REVIEW OF FAST REACTOR ACTIVITIES IN INDIA (1982–83)

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1.0 Introduction:

Survey of historic growth of demand for commercial forms of energy and projections of the demand during the next two decades indicates that it may reach a figure of 0.5 billion tonnes of coal equivalent by the year 2000. This will mean that unless some alternative energy source is established prooven reserves of coal will get exhausted before the year 2100 even if the energy consumption is held constant at this level (which approximately corresponds to an installed capacity of 0.1 Kilo Watt per head). Such an alternative should be established before consuming 20 to 25% of the total reserves of coal so as to leave the valuable reserves of fossil fuels for extended use in the centuries to follow. Fast Breeder Reactors using advanced fuels (capable of obtaining reasonable low specific inventory of 3 Kgs. per MW electric and breeding ratio of 1.3) appear to be the only viable alternate source. Thanks to the Heavy Water Reactor Programme, Plutonium production rate is expected to reach a level of 3000 Kg/year by 2000 A.D. With this Plutonium input in a vigorous programme of deployment of Fast Breeders on a commercial scale, it will be possible to replace coal as a major source of energy production during a transition period of about 50 years (from 2000 A.D to 2050 A.D). During this period it will be possible to maintain a growth rate of about 1.5% per annum in the supply of energy while depletion of coal can be limited to 20–25% of the total reserves. This is therefore both a challenge and opportunity for the Indian Nuclear Energy Programme based on the combination of HWRs and FBRs.

1.1 In this context we are now proposing to design, develop and construct a 500 MW Proto-type Fast Reactor which can be connected to the grid in the late nineties. The Reactor is expected to be the "Head of the series" to be built from 2000 to 2005 for commercial deployment. We are hopeful of obtaining the requisite support for this ambitious step. But as Fast Breeder Reactors constitute the second stage of the Nuclear Programme, success of the First Stage HWRs is likely to influence the overall response of the Government and the public. Capital costs of HWRs are now in the range of Rs.10,000 per Kilo Watt installed. Capital costs of Coal Fired Power Stations lie in the range of Rs.6,000-8,000. Unit energy costs for HWRs is expected to be lower than the unit energycost of Coal Fired Power Stations located not too far away from the Coal Mines. Hence there is a cautious optimism on the Nuclear Front in general and Fast Reactors in particular. Much will therefore depend on the success of Heavy Water Plants and FBTR which is now nearing completion.

2.0 Financial Outlay

Financial outlay for the year ending 31.3.1983 has been 148.9×10^6 Rupees while the same figure for the previous year has been 119.4×10^6 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 contribution 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.

3.0 Fast Breeder Test Reactor(FBTR)

The Reactor is now in a very advanced stage of construction. Reactor Vessel, Rotating Plugs (which provide the top closure and access for fuel handling) and Control Plug (which houses the control rod drive mechanism) have all been delivered to the Site. The installation is in progress. The vessel has been installed, the grid plate has been lowered in its position and mounting of neutronic shields inside will be taken up soon. Most sub-assemblies to be loaded in the Reactor

Vessel except the fissile sub-assemblies have been fabricated and are in the process of delivery to Site. Welding of inlet and outlet pipes is in progress. Intermediate Heat exchangers and primary sodium pumps have been installed and primary piping work is in full swing.

3.1 Testing of both primary and secondary sodium pumps has been completed in water tests. Control rod drive mechanisms have been assembled and are being tested in hot Argon. All equipments except the steam generators are already at Site and steam generators would be arriving at the Site in the next months. Commissioning of the systems has been initiated. Some systems like air-conditioning, ventilation, process water, normal and emergency power supplies have been in operation during the year. Sodium required for the Reactor has been produced and a small purification rig has been built to upgrade the quality from "commercial" to "Reactor" grade.

Two computers constituting the system of on-line plant supervision have been installed in their permanent location in FBTR. These are also used for reactor protection. Out of the two Computers, one works on line while the other one is a stand-by ready to take over from the first computer. The two computers have a common memory and vital systems of the reactor can be monitored by the stand-by computer. While it stands by, it can be used for scientific applications.

On the licensing side, one review by the Working Group constituted by the Safety Review Committee of Department of Atomic Energy (DAE-SRC) has been completed. Modifications/additions proposed by the Group have been discussed and are being incorporated in the design. No difficulties are therefore foreseen to arise in obtaining formal authorisation to start the reactor.

