

**Editorial**

*Concept of breeding and development of technology for fast breeder reactors date back to the initial days of demonstration of nuclear chain reaction. The primary purpose of a fast breeder reactor is to produce more fissile isotopes, from fertile isotopes, than is consumed in fission. Plutonium-239 is produced from the fertile isotope  $^{238}\text{U}$  and  $^{233}\text{U}$  is produced from  $^{232}\text{Th}$ , by neutron absorption followed by two successive beta decays. Thus energy could be produced utilizing fertile isotopes by fast breeder technology. This route could help in providing the mankind with an inexhaustible source of energy to meet the ever increasing demands of energy. This is highly relevant to India as we have limited uranium resources and are blessed with vast resources of thorium. In the fast breeder reactors (FBR), fast neutrons are used both for sustaining the chain reaction and breeding. Fission reaction cross section with fast neutrons is very small and therefore, FBRs need fuel with high fissile content and neutron economy plays a vital role in achieving the desired breeding ratio and doubling time. This demands R&D on the design of the fuel, fuel development, fuel fabrication, structural materials etc. in addition to giving due importance to safety and economics.*

*Efforts over the last six decades resulted in evolving the fast breeder technology and it continues to improve as any other technology. Although only a handful of countries are pursuing this technology, India has made significant progress. It was felt that a thematic bulletin on Fast Breeder Reactors would provide the current status of this technology vis-à-vis our contribution and future direction of nuclear energy programme. Accordingly the Guest Editors Dr. P.R.Vasudeva Rao and Mr. C.R. Venkata Subramani of FCD, IGCAR chose the topics and the experts to contribute articles to this thematic bulletin. I am thankful to all the experts for their cooperation in providing well-written articles in time. Guest editors did a commendable job in bringing out this bulletin and their efforts are gratefully acknowledged. IANCAS readers are extremely fortunate to have the views of Shri S.B. Bhoje, Director, IGCAR on the progress made and our projected future in this technology, and I am grateful to him for sparing his valuable time.*

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## ***From the Secretary's Desk***



Dear Members,

*Greetings to you all!*

*Continuous technology promotion, backed by indispensable research efforts, is the constant endeavour at the DAE that has seen India emerging as the Sixth nation to have built and successfully operated a Fast Breeder Reactor at Kalpakkam. Multiplication of fissile inventory is needed to establish a high power base as to engage thorium in the third stage of nuclear power program. India has demonstrated the capabilities and attained maturity in the indigenous development of engineering facilities towards meeting the objective of building its own nuclear power reactors. India has surmounted many hurdles in its mission and has successfully employed the liquid coolant operating systems with advanced carbide fuel whose performance has excelled all the expectations. This bulletin devoted to "Fast breeder reactors" would stand as a testimony to the enormous strides that India has made in the knowledge of the behaviour of fast reactor fuels and materials.*

*IANCAS compliments our colleagues from Kalpakkam, Dr.P.R.Vasudeva Rao and Mr. C.R.Venkata Subramani, respectively the President and Secretary of the Southern Regional chapter of IANCAS, for their painstaking effort as guest editors for this thematic bulletin. We believe that this bulletin, covering the array of topics from the fuel fabrication, safety, sodium chemistry and economics, by renowned scientists in the chosen field would present excellent appraisal of the fast reactor technology to the readers. The earlier bulletin on "Nuclear Reactors" has served to be cornerstone to comprehend the gigantic technology that has evolved in designing, building and operating the breeder reactors.*

*The 47<sup>th</sup> BRNS-IANCAS National Workshop was organized at Jai Narain Vyas University, Jodhpur during 15-24, 2002. Sri B.Bhattacharjee, Director, BARC inaugurated the Workshop and gave the keynote address on the aims of DAE to work for improving the quality of life through nuclear science. The IANCAS team received many invitations from various schools and colleges for lectures. IANCAS has organized 7 such school/college Workshops by giving lectures on Radioactivity and Applications of Radioisotopes and conducted demonstration experiments on half-life determination and shielding aspects.*

*Announcement inviting the nominations for Annual Tarun Dutta Memorial Award and Prof. H.J. Arnikar Best Thesis Award is included in this issue. I request the members to give wide publicity to this call.*

*IANCAS is grateful to BRNS for encouraging and supporting one of the chief activities, namely organizing the National Workshops on 'Radiochemistry and Applications of Radioisotopes' and appreciating the publication of these thematic bulletins with generous grants every year.*

**G.A. Rama Rao**



### ***From the Editor's Desk***

*Dear Members,*

*Greetings to you all!*

*One of the main objectives of our Association, Indian Association of Nuclear Chemists and Allied Scientists (IANCAS), has been to publish periodic bulletins with articles of contemporary importance and interest. Publication of Bulletins started in the very early days of the formation of IANCAS. Continuous efforts by dedicated members with support from appreciative authorities culminated in Quality Bulletins and in 1993 first "thematic Bulletin" with an expert as Guest Editor was published. Needless to state that there has been continuous improvement of Bulletins both in Quality of the articles and presentation style. There was a time when Editors found it difficult to receive acceptance from experts to write articles. It is gratifying to share with you that the situation has improved and now most of the experts are lending their expertise by timely contribution of articles. But for the cooperation received from the experts both from DAE family and academics, IANCAS could have not succeeded in this activity. IANCAS is grateful to the authors and Guest Editors for their cooperation. The Bulletins are being well received and appreciated, by readers as well as DAE authorities.*

*IANCAS has published 23 thematic Bulletins since 1993 and another two are being planned (Table 1). In addition, another four thematic Bulletins were published to commemorate important occasions. In the last executive committee meeting, it was decided to compile all the articles in the form of Volumes. Compilation work is planned and will be completed by January 2003. I urge all the members to send suggestions in this regard to the Editor, IANCAS, c/o Radiochemistry Division, BARC, Trombay, Mumbai 400 085.*

*A.V.R. Reddy  
Editor*

**TABLE 1. List of Thematic IANCAS Bulletins with Guest Editors and Editors**

Sl. No.	Volume	Year	Title	Guest Editor	Editor
1	9(1-3)	1993	Nuclear Techniques in Research	S.B. Manohar	K.R. Balasubramnian
2	10(1)	1994	Trace Element Analysis	P.K. Pujari	K.R. Balasubramnian
3	10(2)	1994	Solvent Extraction : New Perspectives	V.K. Manchanda	P.K. Pujari
4	11(1)	1995	Radiopharmaceuticals & Nuclear Medicine	N. Ramamoorthy	P.K. Pujari
5	11(2)	1995	Radioisotopes in Earth Science	K.L. Ramakumar	P.K. Pujari
6	13(1)	1997	Nuclear Waste Management	S.K. Samanta	M.R.A. Pillai
7	14(1)	1998	Preservation of Food by ionizing radiations	Arun Sharma	M.R.A. Pillai
8	14(2)	1998	Nuclear Fuel Reprocessing	P.R.V. Rao	M.R.A. Pillai
9	15(1)	1999	Nuclear Technology and Biotechnology	P.S. Rao	M.R.A. Pillai
10	15(2)	1999	Neutron Activation Analysis	A.V.R. Reddy	M.R.A. Pillai
11	15(3)	1999	Radiation Protection and Safety Aspects	Pushpa Raja	M.R.A. Pillai
12	15(4)	1999	Ionising Radiation Sources	B.S. Tomar	M.R.A. Pillai
13	16(1)	2000	Nuclear and Radio Chemistry Research in Indian Universities	K.L. Ramakumar	M.R.A. Pillai
14	16(2)	2000	Environmental Quality Monitoring and Assessment	S. Sadasivan	A.V.R. Reddy
15	16(3)	2001	Actinides	V.V. Ramakrishna	A.V.R. Reddy
16	16(4)	2001	Industrial Applications of Radioisotopes	Gursharan Singh	A.V.R. Reddy
17	17(1)	2001	Heavy Water	D.G. Pradhan	A.V.R. Reddy
18	I(1)	2002	Nuclear Reactors	P.N. Prasad	A.V.R. Reddy
19	I(2)	2002	Analytical Spectroscopy	M.D. Sastry	A.V.R. Reddy
20	I(3)	2002	Utilisation of Research Reactors	S.M. Yusuf	A.V.R. Reddy
21	II(1)	2002	Fast Breeder Reactors	P.R. Vasudeva Rao & C.R. Venkata Subramani	A.V.R. Reddy
22	II(2)	2002/03	BRNS interaction with Research Institutes	R.B. Grover	A.V.R. Reddy
23	II(3)	2003	Nuclear Analytical Techniques		A.V.R. Reddy



## Tarun Datta Memorial Award

*Indian Association of Nuclear Chemists and Allied Scientists (IANCAS) invites nominations for the annual Tarun Datta Memorial Young Scientist Award from eligible scientists for their outstanding contributions in the field of Nuclear and Radiochemistry. The award, carrying an amount of Rs. 5000/- in cash, a citation and a medal, will be presented during the Annual General Body Meeting (AGM) of IANCAS to be held during 2003, the date and venue will be intimated later.*

### **Eligibility**

*Citizens of Indian nationality below 35 years of age as on December 31, 2002 and working at least for the last five years with significant contributions in the field of Nuclear and Radiochemistry or Applications of Radioisotopes for the basic research in any branch of science, are eligible to apply.*

*It may be noted that the award would be given for research work carried out in India.*

### **How to apply ?**

*The application should be as per the proforma given on the reverse side of this announcement. Photocopies of the proforma may also be used. Applicants should submit the proforma along with a summary (not exceeding 500 words) highlighting the significant research contributions, during the last five years. In addition, they should enclose two passport size photographs, proof of age, and reprints of best five published papers in support of the application and a declaration by the applicant ratified by the Head of the Department, Research Guide or Head of the Institution. The declaration should also bring out clearly the contributions of the co-workers.*

*The application, complete in all respects should reach the General Secretary, IANCAS on or before November 30, 2002.*

### **Selection**

*An expert panel will scrutinise the applications and judge the best research contribution for the award. The awardee has to present her/his work by giving a lecture during the AGM of IANCAS the date and venue of which will be intimated later. The awardee will be provided with DA and to and fro first class trainfare, if the awardee cannot get the same from any other source.*

*It may please be noted that the decision of the expert panel is FINAL and canvassing in any form is a disqualification.*

*Completed application may please be forwarded to*

*Dr. G.A. Rama Rao  
General Secretary, IANCAS  
C/o Radiochemistry Division  
BARC, Trombay  
Mumbai 400 085*

# PROFORMA

## Application for Tarun Datta Memorial Award

1. Name in full : .....
2. Present Office Address with Telephone, : .....  
Telex, E-mail and FAX number : .....
3. Date of Birth (Attach certificate) : .....
4. Academic Qualifications : .....  
(Attach certificates)
5. Details of employment : .....
6. Awards / Recognitions : .....
7. Field of specialisation : .....
8. Research experience : .....
9. Number of Publications : .....  
(published and accepted)
10. Signature of the Applicant : .....
11. Signature, Name, Designation and full : .....  
address of the proposer of the  
nomination with Telephone Nos.,  
E-mail and FAX : .....

(Please include (i) two passport size photographs, (ii) brief write-up not exceeding 500 words, clearly bringing out significant research contributions, (iii) reprints of best five published papers, (iv) List of publications and (v) a declaration stating that the work was carried out in India)

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### Declaration

(By Head of the Institute or Head of the Department or Research Guide)

I certify that the research work mentioned by Dr. / Mr. / Ms. .... of  
..... (Name of the Institute) was carried out by him / her. The  
candidate is mainly responsible for the outcome of this work. I request the committee to consider the  
nomination for Tarun Datta Memorial Award.

Signature and Seal



## **IANCAS - Prof. H.J. Arnikaar - Best Thesis Award for the Year 2002**

*The Indian Association of Nuclear Chemists and Allied Scientists is happy to announce the institution of an Annual Award for the best thesis in the field of radiochemistry and Allied Sciences. The award carries a medal, a citation and Rs.5,000/- in cash. The awardee will be provided return I class/AC III tier fare to attend the award function, which will be held during the Annual General Body meeting of IANCAS during 2003, the date and venue will be intimated later.*

### ***Elegibility***

1. *Ph.D. Degree awarded by any of the Indian Universities between July 2001 and June 2002.*
2. *The work reported in the thesis should be in any one of the following fields*
  - *Radiochemistry*
  - *Nuclear Chemistry*
  - *Nuclear Materials*
  - *Radioanalytical Chemistry*
  - *Isotope Production*
  - *Radiotracer Studies*
  - *Radioactivity Measurement or*
  - *Any Allied Fields wherein Radioisotopes are Used*
3. *Age limit : There is no age limit for this award*

### ***How to Apply?***

*The nomination should be sent by the Ph.D. Guide along with 3 copies of the synopsis as submitted to the University and a write up not exceeding 500 words highlighting the aim of the work and the most recently reported work from the thesis should also be submitted, along with reprints or manuscripts. It is essential that a copy of the Ph.D. Degree certificate/provisional certificate from the University is also submitted.*

### ***Last Date : on or before November 30, 2002***

*The application, complete in all respects, should reach the General Secretary, IANCAS, C/o Radiochemistry Division, Bhabha Atomic Research Centre, Mumbai - 400 085.*

### ***Selection***

*A panel of experts nominated by the Executive Committee of IANCAS comprising of members from Department of Atomic Energy and the Universities will select the best thesis. It may please be noted that the decision of the panel is FINAL and canvassing in any form will be a disqualification. The awardee will be given an opportunity to present his/her work during the Annual General Body Meeting of IANCAS.*

# PROFORMA

## Application for the Best Thesis Award

1. *Name and address of the Candidate* : .....  
.....  
.....
2. *Name and affiliation of the Guide(s)* : .....  
.....  
.....
3. *Institute where the work was carried out* : .....  
.....
4. *Name of the University awarding the degree* : .....  
.....
5. *Title of the thesis* : .....  
.....
6. *Year of the award* : .....

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### CERTIFICATE

*I hereby confirm that the work pertaining to the Ph.D. Thesis mentioned above of .....  
..... was carried out under my supervision.*

*(Signature of Guide)*



**An Interview with**  
**Shri. S.B.Bhoje**  
**Director, Indira Gandhi Centre for Atomic Research**  
**(IGCAR), Kalpakkam**



***Shri S.B. Bhoje**, Director, IGCAR, joined the Department of Atomic energy through the 9th batch of BARC Training School. Right from the initial days, he specialized in various aspects of fast reactor technology. He was deputed to Centre d'Etudes Nuclear, Cadarache, France, as a member of the design team of the Fast Breeder Test Reactor (FBTR) to be set up at Kalpakkam. He moved over to Reactor Research Centre (presently IGCAR) at Kalpakkam in 1971 and continued to pursue his passion for fast reactor technology, as the Head of the Nuclear Systems Division (1985), Head of Reactor Operations Division and Station Superintendent of FBTR (1989) and Director of Reactor Group (1992). He took over as Director, IGCAR in Nov. 2000. He has received many awards including the Vasvik award for 1992. He is a Fellow of the Indian National Academy of Engineering. He has published over 150 papers in journals and seminars. He is an internationally reputed authority on fast reactor technology and has represented India in International Working Group on Fast Reactors during 1988-1997.*

Dr. P.R.Vasudeva Rao and Shri C.R.Venkatasubramani, Chemical Group, IGCAR, interviewed Shri Bhoje to obtain his views on fast reactor technology, its relevance for our country, and the status of our technology. Given below are excerpts from the interview:

1. ***On behalf of the Indian Association of Nuclear Chemists and Allied Scientists (IANCAS) and on our own behalf we thank you for giving us this opportunity to know more details about the country's fast breeder programme and its pivotal role in the energy scenario.***

*Welcome*

2. ***The first question that everyone asks is whether the fast breeder technology is essential for our country at this stage?***

*Yes, it is very much essential because, if we take the large population of our country and the present world average per capita energy consumption, we would require an installed capacity of about 500,000 MWe. Our current capacity is only about 105,000 MWe. This gap can be closed only by Liquid Metal cooled Fast Breeder Reactors (LMFBRs). In the long run, we have to use our vast resources of thorium, either in thermal or fast reactors. The energy available from coal is very much limited and there are a lot of pollution problems associated with the use of coal. The energy requirement is the main concern for our country with such a large population. Now, with the PHWR programme and the available uranium, we can generate only 10,000 MWe, whereas our requirement is 500,000 MWe. Our department cannot contribute meaningfully to the energy production if we depend only on PHWRs. Also, the PHWRs can be sustained using our available uranium resources for only about 30 years, whereas fast breeder reactors can make a significant contribution for hundreds of years. Fast reactors can use depleted uranium or low grade uranium. The uranium requirement is also low (about 1.3 tonnes per year per GWe). Hence, the cost of fuel will be very low. Therefore, wherever you locate the fast reactor, the fuel mining and transport requirements will be negligible.*

3. ***You Sir, have lucidly put forth the necessity of FBRs for our country. But, other countries except Russia and China are not proceeding with FBRs. Are we choosing a technology that has been given up by other countries?***

*No country has permanently given up fast reactor technology. They have only discontinued working in this area for the present. The option is still open for them to restart their programmes any moment. The main concern of USA today is nuclear proliferation, as fast reactors handle large amounts of plutonium. They also have large amounts of natural uranium, oil, gas, coal etc. At the same time, they have recently realized the need for energy security and there appears to be a renaissance of nuclear power in USA. In addition, USA already has a very high rate of per capita energy consumption (around 12,000 kWh per person), and they can very well reduce this a little by conserving energy. In our case, our consumption is less than 500 kWh per person, and we have an urgent need to increase the energy consumption. Our programmes have to be based on our needs. We only need to choose a technology which is safe and economical.*

4. ***I am very glad that you brought up the question of safety. Safety is one question which is uppermost in the minds of public. How safe are fast reactors vis-a-vis thermal reactors?***

*It is not appropriate to compare in one statement the safety of the two systems – both have to be safe in order to be accepted. The safety record of the nuclear industry worldwide is excellent, as is our own safety record. However, fast reactors do have a few advantages because of their inherent characteristics. For example, the radiation exposure to the operating personnel and to the general public from a fast reactor is less than one-tenth of the dose due to a thermal reactor. Radioactivity is present only in the primary sodium, which is contained. Secondly with the high thermal efficiency of 40% the thermal pollution is very low. Similarly, the radioactive waste generated is also less. There is a large advantage due to natural convection in the case of sodium cooled fast reactors. Such concepts are being proposed even in the case of thermal reactors now (eg. AHWR). The control of fast reactors is very easy as there is no significant change of reactivity with burn-up. All the temperature coefficients are negative, so the reactor power is very stable to the extent that manual control is adequate. The other important aspect of fast reactor safety is the use of liquid sodium as the coolant. Certain special provisions are made in order to avoid reaction of sodium with air or water and we use a sensitive detection system to detect leaks in the steam generator, as this is where the sodium and water may come into contact.*

5. ***Thank you Sir, for the detailed explanation on the safety aspects of fast reactors. I am sure that this would clear most of the mis-apprehensions about safety of fast reactors. However, in the final analysis the Govt. of India or Public will look at the cost of fast reactor technology. What is the current situation and how is it expected to change in the future?***

*In the late 80s and early 90s it had been shown that all the demonstration FBRs are costly, and a lot of efforts have been put in for cost reduction. Europe has been able to cut capital costs by about 67 %. We have for example, reduced the number of loops in PFBR from 4 to 2. This has resulted in a drastic reduction in the number of pumps, heat exchangers, steam generators, piping and instrumentation, electrical systems, civil area, purification circuits, argon circuits etc. Because of such reductions, the cost of power from fast reactor is now in the range of 0.9 to 1.2 times the cost of power from PWRs. In fact, not only the capital costs, operating costs are also expected to be lower. This is because the number of components to be maintained or replaced is also reduced. Similarly construction time is going to be short.*

6. ***The Fast Breeder Test Reactor (FBTR) at Kalpakkam is the first step in the country's LMFBTR programme. Could you let the IANCAS members know what are the highlights and achievements of FBTR?***

*FBTR is the first fast reactor in any developing country. Initially, we had the design of the primary and fuel handling systems from France. We modified the secondary circuit and added the steam generators. The complete civil design and steam water circuit was indigenous. In fact, quite a large part of the design was indigenous. The primary and fuel handling systems were based on Rapsodie design. 80% of the components have been manufactured indigenously. This is a matter of great pride, as FBR requirements are more stringent than other industries, and all the components have been operating very well. The next challenge was posed by the fuel. After the 1974*

PNE, we did not get the enriched uranium required for fabricating the fuel. We developed a unique mixed carbide fuel after some studies in the areas of core physics and engineering, development of process flowsheet for fabrication, etc. This was another important achievement. Sodium has also been manufactured by Indian industry. We have been able to maintain a very high level of purity of sodium from day one, and there has been no corrosion of any component in sodium.

Initially because of the lack of operating experience, we had a few teething problems, which we have overcome. We had, for example, a lot of scrams and problems with instrumentation which have been rectified. All the sodium systems have been operating very well for the last 15 years, except for a small leak which occurred recently. Even in this instance, there was no radioactivity leak into the RCB or sodium fire. The reactor has been put back on line after all checks in minimum time. Each sodium pump has now crossed 1,20,000 h of operation. They have not been taken out at anytime for maintenance. With this trend, we can safely assume that the pumps can run for the entire lifetime of the reactor.

Another important aspect is that the steam generator, which has been indigenously developed for the first time, has operated without a single leak. This is again unique in the fast reactor experience, as elsewhere in the world, large leaks have been reported in the beginning in steam generators. In fact, it is said that if you are successful with the steam generator, you will be successful with the fast reactor, because this is a critical component.

The mixed carbide fuel has attained 100,000 MWd/t burn-up without any fuel pin failure, which is again a world record. Of course, we have taken a long time, and operated the fuel at relatively low linear power rating; even so, the achievement is remarkable. The turbine, which has again been designed and manufactured indigenously, has also performed very well. FBTR is now generating 2.8 MWe (maximum), which is being fed to the grid. The entire exercise has given a lot of valuable training to the operators. We do need to improve on a few aspects such as the number of shutdowns. The capacity factor is around 30 % and needs to be improved.

- 7. Thank you Sir for the vivid explanation of FBTR's performance. It is a matter of pride that we are now supplying power to the grid from our test reactor. Overall can we say that you are fully satisfied with the FBTR experience?**

Well, I cannot say that I am fully satisfied till we reach the rated power of 40 MWt/13 MWe. Further, FBTR is an irradiation facility. Once we put in place a larger core, we can carry out test irradiation of PFBR fuel and other structural materials. In addition to satisfactory operation, we should also utilize the reactor for the intended purpose.

- 8. The next stage of the LMFBR programme involves the design and construction of the Prototype Fast Breeder reactor (PFBR). In what way have we built all the experience of FBTR into the design of PFBR?**

The major problem in the case of FBTR was lack of experience on the industry side and the consequent lack of confidence, which resulted in delay in supply of components and a longer construction period. We got the design for FBTR from France and we did not have a full appreciation of the design, especially the why of it. We did not have the





**Fast Breeder Test Reactor at Kalpakkam**

*complete background on many of the instrumentation systems. In the case of PFBR, we are making all the decisions and documents ourselves. We have the capability for detailed analysis, development and validation of computer codes. Our codes have been independently checked by AERB. No other country has validated all the codes before the start of construction. This level of confidence did not exist at the time of construction of FBTR. In addition, one of the inputs we had from FBTR was the factors to be optimized for cost reduction. The lay-out of FBTR, for example, is big. We have now gone in for a pool type reactor, and chosen a more proven fuel for reaching high burn- up, easy reprocessing and safety in fabrication.*

**9. What is the current status of PFBR ?**

*The current status of PFBR is that design has been almost completed both on the nuclear side and balance of plant. IGCAR is doing the nuclear design, while Tata Consultants and DCL are doing the design on the conventional side. The design is expected to be completed in a month's time. A large amount of R&D has been completed, and the remaining R&D will be completed before end of 2003, so that there are no open issues thereafter. As your members are aware, we have to get clearances from a number of agencies before the reactor construction is started. For PFBR, we*

have already obtained clearance from state government agencies like the Dept. of Forests, Department of Wildlife and the Pollution control Board of Tamil Nadu. We now await the clearance from the Ministry of Environment & Forest, Govt. of India, which is expected by end of September 2002. Regarding nuclear safety, AERB has given clearance for siting, manufacture of nuclear components as well as excavation.

**10. As on date, what is the scheduled criticality date for PFBR?**

We plan to start construction by January 2003 and the reactor should go critical in about 7 years time. Technology development as well as indigenous development of materials has also been completed. We may have to import some materials, mainly to save time and cost, but that would be a small fraction. The project is thus in a good shape.

**11. What is the level of indigenisation of PFBR vis-à-vis FBTR?**

It is better. First of all, the design is ours, all the materials and technology is available indigenously. There is no lack of know-how, and I expect that it would be a totally indigenous reactor.

**12. The Breeder reactor programme essentially depends on closing the fuel cycle and recycling of the fuel. What is your thinking on this subject?**

We will be reprocessing the FBTR fuel by the end of 2002. The R & D on fuel reprocessing is being focused on processing of high burn-up, short-cooled fuels, with minimum loss of heavy metals. I am confident that we will be able to close the FBTR fuel cycle in about two years' time.

**13. What is the breeding ratio in PFBR?**

Presently, the breeding ratio is about 1.1. PFBR has not been optimized for breeding. I consider that our emphasis initially should be economy and safe operation of the reactor, leading to public acceptance. The question of breeding comes later. The Pu available from the thermal reactor programme, would be sufficient to fuel 60 FBTRs of 500 MWe capacity. After, say, 50 years, if we are constrained by availability of plutonium, we should emphasise breeding.

**14. We have gone from a 40 MWt reactor to a 1250 MWt reactor. What was the basis for this scale-up? Why not 1000 MWe, for example?**

There were a number of factors in the decision to go for 500 MWe reactor. Larger reactors like BN-600 and Superphenix would be in operation. Importantly, a proven design of a 500 MWe turbine was available. The design of the balance of plant, which constitutes almost 50% of the plant, was thus available in the country. The development work required was less. We have a large number of 500 MWe coal fired power stations in India, with similar design of turbine. As the component sizes for the 500 MWe system are only slightly bigger than those for the 250 MWe system, we could increase the capacity significantly by slightly increasing the size of the components. This would also make the reactor more economical. Today the industry is in a position to manufacture very large components, so we have confidence in setting up power stations of high capacity. We have learnt a lot from the experience of other countries which have built power stations of capacity 250, 350, 600, 1200 MWe.

