36 A REVIEW OF THE INDIAN FAST REACTOR PROGRAMME

S.C. CHETAL
Indira Gandhi Centre for Atomic Research,
Kalpakkam, India

Abstract

Fast Breeder Test Reactor (FBTR) in India is ready for restart. Satisfactory progress has been made in the design of Prototype Fast Breeder Reactor (PFBR). Conceptual design work for the important systems and components has been completed. Cost estimation is in progress. Detailed project report for the financial sanction is under completion stage and is planned to be submitted to the Government this year. Draft Safety criteria prepared by a sub-committee on behalf of the Regulatory Board have been discussed and will be issued shortly.

1.0 GENERAL

India's Gross National Product growth rate for the financial year ending March, 1989 is expected to be 9% following quick recovery in post drought year, while industrial production is expected to notch up a growth of nearly 10%. On the energy front, additional 4780 MWe capacity has been installed in the year 1988-89. The total electricity generated has increased to 221 billion units during the year 1988-89 compared to 201.7 billion units in 1987-88. The total installed electricity generation capacity in the country, as on end March, 1988 was 53.3 GWe, comprising of 66.1% fossil fuel based thermal, 31.6% hydel and 2.3% nuclear.

The major highlights on the nuclear front since the last review presented has been the commissioning of the Unit-I of 235 MWe Narora Atomic Power Station (NAPS) and the decision of the Government on the location of sites for 12 new nuclear power reactors with a cumulative capacity of 5940 MWe. This includes 6 units of 500 MWe, 4 units of 235 MWe and 2 VVER type Soviet reactors of 1000 MWe each. The two VVER reactors are in addition to the target of achieving 10,000 MWe of nuclear power generation capacity by the year 2000 using indigenously built 235 and 500 MWe PHWRs. The commissioning of the Unit-I of NAPS is an important milestone for Nuclear Power Corporation as the reactor is based on 'Swadeshi' (indigenous) design and is the first of the standardised 235 MWe units. This design is also being extrapolated for the 500 MWe units. Construction work on other PHWR units at Kakrapar and Kaiga is progressing satisfactorily. Advance procurement action has been initiated for six units of 500 MWe and four units of 235 Mwe. The Unit-2 of NAPS is expected to be commissioned by 1990.

An open debate on the relevance of nuclear power and safety of nuclear power stations has begun in the country. Directorate of Public Awareness has been created in the Nuclear Power Corporation to disseminate the information and dispel fears in the minds of the public regarding nuclear power.

2.0 RESOURCES FOR FAST REACTOR DEVELOPMENT

Financial outlay for IGCAR for the year ending March, 1989 is placed at 457 million Rupees ($ 29.3 million) while the proposed outlay for the coming financial year has been placed at 634 million Rupees. These efforts are to be a certain extent supplemented by other units of DAE. The total scientific, technical and auxiliary staff is about 2100. Total manpower presently employed for the design of PFBR is about 100.

3.0 FAST BREEDER TEST REACTOR

Steps for the normalisation of the reactor continued during the year following the fuel handling incident in May, 1987. This incident was described in the last review. The exercise of solving the bent guide tube problem led to the development of new techniques and remote toolings. The three techniques developed and utilised to assess the bend of the guide tube were optical inspection, ultrasonic air gauging and mechanical disc gauging. With the assessment of bend in the guide tube of the order of 325 mm, it was confirmed beyond doubt that the guide tube could be safely removed from the reactor by cutting just above the first set of equalising holes i.e. around 3 m below zero level. Remote in-situ cutting technique was developed for the purpose and included the cutting tool, a leaktight cup to prevent falling of chips in the reactor and arrangement for holding of the bottom piece. After carrying out several mock-ups, cutting and retrieval of the damaged guide tube was successfully accomplished. The damaged subassemblies were then retrieved with the help of special two fingers gripper for 27 subassemblies and a special hook for one severely bent subassembly. All these operations were carried out after extensive mock-ups.

