IGC newsletter

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## From the Director's Desk

# History and Evolution of Fast Breeder Reactor Design in India

## - A Saga of Challenges and Successes

INSIDE



Nuclear energy is an inevitable source to meet the fast growing energy demands of India and commitment to provide better quality of life to all the citizens. Presently known reserves of uranium in the country can support an installed capacity of about 10 GWe for about 30 years, based on thermal reactors, without recycling the plutonium produced. However, the Fast Breeder Reactor (FBR), in view of its capability to extract at least sixty five times more energy from the same resource, can increase nuclear energy many fold, by contributing about $1/4^{th}$ of the projected energy requirement by 2050. FBR is also an enabling technology to make a transition to large-scale utilisation of thorium resources available in the country.

Realising this, DAE had started FBR programme as early as 1965, by forming a Fast reactor section at BARC under Shri S.R. Paranjpe and preliminary design of a 10 MWe experimental fast reactor was initiated. Dr.Vikram Sarabhai, who took over as Chairman AEC in 1966 took a bold and decisive step to speed up the introduction of the fast reactor technology in India, through effective collaboration with a country having design, construction and operating experience in FBRs. At that time, France, having focused fast reactor programme and an experimental reactor, Rapsodie-Fortissimo, offered to collaborate with India. As the offer was considered attractive, a bilateral agreement was signed in 1969. Under this collaboration agreement, a design team consisting of both French and Indian Engineers was constituted for preparing a preliminary design and a project report for an experimental test reactor (FBTR), to be constructed in India, that could be used as a test bed for materials (i.e. fuel, cladding, etc.) as well as human resource development. A design team of 31 engineers and scientists, including draftsmen under the leadership of Shri. Paranjpe worked at the Cadarache Nuclear Centre in France, for over 15 months to complete the preliminary design. For executing the project at Kalpakkam, a group of about 50 engineers and draftsmen with Shri. Paranjpe as the Principal Design Engineer was shifted from BARC to then RRC in June 1971. The group consisted of S/Shri M.C.Sabherwal (Senior design Engineer-Circuits), S.B.Bhoje (Senior Design Engineer-Block Pile), K.Raghavan (Senior Design Engineer–Control and Instrumentation), M.K. Ramamurthy (Senior Design Engineer-Fuel Handling), S.A.Welling (Senior Design Engineer–

Computer Data Processing System), R.Shankar Singh (Senior Design Engineer-Reactor Physics), R. Subramanian (Senior Design Engineer - Reactor Chemistry), S.S.Kumar (Senior Design Engineer – Reactor Operation Studies) and R.Seshadri (Plant Layout).

The design of FBTR was largely based on RAPSODIE in its primary coolant (sodium) circuit. In RAPSODIE, the nuclear heat was dissipated to atmosphere through sodium to air heat exchanger which was modified in FBTR by adding 4 modules of steam generators (SG) and turbogenerator (TG) in the secondary sodium circuit, similar to 250 MWe PHOENIX reactor which was under construction then in France. The construction of FBTR started in 1972 with Shri N.L. Char as the Principal Project Engineer. All the components except grid plate, one control rod drive mechanism and one primary sodium pump were successfully manufactured indigenously for the first time in the country. A few design modifications in the nozzle and thermal shield at the reactor vessel outlet, clad rupture detection circuit and instrumentation, were incorporated, subsequent to two consecutive leaks reported in the primary circuit of RAPSODIE. FBTR was successfully commissioned in 1985. The criticality and related parameters were predicted very closely for the high plutonium content carbide, even with no experimental support. The credit goes to the Reactor Physics Division, then headed by Shri.R. Shankar Singh with significant contributions made by Dr. S.M. Lee. So far FBTR has completed 20 years of successful operation. Mark-I carbide fuel has achieved a burn-up of 155 GWd/t and a maximum linear heat rate of 400 W/cm without a single clad failure in the whole core. Sodium pumps have shown excellent performance for more than 540,000 cumulative hours of operation. Steam Generator performance has been excellent without incident of any tube leak. FBTR has been a fountainhead of nurturing of competent personnel for the design, construction and operation of subsequent FBRs. The history of IGCAR and FBTR has already appeared in IGC newsletter (volume 61, July 2004 & volume 62, October 2004).

Based on the experience in the design and construction of FBTR, DAE decided to launch the design of 500 MWe Prototype Fast Breeder Reactor (PFBR). For drawing up a plan for setting up PFBR, a Steering Group consisting of Shri P.R. Dastidar, Dr.M.R. Srinivasan and Shri N. Srinivasan was constituted in December 1979 by Dr.Raja Ramanna, the then Secretary, DAE. The Steering Group submitted its report in July 1980 and supplementary report in November 1980. Based on the recommendations of the Steering Group, the Secretary, DAE constituted a PFBR Working Group (PFBR-WG) in March 1981 to proceed with the work of preparation of a feasibility report for PFBR. The WG consisted of Shri N.L. Char (Chairman), Shri S.R. Paranjpe, Shri K.V. Mahadeva Rao, Dr. Anil Kakodkar (current Chairman, AEC) and Shri S.B.Bhoje (Member-Secretary).

FBTR has been a fountainhead of nurturing of competent personnel for the design, construction and operation of subsequent FBRs.

The design proposed by WG in 1983 was basically a 500 MWe capacity prototype reactor, with oxide fuel, 4 primary loops having 4 pumps and 4 secondary loops having 8 intermediate heat exchangers (IHXs) and 12 numbers of steam generator units with each unit comprising of separate evaporator, super heater and reheater module. The core outlet temperature was chosen as 530°C with the design plant life of 30 y, based on the confidence in the high temperature design prevailing at that time. Identifying areas for Research and Development (R&D), technology and manufacturing, WG suggested to plan PFBR on the assumption that external assistance would not be available and R&D should be planned as a national effort. Hence, the participation of capable organizations in the country that could make significant contributions to the FBR programme was foreseen. It was envisaged that the role of IGCAR should be principally to carry out the basic design and R&D. This apart, it should also coordinate R&D and technological development carried out as collaborative projects with industrial units and educational institutes, which would result in a significant reduction of time in evolving designs and manufacturing processes. The WG submitted the feasibility report and preliminary design report in June 1983.

From 1983, design activities were undertaken towards preparing the Detailed Project Report (DPR), under the guidance of Shri S. R. Paranjpe, the then Director, Reactor Group. To accomplish the detailed design activities related to nuclear steam supply system, Nuclear Systems Division (NSD) was formed in 1985 with Shri S.B. Bhoje, as the Head of the Division. In this division, S/Shri A. Selvaraj, S.C. Chetal and B.S. Sodhi were the section heads to look after the design of reactor assembly including core, sodium circuits and fuel handling systems. Realising the need for in-depth analysis of reactor components, two specialist sections were also created: Thermal Hydraulics Section with Shri.G.Vaidyanathan and Structural Mechanics Section initially with Shri.K.K. Vaze and later with Dr.P. Chellapandi.

Parallely, in order to raise the confidence in the operation of mechanisms in sodium and also to understand the complex phenomena such as heat & mass transfer in the cover gas, a few dedicated test loops were commissioned; the largest one commissioned in 1994, called Large Component Test Rig (LCTR), is being used effectively for the qualification of prototype components. Shri R.D. Kale and Shri M. Rajan have been the key persons in developing such unique facilities.

Based on the analyses and validation, in conjunction with the qualification of critical components by testing, design of all the major components was completed and the detailed project report was submitted to the Department in 1989 for the consideration of financial sanction. However, it was decided by Dr.P. K.Iyengar, the then Secretary DAE to examine policy issues of PFBR project. A committee headed by Shri.S.K. Mehta with Shri.S.K. Chatterjee, Dr. Anil Kakodkar, Shri. S.B. Bhoje, Dr. S.M. Lee and Shri. S.C. Chetal as members was constituted in 1992 to assess indigenous manufacture of PFBR components, possible technological improvement in PFBR design, comparative cost of PFBR & PHWR and time frame in which PFBR could be commissioned. The committee submitted its report in 1993. The committee recommended that the cost comparison with PHWR should not be the main consideration for the sanction of the PFBR as the introduction of FBR is considered to be important to maintain the steady growth of contribution by nuclear energy for power generation. It recommended constructing the PFBR with proven mixed oxide fuel and the main objective of PFBR should be techno-economic demonstration. The committee recommended that the efforts should be made to reduce the cost. Further, the fuel for PFBR was envisaged to be available by 2005 at that time. In conclusion, the committee emphasized the need to complete the detailed engineering design of PFBR by end 1995; the R&D activity should be scheduled in a manner that all inputs are available in time for completion of detailed engineering and for safety analysis. With the above objectives, further in-depth studies were undertaken by the design team beginning from 1992.

> Continuation of R&D towards ensuring the robustness of design, definition of manufacturing tolerances based on rational approach, design and R&D support to BHAVINI.

In 1997 when Shri S.B. Bhoje took over as Director, Reactor Group, Design and Technology Group (DTG) was formed with Shri.S.C. Chetal, as Associate Director to undertake the design and technology activities together. In order to have close co-ordination between thermal hydraulics and structural mechanics analysis sections and also to carryout the analysis activities with enhanced emphasis on in-depth understanding of mechanics, by involving outside R&D establishments and academic institutions, Mechanics and Hydraulics Division (MHD) was formed with Dr. P. Chellapandi as its head. When Shri S.B. Bhoje became Director of the Centre in 2002, Shri.S.C. Chetal took over as Director - Reactor Engineering Group in 2002 to integrate design and manufacturing technology development of the entire plant including nuclear, civil, electrical, instrumentation & control and balance of plant aspects. From 1993, activities have been focused to improve the conceptual design to make it economically competitive, (which has resulted in a 2-loop concept with significant design changes) and to carryout necessary R&D activities. The activities carried out during the period 1993 to 2002 formed the basis for preparing the revised Detailed Project Report (DPR), which was submitted during 2002. Government of India accorded administrative approval and financial sanction in September 2003. Since then, continuous design support is being provided to BHAVINI for successful construction and commissioning of PFBR.