**15. What is the level of participation of academics and industries in this project?**

*The academic participation in this project is high. At the time of FBTR, R&D support from academic institutions was very little, whereas for PFBR a large component of R&D has been done in institutions like SERC, Chennai, IIT, Chennai, TBRI and CBRI. We have also started interacting with IISc, Bangalore. We have got good support from these institutions, and very encouraging results. I can say confidently that the component of academic support has been higher for PFBR than for any other DAE project. In fact, Chairman AEC, has said that this is a model to be followed by other units of DAE. The participation of industries in PFBR project is also more. Firstly, they are going to procure the materials for the components as well as take up the manufacture, transport and site installation. There will be more comprehensive supply packages. On the conventional side, such as civil works or steam water system, we will go in for big tenders. However, for critical NSSS components such as the control rod drive mechanism, only those industries who have already participated in the manufacturing technology development will be invited to quote. They have understood our requirements and specifications. Once we give our orders in advance, and give incentives for early completion, the industries should be able to respond well.*

**16. One of the problems that has been agitating the general public is the handling of radioactive waste. What are our plans to handle the radioactive waste which would be generated in PFBR?**

*The radioactive wastes generated by thermal or fast reactors are not different except for the higher amounts of minor actinides in the latter case. For a given level of electricity production, the fast reactor produces lower quantities of waste due to higher thermal efficiency and consequently less amount of fuel is discharged per unit amount of energy produced. One of the advantages of fast reactors is that the other actinides present in waste can be burnt efficiently. Thus, waste management is simpler with fast reactors. However, a lot of R & D has to be carried out before we reach a level of maturity in this area. I believe that we have a lot of time for solving these problems, since the waste produced so far is quite small.*

**17. You mentioned that some countries are still active in fast reactor technology – such as China, Russia and Japan . What are the areas in which we are collaborating with them ?**

*We communicate through IWGFR and other conferences, but otherwise we are indeed working in isolation. After sometime, perhaps, it may be possible to work with other countries. The question is that it is an energy problem for us. The PHWR program has become successful now. Our reactors are operating at 85 % capacity factors. The costs are coming down, the construction time is getting reduced, the confidence has increased tremendously. We will reach the same level of confidence with respect to fast reactors soon. In fact, if the present trend continues, in about 30 years from now, we would become leaders in fast reactor technology. We may have 5 fast reactors in our country by that time, and would be able to offer our technology to others.*

**18. After PFBR what do we expect – are any more reactors already being planned?**

*The Site Selection Committee is identifying two more sites, each to have 4 FBRs of 500 MWe capacity. After PFBR goes into operation, four more reactors of identical design will be constructed before 2020. Parallely, we will design a 1000 MWe reactor, and carry out necessary R&D. The growth of fast reactor capacity can thus be speeded up and the reactor also made more economical.*

**19. With all the years of experience in fast reactor technology, what is your vision for the LMFBR programme, say 20 years from now?**

*We would have commissioned the PFBR and probably another four LMFBRs by then. We would be designing and probably constructing the first 1000 MWe reactor. I believe that over a period of time, we should transfer the technology to the industry, and only carry out R & D within the DAE family. I expect the level of private participation in nuclear power to increase substantially. Even for PFBR, in fact, we have been asked to obtain financial input to the extent of 20% from private bodies. When the energy requirement becomes serious for our country, I expect that more players would emerge on the scene automatically. Looking back, we should acknowledge that it was a wise decision to go in for the fast reactor programme. We are now one among seven countries experienced in fast reactor technology. Only, we have taken a little more time. However, we have learnt a lot in this process. This will go a long way in solving the energy problems of the country and provide us the energy security which is very vital. I would like to emphasise again that fast reactors alone can provide us the energy security. There is no option but to go in for fast reactors.*

***On behalf of IANCAS and on our own behalf, we thank you for patiently answering a number of questions in great detail. We wish that the fast reactor programme proceeds on the lines you have envisaged, and becomes a great success.***

*Thank you.*



## **Fast Breeder Reactors**

*Guest Editor*

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P.R. Vasudeva Rao

## Guest Editorial



C.R. Venkatasubramani

*The world's first fast reactor (Clementine) was designed and built in New Mexico, USA as early as 1946 and this was fuelled by metallic plutonium and cooled by liquid mercury. The Experimental Fast Reactor (EBR 1) was the second reactor built in 1956. This reactor produced 200 kW (e), which was used to light the town of Arco in Idaho in USA. This was in fact, the first instance of a town powered by nuclear electricity !*

*Today, however, the fast reactor programmes in most countries have tapered off, due to various reasons, not related to any drawback of the fast reactor concept. Our country is now one of the few countries in the world, pursuing the fast reactor technology and the decision to go in for fast reactors is expected to pay rich dividends in terms of the energy security of our country. The interview with Sri. S.B.Bhoje, Director, Indira Gandhi Centre for Atomic research, which is spearheading the fast reactor programme in our country, provides an excellent overview of the implications of fast reactor technology.*

*Fast reactors form the second phase of the three-phase nuclear power programme envisioned for our country by the late Dr. Homi Bhabha. They form the important link between the thermal reactor programme, which would generate the plutonium needed to set up fast reactors, and the programme of utilisation of the vast thorium resources of our country through conversion to  $^{233}\text{U}$ . The DAE entered into collaboration with CEA, France to build FBTR – similar to RAPSODIE – FORTISSIMO in the year 1969. This reactor had  $\text{UO}_2$  – 30%  $\text{PuO}_2$  fuel with uranium enriched in  $^{235}\text{U}$  to 85%. In order to overcome the problems arising out of difficulty in obtaining  $^{235}\text{U}$ , first a plutonium rich mixed oxide,  $(\text{U}_{0.24}\text{Pu}_{0.76})\text{O}_2$  fuel system was studied thoroughly; however fuel of this composition was found unacceptable; subsequently, a uranium, plutonium mixed carbide with  $(\text{Pu}/\text{U}+\text{Pu})=70$  was developed at RMD, BARC. The Fast Breeder Test Reactor (FBTR) fuelled by the mixed carbide, was commissioned in 1985. To date, this fuel has achieved a peak burn-up of above 10 % h.a. Presently, the FBTR core is being gradually fuelled with Mark – II fuel-subassemblies containing fuel of composition 55%PuC – 45%UC to raise the power level of the reactor to the designed 42.5 MW(Th).*

*With the rich experience gathered in the design, construction, commissioning and operation of FBTR, a 500 MW(e) Prototype Fast Breeder Reactor (PFBR) is being planned to be set up at Kalpakkam. The reference fuel for this reactor is  $(\text{U}, \text{Pu})\text{O}_2$  having  $\text{PuO}_2$  content of 21% and 28%. The cladding material will be D-9 Stainless Steel. The first few cores of fuel pins required for this reactor will be fabricated at AFFF, BARC, by augmenting the existing facility. Planning for the other aspects of the PFBR fuel cycle is in progress.*

*The new millennium has seen achievement of an important milestone for the Indian FBR programme. The indigenous fuel of FBTR has achieved a peak burn-up of 100, 000 MWd/t. This issue of the IANCAS bulletin on Fast Breeder Reactors thus comes at a very appropriate time. As Shri Bhoje points out in his interview, this is a special achievement, a world record of sorts, in fact, underlining the great strengths in our atomic energy programme. While we join all IANCAS members in congratulating all those who have been involved in the FBTR programme, we also wish the PFBR and the future nuclear programme a great success- our country*

*needs this technology to attain energy security and also increase the per capita energy consumption in our country, which alone would lead to our prosperity.*

*In this issue, we have tried to cover a wide range of subjects related to fast reactor technology. While we are aware of the profile of our readership, we have not been able to avoid a certain measure of technical descriptions which might have limited appeal. This difficulty is inherent when we deal with a complex technology. Readers are most welcome to contact the authors for more details about the subject matter of the articles.*

*We thank the Executive Committee of IANCAS for giving us an opportunity to put together this issue and also profusely thank all the authors of the articles for their invaluable efforts to present a difficult subject.*

# Fast Breeder Reactors : World Scenario



*Shri S. Govindarajan joined the Department of Atomic Energy at IGCAR, Kalpakkam in 1972 from the 15th batch of the BARC Training school. He obtained his BE(Hons) from Annamalai University (1971) and M.Tech. from IIT (Kanpur) (1981). He was involved in the design of FBTR, with special responsibility for core design. He is currently Head, Core Engineering and Component Handling Division and is responsible for the design of the PFBR core and the component handling equipment.*

## Introduction

The potential of breeder reactors to provide inexhaustible source of energy for the mankind was known from the days of the Chicago pile in which the fission chain reaction was successfully demonstrated for the first time. It was Fermi who remarked, soon after the experiment, that development of breeder reactors would end the problem of finding new energy resources for ever. It was also known from the neutronic characteristics of the fissile and fertile nuclei that breeding is favourable under fast neutron spectrum. It is therefore no surprise that the first nuclear reactor to produce electricity was a fast reactor. The development of fast breeder reactors (FBR) has therefore run concurrently with that of thermal reactors over the last six decades. However, certain inherent characteristics of FBR and economic considerations have kept its development in long term perspective.

## Characteristics of Fast Breeder Reactors

The fission cross sections are quite low in fast neutron spectrum and fast reactors require very high fissile concentrations in the core, about 10 times that in thermal reactors, for criticality. In order to conserve total fissile material inventory in the core from economic considerations it is therefore necessary to minimize the volume of a fast reactor core. As a result power density of a fast reactor core is about ten times that in a thermal reactor. This necessitates very efficient cooling medium for fast reactors and liquid metals like sodium are the

suitable candidates. This required development of sodium technology and also of structural materials that are compatible with the coolant. Austenitic stainless steel which is compatible with sodium and capable of operating at high temperatures is extensively used in fast reactors as structural material. It is also required to incorporate a non-radioactive intermediate sodium circuit in the plant between steam generator and the radioactive primary sodium circuit from safety, operation and maintenance considerations. These requirements tend to increase the capital cost of fast reactors. However, the use of sodium has its advantages such as high thermal efficiency, low pressure operation and capability to remove decay heat under natural convection.

## Fuel Utilisation in Fast Reactors

The most attractive feature of fast reactors is the potential to fully utilize the nuclear fuel, most of which is fertile by breeding and recycling. Fuel utilization increases by more than 70 times in fast reactors compared to thermal reactors. This makes the fuel cost of fast reactors to be insensitive to basic uranium price. A typical economic analysis comparing FBR and Light Water Reactor (LWR) may be illustrated [1] as shown in Fig 1. It is seen from the figure that the unit energy cost from the plant is insensitive to uranium price in the case of FBR while it varies significantly in the case of LWR which uses the fuel in once-through mode. The figure also shows clearly that in a scenario of uranium price increase higher capital cost of FBR is affordable. This is a significant fact which had

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guided the FBR design during the early period and had contributed in a way to delay in commercialization of fast reactors.

### Historical Development of Fast Reactors

Over the last six decades the development of fast reactor technology has followed the traditional route of building critical facilities, experimental and test reactors, prototype reactors and demonstration

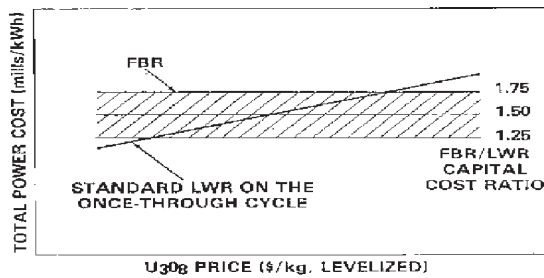


Fig. 1 Transition from the standard LWR on the once-through cycle to the FBR. The horizontal lines represent FBR total power cost for the assumed ratio of FBR / LWR capital costs

reactors. The list of experimental and test reactors is given in Table 1 with certain important details.

The United States had taken an early lead in the development of FBR technology. As many as seven experimental reactors were built there which goes to show the great emphasis that was laid there on this technology. In fact in the early seventies the FBR programme in USA was given equal status with their space programme. EBR II and FFTF operated very successfully and have provided very valuable data on irradiation behaviour of materials and fuel behaviour. Next to USA prominent players in the FBR field were USSR, UK and France. Germany, Japan and India were late entrants.

The construction of prototype and demonstration reactors followed the phase of experimental reactors and these were built in seventies and eighties for technology demonstration and realistic cost estimates. Table 2 lists these reactors.

These reactors have laid a strong foundation for the technology of sodium cooled fast breeder reactors. Phenix and PFR operated very well after a

TABLE 1. Fast Experimental and Test Reactors

Reactor	Country	Date of criticality	Capacity MWt/MWe	Core size litres	Fuel	Coolant
CLEMENTINE	USA	1946	0.025 / —	2.5	Pu metal	Hg
EBR I	USA	1951	1.2 / 0.2	6.0	U metal	NaK
BR 1/ 2	USSR	1956	0.1 / —	1.7	Pu metal	Hg
BR 5/ 10	USSR	1958	5.0 / —	17.0	PuO <sub>2</sub> ,UC, MOX,UN	Na
DFR	UK	1959	60.0 /15.0	120.0	U metal	NaK
LAMPRE	USA	1961	1.0 / —	3.2	Liquid Pu	Na
EFFBR	USA	1963	200.0 /65.0	400.0	U metal	Na
EBR II	USA	1963	62.0 /20.0	73.0	U metal	Na
RAPSODIE	France	1967	40.0 / —	42.0	(U,Pu)O <sub>2</sub>	Na
SEFOR	USA	1969	20.0 / —	566.0	(U,Pu)O <sub>2</sub>	Na
BOR 60	USSR	1969	60.0 / 12.0	60.0	UO <sub>2</sub>	Na
KNK II	Germany	1977	58.0 / 21.0	320.0	UO <sub>2</sub>	Na
JOYO	Japan	1977	100.0 / —	300.0	(U,Pu)O <sub>2</sub>	Na
FFTF	USA	1980	400.0 / —	1040.0	(U,Pu)O <sub>2</sub>	Na
FBTR	India	1985	40.0 / 13.2	55.0	(U,Pu)C	Na

**TABLE 2. Prototype and Demonstration Fast Breeder Reactors**

Reactor	Country	Date of criticality	Capacity MWt/MWe	Core size litres	Fuel	Coolant	Coolant Configuration
BN-350	USSR	1972	1000.0 / 150.0	1900.0	UO <sub>2</sub>	Na	Loop
PHENIX	France	1973	568.0 / 250.0	1300.0	(U,Pu)O <sub>2</sub>	Na	Pool
PFR	UK	1974	600.0 / 250.0	1500.0	(U,Pu)O <sub>2</sub>	Na	Pool
BN-600	USSR	1980	1470.0 / 600.0	2500.0	UO <sub>2</sub>	Na	Pool
SUPER – PHENIX	France	1983	3000.0 / 1200.0	10500.0	(U,Pu)O <sub>2</sub>	Na	Pool
MONJU	Japan	1987	714.0 / 300.0	2300.0	(U,Pu)O <sub>2</sub>	Na	Loop

few teething problems. BN 600 is the most successful prototype reactor logging a most impressive operating record. Of the reactors, only Phenix, BN 600 and Monju are currently in operation. Since the basic premise in the development of FBR technology was protecting against possible rise in uranium prices and conservation of uranium the emphasis had not been on economics while designing these reactors. Superphenix and Monju which were built in eighties cost more than twice the cost of equivalent LWRs. The oil shock in the seventies resulted in conservation of energy and energy demand has saturated in the developed countries. Uranium prices have remained steady and no price escalation is foreseen in the near future. Therefore the fast breeders which are costlier are found unattractive in the changed scenario. In western countries the interest in fast reactors started waning in the nineties. France, a prominent proponent of the FBR technology, took up the design of European 1500 MWe reactor EFR 1500 in the nineties with collaboration from UK and Germany mainly to make it economically competitive with respect to PWRs. Very detailed studies were undertaken and the results indicate that unit energy cost of fast reactors can be brought down within 110% of equivalent PWR cost [2]. Further cost reduction is possible for series construction of reactors. Similar exercise has been carried out in Russia and the results are quite similar.

Apart from the economic considerations there is also the politics of proliferation raised by USA because of the use of plutonium in FBR.

### Current Status

There is slow down in FBR programme in Western countries mainly because of saturation in these countries in energy demand. Proliferation considerations, have also weighed in. France has abiding interest; but has a curtailed program now. All through Russia has shown unabated interest in the development of fast breeders and has recently announced the start of construction of 800MWe BN 800. For Japan which depends on imports for its energy needs the fast reactors would provide high energy security and it has kept its large R & D programme going. China has shown interest in fast reactors during the last decade and construction of the test reactor CEFR of 65 MWt capacity is already in progress near Beijing. A prototype reactor of 300 MWe is being planned as the next stage.

### FBR Programme in India

For India the development of breeder reactors is inevitable as its uranium resources are limited and the vast thorium resources require breeder cycle for exploitation [3,4]. The studies with regard to content of the FBR programme and type of test reactor to be built were undertaken in the early sixties. A collaboration agreement was signed in 1969 with France for technical know-how to build a test reactor in India similar to the French reactor RAPSODIE. A team of Indian engineers and scientists visited Cadarache, France in 1969-70 to finalise the design of India's Fast Breeder Test reactor (FBTR) to be built at Kalpakkam. In order to gain experience with steam generators and the power plant it was decided to add these equipment in FBTR. The construction of



FBTR was started in 1974 and completed in 1984. Critical components of the reactor were manufactured in India with knowhow from France. It was indeed a challenge to the Indian industries as the technical requirements were more stringent than for thermal reactors. The extent of indigenous input to the reactor can be gauged from the fact that only 20% of the total cost of Rupees 800 million was the foreign exchange component, mainly towards know-how and cost of raw materials. Sodium for the reactor was procured from local supplier and about 150 t of sodium was purified to reactor grade at IGCAR.

An important achievement was fabrication of mixed carbide fuel of high Pu content at BARC. This indigenously designed and developed mixed carbide fuel has set a record as mixed carbide fuel is being used as driver fuel for the first time in the world. The first core was designed as a small core of 25 subassemblies since the behaviour of the fuel was not well known. The fuel has operated excellently and now crossed a peak burnup of 100,000 MWd/t without a single pin failure.

After carrying out commissioning of various systems in 1984- 85 FBTR was made critical in October 1985. The reactor produced nuclear steam in January 1993 and reached a milestone when the power level was raised to 10.5 MWt in December 1993. Rolling of the turbine using nuclear steam was achieved in 1996 and this was followed by generation of electricity and synchronisation with grid. The reactor is currently operating at a power level of 17.4 MWt/2.2 MWe.

Other highlights of the operation of FBTR are excellent performance of the sodium pumps, intermediate heat exchangers and steam generator.

As a logical follow-up of FBTR, it was decided to construct a prototype reactor. The working group constituted for studying the technical aspects of the prototype reactor including feasibility of indigenous design and construction recommended a 500 MWe pool type sodium cooled fast reactor to be designed and constructed indigenously. A size of 500 MWe was chosen for the reactor as an optimum from the point of view of scale up from FBTR and economics. The design of 500 MWe prototype fast breeder reactor PFBR was taken up in the eighties and a

review of the design at the end of the decade indicated that there were many areas in which cost effective designs could be employed. The following decade was therefore fully devoted to reviewing and revising the design of the reactor. The design is now complete and the detailed estimates of cost indicate that it is competitive to its PHWR counterpart. Major changes were made in the number of primary circuit components, number of secondary sodium circuit loops and steam generator modules. The construction of the reactor is expected to start in 2003.

### **Description of PFBR**

The main characteristics of the plant is given in Table 3 and the schematic flow sheet is shown in Fig. 2.

PFBR is a sodium cooled, mixed oxide (MOX) fuelled, pool type fast reactor. The core thermal power is 1253 MW and the gross electrical output is 500 MWe. The active core, in which most of the heat is generated by controlled fission, consists of 181 fuel subassemblies (FSA). There are 2 rows of radial blanket subassemblies. 12 absorber rods viz., 9 control and safety rods (CSR) and 3 diverse safety rods (DSR) are arranged in two rings. Enriched boron carbide is used as absorber material for reactor control and shutdown.

A single grid plate is used to support all the core subassemblies (CSA) and to distribute coolant to the CSA. The coolant is pumped by the two sodium pumps into the grid plate and the coolant flows upward through the core to remove the fission heat. The inner vessel separates the hot and cold pools of sodium. The hot sodium flows by gravity through the four intermediate heat exchangers to transfer heat to the secondary sodium which in turn transfers heat to water in the steam generators to produce steam.

The main vessel (12.9 m in diameter) is supported at the top by welding to the outer shell of the roof slab and is free to expand downwards. The total weight of primary sodium is 1150 te. A safety vessel is provided as a second barrier to contain sodium in the remote event of any sodium leakage from the main vessel.

The roof slab supports the large rotatable plug, primary sodium pumps, IHX and heat exchangers of





**TABLE 3. Main characteristics of the PFBR plant**

Thermal power, MWt	1250
Electric output, MWe	500
Core height, mm	1000
Core Diameter, mm	1900
Fuel	(Pu-U)O <sub>2</sub>
In-pile fuel inventory, t	9.2
Pins per fuel subassembly	217
Fuel pin outer diameter, mm	6.6
Maximum neutron flux, n/cm <sup>2</sup> -s	8 x 10 <sup>15</sup>
Fuel clad material	20% CW D9
Absorber material	enriched B <sub>4</sub> C
Primary circuit layout	Pool
Number of primary sodium pumps	2
Number of IHX	4
Primary inlet / outlet temp, K	670 / 820
Primary sodium flow, t/s	6.8
Number secondary sodium loops	2
Number steam generator modules	4 / loop
Steam temperature, K	766
Steam pressure, MPa	16.6

sodium water reaction. Sodium purity is maintained in the primary and secondary circuits by cold traps.

All sodium piping inside the RCB is provided with nitrogen inerted guard piping to ensure that no sodium from primary sodium purification, secondary and safety grade decay heat removal (SGDHR) circuits can enter the containment volume in the event of a leak. Continuous leak monitoring of the inter-pipe annulus is provided by sodium leak detectors.

A passive decay heat removal system is provided taking advantage of the excellent heat transport properties of sodium to remove decay heat from core by natural convection. In case of off-site power failure or non-availability of steam-water

system, the decay heat is removed by safety grade decay heat removal (SGDHR) circuit consisting of four identical loops, each of which can remove 8 MW when the hot pool temperature is 803 K.

The steam-water system adopts the standard turbine that is used in fossil fired thermal power stations of the same rating. The steam cycle employs regenerative feed water heating based on steam bled from the turbine. A turbine bypass of 60% capacity is provided to facilitate bypassing the main steam and restart of the unit after a minor turbine fault. To remove the heat rejected from the steam cycle in the condenser, a once-through condenser cooling water system is employed using sea water.

The plant will be connected to the Tamil Nadu/Southern Regional Grid to transmit the power generated and these connections also provide off-site power supply to the station. A 220 kV substation with five numbers of transmission lines and double circuit ties to Madras Atomic Power Station (MAPS) 220 kV bus is provided. An indoor switchyard is provided to increase the reliability of the electrical equipment against the saline atmosphere.

### Summary

The technology of fast reactors is well proven and there are also strong indicators that they are economically competitive. Currently the programme is being pursued in Russia, France, India, Japan and China. It is expected that western countries who were the pioneers in the field would turn to fast reactors after about two decades because of many attractive features of fast reactor and its potential to facilitate waste disposal.

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# Operating Experience of Fast Reactors



*Shri R.P. Kapoor is a B.Tech in Chemical Engg. from IIT, Delhi and MS in Nuclear Reactor Technology from University of Aston, UK. He joined the department in 1968 and belongs to the 11<sup>th</sup> batch of the BARC Training School. He worked in various capacities to develop the multi-disciplinary fast breeder technology and participated fully in the commissioning and operation of FBTR. He has also steered the commissioning of the 30 kW KAMINI reactor which is utilized for neutron radiography and activation analysis. Shri Kapoor is presently the Associate Director, Reactor Operation and Maintenance Group, IGCAR.*

## Introduction

Fast Reactors have operated well in seven countries viz, USA, former USSR, UK, France, Germany, Japan and India and recently China has also launched its FR programme. India started its programme in collaboration with France in 1969 for technical know-how exchange to build an experimental reactor in the country. The responsibility of construction was totally with India with the participation of Indian industry [1]. Construction of the Fast breeder Test Reactor (FBTR) started in 1974 and completed in 1984. After completing the commissioning activities, the first criticality was achieved in Oct 1985. The reactor has satisfactorily operated for the past 16 years to systematically achieve various set objectives related to understanding of reactor physics parameters, design and operation of high temperature liquid sodium systems, conduct of safety related physics and engineering tests, fuel and structural materials performance, demonstrate generation of electricity, generate a core group of professionals and provide continuous experience feedback to undertake future commercial projects.

This paper focuses on the operating experience with fast reactors with special reference to our country.

## General Characteristics of Fast Reactors [2]

### Neutronics

The neutron spectrum in a fast reactor core is in the high energy range (200 – 600 keV) due to the

absence of moderator. As the fission cross sections are low for high energy neutrons, high fissile concentrations are required (around 20% for commercial and much larger for test / experimental reactors). To reduce total fissile inventory in the core for achieving low fuel cycle cost, core volume for a given power output is kept low leading to high power densities (more than 500 kW/litre). This necessitates use of an efficient heat transport medium as primary coolant. Neutron flux in a fast reactor core is 10 times higher than in a thermal reactor. ( $\sim 7 \times 10^{15}$  n/cm<sup>2</sup>/s) necessitating use of radiation resistant structural material in the core.

### Coolant

The advantages of using liquid sodium as the coolant are discussed in detail elsewhere in this issue. Owing to the high operating temperature and highly reactive nature of sodium coupled with induced radioactivity of primary coolant sodium while passing through the core, certain constraints on the design of process instrumentation are imposed. All the sensors designed for use in sodium should perform their function without coming in direct contact with sodium, i.e., the sensors should be located in pocket/well. On the positive side, the low electrical resistivity of sodium ( $20 \times 10^{-6} \Omega \text{ cm}$  at 400°C) permits exploitation of electrical/ electromagnetic principles of measurement for designing process instrumentation for flow, level and leak detection. Principles of electrical continuity and sodium ionization are also used for designing probes for leak/fire detection. Since liquid sodium is opaque to natural light, under-sodium viewing to

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assess the health of submerged reactor internals is adopted by ultrasonic probes specially developed and used in reactor environment. Sodium density and viscosity, being close to water, permit use of water for various hydraulic experiments to finalise design parameters.

As a consequence to low pressure and high temperature of operation, thermal stresses assume importance and the design of thin walled component and pipe lines has to account for thermal expansion / contraction during operation. Creep and fatigue are the primary damage mechanisms. Thermal stripping phenomenon leading to random temperature fluctuations needs to be considered for designing in-sodium structures and sodium retaining boundaries. Thermal hydraulics of high temperature sodium free levels under inert atmosphere promote convection currents carrying sodium aerosols to upper structures causing circumferential temperature asymmetry. Aerosol deposits in colder narrow passages can hinder relative movement. Design should ensure to suppress these convection currents.

### ***Structural Material***

Austenitic stainless steels of grade 304, 316, 304L (N) and 316 L (N) exhibit excellent compatibility with sodium, have adequate high temperature strength and are extensively used in FRs. For core components, the choice is dictated by resistance to neutron induced void swelling and austenitic stainless steel containing Ti & Si and D-9 cold worked to 20% is currently considered as a suitable material for fuel cladding as well as hexagonal wrapper tube. For steam generators, Cr-Mo steels of type 2.25Cr-1 Mo and 9 Cr-1 Mo are chosen for their good compatibility with sodium and water.

### ***Fuel***

One of the parameters by which efficiency of breeding is measured is doubling time, which is the time required by the reactor to generate enough surplus fuel to start another reactor. Doubling time is indicative of the growth potential of fast reactor system and it depends very much on the choice of fuel. Metallic fuel like U-Pu-Zr alloy fuel is the best in this respect; but due to problems of fabrication and

irradiation stability, ceramic fuels are favoured. The ceramic fuels that have been considered for FBRs are oxides, carbides and nitrides. Of these, the latter two are similar and give better breeding. However, world experience indicates that oxides give the most reliable performance with respect to burnup and for this reason, they are universally preferred.

### ***Safety Aspects***

Controllability and inherent stability of fast reactor has been well established as increase in temperature in the core results in reduction of power due to negative temperature coefficient of reactivity. Sodium as coolant, presents many attractive safety features. Due to its high boiling point (~880°C) resulting in large thermal inertia and its capability to remove decay heat in natural convection mode, consequences of loss of flow accidents are mitigated. Operation of the system at low pressure reduces the probability of pipe leakage and due to presence of double containment, the probability of loss of coolant accident is also negligibly small. Redundancy and diversity are incorporated in the reactor shutdown system in order to ensure high reliability of reactor protection system.

### ***Operating Experience-World Scenario [3,4]***

Fifteen test / experimental reactors have been built and operated satisfactorily in seven countries viz, USA, UK, former USSR, Germany, France, Japan and India during the past 55 years. All these reactors have accomplished their set objectives to enrich knowledge base for this frontier technology. Presently four experimental reactors; two in Russia (BR-10, BOR 60), one in Japan (JOYO) and one in India (FBTR) are operational. China has also launched its fast reactor programme recently and has started construction of 25 MWe experimental reactor (CEFR) with first criticality scheduled by end 2005.

Six prototype / commercial demonstration plants have been built and operated in four countries during the past 28 years viz; two in former USSR (BN 350 and BN 600), two in France (Phenix and Superphenix), one in UK (PFR) and one in Japan (MONJU). The test and prototype reactors together have accumulated more than 310 reactor years of safe and reliable operational experience. Three of the

prototype reactors have since closed down and the present status of the other three reactors is as follows;

- BN 600 reactor (600 MWe) in Russia is operating since 1980 with an overall load factor of 72% and gross thermal efficiency of 42%. Design life of this reactor expires in 2010 and work is under way for life extension for 10 more years.
- Phenix reactor (250 MWe) in France, operating since 1974, is presently undergoing renovation programme and safety upgrade based on current standards and is due to restart by end 2002.
- MONJU reactor (280 MWe) in Japan, commissioned in 1994, is also undergoing improvements based on comprehensive safety review and countermeasures against sodium leak incident which occurred in Nov 95 and is due for restart by 2005.

Russia has started construction of a new fast reactor of 800 MWe capacity (BN-800). India also plans to start the construction of 500 MWe capacity reactor (PFBR) towards the end of 2002. The salient operating experience in the world can be summarized as follows;

- The technology of fast reactors, in particular the technology of sodium coolant has been mastered and it has been shown conclusively that it is possible to build and operate a fast reactor for electricity production safely and reliably with acceptable availability and capacity factors.
- The basic technology of breeding i.e recycling plutonium and thus closing the fuel cycle in a commercially acceptable manner has been demonstrated.
- Reactor safety experience has been good and sodium cooled fast reactors have continued to give particularly low radiation doses to operating personnel and low radioactive releases to environment.
- For achieving high burnup which is a very important factor in the economy of fast reactor fuel cycle, ceramic fuels (Pu-U mixed dioxide and to a great extent carbides and nitrides) have

been irradiated upto 20 atom % burnups (test pins have achieved even higher burnups).

- Large scale experience in decommissioning has not been accumulated. Experience obtained from decommissioning of experimental reactors (EBR-II in USA and Rapsodie in France) has been satisfactory and plans are underway to decommission prototype/commercial plants in UK (PFR), France (Superphenix) and Kazakhstan (BN-350).
- Capital cost of fast reactors being high, for commercial viability, several design optimization studies are being carried out in many countries including India to reduce the cost so that it is competitive with thermal nuclear plants and fossil power plants.
- Like in any technology, there are still certain areas where further development and improvements are necessary and major among these are:
  - (i) Highly enriched fast reactor cores are not in the most reactive configuration, hence in case of core disruptive accident (CDA), reliable means to prevent re-criticality need to be found so that core material can be relocated in a safe configuration e.g, incorporation of core catcher etc.
  - (ii) Sodium coolant void coefficient of reactivity for commercial size reactors is positive near the center of the core and unexplained minor reactivity changes have been observed in some of the reactors. Core design improvements are constantly being made to either considerably reduce the sodium void coefficient or if possible make it negative.
  - (iii) Sodium being highly reactive, sodium leaks resulting in fires and large downtime has been of concern in most of the reactors. While it is difficult to design for zero leak, the number of leaks can be reduced by improvement of design methods incorporating various service conditions to which the structural materials are subjected to. In addition acceptance of the “leak-before-break” approach will also protect against major failures and large leaks.



Development in the field of reliable methods for detection of small leak at an incipient stage, protection against the effect of fires by means of improved segregation and faster sodium dump system, better methods for extinguishing sodium fires and protection against sodium smoke to mitigate the consequences are some of the areas where considerable amount of work is being done.

