IGC Newsletter

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Inside

From the Director's Desk

Path ahead...

I consider it a privilege to have got the opportunity to steer this prestigious institution, of which I have been a part for the past forty years. This Centre has the legacy of able and visionary leaders providing direction to the Centre and I have benefited immensely having got the opportunity of interaction with them. Several years ago, when I stepped into Kalpakkam, it was a small and sleepy hamlet, where the construction activities relating to Madras Atomic Power Station and the then Reactor Research Centre were the main on-going activities. In the intervening years, it has grown into a world class nuclear community, with a large variety of high impact programmes in various domains. I have been nurtured by several leaders of the Centre and immensely benefited by their encouragement and leadership. We have travelled a long way towards achieving the present pride we hold, yet much more remains to be done towards reaching the goal of contributing to the energy security of this country by way of establishing robust sodium cooled fast breeder reactors and associated fuel cycle technologies on a large scale. In the times ahead, there would be continuity in pursuing the R&D towards achieving the goal of mastering the fast breeder reactors. In this article I would like to highlight the present status vis-a-vis the challenges ahead on several of the important activities. I would like to share with you my thoughts about the path ahead and the goals to be set for the Centre.

Fast Breeder Test Reactor

Fast Breeder Test Reactor completed one of its important missions of being a test-bed for fuel irradiation, when the test fuel of Prototype Fast Breeder Reactor was taken to a burn-up level of 112 GWd/t as against the target burn-up of 100 GWd/t. The test fuel sub-assembly has been discharged and is presently undergoing post-irradiation examination at Radiometallurgy Laboratory.

During 2010-11, the reactor power was also increased to a maximum of 20 MWt for the first time, with the turbine generator generating 3.5 MWe. The steam temperature was taken closer to the design value of 480˚C. Several experiments including measurement of flux of neutron and gamma radiation levels above the core and testing of high temperature fission counters for Prototype Fast Breeder Reactor were carried out.

Fast Breeder Test Reactor has seen the first failure of a MARK-I fuel pin in the month of February, 2011. It has been a learning exercise and I am satisfied to note that the sub-assembly containing the failed fuel pin was identified in a single fuel handling operation. The sub-assembly has been subsequently discharged. This has amply demonstrated the capability and maturity of O&M and technical services personnel of Fast Breeder Test Reactor. The analysis of the failed fuel sub-assembly with a burn-up of 148 GWd/t would provide valuable inputs towards understanding the behavior of the unique high plutonium content mixed carbide fuel.

We rededicate ourselves to achieve the target of operating the Fast Breeder Test Reactor with reduction in SCRAMS and improved availability. Life extension studies for Fast Breeder Test Reactor have been completed and the Periodic Safety Review documents submitted to Atomic Energy Regulatory Board. The residual life is estimated to be about twenty calendar years. Relicensing of Fast Breeder Test Reactor for a period of five years is expected this year after review by Atomic Energy Regulatory Board. Seismic re-evaluation of Fast Breeder Test Reactor has been completed as a joint research project of our Centre and Safety Research Institute of Atomic Energy Regulatory Board. Retro-fitting measures based on the review are in progress.

In the coming years, Fast Breeder Test Reactor will embark on a large scale programme of testing the metallic fuels which will be the fuel for future fast reactors for better utilization of our uranium resources. Two types of design are on the anvil, namely the sodium bonded fuel being developed by IGCAR and the mechanically bonded fuel being developed by BARC. Shortly experimental pins of sodium bonded metallic fuel will be loaded in to Fast Breeder Test Reactor, primarily to validate the fuel pin fabrication process.

Ferro boron is an attractive candidate material for use as in-core shield material from the point of view of economics. Fast Breeder Test Reactor will be utilised to validate the design of ferro boron in-core shield material and in-core neutron detectors of higher sensitivity.

Contribution towards construction and commissioning of Prototype Fast Breeder Reactor

IGCAR and BHAVINI are committed to work in a seamless manner to complete the balance construction activities and take up subsequently the commissioning of the reactor. An APEX committee has been constituted at senior level with members from both the organisations and the project is being monitored on monthly basis to achieve the target of completion of construction by March 2012. An internal review mechanism has been constituted for timely supply of balance components from IGCAR as free issue to BHAVINI. The detailed schedule has been drawn for completing the in-sodium testing of two fuel handling machines, which have already been tested successfully in air.

An impressive milestone has been recently reached in the Prototype Fast Breeder Reactor project of successful completion of the last stage of sodium transfer from tankers to reactor storage tanks, thus completing the activity of transfer of 1750 tonnes of sodium to reactor storage tanks.

Design R&D towards Commercial Fast Breeder Reactors

The conceptual design of twin unit (2x500 MWe units) of Commercial Fast Breeder Reactor to be constructed at Kalpakkam has been finalised. The layout for component handling equipment in the fuel and decontamination buildings has been arrived at. Most of the equipments in the fuel and decontamination buildings are being shared between the twin units resulting in improved economy. The equipment to handle radioactive waste has also been integrated with fuel building since a major portion of the active waste generated is associated with component handling. Optimisation of permanent reactor assembly components has been completed and the preliminary cost estimation indicates a saving of ~25 % on specific weight,

which in turn has strong linkage with the economics.

A road map has been established to enhance the safety features of future fast reactors. Towards enhancing the reliability of shut down systems by an order of magnitude, use of both active and passive devices are foreseen. Strolle limiting device will be incorporated in the existing design of control and safety rod drive mechanism to limit the uncontrolled withdrawal of the control rod. Instead of use of an electromagnet in the existing design of diverse safety rod rive mechanism, self actuated shut down system through curie point magnet is planned as a passive shut down system. An additional third shut down system of either liquid enriched lithium- 6 injection system or 90% enriched boron carbide balls is planned to be implemented. These three shut down systems will be extensively tested to validate the design.

In-Vessel sodium purification is an option that is being pursued, where, all purification system components, are kept inside the reactor vessel thereby preventing radioactive primary sodium from exiting the reactor vessel. 1:1 in-vessel cold trap is under manufacture and the performance would be tested in an existing sodium test facility.

Feedback from PFBR to reduce construction time in particular is being incorporated in the design of steam generators. Instead of design with eight steam generator modules with each tube of 23 meters, design with six steam generator modules with each tube length of 30 meters is envisaged to reduce construction time, enhanced safety through ~ 40% reduction in tube to tube sheet welds and ~ 25% reduction in cost. A module with few tubes is planned to be tested in the steam generator test facility.

Indigenous development of electrical and instrumentation items imported for Prototype Fast Breeder Reactor would be planned so as to result in economy for future reactors.

Materials programme will be directed towards development of 15-18Cr oxide dispersion strengthened (ODS) ferritic steels for fuel cladding to meet the ambitious fuel burn-up target of 200 GWd/t and also to facilitate reprocessing by Purex process. Advanced 15Cr 15Ni steel with optimized phosphorous and silicon contents will be studied for improvement in creep properties through optimum thermo mechanical heat treatment, for fuel cladding applications. Special end plug welding technologies will be established for both ODS steels and advanced stainless steels.

As a part of cost optimisation of the future Fast Breeder Reactors, the materials for construction of cold leg sodium and hot leg sodium loop piping and components are proposed to be changed from 304 LN and 316 LN to 2¼Cr–1Mo and Modified 9Cr–1Mo, respectively. In view of this, development of sodium service valves in ferritic steel is being taken up with the participation of valve manufacturers in the country.

Efforts are being made for development and evaluation of boron added modified 9Cr-1Mo steel with improved creep performance for steam generator applications. Exploratory research is underway for utilisation of grain boundary engineering concepts along with thermomechanical processing to enhance resistance to sensitisation for austenitic stainless steels and to minimise irradiation induced embrittlement in ferritic-martensitic steels. R&D for development and long term performance evaluation of advanced materials and coatings for pyro-reprocessing applications will be pursued.

Emphasis will be placed on in-service inspection of reprocessing plant components, and in-sodium viewing and measurements and in-sodium non-destructive evaluation for cracks detection for reactor internals.

Re-irradiation of the fuel pins, in-pile creep experiments and fabrication of fuel sub-assemblies inside hot cells for dispatch of fuel back to Fast Breeder Test Reactor storage after Post Irradiation Experiments, will be the new areas to be taken up.

Code development in the field of thermo hydraulics, structural integrity and severe accidents will be taken up on priority with the support of national and international institutions.

Reprocessing Programme

Campaign of reprocessing of spent fuel with a burn-up of 155 GWd/t has been carried out successfully. The operations have reached a level of maturity in terms of product quality, waste reduction and control on radiation exposure to the operators. The future campaigns in CORAL will be directed towards increasing the number of campaigns and taking up forty pins instead of twenty pins in each campaign.

The construction of DFRP is being monitored on a monthly basis and it is targeted to reduce the construction time by three months.

R&D towards metallic fuel

Introduction of plutonium based metallic fuels in commercial domain is envisaged for achieving high breeding ratio and realisation of shorter doubling time. Metallic fuels have the advantage of excellent compatibility with sodium coolant and amenability of adopting non-aqueous reprocessing routes like pyro-reprocessing. The pyro-reprocessing method is amenable for handling short cooled fuels and would result in the reduction of the waste volumes.

A sustained programme on fuel development, including measurement of fuel properties, fabrication and irradiation of metallic fuel pins (and subsequently of sub-assemblies), and pyrochemical reprocessing has been initiated. The R&D programme has been conceived in such a way that various aspects of the fuel cycle are developed with appropriate test programme, which would yield sufficient data and confidence to design and operate the metallic fuel based reactor and the associated fuel cycle facilities.

To demonstrate the technology for fabrication of sodium bonded test fuel pins, a facility with a train of argon atmosphere glove boxes has been set up at Chemistry Group. Capsule irradiations are planned to be carried out at Fast Breeder Test Reactor with T91 clad with fuel composition varied in a systematic manner. This activity is being pursued in close collaboration with Bhabha Atomic Research Centre.

The engineering scale facility for demonstration of selected unit operations of pyroprocessing has been commissioned, and the processing of uranium metal on 1 kg scale by electrorefining, followed by consolidation of the deposit, has been successfully completed. In the coming years, additional facilities will be established for mastering all the unit operations on an engineering scale, with remotisation and automation. It is targeted to complete the developmental work in time to take up the reprocessing of the metallic fuel when it is discharged from Fast Breeder Test Reactor after the test irradiations.