The Design Group activities cover a wide spectrum of domains starting from: (i) design modifications introduced in FBTR during 1972-85, (ii) conceptual design of four loop concept of PFBR, development & application of analysis tools & methodologies and preparation of safety criteria (1985-1993), (iii) conceptualisation and detailing of two loop concept towards improving economy, design confirmation and validation, consolidation of design criteria for core and sodium coolant systems and components, preparation of preliminary safety analysis reports (PSAR), technology development of large size components, construction of A dedicated structural mechanics laboratory, definition & execution of R&D activities both in-house and at outside institutions, incorporation of outcome of completed R&D results in the design, preparation of supporting documents for obtaining financial sanction and preparation of tender documents for long delivery components (1993-2002), (iv) continuation of R&D towards ensuring the robustness of design, definition of manufacturing tolerances based on rational approach, design and R&D support to BHAVINI, study of innovative design features for FBRs beyond PFBR and preliminary design and analysis for the metallic fuels (since 2004). I would like to highlight some of the important and challenging activities, accomplished by the group.

The four-loop design as conceived by WG in 1983, was basically intended to demonstrate technological feasibility. After 1993, demonstration of technology capability, and enhanced economic feasibility became the focus of the efforts. A task force was constituted with Shri.A. Selvaraj (we lost Shri.Selvaraj during Tsunami in Kalpakkam on 26th December 2004), then Head-Reactor Assembly Section to recommend economically competitive conceptual design for further detailing. The task force through a systematic study, identified various factors affecting the economy and optimised them towards essentially minimizing the steel consumption per MWe power. Accordingly, the number of loops and components were reduced giving due considerations to the international and Indian operating experiences of sodium systems and components. Certain important conclusions of the study were: the capacity factor is not unduly influenced by a large number of components, loops and sodium pumps; are highly reliable and hence a minimum number can be deployed, A minimum of two IHXs per primary pump is required for better pool thermal hydraulics and more SGs are required for the higher plant availability. Further, the main vessel diameter, which is one of the key factors influencing the economy, is to be minimised by the optimum choice of sizes of IHX and pumps and adopting an innovative concept for the in-vessel fuel handling system. Based on these, the task force recommended a 2-loop concept with 2 primary pumps, 4 IHXs and 2 secondary loops with one pump and 4 integrated once through steam generator modules in each loop. The dimensions of primary pump were optimized by eliminating the non-return valve and reassessment of cavitation margin. For the IHX, diameter to height ratio was chosen based on a systematic parametric study to arrive at the minimum diameter. In addition, the tube size was also optimised to arrive at 19 mm outer diameter with 0.8 mm thickness. For the component handling system, a transfer arm concept with a fixed offset of about 530 mm was adopted. Realising that shielding design has a critical bearing on safety and economics of FBRs, a series of experiments with increasing complexity were conducted jointly by IGCAR and BARC at APSARA reactor. Based on the detailed analysis of the test results, the axial, radial and ex-vessel shielding were optimized. It is worth mentioning the role played by Dr. Anil Kakodkar, the then director BARC, in giving priority to the experiments in APSARA reactor. All these efforts helped to reduce the main vessel diameter from 15 m to 12.9 m.

Compact plant layout has a major influence on the safety and economy. Towards achieving this, a concept of inter-connected building for the nuclear island was adopted.

Towards improving thermodynamic efficiency, the plant parameters such as operating temperatures were critically investigated and optimised on the basis of extensive inelastic analysis employing realistic and validated material constitutive models, operating experiences & trends in the international reactors, choice of structural materials and economic considerations. These resulted in raising the core outlet temperature from 530$^0$C to 547$^0$C and consequently, the steam temperature from 480$^0$C to 490$^0$C. In addition, the design plant life was enhanced from 30 y to 40 y. In the process of cost optimisation, a balanced perspective was adopted in choosing steam reheat cycle in place of sodium reheat cycle, which has a marginal advantage in economy and ease of operation with a slight decrease in the thermodynamic efficiency.

The materials were also chosen carefully to achieve economy. The austenitic stainless steel 316 LN was chosen for the main vessel, hot pool components and hot piping; 304 LN for components and piping operating at steady state temperature less than 400$^o$C and modified 9 Cr-1Mo in place of 21/4 Cr-1 Mo for SG. In order to achieve higher burn-up up to 100 GWd/t, an improved stainless steel with Ti addition (D9) having a high swelling resistance and creep behaviour was chosen as the principal material, over SS 316 for the structural materials of fuel subassemblies.

Compact plant layout has a major influence on the safety and economy. Towards achieving this, a concept of inter-connected building for the nuclear island was adopted. This enhances seismic integrity and economy by minimizing the concrete consumption and reducing piping and cable lengths.

In order to raise the confidence in the designs evolved through economic optimisation studies, many challenging issues, in the domains of reactor physics & shielding, thermal hydraulics, structural mechanics including seismic issues, instrumentation & control, sodium and fuel chemistry, materials and metallurgy, NDE including ISI and safety were addressed. For the detailed analyses in the thermal hydraulic and structural mechanics domains, a team of specialists developed many sophisticated software packages. To specify a few, computer codes called FUSTIN to numerically simulate the mechanical consequences of CDA, CONE to carryout vicsoplastic analysis of high temperature components and THYC to study the complex pool hydraulics are of truly international quality. Further, codes were developed for plant dynamic studies viz. DYNA-P & DHDYN and SWEPT for predicting the pressure transients due to sodium water reaction which is one of the critical aspects, considered in FBR design. These codes play very important role for ensuring that the temperature limits for the fuel, clad and coolant are not exceeded. The codes also provide input data for the subsequent structural analysis to demonstrate structural integrity under all the design basis events. Apart from the computer codes developed in-house, a few sophisticated computer codes such as CASTEM2000, PLEXUS, TEDEL, BILBO, INCA and ALICE developed by CEA France were also used for completing many challenging and advanced analyses. Similarly, well-established commercial codes viz. PHOENICS and STAR-CD have added high capability to carryout many state-of-art CFD analyses. With the availability of such sophisticated codes, the computational capability at the Centre in the domain of structural mechanics and thermal hydraulics has reached international mark.

Towards structural design validation of components, many experiments were carried out. To accomplish this, a dedicated state-of-art structural mechanics laboratory was established to carry out specialized experiments in the domain of high temperature design, seismic and fracture mechanics and buckling investigations. Similarly, to validate the thermal hydraulic aspects such as flow distribution in and around IHX inlet regions, gas entrainment, free level fluctuations, thermal stratifications, fluid-elastic instability, etc. that are associated with the pool hydraulics, experiments in water with scaled down models were completed. In the validation exercise, about 40 R&D establishments and academic institutions were extensively involved by way of establishing about 80 collaborative projects related to reactor engineering aspects. Apart from these, academic institutions have contributed through more than 700 BE/ME/PhD student projects.

Further, I would like to specifically mention about the history of safety review of the design. Safety Criteria was evolved in parallel, as the design was proceeding and the then DAE-SRC constituted a committee to finalise the safety criteria. The eight-member sub-committee with Shri S.B. Bhoje as chairman had contributed towards this activity immensely. The accumulated reactor-years of operating experience in sodium cooled fast reactors was taken into account and IAEA guide 50-C-D (1988) was taken as a guideline document. With valuable inputs from IGCAR design team as well as from BARC and NPC, AERB issued a comprehensive safety criteria in 1990.

AERB constituted Preliminary Design Safety Committee (PDSC) for PFBR in July 1988 under the chairmanship of Dr. Anil Kakodkar (currently Chairman, AEC) with 6 other members for reviewing the 4-loop concept. Dr.Anil Kokodkar has provided excellent decisive inputs at all stages of PFBR culminating in its financial sanction and subsequent construction and is now providing guidance for the FBR programme beyond PFBR. In 1995, the committee was reconstituted with Shri G.R. Srinivasan as Chairman, to review the 2-loop design that formed the basis for obtaining financial sanction. The reconstituted PDSC reviewed the conceptual design from point of view of safety and operation and provided valuable inputs to AERB for clearance for manufacture of long delivery nuclear components and civil excavation. In March 2004, the committee was again reconstituted under Shri S.S. Bajaj, for in-depth review of detailed analysis results and R&D aspects. To support this committee, 16 specialist groups have been formed in 2006 and review is in-progress. In order to review aspects very specific to sodium cooled fast breeder reactor systems, a concept of internal safety committee (ISC) was introduced. The ISC consists of experts who have considerable experience in design, construction, commissioning and operation of FBTR and test loops at IGCAR. Dr.S.M. Lee, Shri.R.D. Kale, Dr.S.L. Mannan have been the chairman of ISC. Recently, ISC has been reconstituted with Shri.M. Rajan, Director, Safety Group, as Chairman in May 2006.

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The Design Group has demonstrated the adequacy of design and safety margins through a systematic and comprehensive presentation of design concepts, design approach and data bank. The extent of review can be gauged from the fact that PDSC and ISC had met 63 and 93 times respectively so far and more than 600 design documents ranging from specification, conceptual design, detailed design, structural mechanics and thermal hydraulic analysis details etc. have been reviewed. Majority of safety issues have been addressed with adequate depth. The structural integrity assessment of reactor assembly components under postulated core disruptive accident (CDA), performance of primary sodium pumps, core along with safety rod drive mechanisms during seismic events, buckling of thin shells under seismic loads and reliability of core support structure integrity, analysis of design basis events and control rod worth are a few examples. Safety Review also identified some issues that were subsequently addressed with adequate detailing. Some of the issues, to name a few, are, monitoring of blanket subassembly, flow allocation check through core & leakage flow during commissioning, increased flow for failed fuel subassembly in storage, neutron flux monitoring systems, pulse coded logic system for shut down system, drop time of absorber rods etc. AERB has given clearance for the vault erection up to the level of safety vessel support. It is planned to complete the review processes necessary for AERB to obtain further clearances for completing the vault construction and starting equipment erection.