Barring unforseen circumstances, criticality is expected within two years from now.

4.0 Design Studies for 500 MWe Proto-type Fast Breeder Reactor

As FBTR is nearing completion, design studies for a 500 MWe proto-type have been taken up. This was reported in the last year. This phase of prilliminary design is coming to an end and we will be submitting a Project Report to the Government in the next few months. Main features of the design can be given as:

1) Thermal output — 1250 MW(t)
2) Electrical output — 500 MWe gross
3) Fuel Inventory

Inpile	— 1500 Kg
Out of pile	— 600 Kg
Total	— 2100 Kg

4) Breeding Ratio:

		Normal	Advanced
Oxide	...	1.146	1.351
Carbide	...	1.313	1.425

Doubling Time Years (Linear) : (based on 2% loss in reprocessing and refabrication. Out of pile time = 365 days)

		Normal	Advanced
Oxide	...	47.5	19.19
Carbide	...	19.6	16.41

FUEL

No. of fuel subassemblied	: 180
No. of blanket subassemblies	: 205
Fuel pin dia	: 6.5 mm (Oxide) 8.0 mm (Carbide)
Fuel smear density	: 80% of oxide; 75% of carbide
Bond material	: Helium gas
Cladding	: cold worked AISI 316 with or without Ti modification
Clad thickness	: 0.45 (oxide) & 0.56 mm (carbide) for normal design
F.G.plenum location	: Bottom
No. of pins/subassembly	: 217 (oxide) 169 (carbide)

REFUELLING

Refuelling interval in days	:	240
Max. fuel burn-up	:	138000 MWd/te
Average fuel burn-up	:	90000 MWd/te
Internal storage days	:	240
No. of positions	:	30

REACTOR BLANKET

Axial : Integral	with fuel pin
Material form	pins
Pin Diameter	6.5 (Oxide); 8.8 (Carbide)
Top length	300 mm
Bottom length	300 mm

RADIAL BLANKET

No. of subassemblies	205
Material form	Pins
Pin dia	14.65 mm
Assembly length	1685 mm - active
	5000 mm - total
No. of pins per subassembly	: 61

CONTROL

Material	B4C enriched to 50 in B^{10}
No. of subassemblies	12 - 9 for 1st system
	- 3 for 2nd system
No. of pins/subassembly	19
Pin cladding	AISI 316

REACTOR VESSEL

Height	14.7 M
Diameter	15.0 M

PRIMARY

Type	Pool
No. of loops	4
Pump type	Centrifugal - mechanical
Pump position	Cold leg
Total flow	8000 Kg/sec
Inlet Reactor Temperature	370°C
Outlet Reactor Temperature	530°C

SECONDARY

No. of loops	4
Pump type	Mechanical - centrifugal
Pump position	Cold leg
Total flow	8000 Kg/sec

INTERMEDIATE HEAT EXCHANGER

Number	8
Primary side	Shell
No. of tubes	1549
Tube dia x wall thickness	24 x 1 mm

STEAM GENERATORS

Numbers per loop	3
Type	Vertical straight tube with an expansion bend
Superheater	Yes - steam temp. 480°C
Re-superheater heated by Sodium	Yes - steam temp. 480°C

TURBINE

No.	1
Inlet pressure MPa	16.5
Inlet temperature	460°C

DECAY HEAT REMOVAL

No.	4
Type	Na to Air heat exchanger
Capacity	5 MW per unit (Natural convection)
	7.5 MW per unit (Forced convection)

5.0 Reactor Physics Studies:

5.1 Nuclear Data Evaluation, Processing and Testing:

Nuclear data processing code RAMBHA developed in the centre and successfully commissioned and tested with ENDF/BIV data library has been used to generate multi group cross sections for several materials of interest for the analysis of fast critical experiments and for the neutronics studies of PFBR. Analysis of ZPPR-6-7 critical assembly using the newly generated cross section set has been completed. Analysis of a number of selected fast critical assemblies has been taken up for comparing the various predicted parameters with measurements.

A code system to carry out the sensitivity analysis of the effect of uncertainities in our nuclear data libraries on the core physics parameters predicted by us is under development.