The fuel handling incident has been reconstructed from the data obtained from inspection of the core and subassemblies retrieved therefrom. The incident seems to have been initiated by excessive friction between the gripper and its sealing in the charging flask, resulting in failure of the fingers to open. During the rotation of the plugs from periphery to the core centre, the fuel subassembly has protruded into the core causing damages to the heads of reflector subassemblies and bend in its own foot. Though the control logic forbids rotation of plugs under such a condition, the logic seems to have been inoperative. Subsequently the bent foot of the fuel subassembly has ejected out one of the peripherial subassemblies. The guide tube has a complex interaction with this ejected subassembly resulting in the bend of the guide tube and damages to the foot sections of the some adjacent subassemblies.

All the damaged components have since been replaced with new ones. All operations were carried out in strict compliance with safety requirement, with minimum exposure to operating personnel and no contamination of reactor sodium.

The normalisation of the reactor subsequent to fuel handling incident has provided useful engineering development information. Equally important is the lessson learnt by the operators. Modifications in the handling machine control logics and plug rotation have been carried out to prevent its recurrence.

Parallel to the above activities, commissioning of the remaining circuits have been carried out. The sodium flooding circuit was commissioned and 700 litres of sodium was injected into the reactor vessel to demonstrate the satisfactory working of the system. The hydrogen leak detection system and discharge circuit for the steam generator have been commissioned. The stage has been set for heating of the steam generator using auxiliary boiler and for filling sodium in the steam generators. During the commissioning of steam-water system, problems of excessive vibrations and jamming of shafts have been faced in some of the pumps. The problems have been resolved and the entire system is now ready for operation.

The strength of auxiliary neutron source required for the restart of the reactor and its subsequent operations was estimated and the shielding requirements to transport the activated antimony pin required for this neutron source were also evaluated. All the prerequisites for the reaction rate measurements to be carried out when the reactor is restarted and taken to power operation, such as loading the activation foils in the capsules, sealing them and loading the capsules in the special experimental subassemblies have been completed.

After completion of the requirements from the Regulatory Authorities regarding relicencing of operating staff and approval of procedures for work on pile, the reactor will be restarted and operated at a power of 500 KW for reactor physics experiments. The reactor is expected to start by April, 1989.

4.0 PROTOTYPE FAST BREEDER REACTOR

Satisfactory progress has been made in the design of Prototype Fast Breeder Reactor (PFBR). Conceptual design work for the important systems and components has been completed. Cost estimation is in progress. Detailed Project Report for the financial sanction is under completion stage and is planned to be submitted to the Government this year. Draft safety criteria prepared by a sub-committee on behalf of the Regulatorty Board have been discussed and will be issued shortly. Regulatory Board has constituted Project Design Safety Committee. The design review activities as a consequence of the formation of this committee have begun.

The design is being detailed out on the basis of following parameters and design features.

<u>General</u>

Reactor thermal output	- 1210 MWt
Gross electrical output	- 500 MWe
Plant life	- 30 Yrs
Design capacity factor	- 75 %
Primary circuit concept	- Pool
Reactor coolant temp.	- 653/803 K

General (cont.)

Secondary coolant temp.	- 623/778 K
Steam conditions as turbine	- 16.6 MPa/753 K
Number of primary pumps	- 4
Number of secondary loops	- 4
Proposed Site	- Kalpakkam

Core

Fuel - mixed oxide for the first few cores

metallic fuel to be studied as long term solution

Type of core - Homogenous with two enrichment zones

Number of fuel subassemblies	- 181 (91 + 90)
Max. fuel burn-up initial/ target	- 50,000/100,000 MWD/T
Number of pins in subassembly	- 217
Pin outside diameter	- 6.6 mm

Reactor Assembly

Main vessel outside diameter	- 14 m
Main vessel height	- 12.45 m
Inner vessel	- Single
Top shield of roof slab	- Warm (393 K)
Top shield cooling	- Nitrogen

Circuits

Primary sodium purification	- Ex-vessel
Number of IHXs per secondary circuit	- 2
IHX design	- Straight tube with primary sodium on shell side
Primary Pump	- 1.8 t/s, 0.68 MPa head developed at full power. Non-return valve provided
Secondary circuit layout	- Pump in cold leg at the lowest elevation steam generators without cover gas, surge tank provided on upstream of steam generators.
Number of steam generators per secondary circuit	- 3
Steam generator design	- Once through, sodium reheat, vertical straight tube with expansion bend on each tube. Operation with 2/3 modules foreseen
Decay heat removal (DHR) circuit design	- 2 independent circuits One is normal heat transport path provided with no emergency supply. Safety grade DHR consists of 4 independent circuits with heat rejected from pool to ambient air based on natural convection. Heat capacity of each circuit 8 MWt at 803 K hot pool temperature. 2 circuits adequate to keep maximum temperature close to normal operating state.

