STRUCTURAL MATERIAL EXPERIMENT

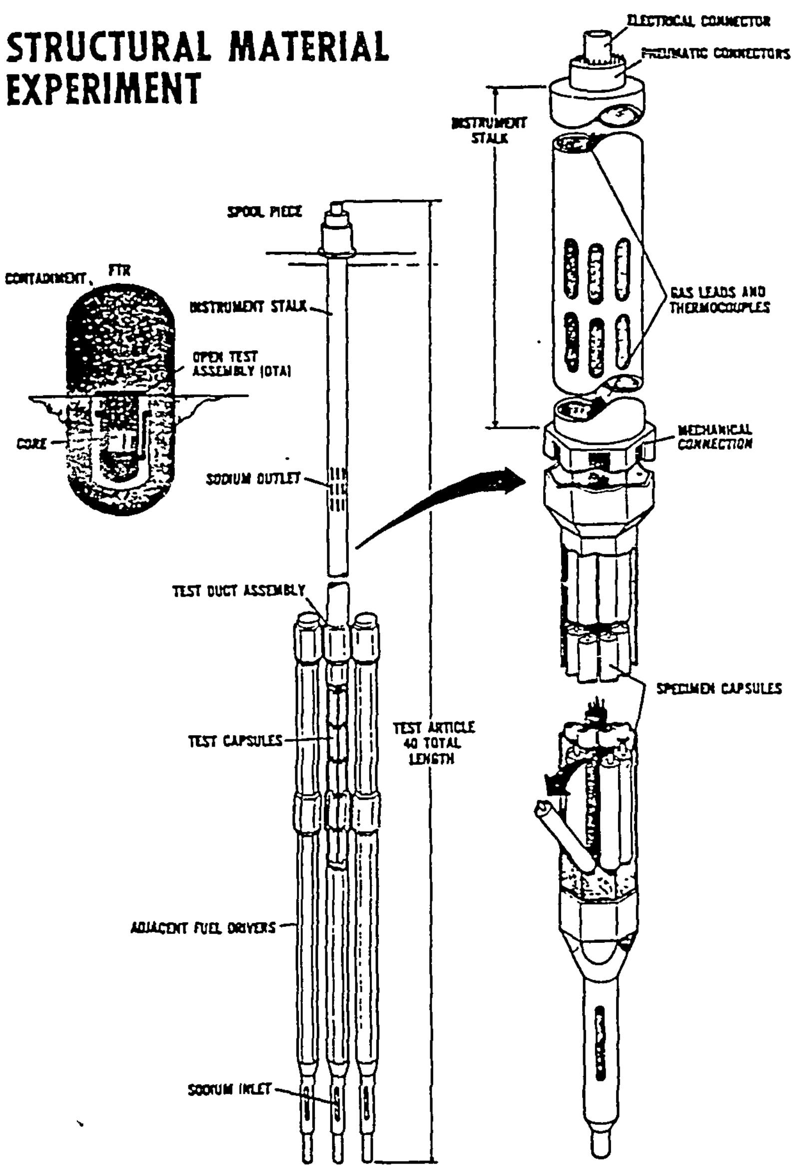

FIGURE 3.1

REVIEW OF FAST REACTOR ACTIVITIES IN INDIA (1983–84)

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1.0 General Background

The last year was very significant for Indian Nuclear Energy Programme as the first indigeneously built heavy water moderated natural uranium reactor called Madras Atomic Power Plant Unit-I was made operational and connected to the grid. The power level has been gradually increased and the reactor has been operating at a power level of 200MWe (temporarily limited by Plutonium build up during approach to equilibrium core loading). The 'plutonium peak' will be crossed shortly clearing the way for raising the reactor to the full power of 235MWe gross. The second unit of MAPP, is well advanced and barring unforeseen difficulties, is expected to become operational during this financial year. This has been a big morale booster for the programme in general and the Fast Reactor Programme in particular as plutonium produced in these reactors is expected to be the inventory for Prototype Fast Breeder Reactor. It may be recalled that in the last report to this group, a reference was made to initiation of some preliminary design studies for such a reactor.