An IAEA code verification project suggested by the Nuclear Data Section of IAEA as an international effort in comparing the accuracy of processing codes used by different groups has been taken up recently.

5.2 Fast Reactor Design and Analysis:

Preliminary Physics design of 500 MW(e) Proto-type Fast Breeder Reactor (PFBR) has been completed.

Relative merits of oxide and carbide fuels were studied and all the neutronic parameters for a carbide fuelled core were evaluated. Assessment of the error bars in the predicted neutronics parameters of PFBR is being carried out using sensitivity analysis methods.

5.3 Fast Reactor Safety Analysis:

The European LOFA Benchmark problem has been studied with our computer code system PREDIS-VENUS and our results are in good agreement with those of other countries.

Considerable progress has been made in the development of computer codes for the predisassembly and disassembly phases of accident analysis using deterministic models. These codes have been used to determine the accident potential of cores fuelled with mixed oxide and mixed carbide fuels. No significant differences have been observed during the analysis of LOFA and TOPA accidents. These have been presented and discussed with DAE-SRC.

Improvements in the deterministic models used in the accident analysis with respect to sodium boiling, clad relocation etc. are being incorporated in the code systems.

Development of suitable computational models for estimating the mechanical damage to the reactor structures following energy release in DBA is being taken up.

Improved and more realistic models were developed for the study of molten fuel coolant interaction (MFCI). Further the improvements in the MFCI calculational models which result in better agreement with experiments are being studied.

A national workshop on Reactor Noise Analysis was conducted at the Centre for two days in which topics like development of suitable sensors and instrumentation, advanced techniques of signal analysis, mathematical modelling of noise source and measurements of noise in reactors and laboratories were presented and discussed. This served to define in more clear terms the R &D programme required to be persued to evolve a meaningful reactor surveillance system.

5.4 Radiation Transport and Shield Design:

Preliminary shield design calculations for the radial and axial internal shields of PFBR have been completed using I-D models to arrive at suitable shield configurations in both directions.

Detailed two-dimensional and Monte Carlo calculations are being carried out for the final shield configuration of PFBR.

Development of coupled transport theory methods and Monte Carlo methods for studying the detailed neutron and gamma transport and streaming in practical shield configurations has been taken up.

6.0 Metallurgical Development (In support of the Fast Breeder Programme)

6.1 In the continuing programme for the generation of base line data on the mechanical and corrosion properties and on the microstructural characteristics of the materials used in FBTR, considerable data have been obtained durin the year on the tensile and creep properties of 316 Stainless Steel cladding tubes in solution annealed and cold worked conditions at test temperatures ranging up to 750°C. Detailed analyses of the large volume of data already generated on the four heats of 316 Stainless Steel plates have given valuable insight into the mechanistic aspects their deformation behaviour, besides providing the correlations governing their creep and tensile strengths and variations in properties from heat to heat.

6.2 High strain (low cycle) fatigue life being an important design parameter of fast reactor meterials studies on the effect of strain rate and grain size on the low cycle fatigue behaviour of type 304 Stainless Steel were carried out at temperatures up to 750°C. It was observed that in general fine grained materials are superior in fatigue endurance at all strain rates and temperatures investigated. Studies on creep-fatigue interactions at 650°C have also confirmed the superiority of fine grained materials.

6.3 The rapidly developing NDT technique of accoustic emission monitoring was used for studying the influence of microstructure (viz. carburisation and carbide precipitation) on the tensile deformation and fracture behaviour of type 316 Stainless Steel. In addition to through their effects on impeding dislocation motion, carbide precipitates were found to influence accoustic emission through grain boundary cracking also. Using techniques standardised and reported last year, eddycurrent test method has been used a complementary technique to ultrasonic method, for evaluating FBTR fuel cladding tubes (20% cold worked 316 Stainless Steel); tube lengths sufficient for more than two reactor cores have been evaluated and cutting plans evolved to obtain tube lengths free from unacceptable flaws.

6.4 In most of the surface hardening applications, ionnitriding is expected to ultimately replace the conventional gas nitriding. Investigations were carried out on the ion-nitriding behaviour of type 316 Stainless Steel by the glow discharge method. An uniform nitrided layer of about 4μm thickness, with a maximum surface hardness of 1000 VHN, was achieved by carrying out cyclic diffusion annealing.

