SRI	Stanford Research Institute, Stanford, Cal., USA
Super Phénix:	French demonstration breeder power plant at Creys-Malville
TNO	Netherlands industrial research organization, Apeldoorn
TOP	Transient Overpower
TREAT	Transient Reactor Test Facility, Idaho, USA
UKAEA	United Kingdom Atomic Energy Authority, London
ULK	Versuchsanlage für Umluftkühlung (Reventing air cooling test facility), Karlsruhe
UNC	United Nuclear Corp., USA
VEC	Variable Energy Cyclotron, Harwell
VIC	"Variable Irradiation Conditions" loop in BR 2
ZEBRA	Zero energy facility at Winfrith, U. K.

REVIEW OF FAST REACTOR ACTIVITIES IN INDIA

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1.0 Introduction

It may be recalled that the 1st stage of India's nuclear energy programme is based on natural uranium fuelled heavy water reactors. Fast breeders have been considered only as a second stage to utilise the plutonium produced by the heavy water reactors. Attitudes and public reactions to nuclear energy are therefore governed by the developments in areas related to heavy water reactors rather than the developments in the field of fast breeder reactors themselves. Although some measure of success has been achieved by the country in increasing the gross national product and reducing the rate of inflation, prices of various commodities have recorded an increase and the same has resulted in an increase in the capital cost of heavy water reactors. But inspite of these increases, these reactors have maintained their competitiveness vis-a-vis coal-fired power stations in areas away from coal fields. On the other hand, there are increased stresses in the transportation system due to increased industrial activity and this is making transportation of increasing quantities of coal to remote parts of the country more and more difficult. Hence, need for nuclear energy is being felt more and more acutely. In this sense, climate for nuclear energy is quite good. While the work on existing projects for generation of electricity has recorded progress, a twin reactor station at Kakrapar in Gujarath State on the west coast of India has been announced and the work has been initiated. Moreover, the current five-year plan provides for start of work on four more reactors of 235 MWe capacity in the next 3 years. A working group has also been formed to evolve a project report complete with cost estimates for a 500 MWe proto-type fast breeder reactor to be made

operational in mid nineties. The group has made significant progress and generated enthusiasm amongst scientists and engineers working on the system.

2.0 Financial Outlay

Financial outlay for the year ending 31-3-'82 has been 119.4 x 10^6 Rupees while the same figure for the previous year has been 111.7 x 10^6 Rupees. Manpower deployed has also remained at around the last year's level. In effect, the level of activity has been maintained at the previous year's. These efforts are supplemented by contributions from Bhabha Atomic Research Centre and other units of Department of Atomic Energy like Nuclear Fuel Complex and Electronics Corporation of India Limited at Hyderabad.

3.1 Fast Breeder Test Reactor (FBTR)

The reactor construction has made further progress during the course of the year. However, due to delays in the construction and escalations in the prices, cost of the project has gone up and is now placed around 685 x 10^6 Rupees excluding fuel. As regards physical progress, the civil works have been completed in 1980 itself and various components were getting ready in the works of the manufacturers. The year 1981 was therefore devoted to tests in the manufacturers' works and installation at site.

Reactor vessel, rotating plugs which provide the top closure and access for fuel handling and control have been assembled in the manufacturers' works to confirm machining and carry out minor adjustments. This was necessary as reactor vessel and rotating plugs excluding the control plug was manufactured by one supplier, control plug was manufactured by another manufacturer and the grid plates were ordered from France.

Fuel handling machines have been assembled in the reactor containment building and their operations verified. These will be further tested on the reactor itself before criticality.

Successful testing of primary sodium pumps in a water loop was reported earlier. During 1981, secondary sodium pump has also been tested successfully in water. Intermediate Heat Exchangers are nearing completion. Steam generator fabrication has made good progress. Manufacturing difficulties encountered in attaining the specified quality of workmanship have been overcome.

Electrical power distribution is completed. All cables for control and instrumentation have been laid. Piping work is in progress, control room panels have been installed and are awaiting connections as basic sensors have not been installed.

Turbo-alternator set has been delivered to the site and its installation is nearing completion.

Barring unforeseen difficulties, we should attain criticality in the next two to three years.

3.2 Design Studies for 500 MWe Proto-type Fast Breeder Reactor (PFBR)

As FBTR is nearing completion, design studies for a 500 MWe proto-type fast breeder reactor have been initiated. A report is to be submitted to the Government bringing out the main features, manpower needs, time-schedule and costs. This work has made good progress. As envisaged to-day, PFBR will have the following features :

Reactor type	: Sodium-cooled, pool-type
Power	: 1250 MW(t) i.e., 500 MW(e)
Reference fuel	: Mixed $\frac{\text{oxide}}{\text{carbide}}$ of plutonium & uranium
Primary & secondary sodium loops	: 4 each
Steam conditions	: 165 atm; 480°C with reheat to 480°C
Control rods	: B_4C
Steam Generators	: Once-through, modular units Three modules per loop Each module to consist of one economiser-evaporator; and one superheater, one re-superheater heated by sodium.