- (iv) Sodium-water reaction in steam generator is another area where large effort is being made to detect the leak at the incipient stage itself so that the unit can be isolated to mitigate the damage consequences. Several diverse methods of leak detection, including hydrogen monitors, pressure measurement, acoustic noise detection etc are being perfected.
- (v) To counter some of the adverse characteristics of sodium lead and lead-bismuth alloy have been considered as an alternative to sodium as coolant, especially in Russia. But its very high melting point (needing additional energy for initial preheating and subsequent recirculation), very high density (needing large pumping power), highly corrosive nature (with stainless steel as structural material) and generation of  $^{210}\text{Po}$  by neutron capture in  $^{209}\text{Bi}$  and  $^{208}\text{Pb}$  (which is radioactive) has restricted its large scale use. The consensus world wide still remains very strongly in favour of sodium which is corroborated by the fact that in the last few years sodium has been chosen as coolant by countries like China and South Korea who have recently ventured into the fast breeder reactor field.
- There is a continuous and encouraging development in the improved methods for non-destructive in-service inspection to demonstrate the integrity of sodium vessels and pipe work and to detect incipient defects so that these can be repaired before they grow to cause leaks. The advancement in repair technology has also helped in minimizing reactor outages and reduce repair costs. These technologies not only help better plant availability but the data generated is utilized for assessing residual life of the components to make a case for plant life extension.



Fig. 1 Fast Breeder Test Reactor at Kalpakkam

## Operating Experience-Fast Breeder Test Reactor [5]

### Description

Fast Breeder Test Reactor (FBTR) is a 40 MWt / 13.2 MWe loop type, sodium cooled, plutonium rich mixed carbide fuelled reactor (Fig. 1). Heat generated in the core is removed by two parallel sodium loops and transferred to the corresponding inactive secondary sodium loops through vertical shell and tube type intermediate heat exchangers (IHX). Each secondary sodium loop is provided with two once-through steam generator (SG) modules. The superheated steam from the SG in both the loops is fed to a common steam water circuit comprising a turbo generator (TG) and a bypass steam circuit with 100% dump condenser (DC). In case of non-availability of turbine, reactor operation is continued for experimental purposes bypassing the steam to the DC. The waste heat from the condenser is finally rejected to an induced draft cooling tower (Fig. 2). Stainless steel (SS 316) is the principal material of construction for reactor vessel (RV), primary and secondary sodium circuits. Since a unique Pu rich U monocarbide fuel with indigenously available resources was chosen as driver fuel [6], as a cautious approach, the core size was kept small initially to assess fuel performance.

### Summary of Operation

After completing the commissioning activities during 1984-85, FBTR achieved first criticality in Oct 1985. The reactor was operated up to 1 MWt till 1992 for physics and engineering tests [7]. The SG

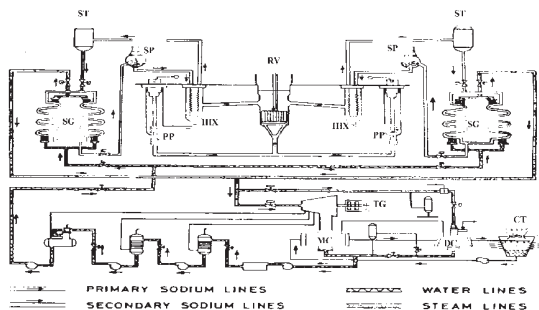


Fig. 2 FBTR Schematic Flow Diagram

and steam water systems were commissioned in 1992 thus providing the heat sink for increasing the reactor power. The SG was put in service in Jan 1993 and the TG was synchronized to the southern grid in July 1997. Since first criticality in 1985, the reactor has completed 10 irradiation campaigns with a maximum operating power of 17.4 MWt/2.5 MWe. During this period, the reactor has operated cumulatively for more than 29,000 h with about 15,000 h at high power. TG was on line for 2400 h generating more than 2 million units of electrical energy. The present reactor core has 35 fuel subassemblies (SA) and it will be expanded in a phased manner to 76 fuel SA core in the next two years to achieve nominal power operation. The main characteristics of FBTR small and nominal core are given in Table 1.

#### Fuel Performance

There was no in-pile irradiation data about the Pu rich U monocarbide fuel chosen as driver fuel for the reactor and the reactor itself served as an irradiation facility for ascertaining the fuel performance in terms of Linear Heat Rating (LHR) and burnup (the amount of energy extracted per tonne of fuel). Hence the core size was kept small and reactor power was increased in stages after assessing the fuel performance through out-of-pile simulation studies, plant observations, post irradiation examination (PIE) of test pins and fuel SA at various burnups [8] and rigorous theoretical analysis. Starting with a conservative LHR of 250 W/cm and fuel burnup of 25 GWd/t, gradually the core size was enlarged and fuel performance parameters enhanced to 400 W/cm (LHR) and more than 100 GWd/t (11 atom % burnup) respectively.

PIE has indicated that the fuel is performing very well with very low swelling rates and no evidence of fuel clad mechanical interaction (FCMI). There has been no fuel clad failure so far. It is proposed to carryout PIE of fuel at 100 GWd/t burnup to assess further enhancement of fuel performance. The performance of this indigenously developed fuel has been excellent and up to international standards. Irradiation experiments are planned to study the performance of mixed Pu-U oxide fuels chosen for PFBR.

#### Nuclear Steam Supply System

- (i) The sodium circuits and components like the reactor vessel, intermediate heat exchanger, mechanical sodium pump, etc. have operated satisfactorily for the past 16 years at a maximum temperature of 445°C [9]. Some salient observations are:
  - Mechanical sodium pumps have operated for 1,20,000 hours satisfactorily.
  - The purity of the coolant sodium has to be nuclear grade (Table 2) and needs to be well maintained not only at the time of charging the circuit but also during regular operation to minimize corrosion of structural materials, avoid plugging of small diameter pipe lines and narrow coolant passages in fuel subassemblies in the reactor core. The purification of 150 tonnes of commercial grade sodium for the reactor was carried out by processes involving melting of sodium, settling and coarse filtration to remove metallic impurities, removal of calcium by converting into insoluble calcium oxide followed by micro-filtration and removal of soluble impurities (mainly oxygen and hydrogen) by the process of cold trapping before charging to the loops. Both the primary and secondary sodium circuits have been provided with on line purification system and monitoring circuit to limit oxygen and hydrogen impurities with provision to take sodium sample from the circuits for chemical and radiochemical analysis.
  - The nuclear grade purity has been well maintained. Electrochemical carbon meter developed in the center was incorporated in one

**TABLE 1. Main Characteristics of FBTR**

Parameter	Small core	Nominal core
Reactor Power MWt/MWe	17.4/2.2	40/13.2
Fuel Mark I	PuC 70%-UC 30%	-
Mark II	PuC 55%-UC 45%	PuC 55%-UC 45%
Fuel pin diameter mm	5.1	
No. of pins/SA	61	
Maximum LHR W/cm	400	350
Maximum burnup MWd/t	1,00,000	
Peak neutron flux n/cm <sup>2</sup> s	2.75 E 15	3.5 E 15
No. of control rods	6	
Control rod material	B <sub>4</sub> C (90% in B <sup>10</sup> )	
Speed mm/s	1.0	0.5
Control rod worth (pcm)	7680	13000
Sodium temperature (°C)		
RV in/out	355/445	380/520
SG in/out	435/290	515/295
Primary sodium loop flow (m <sup>3</sup> /h)	300	550
Secondary sodium loop flow (m <sup>3</sup> /h)	170	300
Sodium inventory (tonnes)	150	
Feed water flow (t/h)	27	70
Feed water / steam temp. (°C)	190/400	190/480
Steam pressure (kg/cm <sup>2</sup> )	125	
Steam generator	Once through type, modular construction	
TG	16 stages, condensing type, 16.4 MWe air cooled	

of the secondary sodium circuits for on line measurement.

- Four incidents of leaks in sodium auxiliary circuits have taken place. The maximum amount in three minor leaks in the inactive circuits was 2.5 liters. Suitable procedural and design modifications were carried out, after identifying the causes, to prevent recurrence. In April 2002, about 75 kg of sodium leaked from the primary sodium circuit. The cause of the leak was the defective manufacturing process adopted in the manufacture of the bellows sealed sodium service valves. Necessary action was taken to dispose the sodium, rectification works carried out, leak detection provisions /procedures strengthened and the circuit and reactor returned to operation. The man-rem

**TABLE 2. Sodium specifications**

Impurity	Commercial grade (ppm)	Nuclear Grade (ppm)
Oxygen	High	< 10
Hydrogen	High	< 2
Carbon	30 – 50	< 30 ppm
Calcium	40	< 10
Sodium	98.5%	99.95%

expenditure during this procedure was 130 person-millirem, which is negligibly small.

- A number of thermal-hydraulic phenomena were observed, studied and satisfactorily resolved. One of them related to the

non-uniformity of the circumferential temperature differential in the upper regions of the reactor vessel wall. This phenomenon caused deflection of the grid plate leading to problems in operation of control rods and fuel handling grippers [10]. Helium injection in the reactor vessel cover gas space minimised the temperature differential and satisfactory operation of control rods and fuel handling grippers was possible.

- In-service inspection of reactor vessel internals has been carried out periodically by indigenously developed periscope and ultrasonic under-sodium scanner to ascertain non-plugging of safety related pipes by observing actual sodium flow, checking abnormalities / sodium oxide deposition on internal surfaces, ensuring mirror like finish of liquid sodium surface free of impurities. These observations have generally revealed healthy conditions of reactor vessel internals [11].
  - The sodium heated once-through steam generators have operated satisfactorily for about 600 days without any SG tube leak. The SG is provided with diverse methods of leak detection, viz, diffusion type and electrochemical hydrogen meters, pressure sensors and rupture discs to safeguard this critical equipment from damage. These instruments have been developed in-house and have performed satisfactorily. The compact air-cooled condensing type turbine, specially designed by BHEL, has operated satisfactorily, to generate electricity.
- (ii) A fuel handling incident took place in 1987 and the reactor could be restored to normal operation only after two years. Special operations to recover from the incident enabled development of novel remote inspection techniques/tools. Remedial measures implemented after this incident have enabled about 375 handling operations in 35 fuel handling campaigns to be carried out satisfactorily.
  - (iii) A number of instrumentation and control problems were faced during the first few years of operation. However, operation of the reactor

in the manual mode by adjusting control rods to compensate for the loss of reactivity due to burnup has been found to be stable. The neutronic instrumentation system has been fully replaced by a state-of-the-art system in 1998. Continuous improvement/modifications in various instrumentation system have helped in reducing the number of trips to one third during the past five years.

- (iv) Biological shield concrete surrounding the reactor block is cooled with demineralised water flowing through embedded carbon steel coils to maintain the concrete temperature within 80°C. This circuit has operated satisfactorily till 2001, when leaks in some of the embedded coils were noticed. These leaks, located in an inaccessible zone, are most probably due to crevice corrosion in the socket welds and were sealed using an on-line procedure. Normal power operation was resumed within a few months of the incident.

#### *Safety Experiments*

It is essential and mandatory to carry out important safety related physics and engineering tests to validate all assumptions made in the safety report to ensure plant safety under various anticipated incidental situations. The data obtained from these tests also help the designer to improve mathematical modeling for better prediction. Summary of salient tests conducted are as follows:

- (i) Temperature, power and void coefficient of reactivity have been measured and found to be negative thus emphasizing the inherent safety features built in the design. Failed fuel detection and localization experiments were performed, by loading perforated natural uranium pin at different locations. The results confirmed the capability of the system to detect failed fuel. Kinetics experiments have helped in providing the time constants for reactivity feed back mechanisms to ensure the stability of the reactor [12,13].
- (ii) Extensive experimental/theoretical investigations lasting for a cumulative period of 2 years were carried out to explain the small reactivity transients observed in Nov 94, Apr 95 and Aug 98. The most probable cause



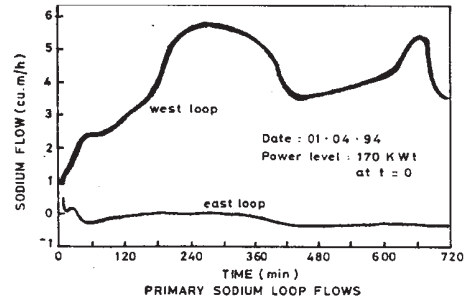
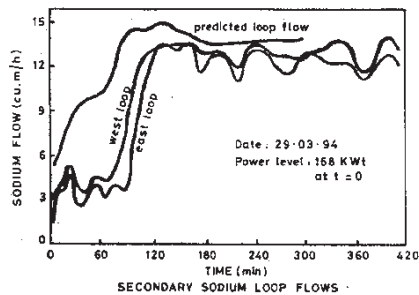


Fig. 3 Natural convection tests

appears to be slight, thermally induced, geometric changes to the small size core. It has been established that the reactivity inputs are small (35-40 pcm), slow, reversible, repeatable and regimes are well defined. These experiments have indicated that this phenomenon is expected to disappear when the core is enlarged to nominal size [14].

- (iii) The capability of normal and emergency cooling systems to safely remove the reactor decay power during various postulated incidental scenario, were tested. Tests for various under-cooling events during power operation viz, tripping of one pump in a coolant loop and failure of offsite power supply revealed that all safety actions take place as per design intent and temperature transients seen by the components are as predicted [12].
- (iv) During station blackout, the sodium circuits have been designed to operate in natural convection mode for ultimate decay heat removal. Two natural convection tests were conducted at 170 kWt power, one by tripping both the secondary sodium pumps and the other by tripping both the primary sodium pumps. The natural convection flows established are indicated in Fig. 3. In both the tests, the flow was sufficient to remove the decay heat.

#### Radiological Protection

The general radiation levels in all the accessible location in reactor containment building (RCB) during operation at 17.4 MWt power varied from 0.1 to 1.5 mR/h. During the past 16 years, there

has been no significant event of abnormal radioactivity release, personnel or area contamination thus fortifying the worldwide view that the fast reactor concept gives low radiation doses to operating personnel and low releases to environment. Integrated leakage rate tests of RCB have indicated that the leakage rate is lower than the permissible limit, thus indicating sound health of the containment structure against release of radioactivity

#### Summary

For India's future growing energy needs, contribution from nuclear energy is inevitable. Fast breeder reactors enhance utilization of available natural uranium fuel resource by about two orders of magnitude. Large operational experience in the world during the past 50 years has established that this is a viable technology which can be safely exploited.

India's fast breeder programme is progressing steadily with satisfactory operation of FBTR for the past 16 years thus demonstrating the mastering of this multi-disciplinary technology for energy production [15]. Satisfactory experience with fuel, sodium systems, steam generators and instrumentation has established that required technological infrastructure exists in the country to deliver high quality products. Operational feedback has been utilized for carrying out safe and optimum design of 500 MWe prototype fast breeder reactor.

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# Chemistry of Fast Reactor Fuels



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## Introduction

The fuel is the heart of the reactor where the fissions take place to produce power. The fuel for FBR can be metal, alloy, oxide, carbide or nitride. The chemistry of these materials has to be understood during the preparation to the stringent specification.

The fuel and structural material undergo 4 to 5 displacement per atom in thermal reactors and 15-40 in fast reactors. The fission products formed also vary from gases to solids with different chemical reactivities. In fast reactors, the production of the fission products is higher and hence the fuel and structural materials should be designed in such a way to withstand the aggressive environment and reactions. The present article deals with the chemistry aspects of fast reactor fuel development mainly with respect to their fabrication, out of pile and in-pile behaviour. A detailed article related to reprocessing appears elsewhere in this issue.

## Fast Reactor Fuel

The choice of fuel in fast reactors is governed by factors such as feasibility of breeding of plutonium, closing of fuel cycle (reprocessing), achievement of high power operation and high burnup, ease of fabrication and fuel cycle economics. The present generation of prototype and large fast reactors use (U,Pu)O<sub>2</sub> with 20-30% Pu as fuel. However, advanced fuels like (U,Pu)C, (U,Pu)N and U-Pu-Zr alloys have much higher breeding potential. The features, which are important are melting point, thermal conductivity, density and absorption cross section of the light elements. The oxide has relatively low heavy atom density, high light atom fraction and poor thermal conductivity compared to the carbide or nitride.

## FBTR Fuel

The fuel used in FBTR is a mixed carbide of uranium and plutonium with a Pu/(Pu+U) ratio of 0.7 for Mark I core and Pu/(Pu+U) of 0.55 for Mark II core. A high fissile content has to be used for a small experimental reactor. This cannot be achieved with oxide of composition (U<sub>0.3</sub>Pu<sub>0.7</sub>)O<sub>2</sub> as this fuel is not compatible with clad and coolant. Also, the fuel will

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have high oxygen pressure during burn up leading to very large oxidation of cladding. If stoichiometry is decreased to an O/M of 1.98, then thermal conductivity loss is not manageable. Further the fuel reacts with coolant in case of breach leading to the formation of bulky  $\text{Na}_3(\text{U,Pu})\text{O}_4$  which causes swelling. These considerations have led to the use of the uranium, plutonium mixed carbide fuel in FBTR. The entire range of physico-chemical properties of the high plutonium content mixed carbide has been studied at BARC and IGCAR.

### ***PFBR Fuel***

The Prototype fast breeder reactor (PFBR) will use a uranium-plutonium mixed oxide which has been most widely used throughout the world and the fuel specifications are given in Table 1.

### **Chemistry During Fuel Manufacture**

#### ***Oxide Fuels***

The method adopted involves blending of  $\text{UO}_2$  and  $\text{PuO}_2$  powders followed by pelletising and sintering. Alternatively,  $\text{UO}_2$ - $\text{PuO}_2$  can be generated directly using coprecipitation, direct co-denitration or gel sphere process. In coprecipitation, uranium and plutonium are either precipitated as ammonium diuranate and plutonium hydroxide or as a mixture of ammonium uranyl-plutonyl carbonate, filtered and dried. In direct thermal denitration, solutions of uranium and plutonium nitrates are heated causing concentration and subsequently direct denitration.

#### ***Preparation of $\text{UO}_2$ and $\text{PuO}_2$***

$\text{UO}_2$  can be prepared by several methods which include direct denitration, ammonium diuranate (ADU) precipitation, ammonium uranyl carbonate (AUC) precipitation and peroxide precipitation. The conversion processes mainly used depend on the precipitation of uranium as uranates or carbonates followed by their subsequent conversion to  $\text{UO}_2$ . The ADU precipitate is amorphous and the size of ADU particles and agglomerates depends on the precipitation conditions. In order to obtain the precipitate, which can yield  $\text{UO}_2$  powder of desired characteristics, it is necessary to carefully control the conditions to obtain a blend of coarse and fine particles or carry out precipitation in two stages. The ADU precipitate is converted to  $\text{UO}_3$  powder by

**TABLE 1. PFBR Fuel Specification**

Fuel	$(\text{U,Pu})\text{O}_2$
$\text{PuO}_2$ enrichment	21 and 28%
Pin diameter	6.6 mm
Linear heat rating	450 W/cm
Burn up	100,000 MWd/Te
Failure rate	Less than 0.0001

#### **Fuel Pin Specification**

Pellet diameter	Annular 5.56/1.6 mm
Pellet density	96 % TD
Smear density	83 %
Pin length	2580 mm

calcination at about  $300^\circ\text{C}$ , which is then reduced to  $\text{UO}_2$  with cracked ammonia at about  $650^\circ\text{C}$ . Since the reduced powder is reactive to oxygen, it is stabilized by partial oxidation to  $\text{UO}_{2.15}$  using  $\text{CO}_2$  or controlled amount of air. The particle size of powder is controlled within 2-10  $\mu\text{m}$  by ball milling and a surface area of  $5\text{m}^2/\text{g}$  is considered optimum.

Several methods are available for preparation of pure  $\text{PuO}_2$  including oxalate or peroxide precipitation or direct denitration. Pu(IV) oxalate precipitation process has been widely employed in the conversion of plutonium nitrate solution to plutonium oxide because of its amenability to continuous operation and relatively good decontamination with respect to impurities. The conversion method based on the precipitation of Pu(III) oxalate and subsequent calcination has also been used. In this method, before precipitation, the plutonium is adjusted to trivalent state, by the addition of 1M ascorbic acid, in the presence of an oxidation inhibitor such as hydrazine. Finally the precipitate is calcined at  $700^\circ\text{C}$  to obtain  $\text{PuO}_2$ .

#### ***Properties of $\text{UO}_2$ and $\text{PuO}_2$***

$\text{UO}_2$  and  $\text{PuO}_2$  are isostructural, completely solid soluble and have similar thermophysical and thermodynamic properties. However  $\text{UO}_2$  is generally hyperstoichiometric and has composition  $\text{UO}_{2+x}$  whereas  $\text{PuO}_2$  is hypostoichiometric with the composition,  $\text{PuO}_{2-x}$ . Such deviations are possible because U and Pu have many valence states. In

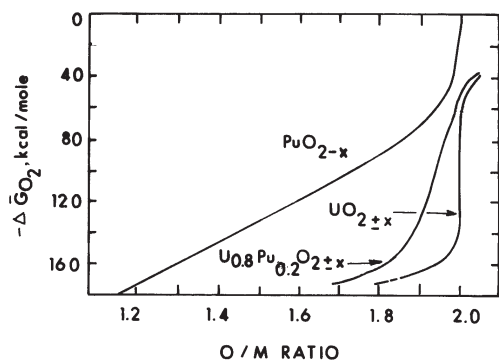


Fig. 1 Calculated oxygen potentials at 2240°C for  $UO_{2\pm x}$ ,  $PuO_{2-x}$  (Fluorier single phase) and  $U_{0.8}Pu_{0.2}O_{2\pm x}$

uranium, the  $U^{4+}$ ,  $U^{5+}$  and  $U^{6+}$  states tend to be most stable, while in plutonium the  $Pu^{3+}$  and  $Pu^{4+}$  states occur most frequently. In the stoichiometric oxides,  $UO_{2.00}$  and  $PuO_{2.00}$ , the heavy metal ions carry a charge of  $4^+$ . Electrical neutrality in the crystal, when the oxygen ions are removed from or added to exactly stoichiometric material requires that some of the cations change valence. Thus, the uranium ions in  $UO_{2+x}$  are a mixture of  $U^{4+}$  and  $U^{5+}$  (or possibly  $U^{4+}$  and  $U^{6+}$ ) and the plutonium ions in  $PuO_{2-x}$  are a mixture of  $Pu^{3+}$  and  $Pu^{4+}$ . The excess oxygen in  $UO_{2+x}$  is accommodated in interstitial sites in the fluorite structure. The deficiency of oxygen in  $PuO_{2-x}$  manifests as vacancies on the oxygen sublattice. The mixed uranium-plutonium oxide can exist as hypostoichiometric oxide,  $(U,Pu)O_{2-x}$  with some Pu in  $3^+$  state or hyperstoichiometric oxide,  $(U,Pu)O_{2+x}$  with some uranium in  $5^+$  or  $6^+$  state.

#### Oxygen Potentials

An important concept in understanding the fabrication and irradiation studies on  $(U,Pu)O_2$  is the oxygen potential. Each uranium-plutonium oxide of the general formula  $(U_{1-y}Pu_y)O_{2\pm x}$  at a particular temperature  $T$  is characterized by a definite partial pressure of gaseous oxygen which is in thermodynamic equilibrium with the solid. The chemical potential of oxygen is given by:

$$\Delta G_{O_2} = RT \ln p_{O_2}$$

$\Delta G_{O_2}$  is another way of expressing the equilibrium oxygen pressure over the material.

Fig.1 shows oxygen potential as a function of oxygen-to-metal ratio (O/M) for uranium oxide, plutonium oxide and a mixed uranium-plutonium oxide. The most significant feature is the very abrupt change in  $\Delta G_{O_2}$  near exact stoichiometry. This feature has profound influence on the chemical behaviour of fuel elements in a reactor. It has been recommended that the (U,Pu) oxide fuel for fast reactors should be slightly hypostoichiometric. However critical control of oxygen potential is necessary to obtain specific O/M values below 2.00. Figure 2 shows the relationship between  $H_2/H_2O$ , oxygen potential, temperature and O/M and at a sintering temperature of 1700°C,  $H_2O/H_2$  ratio of  $10^{-3}$  is required to obtain an O/M of 1.96-1.97 and very special care is necessary to maintain this value during cooling. The kinetics of reduction of O/M can be very slow requiring long sintering times.

#### Advanced LMFBF Fuels

Mixed carbide and mixed nitride fuels have been recognized as advanced LMFBF fuels on the basis of their high thermal conductivity, high heavy atom density and excellent compatibility with sodium coolant. The fabrication of (U,Pu)C and (U,Pu)N is difficult compared to mixed oxide because these fuels are highly susceptible to oxidation and hydrolysis and pyrophoric in some cases, necessitating high purity inert cover gas in glove boxes. Also, the number of process steps are more than double. Stringent control of oxygen content and carbon or nitrogen stoichiometries are needed to avoid metallic second phase and keep the higher carbides or nitrides within acceptable limits.

#### Uranium-Plutonium Carbide

The uranium plutonium mixed carbide fuel is specified as a monocarbide with a small admixture of  $M_2C_3$ . This fuel is fabricated by the carbothermic reduction of a mixture of the oxides of uranium and plutonium. If the pseudo ternary section of the phase diagram (Fig. 3) of the U-Pu-C-O system is considered, then carbothermic reduction starts at a point on the  $MO_2$ -C binary line and proceeds along the path indicated by the dotted line through the removal of CO gas. The  $MC_{1-y}O_y$ - $M_2C_3$  phase field becomes important and the oxygen content of final product depends on the tie line that is reached through CO removal. The gas phase above this phase





spheres can be pressed in to pellets. The sol-gel method of preparation of carbides and nitrides has several advantages over the conventional pellet route, as it uses minimum number of steps with carbothermic reduction leading to dense particles and no milling or grinding of reactive and pyrophoric powder is required. The carbothermic reduction temperatures for the synthesis of MC and MN are lower because of the higher chemical activity and specific surface area of the gel microspheres which favour the kinetics of carbothermic reduction. Excellent micro homogeneity is ensured since heavy metals are mixed in the form of nitrate solution. Carbon is added to U/Pu nitrate solution along with gelation agents to get particles, which have uniformly dispersed carbon in oxide matrix. Carbothermic reduction of gel particles directly yields microspheres suitable either for vibrocompaction or gel pelletisation. For the nitrides, carbothermic reduction is carried out in nitrogen atmosphere instead of vacuum.

### **In-pile Chemical behaviour of Fast Reactor Fuels**

#### ***Oxide fuels***

The steep thermal gradient and consequent oxygen potential gradient, which are set up in the fuel during irradiation, have a profound influence on the fuel chemistry. The oxygen potential is a measure of the driving force for the numerous reactions that take place in the fuel element during irradiation. This potential provides the driving force for attack of the stainless steel cladding, controls the chemical state of many fission products, their interaction with the fuel, their contribution to swelling, their volatility and their redistribution. It controls the vapour pressure of many fuel components, affects the potential for the interaction of the liquid sodium coolant with the fuel matrix in the event of minor breaches in the cladding and affects the intrinsic thermal conductivity of the fuel. In fact, oxygen stoichiometry has a much greater influence on thermal conductivity of U-Pu oxide than does the plutonium content.

There are two essential differences between urania and uranium-plutonium oxide, which involve the effects of oxygen concentration on fuel behaviour. For a given O/M ratio and Pu concentration the U-Pu oxide has a much higher

oxygen equilibrium vapour pressure or more positive oxygen potential (i.e. plutonia is more oxidizing than urania). The second difference is that in the case of  $\text{UO}_{2+x}$ , there is a very large change in oxygen pressure with composition near the stoichiometric composition. For U-Pu oxide the oxygen pressure or oxygen-potential curve is flatter (Fig. 1). More cladding oxidation by U-Pu oxide is likely to occur for a given oxygen potential change because a larger change in oxygen concentration is required than in the case for  $\text{UO}_2$ .

#### ***In-pile Behaviour of Oxide Fuel***

For a fuel in which some fission has occurred, O/M ratio may be defined as the ratio of total number of moles of oxygen to the total number of moles of uranium, plutonium and fission product elements in solid solution in the fluorite phase. Fission product elements can be classified in to two main classes, namely, those which are oxidized (like Sr, Y, Zr, Nb, Ba, and rare earths) and those, which are not (like Kr, Xe, Tc, Ru, Rh, Pd and Ag). The chemical states of Mo, Cs and Rb are important to the oxygen balance in the fuel and the extent to which these elements are oxidized depends on the oxygen potential of the system. The effect of burnup on the O/M ratio depends on the yields of fission products. There is a major shift in mass yields for  $^{239}\text{Pu}$  as compared to  $^{233}\text{U}$  or  $^{235}\text{U}$  for both fast and thermal neutron energies. In both the cases a significant drop occurs in the zirconium isotopes with a corresponding increase in noble metal fission products. As a result when  $^{239}\text{Pu}$  is the fissioning fuel, significant increases in the quantity of available oxygen are to be expected with the increase in burnup.

The mixed oxide fuel is typically taken to a burnup of 12%h.a, but fuels are being developed that can attain 20%h.a burnup. The main areas of concern, which have implications on achievable burnup, are the internal corrosion of the clad at high burnups, fuel-coolant chemical interactions and the reactions of solid fission products. O/M ratio increases with burnup. On the basis of post-irradiation examination of the fuel, the fission products in the mixed oxide fuel can be classified as (a) volatile fission product (F.P) elements (Kr, Xe, Rb, Cs, Br, I, Se and Te), (b) F.Ps forming metallic phases (Mo, Tc, Ru, Rh, Rb, Pd, Ag, Cd, In, Sn and Sb), (c) F.Ps forming oxide phases (Rb, Cs, Sr, Ba,

Zr, Nb and Mo), and (d) F.Ps which dissolve in the fuel matrix (Sr, Zr, Nb and rare earths). Mo is present in the alloy phase in the inner region of high temperature and low oxygen potential, but is a constituent of the oxide phase in the outer regions. Post-irradiation examinations have shown the presence of 'white' metallic incursions containing Mo, Tc, Ru, Rh, and Pd and a 'grey' ceramic phase containing mainly the oxides of barium, molybdenum, uranium, plutonium and zirconium. This can be described as  $[Ba_{1-x-y}Sr_xCs_y][U,Pu,Ln,Zr,Mo]O_3$  and it is a eutectic oxide with a lower melting point than the fuel. The intergranular attack on the clad inner surface could be due to Te, which may also be tied up with Cs as  $Cs_2Te$  depending on the oxygen potential.