Demand for all commercial forms of energy has continued to rise even during the last year as against decrease in the demand for electricity observed in many of the industrialised countries of the Western world. Both industrial production and Gross National Product have registered a significant growth (over five percent). Hence in spite of continued increase in the generation capacity, most of the regional grids in India had to impose restrictions on the consumption of electricity. It is also being recognised that reserves of fossil fuels which to-day appear large are really not large. Consequently, there is increasing realisation of the need to develop nuclear energy as an alternate source of energy. On the other hand, inflation though moderated is still significant (around 10% per annum) and calls for vigil on the economic front. So the Nuclear Energy Programme which appears justified on account of the need and cost per nuclear kwh is likely to be constrained to some extent by higher initial investments in Nuclear Plants.

Public acceptance of nuclear energy is not too much of a difficulty. All the same, to promote greater public confidence, Government of India has created a new organisation called Atomic Energy Regulatory Board reporting to the Atomic Energy Commission. Presently, it is headed by Dr.A.K.Dey, who was formerly the Director of Indian Institute of Technology, Powai, Bombay. The board has powers to lay down safety standards and frame rules and regulations in regard to the regulatory and safety requirements envisaged by the Atomic Energy Act 2963 and the board will ensure their compliance by Department of Atomic Energy (DAE) and non-DAE installations during construction-commissioning operation. The board will be assisted in the performance of its tasks by DAE Safety Review Committee which was earlier the body carrying out such and similar functions in-hours within DAE.

In this context, with cautious optimism and renewed enthusiasm, DAE is approaching the Indian Government to approve a plan of installing 10 million KWe of installed nuclear capacity and attendant expansion of fuel and heavy water production capacities and addition of fuel reprocessing plants. The target is to be reached by the turn of this century.

2.0 Financial Out-lay:

Financial out-lay for the Reactor Research Centre for the year ending 31.3.84 is placed around 137.6 x 10^6 Rupees (Today 1 US Dollar is approximately 10.5 Rupees) while the proposed out-lay for the coming financial year has been placed at 116.6 x 10^6 Rupees. These efforts are to a certain extent supplemented by Bhabha Atomic Research Centre and other units of DAE. Man power deployed has increased somewhat as the Fast Breeder Test Reactor - the main unit in Reactor Research Centre, Kalpakkam in entering in the final phases of commissioning before reaching criticality.

3.0 Fast Breeder Test Reactor (FBTR):

The reactor construction is now nearing completion. The reactor vessel installation was completed in the previous year itself. During the year, erection of internal parts (i.e. neutronic and thermal shields) was completed. This was followed by installation of the top structures consisting of rotating plugs, control plugs, the ball bearing structures and the anti-explosion floor. Different types of subassemblies like steel and Nickel reflector assemblies and dummy fuel subassemblies have also been loaded in the reactor vessel. All components of the primary sodium circuit have been installed and the entire primary sodium piping will be completed shortly. The secondary sodium piping is already complete. Reactor grade sodium required for the secondary sodium system has been produced starting from commercial grade sodium. Purification of sodium for primary circuit has also been initiated. Sodium will be charged into secondary sodium circuit very soon. Two of the four steam generator units have already arrived at the site and the balance two are getting ready in the factory and could be expected within next few months. There has been some delay in the receipt of steam generators, and consequently, it has been decided to attain the first criticality and carry out low power physics measurements without the steam generators. All associated equipments like sodium pumps, control rod drive mechanism, drives for the primary and secondary sodium pumps and fuel handling machines have been tested. Leak test of reactor containment building has been carried out successfully. Leak rate is well below the permitted value of .1% of building volume/hr.

While the auxiliary systems (air conditioners; ventilation, power supplies, process water, emergency power supply etc) have been commissioned, commissioning of main systems like primary and secondary sodium system, preheating and emergency cooling system, biological shield cooling system, flooding system, nuclear instrumentation and control system, failed fuel detection systems based on delayed neutron detection (DND) and gaseous activity monitors will be taken up progressively with the object of attaining the first criticality towards December 84 - January 85.

On the licensing side, after review by the working group constituted by the safety committee, the subject is now being discussed with the Safety

Review Committee of DAE, basically for finalisation of the technical specifications which define the plant operation limits and administrative procedures which supplement the automatic and autonomous reactor protection system.