3.3 Fast Reactor Physics Studies

Nuclear Data Evaluation, Processing and Testing :

The development of the nuclear data processing code RAMBHA was completed and it has been tested for typical materials from ENDF/BIV files to generate multigroup data sets.

A set of routines to generate self-shielded cross sections using multilevel formalism in the resonance region were commissioned and the self-shielding factors so calculated for Pu^{239}, U^{235}, Fe and Na were compared with the measured values reported by the French and Soviet teams.

The processing code for coupled neutron-gamma data libraries, MARS is in the final stages of adaptation to generate data for shield design studies.

A National Workshop on Nuclear Data Evaluation, Processing and Testing was organised at the centre and the needs and status of nuclear data work required for our fast reactor programme was discussed among different groups working in these areas.

Computational Methods and Code Development :

A two-dimensional R-Z code based on analytical solution in one dimension (Z) and finite difference method in the other (R) is being written for rapid flux and criticality calculations.

The optimisation code using linear programming was extended to study the effect of control rod movement and power constraints on the fuel management schemes.

Analysis of Experiments for Validation of Methods and Data :

Heterogeneity calculations for K_{eff} for the assembly ZPR-6-7, using the code system RECRHGM-MUDE which contain improved formulations for the treatment of heterogeneity were made along with central reactivity worth of some materials.

Analysis of a carbide fuelled assembly in SNEAK was made to calculate K_{eff} and reaction rates using our codes and data libraries. Our results compare very well with the experiments as well as those obtained by Karlsruhe group.

An assessment of uncertainties in core physics parameters predicted for a Proto-type Fast Breeder Reactor (PFBR) of 500 MWe capacity, in the absence of any critical experiments programme, was made.

Reactor Design and Analysis :

The following calculations for the physics design of PFBR were made -

- Neutronics parameters for oxide, carbide and advanced oxide and carbide fuels;
- Two dimensional perturbation calculations for reactivity worth and reactivity co-efficients;
- Control rod worth, location and numbers for B_4C and Europium oxide materials;
- Sodium void and Doppler reactivity worth for oxide and carbide fuels.

A study of the effect of heterogeneity and group collapsing on the sodium void and Doppler effect of PFBR was made.

Kinetics and Safety Analysis :

The work on the formulation of dynamic feedback transfer function analysis was completed and the stability analysis code DPCOF was commissioned.

Validation of our predisassembly accident analysis code PREDIS is being made with a LOFA benchmark calculation suggested by NEACRP.

Energy release calculations in an HCDA and efficiencies in an FCI were estimated for PFBR using oxide or carbide fuel.

Improvements in the FCI calculations using a finite difference model and an improved heat transfer model are being incorporated in the code being developed for these studies.

A review of the current trends in the study of equation of state for fast reactor materials was completed and efforts to derive simpler and more convenient equations of state using Van der Waals approach are being made.

Investigations made on the sodium boiling detection by noise analysis method were presented in a paper at the SMORN-III meeting at Tokyo in October 1981.

Radiation Transport and Shielding :

Radial and axial shield calculations for PFBR were carried out using different configurations of shield materials to reduce the secondary sodium activation and fluence levels to permissible values. Activation of IHX and pumps and the neutron flux levels at detector locations were also estimated.

Analytical study of bending shield problem was carried out using exponential transform coupled with angular biassing of source angle and scattering angle Kernels in Monte Carlo methods.

Two computer codes DUST and PLACID developed here for studying neutron and gamma streaming through annular and square ducts and another improved DTF code for one dimensional transport calculations were contributed to the code centres NESC at ANL and RSIC at ORNL.

3.4 Metallurgical Activities (in support of Fast Breeder Reactor Programme)

In the metallurgy programme, the generation of metallurgical data (mechanical and corrosion properties and microstructural characterisation) on the materials used in FBTR continued to be a major activity.

During the year, creep rupture tests on the three different supplies of FBTR clad tubes have been carried out at 600, 650 and 700°C for establishing the applicable creep laws relating steady state creep rate to stress.

Eddy current testing has been standardised and employed as a technique complementary to ultrasonic testing for characterising FBTR clad tubes for defects. Through destructive metallographic studies, the types of defects that are sensitive and insensitive to the two different techniques have also been identified.