In parallel, design of 120 MWe Experimental Metal Fuel Reactor has been initiated. The primary objective of this reactor is to test the fuel and blanket subassembly on 1:1 scale basis and at the same time ensuring that it can sustain its own fuel requirement without any external feed unlike the case of Fast Breeder Test Reactor.

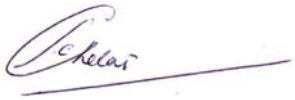
Fast Reactor Fuel Cycle Facility (FRFCF)

FRFCF is a cluster of co-located plants required for, reprocessing of irradiated fuel of Prototype Fast Breeder Reactor, re-fabrication of the fuel and handle associated radioactive waste management. The design of plant of FRFCF is a joint effort of IGCAR, BARC and NFC. FRFCF is being co-ordinated and piloted by IGCAR. The present emphasis is to complete the detailed design and seek the necessary clearances for construction from Atomic Energy Regulatory Board by end September 2011. Tenders for civil packages and long delivery components will be floated immediately on receipt of the financial sanction. FRFCF will be the most challenging project to be executed by IGCAR.

R&D in Basic Sciences

IGCAR right from its inception has placed an emphasis in select areas of basic research. It is my considered opinion that scientists must look for potential areas of applications of their domain knowledge and expertise. A beginning will be made in the areas of surface science for engineering applications and SQUID based magnetoencephalography (MEG) for the medical applications. R&D in basic sciences in the field of radiation damage, nano sciences, chemical sensors and ferro fluids to name a few will continue. It is my desire that our scientists, who pursue open-ended basic sciences have to be amongst the best.

We are now in the process of finalising our programmes for the XII five year plan. Through an elaborate exercise with the young and the senior colleagues, we have arrived at the contours of the R&D that we propose to pursue in the XII plan. An exciting period is ahead for the Centre, in that we will soon witness the commissioning of the Prototype Fast Breeder Reactor and take steps for establishing more fast reactors. I am confident that with the dedication, dynamism and passion for innovations exhibited by our collleagues, we stand poised to make a substantial contribution to the setting up of future Fast Breeder Reactors and fuel cycle facilities in a way that would enhance the acceptance of Fast Breeder Reactors as an important source of energy for the country.



S. C. Chetal
Director, IGCAR

Natural Circulation Sodium Loop - SADHANA

The decay heat generated in the reactor core even after the reactor is shut down is one of the most important safety concerns for any operating nuclear reactor. This decay heat is to be removed even in the event of a station blackout to limit the temperature of the coolant in the primary circuit, thereby maintaining the structural integrity of the reactor components.

In the case of the Prototype Fast Breeder Reactor there are two distinct decay heat removal circuits viz: the operating grade decay heat removal system and the safety grade decay heat removal system. Operating grade decay heat removal system is the normal heat transport circuit available for all the normal reactor operating conditions except when the secondary circuit is not available. Safety grade decay heat removal system ensures the removal of the decay heat from the reactor in the case of non availability of operating grade decay heat removal system. This passive system will be put into operation by opening of dampers on the air side. This system consists of four independent circuits of 8 MWt capacity each. Each circuit is provided with a decay heat exchanger immersed in reactor pool, air heat exchanger, air stack, dampers, expansion tanks, associated piping and auxiliaries. The dampers are kept crack open during normal operation to keep the system poised.

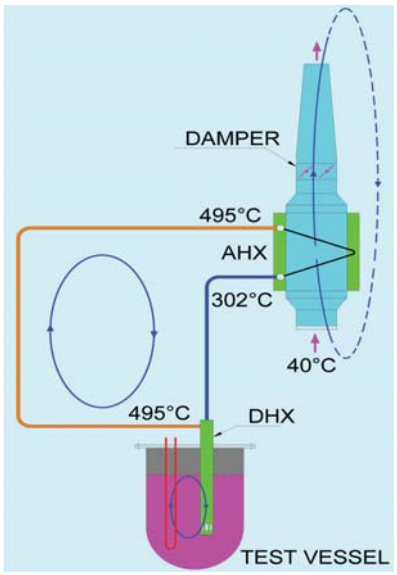


Figure 1: Schematic of the natural circulation paths in safety grade decay heat removal

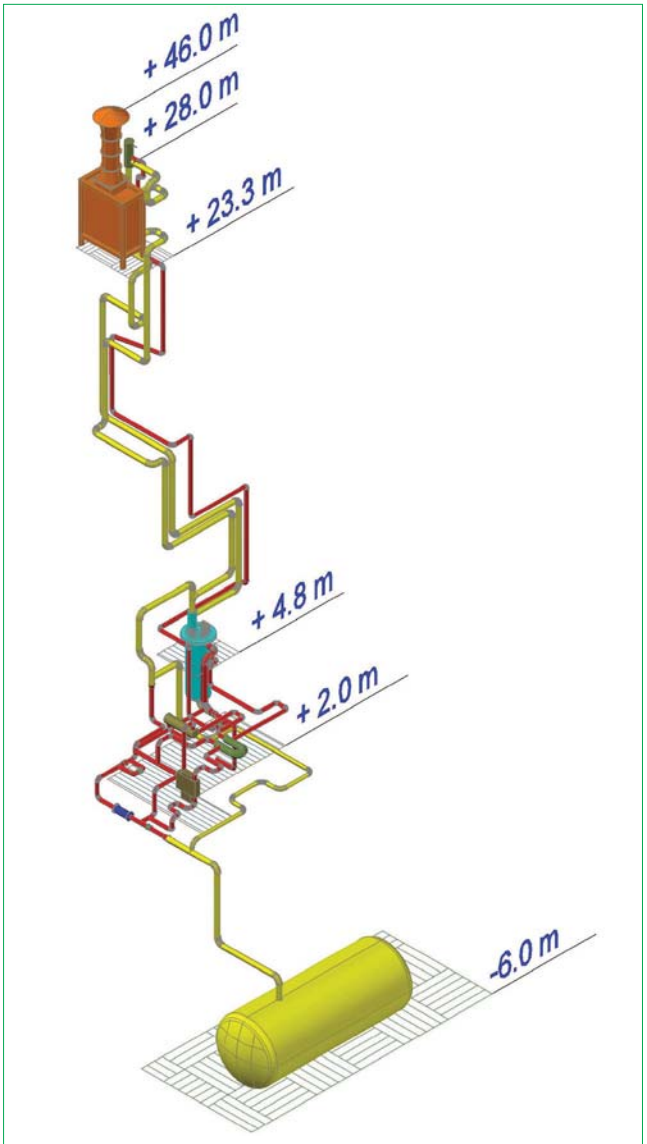


Figure 2: Isometric view of SADHANA loop

An accurate evaluation of thermo hydraulics under natural circulation condition is important for proper utilization and enhancement of passive safety features of Prototype Fast Breeder Reactor. While the interaction between the core and decay heat exchanges has been studied in the ¼th scale model SAMRAT in detail, a separate natrium (Sodium) loop was set up to demonstrate and evaluate the natural circulation flow and heat removed safety grade decay heat removal in Natrium (Sodium) loop was set up to demonstrate and evaluate the natural circulation flow and heat removal in all the three different heat removal paths of safety grade decay heat removal viz: the primary pool, decay heat exchanger - air heat exchanger sodium

Figure 3: View of decay heat exchanger before erection

Figure 4: Internal view of model air heat exchanger

circuit and by natural draught through chimney for air circulation in air heat exchanger (Figure 1).

This scaled down model of the circuit was designed, fabricated, installed and commissioned in Hall Number III of Fast Reactor Technology Group. This 1:22 scaled model facility is based on Richardson number similitude which is the predominant non dimensional number governing buoyancy induced natural circulation flows of the capacity of safety grade decay heat removal in sodium loop is 355 kW and the height difference between the thermal centers of decay heat exchanger and air heat exchanger is 19.5 m (Figure 2). The sodium in Test Vessel-IV which simulates hot pool of Prototype Fast Breeder Reactor is heated by immersion type electrical heaters. This heat is transferred to the secondary sodium through the model decay heat exchanger (Figure 3). The secondary sodium gets circulated in the secondary loop by the buoyancy head developed in the loop due to the temperature difference in hot and cold legs of the loop. The heat from secondary sodium circuit is rejected to the atmosphere through the air heat exchanger (Figure 4). A 20 meter high chimney develops the air flow required to transfer the heat from secondary sodium to the atmosphere through air heat exchanger. Different sodium pool temperatures can be achieved by controlling the heat input to the sodium pool. This is achieved by controlling the input heater power through immersion heaters. Different cooling rates of sodium inside test vessel-IV (Figure 5) can be achieved by admitting cold sodium at required temperature through the test vessel-IV bottom nozzle with the help of an electromagnetic pump. The design temperature of test vessel-IV, secondary sodium loop and related equipments are 600 °C.

Figure 5: Top view of test vessel

Scaling down the safety grade decay heat removal system of Prototype Fast Breeder Reactor to an economically and practically viable experimental system in sodium with minimum distortion in the physical behaviour was a challenging task.

Figure 6: SADHANA Loop with Chimney

During the conceptualization of the experimental system, modeling criteria based on similitude laws were adopted in the system level, equipment level and sub equipment level. The system and equipments were designed based on the design philosophy adopted for the design of safety grade decay heat removal system system for Prototype Fast Breeder Reactor. The process design and the mechanical design were carried out with the help of computational tools for finite element analysis and computational fluid dynamics. Fabrication of the components was a complicated task because of the tight tolerance requirement for the geometrically scaled down dimensions of the equipments.