4.0 Design Studies for 500MWe Prototype Fast Breeder Reactor (PFBR):

Preliminary design studies for PFBR mentioned in the earlier report have been completed and submitted to the Atomic Energy Commission in the form of a Project Report. Main features of the plant have been already reported and would also appear in the IAEA compilation of LMFBR plant parameters. Hence only the important points arising out of the preliminary design study will be mentioned.

The reactor core is being designed to have the same number of subassemblies with same external dimensions while the fuel proper may be mixed oxide or mixed carbide. Power density expressed as MW/Te will be quite comparable for the two fuels. The major difference will be in the linear power and the pin diameter. Using consistant assumptions and models, energy release under Design Base Accident(DBA) conditions like Loss of Coolant (LOCA) or Transient Over Power Accident (TOPA) or TOPA followed by LOCA, have been determined for both fuels and the results are comparable. Mixed carbide fuels do not appear to be more hazardous than the well known and widely accepted mixed oxide fuels. For either fuel, energy release during DBA works out to a sufficiently small value as to permit housing the reactor vessel in an industrial leak resistant structure as is the case of PFR in U.K. or Phenix in France.

Studies have been carried out to see if PFBR can be made inherently safe without scram under the worst conditions of loss of onsite and offsite power. It has been found that by choosing a sodium pump coast down lawwith a halving time of around 60 to 70 secs., one can have adequate negative feedback reactivity and the power and temperatures in the core can be maintained within safety limits.

Possibility of omitting ex-vessel storage of irradiated fuel and the 'A' frame transfer machine for movement of fuel from the pool to ex-vessel has been explored. Short refuelling interval has been adopted for PFBR in order to obtain better breeding ratio and greater average burn-up for relatively low and conservative value of maximum burn-up assumed as design basis. Under such conditions, it becomes necessary to have ex-vessel storage and 'A' frame for fuel handling.

Inelastic analysis of the secondary sodium piping appears to justify the allowable stresses permitted by the power piping code as compared to very low values permitted by ASME code case N-47.

Finally, the cost of PFBR does appear rather high compared to thermal reactors even under conditions prevalent in India. And so significant and persistent efforts are required to make LMFBRs economically attractive.

5.0 Reactor Physics Studies:

5.1 Nuclear Data Evaluation, Processing and Testing

The nuclear data processing code RAMBHA, developed in the Centre has been used to generate multigroup cross sections from the ENDF/B-IV data library for various fertile, fissile, structural and coolant materials of interest to fast reactors. Analysis of ZPR-9, Assembly 31, a carbide fueled benchmark critical assembly has been completed and analysis of some more selected critical assemblies is in progress, using the newly generated cross section set.

The computer code package RECOIL for calculation of displacement damage cross sections has been commissioned and has been run for stainless steel material.

As a part of an IAEA code verification project, using the ENDF/B-V dosimetry library, unshielded cross sections in the SAND-II, 620 group structure were generated and sent to nuclear data section of IAEA for comparison with the benchmark standard results.

Cross section sensitivity studies for fast neutron transmission through sodium were carried out using the sensitivity analysis code system commissioned earlier.

5.2 Reactor Theory, Design and Analysis

The application of continuous space dependent synthesis using axial helmholtz modes for fast reactor diffusion theory calculations has been studied and a two dimensional code written based on this method has been tested against an LMFBR benchmark problem successfully for the efficiency od the method which will be ultimately used for three dimensional hexagonal-z fast reactor calculations.

An analysis of a new generalised first order perturbation formula for the diffusion equation, derived recently by Pomraning has been made to assess its suitability to replace the classical perturbation formula and it is found that the generalised formulation has advantage over the classical formulation only in a limited range.

The dependence of loss of burn-up of discharged fuel on the re-fuelling interval for the prototype fast breeder reactor (PFBR) was studied and the optimum cycle was arrived at from the point of view of burn-up.

Neutronics studies of a small carbide core and the possible nominal core configurations for the Fast Breeder Test Reactor (FBTR) were carried out. An error analysis for the effective multiplication factor calculations of FBTR was also carried out.

5.3 Fast Reactor Safety Analysis

Improved models for the predisassembly phase calculations of accident analysis were incorporated in the computer codes and the DBA studies using these improved models were made for FBTR.

LOFA and TOPA analyses were carried out for the PFBR also with oxide and carbide fuel, using the improved models in the codes.