The principal parameters initiating and controlling the cladding attack are the oxidizing potential, the amount of available oxygen and the temperature of the inner cladding wall. Selected fission products (e.g., Cs and Mo) also contribute to corrosion. Their transport to the cladding wall within the fuel is controlled by the temperature and local oxygen potentials. The fission products Cs, Mo, Te and I which populate the fuel-cladding gap are significant factors in influencing the degree and type of cladding attack.

Uranium and plutonium have been observed to be redistributed within the irradiated mixed oxide fuel with the plutonium concentration being enhanced in the center of the fuel for hyperstoichiometric oxide fuels.

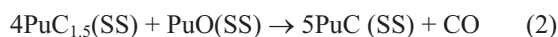
The chemical interaction of the sodium coolant with the U-Pu oxide fuel in the event of any breach in the stainless-steel cladding can cause fuel swelling and cladding rupture. The reaction of sodium and fuel results in three phases, sodium, fuel and  $Na_3MO_4$  ( $M=U,Pu$ ) in local equilibrium. The most reliable method for controlling the sodium-fuel reaction is to limit the oxygen both in the fuel and in the sodium to a level where the reaction product will not be stable.

### Carbides

Because of the higher thermal conductivity of uranium-plutonium carbides, the temperatures attained in the irradiation of carbide fuels are much

lower than those for oxide irradiations under similar power ratings. Any restructuring will be much less than an oxide; grain growth and fission product diffusion will be slow and thus the growth of fission product phases will also be slow and consequently observations on the fission product phases within the carbide fuel are limited.

As the fabrication of FBTR fuel via carbothermic reduction route results in a much higher oxygen and nitrogen level than is specified for normal FBR fuels (with much lower plutonium content), the parameters of interest are carbon potential and the partial pressure of CO, both of which are important in clad carburisation, the latter phase being involved in gas phase carburisation mechanism. Presence of oxygen and nitrogen impurities in the fuel can lead to generation of CO as per the following reaction:



Many of the fission products like Sr, Ba, Mo, R.Es, Rh, Ru, and Tc form carbides. However, there is an overall decrease in C/M ratio as a function of burnup and clad carburisation should decrease theoretically. But experimentally it has been seen that clad carburisation increases with burnup. This has been attributed to the role of oxygen which modifies the chemistry. Migration of the fission products from the center of the pellet to the surface is not very pronounced and is not responsible for any fuel-clad chemical interaction. The carbides can react with stainless steel cladding to form intermetallic compounds with iron and nickel like  $(UPu)(Fe,Ni)_2$  and  $(U,Pu)(Ni,Fe)_5$ , which form low melting eutectics with stainless steel. Thermodynamically this reaction can be initiated at 700°C but kinetically it has been observed only at temperatures above 900°C. At temperatures above 1100°C the reaction rate is fast and the cladding is extensively damaged.

Sodium is compatible with mixed carbide fuel and in fact sodium-bonded fuel is one of the concepts under investigation. The use of sodium bonding, however, requires a better control of the stoichiometry of the fuel due to enhanced rate at which the carbon can be transferred from the fuel to the clad.



If it is assumed that all the Ru, Rh and Pd are present in the form of  $(U,Pu)(Ru,Rh)_3$  and  $(UPu)Pd_3$  type compounds then for monocarbide the carbon potential will be buffered by the  $(Ba,Sr)C_2$  formation.

### ***Nitride Fuel***

A careful control of stoichiometry as well as oxygen and carbon impurities is necessary in the case of nitride fuels also. The presence of sesquinitride phase can lead to pressure build up within the fuel pin. If carbon and oxygen impurities are above 1000 ppm each, there is a likelihood of clad carburisation via carbon monoxide. Nitriding of clad is another aspect needing careful evaluation. The irradiation experience with nitride fuel is very limited; bulk of the experience is with high purity nitride fuel produced by using uranium and plutonium metals. The accumulation of fission products in the fuel is expected to lead to an increase in N/M ratio of the fuel but the effects of this increase are yet to be evaluated. However nitride is currently being considered in view of its lower reactivity with oxygen and moisture and also ready dissolution for reprocessing in comparison to carbide fuels.

### ***In-pile Behaviour of Carbides and Nitrides***

The major factors that dominate the behaviour of MX-type of fuels are the chemical state of the fission products, swelling and gas release and compatibility with clad. These fuels are denser than the oxide and have cubic close-packed structures. Hence they have greater difficulty in accommodating fission products, as a result of which burnup effects are expected to be more significant. Most of the non-volatile fission products form mono, sesqui- or dicarbides. Some of these are partially soluble in the fuel matrix while Zr is fully soluble. Table 2 gives the chemical state of fission products in the oxide, monocarbide and mononitride fuels.

### ***In-pile Behaviour of Metallic Fuels***

The risk inherent in the use of metallic fuels is the formation of low melting eutectics by reaction between fuel and clad. For reasons of safety it is essential to keep the interface temperature substantially lower than the eutectic temperature. Out of pile tests with U-15%Pu-10%Zr alloy indicated that the reaction does not take place below

700°C. However irradiation experiments with this alloy resulted in fuel failure at interface temperatures below 630°C. Some studies have indicated that a soft Zr layer containing up to 20% nitrogen formed near the clad surface can act as a barrier for this reaction at temperatures below 700°C. To overcome the problem of fuel-clad compatibility, the use of other clad materials like Nb, V and Mo can be considered. These materials are highly susceptible to loss of strength due to pickup of non-metallic impurities, particularly oxygen, from sodium.

The behaviour of FPs falls in to three categories. Those FPs which dissolve in sodium end up in the sodium bond above the fuel column and sodium-filled pores. Most metallic elements either dissolve in the fuel alloy or form precipitates. Lanthanides migrate to the periphery of the fuel pin where they precipitate as a separate phase in existing pores. Lanthanides also diffuse into the clad, generating a narrow embrittled layer. Lanthanide diffusion produces an interaction layer, which has been found to extend to 0.1mm in to the clad at 10%h.a burnup. This brittle layer containing nearly 20% lanthanides does not contribute to clad strength and is one of the major concerns in fuel-clad chemical interaction. The diffusion of cladding alloying elements (Fe and Ni) in to the body of the fuel has also been noticed. This can lower the fuel melting point.

### ***Conclusions***

There are many areas in fast reactor fuel technology where chemistry plays a crucial role like fuel manufacture, irradiation behaviour and reactor safety. Among the four types of fuels discussed, the mixed oxide fuel is well developed. A better understanding of the mixed fuel can lead to development of advanced oxide fuel with higher breeding potential. The chemistry of mixed carbide and mixed nitride fuel is yet to be understood completely and irradiation experiments with well-qualified fuel would go a long way in developing the potential of these fuels with higher breeding ratio. Metallic fuels have recently entered the scene of fast reactors and need a careful assessment in view of their superior potential for breeding.

**TABLE 2. The chemical State of fission products in irradiated fuel in fast breeder reactors**

Fission product elements	Likely chemical state in oxide fuels	Likely chemical state in carbide fuels	Likely chemical state in nitride fuels
Kr, Xe	Elemental state	Elemental state	Elemental state
Y, La-Eu	Oxides which dissolve in host matrix	Dissolved in monocarbide or sesquicarbide lattices	Dissolved in mononitride matrix
Ba, Sr	Oxides which can dissolve to a limited extent in fuel and also form separate phases	$Ba_{1-x}Sr_xC_2$	$(Ba_{1-x}Sr_x)_3N_2$
Br, I	$(Cs_{1-x}Rb_x)Br_{1-y}I_y$	$(Cs_{1-x}Rb_x)(Br_{1-y}I_y)$	$(Cs_{1-x}Rb_x)(Br_{1-y}I_y)$
Rb, Cs	$(Cs_{1-x}Rb_x)Br_{1-y}I_y$ and compounds analogous to $Cs_4UO_4$ , $Cs_2U_2O_7$	$Cs_{1-x}Rb_x$ , $(Cs_{1-x}Rb_x)(Sr_{1-y}I_y)$ $(Cs_{1-x}Rb_x)_2Te$	$(Cs_{1-x}Rb_x)(Br_{1-y}I_y)$ $(Cs_{1-x}Rb_x)_2Te$
Se, Te	$(Cs_{1-x}Rb_x)_2Sr_{1-y}Te_y$	$U_2TeC_2$ , $(Cs_{1-x}Rb_x)_2Te$	$(Cs_{1-x}Rb_x)_2Te$
Zr, Nb	Some dissolution in host matrix	Dissolved in monocarbide lattice	Dissolved in mononitride matrix
Mo, Tc,	Usually single phase alloy. Mo may oxidize to $MoO_2$ and form compounds	$(UPu)Mo_{1-x}Tc_xC_2$ Some Mo dissolved in monocarbide lattice	Elemental
Ru, Rh, Pd	Single phase alloy	$(UPu)_2(Ru_{1-x}Rh_x)C_2$ $(UPu)(Ru,Rh,Pd)_3C$	$(UPu)(Ru,Rh,Pd)_3$
Ag, Cd, In, Sn, Sb	Metallic alloys	Alloy phases	Alloy phases

#### Acknowledgement

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# Fast Reactor Fuel Fabrication



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## Introduction

The primary purpose of a fast breeder reactor is to breed fissile  $^{239}\text{Pu}$  from fertile  $^{238}\text{U}$  while generating power and thus increase the fissile resources required for future nuclear power reactors [1]. Because of the requirement of fast neutron spectrum for breeding, fissile enrichment required in a fast reactor fuel is high and a typical commercial fast reactor fuel contains 15-25% plutonium with depleted or natural uranium. However, some test reactors like RAPSODIE, FBTR etc have used higher enrichment ( $^{235}\text{U}$  and  $^{239}\text{Pu}$ ) to reduce the core size.

The cladding and other in-core structural materials should have high temperature capability. In order to reduce the fuel cycle cost, the fast reactor fuel should have high burn up capability (10%h.a to 20%h.a) and the cladding/ wrapper materials etc. of fuel element/ subassembly should be resistant to irradiation induced void swelling, creep and

embrittlement. The core of a fast reactor contains a number of fuel-subassemblies (each subassembly contains no. of fuel elements assembled in a wrapper tube) and is surrounded by blanket region (fertile material).

## Fuel

A fast breeder reactor fuel should have potential for high breeding ratio and high specific power rating in order to achieve shorter doubling time. High breeding ratio is possible with fuel having high heavy atom density and hard neutron spectrum (light elements like oxygen and carbon moderate neutrons) whereas, high specific power rating is achieved with fuel having high heavy atom density and high thermal conductivity.

Based on the experience gained over a long period of time with oxide fuels in thermal reactors and also in some of the fast reactors (PHENIX, PFR, SUPER PHENIX, MONJU etc.) (U, Pu) $\text{O}_2$  (mixed

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oxide) fuel is among the front runners as fuel for Liquid Metal Fast Breeder Reactor (LMFBR). The advantage of mixed oxide fuel is that fuel fabrication flow sheet is well established, fuel elements have shown high burn up capability [2] (>15% h.a) and the spent fuel is compatible with PUREX process of reprocessing. However, the disadvantage of oxide is its low thermal conductivity, low heavy atom density and thus lower breeding ratio [3] (1.1 – 1.2) and higher doubling time (around 20 yrs). Mixed oxide fuel reacts with sodium to give low density  $\text{Na}_3\text{MO}_4$  (M=U,Pu) phase and to take care of this, the initial O/M has to be kept under control. However, if O/M of MOX fuel is kept low, thermal conductivity of the fuel goes down. Hence, designers have to specify optimum O/M value for a particular fuel composition (plutonium content) for good in-reactor fuel performance. High burn up oxide fuels release more than 80% of fission gases [4], hence fuel elements are designed with sufficient space (plenum) to accommodate fission gases and keeping the internal pressure of fuel elements within control.

Non-oxide ceramic fuels like mixed monocarbide (U,Pu)C and mixed mononitride (U,Pu)N are considered advanced fast reactor fuels because of their high thermal conductivity, high heavy atom density and excellent compatibility with liquid sodium. The higher thermal conductivity of these fuels make it possible to operate the fuel at higher linear rating ( $\approx 1000 \text{ Watt/cm}$  compared to  $\approx 450 \text{ Watt/cm}$  for mixed oxide fuels). These fuels can also give a breeding ratio higher than 1.2. However, the fabrication of MC and MN requires more number of process steps than oxide as these have to be synthesised from oxide powder by carbothermic reduction. MC is a highly pyrophoric material and is susceptible to oxidation and hydrolysis, hence it has to be handled in inert atmosphere like nitrogen or argon. MN is less pyrophoric and is less susceptible to oxidation and hydrolysis than MC. However, one of the reasons for the lower efforts on nitride fuels is requirement for using enriched nitrogen (99.65%  $^{15}\text{N}$ ) as the absorption of neutrons by  $^{14}\text{N}$  is more and  $\text{H}_2$  and He are generated as a result of (n,p) and (n, $\alpha$ ) reactions of  $^{14}\text{N}$ . Additionally reprocessing of carbide and nitride fuels is yet to be established on a commercial scale.

MC and MN fuel pins with different cladding material and bonding material (He or Na) in pellet-clad gap have been satisfactorily tested to atleast 10% h.a burn up [5]. Fission gas release at higher burn ups (>15% h.a) probably exceeds 50%, but always remains lower than oxide. These fuels show high swelling rate and high mechanical interaction with cladding. This is because of their lower plasticity. Operating temperature for these fuels is lower than oxide at similar linear ratings with the result that thermal creep as well as irradiation creep rates are lower. Hence these fuels have to be designed by allowing enough voidage/ pellet-clad gap to accommodate swelling. However, introduction of these voids in the form of porosities/ higher pellet-clad gap reduces the thermal conductivity and hence sodium bond between fuel and clad is necessary to exploit full potential of these fuels.

Table 1 gives comparison of the properties of various fast reactor fuel materials.

#### Cladding and Wrapper Materials

A detailed article on this topic appears elsewhere in this issue. The cladding and wrapper tube material in a fast reactor fuel subassembly are subjected to high operational temperature ( $550^\circ\text{C}$ - $650^\circ\text{C}$ ), high energy neutron fluxes of around  $10^{15}$  neutrons/ $\text{cm}^2/\text{sec}$  and chemical interaction with fission products, fuel species and sodium. The neutron dose on cladding/ wrapper material is around 100-200 dpa. Cladding/ wrapper materials should be resistant to irradiation induced void swelling, embrittlement and creep at the operating temperature. Three types of materials, Austenitic Stainless Steel (SS316, SS316-Ti, 15-15 Ti SS, D-9 SS, etc.), high nickel alloys (PE-16, INC 706, etc.) and ferritic/ martensitic SS (HT9, EM10, EM12, etc.), have been used as cladding/ wrapper material [6] for fast reactors. Alloy D-9 has been the proposed cladding/ wrapper material for the Indian PFBR.

#### Fuel Pellet, Fuel Element and Fuel-Subassemblies

##### Fuel Pellet

The fast reactor fuel should have the capability to withstand high burn ups (>10% h.a), and therefore

**TABLE 1. Properties of Fast Reactor Fuel Materials [4]**

Fuel	(U, Pu)O <sub>2</sub>	(U, Pu)C	(U, Pu)N	U-Pu-Zr Alloy
Density (gm/cm <sup>3</sup> )	11.0	13.6	14.3	15.6
Heavy atom Density (gm/cm <sup>3</sup> )	9.7	12.9	13.5	14.0
Melting Temperature (°C)				
Liquidus	2775	2480	2780	1160
Solidus	2740	2325	2720	
Thermal Conductivity at 1000°C W/m/K	2.9	19.6	19.8	35*
Thermal Expansion (20°C–1000°C)	12.6x10 <sup>-6</sup>	12.4x10 <sup>-6</sup>	10x10 <sup>-6</sup>	16.5x10 <sup>-6</sup> *
Dissolution in HNO <sub>3</sub>	Yes	-	Yes	-
Compatibility with Sodium	Poor	Good	Good	Good

\* At 500°C

in order to accommodate swelling because of fission products, the smear density is kept normally lower than thermal reactor fuels. Typically MOX pellets have a smear density of 80-85% TD. This low smear density can be attained either by going for low density pellets, use of high density pellets with high pellet-clad gap or by going for annular pellets of high matrix density (>95% TD). Annular pellets of high matrix density are considered better fuel than lower density solid pellets or high density high pellet-clad gap solid pellet configuration, as they provide better in-reactor behaviour under transients and the central temperature is lower for a given linear heat rating.

#### **Fuel Element and Fuel-subassemblies**

Fast reactor fuel elements are relatively small in diameter than thermal reactor fuel elements. The fuel pellets are stacked in cladding tubes of external diameter of around 5 - 8.5mm. Above and below the fissile pellet stack are pellets of fertile material as axial blanket and space (plenum) for collection of fission gases. Depending on the design, axial blankets are either integral with the fuel pins or separate fertile material elements are manufactured and assembled in the fuel subassembly. A number of fuel pins are bundled together, and this bundle, along with axial blankets is incorporated in the fuel-subassembly. The fuel pins in a subassembly are held in a regular array and inter element spacings are provided either by spacer grids or wire wrapping. A hexagonal subassembly wrapper envelops the fuel

pin bundle. This wrapper channels the liquid coolant flow over the fuel pin array and provides structural strength and stability to the fuel subassembly. In large fast reactor cores, the core is subdivided in two or more zones with respect to fuel enrichment. The central zone contains fuel-subassemblies with lower enrichment whereas, outer zone contains fuel-subassemblies with high enrichment. This arrangement gives almost similar power output from all the fuel-subassemblies.

Table 2 gives details of fuel elements for some of the LMFBRs.

#### **Fuel Fabrication**

Fuel element/ subassembly fabrication is one of the important components of nuclear fuel cycle and the manufacturing process depends on type of fuel (Ceramic/ Metallic), radioactivity and radiotoxicity of fuel materials, reactivity of the fuel materials with ambient and fuel/ fuel element specifications. Since commercial LMFBR Fuel invariably has high plutonium content, special efforts are made to take care of radiological hazards associated with handling of plutonium. Plutonium is primarily an alpha active material having high specific activity and high biological half life, hence this material has to be handled in leak tight glove boxes and the operating area of the plant should have sufficient number of air changes to control the level of plutonium in air. Aged plutonium contains significant concentration of <sup>241</sup>Am which emits 60



**TABLE 2. Fuel Element/ Fuel Assembly Details of LMFBRs**

Reactor	Fuel / PuO <sub>2</sub> content	Pellet shape (Cylindrical)	Pellet dia (mm)	Clad O/D (mm)	Clad Material	Fissile length (mm)	Integral axial blanket# (mm)	Pin length (mm)	Pins / subassembly
RAPSODIE	(U,Pu)O <sub>2</sub> 30%	Solid	4.23	5.1	SS 316	-	T-Nil B-Nil	-	61
FBTR*	(U,Pu)C	Solid	4.18	5.1	SS 316	320	T-Nil B-Nil	531	61
PFR	(U,Pu)C	Annular	-	5.8/6.6	PE-16	914	T-102 B-458	2300	265 / 365
PHENIX	(U,Pu)O <sub>2</sub> 18 & 25%	Solid	5.4	6.5	15-15 Ti SS	850	T-Nil B-300	1800	217
SUPER PHENIX	(U,Pu)O <sub>2</sub> 14 & 17%	Annular	7.5	8.5	15-15 Ti SS	1000	T-300 B-300	2700	271
MONJU	(U,Pu)O <sub>2</sub> 15 & 20%	Solid	5.45	6.5	PNC SS316	930	T-300 B-350	2800	169
PFBR	(U,Pu)O <sub>2</sub> 21 & 28%	Annular	5.56	6.6	D-9 SS	1000	T-300 B-300	2600	217

\*Mark-I core, (U<sub>0.3</sub>Pu<sub>0.7</sub>)C  
Mark-II core, (U<sub>0.45</sub>Pu<sub>0.55</sub>)C

#T-Top side of fissile column  
B-Bottom side of fissile column

keV gamma rays. Some of the isotopes of plutonium also emit neutrons due to spontaneous fission. Additionally, neutrons are emitted by ( $\alpha$ ,n) reaction with light elements. Hence, fast reactor fuel fabrication will require shielding arrangements to reduce neutrons and gamma radiation. The equipment and process used should be amenable to remotisation/ automation to reduce operator dose and to get high throughput and should be designed to avoid criticality through engineered design safety features, geometry control and inventory control backed up by administrative control.

#### **Mixed Oxide Fuel Pellet Fabrication**

Mixed oxide fuel pellets for fast reactor are normally prepared by using powder metallurgy techniques starting with UO<sub>2</sub> and PuO<sub>2</sub> powder as feed materials. Fig. 1 gives a typical flow sheet for fast reactor mixed oxide fuel fabrication.

Some of the important mixed oxide fuel specifications are composition (U, Pu content), homogeneity, dimension, density, O/M ratio, metallic/ non-metallic impurities, the gas content of

the pellet, etc. In-homogeneity of plutonium distribution in the fuel affects in- reactor fuel performance (plutonium rich agglomerates leads to hot spots) and dissolution behaviour of the spent fuel during reprocessing. O/M ratio influences the fuel-clad and fuel-coolant compatibility, thermal conductivity, plasticity and U/Pu redistribution of the fuel. The gas content of the pellets is important because high amount of gas release during irradiation may lead to fuel element pressurisation. High amount of impurities may lead to fuel dilution as well as deterioration in chemical/ metallurgical properties of the fuel.

Mixing and milling of UO<sub>2</sub> and PuO<sub>2</sub> powder is a very important process step as it controls the degree of homogeneity of plutonium distribution. Milling machines like centrifugal ball mills, bead mills and attritors, have been used by various fuel fabricators. Attritors have been found very efficient and impurity pick up from vessel wall is also found negligible. BNFL, UK and AFFF, BARC have used this machine in mixed oxide fuel fabrication [7] line. Use of co-precipitated [2,8] (U,Pu)O<sub>2</sub> powder gives a

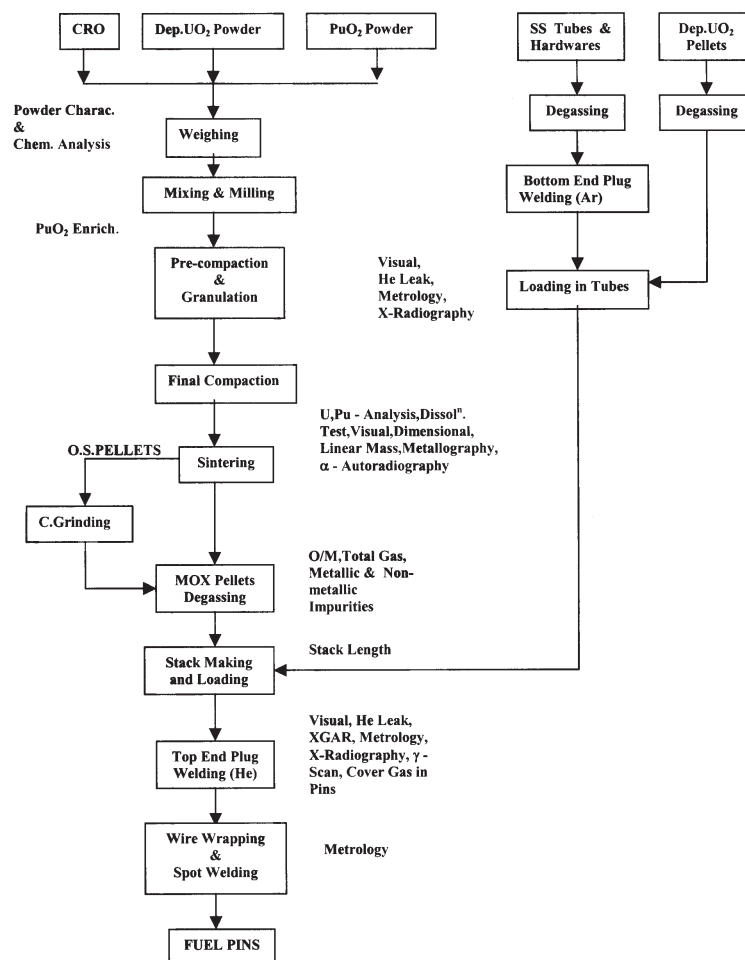


Fig. 1 PFBR fuel pin fabrication flowsheet with QC steps (proposed)

higher degree of homogeneity as uranium and plutonium are mixed in liquid state. High homogeneity fuel can also be obtained by using soft (U,Pu)O<sub>2</sub> microspheres produced by gelation route, as feed materials followed by direct compaction. This technique is known as SGMP [9] (Sol-Gel Microsphere Pelletisation). This process gives less dust problems as fine powder is not handled during fuel fabrication. Process steps like mixing/ milling, pre-compaction and granulation, are not required in SGMP technique, as feed material, (U,Pu)O<sub>2</sub> microspheres are free flowing. This process is more amenable to remotisation and automation.

Granulation helps to improve flowability of the milled UO<sub>2</sub> and PuO<sub>2</sub> powders to facilitate automatic

compaction. Granulation is done either by pre-compaction of the powder at low pressure followed by breaking them in granules by using planetary / oscillatory granulator or by mixing the powder with sufficient quantity of binder in a Z-blade mixer [8] and forming paste by kneading, drying the paste into crumbly dough and then granulation by using screen granulator. Another technique under development at AFFF, BARC, Tarapur is Spherodisation/ Extrusion [10] technique, wherein the powder mixture along with the requisite amount of liquid binder is first extruded and then spherodised followed by drying.

Sintered density of fuel pellets depends on green density of compacts. Most of the pellet defects

e.g. lamination, end capping, end chipping, dimensional deviation, cracks, etc., have their origin in wrong compaction practices. The number of pellets per kg. of fuel in a fast reactor is much higher than that for thermal reactors. Hence, the compaction machine should have high throughput. A mechanical press gives high throughput but requires granules in narrow size range to get pellets of reproducible density. Hydraulic press can give reproducible density compacts but throughput is lower. Rotary compaction machine gives high productivity with good quality granules and has been used by fuel fabricators in Belgium [11]. AFFF, BARC has also inducted a rotary compaction press for fabrication of fuel pellets [10].

Sintering of mixed oxide pellets is done in a molybdenum resistance furnace under controlled reducing atmosphere ( $\text{Ar}/\text{H}_2$  or  $\text{N}_2/\text{H}_2$ ) at around  $1650^\circ\text{C}$ . The batch type sintering furnace is preferred over continuous sintering furnace because of lower oxygen potential inside the furnace due to which achieving hypo-stoichiometry is easier. Further, batch furnace requires less logistical support e.g. cover gas, cooling water, power and manpower. It is easy to maintain and more amenable to mass control from criticality consideration.

The purpose of the sintering step is to get acceptable density, dimension, homogeneity of plutonium distribution and slight hypo-stoichiometry ( $\text{O}/\text{M} < 2.00$ ). Use of optimum die size, pressing parameters and sintering parameters produce most of the MOX pellets of diameter well within the diametral tolerance ( $\pm 0.04$  mm). The oversize pellets if any, are ground to dimension by dry centreless grinding method. Dry grinding is done to take care of criticality hazards and to avoid increase in O/M during grinding.

Pellet density is influenced by powder characteristics, milling parameters, thermal treatment of powder [11,12,13], use of pore formers, etc., apart from green pellet density and sintering parameters like temperature, time, gas flow rate,  $\text{H}_2/\text{H}_2\text{O}$  ratio in the gas.

Some fuel fabricators are of the opinion that to attain significant and reproducible hypo-stoichiometry with  $\text{UO}_2\text{-PuO}_2$  fuel on industrial scale is not easy [11]. The O/M of the fuel can be

controlled by addition of carbonaceous substance like Sterotex or Carbowax in the powder mix [14] and post sintering thermal treatments, but on industrial scale this has not been demonstrated. Hypo-stoichiometric fuel pellets if not stored in high purity inert atmosphere/ vacuum, pick up oxygen and thus O/M gets increased [14]. The off gas content of the sintered pellets is kept within the specifications by degassing the pellets in vacuum or inert gas like argon and storing the degassed pellets in controlled environment.

#### *Scrap Recycling*

Two types of scraps are generated during fuel fabrication namely, Clean Rejected Oxide (CRO) and Dirty Rejected Oxide (DRO). CRO scraps are materials which are rejected due to physical defects, whereas DRO scraps are fuel materials which are having chemical impurities. CRO is mostly recycled by dry route by converting it to powder by process steps consisting of crushing/ milling, oxidation and reduction and then adding this powder with virgin powder to make fuel pellets. Another promising route is microwave dissolution and de-nitration technique [15] which can give powder that can be either used directly or mixed with virgin powder in higher proportion than dry route. DRO is recycled by wet route consisting of dissolution, precipitation, calcination and reduction. The wet route has the disadvantage of generation of liquid waste; however, the wet route has to be adopted for recycling of DRO to get pure feed material.

#### *Fuel Pin Welding*

The cladding material mostly used in fast reactor fuel is Austenitic-Stainless Steel (SS316 M for FBTR and D-9 SS for PFBR). End cap welding is done by TIG welding process in argon and helium for bottom and top end plug respectively. Accepted fuel pellets are stacked and loaded (along with insulation pellets or blanket pellets) with plenum, spring/ spring support in bottom end plug welded tube. Top end plug welding is done in helium as helium is used as bonding gas in the pellet-clad gap region. Sometimes Pulse-TIG welding (low heat input) is also used instead of continuous TIG welding to eliminate weld defects [16] e.g. root pocket, lack of penetration, etc. Advanced techniques like microplasma welding are under

evaluation at AFFF, BARC for end cap welding because of independence of weld penetration with arc gap.