On the basic side, as part of a programme aimed at correlating properties with microstructural features in materials, the creep rupture properties of type 316 stainless steel have been extensively investigated as a function of grain size. Similarly the tensile properties of Nimonic PE16 superalloy have been correlated with the type, morphology, amount and distribution of second phase precipitates resulting from different thermomechanical treatments.

Another area that has received attention is the crevice corrosion in type 316 stainless steel in aqueous chloride solution. The compositions of the critical crevice solution were determined for different combinations of pH and chloride ion concentrations chosen on the basis of a mathematical model which considers all possible hydrolysis processes occurring in the crevice. The studies have also revealed the new finding that texture has an important influence on the crevice corrosion behaviour of type 316 stainless steel.

A phase analysis attachment for the Scanning Electron Microscope was designed, fabricated and calibrated. It is now possible to use this equipment to obtain detected images of specific phases from signals of secondary electrons, back-scattered electrons and absorbed current. Several stainless steel weld samples were satisfactorily examined using the above attachment to estimate the ferrite contents.

The Field Ion Microscope built by the Technical Physics Division of BARC has been commissioned and utilised to image dislocations and grain boundaries in tungsten. This instrument has been successfully converted into an Imaging Atom Probe by an in-house development of the necessary instrumentation.

Among the various in-cell equipment for Radip-Metallurgy Laboratory, development efforts have been successfully completed on the profile meter for fuel subassembly, computer based gamma scanning system, and one of the six lines of the biaxial stress rupture systems.

3.5 Reprocessing Programme of the Fast Reactor

The Engineering Laboratory was operational during the year carrying out R&D activities in the development of equipment in the field of process engineering. In the inactive chemical laboratory, refinement of analytical methods by adaptation of known methods for remote application is being carried out. The laboratory has been tested for handling of radioactive material which is likely to commence soon. Containment box for the lead shielded facility is getting ready. With this the work on lead shielded cell will assume accelerated progress. The one cycle integrated equipment test will be carried out with uranium for evaluation of equipment and system in concrete cells.

Work on direct dissolution of uranium carbide and oxidation of organic acid electrolytically is in progress. Parallel work has commenced on direct oxidation of carbide as well as pyro-hydrolysis.

A dissolver system of thermosyphon type is being fabricated for the evaluation of powder distribution and also heat transfer characteristics.

Electrolytically aiding the dissolution of oxide with a redox system is being attempted in a small glass dissolver.

A continuous dissolver of rotating type has been tested for material movement. The hull delivery system for the basket is also under test.

Single stage centrifugal contactors developed were successfully tested in a bank for evaluation of efficiency and reliability. The above are to be tested for evaluation in hot cells. A three-stage contactor developed was subjected to extensive test with uranium and for hydraulic stability and was found to meet the requirements satisfactorily. This contactor is to be tested now for remote maintenance in lead cells.

Work on horizontal baffle contactor has indicated that the mixing was equally efficient.

Air pulsed mixer settler was tested with uranium-thorium system with the new pulsing system.

A magazine for feeding pins to the shear is under fabrication. The system of pneumatic chopper in combination with feed mechanism and dissolver has to be tested. Feed clarification system using a horizontal vertical sintered filter element was tried. The system is being tried for remote replacement of the element and removal of solid. Pneumatic system with suitable diverters for the conveying stations is being tested.

Feed metering system using constant height variable orifice device has been studied for 3mm/1.6 mm orifice and the precision of the system was found to be excellent.

Various off gas filtration systems and scrubbing system suitable for lead cells as well as pilot cells were studied. Deep bed fibre filter has been designed for both the applications. A fluidised sand bed filter is being planned as the pre-filter in the one-cycle pilot cell tests. Experiments on absorption of iodine using mercuric nitrate and for recovery of mercury and fixation of iodine are in progress.

Mock-up studies have been carried out to finalise the cask design for the transport of radioactive material between facilities and the cask design finalised. Work on development of reconversion step for refinement like use of centrifugal decanter, etc., is in progress.

3.6 Radiochemistry

During the past one year the emphasis has been on sodium chemistry. The electrochemical oxygen meter made out of a long tube of yttria doped thoria continued to work satisfactorily. One such meter was used to measure the equilibrium oxygen level in sodium for the reaction between sodium and uranium oxide.

Two types of electrochemical carbon meters are being experimented with. In the first one, a reference electrode dips into a molten electrolyte consisting of the eutectic mixture of Li_2CO_3 and Na_2CO_3 which is separated from sodium by an iron membrane. The e.m.f. is measured between the reference electrode and the iron membrane which is the equilibrium with sodium. The early models based on graphite or carburized iron as reference electrode failed within a few hundred hours. Studies are at present progressing on a new reference electrode which appears to give promising results. The second carbon meter takes advantage of the fact that carbon diffusing through an iron membrane can alter the CO/CO_2 ratio maintained on the non-sodium of the membrane. This can be measured as an oxygen potential using an oxygen probe making use of calcia doped zirconia solid electrolyte tube. This method is also giving encouraging results.