Dynamics of fuel vapour pressure saturation in the voided LMFBR cores during transient heating in a DBA were studied and the effect of pressure lagging the temperature on the energy release during the disassembly phase of accident has been evaluated.

A study of the sensitivity of heat transfer parameters on sodium boiling propagation, fuel slumping, and subsequent effects in LMFBRs under unprotected loss of flow accidents (LOFA) has been carried out.

A multizone fuel coolant interaction (FCI) model has been developed which includes the heat exchange between the interaction zones and leakage to the surroundings. The formulations were developed in Lagrangian system and a code MULTIFCI has been written and tested successfully.

5.4 Reactor Noise Analysis

Investigations of the reactivity fluctuations caused by the boiling of sodium in LMFBRs have been carried out with regard to the likely frequency range in which the local boiling may be reflected under LMFBR operating conditions and the minimum detectable size of the sodium vapour bubble by such an analysis.

A study of neutron noise transmission characteristics of multiplying media and neutron noise source localisation in LMFBRs has been carried out using neutron wave propagation technique.

5.5 Radiation Transport and Shielding

A two dimensional analysis of the Japanese shielding benchmark experiment for sodium was carried out using the discrete ordinate transport code DOT in S_4P_0 approximation. The neutron fluxes and the gamma doses were compared with the measured values.

A series of computational studies for the preliminary design of the internal bulk shields for the PFBR were carried out. These included the radial shields, upper axial integrated shields and the Monte Carlo simulation of neutron streaming through the lower axial shield and the grid plate.

The effect of scattering anisotrophy on the optimal choice of parameters for exponential transform in deep penetration Monte Carlo simulation has been studied. It is found tnat the scattering anisotropy influences significantly the optimal parameter if the scattering probability is high.

An optimising scheme in which the selection of a biassing parameter is automatically adjusted for simulating neutron and gamma transport in a practical shield problem, has been formulated and its performance has been studied against a sodium shielding benchmark experiment.

6.0 Materials Development Programme

Materials Development Laboratory of Reactor Research Centre is concentrating its efforts on generation of data on materials of direct interest to the Fast Reactor Programme. Some idea of the activities of this laboratory can be obtained from the following programmes of the laboratory.

- i) Studies on compatability of austenitic stainless steels with static and dynamic sodium.
- ii) Studies on pitting and crevice corrosion of stainless steels in aqueous solution containing chloride ions.
- iii) Studies on sensitization of stainless steels.
- iv) Studies (by simulation) on the clad/fission products interaction
- v) Studies on atmospheric corrosion of stainless and low alloy steels.
- vi) Studies on oxidation behaviour of Cr-Mo ferritic steels.
- vii) Study of phase separation techniques in stainless steel weldments and kinetics of the growth of secondary phases.
- viii) Studies on stress corrosion cracking of stainless steels.
- ix) Influence of grain size on the high temperature mechanical properties of austenitic stainless steels.
- x) Correlation between mechanical properties and microstructure of Nimonic PE-16 superalloy.
- xi) Studies on low cycle fatigue behaviour of reactor structural materials.
- xii) Testing and evaluation of FBTR materials for creep, tensile and impact properties.
- xiii) Evaluation of the efficiency of various stress rupture parameters.
- xiv) Influence of cold work on the stress rupture properties of 316 stainless steel.
- xv) Evaluation and testing of 316 stainless steel plates from ASP, Durgapur for suitability for fast reactor.
- xvii) Influence of environment on creep rupture properties of type 304 stainless steel.
- xviii) Linear elastic and general yielding fracture mechanics studies on reactor materials.

Significant results obtained can be summarised as:

6.1 Influence of Grain size on Crevice corrosion of Austenitic Stainless Steels

The critical crevice potential, E_{cc} and average grain diameter (d) for type 316 stainless steel in 0.5 M NaCl solution have been found to fit the following two equations:

i) $E_{cc} = 475\text{-}99\ d^{-\frac{1}{2}}$; Correlation coeff. = -0.95

ii) $E_{cc} = 297\text{-}13\ d^{-1}$; Correlation coeff. = -0.94

A sample having grain dia as large as about 5 mm was prepared and the E_{cc} measurement was carried out on a single grain (no grain boundary under the crevice). E_{cc} value was obtained as +250 mV (SCE) in this case. This value should correspond to the intercept at d=0 in theabove equations. As this value is quite close to the intercept in equation (ii), it seems that this equation would be more appropriate than the equation (i) even though it had slightly lower correlation coefficient.