The pins are cleaned and decontaminated. Spacer wire is wrapped around the pin in a wire wrapping machine by fixing the beaded end of the wire in the bottom plug groove and spot welding the wire-end at the top plug.

#### ***Mixed Carbide/ Mixed Nitride Fuel Fabrication [17]***

The feed materials for fabrication of MC pellets are  $\text{UO}_2$  powder,  $\text{PuO}_2$  powder and Graphite powder. These powders are weighed in proportion and milled in a ball mill for sufficient time to get homogeneous powder mixture. Graphite is added slightly less than stoichiometric amount to keep the  $\text{M}_2\text{C}_3$  phase within specified limits. A certain percentage of  $\text{M}_2\text{C}_3$  is always specified in carbide fuel to ensure absence of metallic uranium and plutonium phase. The milled powder mixture is pre-compacted at low pressure and then synthesized by carbothermic reduction in a vacuum furnace at around  $1500^\circ\text{C}$ . Clinkers of MC are then crushed and ground to sinterable grade powder and mixed with binder/ lubricant to facilitate compaction. The powder mixture is pre-compacted/ granulated and then compacted to green pellets. The compacts are sintered in  $\text{Ar} + \text{H}_2$  mixture. Fuel pins are then fabricated as in the case of oxide. Fig.2 gives the flow sheet used for FBTR fuel fabrication at RMD, BARC.

MN fuel is also prepared almost in the same way as that of MC fuel with carbothermic reduction of  $\text{UO}_2$ ,  $\text{PuO}_2$  and graphite powder in nitrogen atmosphere. All other process steps remain same as that of MC fuel/ fuel element fabrication.

Another alternative method for MC and MN fuel pellet fabrication is SGMP route, where gel microspheres containing uranium, plutonium and carbon are synthesised to soft MC and MN microspheres followed by compaction and sintering.

#### ***VIPAC Fuel***

Though pellet-in-tube type of fuel pins are the most commonly used fuel in nuclear power reactors because of their better defect behaviour, a lot of

development work has been done to fabricate and irradiate vibro-compacted fuel elements in fast reactors [8]. The advantage of VIPAC fuel is that fuel does not have to meet stringent fuel pellet specifications like dimension and physical defects. In VIPAC fuel pin fabrication, high density particles (2 – 3 size ranges) of fuel material are filled in clad tube and vibro-compacted to get smear density of  $\approx 80\%$  TD (similar to pellet fuel). Both dry and wet route can be used to get high density MOX fuel particles.

#### ***Dry Route***

Two dry routes which have been studied are (a) sintered pellets made by conventional powder metallurgy route are crushed and fuel particles are then classified and (b) Granules are prepared from green powder mix, then classified in different size ranges and sintered to get high density particles.

#### ***Wet Route***

Microspheres of  $(\text{U,Pu})\text{O}_2$  in different size ranges are produced by external or internal gelation technique from uranium – plutonium nitrate solution, followed by calcination and sintering in  $\text{Ar}/\text{H}_2$  or  $\text{N}_2/\text{H}_2$  mixture to get sintered microspheres of high density ( $>95\%$  TD). The advantage of wet route compared to powder based dry route is reduction of operator dose and amenability of the process to automation. Sol-gel process is reported [18] to have been used to manufacture MC and MN fuel particles and has been vibro-compacted to manufacture fuel pins.

In some reactors viz. PFR, VIPAC (vibro-compacted) fuel of 78-83% smear density has been test- irradiated, but their performance was not satisfactory. The reason was segregation of fine particles, higher central temperature [8].

#### ***Desired features in a Fast Reactor Fuel Fabrication Plant***

A fast reactor fuel fabrication plant should aim to be a “Safe and Secure Automated Fabrication Facility”, as the plant handles large quantity of plutonium. Plutonium is a strategic nuclear material (SNM) and hence access to the plant and SNM should be under strict control. The plant, process flowsheet and equipments should be designed/

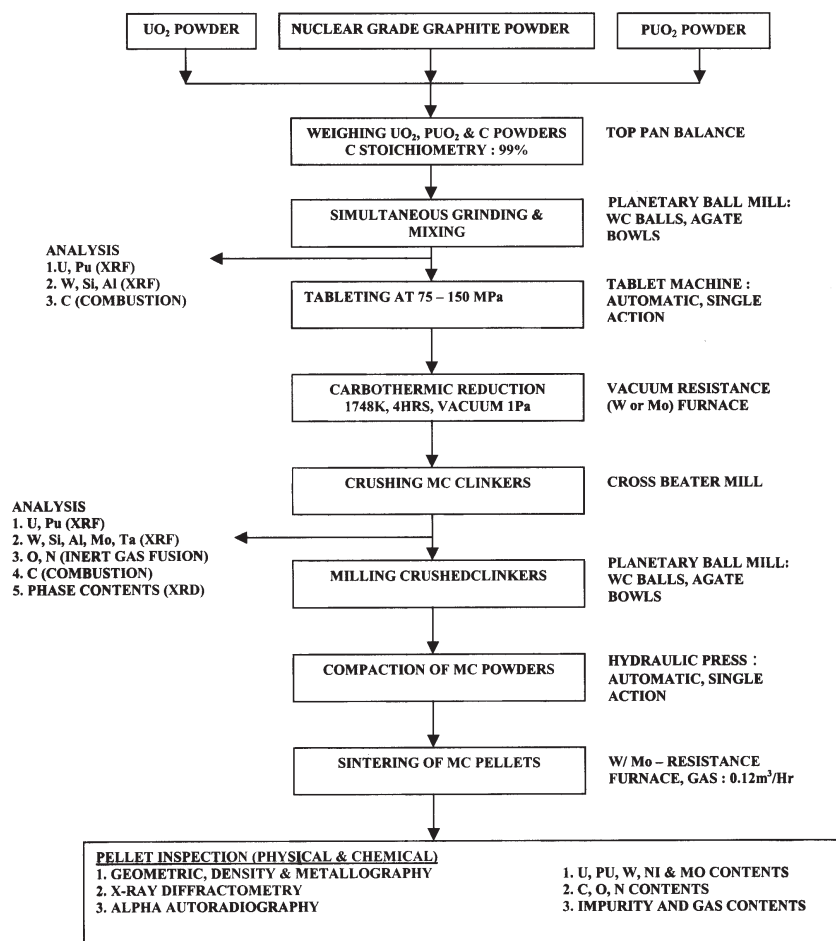


Fig. 2 Mixed carbide fuel pellet fabrication flow sheet with QC steps

selected for safe operation both from criticality safety and personnel exposure. Criticality safety can be attained by engineered features, geometry control, mass control, avoiding use of moderating materials in the process as far as possible and use of neutron absorbing materials where ever feasible. Personnel exposure can be reduced by adopting a process which generates less radioactive dust, has less number of process steps, is amenable to remotisation and automation and uses robust equipment requiring less maintenance.

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# Structural Materials for Fast Breeder Reactors



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## Introduction

Fast Breeder Reactor (FBR) components operate under a hostile and demanding environment of high neutron flux ( $\sim 10^{15}$  n.cm<sup>-2</sup>s<sup>-1</sup>, about ten times higher than in thermal reactors), liquid sodium coolant and elevated temperatures. Hence, FBR technology poses new challenges in the area of materials development. Efficient, economic and safe production of power necessitates optimised choice and design, fabrication and evaluation of structural materials. This paper discusses the materials selected for the main components of the proposed 500 MWe sodium cooled Prototype Fast Breeder Reactor (PFBR). The major factors considered in the

selection of materials include operating conditions, availability of design data in nuclear codes, ease of fabrication, international experience and cost as also experience with the currently operating Fast Breeder Test Reactor (FBTR) at Kalpakkam. An attempt has been made to minimise the number of materials and welding consumables in order to avoid mix up of materials during fabrication and to reduce the cost of Research and Development on materials development and characterisation. Stringent specifications have been drawn up for PFBR materials with respect to chemical compositions and other mechanical properties with a view to improve the reliability of components. Figure 1 gives a

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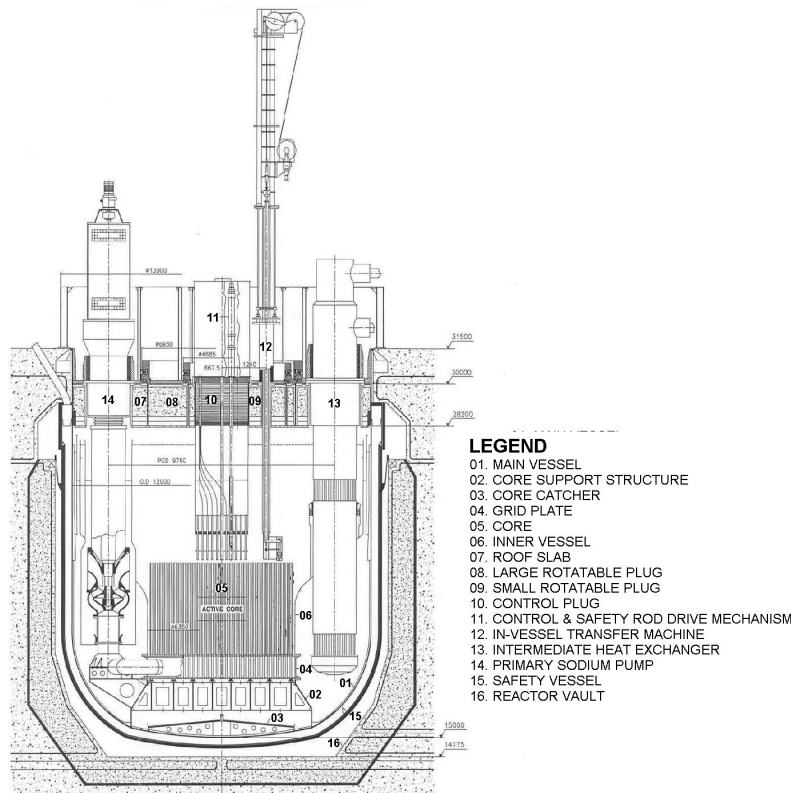


Fig. 1 Schematic of PFBR reactor assembly

schematic view of the reactor assembly. Table 1 gives the PFBR characteristics and operating conditions.

### Considerations for Materials Selection

**Core Structures :** The Fuel subassembly (Fig.2) encounters unique materials problems like void swelling, irradiation creep and irradiation/helium embrittlement. Hence clad and wrapper materials should have, in addition to compatibility with sodium, corrosion resistance, adequate end-of-life creep strength and ductility, and good radiation damage resistance. The fuel burn-up achievable is limited by the void swelling of the wrapper material.

**Out-of-Core and Sodium Circuit Components** must possess adequate high temperature mechanical properties such as creep, low cycle fatigue (LCF) and creep-fatigue interaction, and should be compatible with liquid sodium coolant.

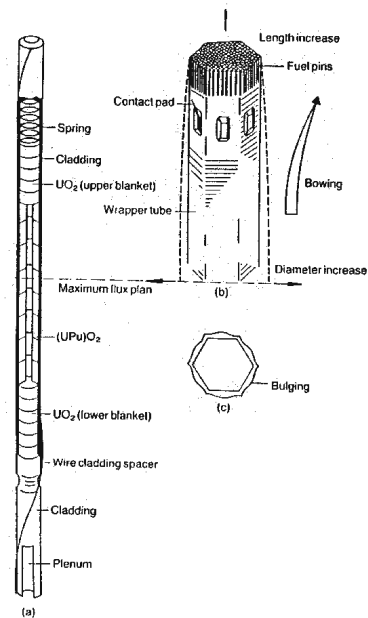


Fig. 2 Schematic of fuel subassembly showing the cut out of fuel pins, bulging and bowing.

**TABLE 1. PFBR Components and Conditions**

<p><b>Operating Conditions of PFBR components</b></p> <p><b><i>Fuel Subassembly (Fig. 2)</i></b></p> <p>Core Structurals (Clad and Wrapper): Flux <math>\sim 10^{15}</math> n/cm<sup>2</sup>s<sup>-1</sup> (2 orders &gt; in Thermal Reactors)</p> <p>Residence Time of Fuel Subassembly in the Reactor: <math>\sim 2</math> yrs.</p> <p>Fuel Clad Temp.: 673-973 K (steady state operating conditions); upto 1273 K (transient conditions)</p> <p>Wrapper Temp. : 673-873 K (steady state operating conditions); upto 1073 K (transient conditions)</p> <p>Maximum Neutron Dose on both Clad and Wrapper: 85 dpa for the Initial Target Burn-Up  <math>\sim 200</math> dpa for the Desired Burn-Up</p> <p>(dpa: displacements per atom, i.e., average number of times that an individual atom is displaced from its lattice site, a measure of material damage from neutron irradiation)</p> <p>Major Loads on Clad: pressure from accumulated fission gas release from fuel matrix (<math>\sim 5</math> MPa) and moderate fuel-clad interaction (especially during transient over power (TOP) incidents)</p> <p>Major Loads on the Wrapper: internal pressure due to sodium coolant (<math>\sim 0.6</math> MPa) and the interaction loads at the contact pads due to bowing of the subassemblies under temperature and swelling gradients.</p> <p><b><i>Reactor Assembly/Primary Circuit</i></b></p> <p>Min. Na temperature (temp.) of primary pool (Core Inlet Temp. under normal operation): 670 K</p> <p>Mean above Core Temp.(core outlet temp. under normal operation): 820 K</p> <p>Core Outlet Temp. under transients (arising due to failure of pumps, rupture of pump to grid plate pipe, uncontrolled withdrawal of control rod etc.): 923 K</p> <p>Operating Environ.: Liq. Na with Ar gas or Nitrogen plus Na vapour based on component site</p> <p>Fast Fluence: <math>10^{21}</math> to <math>10^{22}</math> n/cm<sup>2</sup> for Near Core Components like GP; Radiation Not Important for Other Components</p> <p>Primary Na Circuit: Primary Stresses Low, but Secondary Thermal Stresses High</p> <p><b><i>Steam Generator (SG)</i></b></p> <ul style="list-style-type: none"> <li>• A single wall separates the water/steam from sodium</li> <li>• Very high reactivity of sodium with water makes the SG a critical component</li> <li>• Sodium Inlet temperature: 798 K; Steam Outlet Temp.: 766 K.</li> </ul>
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Steam Generators should have good high temperature mechanical properties including creep and LCF. Resistance to loss of carbon to liquid sodium and consequent reduction in strength, resistance to wastage in case of small leaks leading to sodium-water reaction, and resistance to stress corrosion cracking (SCC) in sodium (caustic) and water (chloride) media are other important considerations.

Roof Slab materials should have adequate through thickness ductility to avoid lamellar tearing in weld joints. Weldability and fabricability of the above materials and the properties of the weldments

in reactor conditions also decide the choice of the material for particular applications.

## Materials for PFBR Components

### Core Structural Materials

20% cold worked (CW) Alloy D9 (15Cr-15Ni-Mo-Ti-Si) (Table 2) has been chosen for the initial core of the PFBR. The formation and growth of voids and consequently swelling is sensitive to nearly all the metallurgical variables like chemical composition and thermo-mechanical history, and irradiation parameters like fluence, dose

**TABLE 2. PFBR Material Specification: Composition wt%, unless specified otherwise (single values denote maximum permissible)**

Element	PFBR D9 Core Material	PFBR 304L(N) SS	PFBR 316L(N) SS	PFBR Mod. 316(N)SS Electrodes	PFBR Mod.9Cr-1Mo Steel Tubes	PFBR Mod.9Cr-1Mo Filler Wire	PFBR Mod.9Cr-1Mo Electrodes
C	0.035-0.05	0.024-0.03	0.024-0.03	0.045-0.055	0.08 – 0.12	.08-.12	.08-.12
Cr	13.5-14.5	18.5-20	17-18	18-19	8.00 – 9.00	8.0-9.5	8.0-9.5
Ni	14.5-15.5	8-10	12-12.5	11-12	0.20	0.6-1.0	0.6-1.0
Mo	2.0-2.5	0.5	2.3-2.7	1.9-2.2	0.85 – 1.05	0.85-1.05	0.85-1.05
N	0.005	0.06-0.08	0.06-0.08	0.06-0.10	0.03 – 0.07	0.03-0.07	0.03-0.07
Mn	1.65-2.35	1.6-2.0	1.6-2.0	1.2-1.8	0.30 – 0.50	0.5-1.2	0.5-1.2
Si	0.5-0.75	0.5	0.5	0.4-0.7	0.20 – 0.50	0.2-0.4	0.2-0.3
P	0.02	0.03	0.03	0.025	0.02	0.015	0.01
S	0.01	0.01	0.01	0.02	0.01	0.01	0.01
Ti	5-7.5xC	0.05	0.05	0.1(Ti+Nb+Ta)	-	-	-
Nb	0.05	0.05	0.05	-	0.06 – 0.10	0.04-0.07	0.04-0.07
Cu		1.0	1.0	0.5	-	-	-
Co	0.05	0.25	0.25	0.2	-	-	-
B	10-20 ppm	0.002	0.002	0.002	-	-	-
Misc.				$\delta$ -ferrite: 3-7FN as per WRC-92 FN diagram	V:0.18-0.25 Al:0.04 Sn:0.02 Sb:0.01 Ti:0.01	V:0.15-0.2 2 Ni+Mn $\leq 1.5$	V:0.15-0.22 Al:0.04 Ni+Mn $\leq 1.5$

**Inclusion Contents in 304/316 L(N) SS (max.)**

Type	Thin	Thick
Type A (sulphide)	1	0.5
Type B (alumina)	2	1.5
Type C (silicate)	2	1.5
Type D (globular oxide)	3	2.0
A+B+C+D	6	4.0

rate and irradiation temperature. Peak swelling in austenitic SSs takes place generally in the temperature range 673-973K during irradiation in FBRs. The fluence dependence of swelling can be described as an incubation dose, a transient period with low rate of swelling followed by an acceleration to a regime of near linear swelling rate. Improvements in swelling resistance in advanced alloys are by way of a longer incubation and transient

regimes. Core component materials have evolved continuously from the first generation materials of 304 and 316 SS grades (which reach unacceptable swelling at doses higher than 50 dpa) by the addition of stabilising elements, changes in the chemical composition of major and minor elements and by modifications of the metallurgical structure by cold working. Increased Ni content and decreased Cr content for improved creep and irradiation



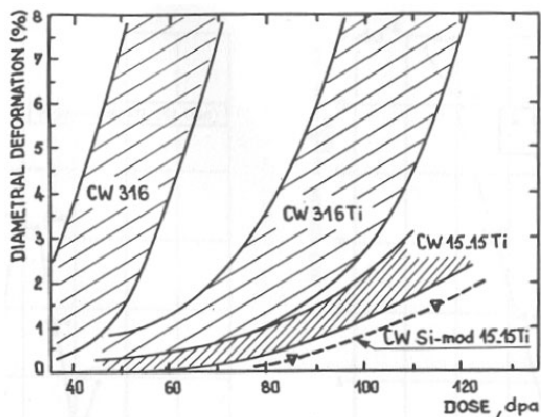


Fig. 3 Variation with dose of the maximum diametral deformation of fuel pins irradiated in Phenix for various cladding materials.

resistance, optimised addition of solutes Ti, Si, P, B and C plus optimised CW level of 20% have resulted in the advanced alloys D9 and D9I. Figure 3 shows the swelling resistance of various austenitic SSs. The major difference in the four alloys shown in Fig. 3 is the increase in the incubation dose for swelling. CW15-15Ti has reached a record dose of 140 dpa without excessive deformation. Ti/C ratio is known to play an important role in determining irradiation behavior. Maximum swelling resistance in CW 15-15Ti in a high carbon grade (C 0.08 to 0.12 wt%) has been obtained when Ti/C ratio is below the stoichiometric composition i.e. when the material is understabilized (i.e., Ti content of less than four times the carbon content in weight percentage). The reason for this behavior is the synergistic interrelation between freely migrating carbon and the formation of finely dispersed TiC particles.

Minor elements having high neutron absorption cross-section and impurities affecting weldability have been kept to a minimum (Table 2). Permissible inclusion contents are stringent so as to minimise radiation embrittlement and sodium attack. Recent international alloy development efforts and in-reactor experience have led to modified D9 known as D9I (with basic composition same as that of D9 but with additions of P, Si and B in the ranges of 0.025-0.04 wt%, 0.7-0.9 wt%. and

0.004-0.006 wt% respectively) and will be considered for future cores of PFBR.

### Reactor Assembly and Circuit Materials

Austenitic SSs are chosen as the major structural materials in view of their adequate high temperature mechanical properties, compatibility with liquid sodium coolant, good weldability, availability of design data, good irradiation resistance and above all the fairly vast and satisfactory experience in the use of these steels in sodium cooled reactors. Monometallic construction is preferred for liquid sodium systems because of the concern of interstitial element transfer through liquid sodium due to the differences in activity in a bimetallic system. Hence, austenitic SSs are employed in the entire liquid sodium system (except steam generator), even if the temperatures of some components are low enough to use less expensive ferritic steels.

For PFBR, low carbon austenitic stainless steel types 304 and 316, alloyed with 0.06-0.08 wt% nitrogen, designated as 304L(N) SS and 316L(N) SS respectively have been selected for the structural components (Table 2). 316L(N) SS will be used for components experiencing relatively higher temperatures (above 770 K) while 304L(N) SS has been selected for the rest of the structural components since cost of 304L(N) SS is less by 20% in terms of material cost. For Grid plate, though temperatures are not in the creep region, 316L(N) SS is preferred over 304L(N) in view of better ductility after irradiation. Low carbon grades have been chosen to ensure freedom from sensitisation during welding of the components to avoid risk of chloride stress corrosion cracking during storage in coastal site. Since low carbon grades have lower strength as compared to the normal carbon containing grades, nitrogen is specified as an alloying element to make up for the loss in strength resulting from reduced carbon levels.

Although 304L(N) and 316L(N) are specified by ASME with nitrogen in the range of 0.10 to 0.16 wt%, for PFBR, nitrogen content is limited to 0.08 wt% in view of improved weldability, code data availability and for minimising scatter in mechanical properties; specification of minimum Mn and maximum Co level are for improved weldability and

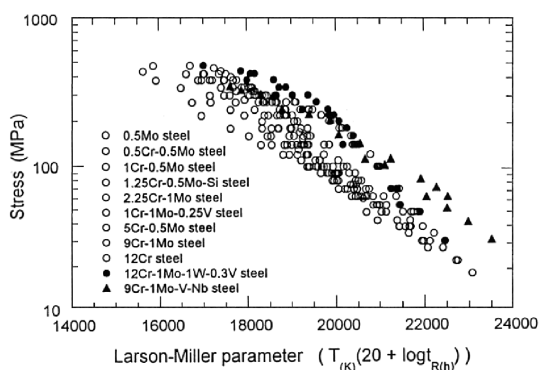


Fig. 4 Creep-rupture strength of eleven types of ferritic heat resistant steels. Open circle is used as the symbol to represent nine different grades of ferritic steels which converge at long test durations

reduced Co<sup>60</sup> activity. A specification for inclusions has been added keeping in view that sulphide inclusions are most detrimental especially from welding considerations, and globular oxides are least harmful. A grain size finer than ASTM No. 2 is specified so as to achieve optimum high temperature mechanical properties.

#### Steam Generator (SG) Materials

To ensure high integrity and reliability of the critical tube to tube sheet welds, PFBR SG will be manufactured in mono-metallic material (tube, shell and thick section tube-sheet/plate). Modified 9Cr-1Mo ferritic steel has been selected for all the steam generator components, in line with recent trend favours. The chemical composition (Table 2) is controlled within close limits to avoid scatter in the mechanical properties. Lower limits are specified for residual elements, like sulphur, phosphorous and silicon to improve weldability and reduce the inclusion content to ensure a high degree of cleanliness. In addition, this steel is less sensitive to minor variations in Post-Weld Heat Treatment (PWHT) conditions. Modified 9Cr-1Mo steel exhibits higher creep strength than most of the other materials (Fig. 4); also, Mod.9Cr-1Mo steel, unlike several other ferritic steels, does not exhibit a drastic reduction in creep strength at longer durations due to the stability of its microstructure. This is the most

important aspect favouring the selection of Mod. 9Cr-1Mo steel for steam generator. Further, the higher creep strength of Mod. 9Cr-1Mo steel allows the use of comparatively thinner tubes and also facilitates lower heat transfer area. Other properties like LCF creep-fatigue interaction are also better than other ferritic steels.

#### Welding Consumables and Properties of Welds

##### Austenitic SSs

Welding is extensively employed in the fabrication of FBR components. Weld metal cracking and heat affected zone (HAZ) cracking are major areas of concern in welding austenitic stainless steels. Weld metal cracking can be controlled by optimising the chemical composition of the welding consumables. The optimised chemical composition for 316(N) SS welding electrodes is given in Table 2. Carbon in the range of 0.045-0.055 wt% and nitrogen in the range of 0.06-0.1 wt% are specified to provide weld joints with improved creep strength and freedom from sensitisation in the as-welded state. In addition, ferrite in the weld metal is specified to be between 3-7 ferrite number (FN) to promote ferritic solidification mode. A minimum of 3 FN is specified to ensure freedom from hot cracking in the weld metal. Because delta-ferrite undergoes phase changes to carbides and brittle intermetallic phases at high temperatures, an upper limit of 7 FN has been specified. HAZ cracking is avoided by specifying lower permissible limits for P, S and Si and also by specifying limits on B, Ti and Nb which are not specified in the ASTM standards for the base metal. 316(N) SS electrodes would be utilised for welding of both 316L(N) SS and 304L(N) SS base materials. This would avoid any mix-up of electrodes in welding if a different electrode is selected for 304L(N) SS. 16-8-2 filler wire will be used for TIG welding since this composition has better microstructural stability, creep strength and toughness.

##### Modified 9Cr 1Mo Steels

For the welding of Mod. 9Cr-1Mo steel, consumables having composition closely matching with that of the base metal are normally employed. However, achieving required toughness in the weld

metal after PWHT has been an important issue steel, especially in the case of shielded metal arc (SMA) welds. Use of only alloyed core wire for making SMA welding electrodes is permitted. The specified PFBR composition is given in Table 2.

#### ***Trimetallic Transition Joint***

As the main structural and piping material is austenitic 316LN SS, and the SG material is Mod. 9Cr-1Mo steel, a dissimilar weld involving these two materials is inevitable in the construction of PFBR. A large number of premature failures of the direct joint involving austenitic stainless steel and ferritic steel operating at high temperature have been reported in the past, mainly from fossil power plants. The failures are mainly attributed to (i) large differences in the thermal expansion coefficients of these two steels, which lead to generation of thermal stresses during start-up and shut-down, (ii) difference in the creep strength of these materials, and (iii) carbon migration from ferritic steels to austenitic steels leading to formation of soft zone near the interface. Introduction of an intermediate piece of material, having thermal expansion coefficient value between austenitic SS and ferritic steel, can significantly reduce the thermal stresses generated. Accordingly, a trimetallic joint configuration in which an Alloy 800 spool piece welded to 316L(N) SS pipe on the one side, and Mod.9Cr-1Mo steel pipe on the other side, is chosen for this dissimilar joint. For welding Alloy 800 to Mod. 9Cr-1Mo, Inconel 82/182, welding consumable is recommended. For welding Alloy 800 to 316L(N) SS, 16-8-2, filler wire is selected.

#### **Roof Slab and Hard Facing Materials**

Because of less demanding environment faced by the roof slab, a carbon steel equivalent to ASTM A516 Grade65 Steel, namely, A48P2 (French RCC-MR Code) has been chosen with particular emphasis on cleanliness and lower residuals and inclusions to ensure better weldability and through-thickness ductility leading to better resistance to lamellar tearing of welds. Nickel-base Colmonoy alloys have been selected and demonstrated as suitable replacement for the cobalt-base Stellite alloys as the hardfacing material for the Nuclear Steam Supply System components of PFBR. This choice of hardfacing material is aimed at

keeping induced radioactivity to the minimum for maintenance and decommissioning purposes.

#### **Indigenous Development and Production**

It was important to develop indigenous sources for materials needed for the construction of PFBR in view of the long term programme to construct a series of FBRs. These materials are required in different product forms like plates, forgings, tubes, rods and hollow bars. Dimensions, grade and tonnage of stainless steel required in various product forms for the 500 MWe PFBR are listed in Table 3. The close control of chemical composition and very low inclusion contents specified for these materials necessitate use of special steel making processes like vacuum arc remelting (VAR), vacuum oxygen decarburisation (VOD), argon-oxygen decarburisation (AOD) or similar secondary refining processes or electro-slag remelting (ESR) for 9Cr-1Mo steels. Very stringent non-destructive examinations are also specified in order to detect the presence of flaws which can influence the performance of the components during service. It is now possible to manufacture stainless steel ferritic steel and low alloy steel of close chemical composition and low inclusion contents within the country. A collaborative effort with organizations such as SAIL, MIDHANI and NFC has resulted in the development and production of plates of larger dimensions within the country than ever produced as well as manufacturing of long tubes for SG applications. Another important achievement has been the production of cladding tubes and hexcans required for the core of the reactor, meeting very stringent specifications for both chemical composition as well as dimensional requirements. Indigenous development of welding consumables is also underway in collaboration with various manufactures.