Significant lessons learnt from operating experience of FBRs have been given due consideration, when we are embarking on indigenous design. FBR programme, both for research and power production, have been pursued in select number of countries. A great deal of experience totalling to about 350 reactor years has been exclusively accumulated in operating experimental, demonstration and prototype fast spectrum nuclear systems. As indicated in a Russian report by Dr. A.A. Rinejski the cost of developing, testing and operating such systems is estimated to be more than around $50 billion in the world as a whole. The experiences have been taken into account in design, analysis and validation. A few examples are highlighted here. The choice of 20% cold worked special stainless steel (D9) has been chosen for major core components like clad, hexcan etc. The undesirable experience with stabilised austenitic SS grades such as 321 & 347 as also ferritic steel of 15M D3 with low molybdenum has been factored in design. The inadvertent blockage of the coolant inlet port in sleeve of one of the fuel subassembly in EBR-II reactor leading to melting of fuel has resulted in adoption of multiple coolant entry paths in grid plate to preclude occurrence of such an event in the Indian context.

The initiating mechanisms of vibrations of thermal baffles of French reactor SPX-1 have been understood and design of reactor structures modified to preclude the same. Generic design failures in the weld at the top of IHX in Phoenix were due to large thermal gradients between the inner and outer shells of outlet header. The cause was attributed to improper mixing of coolant (secondary sodium) streams from tubes and lack of flexibility at the junction. This is addressed in PFBR by incorporating a proper mixing device above the top tube sheet and appropriate secondary sodium flow distribution in the tubes.

> Further, innovation in design and extensive testing of such designs is the key to success in evolving a robust design for future FBRs in India.

It is well acknowledged that reliability of the SG is a major factor in governing the availability of FBR. Hence, greater care was exercised at all stages viz material selection, design and manufacture. Apart from material selection, international experiences with SG in early FBR, such as PFR (UK), BN 350 (CIS), BN 600 (Russia) was not satisfactory. The design and manufacturing of tube to tubesheet joint is critical for the success of operation of SG based on the experience. For PFBR, the same is accomplished by incorporating a raised projection (spigot) in tube sheet leading fully radiographable butt-welded joint to tube. The origin of failure of components such as mixing Tee in SPX-1 and tank of a secondary sodium pump in Phoenix have been traced to thermal striping phenomenon. The lattice plate in above core structures of reactor assembly of FBR and other locations of mixing of hot and cold streams of sodium are prone to failures under this phenomenon. In PFBR, all such locations of concern have been carefully identified and detailed studies w.r.t analysis and experiments have been completed. A detailed creep-fatigue damage evaluation has been carried out based on which recommendations are made for DT limits for each of the critical locations. Mixing devices are incorporated wherever necessary and geometry of some of the components such as lattice plate, inner vessel etc are arrived respecting the recommended temperature limits for avoiding the failures due to thermal striping phenomenon. Considering the leakage of sodium from thermowell of secondary sodium piping in a Japanese Fast Reactor MONJU, the design of all thermowells was critically reviewed and confirmed that there' is no such risk in the PFBR thermowells due to their special and favourable geometrical features.

It is no doubt that the safe operability of the reactor and the mechanical integrity of its components / systems are of great practical importance from both safety and economic considerations. Realising this aspect quite early, the design of PFBR has been evolved with adequate in-service inspection provisions. The required gadgetry and the technologies have evolved concurrently with the evolution of reactor design. The state-of-art inspection for the reactor assembly includes volumetric inspection of main vessel by remotely controlled robotic carriage, viewing of the vessel internals in cover gas space by a periscope and viewing of the plenum above the core by a under sodium ultrasonic scanner. With these measures and several other features specific to sodium systems and piping, we can confidently say that the requirement of ISI has been well factored into the design of PFBR. The technologies w.r.t. repair of reactor structures occupy our current attention. BARC has made significant collaborations in realizing in-service inspection equipment.

Major design activities of PFBR have now been completed. The balance design activities including obtaining PDSC clearance for all the revised PSAR chapters towards AERB clearance for the erection of major equipments will be completed within next 3-4 months. A few important activities like preparation of Final Safety Analysis Report, R&D targeted before commissioning of reactor and full support to BHAVINI to construct and commission PFBR, will be on-going activities in phase with the progress of PFBR.

DAE has committed to the nation an ambitious energy share from FBR, about 200 GWe in 2050. To achieve this objective, a well-conceived roadmap has been established. Accordingly, it is planned to construct by 2020 four additional 500 MWe oxide fuelled FBR plants adopting twin unit concept. During this period, metallic fuelled FBRs will be developed and one out of four units will be operated with the metallic fuel (design will have the flexibility to accept the metallic fuel, though the reference design is for oxide fuel). Beyond 2020, a series of 1000 MWe Metallic fuelled Reactors will be constructed.

The present design of PFBR has improved considerably in meeting the objective of economy, safety and operation, based on in-depth design, reliability analysis, optimization studies, in depth R&D through in-house and collaborative projects, critical review by national and international experts and incorporation of operational experiences of FBRs world wide including hands on experience with FBTR. Further, innovation in design and extensive testing of such designs is the key to success in evolving a robust design for future FBRs in India. The first step would be the finalisation of the conceptual design of 500 MWe oxide fuel reactor (FBR 500), with improved economy and enhanced safety taking PFBR as a reference benchmark. Based on our reviews and rigorous discussions, directions have been identified clearly to move ahead to achieve the objectives of economy, safety and high breeding. Some of the improvements include critical re-look at plant parameters to fully examine the existing margins, deletion of main vessel cooling, in-vessel primary sodium purification, integrated control plug with small rotatable plug, thick plate concept for top shield, anchored safety vessel, increasing the plant design life etc. Increasing the burn up to 200 GWd/t and above, plant availability more than 90 % and reducing the construction time from 7 years to 5 years are some of the challenging tasks ahead of the Design Group.

It is well realized by us that for faster growth of nuclear power in India, development of metallic fuel which yields high breeding ratio and in turn shorter doubling time is essential and must be introduced expeditiously. Reactor physics aspects for 500 MWe and 1000 MWe FBR designs have been studied, and it has been demonstrated that the doubling time of 8-10 years is achievable. Further, preliminary safety analysis, completed for 500 MWe reactor indicates that, due to lower average fuel temperature and strong axial thermal expansion, the reactor is safe. These results provide high confidence for the development of metallic fuelled FBR, for the FBR design community internationally.

With the dedicated and continuous mission oriented efforts, put forth over the years, the engineers / scientists in Design Group have attained high level of competence and knowledge to design and develop robust FBR plants that can meet the sustainability requirements on par with international standards w.r.t economy and safety. The successful commissioning of PFBR and ensuring high availability factor is the milestone for which Design Group has worked relentlessly and is looking forward to experience this success. A Peer Review Committee with Chairmanship of Dr.K.Kasturirangan, during its critical and independent assessment of quality of R&D pursued in engineering science, stated in May 2006 that IGCAR has comprehensive facilities and expertise and is on the right path to achieve leadership in FBR Technology. There is no doubt that this Group at IGCAR is truly a national asset and is destined to play a key role in making India a global leader in the fast reactor technology by 2020.

I would like to conclude by saying that all the directors of the Centre starting with former Directors, Shri. N. Srinivasan, Shri. C. V. Sundaram, Shri. S. R. Paranjpe, Dr. P. Rodriguez and Shri.S.B. Bhoje have kept the focus of design group and accelerated the pace of achieving high level of competence through judicious experimental, analytical and collaborative approaches. I am walking on the path of success with emphasis on synergy within the Centre, India and also internationally, doing everything possible so that we can attain the global leadership in this technology by 2020. I am convinced that this technology is of high relevance and significance to India and is an almost inevitable solution to provide sustainable energy in 21st century internationally.

Baldev Raj  
*Director*

# Investigation of Thermal Striping Risks in Fast Breeder Reactors

Sodium is the coolant in Fast Breeder Reactor (FBR) which remains in liquid state up to about 1153 K at ambient pressure, and hence no pressurization is needed for the normal operating temperature of 820 K. Hence, the system design pressure is low for the reactor assembly components. However, both hot and cold pools can co-exist within the main vessel with a large temperature difference (150 K) and which is the source of high temperature gradients during steady as well as transient conditions. Apart from this, the temperature changes in the sodium which has high heat transfer properties induces high thermal stress on the austenitic stainless steel which has high thermal expansion coefficient & low thermal conductivity. In view of these characteristics, most of the failure modes in sodium cooled fast reactor are of thermal origin. Among them, thermal striping is one of the critical failure mechanisms which has been reported in FBR components operating in the world.

In an idealised situation, when two streams at different steady temperatures happen to mix, the temperature in the vicinity of junction would not be steady and would rather fluctuate in a random manner (Fig.1). This random temperature fluctuation is called thermal striping (TS). The temperature fluctuations have a wide frequency spectrum ranging from 1 Hz to 20 Hz, with a dominant frequency of about 10 Hz. If thermal striping occurs in the vicinity of a metallic structure, the structural wall surface would be subjected to the temperature fluctuations, the magnitude of which depends upon the fluid characteristics. In case of sodium whose heat transfer properties are excellent, the adjoining metal surface is subjected to similar fluctuations without any significant attenuation. Under such fluctuations, crack initiation might occur due to high cycle fatigue damage mechanics.