6.2 Pitting and Crevice Corrosion Studies on 'SEA-CURE' Material

A ferritic stainless steel with trade name 'SEA-CURE' (Chem. comp. in wt. % : C = 0.15, Mn = 0.35, P = 0.21, S = 0.004, Si = 0.20, Ni = 2.06, Cr = 27.16, Mo = 3.36, N = 0.022, Ti = 0.54, Co = 0.15, Al = 0.80 and Fe = balance), a new material developed for condenser tubes in sea water applications has been evaluated for resistance to localised corrosion attack. A few potentiodynamic experiments conducted on this material in 0.5 M NaCl solution at room temperature have shown that it is immune to pitting and crevice corrosion attack. Such a high resistance against localized corrosion is considered to be mainly due to higher Cr, Mo and N contents in the alloy.

6.3 Development of TTS diagrams for AVESTA Stainless Steel Material at Different Cold Work

In order to develop time-temperature sensitization (TTS) diagrams for AVESTA 304 stainless steel material at various degrees of cold work, samples were cut from 7 mm thick plate (in mill-annealed condition) and were cold rolled to various reductions in thickness to achieve 0, 5, 10, 15, 20 and 25% cold work. Samples were heat treated between 823 and 973 K for various time periods ranging from 0.25 to 200 h. Intergranular corrosion tests are being carried out as per ASTM A-262 practice E.

6.4 Microstructural Instability of Type 316 Stainless Steel Weld Metal at High Temperatures

The aged samples were tensile tested at room temperature using a strain rate of 4 x 10 s^{-1}. The results obtained are tabulated below:

Ageing Temp. (K)	Ageing time (h)	Y.S (MPa)	UTS (MPa)	Elongation (%)
As welded		330	499	37.2
	200	323	515	33.5
773	2000	332	516	31.0
	5000	336	543	33.0
	20	292	561	25.6
873	200	281	577	26.6
	2000	340	599	19.2
	5000	315	581	16.8
	20	321	618	21.8
973	200	308	597	13.1
	2000	317	600	9.7
	5000	308	622	10.4

Fractographic studies are in progress.

6.5 Characterisation of Precipitates

The aged samples were dissolved in HCl-CH_3OH electrolyte at 1.5 V to separate the secondary precipitates from the matrix. The extracted precipitates were then analyzed, using x-ray diffraction to identify the phases present. The data are tabulated below:

Ageing Temp. (K)	Ageing Time (h)	Total Weight % ppt.	Phases present
As welded		6.2	delta ferrite
	200	*	*
773	2000	0.87	delta ferrite, sigma, $M_{23}C_6$, Chi
	5000	1.67	delta ferrite, sigma, $M_{23}C_6$, Chi
	20	4.30	Chi, sigma, $M_{23}C_6$
873	200	5.80	Sigma, $M_{23}C_6$
	2000	6.31	Sigma, $M_{23}C_6$
	5000	8.74	Sigma, $M_{23}C_6$
	20	8.04	Sigma, $M_{23}C_6$
973	200	8.63	Sigma, $M_{23}C_6$
	2000	11.18	Sigma, $M_{23}C_6$
	5000	14.73	Sigma, $M_{23}C_6$

*Analysis in progress.

Chemical separation of the phases is in progress to assess the growth kinetics of sigma and $M_{23}C_6$ carbide phases.

6.6 Oxidation Studies

Oxidation studies on 2¼ Cr - 1 Mo steel and 9Cr - 1Mo steel have been carried out in air and oxygen environment in the temperature range of 773-1273K for a fixed duration of 6h. Following results were obtained:

i) Rate of oxidation of 2¼ Cr-Mo Steel increases with increase in temperature in both the environments. The parabolic rate constant increases with increase in temperature upto 1073K. At 1173K the steel undergoes breakaway oxidation. There is some difference in the oxidation kinetics in the two environments. The reason for this difference can be established by carrying out further experiments.

ii) Rate of oxidation of 9Cr-1Mo steel is about an order of magnitude less than that of 2¼ Cr-1Mo steel in the temperature range 773-1173K. Moreover the oxidation kinetics is logarthmic in nature as compared to the parabolic one in 2¼ Cr-1Mo steel. The breakaway oxidation in 9Cr-1Mo steel occurs at 1223K and at 1173K in 2¼Cr-1Mo steel. The oxide layer formed upto 873K seems to be iron rich and Cr_2O_3 forms at 973-1073 and 1173K. The oxide layer formed at 1273K after breakaway oxidations seems to be similar to that formed on 2¼Cr-1Mo steel after breakaway oxidation. There was no significant difference between the two environments on the oxidation behaviour for this alloy.

