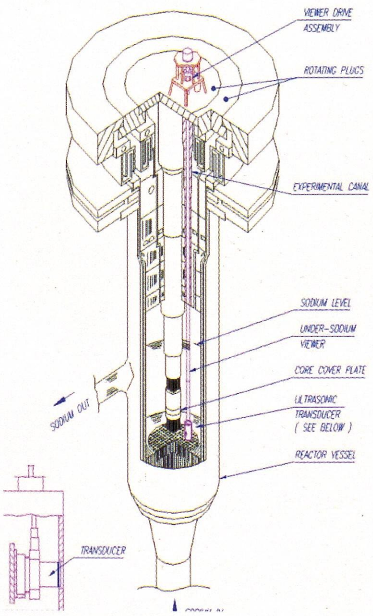
The electromagnetic pumps using sodium conductor find wide application in experimental facilities and in auxiliary systems of FBR, where their operational simplicity outweighs their poor efficiency compared to centrifugal pumps. Design, development, construction and testing of electromagnetic pumps of different types were carried out.



Annular Linear Induction Pump

ULTRASONIC UNDER SODIUM VIEWER

The opacity of sodium excludes conventional optical methods for inspection of vessel internals. Sodium is transparent to ultrasound, with the help of which it has been possible to scan and view the objects. However, the transducer has to function at high temperatures. A 'spinner' type scanner developed in-house is deployed in FBTR.

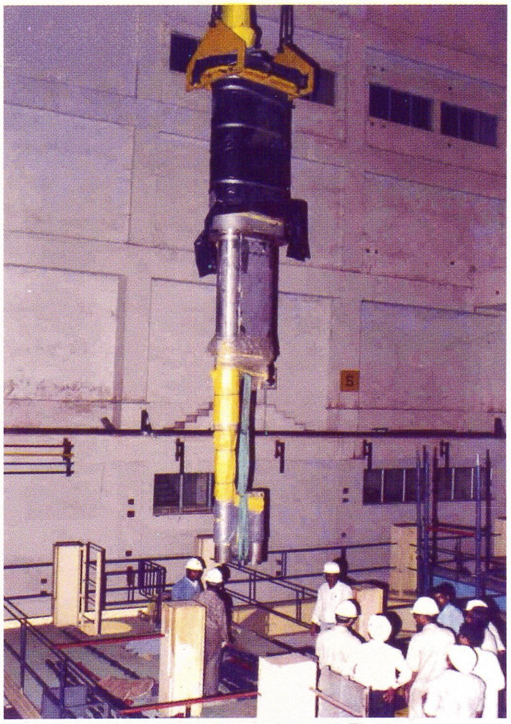


Ultrasonic under - sodium viewer in FBTR

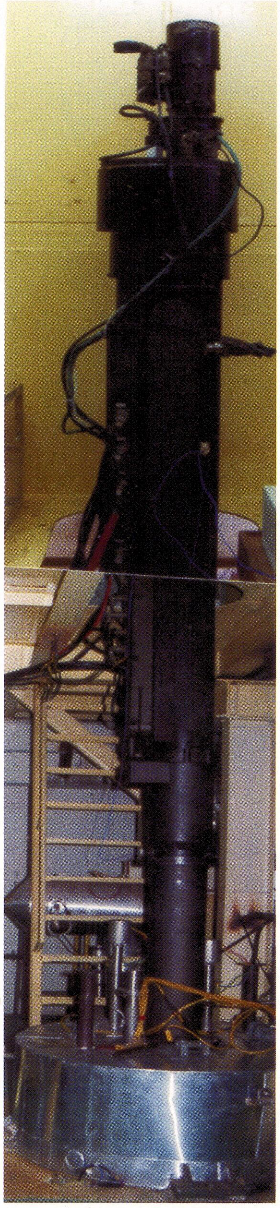
TRANSFER ARM TESTING

Transfer arm is used for handling the core subassemblies in the reactor. The mechanism was manufactured and tested in air at room temperature. Operations such as gripping, lifting, moving to another radial position, lowering, releasing the subassembly and positioning were checked.

The sodium test facility for testing transfer arm at high temperature is setup in LCTR. The sodium vessel holds 40 m^3 of sodium. The vessel is 14 m long and 2.2 m diameter and weighs around 200 kN.