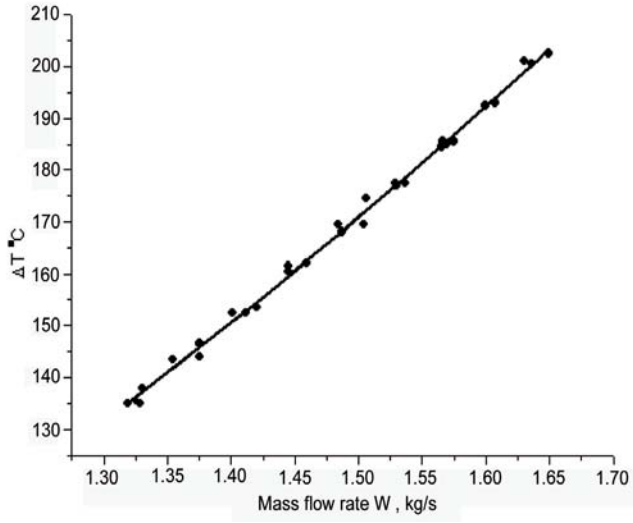


Figure 7: Steady state Natural Circulation flow in SADHANA

Table 1: Comparison between design predictions and the experimental results for SADHANA

Parameters	By Design	Experimental
Sodium pool temperature	526°C	526°C
Inlet temperature of sodium to decay heat exchanger	302°C	303°C
Outlet temperature of sodium from decay heat exchanger	495°C	498°C
Inlet temperature of sodium to decay heat exchanger	495°C	495°C
Outlet temperature of sodium from air heat exchanger	302°C	305°C
Sodium flow rate	6.0 m^3/h	6.55 m^3/h
Air inlet temperature in air heat exchanger	40°C	30°C
Air out let temperature of air heat exchanger	300°C	267°C
Heat removal capacity	355 kW	396 kW

Safety grade decay heat removal in sodium loop (Figure 6) was commissioned with all the systems in place. Experiments in safety grade decay heat removal in sodium were conducted at different sodium pool temperatures. At 550 °C sodium pool temperature, the secondary loop removes 19.4% more power than its rated capacity. Figure 7 shows the natural circulation as a function of the difference in hot and cold leg temperatures.

The one dimensional design calculations of the safety grade decay heat removal in sodium loop at a pool temperature of 526 °C secondary loop was expected to remove 355 kW with a sodium flow of 6 m^3/h in the secondary. At this condition the experimental observations show a heat removal of 396 kW with a sodium flow of 6.55 m^3/h. The temperature drop/gain in air heat exchanger and decay heat exchanger remained close to the expectations. Table 1 shows the comparison between design predictions and the experimental results for the same conditions. The viability of the fully passive Prototype Fast Breeder Reactor decay heat removal system was successfully demonstrated by experiments in SADHANA in sodium facility and was a great achievement towards establishing fast reactor safety.

(Reported by K.K.Rajan & colleagues,
Fast Reactor Technology Group)

Production of ^{89}Sr in Fast Breeder Test Reactor

The most common pain syndrome encountered in cancer patients is metastatic bone pain. It is seen in up to 70% of patients with prostrate and breast cancer, and up to 30% of patients with lung, bladder and thyroid cancers. In addition to pain, common complications include skeletal fractures, hypercalcemia and spinal cord or nerve root compression, all of which affect mobility and sleep, greatly reducing the patient's quality of life. Management of bone pain includes analgesia, radiotherapy, radiofrequency (RF) ablation, hormones, chemotherapy and surgery. Radiotherapy uses bone seeking isotopes like phosphorous (^{32,33}P), strontium (^{89}Sr), etc.

Strontium is an element that behaves biologically like calcium. Strontium-89 chloride localizes selectively in bone, especially bone cells that are rapidly dividing (such as areas with bone metastases that are causing the pain). It remains in the bone for several weeks providing pain relief – mean duration of relief is about six months but can last up to fourteen months. The radiation emitted is absorbed almost completely within this area, maximizing the efficacy of the treatment. The usual therapeutic dose is 148 MBq (4 mCi). The compound is currently marketed under the brand name "Metastron" in USA and Canada and costs about USD 3000 per dose.

Preliminary experiments have been carried out in Fast Breeder Test Reactor to determine the feasibility of producing ^{89}Sr and are described below.

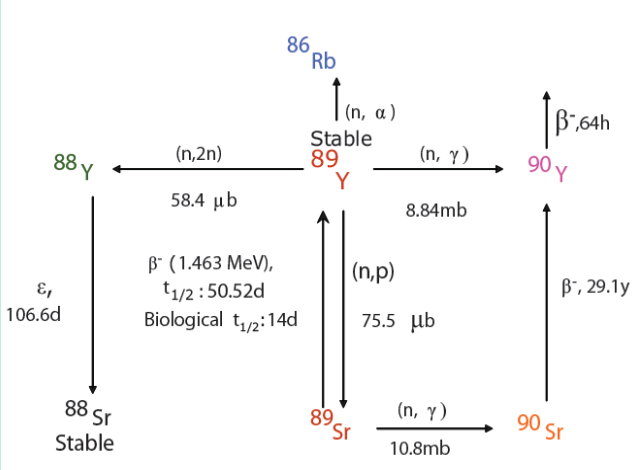


Figure 1: Reaction scheme for production of ^{89}Sr from ^{89}Y

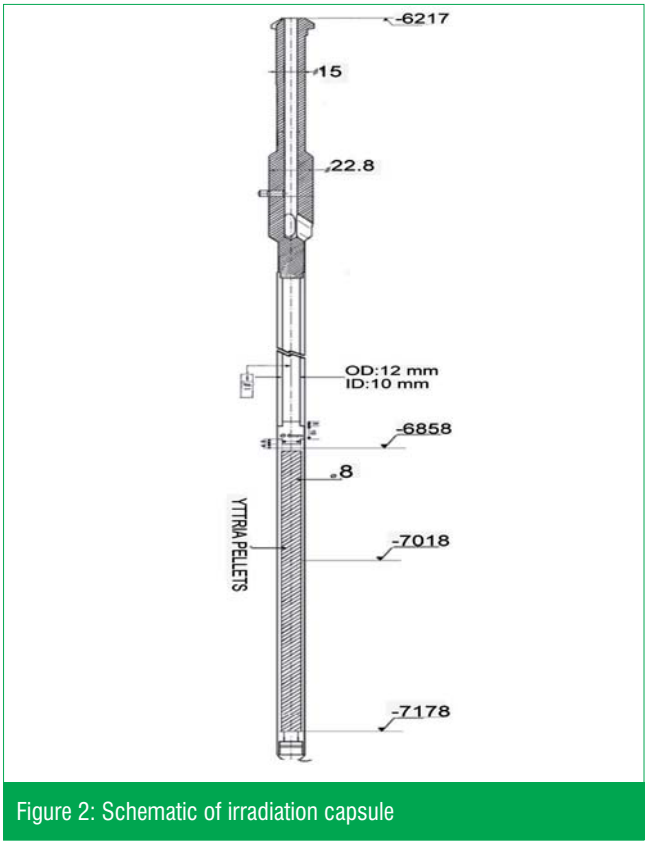


Figure 2: Schematic of irradiation capsule

Production routes

^{89}Sr is mainly produced using the (n,γ) reaction on enriched strontium (^{88}Sr>99.9%) targets in thermal reactors. Enriched targets are necessary to avoid the formation of ^{85}Sr – an undesirable impurity in the present context – and to increase the yield of ^{89}Sr. This radionuclide is also produced in fast reactors using the (n,p) reaction on ^{89}Y. One of the major advantages of using the (n,p) or (n,α) reaction is that the element produced is different from the starting element enabling us to chemically separate the desired radionuclide with very high specific activities. In fast reactors, specific activities of about 19 kCi ^{89}Sr per gram of strontium have been achieved vis-a-vis about 10 Ci/g strontium in thermal reactors. However, the penalty paid lies in the more complex procedures required for the introduction and removal of samples from the sodium covered fast reactor pile, formation of undesirable radionuclides due to other reactions {e.g. ^{88}Y due to the ^{89}Y(n,2n) reaction} and complex chemical separation procedures.

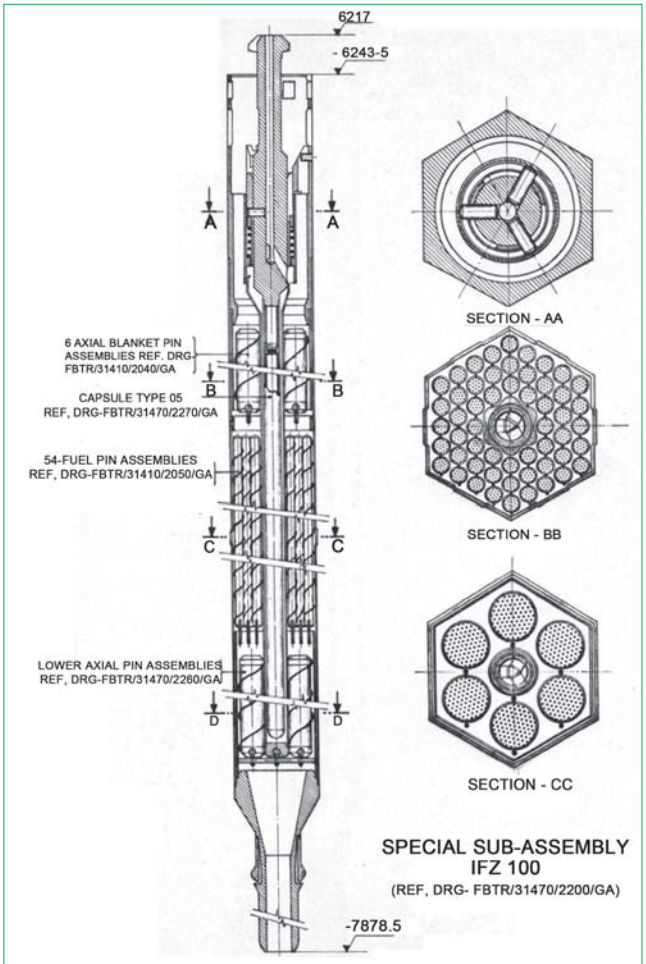


Figure 3: Sketch of special fuel sub-assembly

Natural yttrium – as yttria, Y_2O_3 – was used as the starting material for the production of ^{89}Sr. Yttrium has only one stable isotope ^{89}Y, and the various nuclear reactions that take place when this nuclide is irradiated in a fast flux is shown in Figure 1. ^{89}Y undergoes the (n,p) reaction to form the nucleus of interest, ^{89}Sr, the (n,γ) reaction to form ^{90}Y, the (n,2n) reaction to form ^{88}Y and the (n,α) reaction to form ^{86}Rb. The ^{89}Sr formed also undergoes neutron capture to form ^{90}Sr and also decays by negatron decay to the starting nucleus, ^{89}Y.