#### **R&D Activities**

Extensive R&D programmes have been under way on the development and characterization of fuel and structural materials selected for PFBR. As creep, fatigue and creep fatigue interaction play a dominant role in the performance of materials for FBR, extensive studies have been carried on austenitic stainless steel, ferritic steels and their weldments with a view to understanding their high temperature

**TABLE 3. Grade, size and tonnage of steels required for PFBR**

Grade	Product	Size W x L x T (mm)	Quantity (tonnes)	Application
316L(N)	Plate	2500 x 4000 x 1.25-10	100	Main vessel, inner vessel, IHX, etc.
		2500/3000 x 8000 x 12-20	450	
		2500/3000 x 8000 x 25-40	325	
		2500/3000 x 8000 x 55-120	125	
304L(N)	Plate	2500/3000 x 8000 x 3-4	300	Tanks, pumps, safety vessel, IFTM
		2500/3000 x 8000 x 6-25	800	
		2500/3000 x 8000 x 25-50	20	
304L(N)/ 316L(N)	Forgings	835-2350 mm OD x 345-1860 ID x 100-160 mm thick	80	Heat exchanger tube sheet, pumps
316L(N)	Seamless tubes	19 OD x 0.8 mm WT x 8 m	50	Heat exchangers
304L(N)/ 316L(N)	Seamless pipes	15-200 mm nominal size Sch. 40	350	Sodium and gas piping
316L(N)	Rounds/ Hollow Bars	60-325 mm diameter x 4-5 m long	300	Control rod drives, Fuelling machines
Mod.9Cr- 1Mo	Plates	2000/3000 x 5000 x 12/24 mm	150	Steam generator
	Seamless tubes	17.2 OD x 2.3 mm WT x 23m	100	Steam generator
	Forgings	850-1510 OD x 375-1185 ID x 80-350 thick	150	Ring Flanges
		1250 OD x 260 thick		Tubesheet

mechanical behaviour. These include evaluation of creep rupture behaviour of 316 LN and its weldments [1], time and temperature dependant degradation mechanism in high temperature low cycle fatigue on 316 LN [2], influence of grain size and cold work on creep and low cycle fatigue of austenitic stainless steel [3, 4], evaluation of dynamic fracture toughness [5] and quasi-static fracture properties of austenitic and ferritic steels. Dynamic materials modelling has been employed to identify safe forming windows in the strain rate-temperature space as well as to achieve desirable microstructural features [6,7,8]. Extensive intergranular corrosion (IGC) and intergranular stress corrosion cracking (IGSCC) studies have been carried out in chloride and caustic environments. Time temperature sensitization curves have been generated to arrive at permissible cooling rates during fabrication [9]. The role of cold work in influencing corrosion behaviour has also been assessed. Corrosion studies have revealed that

addition of nitrogen to 316 L retards the sensitization kinetics.

Weldability studies have been conducted to assess the role of various elements on hot cracking and to arrive at optimum composition for stainless steel welding consumable [10]. The weldability of D9 needs further evaluation since it is fully austenitic and contains titanium both of which promote hot cracking. Nitrogen presence in 316LN has been found to increase cracking tendency particularly when sulphur levels are high. High temperature low cycle fatigue behaviour of weldments has been examined in details to understand factors responsible for poor life of weldments [11,12]. In order to study mechanical behaviour of materials in the sodium environment facilities for creep and fatigue testing in sodium have been set up. These investigations would lead to more realistic life assessment of PFBR components.

## Summary

The choice of structural materials for FBRs is governed by a combination of a unique set of operating conditions and a major concern for reliability and safety of the components. High temperature mechanical properties such as creep, low cycle fatigue, compatibility with liquid sodium coolant, stress corrosion cracking, void swelling and weldability are some of the important considerations that are to be addressed in the choice and development of FBR materials. Systematic and exhaustive studies have been carried out to understand these properties in the case of materials used in the construction of FBTR and materials selected for PFBR. Special emphasis has been given to study the properties of welds. The success of FBRs as an economically viable alternative source of electric power lies in increasing the burn-up of the fuel in order to reduce the fuel cycle cost. Developing all the materials indigenously through collaboration with other organizations has achieved commendable success. A more exhaustive and detailed coverage of both fuels and structural materials including related R&D is given in the two recent publications [13,14].

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# Chemistry of Sodium Coolant in Fast Reactors



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## Introduction

Liquid sodium metal is the coolant of choice for fast reactors as it has highly favourable nuclear, physical and chemical properties [1,2]. Sodium does not moderate neutrons. It has a low melting point, which makes it easy to maintain in liquid form. It has a high boiling point making it possible to have a low-pressure primary system, which is easy to design and construct. Thermal convection in liquid sodium sets in easily facilitating positive decay heat removal from the core in the case of pump failures. The reasonably good heat capacity of sodium results in a slow rise of reactor temperature in the event of emergency cooling system failure. This coupled with the high boiling point, provides ample time to take remedial action. The temperature and power reactivity coefficients due to sodium are negative for a fast reactor. Pure sodium is highly compatible with austenitic stainless steel, which is the major structural material for fast reactors. The only negative point is its high chemical reactivity. Hot sodium catches fire in air and reacts explosively with water. Though it appears a bit difficult, the accumulated experience with more than 20 fast reactors for well over 280 total reactor years has shown that it is possible to design, construct and operate safe sodium coolant systems that can withstand high cycling stresses and prolonged exposures to high temperatures without leaking sodium. This article highlights some of the chemistry aspects of sodium relevant to its use as fast reactor coolant.

## Properties of Sodium

### Physical and Nuclear

Solid sodium has a body centered cubic structure. It melts at 370.8 K to give a silvery liquid and boils at 1156 K. Liquid sodium has a density of  $0.9 \text{ g cc}^{-1}$  at 473 K and a specific heat of  $1383 \text{ J kg}^{-1} \text{ K}^{-1}$  at 371 K. Its viscosity (0.45 cps at 473 K) is similar to that of water making it possible to test sodium-pumping components using water. It has a high thermal conductivity ( $0.84 \text{ J cm}^{-1} \text{ K}^{-1} \text{ s}^{-1}$  at 473 K).

Liquid sodium has a neutron absorption cross section of 0.87 mb for fast neutrons. The (n, $\gamma$ ) activation product of sodium,  $^{24}\text{Na}$ , is short lived ( $t_{1/2}=15 \text{ h}$ ). The (n, 2n) product,  $^{22}\text{Na}$ , does not build up much as it has a high absorption cross section for thermal neutrons. The negative reactivity coefficients of sodium for temperature and power result in stable negative feedback providing intrinsic safety [3]. The sodium void coefficient is also negative for small reactors and is only slightly positive for large reactors. This can be taken care of by reactor design [4]. In fact, a net void reactivity coefficient of  $\sim 0$  for the upper core and sodium has been achieved for BN-800 [3].

### Chemical Properties

Being an alkali metal, sodium is highly reactive and forms compounds with all the non-metals except nitrogen and noble gases. On exposure to dry air,

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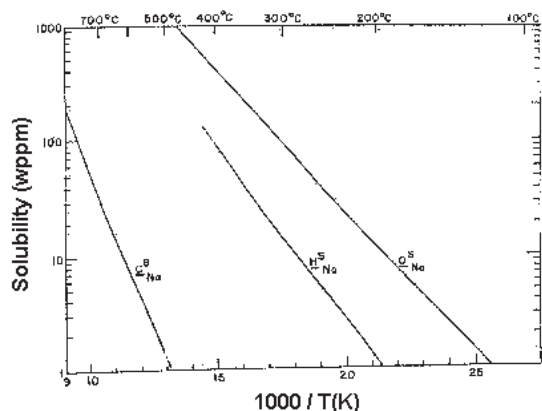


Fig. 1 Solubility of O, H, C in Sodium

sodium reacts with oxygen to give sodium oxide and sodium peroxide. Liquid sodium burns in air to give sodium oxide above 453 K. Liquid sodium reacts with hydrogen to give sodium hydride, the kinetics of the reaction being slow below 523 K and fast only above 573 K. Solubility of  $\text{Na}_2\text{O}$  and  $\text{NaH}$  in sodium is a very steep function of temperature. Solubility of oxygen in sodium is 1 ppm at 371 K and 420 ppm at 673 K and that of hydrogen is 2 ppb and 94 ppm respectively. This property is made use of in sodium purification where sodium is passed through a cold region called cold trap at  $\sim 400$  K to precipitate the oxide and hydride and obtain pure sodium. Carbon solubility in sodium is very low (4 ppb at 573 K).

Solubility of the constituents of stainless steel, such as Fe, Cr, Mo, Mn and Ni in sodium are very low and are in ppm ranges at the reactor temperature (Fig. 1). When the oxygen levels are higher than 3 ppm above 700 K, sodium reacts with chromium to give sodium chromite. This is the main product formed in sodium circuits and its formation controls oxygen potentials in sodium. Ternary oxides of other metals such as  $\text{Na}_4\text{FeO}_3$  and  $\text{Na}_4\text{MoO}_5$  can form only at much greater oxygen levels, usually not encountered in sodium systems. Coinage metals, low melting metals and noble metals have high solubilities in sodium. They also form intermetallic compounds such as  $\text{NaAg}$  and  $\text{Na}_2\text{Au}$  [5-7]. Sodium reacts exothermically with water to give  $\text{NaOH}$  and  $\text{H}_2$ . Sodium reacts with alcohols to form alkoxides. Hydrocarbons can crack on exposure to sodium above 600 K to produce methane, hydrogen and amorphous carbon.

TABLE 1. Specifications for PFBR sodium

Impurity	Specified (ppm)
Oxygen	< 30
Carbon	< 30
Calcium	< 10
Iron	< 5
Halogens	< 20
Sulphur	< 30
Barium	< 5
Silver	< 1
Zinc	< 2
Potassium	< 300
Lithium	< 10
Boron	< 5
Uranium	< 0.01
Sodium	99.95%

### Sodium Quality Control and Monitoring

Pure sodium is quite compatible with austenitic stainless steel. But presence of impurities such as oxygen in sodium even in ppm levels can enhance corrosion and activity transport. Carbon in sodium can carburize stainless steel and make it brittle. Trace metals can get activated and give rise to long lived radionuclides. For this reason, stringent specifications are imposed for reactor grade sodium (Table 1).

Sodium metal is commercially produced by Down's process. In this process molten  $\text{NaCl}$  is electrolysed at about  $650^\circ\text{C}$  using graphite anode and steel cathode. Commercial sodium is too impure for its direct use in fast reactors.

To get reactor grade sodium, commercial sodium has to be filtered near its melting point using coarse and fine filters in that order and then cold trapped in a loop to remove oxygen. This sodium has to be chemically analysed for both non-metallic and metallic impurities before charging it into the reactor circuits. Obtaining a representative sample of liquid sodium for chemical analysis is difficult because of impurity segregation during cooling and

contamination while handling. Though many different techniques are in use at various fast reactors, overflow sampling technique is believed to provide the most reliable samples free from segregation problems.

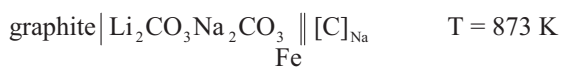
Monitoring sodium coolant for impurities like oxygen, hydrogen and carbon will help detect maloperations in a fast reactor such as air leak into sodium, steam generator leak and oil-leak through the pump shaft seals respectively. The oxygen sensor for sodium monitoring is a galvanic cell based on costly yttria doped thoria solid electrolyte which is hard to get commercially. A zirconia based oxygen sensor, which operates at 523 K, has been developed as an alternative at IGCAR for this purpose. The cell can be given as



The hydrogen detection system used at FBTR measures the hydrogen flux through a nickel membrane using a mass spectrometer. The membrane is kept exposed to sodium at 723 K on one side and vacuum on the other. Electrochemical sensors based on a hydride-ion conducting solid electrolyte  $\text{CaCl}_2\text{-CaH}_2$  have also been developed and tested in FBTR. The electrochemical cell of the hydrogen sensor can be given as



An electrochemical carbon meter, which can measure the carbon activity in sodium and thus detect any oil-leak into sodium, has also been developed at IGCAR. The cell can be represented as



All these electrochemical sensors provide an emf output, which varies with the concentration of the impurity in sodium [5]. Monitoring the argon cover gas for hydrogen is useful in detecting steam generator leaks at low sodium temperatures (<523 K) as obtained during reactor start up or at low temperature operations. Under these conditions, the in-sodium sensors may not work well as the kinetics of hydrogen dissolution in sodium is slow. Monitoring the cover gas for methane using a gas chromatograph can help detect oil leaks, as oils crack

on contact with hot sodium to give methane and carbon particles. Nitrogen in cover gas can be measured using a gas chromatograph. Its presence indicates air leak into sodium. All the above facts point to the need for continuous monitoring of sodium and argon cover gas in a fast reactor and the importance of a well equipped chemical lab as a part of the reactor building needs no emphasis.

### Sodium Corrosion and Activity Transport

Localized electrochemical corrosion is absent in sodium. However, liquid sodium can slightly dissolve the constituents of stainless steel at high temperatures, which can get transported and deposited at different regions [5].

This mass transfer is influenced by impurities such as oxygen as it can form ternary oxides. The different processes involved in the corrosion by liquid sodium include i) loss of material leading to reduction in wall thickness and ii) preferential leaching leading to modified layer of inferior properties

Based on data generated in the last 40 years on sodium corrosion, thickness loss is estimated to be less than a micron per year, which is considered negligible. Similarly, degraded layer due to differential leaching is estimated to be about 40  $\mu\text{m}$  in 40 years and it needs to be factored only in thin components such as IHX having a wall thickness of 0.8 mm and design life of 40 years. Even after accounting for the total corrosion allowance (100  $\mu\text{m}$  on either side of the tube) the remaining thickness in primary sodium system components is sufficient to take care of design loads. Substantial research has been carried out on carburization and models established for prediction of long-term carbon transfer. Operating experience and the model predictions indicate no serious problems on this count.

Corrosion in fast reactor primary system has a strong influence on activity transport. Radionuclides produced in the reactor core by neutron activation ( $^{54}\text{Mn}$ ,  $^{60}\text{Co}$ ,  $^{51}\text{Cr}$ ) can get slowly released into sodium by corrosion processes. In the event of a fuel pin failure some fission products ( $^{137}\text{Cs}$ ,  $^{132}\text{Te}$ ,  $^{140}\text{Ba-La}$ ) are also released into sodium. These nuclides can potentially travel along the primary

coolant circuit and depending on the nature of the individual nuclide can get deposited at various positions/components in the primary circuit [8,9]. The radiation field due to these deposits may restrict the approach to reactor components for maintenance and repair leading to longer reactor downtime.

Most of the radionuclides released into sodium at the core, deposit on the system walls immediately downstream. Only  $^{54}\text{Mn}$  has been found to migrate farther and reach the cold leg, possibly because of its greater solubility in sodium and high release rate from austenitic stainless steel. In fact, more than 90% of corrosion product activity found in the cold leg in most reactors is due to  $^{54}\text{Mn}$ . Among fission products,  $^{137}\text{Cs}$  is the problem nuclide. Almost the entire  $^{137}\text{Cs}$  inventory in a fuel pin is released into sodium in the event of pin failure. Owing to its high miscibility,  $^{137}\text{Cs}$  gets distributed throughout the primary coolant system. The other radionuclides released from a failed fuel are  $^{95}\text{Zr}$ (6%),  $^{144}\text{Ce}$ (5%) and  $^{140}\text{Ba-La}$ (6%). Among these, only the last one can reach the cold leg to some extent after prolonged operations.

The release, transport and deposition of radionuclides in a reactor system is a very complex process. Consequently, only a model, which takes into account all the parameters of relevance, can predict the possible radiation fields due to radioactive deposits in different parts of a reactor system. Codes that can do this have been developed in various countries (e.g. CORONA code in FRANCE and PSYCHE code in JAPAN). The transport of the radionuclides released into sodium can be minimised by using suitable radionuclide traps. Nickel foil at 823 K located at the top of fuel subassembly can trap  $^{54}\text{Mn}$  and prevent it from migrating. Reticulated vitreous carbon (RVC) exposed to sodium at 473 K can remove  $^{137}\text{Cs}$  from sodium very effectively. PFBR will have an in-core Cs trap which can be used whenever  $^{137}\text{Cs}$  levels in sodium go up.

### Fuel-Coolant Reaction

When the mixed oxide comes in contact with sodium after a fuel pin failure, it reacts with sodium to give a compound sodium urano-plutonate ( $\text{Na}_3\text{U/PuO}_4$ ) (Fig. 2).

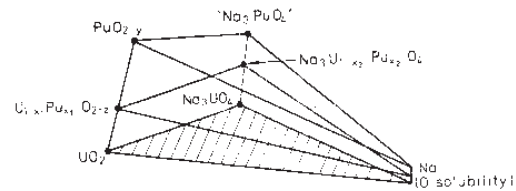


Fig. 2  $\text{Na}-(\text{U,Pu})\text{O}_2\text{-Na}_3(\text{U,Pu})\text{O}_4$  ternary phase diagram

It is a low-density compound whose formation leads to fuel pin swelling. This is an oxygen consuming reaction and can progress till the oxygen potential of the fuel equals that of the product. The thermodynamics of this reaction is well understood and it is possible to estimate the extent of the reaction and the swelling with sufficient confidence [10].

The volume increase due to fuel-coolant reaction can easily be accommodated by the hot operating fuel as it is sufficiently plastic. As a consequence, the reactor can be operated with a failed fuel pin for some time till the delayed neutron signals reach the set threshold though release of some fission products like  $^{137}\text{Cs}$  can take place in the meanwhile. Only in high burn up fuels with large uniform stress on the clad, the breach can open up more. Fuels that failed in the core can be moved to the internal storage position and kept there without further defect propagation.

Fuel-coolant chemical reaction becomes a matter of concern only when the clad failure occurs after the fuel is moved over to storage position where the fuel is not on power. Here the stress due to swelling does not get distributed since the fuel cannot creep at these temperatures and this can sometimes lead to severe pin ruptures. For virgin fuel, the kinetics of fuel-coolant reaction is slow below 700 K. For high burn up fuels, however, the threshold temperature may be much lower. When the breach is large, pure sodium, which is low in oxygen, can reduce the fuel and facilitate its release. If the oxygen in sodium is high, a large amount of fuel can react and the pin can disintegrate. However these are low probability events.

## Sodium Fire

Due to the high chemical reactivity of sodium, in the event of exposure to air, the oxidation reaction of liquid sodium can occur in a run-away manner and lead to fire. The ignition temperature for sodium is around 593 K in damp air, 473 K in dry air and 393 K in stirred condition [11]. The severity of consequences of sodium fire depends on the efficiency of dispersion, sodium to air ratio, the non-adiabatic nature of the container and its contents. Spray fires are more severe than pool fires. In a spray fire incident at Almeria, Spain in August 1986, 14 t of sodium spilled over half an hour giving rise to temperatures up to melting of metallic structure. However, the damages were limited to fire zone and did not lead to any large-scale destruction [12]. This is because the intensity of sodium fires is much less (15 times) when compared with hydrocarbon fires due to the following factors. They are: (1) the low combustion rate, (2) large latent heat of evaporation, (3) lower heat of combustion and (4) lower flame height (Fig. 3). The sodium fire produces dense smoke that reduces visibility but cuts down the heat radiation permitting closer approach for fire fighting.

The progression of a sodium pool fire is as follows. Rapid oxidation of the surface takes place initially. The oxide thickness grows and wrinkles giving rise to oxide nodules. Around 623-723 K, small flames appear on the tip of the nodules, which act as wicks. This proceeds till the pool temperature reaches 837 K. Around this temperature, the heavier oxide layer sinks to the pool bottom. The ensuing vapour phase combustion consumes the sodium and eventually burning rate declines. Both sodium monoxide and sodium peroxide are formed at varying ratios depending on the conditions. About 30% of sodium is released as aerosols.

The best way to extinguish the sodium fire is to exclude oxygen from the metal by spreading dry chemical powder, which is a mixture of sodium carbonate and sodium bicarbonate. Graphite based powders (90% graphite and 10% sodium/lithium carbonates) are also used. Metallic collection trays are provided at possible sodium leak areas. These are boxes with slotted and sloped covers to facilitate the sodium drainage and at the same time limit the exposed area so that oxygen supply is limited. Even

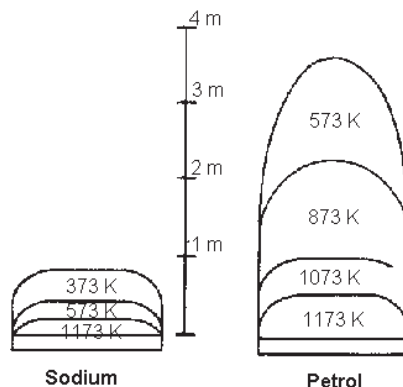


Fig. 3 Flame characteristics of sodium and Petrol

partitioning the fire area for inert gas flooding is resorted to.

## Sodium-Concrete Reaction

Proper choice of concrete is essential to minimize damages when a sodium leak occurs. When hot sodium comes into contact with concrete, free water is released as steam, which reacts exothermically with sodium leading to hydrogen explosions. At higher temperatures ( $>473$  K) release of chemically bound water occurs. The Sodium hydroxide produced can chemically react with granite, which forms the coarse aggregate. The exothermic reactions of sodium with coarse and fine aggregates can build up internal pressure in the solid causing cracking and swelling. Based on extensive studies carried out, limestone has been identified as the best available coarse and fine aggregate material for making the concrete for use in sodium area. Use of metallic liner with back up thermal insulation avoids both thermal and chemical attack of sodium on concrete. Reducing the water content without compromising on strength is another such approach. Based on preliminary studies at IGCAR, dunite is identified as another suitable coarse aggregate material. An aluminous concrete that does not interact much with sodium has been proposed in France [13].

## Sodium Water Reaction

Sodium reacts readily with water to form NaOH and  $H_2$ . This reaction is highly exothermic. In the event of a steam generator leak due to a manufacturing defect, the high-pressure steam



entering hot sodium takes the shape of a reaction flame (Fig. 4). The caustic produced can attack the tube wall and lead to the expansion of the leak. The lengthening flame can cut an adjacent steam carrying tube leading to large-scale steam release, sodium-water reaction and explosion. So, the steam leaks have to be detected at the very inception by using on-line hydrogen meters and remedial actions taken. This has been done successfully in all operating reactors. In a few instances where the leak resulted in explosions, the reaction products have been safely let out through collapsing passive action membranes into special containers to separate the sodium from hydrogen, which is then dumped through a pipe into the atmosphere [3]. Modular steam generators suitable for limiting the damage have been designed which has led to their greater availability. The thermal effect of a large-scale steam release is quite severe. In one such event in PFR about 40 steam-carrying tubes were destroyed. However, it can be said that the technology to design and operate the steam generator in a reliable manner has been mastered.

### Sodium Removal and Disposal

The reactor components need to be cleaned free of sodium before repair. Various techniques used for this purpose include alcohol dissolution, water vapor-nitrogen process, water vapour-CO<sub>2</sub>-nitrogen process, water vacuum process and vacuum distillation. The choice depends on the type of component to be cleaned.

Alcohol dissolution using ethanol is employed for small and delicate components. Large components such as sodium pump, IHX, SG are cleaned by water vapour-CO<sub>2</sub>-inert gas process [14]. This method is recommended for PFBR as it is free from possible caustic stress corrosion attack and the reaction product layer formed is porous. Where faster cleaning is required, such as in spent fuel subassembly cleaning, steam-nitrogen process is used since the rate of steam introduction is greater and the N<sub>2</sub> can also remove the decay heat. Components like cold trap can be cleaned by vacuum distillation.

Sodium disposal in small scale is used to be carried out by dissolution in heavy alcohol like ethyl carbitol. However, an explosion in Rapsodie in

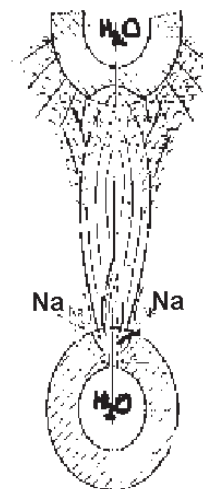


Fig. 4 Sodium water reaction flame

March 1994 during one such campaign indicated that the alcohol could decompose to give gaseous products under some conditions [14]. Since then, use of heavy alcohols is not recommended for disposal. Reaction with air and injection into pool of 10M NaOH are the other techniques adopted for sodium disposal.

### Summary

Liquid sodium is the coolant of choice in fast reactors as it has the necessary nuclear, physical and chemical properties required for a fast reactor coolant. Its high thermal conductivity, high boiling point and the non-moderating properties are the crucial factors. Reactivity coefficients of temperature and power are negative. The void coefficient is also negative for small reactors. However, it has a high chemical reactivity towards air and water that necessitates special design features in sodium systems. This has been effectively mastered. The technology for sodium production, methods for quality control and sensors for on-line monitoring are available. Sodium loops to verify the technologies and components have been successfully operated. Corrosion of austenitic stainless steel in sodium is negligible. Carburization and decarburization phenomena also do not pose any problem. Problems of <sup>54</sup>Mn transport and <sup>137</sup>Cs release into sodium by fuel pin failure are easily overcome by incorporating radionuclide traps.

Sodium fire is much less intense and means of containing and extinguishments are available. The chemical interaction between sodium and low plutonium fuel is mild and the reactor can be operated with failed fuel pin in the core. Methods to remove sodium from components and its disposal have been sufficiently developed.

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# An Integrated Approach to Reprocessing and Waste Management of Fast Breeder Reactor Fuel



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**Shri R. Natarajan**, a chemical engineer from the 19<sup>th</sup> batch of BARC training school is currently the Associate Director, Reprocessing Group, IGCAR. His current responsibilities include erection and commissioning of the pilot plant, called, Lead Mini Cell (LMC) and the demonstration plant for processing irradiated FBTR fuels and in the planning of fuel cycle facilities for PFBR at Kalpakkam. He was involved in the J-Rod process campaigns at RDL for the recovery of U-233. This material was useful in fabricating the fuel for KAMINI reactor and the PFBR fuel pin simulation studies. He has evolved the process flowsheets for the Fast Reactor Fuel Reprocessing plant for FBTR and PFBR. He has over 60 technical presentations and publications to his credit.



## Introduction

The success of Fast Breeder Reactor (FBR) programme in India will depend on reprocessing the irradiated fuels with a high burnup of over 100 GWd/t after a short cooling (typically about 6 - 12 months) with less than 1% fissile material loss for the whole fuel cycle. The reprocessing of fast reactor [1,2] fuels is a major task due to high concentrations of plutonium (Pu) and high specific activity [3]. The highly matured Plutonium Uranium Extraction (PUREX) process is likely to remain the workhorse of the FBR fuel reprocessing for the next 2-3 decades. This paper enumerates a road map of four

phases through which the Fast Reactor Fuel reprocessing technology is being developed in India.

## Four Phases of FBR Reprocessing Programme

Phase I: To understand and provide solutions for the challenges of fast reactor fuel reprocessing, the following activities are undertaken in Phase I: 1) Process formulation and validation 2) Design and development of equipment 3) Development of Remote handling systems 4) Development of measurement and monitoring systems 4) Material development 5) Waste volume reduction and management.

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Phase II: Construction and operation of a pilot plant called, Lead Mini Cell (LMC), to demonstrate the process flow sheet and optimise the process parameters

Phase III: Construction and operation of Fast Breeder Test Reactor (FBTR) Fuel Reprocessing Plant (FRFRP) to gain experience in reprocessing of fast reactors fuel with high availability factors and plant throughput

Phase IV: Construction and operation of the commercial plant, Proto type Fast Breeder Reactor (PFBR) Reprocessing Plant (PFRP) for reprocessing of PFBR fuel.

## **Process Development**

### ***Dissolution***

Direct dissolution of carbide fuels of FBTR has been demonstrated in laboratory scale upto 50 GWD/T burnup. Destruction of organics formed during the dissolution step is required, since these compounds interfere with the subsequent extraction steps. This could be carried out either by extended boiling in concentrated nitric acid or electrochemical methods or with ozone. The dissolution studies for 100 GWD/T fuel will be taken up in LMC. Electrolytic dissolver will be used for this purpose. It is also planned to condition Pu to Pu (IV) in the same dissolver by suitable choice of electrodes and application of electric potential. Mixed oxide fuels have been dissolved in nitric acid containing HF [4]. It is planned to use electro oxidative dissolution technique for PFBR fuel so that the use of corrosive HF could be avoided. This will also reduce the loss of Pu due to fluoride complexing.

Undissolved solids in the dissolver solution of LMC will be characterized and if they contain unacceptable levels of Pu, a suitable method to recover Pu will be evolved. It is also planned to characterize the depth of penetration of fission products and Pu in the hull. This will aid in establishing a process to leach the hulls and reduce the Pu content so that an effective hull disposal philosophy could be evolved.

### ***Solvent Extraction***

The simulation code SIMPSEX [5], developed at IGCAR will be validated during the reprocessing of FBTR fuel in LMC. It is planned to study the extraction behavior of fission product nuclides like Tc, Ru, Zr and Np during the extraction runs of LMC. This will aid in optimising the operating parameters of the extraction units. Third phase formation poses a serious problem from the point of criticality as well as the plant availability. Suitable models have already been developed [6] to predict third phase formation during extraction.

### ***Solvent Degradation***

One of the major concerns in deploying PUREX process for FBR fuel reprocessing is solvent damage. Reduction in solvent damage is achieved by reducing the contact time during extraction. During the LMC campaigns, the solvent damage will be assessed and suitable cleaning methods will be evolved.

### ***Diluent Washing***

The aqueous waste streams from solvent extraction cycles will have both entrained solvent as well as traces of dissolved TBP. For removal of dissolved TBP, different designs like pulse columns and mixer settlers will be tested.