Four major sodium leaks have occurred in the operating plants due to thermal striping. In 250 MWe Phoenix reactor, 2 sodium leaks were detected during the inspection campaign in the secondary sodium circuit after operation of about 90,000 h. The material of construction was AISI 304. The first failures as on the weld line in the expansion tank in the vicinity of sodium discharge area (Fig.2). The temperature of the hot leg is 823 K and the cold temperature of sodium in the tank is 623 K. The second one was found at the C-seam weld on main pipeline near the mixing Tee where the temperature difference ($\Delta$T) between the hot (703 K in the branch line) and cold sodium (613 K in the main line) is 90 K. Taking into account the proximity of mixing zone from the wall, the $\Delta$T between the hot and cold sodium is 170 K at the failure location. The low frequency displayed is 0.017 Hz and even less. In SPX reactor, sodium leak was observed on a circumferential weld downstream of a Tee junction in an auxiliary pipe of the secondary sodium circuit (Fig.3). The crack was due to thermal striping of $\Delta$T 220 K caused by partial opening of valve. The duration of the unexpected flow was estimated as 4 h. In BN 600 reactor, after operation for about 10 y, primary sodium leaks took place in a cold trap. Thermal striping was identified as a possible reason. The associated $\Delta$T is estimated as 160 K (possible if cold sodium leak exists). In the international

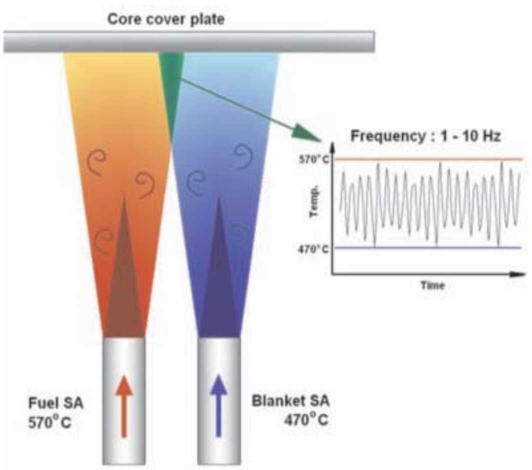


Fig.1: Thermal striping phenomenon

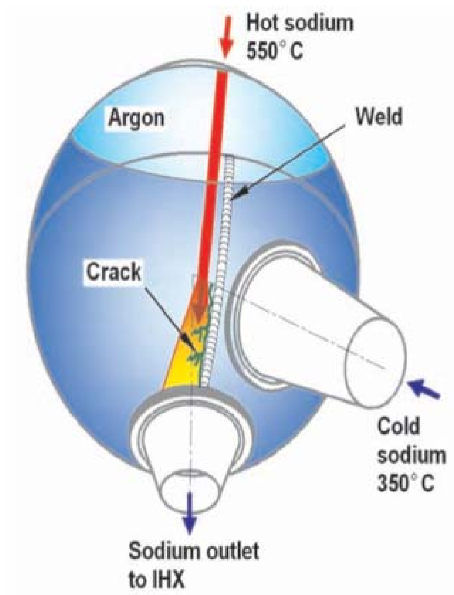


Fig.2: TS in Phoenix expansion tank

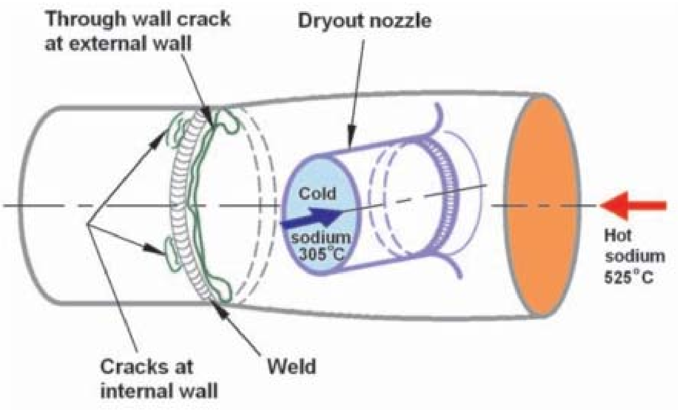


Fig.3: TS in SPX-1 Mixing Tee

test facilities, viz. SUPERSOMITE (AEA, Risley) and FAENA (CEA-Cadarache), thermal striping cracks were simulated on AISI 316 steels. In SUPERSOMITE tests on cylinder with circumferential TIG welds, typically, the ΔT of 210 K and 180 K caused crack initiation and some limited crack propagation after about 1000 h and 10,000 h respectively. The FAENA tests indicated that the ΔT of 192 K (equivalent strain range = 0.382 %) and 378 K (equivalent strain range = 0.75 %) caused cracks of about 50 m size after $1x10^6$ and $1.6x10^4$ cycles respectively.

In Prototype Fast Breeder Reactor (PFBR), there exist a few critical locations in the hot pool viz. (i) region below control plug where the hot sodium jet from fuel sub assembly (SA) and colder jet from blanket SA mixes (Fig.4a), (ii) region below control plug where the hot sodium jet from fuel SA and colder jet from control SA mixes (Fig.4b), (iii) region in the vicinity of inner vessel where the stratified layer exits with high axial temperature gradient (Fig.4c) and (iv) region in the vicinity of main vessel where hotter primary sodium outlet from intermediate heat exchanger (IHX) joins with the cold pool (Fig.4d). Thermal striping issues have been addressed in-depth for PFBR by establishing acceptable temperature fluctuation limits, quantifying the temperature fluctuations at various locations in the reactor assembly and by experimental validations.

Temperature fluctuations due to thermal striping which have high frequency spectrum can create high surface strains. However it decays rapidly with depth of penetration. Due to this decaying characteristic, the crack does not grow deeper beyond certain depth, particularly, in the case of high frequency striping. However, the low frequency component can penetrate deeper. This is clear from the Fig,5 which shows the crack driving force (stress intensity factor) Vs crack depth. With the presence of a wide frequency spectrum, high frequency component creates crack initiation within short time due to accumulation of a large number of cycles without causing deeper cracks and low frequency component can create deeper crack only after a long time. This implies that there exists a narrow frequency range which can create deeper crack within relatively shorter time. This has been arrived at a simple plate with an initial crack size of 0.5 mm (Fig.6).

With an objective of establishing thermal striping limits for PFBR, a fracture mechanics-based method has been derived taking into account the above mentioned frequency dependent through-wall stress decay. The UK-fatigue design procedure

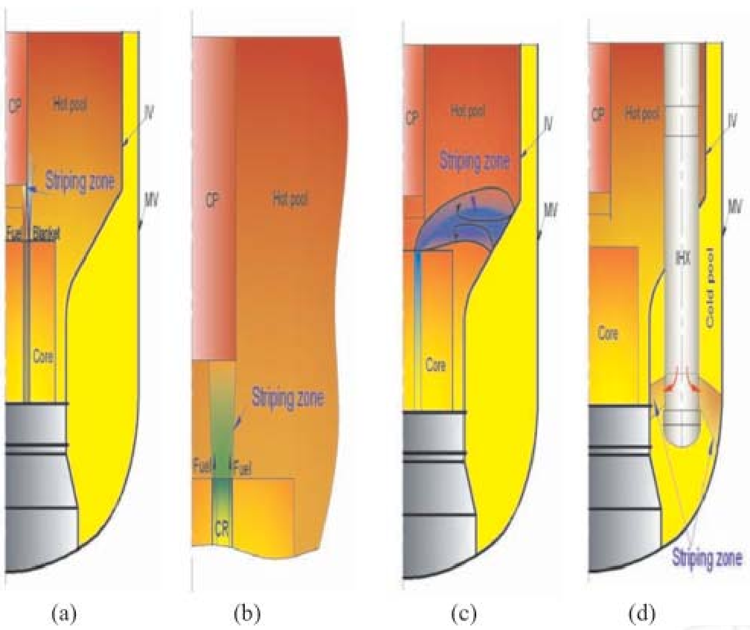


Fig.4: Critical locations where thermal striping occurs in the hot and cold pools of FBR

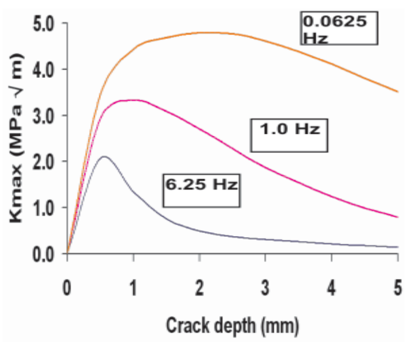


Fig.5: Stress intensity Vs crack depth

is used to obtain the crack size from the cumulative creep-fatigue damage. Based on the method, thermal striping limits have been derived as a function of accumulated creep-fatigue damage and frequency. Thermal striping limits are more stringent for lower frequencies and larger creep-fatigue damages. Since the frequency spectrum is not available at the design stage, the limits are derived conservatively based on 0.01 Hz, the lower most frequency possible. Further, the limits are found to be independent of structural wall thickness, for thickness greater than 5 mm.

From the data collected from operating experiences of Phoenix, FAENA, SUPERSOMITE, acceptable ΔT values corresponding to the life of $1x10^9$ cycles (45 y design life with 75 % load factor) are derived. The corresponding extrapolated values are: 68 K, 63 K and 144 K based on Phoenix, FAENA, SUPERSOMITE data respectively. It is worth mentioning that at all the failure locations in the operating reactors, creep-fatigue damages due to major load cycles were present. Hence the values are on lower side (68 K). For SUPERSOMITE, failure has been detected after certain crack propagation and hence, it yields a higher value (144 K). The lower value derived from FAENA could not be explained because of limited information available. Investigation of test data indicates that there is a need of a factor of 1.2 to be applied on the derived thermal striping limit towards ensuring conservatism.

In order to establish design analysis methodology applied to thermal striping phenomenon based on failure experience of Phoenix expansion tank, a co-ordinated research programme (CRP) involving eleven institutes from seven countries including India was organised by IAEA during 1996-1999. Analysis methodology developed by IGCAR could predict the crack initiation life fairly satisfactory.