A small sodium chemistry loop has been set up to test the above meters and to intercalibrate analytical techniques involving (a) sampling and analysis; (b) foil equilibration; and (c) on-line meters.

In the analysis of metallic impurities in sodium, volatile impurities like Zn & Cd were found to be difficult to determine by the standard procedure that has been adopted in our laboratory which involves the distillation of sodium followed by atomic absorption spectrophotometry. A direct method for determining these impurities, without resorting to the distillation step in which volatile impurities get lost, has been developed based on furnace AAS. Another method based on anodic stripping voltammetry is also being developed for this purpose.

Studies have also been initiated to measure the activities of different carbon bearing species in sodium. Preliminary results show that the addition of different carbon compounds to carbon-free sodium results in different thermodynamic activities.

3.7 Study of Fast Reactor Safety Problems

Primary Containment Studies :

Despite the current shift of emphasis from scale model tests to development and use of elaborate computational models for the assessment of structural integrity under postulated accident conditions in fast reactors, idealised scale model experiments still serve the useful purpose of providing, by extrapolation, estimates of pressure, impulse and deformation without expense or loss of time. Tests in open water filled cylindrical vessels with a charge of high explosives have enabled the determination of appropriate scaling laws for extrapolation. These experiments are being repeated with thin-walled vessels and slower explosives.

Codes are being developed for understanding shock wave structure interactions to permit the design of reactor vessel and containment structures to contain the consequences of a design-based accident.

Particle size spectrum studies have been made for fuel aerosols generated in rapid transients, achieved by deposition of electrical energy stored in a capacitor bank. Characterisation of aerosols produced in slower fuel vaporisation transients made possible by direct electrical heating, is being carried out. Melt down studies with irradiated fuel for investigating release behaviour of fission products are proposed to be done at a later date, in hot cell facilities.

Secondary Containment Studies :

Activities in this area are principally oriented towards the study of aerosols of sodium and its oxides, in particular, their coagulation and settling behaviour. In experiments carried out so far the effect of humidity has been investigated for initial mass concentrations in the range of 80-500 mg/m^3 and a starting

size of 0.9 micrometer MMD. These results have been compared with theoretical calculations using computer codes. Attention is also being focussed on factors such as deposition on walls and duct, and effect of electrostatic charge on aerosol behaviour in the current phase of these studies. In the next stage, use of spray mode of sodium combustion to generate larger initial concentrations is planned.

Accidents not directly affecting core :

Codes for studying the consequences of sodium water reactions in steam generators, superheaters and resuperheaters are being developed.

To study the potential for damage of sodium-water reactions in the steam generator, preliminary design has been completed for a test rig, which includes provisions for measurements of various characteristics of the transients.

Fuel Coolent Interaction Studies :

Following the preliminary studies carried out with tin-water systems for the purpose of standardising the experimental set-up, a series of out-of-pile experiments with UO_2-water and UO_2-Na systems have been conducted. Melting of the UO_2 pellet is achieved by direct electrical heating and the contact with liquid sodium established by dropping the molten UO_2. Characteristics of the pressure pulses generated namely peak pressure, risetime, and impulse are measured. In the next phase, these experiments will be repeated with uranium carbide sodium system.

Development of computer codes for assessing mechanical damage potential for fuel coolant interactions is in progress. Refinements have been made in existing computer codes by way of (a) replacement of the acoustic constraint by a mixed (acoustic + inertial) constraint (b) incorporation of vapour blanketing and improved heat transfer calculations; (c) inclusion of time dependent fragmentation in a parametric manner; and (d) accounting for latent heat of fusion of fuel by appropriately modifying the initial fuel temperature.

Instrumentation Development :

Ultrasonic techniques are being applied for development of safety related instruments. Presently, work is in progress on the development of instrumentation for (a) detection of local boiling in sodium; (b) detection of failure of structural members submerged in sodium through detection of acoustic emissions; and (c) detection and location of failed fuel subassemblies in the fast reactor core, based on the principle of scattering of ultrasonic radiation by the fission gas bubbles emerging from the defective fuel channel.

Dynamic calibration technique for shock pressure transducers has been developed using the dependence of shock velocity on the shock overpressures.

A standard shock generator using the exploding wire technique has been developed and used for the study of decay characteristics of shock overpressures in liquid media, in connection with the fuel coolant interaction studies, for evaluation of thermal to mechanical energy conversion.