Irradiation capsule and special fuel sub-assembly IFZ-100

The yttria pellets were obtained by compacting the powder either with or without a binder like zinc oxide. The pellets were sintered at 1600°C for five hours and then loaded into the irradiation capsule in an inert atmosphere glove box and sealed by welding. A schematic of the irradiation capsule is shown in Figure 2. The length of the pellet column was 300 mm and the column was arranged to be in line with the fuel column of the fuel pins in the core. The outer diameter of the irradiation capsule was 12 mm and the inner diameter 10 mm. The yttria pellets had a diameter of about 8 mm to facilitate easy loading and removal from the irradiation capsule.

The irradiation capsule was then locked in a special fuel sub-assembly, IFZ100 and loaded into the central position of core. Figure 3 shows the sketch of the IFZ100 with the irradiation capsule locked in it. This special irradiation sub-assembly – of RAPSODIE design – is similar to normal Fast Breeder Test Reactor fuel sub-assembly except that the central seven fuel pins along with the top and bottom central axial blanket pins have been removed to make place for a tube sheath (guide tube) with a bore diameter of 13 mm. The irradiation capsule can be inserted into the tube sheath and locked. A fine hole is provided at the bottom of the tube sheath for draining the sodium when the capsule is taken out after irradiation. There will be a trickle flow of sodium around the capsule (i.e. in the annular space between the irradiation capsule and the tube sheath) but for computing the temperature of the pellets, this flow is neglected and it is assumed that the sodium in the annular space is stagnant.

Computation of specific activities

Table 1 lists some of the nuclear properties that have been

Table 1: Relevant Nuclear Data

S. No	Nuclear Property	Value
1	^{89}Y (n,γ) cross section in FBTR	8.8435E-03 b in core centre
2	^{89}Y (n,p) cross section in FBTR	7.5592E-05 b in core centre
3	^{89}Sr (n,γ) cross section in FBTR	0.010872 b in core centre
4	^{89}Y (n,2n) cross section in FBTR	5.842E-05 b in core centre
5	FBTR core centre flux	2.42102 E + 15
6	^{89}Sr decay mode	β^-
7	^{89}Sr half life	50.53 days
8	^{89}Sr β^- max	1.46 MeV
9	^{89}Y (n,p) – energy of emitted proton	~ 1 MeV
10	^{90}Sr decay mode	β^-
11	^{90}Sr half life	28.79 a
12	^{90}Sr decay mode	β^-
13	^{90}Y half life	64 h
14	^{88}Y half life	106.65 d
15	^{88}Y decay mode	E
16	^{88}Y gamma energies and intensities	$E\gamma$ – 898 keV $I\gamma$ – 92.7%

Table 2: Radionuclide yields in FBTR core centre and BOR-60 reactor–30 days irradiation campaign

S.No	Irradiation Position	Radionuclide yield (Ci/g Y)
1	^{89}Sr yield in FBTR	0.0113
2	^{89}Sr yield in BOR-60	0.01
3	^{88}Y yield in FBTR	0.0046
4	^{88}Y yield in BOR-60	0.003
5	^{90}Sr yield in FBTR	1.98E-09

used to compute the amount of ^{89}Sr that could be formed per gram of yttrium irradiated at the core centre of Fast Breeder Test Reactor. One group cross sections averaged with Fast Breeder Test Reactor core centre flux were obtained from Reactor Physics Division, Reactor Engineering Group. Assuming an irradiation period of one month, the Batemann equation was used to compute the activity of ^{89}Sr and ^{90}Sr that are produced. Table 2 shows the ^{89}Sr yield in the core centre of Fast Breeder Test Reactor and also shows the activity produced in the BOR-60 reactor. As can be seen from this table, we can produce about 11.3 mCi ^{89}Sr per gram of yttrium in the core centre (for an irradiation period of 30 days) and the activity produced in the core centre position corresponds well with the activity produced in the BOR-60 reactor. However, in addition to the product of interest, a number of other radionuclides are also formed. The target nucleus, ^{89}Y will undergo the (n,γ) reaction producing ^{90}Y. This is a pure beta emitter with a half life of 64 hours and the amount produced via this route would vanish very soon. The (n,2n) reaction on the target nucleus would lead to the production of ^{88}Y; a radionuclide with a half life of 106.65 d. This nuclide decays by electron capture and emits two high energy and intense gammas during this decay. This would result in the dose due to the sample being significant; however as the yttrium is chemically separated and discarded, this would not affect the use of ^{89}Sr. The high dose due to this radionuclide (about 50 R/h per pellet) necessitates that the chemical separation processes are carried out in shielded cells. ^{90}Sr is also produced due to neutron capture by ^{89}Sr. Production of ^{90}Sr is not desirable as this nuclide is long lived and as per regulatory standards it should not be present in ^{89}Sr at an activity level above 10^{-4}% of the ^{89}Sr activity. This nuclide decays to ^{90}Y which is a hard beta emitter and would be in secular equilibrium with its parent. Our computations showed that the activity of ^{90}Sr in the sample is below regulatory limits.

Scheme of irradiation

The IFZ100 with the irradiation capsule was loaded in the core

Table 3: Quantification of the ^{89}Sr recovered

Technique	^{89}Sr /g yttria (mCi)	% Computed yield
Computed	14.21	-
Liquid Scinitllation	14.18	99.8
Cerenkov	13.58	95.6
Gamma Spectrometry	13.03	91.7

centre position of FBTR. Irradiation was carried out for a total of 73 days in two steps with a break of 40 days. At the end of the irradiation, the IFZ100 was transferred from the core centre position to the storage location. The irradiation capsule was then transferred to a dummy outer sheath using the capsule transfer gripper. The outer sheath along with the irradiation capsule was brought to the Radiometallurgy Laboratory. Here, the irradiation capsule was dismantled from the outer sheath sub-assembly and the irradiation capsule cut open using a laser cutting machine. The pellets were transferred from the irradiation capsule to a new container. This container was transferred using shielded cask to the Radiochemistry Laboratory hot cells for chemical processing. Preliminary experiments were carried out using one pellet at a time before taking up bulk processing.

Dissolution and chemical separation

The irradiated pellets were dissolved by refluxing with concentrated nitric acid and the dissolution kinetics established by monitoring the activity of ^{88}Y in the dissolver solution as a function of time. It was seen that the complete dissolution was achieved in about 40 hours. The process was

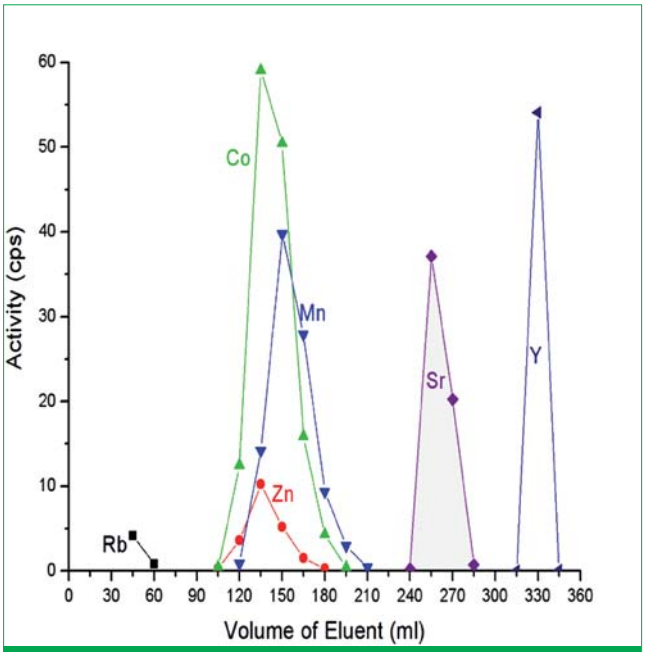


Figure 4: Elution profile

Table 4: Specifications for carrier free ^{89}Sr chloride		
1	Appearance	Transparent colorless solution
2	Concentration activity on delivery day	> 0.5–2.5 mCi/cm^3
3	Radionuclide admixture content on delivery day (% per ^{89}Sr content)	0.01
	a) beta particles energy from ^{32}P up to ^{35}S	< 0.2
	b) ^{90}Sr	<2.3 x 10^{-4}
	c) gamma-emitting radio nuclides	<0.4
	d) Chemical admixture content, (mkg/cm^3)	Al :<2.0, Fe :<5.0,-Pb: <5.0
	e) Molar concentration of hydrochloric acid (m/L)	0.05–0.5

modified by refluxing under pressure and this modification resulted in a lower dissolution time. The bulk yttrium in the dissolver solution was then removed by solvent extraction using 100% TBP which extracts the yttrium to the organic phase and leaves the strontium in the aqueous phase. Extraction profiles were monitored using high resolution Gamma spectrometry and it was observed that after three extractions, only 0.03% of the initial yttrium content was left behind in the aqueous phase. The aqueous phase contained almost all the zinc and strontium (monitored using ^{65}Zn and externally added ^{85}Sr) whereas almost all the ^{160}Tb and most of the ^{141}Ce were removed into the organic phase along with yttrium.

The aqueous fraction contained ^{89}Sr along with trace levels of yttrium and other radioactive impurities. Hence the strontium source was purified by ion exchange chromatography using DOWEX 50 W x 8. The column was conditioned and loaded using 0.1 M nitric acid. Radioactive impurities like ^{65}Zn, ^{54}Mn, etc were eluted using 0.5 M nitric acid. Elution of ^{89}Sr was carried out using 1 M nitric acid while the yttrium was removed from the column using 8 M nitric acid. The purification procedure was standardized using ^{85}Sr carrier as ^{89}Sr is a pure beta emitter and Figure 4 shows the elution profile obtained.

Table 5: Chemical and radiochemical characterization of ^{89}Sr produced		
S.No	Element	Concentration (ppm) or activity (% ^{89}Sr activity)
1	Fe	41
2	Y	0.5
3	Al	< 0.5 (Below detection limit)
4	Ti	< 0.1 (Below detection limit)
5	Pb	<1 (Below detection limit)
6	^{54}Mn	1.0E-06
7	^{65}Zn	6.8E-06
8	^{133}Ba	9.3E-07

Quantification of the ^{89}Sr yield

^{89}Sr is a pure β^- emitter with E_{β^-} (max) of 1.49 MeV and has a gamma of 0.909 MeV with intensity of 0.001%. In view of the high beta maximum, it emits Cerenkov radiation in addition to the above mentioned beta and gamma radiation. Hence, the quantification of the ^{89}Sr was carried out using all three methods: high resolution Gamma spectrometry, Cerenkov counting as well as liquid scintillation counting. The results are shown in Table 3 which also shows the computed yield per gram of yttria. It can be seen from this table, that we are able to recover more than 90% of the computed yield.