Applying the above methodology, thermal striping limits has been established as a function of creep-fatigue damage for PFBR (Fig.7). The following are some important limits extracted from the design chart:

- Location-A: For the core cover plate of control plug wherein the low cycle creep-fatigue damage is negligible (~0.0), the acceptable thermal striping limit is 60 K.
- Location-B: For the lower portion wherein the creep-fatigue damage is ~0.2, the acceptable limit is 52 K.
- Location-C: For the locations on inner vessel or main vessel wherein the accumulated creep-fatigue damage may be moderate (0.5), the acceptable thermal striping limit is 40 K.

In order to determine the temperature fluctuations at various locations in the hot and cold pools of reactor assembly, complete 3-D thermal hydraulics analysis is performed in two stages. In the first stage, global analysis is performed to derive the overall temperature distributions. In this, the control plug and Intermediate Heat Exchangers (IHX) are treated as porous media and appropriate flow resistances are specified based on porous body formulations. In the second stage of local analysis, velocity and temperature oscillations are obtained at the selected locations where thermal striping is possible. The results are summarised in Fig.8. It has been found that temperature attenuation within the boundary layer is about 50 % which means that the actual metal sees the temperature fluctuations much lower than the fluid temperature oscillations.

Comparing the temperature ranges at various locations with the corresponding allowable temperature ranges on the metal wall at the respective locations, it is concluded that the PFBR is free from risk of thermal striping.

Towards understanding the thermal striping behaviour with an idealised model, a test setup involving two water jets with a temperature gradient of 90 K, achieved by maintaining cold water at 278 K and hot water at 368 K, which impinge of the rigid horizontal plate, has been established at Structural Mechanics Laboratory, IGCAR. In this setup, various parameters such as, horizontal location w.r.t. central line between the jets, elevation of rigid plate w.r.t outlet nozzle of water jets, water flow rates and temperature difference between hot and cold jets are studied. The test setup is shown along with the temperature record in Fig.9. Numerical simulation with a state-of-art code STAR-3D is in-progress to

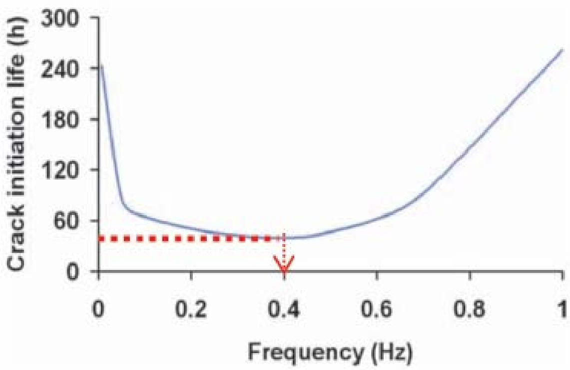


Fig.6: Concept of critical frequency

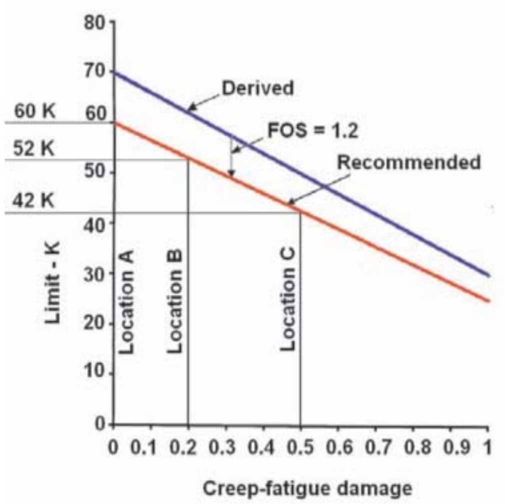


Fig.7 Thermal striping limits for PFBR

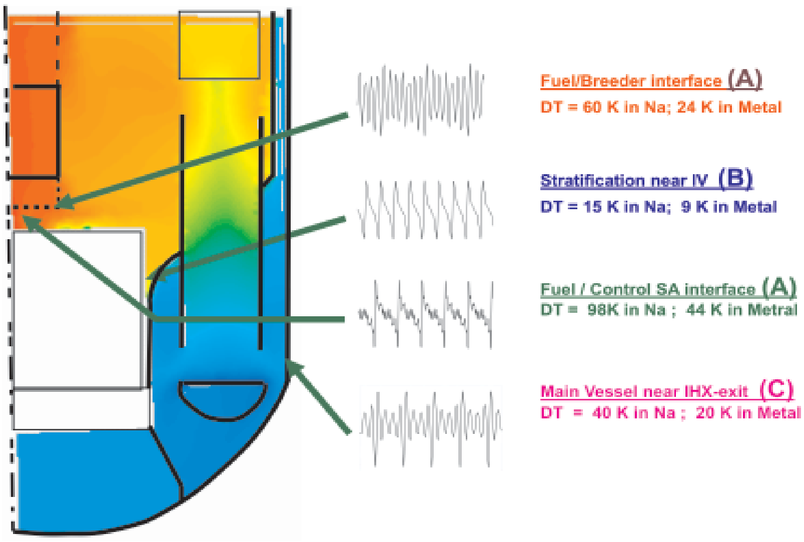


Fig.8: Thermal striping zones and temperature fluctuations in hot and cold pools of PFBR

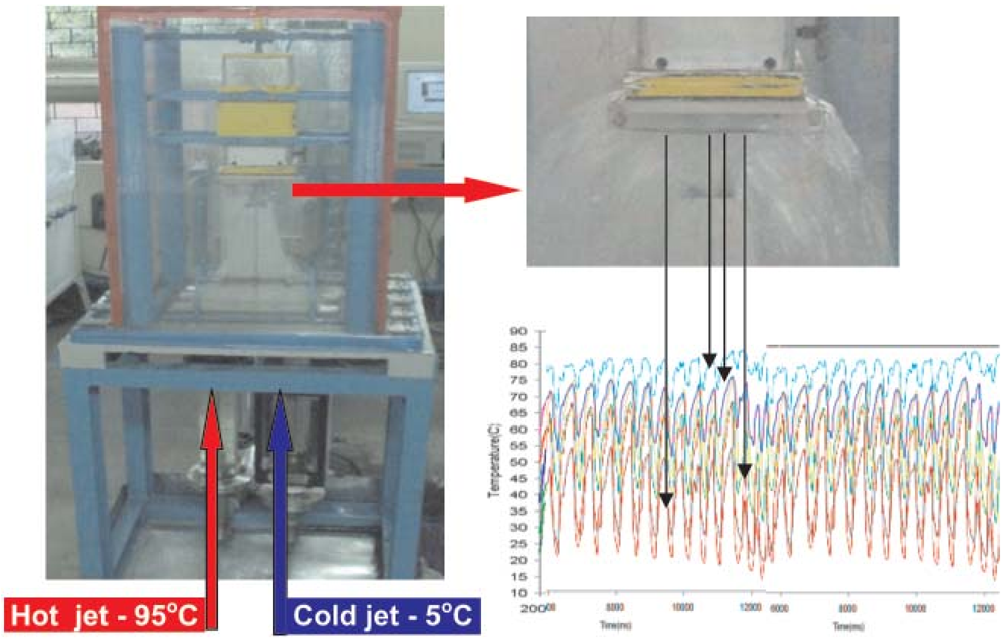


Fig.9 Simulation of thermal striping with water jets

predict the experimental observations. The test setup has been developed jointly by IGCAR and IIT Chennai under a collaborative project.

With the limited experimental data base available, thermal striping limits are established with conservative margins. Hence, R&D works are continued towards further understanding the metallurgy and mechanics of thermal striping phenomenon jointly by IGCAR and IIT Chennai. Water tests are continued in Structural Mechanics Laboratory and further it is also planned to conduct tests in sodium to simulate the temperature fluctuations in scaled models representing the real situations at in-sodium test (INSOT) facility to investigate the metallurgical aspects. It is strongly hoped that the results obtained so far from the investigations completed and further results that would be obtained through systematic R&D activities planned under collaborative projects, would result not only in establishing realistic thermal striping limits, but will also contribute significantly to get better insight into the science behind this complex thermo-mechanical phenomenon.

*P.Chellapandi, R. Srinivasan, K. Velusamy and S.C.Chetal, Reactor Engineering Group*

# Intelligent Displacement Measuring Device for Fast Breeder Test Reactor

FBTR is a sodium-cooled loop-type reactor, where the reactor vessel is suspended from the top. The vessel is subject to a lot of mechanical, thermal and irradiation stresses, as a result of which, it may tilt to one side. Since this is a safety hazard, any such tilt must be detected in time and corrective actions taken. So the displacement of the reactor vessel in all the three directions (X, Y and Z) has to be monitored periodically. For this purpose a displacement measuring device (DMD) assembly exists at FBTR. The displacements are measured at two locations in FBTR, at 45º and 315º from the sodium inlet axis, as shown in Fig. 1. Another view is shown in Fig. 2. The displacement of the reactor vessel along the X and Y axes is measured with respect to a plumb wire suspended from the vessel.

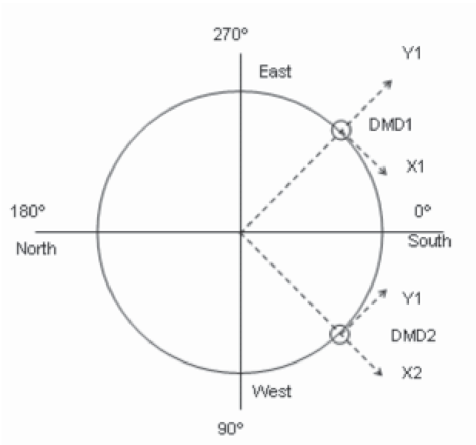


Fig. 1 DMD Locations in FBTR

The DMD assembly is shown in Fig. 3. Plumb wires are suspended from the reactor vessel at the two points where displacement is to be measured. A rectangular metallic frame is positioned at B8 cell such that the plumb wire passes through it. This frame is moved along the X or Y axis by an AC synchronous motor until it touches the plumb wire. The frame is connected to a potentiometer slider whose body is fixed to a mechanically rigid platform. The distance traveled by the frame with respect to the initial position to establish contact with the plumb wire, is the measure of the reactor vessel's displacement in that axis.