Fuel and Handling Components

In-vessel handling	- Two rotatable plugs with a single in-vessel transfer machine using straight pull mechanism
Ex-vessel handling	- Rotatable shielded leg with single transfer pot
Spent fuel storage	- Under water

Balance of Plant

Turbine	- Single. Standard 500 MWe set as used for fossil fired stations without any modification
Boiler feed pump	- 2 - 50% turbo driven 1 - 50% motor driven
Capacity of diesel generators	- 4 units with each unit of 2 MW capable of meeting 50% demand

4.1.1 Reactor Physics and Shielding

Feasibility of locating a neutron guide tube in the core to increase the neutron flux levels at the neutron detector location at the bottom of reactor vessel was investigated.

Calculations of dynamic power, flow and inlet temperature reactivity coefficients were completed and an assessment of the stability of PFBR using these coefficients has been made. Maximum permissible control rod speeds were evaluated based on the maximum reactivity rates permitted during startup and raising of power to meet the safety requirements in such operations. Inherent safety analysis of PFBR with different fuels and with GEM assemblies was carried out.

Shielding requirements of the top shield, reactor vault, complementary shielding for various openings, annular gaps in the roof slab due to penetrations and step regions were evaluated for specified dose criteria.

The neutron streaming through the upper axial shields of fuel subassemblies was estimated using two-dimensional transport theory code. Corrosion products activation, release and subsequent deposition on IHX and pump were estimated. Contamination in Reactor Containment Building (RCB) due to leakage of cover gas during normal operation and fuel pin failure conditions were estimated. A model for the storage and recycling of argon cover gas with purification under fuel pin failure conditions was formulated and studied. Site boundary doses due to release of activity through stack were computed for various scenarios.

4.1.2 Reactor Assembly

Further progress has been made towards completion of remaining conceptual design work of reactor assembly. Conceptual design work of auxiliary grid plate, pipe from pump to grid plate, top shield cooling system, primary and secondary shut down system, and failed fuel localisation module has been completed. General assembly drawings for core subassemblies and important reactor assembly components have been prepared. 2 additional problems defined in the IAEA-CRP on code-code verification have been solved using inhouse core mechanics codes.

THYC-2D code has been improved to include turbulence effects and transient capability. The inner vessel distribution has been determined as a function of the location of lower shell redan junction with respect to core. Temperature distribution in the control plug under steady state has been evaluated. Improving the code for 3D analysis is being continued.

Significant further progress has been made towards the structural analysis of reactor assembly main components. Core disruptive accident analysis carried out by in-house FUSTIN code has demonstrated the ability of the primary containment to withstand energy release of 200 MJ. The code is being improved to include internal components and porous model representation of non-axisymmetric components. The code has been validated against the international bench mark problem. Buckling analysis of the main vessel bottom torispherical dished head has demonstrated adequate factor of safety against buckling under internal pressure.

4.1.3 Sodium Circuits

Conceptual design of sodium main and associated important systems and components has been completed. P&I (Process and Instrumentation) diagrams and operation notes of the sodium main and associated circuits have been detailed out. Further progress has been made in the design of primary sodium pump, IHX, steam generators and sodium main piping. Tender drawings for manufacure of IHX and evaporator and reheater of steam generator for technology development have been prepared. Piping flexibility analysis for secondary sodium main and decay heat removal circuit has been completed. Design of pump hydrostatic bearing has been completed, and conceptual design of mounting the flywheel on the pump shaft and spherical seating arrangement for the shaft has been carried out.

4.1.4 Fuel and Component Handling

Design of straight pull in-vessel transfer machine has been further detailed out. General assembly and main subassemblies drawings have been prepared. Conceptual design of inclined fuel translift, flasks for handling of primary sodium pump, IHX and CRDMs, and layout of fresh fuel storage area has been completed.