Chemical quality control of ^{89}Sr source

^{89}Sr is administered as chloride for patients suffering from bone metastases and the final product specification for carrier free strontium chloride is shown in Table 4. Efforts are underway to develop appropriate standardization procedures to conform to these specifications. Table 5 gives the results of chemical and radiochemical characterization of the source. It can be seen from this table that the major impurity is Fe along with some activated products of stainless steel. This impurity arises due to transfer of material from the SS capsule to the pellets and would not be present in the next run it is planned to encapsulate the yttria pellets in quartz before insertion into the irradiation capsule.

Conclusion

Preliminary experiments carried out have revealed the feasibility of producing ^{89}Sr in Fast Breeder Test Reactor by irradiation of yttria powder. The yields are similar to what is obtained in either BR-10 or BOR-60 reactors. Chemical procedures required for carrying out the dissolution of the irradiated pellets as well as separation and purification of the desired product have been demonstrated.

(Reported by C.R.Venkata Subramani & colleagues Chemistry, Reactor Operation & Maintenance and Metallurgy & Material Groups)

Young Officer's FORUM

Modeling and Simulation of Backup Control Room for PFBR Operator Training Simulator

Ms Rashmi Nawlakha did her B.E. degree in Electronic Instrumentation and Control Engineering from Rajasthan University (Jaipur). She is the Homi Bhabha award winner for topping the Electronics and Instrumentation discipline from the 2nd batch of BARC Training school at IGCAR. She joined as Scientific Officer in September 2008 and is presently working in Computer Division, Electronics & Instrumentation Group.

One of the key elements considered for safe operation of the Nuclear Power Plant is the knowledge possessed by the operators. It is widely recognized that simulators play an essential and extremely important role in establishing a viable media for Nuclear Power Plant Operator Training Programme. It provides a platform for training the operators on normal and emergency conditions including all types of scenarios related to Steady State and Transient Conditions of the reference plant. Introduction of Full Scope Replica Simulator in the operator training programme is a major step towards enhancing the capability of the operators and significantly improving the safety of the plant.

High Fidelity Full Scope Training Simulator for Prototype Fast Breeder Reactor is being developed at Computer Division, in collaboration with Reactor Engineering Group. As per the Atomic Energy Regulatory Board guidelines, it is mandatory to have the Operator Training Simulator in place, so that the operators can be trained and qualified before getting deployed for the main plant operation. Development of Prototype Fast Breeder Reactor Operator Training Simulator follows the ANSI standard, ANSI-3.5-1998 for operator training and examination, IAEA–TECDOC–995 and IAEA–TECDOC–1411.

The operators are trained to operate main control room, handling control room and backup control room for smooth functioning of the plant. Control room facilitates remote monitoring and control of the entire plant. Backup control room is provided for safe shut down of the reactor and initiate decay heat removal in case of non availability of control room due to fire or any other reason. Further, the backup control room panels are fitted with instruments and controls to monitor all the important plant parameters. This is mainly to ensure that the reactor is maintained in safe shut down state and the decay heat is being removed.

Modeling and simulation of backup control room is significant as it helps the operators to understand the instruments and controls provided in backup control room and get trained on the panels so that they can take the necessary control actions and initiate safe shutdown of the plant in minimum possible time, incase of any emergency situation which otherwise can lead to a major accident.

Brief description of backup control room

Backup control room consists of two conventional panels accommodating indicators, recorders, lamps, control switches, annunciation windows, personal computers with its accessories, power supply unit etc. Backup control room is equipped with instruments to monitor neutron flux, control rod position, primary sodium inlet/outlet temperature, primary sodium flow, individual safety grade decay heat removal performance related signals, Safety grade decay heat removal

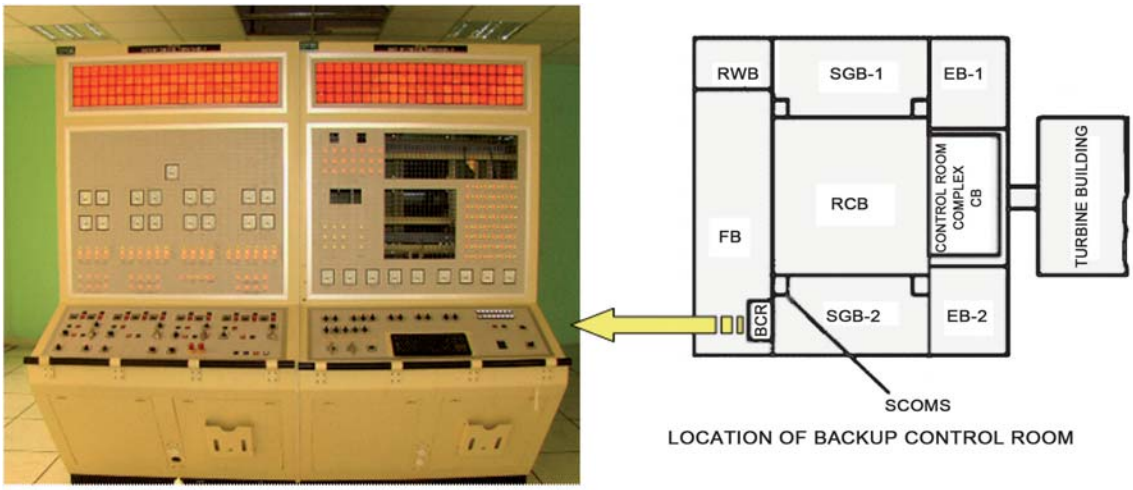


Figure 1: Backup Control Room

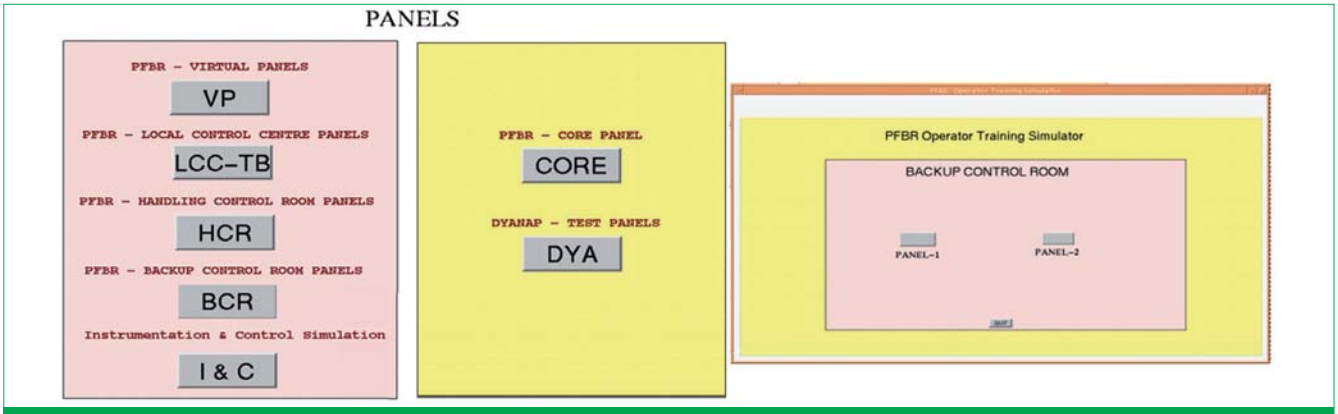


Figure 2: Main Virtual Panel Screen for BCR

damper status, diesel generator status, primary pump pony motor status etc. Control switches are provided to shutdown the reactor, to operate safety grade decay heat removal dampers and to start the diesel generator.

Backup control room is located in a physically and electrically separated building from main control room (Figure 1). It is possible to reach the backup control room easily and within five minutes. Backup control room remains unmanned during normal operation. When main control room is not available, two operators occupy backup control room to carry out the intended functions for required duration.

Modeling and simulation of BCR

A Simulation Software is used for development of Prototype Fast Breeder Reactor Simulator. It consists of three basic modules, namely Process Module for Conventional Process Simulation, Logic Module for logic simulation and Virtual Panel Module for creating soft Panels.

The Virtual Panel Modeling for backup control room is carried out using SL-GMS (Graphic modelling tool) which is a part of simulation tool. Virtual panel is the emulation of Hardware panel and this is helpful in the development stage of simulator. As a first step towards modeling of backup control room, the design notes of I&C system, related drawings and panel data are collected from Designers. These design notes and drawings are understood and a base document for carrying out modeling is prepared.

Virtual panel simulation

According to backup control room panel drawings, the virtual panels are created on SL-GMS. The panel is separated in three screens like annunciations, display and control switches screens to create virtual panel. All the meters, indicators, recorders and control switches are first created on SL-GMS tool. Then naming of all the instruments, fixing of ranges and units have been carried out.

In main virtual panel screen, a provision is provided to call dynamic screens for backup control room panels. This will call a screen to select backup control room panels (panel-1 & 2) as shown in Figure 2.