The potentiometer is provided with a fixed DC excitation, and the ratio of the variable resistance to the total resistance is measured by the DMD controller. The X and Y displacements in other locations are measured in a similar way by the dedicated frame, motor and potentiometer setup. Displacements are measured in both directions, i.e., moving the motor in forward direction until frame touch, and moving the motor in reverse direction till frame touch. This is done in order to achieve dual confirmation of the measured displacement. The displacement in Z-direction is measured using a spring-

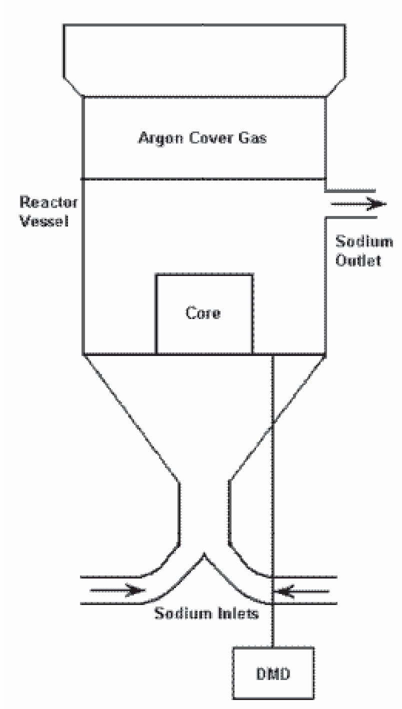


Fig. 2: DMD Setup: Another View

loaded potentiometer whose slider follows the movement of reactor vessel in that axis.

The measurement of displacement in a particular axis (for X axis and Y axis) involves switching of the motor, acquisition of potentiometer reading, detection of plumb wire and frame touch and calculation of final displacement.

While commissioning the DMD assembly in FBTR, initial readings were taken and kept as reference for further calculation. Hence difference in current reading and the initial reading gives the displacement of vessel. However, the final displacements in Y1 and X2 directions need temperature correction due to thermal expansion of the reactor vessel radially. Temperature correction is not required for the other axes. For example, if the reference value for X2 axis (X-axis in location 2) in forward direction is R0 and if the displacement obtained in the same axis in the same direction after a period of time is R1, then the final displacement is calculated as (R1 –R0) – Rc, where Rc is the temperature correction. Hence, the temperature is also to be measured.

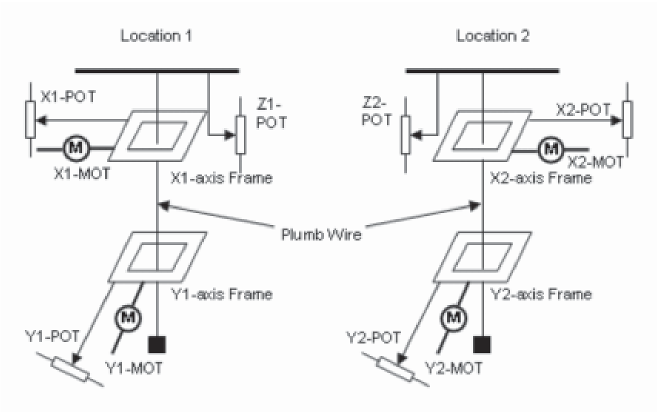


Fig. 3: Diagram of Reactor Vessel Displacement Measuring Device

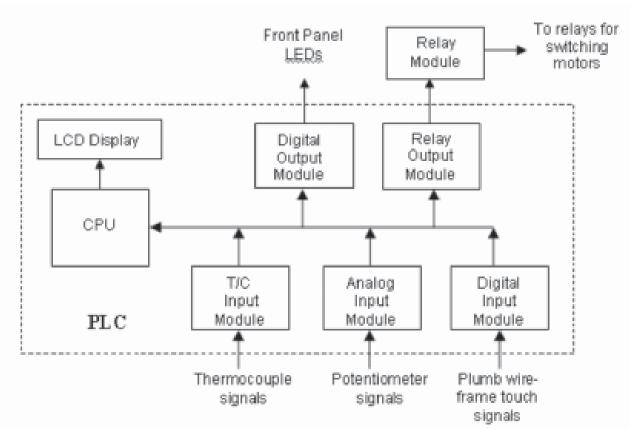


Fig. 4: DMD Controller Block Diagram

Thermocouples are present at the grid plate of the reactor, which provide the temperature signals to the system.

A Programmable Logic Controller (PLC) based DMD Controller has been designed as part of the DMD assembly described above. The program was developed with a Windows-based software package in ladder logic using standard Relay Ladder Logic (RLL) instructions. The basic components of the system are the PLC base or back plane, CPU, LCD Display module, and input/ output modules. The architecture of the DMD controller is shown in Fig. 4.

Two analog input modules are used for reading the potentiometer input signals and a thermocouple input module reads in the temperature signals. A digital input module is used for reading the commands, a digital output module for providing front panel displays, and two relay output modules for driving the motors. The base includes a built-in power supply. An optional lithium battery is available to maintain the system RAM retentive memory when the system is without external power.

With reference to Fig.3, there are four frames (for four axes, namely X1, Y1, X2 & Y2) to measure the displacements in the respective axes. The controller controls the four motors, one at a time, to get the displacement in the selected axis both in forward and backward directions.

The controller accepts 6 potentiometer signals (for X, Y and Z axes position measurements at locations 1 and 2) as well as the fixed DC voltage that is applied across the potentiometer, through the analog input module. To make the measured displacement insensitive to voltage variations, the ratio of potentiometer output voltage to the DC voltage is used to measure the displacement. The full-scale voltage corresponds to a displacement of 50 mm, and so the voltage can be converted into displacement. The analog inputs are optically isolated from the PLC logic. The module has a resolution of 12 bits.

The CPU also accepts four thermocouple signals from the reactor vessel through the thermocouple input module for temperature correction. The thermocouple module is configured to accept 4 differential K type thermocouple inputs and has a resolution of 0.1 ºC.

For measuring displacement, the controller must drive the four motors, namely X1, Y1, X2 and Y2 axes in forward as well as reverse directions. The relay output module is used to activate relays to switch the motors ON/OFF in forward or reverse direction in location 1 or 2.

Basically the system has two modes of operation: Manual and Auto. In the manual mode of operation, the operator specifically selects the location, axis and direction for which he wishes to take a reading. In the auto mode, the operator simply presses a button, and the system takes care of all the measurements.


Fig. 5 DMD controller Front Panel Photograph

The location for displacement measurement (1 or 2) is selected using a front panel toggle switch. The axis and direction (X-Forward, X-Reverse, Y-Forward, and Y-Reverse) are selected through a 4-in-1 radio button switch on the front panel. The mode of operation (Auto or Manual) is selected using a toggle switch. The temperature readings can also be displayed at any time, by pressing the temperature switch.

In addition to this, there is also a Move-Motor switch on the back panel of the instrument. This is a toggle switch and it is used to enter the Move-Motor mode of operation. In this mode, any motor can be rotated in any direction independent of the frame touch status or extreme limits. This mode is used to bring the frames back to proper position in case the motors have rotated too far. It is entered only under abnormal conditions to set the system right when it shows erroneous operation.

When the frame touches the plumb wire, the motor must be stopped and the reading must be taken. The plumb wire - frame touch signals for the X1, Y1, X2 and Y2 axes are sensed through the digital input module. The selected location, axis and direction, the modes (Auto or Manual, Move-Motor or Normal), print command, and temperature display input are also read in through the digital input module. The digital output module is used for giving front panel indications.

The front panel of DMD controller is shown in Fig. 5.

The PLC-based DMD controller device is operator-friendly and its features like the Auto mode help in automating the entire displacement measurement process. The device has been now installed in FBTR and is found to be working satisfactorily.

(K. Palanisami, Saritha P. Menon, S. Ilango Sambasivan, *Electronics & Instrumentation Division, Electronics & Instrumentation Group)*

# IGCAR Campus Network: Upgradation from ATM to Gigabit Technology

The revolutionary developments in the Computers & Communication Technology in the last two decades have brought phenomenal changes in the way we compute and communicate. These technological changes have raised such high expectations that today every Scientist and Engineer want the results of any computation, the graphical view of any engineering analysis, almost immediately right on his desktop. He wants to view, edit, share any file, of any size almost instantaneously. He wants to interact, chat and view in real time. All this requires high bandwidth communication networks connecting the user desktops, computing servers, information servers etc. installed throughout the campus. If establishing such high-speed networks and managing them with a high uptime is a challenge, especially in big campuses, maintaining them against obsolescence is a much bigger challenge.