4.1.5 Materials

The materials of construction for NSSS have been selected and are as follows:

Component	Material of Construction
Core subassemblies (Clad and wrapper)	20% CW 15Ni-14Cr+Mo+Ti+Si (as per ASTM A 771) 9-12% Cr-Mo steel to be developed as long term solution for wrapper
Main vessel, Inner vessel, Core cover plate, Grid plate, IHX, Main sodium piping	SS 316 LN
Safety vessel, Sodium pumps, Tanks, Auxiliary sodium circuits	SS 304 LN
Steam Generator	
Evaporator	2.25Cr-1Mo
Superheater & Reheater	9Cr-1Mo

4.1.6 Instrumentation and Control

Control schemes have been prepared for primary and secondary sodium systems and other associated systems. Preliminary schemes have been worked out for the neutronic instrumentation, safety logic and clad rupture detection system.

4.2 Design of BOP

Further progress was made with the assistance of consultants in the design of power plant, control and electrical systems, buildings and structures and auxiliary systems for the preparation of Detailed Project Report. Structural design of reactor building, steam generator buildings, turbine building, pump house, decontamination building, fuel handling building and control building etc. were carried out. Design of steam, feed water, condensate systems, condenser cooling water system, auxiliary systems such as airconditioning and ventilation system, compressed air, service water, domestic water and fire water systems were finalised with the preparation of P&I diagrams. Piping and ducting layout was also completed for the steam water system and ventilation systems. Interconnection of PFBR to the grid was finalised. Switching scheme and switchyard layout have been finalised. Station power supply schems have been designed. Specifications for major equipment have been finalised. Preparation of write-up for Preliminary Safety Analysis Report is in progress.

5.0 ENGINEERING DEVELOPMENT FOR PFBR

5.1 Pool Hydraulic Studies

Local velocity was measured in 1/24 scale perspex model using propeller anemometer. Magnitude of velocity in radial and circumferential directions has been mapped in hot pool. Direction of flow in hot pool was also mapped using dye, weightless thread and air bubbles. It is planned to measure local velocity around the periphery of IHX windows using 3 mm diameter miniature propeller anemometers.

5.2 Surge Tank

Experiments have been performed using 1/38, 1/32 and 1/22 scale geometrically similar transparent glass models with bottom inlet/side outlet configuration. The minimum height (H) required to avoid gas entrainment was found and a correlation of the form $H/D = 1/4\ Fr^{0.5}$ was found to be fitting within $\pm$ 5%. Plate type internal structures were also found to be the most effective form of reducing the above height, with diameter and location of the plates playing an important role.

5.3 Natural Convection in Cover Gas

An experiment was conducted to study the effectiveness of horizontal gap type convection barrier for prevention of natural convection and sodium deposition in vertical annuli. It was observed that an horizontal gap of 5 mm was almost completely closed by the condensing sodium vapour with 100 Kg being the hold-up of sodium in the test vessel at 803 K. Further work is being planned including the study of dip seal type convection barrier.

5.4 Large Component Test Rig (LCTR)

The design of the rig for testing the full scale reactor components has been completed and two sodium storage tanks of 50 m^3 capacity each have been fabricated. To study the heat and mass transfer related problems connected with roof slab of PFBR, an experimental test vessel was designed and its fabrication taken up. The 3 m diameter stainless steel vessel will have a top structure similar to the roof slab. 43 m high Engineering Hall to house the above rig is under construction.

5.5 Hydraulics of IHX

The manufacture of 1/6th sector model of the IHX was completed. This was erected in the 300 mm diameter pipe water loop. Experiments are soon to be commenced to study the shell side pressure drop, flow distribution and corss-flow induced tube bundle vibration.

Further measurements on the three tube model were carried out to establish mode shapes and response to external excitation using electrodynamic shaker. Results were in good agreement with predictions.

5.6 Sodium Pumps

Indigenously built 50 m^3/h centrifugal sodium pump was tested in sodium and its performance was found to be satisfactory. Recognising the requirements of Flat Linear Induction Pumps (FLIP) for auxiliary circuits of PFBR, a programme on the FLIP development was commenced a few years ago. More recently a FLIP with a capacity to deliver 20 m^3/h at 0.5 MPa pressure and at sodium temperature of 803 K was constructed indigenously. The pump was then tested in a sodium loop at temperatures ranging from 473 to 773 K to evaluate its performance characteristics. The measured pressure developed by the pump was lower than the theoretically predicted value and this short-fall in performance is being analysed. This pump, however, successfully passed the endurance for 1000 h at a sodium temperature of 673 K and 500 h at 773 K.