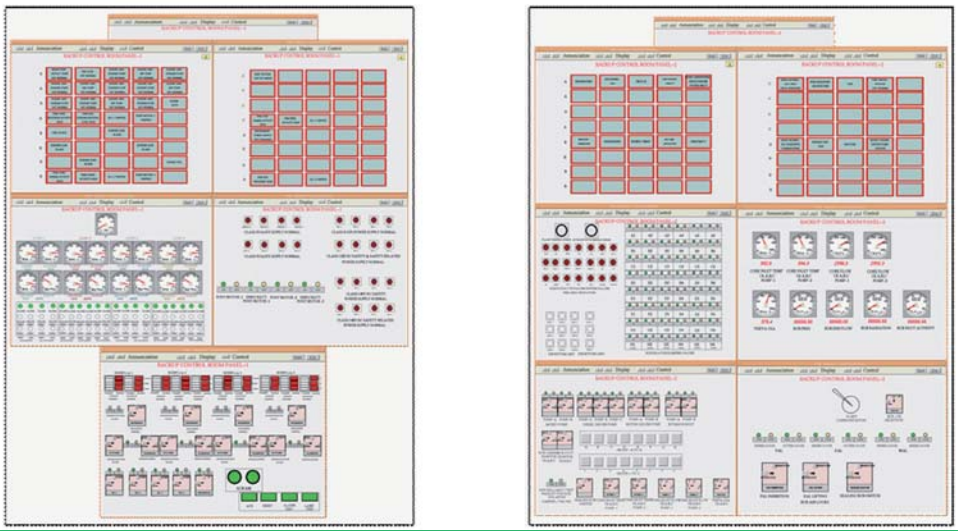


Figure 3: Virtual Panels for BCR Panel-1 & 2

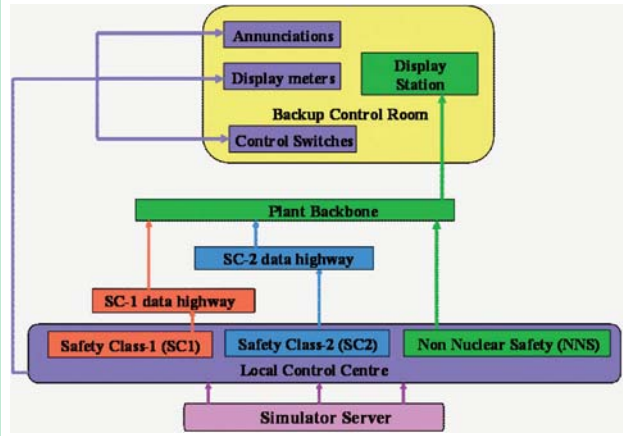


Figure 4: Data flow from simulator server to BCR

Figure 3 shows the virtual panel screens for backup control room panel-1 and 2. The functions for the annunciations, indicators, meters and control switches are also written behind the instruments in SL-GMS(graphic modeling tool). These are known as sub models. Once a sub model is created, it can be used for a number of screens wherever required. The function to select meter range and appropriate unit is written in "edit dynamics" screen.

Simulation of control logic for backup control room

The logic is developed using the logic tool to control the position (open/crack open/close) of air dampers (Pneumatic/Electrical) of safety grade decay heat removal. For this, first a simulator logic sheet has been prepared to control the position of air dampers. Then the appropriate variables are assigned according to the naming convention. The relevant variables are entered into the logic database.

The logic is developed for a two state selector switch to select backup control room or main control room. On selection of backup control room, the control is transferred to backup control room panels thus allowing the operator to SCRAM the reactor through backup control room panels. On selecting main control room, the control operations can not be carried out through backup control room. The control is transferred to control room at that time. The logic is developed to SCRAM the reactor manually. If backup control room is selected and manual SCRAM button is pressed then the reactor will come to shutdown state and the safety grade decay heat removal air dampers will be opened to remove the decay heat.

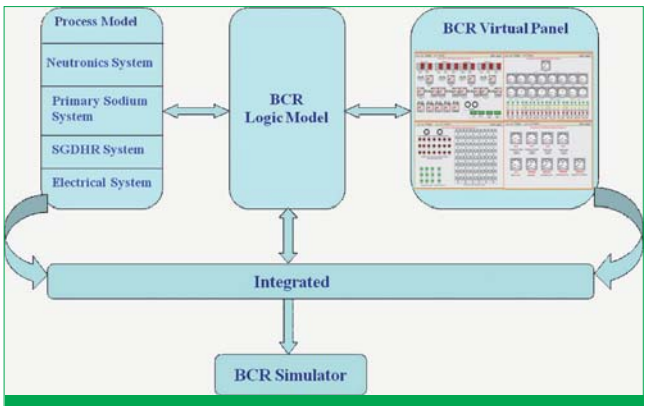


Figure 5: Integration of BCR Models in Simulator Environment

An external code is developed for acknowledge, reset, alarm test, lamp test to incorporate alarm window flashing, audio alarm, clearing alarm and checking the healthiness of all the digital output signals. Alarm test and lamp test pushbuttons are provided on backup control room panel-1 to check the working conditions of all the annunciations and lamps. Two more pushbuttons are provided on backup control room to acknowledge and reset the annunciations.

The code is integrated with the simulator environment. The parameters are subscribed from the process models, the logic compares the run time values with their corresponding thresholds and if any parameter crosses the threshold, the annunciation starts glowing on virtual panel. The input and output variables to the code are entered in the subscription and publishing file, respectively and all the variables are entered in signal database.

Signals come to backup control room

The important signals coming to backup control room are Log N, air heat exchanger sodium inlet/outlet temperature, air heat exchanger, air outlet temperature, sodium flow to air heat exchanger, air inlet/outlet damper position, air damper control for safety grade decay heat removal loop-1,2,3 & 4, Manual SCRAM, control safety rod/diverse safety rod bottom limit, core inlet temperature, core flow of pump-1 & 2, temperature of central sub-assembly, channel selection switches, backup control room/control room selection etc. Figure 4 shows the data flow scheme from simulator server to backup control room panels.

Integration and testing

The Process, Logic and the Virtual panel models are integrated to each other with a common database. The simulator environment establishes a communication path among its components through MDSM (Messaging & Data sharing mechanism). The process model of neutronics system, primary sodium system, safety grade decay heat removal system and electrical system are required for backup control room simulation. These process models generate input and output signals for every component. The control logics are integrated with the process models and

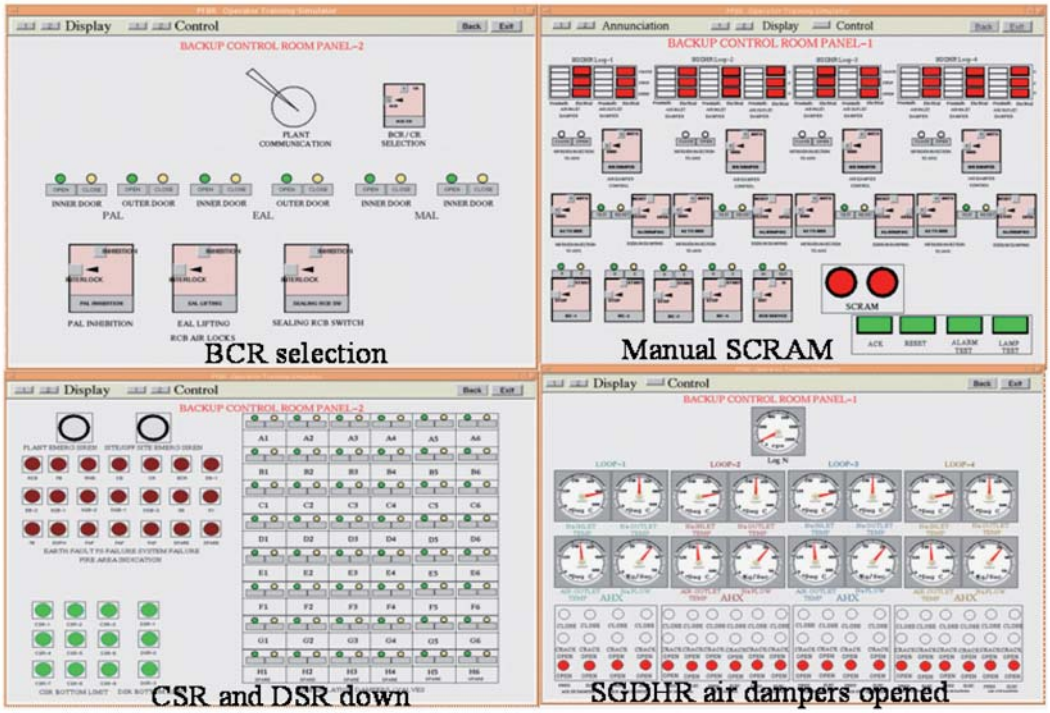


Figure 6: Manual SCRAM from BCR panels

virtual panel by attaching the appropriate variables. The logics generate trip signals, open/close signals and alarm signals. The virtual panels display alarms, indicators, recorders and provide the control switches to monitor and control the reactor status. The integration among all these modules is shown in the Figure 5.

After all these phases, the most important phase is the testing phase of the model. The testing comprises of two parts, Standalone testing and Integrated testing. In standalone mode, the individual meters, indicators, recorders and control switches are tested. If individually every meter is working fine and there is no error found on logic sheet, then integrated testing is carried out. In this, the process model, logic model and the backup control room virtual panel is integrated and working simultaneously to show the desired results.

Results and discussion

The virtual panel screens are shown when the SCRAM is initiated from backup control room if any emergency situation occurs. For shutting down the reactor through backup control room, first backup control room selection is done using the selector switch. Then Manual SCRAM round pushbutton is pressed shown by red color (Figure 6).

The bottom limit of control safety rod and diverse safety rod is displayed on backup control room panel-2. Initiation of MANUAL SCRAM signal leads to falling down of all control safety rod and diverse safety rod to shut down the reactor. This can be seen on the backup control room panel-2 screen with green color lamps indication. When reactor comes to shutdown state, Safety grade decay heat removal air dampers comes to fully open position. The red lamp indications confirm the fully open position of air dampers to remove the decay heat after shutdown.

Conclusion

The operators can be trained on backup control room simulator for all the controls provided on panels. Operator performance can be evaluated by using instructor station facility and simulating the significant events from instructor station and observing the response of the operator. The instructor records the activities carried out by trainee after initiating the event and thus imparting full fledged knowledge about backup control room operations.

Backup control room virtual panels are integrated with the Process models and Logic models. Process models generate the process signals in real time i.e. 200 ms cycle time. The run time parametric values are dynamically linked to logic models and the signals/parameters are compared with appropriate design values and the output generated is passed on to the virtual panels for alarm display, indication and control. A log message is generated for the connected parameters and the information is stored. The model is tested in the simulator environment and the simulated parameters are compared with the design values and found to be satisfactory.

(Reported by Rashmi Nawlakha,
Computer Division, Electronics & Instrumentation Group)

Young Researcher's FORUM

Radionuclide trap for use in Fast Reactor

Shri Prasanta Jana obtained his M. Sc. degree from Bengal Engineering and Science University, Shibpur, West Bengal. He is pursuing his doctoral degree in chemistry under the guidance of Dr. V. Ganesan, Head, Materials Chemistry Division, Chemistry Group. He is working on the development of materials for radionuclide trap for fast reactor. He has three publications in international journal. He has attended a number of national/international conferences and received best paper presentation awards on three occasions.