IGCAR had a Campus network based on Asynchronous Transfer Mode (ATM) Technology connecting about 50 Local Area Networks established in different Laboratories of the Centre. These laboratories are spread all over the big campus requiring more than 10 kms of Fibre Cable for connectivity. The network

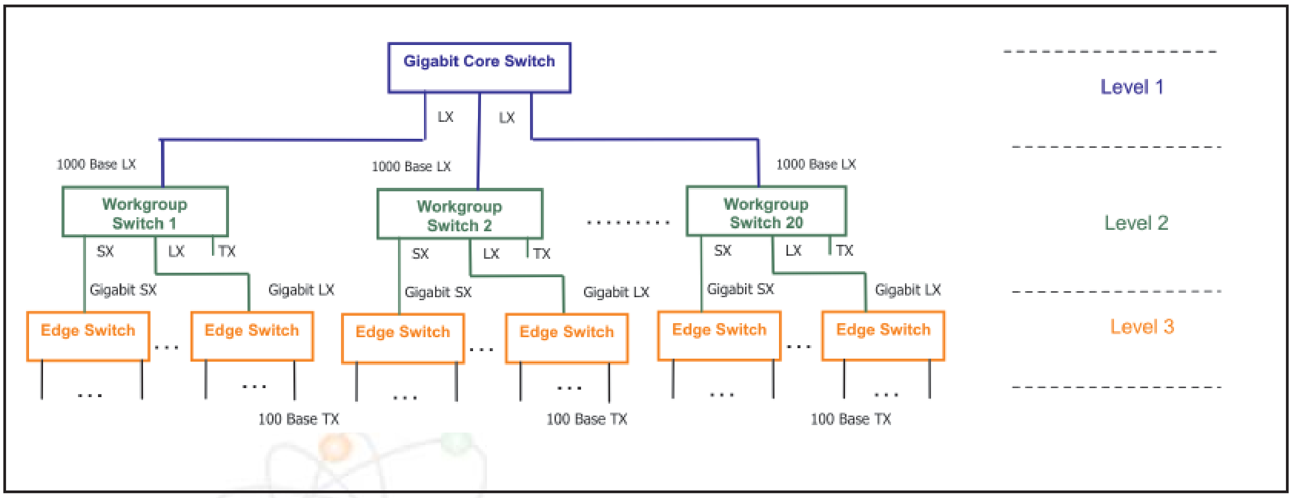


Figure1: Upgraded Setup of Campus Backbone Network with Gigabit Technology

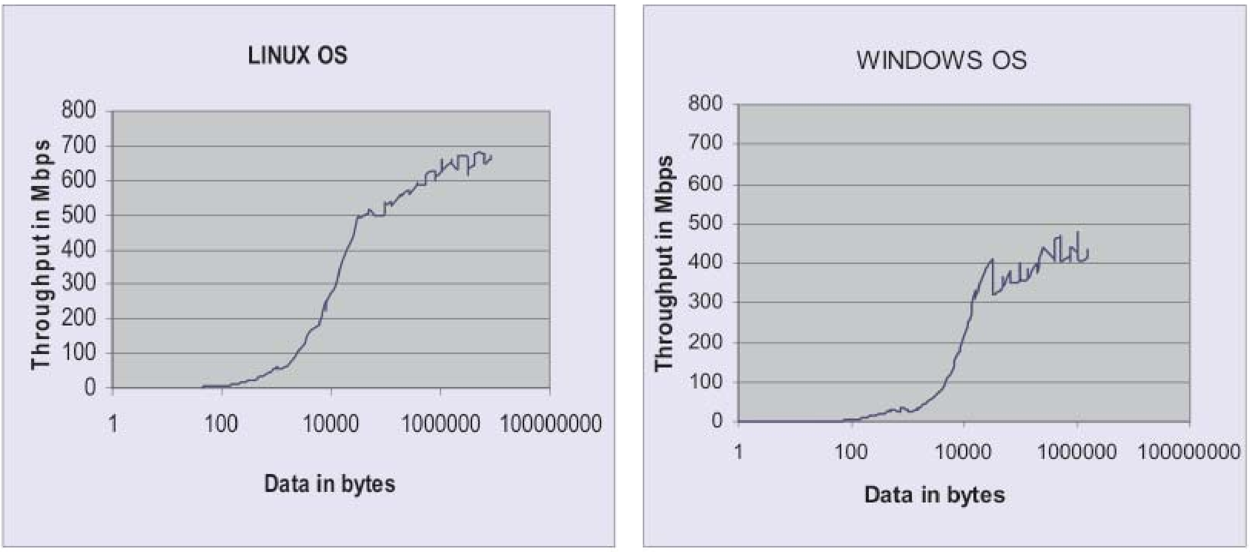


Fig. 2: Throughput performance of the Campus Network

was based on a three tier architecture having ATM Core Switch supporting 155 Mbps with 20 ATM Fibre Ports in the first level, about 20 Work Group Switches having ATM uplinks and 100 Base FX/TX down links in the 2nd level and about 50 edge switches with 100 Base Fx/Tx uplinks and 100 Base Tx fast Ethernet downlinks in the third level. The whole campus network was established using composite fibre cable and provides connectivity to the High Performance computing Servers, Information Management Servers, Internet, E-Mail, Web Servers, Digital Library etc. The total network was configured in to about 20 VLANs for better management.

The upgradation of the Campus Network was necessitated due to the following reasons:

a) Obsolescence of existing ATM hardware
b) Non-availability of spares
c) Cost of Maintenance
d) Need for higher network speed

Once, it was decided to upgrade the network, a thorough market survey was done. The popularity of Ethernet in terms of technological knowledge base, its downward compatibility has made the Ethernet a logical choice. Its high volume adaptation has resulted in unprecedented reduction in prices of all its components. The sheer volumes also ensure a better maintenance support. Hence, it was decided to go in for Gigabit Ethernet Technology. It is a fact that designing a new network / system is far more easier and interesting than designing a retrofit. The latter is more complex and hence becomes a challenge. Since, the upgradation is for an existing network, any change or addition has to be retrofitted. While doing so, it was decided to follow these important guidelines:

a) The upgradation has to be transparent to the end user. i.e. users need not have to change the IP address, Netmask, Gateway etc. The VLAN number and IPX network number remains the same.

b) The downtime of the network due to upgradation has to be very minimum, as the Scientific Community requires services like E-Mail, Internet all the time. To ensure minimum downtime, it was decided that both the ATM and Gigabit networks were configured to co-exist during switch over time. One VLAN by VLAN are switched from ATM to Gigabit Network. In the end, though it required more effort, it ensured smooth switch over and a down time of less than five minutes.

In the ATM network, most of the Work Group Switches and Edge Switches are connected through Multimode Fibre Cable. However, for the gigabit speed, multimode fibre cable does not support the required distances and the singe mode cores of composite cable have to be connectorised in most cases.

The new enterprise switch based on gigabit technology is a modular type to enable future expansion. It supports 32-gigabit ports with a switching speed of 64 Gbps and throughput of 48 Mpps. The switch has redundant switching engine and redundant power supply, supports trunking, mirroring and all protocols required for our environment. The Work Group Switches have all gigabit ports (Fibre/Copper) and the edge switches have Gigabit uplinks and Fast Ethernet downlinks.

The Gigabit Enterprise Switch, Work Group Switches and Edge Switches were configured and installed at various buildings. The Single Mode Fibre Cores were connectorised and tested. Then, all the VLANs were shifted from ATM to gigabit enterprise switch one by one. The set up of the new gigabit campus network is shown in figure 1. The total network is being managed with a state of the art Network Management System.

Once the network was commissioned, it was thoroughly tested to ensure the functional requirements. The various in-house applications and systems were used to ensure that all the services and protocols are working satisfactorily. Open source tools like "Ethereal" were used to trace various services and protocol communications. The network performance was tested using an open source software called Network Protocol Independent Performance Evaluator. These tests were conducted both in Linux and Windows Operating Systems which are mostly used in the Centre. The typical results achieved in both these Operating systems across these switches are plotted in Figure 2. It can be observed that a throughput of 650 to 700 Mbps (with Linux OS) was achieved in the network against a theoretical limit of 1 Gbps which is quite satisfactory.

The total project was planned both at Macro and Micro levels and was executed smoothly with least interruption to services and the upgradation of the campus network was completed successfully.

*(S.A.V.Satya Murty, Computer Division, Electronics & Instrumentation Group)*

# Forum for Young Officers

## CORAL: A Stepping stone to the FBR Fuel Reprocessing Technology

**N. Designan**
DOB: 10.01.78

N. Designan did his M.Sc. (Chemistry) from the University of Hyderabad in 2000. He is from the $46^{th}$ batch of BARC Training School and joined IGCAR in September 2003 as Scientific Officer (SO/C).

The success of the fast reactor technology in India which is the $2^{nd}$ stage of its nuclear power program, apart from the successful operation of the fast reactor itself, depends also on closing the nuclear fuel cycle by recycling the plutonium (Pu) and uranium (U) from the spent fuel back to the reactor through Fast Reactor Fuel Reprocessing (FRFR) technology. In reprocessing the Pu and U present in the spent nuclear fuel are separated from their associated fission products and finally from each other through the Plutonium Uranium Extraction (PUREX) process. The separated Pu and U are then chemically converted to their solid oxide forms which are used for refabrication as fresh fuel rods to be recycled back to the reactor.

The progressive reprocessing of the fast reactor spent fuel is being taken up from a pilot plant scale at the CORAL (COmpact Reprocessing of Advanced fuels in Lead shielded cells) facility at IGCAR through to a demonstration scale at the Demonstration Fast reactor fuel Reprocessing Plant (DFRP) and ultimately to the commercial scale. As the plant throughput and requirement increases enormously in the above order, meeting the plant requirement and issues relating to criticality becomes challenging. This calls for new strategies for reprocessing the fast reactor spent fuel at a larger scale rapidly.

CORAL provides an opportunity to understand and overcome the challenges of the FRFR technology both in process and development of equipments. Unlike in Thermal Reactor Fuel Reprocessing (TRFR) where the Pu concentration is very low, FRFR poses a lot of challenges. CORAL adds more dimension to the challenges by reprocessing the mixed carbide (MC) spent fuel with 70% Pu content which is handled for the first time in the world. To mention a few, development of single pin and multi-pin choppers, dissolution of fuel with high Pu content, the retention of Pu in lean organic[1],

1 Lean Organic: During solvent extraction process an organic solvent is chosen for which the distribution co-efficient (Kd) values of the species of our interest present in an aqueous feed are optimum such that they can be extracted into an organic phase (extraction) and re-extracted back (stripping) into a different aqueous phase enabling their separation from the rest of the species present in the mother aqueous phase. This organic phase after the stripping is termed as the lean organic as they are lean in concentration of the species of our interest.

recovery of Pu from heavy organics, solvent degradation, electrolytic partitioning of Pu and U, remote maintenance of hot cell facility etc. CORAL during its various reprocessing campaigns with fuel of successively increasing burn-up has addressed these issues and various methods were developed to overcome them.