5.7 Instrumentation

Design and assembly of a miniature (12 mm dia) eddy current flowmeter sensor was completed. This type of sensors are intended for in-core application.

Leak simulation experiments were carried out, as part of efforts to develop acoustic leak detection techniques for steam generator. Initial low pressure experiments at 1.2 MPa with argon/helium leaking through a micro hole into static water were completed. Signal level, frequency spectrum and attenuation measurements were carried out. Efforts to localise leak by triangulation technique were not fruitful. High pressure gas injection experiments were commenced recently.

In the context of the fuel handling incident, an ultrasonic under sodium viewer was developed and deployed as a 'sweep arm' in FBTR reactor vessel to confirm that no sub-assembly was projecting from its original position under the core cover plate mechanism, which would then enable safe plug rotation. After successfully ensuring the same, the viewer's imaging/viewing capability was also demonstrated by obtaining an image of the lowermost part of the mechanism over a 150 mm height.

6.0 METALLURGY AND MATERIALS

6.1 Materials Development

6.1.1 Creep Testing of SS 316 Welds

Creep rupture tests were carried out on SS 316 weld metal at 923 K and 873 K in stress ranges of 96-196 MPa and 120-275 MPa respectively, with rupture lives upto 5000 h. The rupture strengths were nearly 20% lower and the rupture ductilities were nearly half of that of the base material. At 923 K, rupture ductility was found to decline rapidly after 100 h due to almost complete decomposition of delta-ferrite into sigma and carbide phases. At 873 K, however, the transformation was sluggish so that the rupture ductilities were found to be almost constant for rupture lives upto 3750 h. At all the test conditions the weld metal had strengths lower than the ASME allowable stresses for base metals.

6.1.2 Creep Deformation Studies on 2.25Cr-1Mo Steel Weldments

Creep tests of 2.25Cr-1Mo base metal, weld metal and welded joints at 773 and 823 K in the stress range 130-300 MPa had been carried out. Welded joints fractured in the soft intercritical region of heat affected zones, and showed inferior creep rupture life and ductility, and also inferior minimum creep rate than either the base or the weld metal. Presence of soft intercritical region increased the creep rate in the hard weld zone over that expected from the all weld metal data, giving rise to "off loading" effect. Modified White and LeMay model by Yoshio Monma et al was found to be applicable for predicting the minimum creep rate and rupture life of welded joint from the corresponding base metal and weld metal properties.

SEM studies of creep cavities of interrupted and fractured base metal crept specimens showed a large number of cavities, associated with sulphide, carbide and silicate particles, displaying a range in size and morphology and heterogenously distributed on the prior austenite grain boundaries; the cavity density (nos/area) increased with creep exposure time and applied stress.

Base metal at 823 K in the stress range 150-240 MPa was found to be notch strengthened, the rupture life increased by a factor of 3 to 42, the strengthening factor increasing with the applied stress.

6.1.3 Fatigue Design Curves for 2.25Cr-1Mo

Low cycle fatigue properties of 2.25Cr-1Mo steel in normalised and tempered (N+T) condition has been evaluated at 773 K and the Coffin-Manson plot was found to exhibit a two slope behaviour. Fatigue design curves have been formulated for 2.25Cr-1Mo steel under N+T condition for various temperature ranges namely $T < 700$ K, $700 \text{ K} < T < 823$ K and $823 < T < 873$ K using the data from literature. Creep fatigue interaction diagram has also been computed; due to enhanced oxidation effects, linear damage creep fatigue rule $\sum(N/N_f) + \sum(t/t_R) = D$ has to be modified with $D < 1$.

6.1.4 Mechanical Properties of 9Cr-1Mo

Evaluation of tensile, creep and low cycle fatigue properties of thick section forged 9Cr-1Mo steel in quenched and tempered (Q+T) and in simulated post weld heat treated (SPWHT) condition have been carried out. Yield strength of both Q+T and SPWHT conditions lie above ISO data for thin section material in Q+T condition, while yield strength of the thermally aged SPWHT material is inferior to the ISO values. At 873 K, for both Q+T and SPWHT materials, double logarithomic plots of rupture life vs stress exhibited two slope behaviour; also the rupture lives in the SPWHT condition were found to be inferior to the corresponding lives for Q+T condition. Over the temperature range 723-793 K, thick section material in SPWHT condition exhibited lower fatigue life compared to thin section material. Fatigue design curves were formulated using the above data over the temperature range 723-793 K. Also using the experimental data available from hold time tests, linear damage rule namely $\sum(N/N_f) + \sum(t/t_R) = D = 1$ has been proposed for 9Cr-1Mo steel under creep-fatigue conditions.