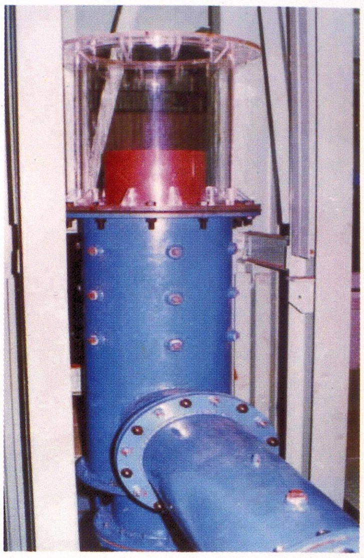
Transfer Arm - Lower Part



CSRDM - Upper Part

ABSORBER ROD DRIVE MECHANISMS

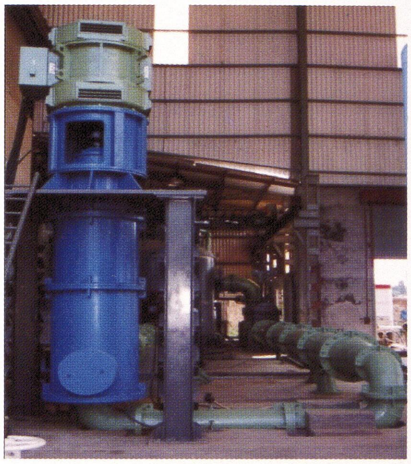
PFBR is provided with two different types of absorber rod drive mechanisms, viz. Control and Safety Rod Drive Mechanisms (CSRDM) (9 Nos.) and Diverse Safety Rod Drive Mechanisms (DSRDM) (3 Nos.). They are used for startup, controlling reactor power and safe shut down of the reactor during abnormal conditions. Many prototype models of the parts of these mechanisms were subjected to functional tests. The full scale prototypes of both the mechanisms are being qualified by sodium testing at temperatures upto 883 K.



Steam Generator Inlet Plenum - 3/5 Scale Model

TESTING OF HEAT EXCHANGER MODELS

The Intermediate Heat Exchanger (IHX) in the primary circuit and Steam Generator (SG) in the secondary circuit are two critical components in the heat transport system. In order to validate the design, hydraulics and vibration experiments are conducted in water and air models. They include a 60° sector full length model and two models of scale 1/2 of IHX. A circular model of 3/5 scale is tested in water to select a suitable flow distribution configuration for the SG inlet and outlet plena. A full scale 60° sector model of SG was tested in water for vibration and pressure drop.



Hydraulic Testing of PSP

HYDRAULICS DEVELOPMENT OF SODIUM PUMPS

Development work on primary sodium pumps (PSP) involved extensive hydraulic and cavitation testing on a 1/2.75 vertical model in water. Hydraulic testing comprised of performance testing (H/Q, P/Q, h/Q and NPSHR/Q) at two different speeds. Cavitation testing involved visual cavity length measurement under stroboscopic lighting, paint erosion tests and buffed SS specimen erosion tests. Similar studies were carried out for secondary sodium pump (SSP).

SSP Model - during testing

ROTOR DYNAMICS STUDIES ON PSP

A test rig is built to study the behaviour of Primary Sodium Pump Rotor Assembly and to validate the spherical seat support concept, hydrostatic bearing leak flow and the performance of rotating parts like seals and coupling. The rig has a rotor of 7100 mm length and 320 mm diameter, weighing 3 t including all rotating parts. The vibration behaviour of the rotating system and orbit plot of the hydrostatic bearing journal are studied.

Rotor dynamic studies

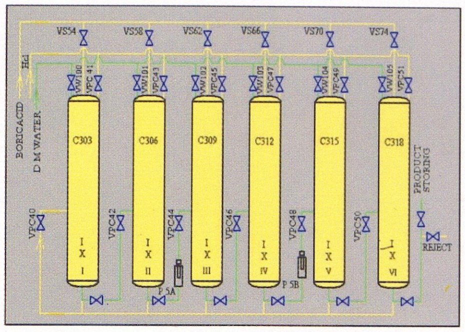
HYDRAULIC TESTING OF FUEL SUBASSEMBLIES

Fuel, blanket, absorber and other core subassemblies were tested in water in a hydraulic test rig to measure pressure drop in the subassemblies and assess the pressure drop and risk of cavitation in flow zoning orifice plates. To achieve the required flow zoning, orifice plates and other flow restrictions were tested with different combinations to select cavitation free flow gags. The experimental data were extrapolated to reactor conditions, using similitude parameters.

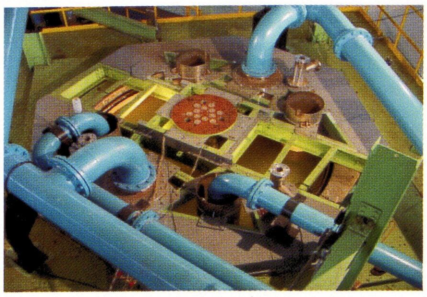
Cast Orifice Plates for FSA

BORON ENRICHMENT PLANT

Boron Carbide, enriched in ^{10}B isotope, is used in absorber rods, in fast reactors. The natural abundance of ^{10}B is 19%, whereas the absorber rod requires an enrichment of 65%. Ion exchange chromatography process is developed to produce enriched (65%) ^{10}B at the rate of 15 kg/a. R&D experiments are being carried out to study the effect of various parameters for improvement of the plant performance.