### ***Partitioning***

As FBR fuel contains U as well as Pu, there may not be need for separating pure U and Pu. But if the core requires Pu of different concentrations, separation of Pu and U will be required. In-situ electrolytic partitioning is being tested for FBR flowsheets to achieve this separation. This will reduce the addition of external uranium (as uranous solution) which is a feature of conventional PUREX flowsheets. Extensive work [7] carried out in PREFRE, Tarapur, will form the basis of this processing step. Mathematical modeling of this process has been carried out at IGCAR, which will be validated during the FBTR fuel reprocessing campaigns in LMC. Partitioning by direct precipitation of Pu [8] from product solutions containing U and Pu, will also be explored.

### ***Reconversion***

Reduction of waste volume is achieved by direct denitration of mixed nitrate solution from the solvent extraction cycle using microwave. Studies have been conducted [9] at Advanced Fuel Fabrication Facility (AFFF), Tarapur for mixed oxide scrap recycling and a prototype microwave system for glove box applications has been designed. This technology is proposed to be implemented at FRFRP. The sol-gel process is also being evaluated at IGCAR for reconversion as this step reduces the powder-handling step during fuel fabrication.

### **Design and Development of Process Equipment**

#### ***Fuel Pin Chopper***

FBR fuel subassembly (SA) has hexagonal sheath around the fuel pins. The sheath has to be removed before the fuel pin shearing operation. Presently, the FBTR fuel SAs are dismantled by mechanical slitting of the hexagonal sheath. A system is being developed with Nd-YAG laser supplied by CAT, Indore for sub assembly dismantling. This will be tested at the Post Irradiation Examination facility at IGCAR and the prototype will be demonstrated in the Head End Facility for FBTR and PFBR subassemblies.

Single pin chopping is adopted since it is easier to design the shear unit components for remote maintenance. This will also assist in reducing the basket temperatures by ensuring that the highest rated sections of the pins are evenly distributed. Crimping is also less by this process. The single pin chopper developed at IGCAR is designed for remote replacement of the components. For PFBR, to meet the throughput requirements, multipin chopping is being evaluated.

#### ***Dissolver***

Electrolytic dissolution process has been developed at IGCAR to destroy the organics and to aid in dissolution. A titanium dissolver has been fabricated since the corrosion rates of SS 304L are very high in this highly oxidizing environment [10]. A thermosyphon type batch electrolytic dissolver made of Ti has been already fabricated and installed in LMC. Scaling up of this unit is possible without much uncertainty for FRFRP. For PFRP, a batch

dissolver will be first commissioned in FRFRP. The dissolution rate data obtained on irradiated fuel at FRFRP will be used to design the PFRP dissolver. To reduce the fuel inventory it is planned to undertake the design of a continuous rotary dissolver for PFRP.

### ***Clarification Systems***

During dissolution of the irradiated fuel, a fraction of the platinum group fission products remain as insoluble residues. The amount of residue is more in case of FBRs fuelled with Pu, as the fission yield of platinoid group elements from plutonium fission is nearly twice that from uranium fission and the insoluble residues increase with increase in burn-up of the fuel. The fine particles of these platinoid fission products physically distribute themselves between the aqueous and organic phases, leading to poor decontamination. These insoluble residues are also the major cause for the interfacial crud formation in the extraction equipment, upsetting the hydraulic behavior of these units. A high-speed centrifuge that can be operated and maintained remotely is deployed in LMC. Based on the feedback from LMC operation, an improved version will be used in FRFRP to remove the aqueous droplets entrained in the organic phase.

### ***Extractor***

TBP undergoes radiolytic degradation to a significant extent with marked reduction in decontamination factors (DF) beyond an integrated dose of 0.3-wh/l. This necessitates the use of fast solvent contactors like centrifugal extractors with integrated phase contact time of 2 to 3 min. in the high active coextraction cycle. The centrifugal extractors also reach steady state operating conditions very quickly. The choice of contactors for the subsequent cycles would be based on the economic and operational convenience. Extensive work has been carried out at IGCAR towards the development of centrifugal extractors of capacities varying from 5 to 1000 l/h [11,12]. Fixed weir type design and pneumatic backpressure weir control type designs will be evaluated during the operational phase of LMC. This will enable standardizing the design for future plants.



### **Solvent Recovery Systems**

Alkali wash can remove only the TBP degradation products and is not effective for removing diluent degradation products. R&D work is being carried out for vacuum solvent distillation studies for removing such degradation products. This step will significantly increase the effective solvent usage. This process will be implemented in PFRP.

### **Remote Handling Systems**

The development of remote handling devices is of paramount importance in FBR reprocessing plants, as it is extremely difficult to design these plants for direct maintenance, because of the high  $\alpha$  and  $\gamma$  contamination. Many gadgets are required which are amenable for operation using master slave manipulators. Indigenisation of  $\alpha$ -tight transfer system and master slave manipulator (MSM) booting are in progress. Large-scale remotisation is required to reduce the manual intervention in sampling and analysis, decontamination, inspection and maintenance.

### **Development of Measurement, Control and Monitoring Systems**

#### ***Metering of Active Process Liquid Streams***

The analysis of the flowsheet proposed to be used for reprocessing FBTR fuel indicates that more than  $\pm 10\%$  variations in flow rates to the solvent contactors will result in Pu loss or inadequate stripping. This kind of precision in flow rate is difficult with airlift pumps. Constant volume feeders are being developed at IGCAR, which can provide exact metering, independent of other process loads or variations in densities.

#### ***Online Monitoring of Raffinate and Product for Pu and Fission Products***

There is a long felt need for online or quick process control analysis of Pu in raffinate and loaded organic stream. This will increase throughput of the plant by reducing recycle requirement and also reduce the waste volume and the hold up volumes. The on-line measurement of low levels of Pu in the raffinate stream with high gamma background will

pose a challenge. R & D work has been initiated in this area.

### ***Hull Monitoring***

The determination of Pu in the hulls is difficult because of high  $^{60}\text{Co}$  background. A method using  $^{144}\text{Ce}$  is planned to trace the Pu in the hull in LMC. Neutron interrogation techniques are under development at IGCAR in collaboration with CAT, Indore. This system will be deployed in FRFRP.

### ***Inspection Techniques***

Dissolver, evaporator and high active liquid waste storage tanks are susceptible to corrosion failures. Inspection of LMC dissolver is proposed to be carried out with immersion ultrasonic technique for wall thinning measurement and laser triangulation for inner surface profiling. Once they are demonstrated in LMC, the equipment for FRFRP will be designed incorporating provisions for these measurements. Visual inspection techniques with suitable robots will be tried in RDL waste tank farm. Suitable gadgets are under development for inspection of evaporators of PFRP.

### ***Fluid Control Devices***

Fluidic devices like reverse flow diverters and vortex diodes are being developed for possible deployment in future reprocessing plants in the active liquid pumping as these devices do not require any maintenance.

### ***Material Development***

AISI type 304L austenitic stainless steel will continue to be the main material of construction for the FBR reprocessing and waste management plants. In order to improve the performance of this material, IGCAR and MIDHANI, Hyderabad are working towards optimizing the specification of tramp elements and carbon. Different grades of stainless steels produced under this collaboration, as well as those available in the commercial market, are being tested in the nitric acid loop at IGCAR. This will yield suitable guidelines for specifying the stainless steel grades and also welding procedures for future reprocessing plants.

Development of corrosion resistant alloys like Ti-5%Ta-1.8%Nb for dissolver and evaporator are

also being undertaken. Dissimilar metal joining for Ti-SS has been already developed and deployed [13].

Design of large capacity FBR fuel reprocessing plants, such as PFRP, for aqueous process flowsheet is challenging due to criticality considerations. Modular design, with low Pu inventory in each module, is economically unviable. Design of storage tanks and process vessels with poison tubes (containing materials such as gadolinium) is one option. Another option is cladding the stainless steel vessels with cadmium or borated coatings. Preliminary criticality estimates indicate that there is substantial reduction in the neutron population in a typical design based on the poison incorporation in tanks and vessels.

### **Waste Management**

The wastes generated from the fast reactor fuel cycle are almost similar to that of thermal reactor fuel cycle. The concern of wastes comes from the amount of actinides that go into these wastes, requiring special handling methods. Another concern is the requirement of handling short cooled fuels, which increases the waste volume. An integrated approach is required in optimizing the cooling times.

Though the quantity of solid wastes generated from the reprocessing facilities are low, special techniques have to be developed to treat them, as they contain  $\alpha$ . Waste volume reduction techniques, such as hull compaction and packaging, should be demonstrated for FBTR wastes before adapting these techniques for PFRP. The present waste assaying techniques need refinement to make them more accurate.

Vitrification of high level liquid wastes of FBR fuels require new glass formulations in accordance with the spectrum of fission products and the dissolver residues. Developmental work is already under way in BARC for the simulated composition of FBR wastes.

### **Lead Mini Cell (LMC)**

The pilot plant LMC located in the Reprocessing Development Laboratory (RDL) is in

the advanced stage of commissioning. The objectives of this plant are the following:

- Demonstration of the operation and maintenance of chopper, dissolver, centrifuge, centrifugal extractors and partitioning contactors
- Demonstration, testing and establishing the process flowsheet parameters with respect to Pu and U recovery, decontamination, solvent stability, third phase formation
- Establish optimum cooling times by progressively operating the plant with spent fuels of higher burnups and shorter cooling times
- Demonstration of dissolution process with respect to completion of dissolution and destruction of dissolved organics
- Demonstration of remote handling devices like sampling, sample conveying, incell crane etc., as well as remote analytical devices
- Demonstration of PLC based systems for plant operation and control

### **General Description**

The LMC facility comprises of lead shielded hot cell (Fig. 1) with 250 mm or 200 mm thick lead shielding (depending upon the  $\beta$ ,  $\gamma$  radioactivity in different zones). An  $\alpha$ -tight stainless steel containment box is housed inside the lead shielding. The cell is provided with radiation-shielding windows and articulated arm type master slave manipulators to facilitate remote operation and maintenance of different equipments and systems. An  $\alpha$ -tight blister box provides access for direct maintenance of small gadgets, which can be brought out of the cell. Specially designed tanks have been installed to store solutions containing Pu to avoid criticality. About 2 km of intricate stainless steel piping involving 3000 bends and 2000 X-radiography joints has been successfully completed within the limited area of the facility. Welding of pipes and joints in an area of high density of piping was a challenging task for the welders. About 35 process vessels and 30 equipments are installed in this compact hot cell. Since large

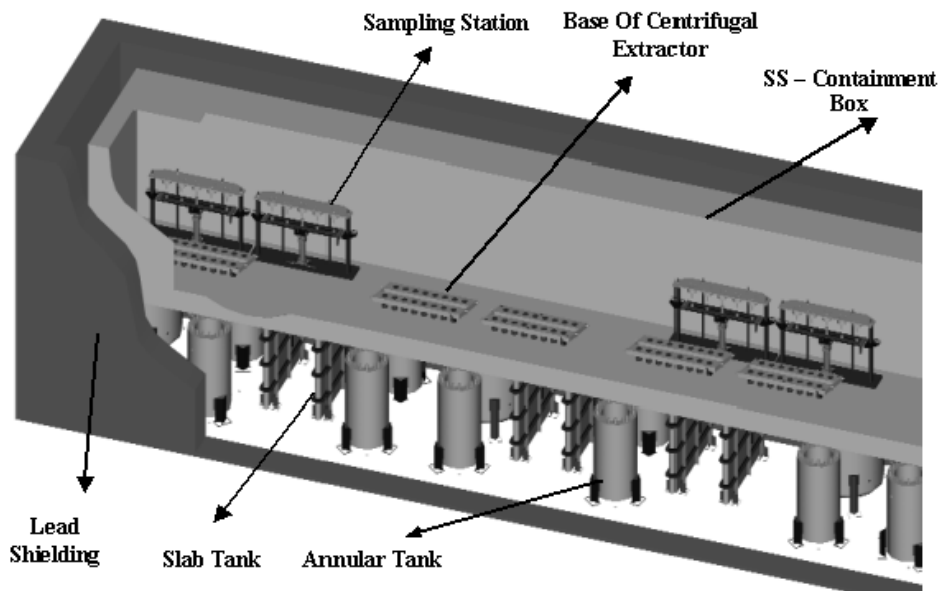


Fig. 1 Lead mini cell (cut away view)

concentrations of plutonium are handled, the hot cell facility is designed with criticality control measures.

LMC is designed on a no-direct maintenance concept. Several remote handling systems and gadgets are being installed. One example is an incell crane of 50 kg capacity, in which all the drives are brought out from the cell to an  $\alpha$ -tight glove box. The incell crane is made of austenitic stainless steel material and can be maintained without breaking  $\alpha$  containment. Sampling devices, which can be operated and maintained remotely are also installed. For carrying out remote operations, improved articulated arm type master slave manipulators (MSM) are used with a pay load of 5 kg and 7 degrees of motion. The arm is equipped with an airtight tong (RRT) which can be remotely removed. The RRT can be fitted into different types of jaws remotely for different operation.

#### Process Description

The process (Fig. 2) used in the plant is an adapted version of PUREX based on chop leach process. The fuel pins will be dismantled in RML hot cells and loaded in magazines. These magazines will be kept in special  $\alpha$ -tight containers and be transported to LMC in shielded casks. The fuel will

be chopped in the single pin chopper and then dissolved in an electrolytic dissolver. After feed clarification and three cycles of solvent extraction, pure Pu and U are separated. The third cycle has a facility for partitioning and electrolytic in-situ partitioning will be tested in this facility.

The high active aqueous raffinate waste and the organic wastes from this plant will be stored in stainless steel tanks, from where they will be pumped to FRFRP waste storage tanks, and then to WIP for final disposal.

#### Fast Reactor Fuel Reprocessing Plant (FRFRP)

FRFRP is a plant commensurate with the capacity of FBTR discharge. This plant has been located alongside the Kalpakkam Reprocessing Plant (KARP) to utilise the common service facilities like chiller unit, boiler, compressed air, etc.

The objectives of FRFRP are the following:

- Reprocessing of FBTR spent fuel on a regular basis
- Waste volume reduction, acid recovery and reuse

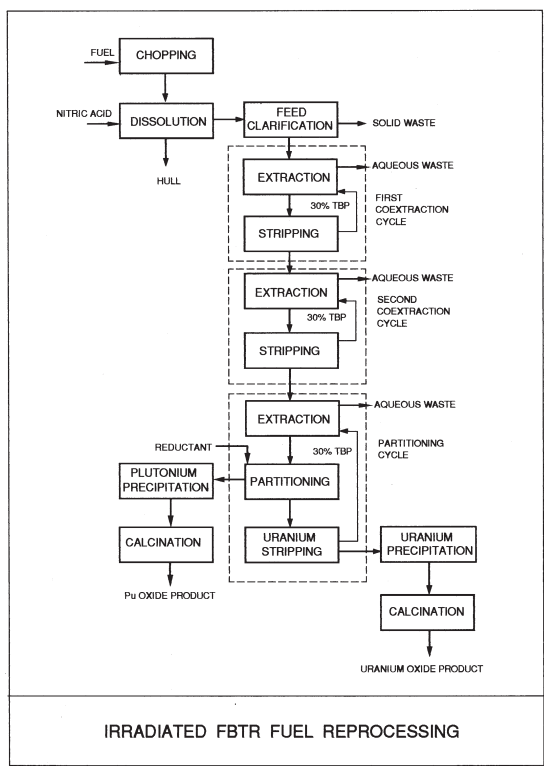


Fig. 2 Process flow sheet

- Demonstration of laser dismantling of fuel subassemblies, continuous precipitator for Pu/U reconversion and robotic based sampling systems
- Demonstration of improved construction materials for dissolver and evaporators
- Demonstration of techniques for on-line monitoring of Pu as well as selected fission products in process streams and monitoring of leached hulls
- Demonstration of PFBR oxide fuel reprocessing
- Evolving in-service inspection techniques for fuel reprocessing plants

### General Description

The design philosophy of the plant is based on remote operation and maintenance. The layout of the cells is such that all equipment requiring similar operation and maintenance aids are located together. There is extensive use of MSMs in this plant. Another important design improvement is the integration of the sampling system with the cell with provision for insitu process analysis. This will obviate the need for transporting high active solutions to the control laboratory. The manrem radiation exposure is also reduced by providing automated remote sampling systems.

### Process Description

The process flowsheet will be similar to the one used in LMC. The process parameters will be based on the operating experiences in LMC. FRFRP will have certain additional features like waste volume reduction and solvent washing for reuse. The aqueous raffinate from the three cycles are concentrated by evaporation and sent to high active waste storage tanks. To reduce the waste volume, acid killing is adopted. Solvent wash cycle is also planned, to remove the degraded organic products from the solvent before recycle. Centrifuges will be used for handling loaded organics to improve the decontamination factor. Acid recovery systems have been introduced to reduce the volume of waste generated.

### PFBR Fuel Reprocessing Plant (PFRRP)

As the maximum Pu content in any subassembly of PFBR is expected to be not more than 28%, the reprocessing plant will be designed for that concentration from the criticality point of view.

The flowsheet will be similar to the one that will be demonstrated in FRFRP. Certain additional features to be introduced in the plant are as follows:

- Provisions for transuranium/Pu recovery from raffinate streams
- Improved flow control of feed streams for better DF per cycle
- On-line accounting of Pu in various areas of the plant and on-line/in-line corrosion monitoring

- Experimental cell for Pyrometallurgical/Pyrochemical reprocessing of mixed oxide fuels

PFBR fuel is a mixed oxide, whereas FBTR uses a mixed carbide with high Pu content. The head end operations for PFBR fuel will be tested at FRFRP where an exclusive chopper and a dissolver will be installed for this purpose. After head end operations, the processing will be carried out with the same process equipment as the ones used for FBTR fuel processing on campaign basis. This strategy will not only enable qualification of the chopper and the dissolution process and equipment for PFRP but also will enable processing the initial discharge of PFBR.

### **Beyond PFRP**

There are some areas where development efforts will be required on a longer time scale, extending beyond PFRP design phase. They are basically aimed at addressing criticality considerations, reducing the transuranium (TRU) elements in the waste and waste immobilisation.

Criticality problems can best be overcome by avoiding moderators in the processing step. Pyrochemical processing appears to be the best bet for this purpose [14]. This simultaneously gives a solution to the recycling of TRUs also. But since these processes are conducted at high temperatures, the development of this complex process should be undertaken parallelly with other aqueous processes. Thus, in future, we will have options based on our own experience.

### **Non-aqueous Processes**

Pyroelectrochemical reprocessing using molten salts has been successfully applied to oxide fuel (RIAR, Russia), metal fuel (Argonne National Laboratory, US) and nitride fuel (JAERI, Japan) with suitable modifications.

The benefits include high chemical stability of salts, absence of neutron moderator, very low volume of high level liquid waste and final products ready for fuel manufacture

At IGCAR, a beginning has been made in this field with laboratory studies [15] on the separation of

Zr, Ce, Mo and Pd from U and on the recovery of UC by molten salt electro refining.

### **Aqueous Processes**

The draw back of TBP for application to the high Pu bearing fuels of FBRs is the possible formation of the second organic phase (often referred as third phase) at high Pu concentrations caused by the limited solubility of TBP-Pu solvates in diluent. Work at IGCAR has indicated that solvates with trialkyl phosphates having longer carbon chain have better solubility. Amides are also reported as alternate extractants for reprocessing. The extractant Octyl phenyl CMPO and selected diamides have also been identified as extractant systems for recovery of minor actinides from the high level waste generated in Purex process. The development of these extractants has been carried out at BARC. Further efforts are required to integrate the processes for minor actinide recovery with the Purex process in order to recycle minor actinides.

### **Waste Immobilisation**

The primary aim of the designer is the reduction of volume of vitrified HLLW waste and increase in the stability of the matrix. Presently the matrices considered are glass and SYNROC. Among glass matrix, boro-silicate glass is preferred over phosphate glass due to less corrosion problems though the formation temperature is low in the later. In glass matrix, removal of fission products, particularly Cs and Sr is required for maximising the loading. Cs and Sr are heat-generating radionuclides and higher loading of these elements leads to higher and unacceptable central temperature in the final glass waste. Cs and Sr will have recovery value as heat sources or can be immobilised separately as a ceramic waste form. SYNROC, a titanate ceramic consisting of zirconolite ( $\text{CaZrTi}_2\text{O}_7$ ), hollandite ( $\text{BaAl}_2\text{Ti}_6\text{O}_{16}$ ), perovskite ( $\text{CaTiO}_3$ ) and the titanium oxide ( $\text{Ti}_n\text{O}_{2n-1}$ ) seems to offer low leachability, excellent long term stability, and increased loading per unit mass, as compared to glasses. However more studies are required to prove these properties and also develop the technology for the production of SYNROC in large quantities. This work is being pursued in a systematic manner by active collaboration between BARC, IGCAR and other national labs of repute.



Studies concerning partitioning, transmutation and specific conditioning of the main long lived radionuclides present in high active waste have been very encouraging in reducing the long term toxicity of the TRU elements and these should be pursued with the purpose of exploitation in FBR fuel reprocessing plants. These studies are especially relevant to the fast reactor fuel cycle in view of the fact that fast reactors can be effectively used to “burn” the minor actinides.

### Conclusion

The cost of reprocessing and waste management constitutes around 25% of the cost of power sold from a typical FBR plant of 500 MWe capacity. In this, the cost due to the interest on capital constitutes the major component. Substantial reduction in the cost of power could be achieved if the capital cost and the construction time of the plants are reduced. Also, the development of high performance fuels and core components will reduce the processing requirement both at the back and front end of the fuel cycle, which will reduce the cost of power further. As the Indian power planners have envisaged significant contribution from the fast reactors in the coming decades, systematic development and management of this technology will help in fulfilling the objectives of abundant power on a cost effective basis.

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# Fast Breeder Reactor Safety



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## Introduction

In nuclear reactors, energy is released due to fission of heavy nuclides like isotopes of uranium and plutonium. The fission products that are released in the process are radioactive. Multiple barriers (fuel clad, primary heat transport boundary and secondary containment) are provided between fission products and the environment. Potential of accidental release of these materials exist in any nuclear reactor. The basic objective of reactor safety is to prevent accidents and if they occur, mitigate their impact on operating personnel, members of the public and the environment. Indeed, the objective of reactor safety is also to protect the plant investment.

Fast Breeder Reactors (FBR) are the reactors that make use of fast (energy in keV region) neutrons for causing fission of atoms. This enables better breeding of fissile material along with producing the power. Plutonium-uranium fuel is used in fast reactors – again for better breeding considerations. Since there is no moderator in the core, the core is small in fast reactors. It results in higher power density in the core. In order to remove this heat from small volume, a coolant with very good heat transfer properties and minimum neutron moderating characteristics is needed. So almost, in all FBR, liquid metal, sodium is used as coolant. Thus fast reactor with Pu-U fuel cycle and sodium as coolant have the following safety related characteristics:

- Short prompt neutron life time because fission chain reactions are maintained by fast neutrons in fast reactors. This would result in rapid power buildup in case the reactor attains super prompt criticality.
- Small delayed neutron fraction of plutonium enriched fuel. This results in lower margin to super prompt criticality.
- All the reactivity coefficients are not negative, particularly in medium sized and large sized reactors, the sodium and structural material (SS), reactivity coefficients are positive. The coefficients also have different time constants. This may lead to stability problems.
- Reactor core is not in the most reactive configuration, and hence there is potential of its acquiring more reactive configuration particularly under conditions of core melting.
- The sodium void worth of reactivity is positive for large reactors and therefore, reactivity-initiated power transients may occur on removal of sodium from the core, due to sodium heating on flow reduction or transformation of superheated liquid sodium into sodium vapor or sodium vaporization in molten fuel coolant thermal interaction or gas passing through the core.

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- Molten fuel-coolant interaction may also result in conversion of large passive thermal energy of fuel into mechanical work in case of core meltdown accident.
- Sodium is a very reactive material and catches fire when it comes in contact with air and reacts violently with water.

The above characteristics raises doubt about the intrinsic safety of FBRs. However, as explained in section 2, these characteristics are not a matter of concern at all. In fact certain design features like high thermal to electrical conversion, choice of structural materials, pool concept and basic fissile material breeding, result in unique intrinsic safety in fast reactors against accident, ease in operation and less environmental pollution. These are discussed in section 3. The safety design that is adopted to provide safety during normal operation and accident is discussed in section 4. Section 5 provides conclusions.

#### **Safety Related Characteristics**

##### ***Short Prompt Neutron Lifetime and Small Delayed Neutron Fraction***

In fast reactors, prompt neutron life time is of the order of a fraction of a microsecond which is approximately 2 to 3 order of magnitude shorter compared to thermal reactors. Thus, super-prompt critical transients will be extremely fast in fast reactors. However, the Doppler coefficient of reactivity as a feedback mechanism is sufficiently high, negative and promptly available in fast reactors and is able to quench the super prompt critical excursions. This has been shown [1] during a planned experiment in SEFOR reactor in USA, where a power transient created by positive reactivity got arrested by Doppler coefficient of reactivity of the reactor. Moreover, reactor kinetics in sub-prompt critical regime (reactivity  $< 0.8\beta$ ) is governed solely by delayed neutrons and the reactor kinetics behavior is independent of prompt neutrons. The delayed neutron fraction in fast reactors is approximately half of the value for thermal reactors and hence the small value of dollar unit of reactivity. However in fast reactors, the reactivity changes associated with burnup and changes in temperature and power, are also small. There is no xenon

poisoning in FBRs. This results in relatively lower reactivity worth requirement for control rods. Thus, lower amount of reactivity is added for the changes in temperature and/or power and uncontrolled withdrawal of a control rod. Operating experience in the world of about 200 reactor years, including reactors of the size of 1200 MWe has proved that short neutron life time and small delayed neutron fraction are of no concern and reactors can be controlled and operated quite safely.

#### ***Reactor Stability***

Fast reactors are designed with overall negative power coefficient of reactivity, and fuel pin size being small and sodium with good heat transfer properties being used as coolant, the involved heat transfer time constants are small and fast reactors are stable with respect to reactivity fluctuations. Nevertheless, the EBR-I core reflected instabilities arising from the positive power coefficient of reactivity [2]. Initially, it was thought that the instability arise due to positive Doppler coefficient of fissile material. However, the detailed analysis of the incident indicated that the movement of fuel due to bowing of fuel subassemblies contributed to the positive power coefficient of reactivity [2]. Subsequently, the mechanical design of the core was changed by providing spacer pads between fuel subassemblies at appropriate locations. Thus, the EBR-I type instabilities are ruled out in the present day fast reactors. In fact, in the case of incident of loss of flow, the spacer pads get heated due to coolant temperature rise and effect the overall outward radial movement of subassemblies. This results in negative reactivity in the system. This is considered as an important mechanism for passive shutdown of fast reactors for transient under-cooling incidents.

#### ***Reactor Core Configuration***

Fast reactor cores are not in the most reactive configuration. This implies that under conditions such as fuel melting/slumping, the core can acquire more reactivity. However, a large number of experiments which have been carried out in facilities like CABRI in France and TREAT in USA and in other countries to study the behavior of molten fuel under melting conditions have shown that this is not a serious problem. In these tests [3,4], the fuel sweeps out of the core and results in reactivity

reduction. Moreover, sufficient design measures are provided in the design (see section 4) to prevent melting of fuel/breach in clad and shutdown of the reactor on detection of any fault leading to core over heating or core under-cooling. Further, it is observed that in the case of unprotected (shutdown system failing) Transient Overpower (TOP) incidents, fuel pin breaches in the upper part of the core and the molten fuel coming out of the pin sweeps out of the core due to benign fuel coolant interaction, fission gas pressure and coolant flow and results in reactivity reduction [5]. For incidents like unprotected LOF (Loss of Flow), molten fuel gets swept out of the core due to shearing force of sodium vapor and/or fission gas released at high pressure, again leading to reactivity reduction. It is also observed [6,7] that radial expansion of the core due to spatial pad heating and differential control rod expansion, leads to reduction in reactivity and consequently reduction in the power of the reactor without core melting or sodium boiling. Fast reactor accidents (TOP & LOF) have been analysed by various models [8,9] and it has been shown that energy release in these accidents is small. Pintolite detonated experiments conducted at TBRL Chandigarh [10] on PFBR scaled down model (1/13) have shown that for the conservative energy release calculated, the main vessel and top shield would remain intact and the integrity of pump, intermediate Heat Exchanger and Decay Heat Exchanger will not be jeopardized under accident conditions.

#### ***Positive Sodium Void Worth***

On loss of sodium due to sodium boiling or gas passing through the core, the neutron spectrum hardens. This leads to larger neutron leakage (negative reactivity effect), increased fast fission from fertile material (positive effect) and emission of more neutrons per neutron absorbed (positive effect). In a medium sized or larger sized fast reactor with homogeneous core, it results in a net positive sodium void worth. However, the homogeneous core design has been adopted world wide due to better breeding ratio and ease in operation and control. Suitable design provisions are made so that no gas passes through the core. The core temperature and the reactivity are monitored to ensure sodium boiling detection and arrest of its propagation. Even if shutdown system fails and LOF or TOP accident

takes place, accident analysis shows that as long as sodium void worth is in the range of 4 to 6 \$ or less, the energy release in the core depends on reactivity changes due to fuel movement and not on the sodium void worth [11,12]. Thus positive sodium void worth of 4 to 6 \$ is acceptable in the present day fast reactor design.

#### ***Sodium Liquid Superheat***

Laboratory experiments [13] have shown that the onset of moiling in liquid sodium can occur at temperatures significantly higher than the boiling point (superheating). This phenomenon can lead to sudden evaporation of sodium resulting in reactivity changes. However, the processes necessary to attain such levels of super heat are now well understood and there is essentially universal agreement that such conditions cannot exist in an operating fast reactor. Of the various mechanisms identified, the presence of small quantities of inert gas absorbed in sodium coolant from argon cover gas, provides sufficient nucleation sites to prevent significant superheat in sodium [14,15].