6.1.5 Development of Transition Joint for Steam Generator

The steam generator circuit of PFBR necessitates a joint between 2.25Cr-1Mo steel and austenitic stainless steel. To improve the service life of this joint, a trimetallic transition joint with an intermediate piece of Alloy 800 is being developed. This development programme includes metallurgical studies on this trimetallic joint to select proper welding consumable and optimum PWHT, and screening the parameters through an accelerated test. For welding the 304 SS/Alloy 800 joint, the study has revealed that 16-8-2 consumable would be preferable, because of its lower microfissuring tendency and reduced mismatch in the coefficient of thermal expansion. It was also found that lowering of the PWHT temperature improves the mechanical properties and the microstructural condition of the Alloy 800/2.25Cr-1Mo steel joint.

6.1.6 In-Sodium Test Facility

Construction works have commenced for setting up the facilities for creep, fatigue and creep-fatigue interaction testing of fast reactor materials and their welds in flowing sodium. Major loops components, structural materials etc. are in various stages of procurement; the facility is expected to be fully operational by mid-1992.

6.2 Metallurgy

6.2.1 Stress Corrosion Cracking Studies

Specimens of 9Cr-1Mo steel (which is a candidate material for the sodium-to-air heat exchanger) in hardened, normalised as well as normalised and tempered conditions were subjected to stress corrosion cracking test in NaCl solution. It was found that the tempered steel specimens did not fail in contrast to hardened or normalised specimens. Hence it was concluded that for use in marine environment the steel should be in normalised and tempered condition.

6.2.2 Compatibility Studies on Antimony Oxides with Stainless Steel

Compatibility studies of Sb_2O_3 and Sb_2O_4 with AISI type 316 stainless steel have been completed for the purpose of suggesting an alternate solid oxide containing antimony, to the low melting (mp 930 K) valentinite (Sb_2O_3) use in neutron start up facility of FBRs. An assessment of the suitability of $MgSb_2O_4$ in lieu of SB_2O_4 has been made on the basis of thermodynamic data.

6.2.3 Thermodynamic Activities of Metallic Elements in Stainless Steel

The thermodynamic activity of Ni in AISI type 316 LN stainless steel was also determined using meta stable emf technique earlier developed in this laboratory. With this it was possible to compare the activities of all major metals in the three types of stainless steels namely 304, 316 and 316 LN for an assessment of the thermodynamic driving force for the formation of corrosion products on these stainless steels.

6.3 Post Irradiation Examination

Two hot cells were commissioned temporarily and an antimony pin, forming part of a neutron source assembly, was assembled and welded remotely. The stringent quality requirements for the weld were met. Experimental capsules for irradiation in FBTR to determine the reaction rates have been assembled and welded using specially developed welding jigs and fixtures. Construction of a separate set of lead shielded hot cells for carrying out mechanical property tests on irradiated structural material samples is well underway. Construction of the building to house these cells has been nearly completed. Work related to installation of a 4000 m^3/h nitrogen gas recirculation and purification system is nearing completion.

6.4 Non-Destructive Examination

6.4.1 Eddy Current Testing

The finite element code developed earlier for prediction of signal trajectories has been standardised to do various functions such as optimised probe design, optimisation of frequency of operation, and prediction of magnetic flux line contours. Eddy current testing (ECT) was carried out using magnetic saturation method on a 2.25Cr-1Mo tube with artificial defects such as through hole and flat bottomed holes of various diameters and depths. It was found that magnetic saturation enhances sensitivity of ECT to a very great extent.

6.4.2 Ultrasonic Testing

Improved reliability in detection of defects like voids and linear defects has been obtained in ultrasonic testing of thick stainless steel weldments by application of signal analysis techniques such as cluster analysis and demodulated auto correlation.