6.7 Frequency effects on 10% Cold Worked AISI 304 Stainless Steel in High Strain Fatigue

Low cycle fatigue (LCF) studies on prior cold worked (PCW) AISI 304 stainless steel at a cycling frequency of 0.1 Hz have shown that 10% pcw condition gives a lower bound endurance. Hence, further investigations were carried out in 10% pcw condition to determine the influence of frequency in the range 0.001-1.0 Hz. The total axial strain ranges employed in testing were in between 0.5 and 1.2%. The preliminary results obtained indicate the following:

i) Fatigue life decreases with a decreasing frequency upto 0.01 Hz at all strain ranges with a drastic life reduction at lower strain range (0.5%).

ii) For a strain range of 1.2% testing with a lower frequency of 0.001 Hz resulted in increase in fatigue life from that observed at a frequency of 0.01 Hz indicating a recovery in fatigue life.

iii) Number of cycles for saturation have shown the same trends exhibited by fatigue life with a decreasing frequency.

iv) In general, the tensile stress amplitude for the first cycle and the saturation stress amplitude at half the separation life have been found to decrease with decreasing frequency.

v) Plastic strain developed during first cycle and plastic strain computed at saturation level were found to be minimum for a cycling frequency of 0.1 Hz, with an increase in both these values at frequencies lower than 0.1Hz.

vi) Dynamic strain ageing has occured on testing at and frequencies lower than 0.01 Hz.

6.8 Creep Fatigue Interaction Studies on AISI 304 Stainless Steel

Creep fatigue interaction tests at 923K on coarse grained AISI 304 are under progress. The introduction of one minute hold period at maximum tensile strain ($\pm$1.0%) into the fatigue cycle reduced the fatigue life considerably from that has been recorded during continuous cycling tests. Increasing the dwell period to 10 minutes resulted in an increase in fatigue life over that observed at shorter dwells. Preliminary examination of the fracture surfaces have shown pure wedge type inter granular cracking at shorter dwells and a mixed wedge and cavitation type intergranular cracking under 10 minute hold time conditions.

6.9 Exploration Test for Determining Peak Strain Ageing in Low Cycle Fatigue

An exploration test for possible cyclic strain ageing interactions was deviced by means of determining incremental temperature effects. This peculiar test was accomplished at a single strain range, at sufficiently low frequency and on a single test specimen by cycling at a specific temperature for ten complete strain cycles and then raising the temperature incrementally to higher levels. Starting at a temperature of 673K and raising the temperature in 50 degree increments, the cyclic strain ageing spectrum in the form of hysterisis loops has been established over the temperature range of interest; and the temperature of maximum cyclic strain ageing is quickly determined. Further tests have also been carried out to cyclic saturation on companion specimens to determine the peak strain ageing effect on 75 m grain size 304SS samples employing a strain range 0.8% and a cycling frequency of 0.01 Hz. The salient features that can be obtained from these tests conducted to saturation can be seen from the following results:

Temp. (K)	Stress Amplitude (MPa) I Cycle	Stress Amplitude (MPa) Saturation	Plastic strain range (%)
673	131	248	0.459
773	143	270	0.414
823	173	278	0.453
873	160	228	0.498
923	159	224	0.523

It is clear that there was a pronounced increase in stress amplitudes for the first and saturation cycles with associated minimum in plastic strain per cycle in the peak strain ageing temperature regims.