#### ***Fuel Coolant Interaction***

A large amount of R&D has gone into studying the nature of fuel coolant interaction (FCI) in fast reactors. It may be noted that an energetic FCI in the central part of the core can lead to large reactivity addition rates and in turn in a severe transient over power accident (TOPA). Similarly FCI at the end of disassembly of the core, following core meltdown accident may translate large amount of thermal energy into mechanical energy. Tests conducted in reactors and out- of- pile loops have shown that FCI is normally benign under reactor conditions[15]. For an energetic FCI, the fuel coolant interaction temperature should be larger than the spontaneous nucleation temperature of the coolant [5]. For oxide fuelled fast reactors, the said FCI temperature is only 1230 K [5] while the coolant nucleation temperature is 2300 K. Thus an energetic FCI in fast reactors is ruled out.

#### ***Sodium Coolant - A Reactive Material***

Sodium is a reactive material and catches fire if it comes in contact with air and reacts violently if it comes in contact with water. Therefore, specific design provisions are needed to eliminate or reduce

the risk from sodium fire and sodium water reaction. To reduce the probability of sodium fire due to leak, inert gas blankets and double walled pipe construction is provided in all pipelines inside RCB. The main vessel is normally provided with another vessel called safety vessel. Provision is also made to drain sodium into the dump tank if a leak develops anywhere in the system. Ability to detect very small leaks (as small as 100 g/h) enables adoption of effective methods to limit the consequences of leaks from sodium systems. Sodium-water reaction can occur in steam generators due to leak in tubes carrying water at a high pressure. The corrosive sodium hydroxide promotes wastage of adjoining tubes. Therefore, provisions are made to monitor continuously the steam generator to detect the leak before it grows and to take measures to isolate sodium and water to minimize the quantities reacting in the event of leak. It has been possible to show by experiments and calculations that sodium-water reaction in steam generators can be handled safely without damage to adjoining equipment or harm to operating personnel.

#### **Unique Safety Features**

Some of the unique favourable operational features of FBR due to factors like temperature and pressure range of operation, breeding, selection of material and design concepts are listed below.

#### **Low Thermal Pollution**

Thermodynamic efficiency of FBR (~ 40 %) is significantly higher than that of thermal reactors (~30 %). Therefore, thermal pollution of the environment is less for FBR. The heat rejected to the environment for 1 GWe produced is 1.5 GWt for FBR against 2.4 GWt in thermal reactors.

#### **Low Radiological Impact**

During normal operation of fast reactors, the radioactivity release to the environment and the exposure of operating personnel is significantly lower than that from thermal reactors. This is because of good fuel performance (less pin failures), radioactivity contained within primary sodium circuit, leak tightness of sodium circuit ensured by safety provisions such as guard vessel and pipes, minimum maintenance and inert argon cover gas above sodium free level. Long operating experience

of PHENIX, BN-350 and BN-600 reactors have shown that exposure of personnel and emission from these plants are two orders of magnitude lower than those from the thermal reactors of comparable capacity. Fast reactors make full utilization of natural uranium and thorium and energy extracted from fuel is 60 to 80 times more than the thermal reactors. Consequently, the environmental impact of operations like mining and subsequent processing is correspondingly lower in fast reactors [16]. Further, in FBR, less fuel is burnt to produce a unit of electricity. This results in reduction of radioactive waste generation per unit of electricity produced by about 30% compared to the thermal reactors.

#### **Easy Reactor Control**

In FBR, temperature and power coefficient of reactivity are negative. In the event of disturbances in primary and secondary sodium flow or feed water flow, the reactor stabilizes to a new power level even without the corrective action of the operator. This has been amply proved by successful and long operation of reactors like PHENIX, BN-350 and BN-600. Also FBTR operation has been very stable without any automatic control of parameters like power and sodium flow. Therefore, it is easy to control the reactor.

#### **Efficient Sodium Coolant**

Sodium as coolant provides many attractive safety features to FBR. Large margin between the normal operating sodium temperature (845 K) and the boiling point (1160 K) of sodium can accommodate significant temperature rise in the event of mismatch of heat generation and heat removal, without the system being pressurized. Maximum pressure in the primary sodium circuit during normal operation is less than 1 MPa. Hence all the faults associated with the de-pressurisation of coolant are absent in FBR. The primary stresses in vessels and piping remain low leading to low probability of failure.

#### **Easy Natural Convection**

High thermal conductivity, low viscosity and large difference between the temperature of hot sodium at 820 K and ambient air at 310 K, coupled with significant variation of sodium density with temperature, permit decay heat removal through



natural convection mode. This has been demonstrated in all the FBR by tests.

### ***Pool Type Concept***

Pool type concept provides large thermal inertia and hence more time to the operator to act in case of exigencies during reactor operation. The core is configured with adequate shielding to limit radioactivity of secondary sodium and also to reduce the fluence on the structural components such as grid plate, core cover plate and main vessel (MV) ensuring low material property degradation on account of radiation. The MV has no nozzle penetrations and thus offers high structural reliability. Simple shape permits In Service Inspection of the vessel to assess its structural reliability.

### ***Material Choice***

The MV is made of highly ductile AISI 326 LN material for which leak before break criteria applies. The incorporation of a safety vessel around the MV ensures continued core cooling under all conditions, thus removing Loss of Coolant type of accident out of the design basis events. There is also continuous monitoring of leak from main and safety vessels.

### ***Safety Aspects in Design***

The following safety design measures are provided for proper control of the reactor, safety against operational transients, to prevent occurrence of accident and to mitigate their consequences.

### ***Design Approach***

Components and systems are classified depending on their nuclear safety significance and are designed as per established standards, codes and guides issued by regulatory bodies like NRC in USA and AERB in India. Systems are also designed as per seismic categorization and applicable codes. Sodium components are designed for both, Operating Base Earthquake (OBE) and Safe Shutdown Earthquake (SSE). External events design inputs like earthquake, wind speed and design basis flood etc. are determined based on regulatory bodies requirements. Quality assurance measures are followed in design, construction, commissioning and operation. In service inspection provisions are provided for all the important safety and safety

related components and systems. FBR technology has matured over the years. The operating experience of about 200 reactor years exists today. The feedback from this operating experience goes in the design of present day FBRs. In addition, design validation is done experimentally in the engineering laboratories and design is analyzed and improved by analyzing all the local and global conceivable events.

### ***Negative Reactivity Coefficients***

As mentioned above, the temperature and power coefficient of reactivity are designed to be negative so that any off normal increase in temperature or power leads to a reduction in reactivity and the consequent reduction in power. The expansion of coolant and structural steel result in small positive reactivities which are compensated by negative and prompt reactivity effects like Doppler and fuel expansion. There are also negative but slow reactivity feedback resulting from grid plate expansion and spacer pad expansion and differential control rod expansion which tend to shutdown the reactor for transient under cooling incidents. The negative reactivity coefficients provide inherent safety to the reactor. Fig. 3 provides some results of SPX1 analysis [7].

### ***Radial Entry Sleeves***

To prevent total flow blockage in fuel and blanket subassemblies, multiple radial openings in sleeves in grid plate and in the foot of all the subassemblies are provided.

### ***Inertia on Primary Pump***

Flywheel is provided on the pump to achieve gradual flow coast down during off-site power failure and pump trip. This prevents fuel pin clad damage.

### ***Emergency Power Supply***

Emergency power supply from diesel generators backed up by battery are also provided to the primary pumps.

### ***Core Monitoring***

The objective of core monitoring is to detect any fault and ensure safety of the core. Neutron

detectors are provided to monitor the power and provide signals for safety action by SCRAM on parameters like linear power, period and reactivity. This is to protect the reactor from reactivity initiated transients. The SCRAM of the reactor on reactivity protects the reactor against the growth of small local effects like boiling and fuel melting. Neutron monitoring system is triplicated to permit 2/3 logic. Monitoring of sodium temperature is done by providing two thermocouples at the outlet of each fuel SA. SCRAM action is initiated on individual subassembly temperature rise for protection against local fault. SCRAM action is also initiated if average temperature of all the subassemblies and mean core temperature rises above certain thresholds. The flow delivered by the primary sodium pump is measured and power to flow ratio is also monitored. This is another parameter that takes care of events leading to global mismatch of power and heat removal. Detection of fuel pin failure is carried out by monitoring the cover gas activity as well as the delayed neutrons in the primary coolant circuit. These provisions ensure that, there are at least two usually diverse safety parameters to protect the reactor against any event.

#### ***Shutdown system (SDS)***

On detection of any abnormality in the reactor, shutdown of the reactor is assured by two independent, fast acting, diverse shutdown systems. Reactor shutdown is accomplished independently by both the systems by free fall of neutron absorber rods even when one rod remain stuck. The response time shutdown system is prompt enough to take care of reactivity transients leading to a few dollars per second reactivity addition rates. The SDS is designed to function even in a distorted core that may result due to seismic conditions. The unreliability of SDS is normally less than  $10^{-6}$  per reactor year (ry).

#### ***Decay Heat Removal System (DHRS)***

Heat is generated in fuel due to decay of fission products even after shutdown of the reactor. It is about 1.5 % and 0.7 % of nominal power after 1 h and 1 d respectively after reactor shutdown. Therefore, to maintain core integrity after reactor shutdown, the decay heat is to be removed. Thus, Decay Heat Removal (DHR) system is another important safety system which needs very high reliability. Sufficient

redundancy and diversity is provided in decay heat removal also.

The unreliability of each shutdown system and decay heat removal system is less than  $10^{-6}$ /ry such that core damage frequency is less than  $10^{-6}$ /ry.

#### ***Plant Layout***

Safety considerations in plant layout include RCB, steam generator building and fuel building to be kept on common raft so as to minimize the risk of secondary sodium pipe leak and fuel handling untoward incidents in case of an earthquake event. Safety related buildings are kept outside the trajectory of turbine missile.

#### ***Radioactivity monitoring and Reactor Containment Building***

Potential sources of radioactivity in the plant are the fission products in the fuel, primary sodium, primary argon cover gas and washings from the spent fuel handling and component handling systems. Design provision are provided to avoid exposure of the operating personnel and public at site boundary in excess of acceptable dose limits. Primary containment is provided by main vessel and top shield boundaries. The Reactor Containment Building (RCB) provides the secondary containments. Even for severe accidents like core disruptive accident, RCB is designed such that it can withstand the pressure build up due to the sodium fire and the dose rates for the public at the site boundary are less than the acceptable limit prescribed by regulatory bodies.

#### ***Event Analysis***

All the conceivable events that can cause the increase of fuel, clad and coolant temperatures beyond their Design Safety Limits (DSL) are postulated and analyzed using computer codes validated against theoretical benchmark and experimental results. The events that have probability of occurrence more than  $10^{-6}$  per reactor year are called Design Basis Event (DBE). Based on the analysis of each DBE, it is ensured that two diverse SCRAM parameters are available to shutdown the reactor before any DSL is crossed. The response time of sensors, processing electronics delays and response time of electromagnet of

shutdown system are taken into account in the analysis.

The events that have probability of occurrence less than  $10^{-6}/\text{ry}$  are called Beyond Design Basis Events (BDBE). These incidents are analysed to get design input for RCB and working out emergency evacuation plants. BDBE events occur only if shutdown and decay heat removal system fail. It has been established that the energy release in these accidents is small and the primary containment (primary sodium boundary) and its vital internals like pump, intermediate and decay heat exchangers, remain intact. However, seals in the top shield may fail and, a small quantity of sodium can come in the RCB. The design of RCB takes care of pressure buildup in RCB due to sodium fire and the leakage of radioactive fission products into the environment.

### Risk Analysis

To provide a basis for quantitative comparison of risks from accidents, the risk is defined as [8],

$R = (\text{Probability of an accident}) \times (\text{magnitude of its consequences})$

If the probability of occurrence of an accident, it is  $P_i$  and  $C_i$  is the consequence in terms of monetary loss or fatality, then

$$R = \sum P_i C_i$$

Generally, single large accidents are less tolerated than the more small accidents (air crashes Vs traffic accidents). This kind of non-linearity is taken into account, by the following modified equation,

$$R = \sum P_i C_i^\gamma$$

where the constant  $\gamma$ , is normally taken as 1.2. Risk studies have also been conducted for fast reactors, i.e., CRBRP in USA [17] and SNR-300 in Germany [18] and both these studies have concluded that risks from FBR are less than those from LWR.

### Summary

Fast reactors have a number of good safety features that bring in them the intrinsic safety. Some initial fears about their safety have been assuaged. Choice of good design concepts and structural materials, use of established standards, codes and

guides in design, good quality assurance practices in design, construction, commissioning and operation and feedback from the operating experience of 200 reactor years add to the intrinsic safety of these reactors. Extensive core monitoring, highly reliable (unreliability  $< 10^{-6}/\text{ry}$ ) shutdown and decay heat removal systems and analysis of all the conceivable events assure that any fault if developed, will be terminated without crossing the design safety limits of fuel, coolant and structural material. Results of the analysis of beyond design basis events that have probability of occurrence less than  $10^{-6}/\text{ry}$  ensure incorporation of mitigative measures in the design of primary and secondary containment such that risk to the public is minimized. Risk analyses carried out in USA and Germany show that the risk from fast reactor is less than the thermal reactors.

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# Fast Breeder Reactor Economics



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## Introduction

For a large country, like India, long term energy security, mainly based on indigenous resources, is an important and inevitable need, arising from economic, global environmental and strategic considerations. These considerations will dictate the optimum composition of our energy mix. Considering the existing reserves of coal in India and rather meagre reserves of uranium, it remains a certainty that fast reactor based nuclear energy systems will have to be an important component of the Indian energy mix in the long term to meet an enhanced rate of energy consumption. To date, Fast Breeder Reactors (FBR) have been built and operated worldwide, ranging from research reactors of some tens of MW thermal output to the 1200 MWe Superphenix (SPX-1). The fast reactor has thus reached technological maturity with ~ 310 reactor years of operation worldwide. Its commercialisation vis-à-vis the established reactor systems such as PWRs and PHWRs will depend on its generating cost in fully developed form, with mature design and with the benefit of series production. For the success of fast reactors, efforts should be directed towards both reactor design and fuel cycle. The reactor has to combine safety with competitiveness. The fuel cycle, in particular reprocessing, has to be at an acceptable cost.

Despite the indisputable need for FBRs in India, it is worth noting that the 500 MWe pool type Prototype Fast Breeder Reactor (PFBR), whose

construction is to commence shortly, itself has to be economically attractive in the context of present governmental approach for investments.

## Economic Comparisons of Initial Designs

Economic comparison data is available from France and Russia. For Superphenix, a prototype 1200 MWe unit in France, the construction cost/kw of installed capacity has been reported to be around 2.5 times that of PWR operating in France [1]. In Russia, the BN-600, a 600 MWe unit, lagged the VVER-1000 unit by a factor of 1.55 on specific capital cost. Considering the differences in power and site location, this difference decreases to a factor of 1.2-1.3 [2]. In brief, the first experimental prototypes of fast reactors did not match light water thermal reactors in capital cost or in the unit energy cost.

## Economic Comparison of Present Designs

It is unfair and misleading to make comparison of the prototype FBRs with matured PWR units. The PWRs have benefited from many years of experience of construction and more importantly from the benefits of scale arising from batch production. On the other hand, Superphenix and BN-600 were the first of a kind and were built as single units.

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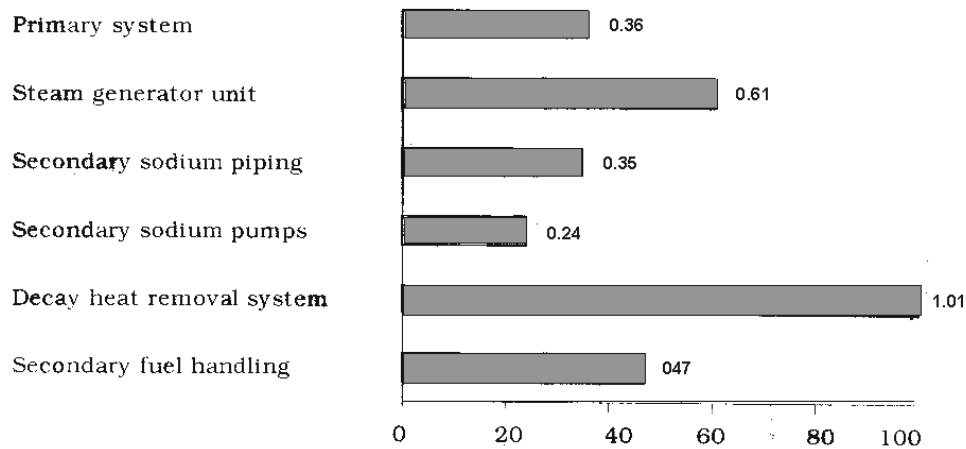


Fig. 1 EFR Vs Superphenix : Comparison of Specific Steel Weight In t/MWe

#### European Fast Reactor (EFR) vs Superphenix & Economics of EFR

In Europe, the construction of fast reactors and design studies for larger units in France, Germany and UK led to the development of the combined European expertise on fast reactor technology. As a result of this collaborative venture, design of European Fast Reactor (EFR) with generating capacity of 1500 MWe was pursued. The major objectives set for EFR were safety level comparable with that of future LWRs, and potentially competitive electricity generating costs compared to future LWRs. Construction and operating experience of FBRs, in particular Superphenix, provided a wealth of information allowing the simplification and optimisation of EFR. There has also been considerable progress of knowledge in structural mechanics and design rules. As a result, the design of EFR had achieved substantial investment cost reductions for the nuclear steam supply system as illustrated in Fig 1 which displays specific weight of steel in t/MWe employed for the main systems of Superphenix and EFR [3]. Specific weight is recognised as a sound concept for economic comparison of designs. The design of the core and fuel for high burn up is an effective way of reducing fast reactor fuel cycle costs as this reduces the annual fuel requirement. EFR core was contemplated to operate at target peak burn-up of 20 at % against 8 at % for Superphenix first core. In the

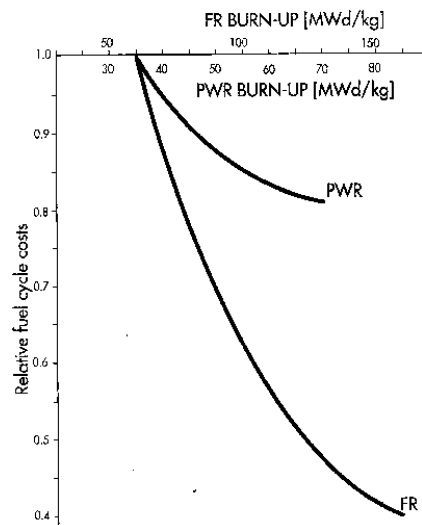


Fig. 2 Potential Fuel Cycle Cost Reduction by Increasing Average Burn Up In PWR &

case of PWR utilising low enriched uranium, increasing burn-up is accompanied by an additional cost arising due to higher enrichment requirement, thus partially off-setting the benefit from reduced annual fuel charges. The effect of variation in burn-up on the PWR and the FBR fuel cycle costs is shown in Fig 2 [3].

In brief, the approach to economic competitiveness of EFR was based on the following:

1. Investment cost saving due to
  - (a) simplicity and compactness of the design
  - (b) decreasing the number of safety graded systems while maintaining compliance with the required safety standards
  - (c) reduction in number of components and systems, reduced component weights, and building sizes
  - (d) Improvement in construction, manufacture, and erection methods to allow shortening of the construction time
2. Fuel cycle cost savings because of high burn-up

The adoption of a 3 pumps/6 IHX arrangement for the primary system for EFR against 4 pumps/8 IHX for SPX-1 combined with an improved in-vessel fuel transfer system resulted in remarkable reduction in main vessel diameter (17.2 m for EFR compared to 21 m for SPX-1) in spite of an increase in reactor thermal power by 20 % as compared to SPX-1.

The benefit of reduced component size and simplification in secondary sodium circuit layout has also influenced civil works. The volume of nuclear island buildings for EFR is lower than that of SPX-1.

The design exercise of EFR has definitively improved the economics in comparison to SPX-1, although it will still be a prototype in nature. As one can normally expect, a prototype is more expensive than a standard plant. There should be increasing benefits from series construction and learning effects. The studies for series construction for SPX-1 type indicate that the investment cost could be as low as 58 % of that of the prototype (Fig 3) [4]. For EFR, the savings corresponding to the elimination of its first-of-a-kind costs would be smaller than indicated by SPX-1. However, savings corresponding to series effects would apply to EFR. In addition, the effects of increasing the capacity of fuel fabrication and reprocessing facilities dedicated to FBR will contribute to reducing fuel cycle cost. The comparison of generating costs between EFR and PWR is shown in Fig 4. It shows that even as compared to the very efficient PWR (EPR), the

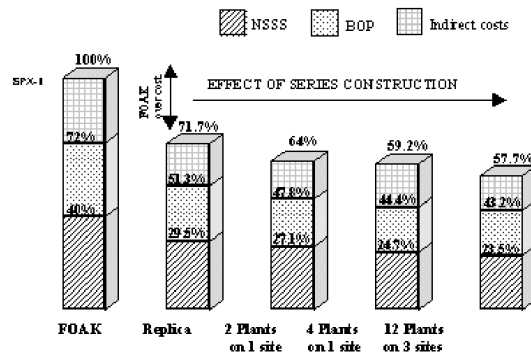


Fig. 3 FBR Investment Cost Reduction

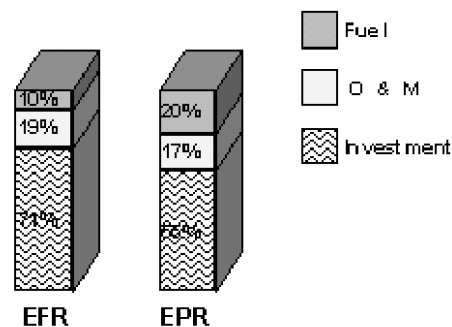


Fig. 4 EFR vs EPR Generation Cost Comparison (costs normalised to 100% for EFR)

series EFR is very close to achieving competitiveness.

#### Russia

In Russia, experience of BN-600 has been incorporated in designing the BN-800. The economic characteristics of the BN-800 design were improved in that the margins built into the BN-600 prototypes could be eliminated. The increase in power of the BN-800 reactor was accomplished with practically no increase in absolute material cost of the reactor [5]. The new BN-600M reactor has significantly lower material consumption on account of the use of fewer heat transfer loops and an integral (tank) steam generator configuration (see Table 1). The design of new generation of thermal reactors with improved safety characteristics has reduced the difference in the economic performance of fast and thermal reactors.

**TABLE 1. Comparative Characteristics of Fast Reactors [5]**

Characteristics	BN-600	BN-800	BN-600M
Thermal Power, MW	1470	2100	1520
Gross electrical output, MWe	613	793	647
Number of sodium loops	3	3	2
Material consumption in reactor, t/MW	13	10.5	10.4 (8.23)*

\* Figure in bracket correspond to the design with an integral steam generator  
Note: Specific material consumption for VVER-1000 is 8.44 t/MW

### **Japan**

For DFBR, a 600 MWe top entry loop type reactor, the initial target was 150 % of that of LWR based on the material quantities. The goal was achieved in the Phase 1 study in 1996 with estimated value of 146 points (100 points for 1000 MWe LWR). In the design study Phase 2, the target was to reduce it by 10 % or more. The target has been achieved by improvements in the design of reactor system like elimination of reactor vessel cooling system, reduction in thickness of reactor vessel, reduction in diameter of reactor vessel, compact secondary circuit by integration of electromagnetic pump, steam generator and cold trap in a single vessel, simplification in fuel handling arrangement and efficient layout of the nuclear buildings. In Phase 2 design, the cost has been brought down from 146 to 131 points [6]. If DFBR had simply been a scaled up version of 280 MWe Monju built at relative high capital cost, its capital cost would have been 3.5 times that of a LWR [7].

### **USA**

Fast reactor designs have been directed towards small and medium size modular units in view of difficulties confronting nuclear power in the United States. With time, the unit size has been changed from small to medium. General Electric has

changed the size of PRISM (Power Reactor Innovative Safe Module) in 1988 from 3 blocks of 421 MWe each with each block consisting of 3 reactors of 425 MWt capacity to SUPER-PRISM design in 1999 comprising of 3 blocks of 760 MWe each with each block consisting of 2 reactors of 1000 MWt capacity [8] (Multiple reactors with single turbine concept). The PRISM design has the basic features of factory fabrication of all components, accelerated learning through the modular concept, low development cost, shorter and less expensive route to commercialisation with significantly lower financial risk. Unit energy cost for S-PRISM is shown to be lower than monolithic design and the cost for series unit is competitive with alternative generating systems. The choice of USA for modular size is contrary to the approach followed in other countries pursuing FBR programmes.

### **Economics of PFBR**

Significant design modifications have been incorporated into the present design compared to the earlier design to improve economics without compromising safety taking benefits from the in-house studies and design improvements reported for foreign reactors. The number of heat transport loops has been reduced from 4 to 2. The design being detailed now has 2 primary pumps, 4 IHX and 2 secondary pumps. Sodium reheat is replaced by steam reheat with turbine retained as similar to a BHEL make for thermal power station. Total number of steam generators is 8 of integrated once through compared to 36 split up type with separate evaporator, superheater and reheater. The plant temperatures have been raised through improvements in elevated temperature design with steam temperature raised by 10 K and core outlet by 17 K. Reactor assembly has been made more compact with diameter of main vessel reduced to 12.9 m from 14 m. Plant layout is more compact with safety related buildings interconnected providing advantage of enhanced safety and economics. A rectangular shape has been selected for the Reactor Containment Building in place of circular shape because of low pressure in the building under core disruptive accident conditions. Reduction in number of sodium components and circuits has resulted in savings in electrical and instrumentation and control systems. Reduction in number of loops/ components

and advanced design of sodium pump, IHX and steam generator will result in reduced construction time, and capital cost. The specific capital cost and unit energy cost for PFBR are comparable to that of 540 MWe Indian PHWR.

### Economics in Indian FBRs beyond PFBR

PFBR, being prototype, will provide guidance for future design and construction. The cost of series of PFBR will itself be less than that of first-of-a-kind due to savings in construction time, twin or four units at a site, advantage of industrial economics (industry charges for steam generator per unit for example is lower with more number ordered). Balance of plant provides relative less scope for cost reduction as PFBR unit in that respect is not very much different from thermal power stations. It is expected that series PFBR on multiple units at a site would cost ~ 65 % of the first unit and would be built in ~ 5 years against 7 years targeted for first unit. Further, cost reductions are possible with new designs. It is likely that future units would be 1000 MWe and will facilitate cost reduction through economy of scale. Design modifications having a strong impact on NSSS economics will be studied in detail and suitably incorporated. Few examples are elimination of main vessel cooling resulting in reduction in diameter of main vessel, integrated primary sodium purification outside the main vessel, reduction in number of steam generators and with longer tube length of ~30 m against 23 m, reduction in design margins based on the initial operation of PFBR in terms of heat transfer area and pressure drop, raising design temperature and improvement in cycle  $\eta$ , elimination of operating basis earthquake from design, no specific design provisions for Beyond Design Basis Events, reduction in design margins through extensive R&D, and compact layout of components inside the nuclear island buildings in particular, to reduce volume of buildings. Design will be retained for single turbine as design with 2 TG is more expensive by ~ 30 % compared to single unit. The fuel burn-up target of 20 at % will be easily met by improvements in wrapper and clad material. The design life will be increased from 40 to 60 years to reduce depreciation portion in unit energy cost. The reactor of higher capacity of 1000 MWe under series will be targeted for 5 years construction time and is expected to provide unit energy cost ~ 50 %

that of PFBR under identical financial conditions. It is to be noted that economies of scale are very effective in reducing the generating cost of electricity by nuclear power plants and are essential in the strive for greater competitiveness. The economic studies for 3x500 MWe FBR units vs 1500 MWe FBR unit of same basic design features indicated that the unit energy cost of 500 MWe design variant would be ~ 50% more than the 1500 MWe design [9].

### Summary

Cost of initial fast reactors was high because of prototype nature. Detailed design studies in many countries have shown that potential for cost reduction is large for reactor as well as fuel cycle. One can hope that fast reactors will supply safe and economic source of energy, helping to assure a radiant future for all mankind.

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## Announcement

### DAE-BRNS Symposium on Nuclear and Radiochemistry (NUCAR 2003)

*The sixth biennial symposium "Nuclear and Radiochemistry" (NUCAR 2003) is being organised by the Board of Research in Nuclear Sciences (BRNS), Department of Atomic Energy at Bhabha Atomic Research Centre, Mumbai, during February 10-13, 2003. The objective of the symposium is to provide a forum for effective interaction among the scientists in the areas of Nuclear and Radiochemistry and Applications of Radioisotopes for the advancement of these disciplines. It is hoped that scientists engaged in research in these areas from National Laboratories, Universities and Research Institutes from India and abroad, will actively participate in large numbers.*

#### Scope

- Nuclear chemistry
- Chemistry of actinides and reactor materials
- Spectroscopy of actinides
- Radioisotope applications
- Chemistry of fission and activation products
- Radioanalytical chemistry
- Radioactivity in environment
- Nuclear instrumentation

*The scientific programme of the symposium will include invited talks by eminent scientists as well as contributed papers. An important feature of this symposium would be half a day special seminar on **APPLICATIONS OF ACCELERATORS INCLUDING ACCELERATOR DRIVEN SUBCRITICAL SYSTEMS, MEDICAL CYCLOTRONS, LOW ENERGY ACCELERATORS** etc.*

Last Date for Receipt of Manuscripts : October 31, 2002

#### Registration Fee

Rs.100/- for research scholars, delegates from National lab.  
Rs.250/- for delegates from PSUs  
Rs.500/- for other delegates

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