In the technology of metal forming, warm working is known to offer good scope for energy saving in addition to resulting in improvements in room temperature mechanical properties. An investigation on the warm rolling of type 316 Stainless Steel was carried out in the temperature range of 550 to 850°C. Two new experimental techniques determination of the rolling temperature using continuous cooling curves and determination of the mean strain rate during rolling using longitudal strain, have been developed and used in this work. By warm rolling it was possible to increase significantly the room temperature tensile strength with very little loss of ductility. Typically, the yield strength at room temperature was raised to 410 MPa from a value of 250 MPa corresponding to the solution annealed material with an attendant drop of only 10% in the total enongation from ~ 55 to 45%.

6.5 Stress corrosion studies on the morphology of the fractured surfaces of cold worked stainless steels of types 304 and 316 revealed that a transition in crack propagation mode occurred from transgranular to intergranular in type 316 Stainless Steel for all degrees of cold work while such a transition took place in type 304 Stainless Steel only above 26% cold work. The texure dependance of crevice corrosion behaviour in type 316 Stainless Steel reported earlier was further investigated in types 304 and 310 Stainless Steels. A direct correlation of the crevice corrosion potential with the stacking fault energy of the alloys was observed. Thus rolled plates of alloy 310, having the highest stacking fault energy and maximum anisotropy, showed marked variations in crevice corrosion resistance in the three perpendicular cross sections. The variation in crevice resistance was less marked in type 316 Stainless Steel, and the least in type 304 Stainless Steel. Investigations on the kinetics of the postbreakaway oxidation behaviour of two ferritic alloys, viz., $2\frac{1}{4}$ Cr–1Mo and 9Cr–1Mo steels, showed that the breakaway oxidation at the transitional stage occurs by the rupture of the protective chromium-rich mixed oxide scale resulting in the formation of a duplex layer consisting of iron oxide at the outer layer and a spinal of oxides of chromium and iron at the inner layer.

6.6 $NaCrO_2$ is an important corrosion product in the Stainless Steel liquid sodium system. Threshold oxygen concentration in liquid sodium for corroding 304 Stainless Steel was predicted by the thermodynamic assessment of the EMF data obtained from the $NaCrO_2$–Cr_2O_3–Na_2CrO_4 system along with the free energy data on $NaCrO_2$ available in the literature.

7.0 <u>Radio Chemistry Activities in Support of Fast Breeder Reactor</u>

7.1 <u>Radiochemical and Chemo-technical Studies</u>

The Radiochemistry Laboratory is pursuing R&D work in sodium chemistry and fuel chemistry. In sodium chemistry, we have completed the chemical quality control of the commercial sodium procured for FBTR. The various techniques developed for this purpose are adequate for the determination of trace level impurities in reactor grade sodium.

<u>Estimation of Chloride in Sodium</u>

Matrix interference was observed in the estimation of very low levels (less than 5 ppm) of chloride in sodium both by spectrophotometric as well as by ion selective electrode methods. Vacuum distillation of sodium was tried to remove the matrix. Sodium chloride standards were added to sodium before distillation in order to establish that no loss of chloride occurs during distillation. Recovery of standards added (5.10 ug) was 95 to 100% .

<u>Analysis by Atomic Absorption</u>

The direct method of determination of trace metals in sodium using graphite furnace AAS is being utilised for estimation of lithium and potassium in sodium. Standardisation work is in progress for the same.

<u>Analysis of Steels by X-ray Fluorescence</u>

Defects in the vacuum system of the instrument were rectified and the instrument was made working in the manual mode. Analysis of Stainless Steel for different constituents was taken up. Calibration for nickel has been completed and calibration for other elements are in progress.

7.2 Meanwhile, a programme to measure solubilities of metals in liquid sodium has been taken. The solubility of manganese in sodium has already been reported. Such data will be useful in modelling activity transport in the primary sodium loops of fast breeder reactors.

7.3 Different types of electrochemical meters are under development for the on-line monitoring of oxygen, carbon and hydrogen in sodium.

<u>Electrochemical Oxygen Meter</u>

A few experiments were carried out to study the feasibility of using YDT shield electrolyte for oxygen measurement in sodium in the diffusion controlled mode. The measured currents were lower than expected. More experiments with increased electrolyte surface and a new barrier for oxygen diffusion from bulk sodium to the electrolyte surface are planned.