I was involved in addressing the issue of recovering Pu from various non-product streams in addition to the regular reconversion operation. After some extensive laboratory studies, methods were developed and demonstrated at the plant scale. During the various reprocessing campaigns at CORAL, the Pu from the entire lean organic stream was recovered based both on the single point and multiple point introduction of uranous solution. As a back up, alternate treatment methods using carbonate as the strippant were demonstrated on an experimental basis and scaled up to the plant requirement. During the reprocessing of the FBTR spent fuel with burn up of 100 GWd/T, the activity due to $Ru^{106}$ was high. The $Ru^{106}$ isotope to some extent followed the path of Pu & U leading to higher activity of the $1^{st}$ cycle lean organic stream. Prior to the recovery of Pu from this stream, a wash step was added to reduce the activity significantly. The resulting lean organic stream with lesser activity was treated by uranous stripping method.

The solvent (TBP, tributyl phosphate) used in the plant gets degraded due to the higher activity of the fuel. The degraded products (DBP & MBP) render the solvent inefficient for further use. Hence they are washed with sodium carbonate to remove the degraded products on a regular basis. But sodium carbonate adds to the waste volume. It also needs heating treatment. A solvent wash method based on ammonium carbonate instead of sodium carbonate was developed and followed to overcome this. Moreover ammonium carbonate does not add up to the impurities in the final product. The solvent is washed and recycled as long as its physical properties like surface tension etc. do not change.

### Recovery of Pu from first cycle lean organics:

The recovery of Pu from the lean organic generated in the first cycle was very challenging as it was highly active due to the $Ru^{106}$ isotope as mentioned above.

For this, the lean organic phase is treated with nitric acid which reduces its activity significantly without changing the concentration of Pu and U. The resulting lean organic is then washed with uranous in dilute nitric acid wherein Pu and the excess uranous get stripped to the aqueous phase leaving the organic phase with negligible amount of Pu. The retained U in the lean organic phase is recovered by employing the ammonium carbonate method prior to their disposal. The Pu and U in the aqueous phase are recovered through the well established oxalate precipitation and ADU precipitation route respectively in that order. Using the above method a significant quantity of lean organic from the first cycle was treated to recover Pu and U.

The above mentioned activities are a small part of the developmental activities that were undertaken at CORAL during the course of the various irradiated FBTR mixed carbide fuel reprocessing campaigns. Thus CORAL offers a number of opportunities for young budding engineers and scientists to be innovative in understanding the FRFR technology. With the development of above and various other methods, the group is confident enough to handle the spent fuel of higher burn up which would pave way for successful operation of the DFRP and the FRP. Thus CORAL serves as the stepping stone for the FBR reprocessing technology in India.

*(N. Desigan and colleagues, Fuel Reprocessing Process Division, Reprocessing Group)*

# Peer Review of Engineering Sciences Activities

The R & D activities in engineering sciences being carried out at IGCAR have over the years been growing consistent with the challenges and have matured to enable design of 500 MWe Fast Breeder Reactor Project (PFBR) giving due consideration to economy so that it will be the forerunner of many FBRs to be constructed in future. With the objective of having a critical and independent evaluation of R & D activities of the Centre in engineering sciences, particularly in the context of embarking upon a multi-fold increase in power generation planned through FBRs and taking reprocessing technologies to high level of robustness, a peer review of engineering sciences activities was conducted during May 3rd and 4th 2006. The review panel consisted of eminent experts Dr. K. Kasturirangan, former Chairman, ISRO and Director, NIAS, Prof. S.P. Sukhatme, Former Chairman, AERB, Prof. M.S. Ananth, Director, IIT-Madras, Prof. M.L. Munjal, IISc, Prof. S. Ranganathan, IISc, Prof. R.K. Shyamasundar, TIFR and Shri. M.S. Konnur, Former Director, FCRI. The activities reviewed included fast reactor engineering & technology, materials technology, safety engineering, electronics & instrumentation engineering and reprocessing technology. The review was aimed to provide important inputs concerning the level of excellence in the programmes, effective utilization of facilities and human resources and suggestions for (a) mid-course corrections in on-going activities to add value, (b) collaborations, both within DAE units and with national research centres and academic institutes, (c) new programmes towards

Members of Peer Review Committee listening to a presentation (R to L) -Shri.M.S. Konnur, Prof. M.S.Ananth, Prof.S.Ranganathan, Dr. K. Kasturirangan, Chairman, Dr.Baldev Raj, Prof. M.L.Munjal, Prof.S.P. Sukhatme and Prof. R.K.Shyamasundar meeting the objectives of the centre, (d) increased synergism and (e) road map to meet the planned growth of FBRs.

In his welcome address, Dr. Baldev Raj, Director, IGCAR explained the role of IGCAR in providing energy security to the country, the future growth of FBRs, the methods by which unit energy cost can be brought down, the importance of mastering reprocessing technology and use of metallic fuels in the growth of FBR program. He concluded by stating that the mission of the centre is to work towards providing robust option of energy security to India and achieving "Global Leadership in FBR Science & Technology" by 2020. In this context, it is very important to have the best wisdom available in the country, in various disciplines, to review the progress made by the Centre. It is envisaged to have the same group of experts to review and monitor periodically the progress of the R&D and the actions taken based on the recommendations and their impact as well as to suggest further strategies.

The review process included 31 oral presentations by various engineering groups and visits of the peers to the engineering laboratories, FBTR and BHAVINI. Detailed discussions were held during the presentations. The committee also had an in-depth discussion with Director, IGCAR and Shri S.C. Chetal, Director, REG at the end of the meeting. The committee complimented the participants for the excellent and passionate presentations and the quality of R&D that is being pursued to meet the challenges in mastering FBR technology. The committee observed that IGCAR has comprehensive facilities and expertise and is on the right path to achieve leadership in FBR technology. The interaction of IGCAR with other R&D and academic institutions within the country was well appreciated and the need for enhancing international collaborations was emphasised. Taking into consideration the enhanced role of FBRs in energy production, need for increasing the manpower commensurate with the challenges was emphasised. The committee suggested IGCAR to work towards enhancing public awareness addressing all the issues related to nuclear power specifically related to fast reactor and associated fuel cycles, as this would be a major component of nuclear energy beyond 2025. It was suggested to have joint working groups involving experts from DAE, ISRO and DRDO. It was felt that this approach would enhance expertise of all the participating strategic departments and accelerate development establishing enhanced synergy.

To make a significant contribution to the power generation, it is imperative that metallic fuel based FBRs with the highest breeding potential are constructed at the earliest. The technology of reprocessing of metallic fuel is challenging and it warrants large developmental efforts. The committee observed the need for augmenting the manpower for this critical assignment where challenges in the design of plant, development of equipment and process are overwhelming.

Dr. Baldev Raj thanked the Chairman and the members of the committee for their excellent suggestions based on their deep involvement in the Centre's programme and their rich experience. He assured the committee that their suggestions would be considered with full earnestness for implementation and he would keep the Chairman and the members posted with the developments so that the Centre can continue to have the full benefit of the peers. The Chairman and the members assured full support and appreciated deep commitment to the excellence with relevance being pursued by the Centre and wished the Centre all the success in achieving the national expectations and aspirations to achieve world leadership in FBR science and technology by 2020.

*(Reported by K. Velusamy, Reactor Engineering Group)*

# ISO 9001:2000 Certification for IGCAR Laboratories

Dr. Baldev Raj, Director, IGCAR receiving ISO 9001:2000 certificate from Dr. C Venkatraman, Head of Certifications, BVQI (India), Chennai

Indira Gandhi Centre for Atomic Research (IGCAR) has received ISO 9001:2000 certification from BVQI, for twelve R&D laboratories along with the DAE Hospital laboratories. The IGCAR laboratories are in the fields of Mechanical Metallurgy, Materials Technology, Non-Destructive Evaluation, Corrosion Science & Technology, Physical Metallurgy, Design & Development of Electronic Instrumentation & Control Systems, Management of Computing & Data Communication Facilities, Quality Assurance, Structural Mechanics, Chemical Characterisation, Radiological Safety and Reactor Safety Engineering.

IGCAR is the first research Centre under the Department of Atomic Energy to receive ISO certification for its laboratories. The Certificate was handed over to Dr. Baldev Raj, Director, IGCAR by Dr.C. Venktaraman, BVQI, in a function held on April 5th, 2006. Speaking on the occasion, Dr. Baldev Raj, Director of the Centre said that IGCAR will consistently endeavour to achieve world leadership in multidisciplinary scientific and engineering research & development activities related to fast reactor and associated fuel cycle technologies to satisfy the energy needs of the country. Dr. S.L. Mannan, Director, Metallurgy & Materials Group, and Chairman, Management Review Committee added that the entire certification process of all the laboratories was completed in a short period of about 12 months. Two years ago, the Reactor Engineering Group was awarded ISO certification for design and engineering of fast breeder reactor for power generation applications.

Dr P.R.Vasudeva Rao, Chairman, Editorial Committee
Members: Dr.G.Amarendra, Shri M.Ganapathy, Dr.K.V.G.Kutty, Dr.Mary Mohankumar, Shri G.Padma Kumar, Dr. B.Purniah, Shri Shekar Kumar, Shri M.Somasekharan, Shri R.Srinivasan, Shri R.V. Subba Rao, Shri K.V.Suresh Kumar.
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