6.10 Effect of Thermal Ageing on the Room Temperature Tensile Properties of an AISI 316 Stainless Steel

The effect of thermal ageing at the temperatures of 823, 923, 1023 and 1123K for periods ranging from 10 to 1000 h on the room temperature tensile properties have been investigated:

i) Optical metallography revealed that qualitatively there was a considerable variation in the matrix and grain boundary precipitation density with variation in both time and temperature.

ii) Yield stress decreased with increase in ageing temperature for any fixed level of ageing time; the rate of decrease of yield stress was greater for longer ageing periods.

iii) On ageing at temperatures of 823 and 923K yield stress increased with increasing ageing time whereas ageing at higher temperatures showed a reverse tendency.

iv) UTS was also found to decrease with increase in ageing temperatures. The rate of decrease was considerably larger for ageing periods of 500 and 1000 hrs compared to lower ageing periods.

v) Ductility at room temperature measured in terms of either uniform elongation, total elongation or reduction in area decreased with ageing time at fixed ageing temperatures.

vi) SEM studies have shown purely intergranular fractures at highest ageing temperature and time and is attributed majority to the heavy sensitization.

6.11 Effect of Prior Cold Work (PCW) on Creep Properties of AISI 304 SS

The creep tests in air on solution annealed and 10% PCW specimens at 973K and at stresses ranging from 120-200 MPa are in the advanced stages of completion. Test at 973K for 20% PCW condition with a stress level of 160 MPa has been completed. The results analysed so far for a stress level of 160MPa reveal that rupture life increases, rupture ductility and minimum creep rate decreases with increasing prior cold work. The studies on substructure and creep damage for different testing conditions are planned.

6.12 Creep Tests on Heat E5158 of VIRGO 14SB Material

Creep rupture tests have been completed at 923K and at a stress level of 200 MPa. Creep rupture tests at a stress level of 150 MPa are in progress.

7.0 Chemistry Programme

7.1 Characteristics of Reactor Grade Sodium

Reactor grade sodium, prepared in the Reactor Engineering Laboratory by purifying commercial sodium, has been characterized by chemical analysis employing the various techniques developed earlier. It was found that the purified sodium meets the specifications for reactor applications.

7.2 Development of On-Line Meters

Development of on-line meters for oxygen, carbon and hydrogen was continued. Oxygen meters based on yttria-doped thoria (YDT) solid electrolytes have been used, calibrated against cold trap temperature in small sodium loops and have been found to give near theoretical response. They have also been used to measure the oxygen potentials in the $Na-NaCrO_2-Cr$ and the $Na-Na_3UO_4-UO_2$ phase fields.

An electrochemical hydrogen meter has been fabricated and tested in a small pumped sodium loop. It is based on a $CaCl_2$-5 wt % CaH_2 electrolyte contained in an iron membrane capsule dipping in sodium which forms one electrode. The reference electrode is Li/LiH contained in another thin-walled iron tube which is located in the electrolyte. This meter has shown near theoretical response and a sensitivity of 0.15 pcm in hydrogen concentration.

Further studies with the electrochemical carbon meter described earlier has shown that equilibria involving carbon in sodium steel systems are rather sluggish.

7.3 Solubility Measurements

The solubilities of Molybdenum and Vanadium have been measured in oxygen free liquid sodium in the temperature range of 500-720K. The solubility of molybdenum can be given by the equation.

$$\log S(ppm) = 2.738 - \frac{2200}{TK}$$

The heat of solution of molybdenum in liquid sodium is calculated to be -42.123 KJ/mole. The measured solubility of vanadium shows a sharp fall above a temperature of $400^{o}C$, probably due to the formation of intermetallics.

Appropriate analytical techniques based on atomic absorption spectrophotometry were developed for determing traces of these elements in sodium.

7.4 Fluence Measurement

A very accurate method has been developed for determining the isotopic composition of boron by thermal ionization mass spectrometry. In this technique not only is the isotopic fractionation effect minimised, but correction for this is in-built. Using this technique, it would be possible to measure the fluence seen by cladding and other structural materials from the isotopic compositions of the boron impurity in them.

7.5 Thermodynamics of Fast Reactor Fuels

As part of a programme to generate thermodynamic data on fission product compounds which are likely to be found in the mixed oxide fuel under irradiation, compounds such as $BaZrO_3$ and $SrZrO_3$ were stydied using high temperature calorimetry. A detailed study of $(U, Ce)O_{2+x}$ is under way using oxygen potential measurements, thermal analysis and x-ray diffraction. In order to assess the role of fission product tellurium in fuel-clad chemical interaction, the iron-tellurium system has been investigated using Knudsen cell mas spectrometry.