6.4.3 Accoustic Emission

Acoustic emission (AE) studies during plastic deformation of AISI 316 stainless steels have been carried out with grain size and strain rate as the variables. The AE signals obtained during the experiments were subjected to a pattern recognition routine which compares the signals with a reference signal and identifies specific features of the AE signals. A frequency shift was observed for different grain sizes. The signals were also subjected to cross power analysis and features like centroid and average spectral energy were identified. AE studies have also been carried out on 2.25Cr-1Mo steel during tensile deformation (till fracture). Analysis in time and frequency domains was done for two microstructures viz. ferrite-pearlite and bainite.

7.0 RADIOCHEMISTRY

7.1 Sodium Chemistry

7.1.1 Studies on Electrochemical Hydrogen meter

$CaCl_2$ - CaH_2 based electrochemical hydrogen meter earlier developed in this laboratory was used for studying the reaction of sodium with rust (FeOOH) and Terrso oil used in the mechanical seals of centrigufal sodium pumps in FBTR) in the temperature range of 623 to 748 K. The reaction between rust and sodium is observed to be slow at 623 K while at 723 K it is found to be rapid. Similarly, the reaction betwen oil and sodium is slow at 623 K but above 673 K, it proceeds fast. Gas chromatographic analysis of the argon cover gas over sodium showed methane to be the major gaseous product formed in sodium-oil reactions. The formation of methane was found to obey parabolic rate law indicating the reaction to be diffusioin controlled. The response of the electrochemical hydrogen meter to additions of rust as well as oil were found to be rapid, qualifying the meter for detection of ingress of these hydrogen bearing compounds into sodium.

7.1.2 Studies on Na-M-O Systems

There is an on-going programme on studying phase equilibria and thermochemistry in Na-M-O system where M stands for the metallic constituents of stainless steels. As part of this programme the phase diagram of the Na-Mo-O system was already worked out and the discrepancies in the threshold oxygen levels in sodium for the formation of

$NaCrO_2$ in sodium-steel systems have been cleared. At present studies are going on on the Na-Fe-O, Na-Mn-O, Na-W-O systems.

In the Na-Fe-O system it has been shown that at low temperatures (below 630 K) sodium and iron exist in equilibrium with Na_2O. At high temperatures (above 760 K) the ternary phase in equilibrium with metals is established as Na_4FeO_3. Studies using various techniques such as DTA, in-sodium equilibriation and oxygen potential measurements indicate a discrepancy in the intermediate temperature region. This is being investigated further.

7.2 Fuel Chemistry

7.2.1 Thermal Analysis Studies

Ignition behaviour of uranium dicarbide, prepared by carbothermic reduction, was studied employing the thermo-analytical method. The ignition temperature was measured under various experimental and sample conditions. Dependence of the ignition temperature on factors like heating rate, sample size, physical form of sample, presence of moisture etc was also analysed.

7.2.2 Carbon Potential Studies on Carbides of Uranium and Uranium-Plutonium Mixed Carbide

As part of a study aimed at investigating the possibility of clad carburisation in FBTR fuel, the carbon potential of the mixed carbide of uranium and plutonium was measured using a novel technique developed in this laboratory. Initial experiments were carred out using UC, prepared by the carbothermic reduction of UO_2. This pellet was equilibrated in sodium and the carbon potential was measured using the carbon meter. By successive gettering, using annealed stainless steel foils, enough carbon was removed to bring the system to a pseudo-binary phase field of $UC_{1-x}O_x - UO_2$. The measured carbon potential was found to be in good agreement with theoretically predicted values from the free energy data, in the temperature range of 850-960 K.

The carbon potential of the mixed carbide of FBTR composition was found to be in the range of -76.03 kJ mol^{-1} at 847 K to -69.28 kJ mol^{-1} at 913 K. From an earlier measurement of the carbon activity of 18/8 austenitic steel, it became evident that the fuel is not likely to carburize the clad in an isothermal condition.

7.3 Process Chemisitry

7.3.1 Head-end Steps for Carbide Fuel Reprocessing

Studies were carried out on the conversion of carbide to nitride as a possible head end step for FBTR fuel reprocessing. The studies indicated that the carbide can be quantitatively converted to nitride which can be dissolved in the nitric acid. The carbon precipitates out as graphite and can be filtered out. Quantitative studies on the carbon content of the solution obtained indicated that the solution contains very little dissolved carbon. The possibility of adapting this method for fuel pins is being looked into.