Owing to the high neutron flux encountered in the core of fast reactors, a large number of activated products such as ^{54}Mn, ^{58}Co, ^{60}Co, ^{59}Fe, ^{51}Cr etc. are produced in structural steel and clad materials. These radionuclides gradually get released into the coolant stream as a result of corrosion by liquid sodium. In addition to this, in the event of fuel pin failure, volatile fission products such as ^{134}Cs, ^{137}Cs, ^{131}I etc. are released steadily into the coolant stream. All these radionuclides get transported to out-of-core components by sodium coolant. On account of the polythermal nature and complexities of the construction, these radionuclides get deposited at various locations of coolant circuits. Since most of these radionuclides are fairly long-lived with relatively high gamma energies, they pose radiation exposure problems during regular operation and maintenance of the reactor system, resulting in long plant shutdowns and demanding high radiation shielding requirements. In addition, these radionuclides also contribute significant problems during decontamination and disposal of primary components of fast reactors. Such radionuclides, produced during reactor operation, need to be contained.

One of the methods of eliminating or minimizing the radiation burden caused by these radionuclides mentioned above is to develop and incorporate suitable radionuclide traps to concentrate and contain them at suitable locations (in-core or out-of-core) in primary system. Data from the operational experience of fast reactors indicate that the major radiation burden is due to the relatively long-lived and high gamma active corrosion product ^{54}Mn and fission product ^{137}Cs in the event of operation of reactor with failed fuel pins.

Development of Caesium trap

Successful operation of caesium (^{137}Cs) trap employing available carbon foam namely reticulated vitreous carbon as trap material was demonstrated in reactors such as EBR-II, FFTF, KNK-II, BN-350, BOR-60, BR-10 and Rapsodie. This report deals with in-house development of carbon foam, as import substitute for reticulated vitreous carbon, for trapping fission product ^{137}Cs from liquid sodium.

Synthesis of carbon foam

The carbon foam termed as home made porous carbon was synthesized from sucrose and characterized by various techniques such as scanning electron microscope, X-ray diffraction and thermogravimetry. The density and porosity of home made porous carbon are 0.10 g/cc and 95.5%, respectively. Figure 1 shows photograph of home made porous carbon. Figure 2 shows the secondary electron image of home made porous carbon. From the images, the average pore size of home made porous carbon was determined to be 700 μm.

Figure 1: Photograph of home made porous carbon

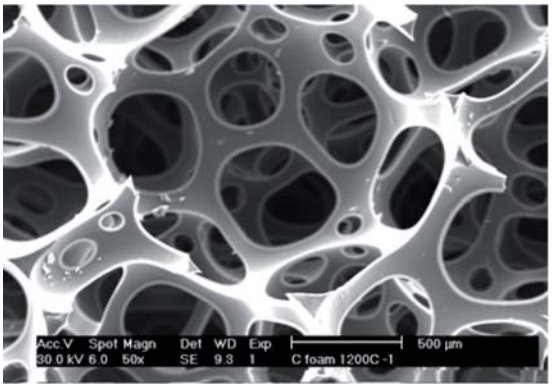


Figure 2: Secondary electron image of home made porous carbon

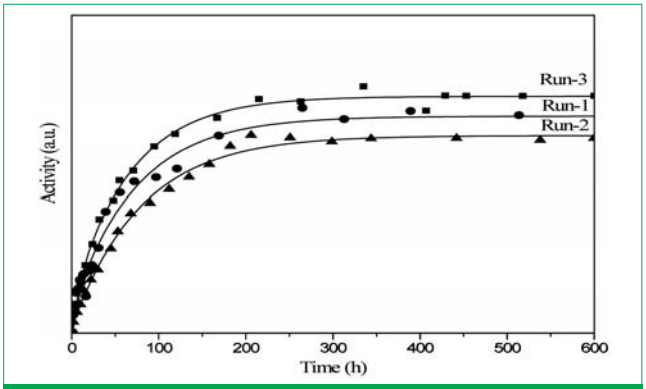


Figure 3: Caesium (^{137}Cs) uptake behavior of home made porous carbon

Studies on caesium trap

The caesium (^{137}Cs) trap experiments were conducted in a static sodium system at 473 K. In ^{137}Cs trap studies, home made porous carbon in cylindrical form as samples were used. Three experiments were conducted namely Run-1, Run-2 and Run-3. The ^{137}Cs activity taken in sodium for these three runs were 15, 13 and 16 μCi, respectively. Amount of ^{137}Cs trapped was measured by standard gamma counting technique employing a NaI(Tl) detector. The ^{137}Cs uptake behaviour is shown in the Figure 3. It was observed that in all the runs, home made porous carbon was found to saturate after 200 hours.

In these three runs, the ^{137}Cs activities trapped by home made porous carbon from sodium were 11, 10 and 12 μCi, respectively. The trapping efficiencies of home made porous carbon for ^{137}Cs in Run-1, Run-2 and Run-3 are 73%, 77% and 75%, respectively. The distribution coefficients (K_D) of ^{137}Cs between home made porous carbon and sodium are found to be in the range 4.9-6.0×10^2. It is to be noted that the trapping efficiency of home made porous carbon for ^{137}Cs is appreciable enough to be used as a trap material for trapping ^{137}Cs from primary sodium of fast reactor.

Development of Manganese trap

Experiments carried out in EBR-II reactor showed that rolled nickel sheet (as trap material loaded in the top of fuel sub-assembly) with dimples made to maintain interlayer spacing is effective in trapping ^{54}Mn. Small interlayer spacing is most efficient in trapping the radionuclides but results in high-pressure drop while a lower pressure drop is suitable to reactor conditions. The actual design is in-between these factors as the radioactivity adsorbed is proportional to the surface area of trap material.

There will be several advantages if nickel is used as a foam for trapping activated corrosion products from primary sodium of fast reactor. Trapping efficiency of nickel foam will be higher (nearly three times) than that of nickel foil as the surface area to volume ratio (S/V) of nickel foam (40 cm^{-1}) is three times higher than that of nickel foil (13 cm^{-1}). Nickel foam can be loaded in the top of fuel sub-assembly easily, because engineering modification of the foam is not required. As the porosity of nickel foam is very high (97.5%), the pressure drop of sodium flow will be less compared to that by employing nickel foil as manganese trap in form of fuel sub-assembly.

A study to indentify the suitability of available nickel foam and foil for trapping manganese from liquid sodium was also carried out.

Studies on Manganese trap

The manganese trap experiments were carried out at 773 K using nickel foam and foil in a static sodium system. Amount of manganese trapped by the nickel foam and foil was analyzed by atomic absorption spectroscopy. Secondary electron image of nickel foam and foil exposed in sodium (containing manganese) for 100 hours are shown in the Figure 4.

The amount of manganese trapped by the nickel foam and foil were in the range of 1.25-7.6 wt% and 0.75-7.35 wt%, respectively. Figure 5 shows the concentration profile of manganese trapped in nickel foam and foil indicating that manganese concentration is increasing with increasing exposure time in sodium. It is

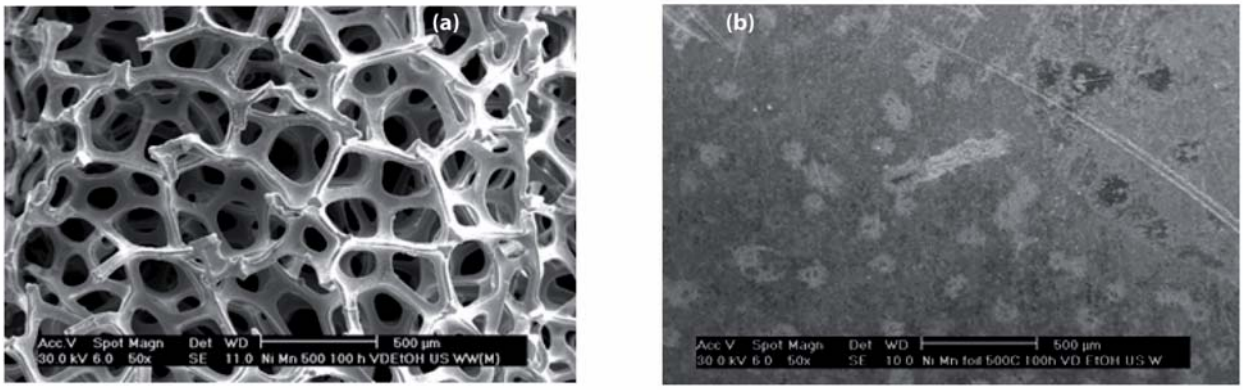


Figure 4: Secondary electron image of (a) nickel foam (b) nickel foil exposed in sodium.

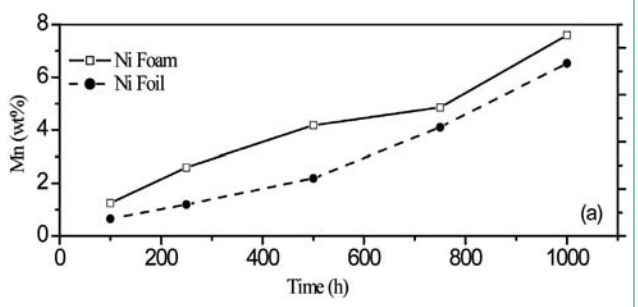


Figure 5: Concentration profile of manganese trapped in nickel foam and nickel foil as a function of time.

observed that concentration of manganese on the nickel foam samples is higher than that of nickel foil samples. This indicates that nickel foam has higher trapping efficiency for manganese from sodium than that of nickel foil. Trapping of manganese by nickel foam is more efficient at lower sodium exposure time (nearly two times higher than that of nickel foil exposed in sodium for 100-500 hours).

The microstructure of nickel foam exposed in sodium remains unaltered, which in turn, indicates that the mechanical properties are retained even after sodium exposure at 773 K. The same effect is also observed in nickel foam exposed in sodium in the dynamic system as well. The corrosion rate from weight loss measurement in dynamic system (sodium loop) at 773 K for 1000 hours was found to be 7 μm/a. Therefore, the corrosion rate of nickel foam (7 μm/a< 25 μm/a) indicates that it is highly compatible in sodium in spite of its large surface area.