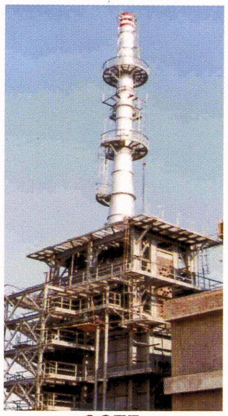


Boron Enrichment Process


Top view of pool hydraulics model

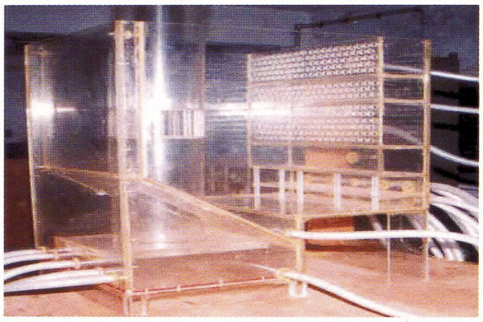
POOL HYDRAULICS

1/4th scale model of the Reactor Assembly is built to study the Reactor Pool Thermal Hydraulics. The facility is used to conduct various thermal hydraulic and flow induced vibration studies. They include studies such as optimum location of core thermocouples, thermal striping on control plug, inner vessel & thermal baffles, flow pattern and temperature profile in hot pool, free level fluctuation and gas entrainment.


SGTF

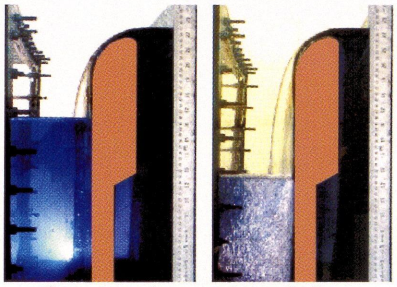
STEAM GENERATOR TEST FACILITY

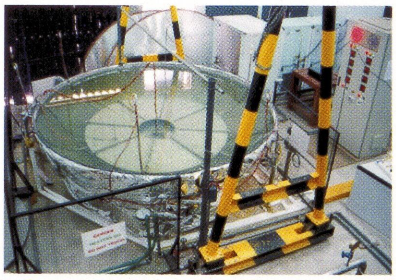
5.5 MWt Steam Generator Test Facility to verify the complete thermal hydraulic performance of an industrial size sodium heated steam generator simulating all conditions of operation including transients has been setup. The SG is instrumented to get data for validation of multi-dimensional thermal hydraulic codes. The steam conditions are typical of fast reactor (17 MPa, 765 K). The facility includes special instruments for detection of sodium–water reaction.


Hot pool model for gas entrainment studies

GAS ENTRAINMENT INVESTIGATIONS

The mechanisms of argon entrainment in hot pool sodium are investigated by experiments on scaled down models in water. Another major source of gas entrainment is due to flow over the crest of main vessel thermal baffle. The shape of the weir-crest, in association with fall height, is optimized experimentally to minimize the entrainment by way of ensuring that there is no flow separation.


Gas entrainment in weir flow

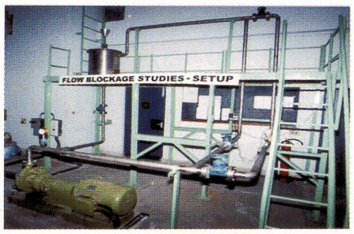

Core catcher experimental setup

HEAT TRANSFER IN CORE CATCHER

To study the heat transfer characteristics and for the design validation of core catcher, a 1/4th water model has been setup. In the experiments, axial and radial temperature profiles on core catcher plate, heat shield plate and fluid zone above and below cover plate have been obtained.

FLOW BLOCKAGE STUDY

An experimental water test loop has been installed to study the blockage phenomenon in fuel subassembly. The loop contains a 37-pin configuration PFBR test subassembly and a pump of capacity 30 m^3/h to circulate the water continuously. Based on the experimental results the blockage can be characterised as a function of location, time, debris size and the resultant hydraulic effects can also be predicted. The studies also help to estimate the Maximum Credible Blockage (MCB) that can be formed in a fuel subassembly.



Experimental setup for flow blockage studies

SODIUM FIRE

The overall experimental programme is planned to understand the pool / spray fire models and associated pressure rise in the respective buildings and further to demonstrate the fire extinguisher capabilitiy. The performance of leak collection trays have been carried out by pouring 30 kg of sodium at 823 K to optimise the sites of leak collections. The facility will also be used to develop suitable sodium resistant concrete to be used in PFBR.


Sodium fire

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