8.0 FUEL REPROCESSING

8.1 Process Development

8.1.1 Dissolution

Mixed carbide dissolution gained further progress marked by parameteric studies on various aspects. Scale of operation of dissolution of unirradiated mixed carbide pin was stepped up. Experiments relating to electro-oxidative dissolution of mixed carbide was carried out. Methods to reduce the time of dissolution, special coating for electrode to reduce the corrosion rate, and design of single limb dissolver are the current areas of study.

8.1.2 Electrolytic Partitioning

Experiments were conducted to study effect of process parameters with in-situ electrolytic partitioning of U+Pu in mixer settlers. Equipment design for plant scale use have been finalised.

8.2 U-233 Separation Facility

Facility to reprocess irradiated thorium metal and oxide rods for separation of U-233 was commissioned. In this facility, rods irradiated in the CIRUS reactor at BARC were reprocessed successfully. This facility has served to qualify newly developed equipment and plant concepts for remote maintenance in a radioactive environment and to train operating staff. The experience gained would be utilised in setting up a reprocessing plant for treating FBTR fuel. U-233 separated will be used for fuel development work for FBTR, and for fabrication of fuel elements for the Neutron Source Facility being set up at IGCAR.

9.0 SAFETY

9.1 Sodium Fires and Aerosols

In the area of safety studies on sodium systems, release of radioactive manganese and cesium from fires of contaminated sodium was studied. Experiments were carried out in a stainless steel vessel of 600 l capacity with about 100 g of sodium premixed with 1-10 microcurie of radioactivity. Aerosol to sodium ratios of specific activity of the isotopes were measured by radiochemical analysis to characterise the release behaviour. The experiments with respect to cesium-134 are being scaled up in view of the higher specific activity ratios associated. A stainless steel vessel provided with knife edged flanges to withstand the high temperature of sodium fire has been readied to carryout fires with upto 1 Kg of sodium. Release behaviour of sodium oxide aerosols from sodium fires was also studied in small scale experiments as a function of oxygen concentration, air flow rate and pool depth. Sodium oxide aerosol release rates appeared to be insensitive to changes in ambient relative humidity, while increasing pool depth decreased the release rate significantly. A test facility is being set up to carryout studies on large scale sodium fires involving several hundred kilograms of sodium, so as to determine pressure/

temperature evolution and aerosol releases as also to evaluate materials/methods appropriate for fighting large scale sodium fires.

9.2 Sodium-Concrete Interaction

To investigate the hazards associated with sodium-concrete interactions, a small scale experimental assembly was set up and a trial run was carried out. In the experiment conducted, limestone concrete block with surface area about 190 cm^2 was contacted in a test chamber (volume 2201) in argon atmosphere with hot sodium (temperature 433 K) dumped from a melt-pot. In the contact duration of about 7 minutes before sodium solidifcation, the evolution of gases was below the detection limits of 1% carbon dioxide and 1% relative humidity. No visible signs of spallation or cracking were seen on the concrete surface.

9.3 Exploding Wire Experiments

In the context of studies releated to transmission of pressure pulses through simulated systems of reactor components/structures, exploding wire technique has been used for generation of standard pressure pulses. In experiments conducted with long (20 cm) resistive wires, peak pressure decay with distance from the wire was studied as also the pressure variation along the length, close to the wire. The cylindrical shock pulses generated were found to have good reproducibility. Unlike the regular shock pulse shapes observed with copper wires, the nichrome wire explosions yielded compound peaks. The discharge time constant for an exploding wire of a given type of material appeared to have no significant influence on the pressure peak and pulse width. In extension of these studies, a 100 kJ condenser bank is now being installed. Design work has been carried out on related items like triggered spark gap switches, slow and fast energy dump devices, condenser racks and PC based control system. Procurement and installation activities are underway.

SUMMARY

The Indian Fast Reactor Programme is now focussed on FBTR and PFBR. The problems faced in FBTR due to fuel handling incident in 1987 have been overcome and the reactor will be made critical by April, 1989. 10 MWt power operation is expected by the year 1989. Detailed Project Report for PFBR would be submitted this year.