8.0 Reprocessing Programme

Development of process equipment system, such as single pin chopper, solvent contractors, feed clarifiers, dissolver etc. has been completed. Presently they are tested for their reliability in an integrated operation on one cycle of solvent extraction using U. Solvent extraction models were developed for the U-Pu-TBP system. Computer studies are carried out for optimisation of flow sheet. Experiments on dissolution of U carbide in nitric acid indicate that the organics could be mostly oxidised electrolytically and the product solution has characteristics suitable for solvent extraction subsequently. The electrolytic system was tested for its viability on experiments with dissolution of sintered Pu oxide also and it was found to have favourable characteristics. The Alpha laboratory was commissioned and is engaged in collection of relevant solvent extraction equilibrium data for flow sheet studies and application of electrolytic process for separation of U and Pu. Lead shielded facility is ready for commissioning.

Design work is in progress on setting up of a pilot plant for reprocessing of fast reactor fuel on 15 to 20 Kg per day.

9.0 Safety Research Laboratory

Release characteristics of fission product iodine in fires involving primary sodium have been studied, using sodium spiked with radioiodine. Effect of relative humidity on autoignition temperature of liquid sodium had been studied. Laboratory scale experiments for the benign digestion of sodium in waste matrix have been successfully completed. Work is in progress to adopt this technique for digestion of sodium in the range of 10 to 20 Kg encountered in the normal operations.

In the area of fuel coolant interactions, techniques for melting UO_2 under sodium have been developed.

Detailed calculations of build up factors in lead, iron, water and other material were done.

A non destructive method of estimating water content in concrete using the back scattered radiation spectra was developed. A detailed analysis of the data generated in the studies on the use of rare earth oxide in concrete reactor shields was carried out to complete the IAEA Research Contract work.

A state of art review was carried out on fission product release and transport following loss of coolant accidents in water reactors.

A microcomputer based simulator is being built for the purpose of training the staff for handling any emergency involving the release of radioactivity to the environment from nuclear facilities

A REVIEW OF THE UNITED KINGDOM FAST REACTOR PROGRAMME – March 1984

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ABBREVIATIONS USED IN THIS REVIEW

AEA	United Kingdom Atomic Energy Authority
AEEW	Atomic Energy Establishment, Winfrith
AERE	Atomic Energy Research Establishment, Harwell
AGR	Advanced Gas Cooled Reactor
AKUFVE	Apparatus for continuous study of distribution factors in liquid/liquid extraction (Swedish acronym)
ASD	Alternative Shutdown Device
ASME	American Society of Mechanical Engineers
BCD	Burst Can Detection
BNL	Berkeley Nuclear Laboratories (CEGB), Gloucester
BNFL	British Nuclear Fuels PLC
CDFR	Commercial Demonstration Fast Reactor
CEA	Commissariat à l'Énergie Atomique
CEGB	Central Electricity Generating Board
CFR	Commercial Fast Reactor (The future series of LMFBR following CDFR)
COGEMA	Compagnie Generale des Matieres Nucleaires
CSNI	Committee on the Safety of Nuclear Installations
CTS	Central Technical Services, RNE
c.w.	Cold Worked
DFR	Dounreay Fast Reactor
DNE	Dounreay Nuclear Power Development Establishment
DPA	Displacements per Atom
EDTA	Ethylene Diamene Tetra Acetate
efpd	Effective full power days
ESR	Electro-slag refined
ERA	Electrical Research Association
FRCC	Fast Reactor Co-ordinating Committee, EEC
HAZ	Heat Affected Zone
HCDA	Hypothetical Core Disruptive Accident
HM	Heavy Metal
IHX	Intermediate Heat Exchanger(s)
IRD	International Research and Development Co, Newcastle
ISAT	Individual Sub-assembly Temperature Monitor
JEF	Joint Evaluated File
KfK	Kernforschungszentrum, Karlsruhe
MCTR	Mechanical Components Test Rig
MEL	Marchwood Engineering Laboratory, CEGB
MFCI	Molten Fuel/Coolant Interaction
MFTF	Molten Fuel Test Facility, Winfrith
MI	Mineral Insulated
MIG	Manual Inert Gas (Welding)