<u>Electrochemical Hydrogen Meter</u>

The meter re-assembled with purified electrolyte showed near theoretical behaviour when tested in a sodium capsule.

<u>Electrochemical Carbon Meter</u>

The electrochemical carbon meter

$$Sn,\ SnO_2\ /\ CSZ\ /\ \underset{(SrCO_3)}{CO_2}\ ,\ CO\ /\ \underset{Ni}{C}(Na)$$

was tested in a capsule with SrO_3 in the CO, CO_2 compartment replacing $CaCO_3$. Hydrogen in sodium was found to interfere with the CO, CO_2 equilibrium.

The electrochemical carbon meter

$$C\ \underset{Ni}{/}\ Na_2CO_3\ ,\ Li_2CO_3\ \underset{Fe}{/}\ C\ (Na)$$

was tested with the new S.S. capsule in place of the nickel capsule. S.S. foils annealed at 1050°C were introduced in the sodium to determine the carbon activity in sodium by the foil equilibration method. The meter failed after 9 days of operation due to corrosion from sodium side.

view to testing a number of these meters in flowing sodium, a sodium loop is being set up. This loop will also be used for activity transport studies.

7.4 Various experimental techniques necessary for thermodynamic studies on fuel materials have been set up and standardized. These include Knudsen cell mas spectrometry, high temperature X-ray diffraction, high temperature microcalorimetry and thermal analysis. A thermal conductivity measuring apparatus, which makes use of a laser flash technique, has been designed and fabricated. To prepare and characterize fuel materials for these studies, a preparation laboratory has also been set up. Work with radioactive materials has only just commenced in this laboratory. It has now become possible to take up process chemistry work related to the reprocessing of FBTR fuel.

8.0 Summary of work in Reprocessing Programme

Development of process equipment/systems such as single pin chopper, solvent contactors, feed clarifier, dissolver etc. is in the advanced stage. An one cycle operation, integration equipment developed to evaluate reliability is to commence shortly. Computer studies on solvent extraction flow sheets are continued. Experiments with direct dissolution of Uranium carbide indicate favourable results for electrolytic oxidation organic acids produced. Alpha laboratory ready for process chemistry and flow sheet evaluation studies. Flow sheet studies on U, Th systems were carried out. Lead shielded facility for reprocessing studies on 100gm scale of fuel is getting ready. Work in progress for a preliminary design of a pilot plant for reprocessing Fast Reactor fuel on 15-20 Kg/day.

9.0 Summary of Studies carried out in the Safety Research Laboratory

In experiments conducted to simulate the FCI phenomenon, attention has been given to the size distribution of the debris particles characterising the events. The out-of-pile experiments involve melting of a UO_2 pellet (2 g; or 30 g) by direct resistive heating and contacting of the molten mass with liquid sodium or water as the coolant. Contact with liquid sodium is established by dropping the molten mass while in the case of water, the UO_2 pellet (either bare or enclosed in an alumina sleeve) is kept submerged in the liquid. The debris particles arising due to interaction of the fuel with liquid sodium, on separation from bulk sodium, showed a range of mass median diameters of 400 to 840 m. A separate control experiment indicated that additional fragmentation is introduced by the separation process itself. The UO_2(submerged) – water events led to larger sized debris particles with MMD's in the range of 4-7 mm.

Attempts are being made in experiments on the UO_2 – Water system to determine the location of the zone of interaction by simultaneous recording of pressure signals from multiple piezoelectric pressure transducers.

Experiments with the molten UC-Sodium are under way.

9.1 BEHAVIOUR OF SODIUM OXIDE AEROSOLS

In this area, experimental studies are initiated to cover the behaviour of iodine and cesium in fires involving contaminated sodium. While release behaviour of the elements is being examined for the large scatter in the reported results, the size behaviour of the aerosols is being studied with respect to the radioactive contominant as a function of both time and relative humidity. The mass median diameter associated with the ^{131}I content of the oxide aerosols, collected immediately after their generation (t=5 to 10 min.) from sodium fires, is observed in these experiments, to coincide with the MMD for the oxide content. These experiments are carried out in a chamber of 5.4 m^3 capacity by burning sodium containing carrier free ^{131}I