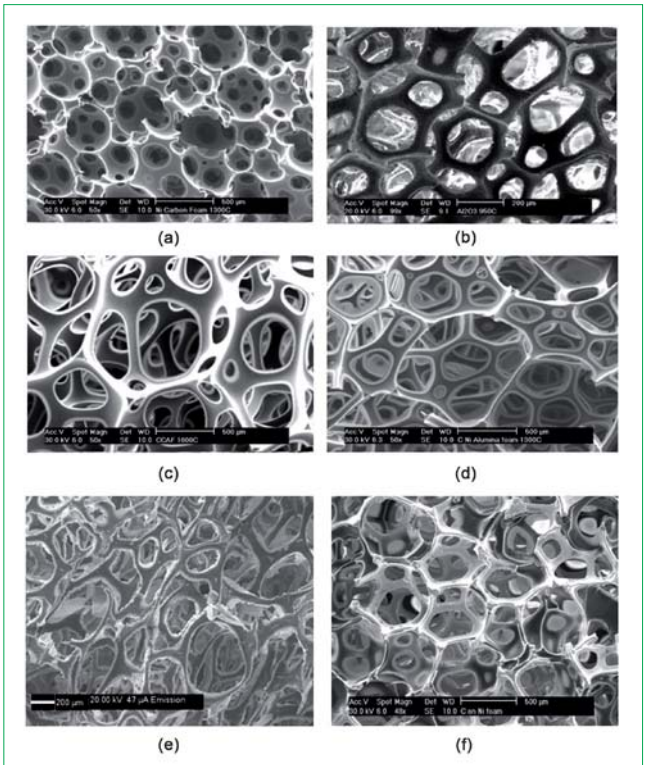


Figure 6: Secondary electron image of (a) nickel dispersed carbon foam (b) alumina foam (c) carbon coated alumina foam (d) nickel dispersed carbon coated alumina foam (e) nickel aluminate alumina foam and (f) carbon coated nickel foam"

Nickel foam was also exposed in sodium at 673 K without manganese. It was observed that nickel foam trapped approximately 4 ppm of manganese (analyzed by neutron activation). Nickel foam was also exposed in sodium loop at 773 K for 1000 hours. It was found that around 700 ppm of manganese was trapped by the nickel foam. These experiments indicate that nickel foam is very efficient to trap manganese even at low concentration present in the sodium system.

Multitrap development

As mentioned above, carbon foam is employed for trapping fission products and it has been demonstrated that nickel foam could be employed for trapping activated corrosion products from primary sodium system of fast reactors. For the installation of various traps such as caesium and manganese traps in the reactor core, separate space and necessary shielding for the traps are required in the active building of a reactor. Due to space constraint in the active building of fast reactor to accommodate various radionuclide traps, it is prudent to search for the material, which can trap fission and activated corrosion products radionuclides together from primary sodium.

Synthesis of mutitrap materials

For the purpose mentioned above, various composite materials consisting of carbon and nickel termed as nickel dispersed carbon foam, nickel dispersed carbon coated alumina foam and carbon coated nickel foam were synthesized by solution combustion route using sucrose as a base material. During the synthesis of nickel dispersed carbon coated alumina foam, intermediate product such as alumina foam, carbon coated alumina foam and nickel aluminate alumina foam were obtained. The solution combustion route is applied for the first time for synthesizing alumina foam and carbon coated alumina foam. Figure 6 shows secondary electron images of synthesized foam materials. The foam materials such as home made porous carbon, nickel dispersed carbon foam, alumina foam, carbon coated alumina foam, nickel dispersed carbon coated alumina foam and nickel aluminate alumina were synthesized without using template as a skeleton material. However, in the case of carbon coated nickel foam, nickel foam was used as a template material and sucrose as a carbon source. Suitability of synthesized composite foam materials to be a multitrap for trapping fission and activated corrosion products from primary sodium will be carried out near future.

(Reported by Prasanta Jana, Materials Chemistry Division, Chemistry Group)

Conference/Meeting Highlights

Report on the 7th CEA-IGCAR Annual Seminar on Liquid Metal Fast Reactors Safety

April 13-21, 2011

A series of four specialists meetings with different themes under the aegis of the 7th CEA-IGCAR Annual Seminar on Liquid Metal Fast Reactors Safety was organized at IGCAR during April 13-21, 2011.

Meetings were conducted on the following themes:
(1) Atmospheric dispersion & Aerosol studies
(2) Instrumentation for measurements in sodium
(3) R&D needs of sodium fast reactors
(4) Severe accidents R&D needs for sodium fast reactors.

Delegation from CEA and participants from IGCAR with Shri S.C.Chetal, Director, IGCAR during the meeting

The Specialist meeting on "Atmospheric Dispersion and Aerosol Release" was held on April 13, 2011. Welcoming the CEA delegates, Shri S. C. Chetal, Director, REG, IGCAR, in his introductory remarks, briefly outlined the genesis of the CEA-IGCAR collaborations. Dr. C. Latge, thanked IGCAR for organizing this meeting and briefly outlined the status of the various on-going collaborative projects. After a brief introduction of all the participants, presentations were made by Shri P.T. Rakesh on implementing agreement on mesoscale and microscale atmospheric dispersion for impact studies on nuclear sites, Dr. V. Subramanian on "Status of Sodium Aerosol Carbonation Studies at IGCAR" and Shri Amit Kumar on the activities on "Sodium Aerosol Studies in Cover Gas Region at IGCAR". Dr. C. Latge made

Delegates from CEA holding discussions with Shri S.C.Chetal, Director, IGCAR and participants from IGCAR during the meeting "Instrumentation for Measurements in Sodium"

Conference/Meeting Highlights

Shri S.C.Chetal, Director, IGCAR and senior colleagues of the Centre during discussions with the members of CEA delegation during the meeting on "R&D Needs of Sodium Fast Reactor"

presentations on behalf of CEA highlighting the activities and the salient results on all the above mentioned topics. The technical meeting was followed by a visit to the sodium aerosol facilities and atmospheric dispersion facilities and a demonstrative experiment was carried out in the presence of the visitors in order to show the possibilities for future experiments.

The second meeting on the theme "Instrumentation for Measurements in Sodium" was conducted during April 14-15, 2011. The meeting started with the introductory remarks by Shri P.Kalyanasundaram, then Director, FRTG followed by Dr. C. Latge of CEA, France. Presentations were made on various the topics including core monitoring, detection sensors and in-service inspection by delegates from IGCAR and CEA. A discussion on the potential areas of future collaborations also took place.

The third meeting on April 18, 2011 was conducted on the theme "R&D needs of Sodium Fast Reactors". The meeting had presentations on overviews of ongoing R&D for Sodium Fast Reactors, R&D need for CFBR, R&D facilities at CEA and experimental facilities at IGCAR. The team then visited Steam generator test facility, Large component test rig, In-Sodium testing, Sodium water reaction test rig and SAMRAT facilities.

Participants from the CEA Delegation with Dr. Baldev Raj, the then Director, IGCAR, during the concluding session on April 21, 2011

The fourth and final meeting was on "Severe accidents R&D needs for Sodium Fast Reactor" during April 19-21, 2011. Dr. P. Chellapandi, Director, NSEG presented an overview on severe accident studies at IGCAR and the CEA gave an overview of the R&D programme in severe accidents. A number of speakers presented on safety aspects, studies on accidents, fuel coolant interaction, sodium fire, qualification of fire resistant cables, tiles and extinguishers and imaging sodium fires.

The concluding session on April 21, 2011 reviewed the "Status on sodium cooled fast reactors, current and potential collaborations". The summary of the Technical meetings were also presented in the concluding session.

(Reported by M. Sai Baba, RMG)

News & Events

Inauguration of UGC-DAE CSR, Kalpakkam Node

April 28, 2011

Dr. S. Banerjee, Chairman, AEC & Secretary, DAE, Dr. Baldev Raj, the then Director, IGCAR, Prof. Ajay Gupta, Centre Director, UGC-DAE CSR, Indore Centre, Dr. G. Amarendra, Scientist In-Charge, UGC-DAE CSR, Kalpakkam Node during inauguration of Kalpakkam Node

UGC-DAE CSR, Kalpakkam Node building was inaugurated by Dr. S. Banerjee, Chairman, AEC & Secretary, DAE on April 28, 2011 in the presence of Dr. Baldev Raj, the then Director, IGCAR and Prof. Ajay Gupta, Centre Director, UGC-DAE CSR, Indore Centre. Senior scientists from IGCAR as well as research scholars participated in the inaugural function. The construction of the building had commenced in April 2009 and was completed within two years. Advanced experimental facilities have been installed in physical and engineering science wings and collaborations with various academic institutes and universities have been initiated using these facilities.

(Reported by G. Amarendra, UGC-DAE CSR, Kalpakkam Node)

Dr. Baldev Raj, the then Director, IGCAR, Shri S.C.Chetal, Director, IGCAR and senior collegues of the Centre with Dr. Sr. Jasintha Quadras, FMM Principal, Dr. Geetha Swaminathan, Vice Principal & Associate Professor, Dr.Rukmani Srinivasan, Associate Professor and Dr.Juliana Joe, Associate Professor from Stella Maris College during the signing of MoU

A Memorandum of Understanding (MoU) for academic and neighbourhood development was signed between IGCAR and Stella Maris College, Chennai on April 29, 2011. A team from Stella Maris College led by Dr. Sr. Jasintha Quadras, FMM, Principal, Shri S. C. Chetal, Director, IGCAR, Dr. Baldev Raj, the then Director, IGCAR and Senior Colleagues of the Centre were present during the signing of MoU.

Visit of Dignitaries

Delegation from M/s. Terrapower, USA with Shri S.C.Chetal, Director, IGCAR and senior colleagues of the Centre

A delegation from M/s. Terrapower, USA, led by its executive vice president Mr.David McAlees, along with Shri Ashok Misra, Chairman-India, Head of Global Alliances visited the Centre on May 20, 2011. The team had discussions with Shri S.C.Chetal, Director, IGCAR, and senior colleagues of the centre on possible avenues of collaboration.

IGC Colloquium on "Fukushima –Regulator Perspective" was delivered by Shri S.S.Bajaj, Chairman, Atomic Energy Regulatory Board on May 13, 2011

Shri S.S.Bajaj, Chairman, Atomic Energy Regulatory Board during his visit to the Centre on May 13, 2011 delivered the "IGC Colloquium " on "Fukushima –Regulator Perspective ".

Forthcoming Meeting / Conference

6th International Conference on
CREEP, FATIGUE AND CREEP-FATIGUE INTERACTION (CF-6)

January 22-25, 2012

The 6th International Conference on Creep, Fatigue and Creep-Fatigue Interaction (CF-6) will be organised at IGCAR during January 22-25, 2012. CF-6 is being organised jointly with the Kalpakkam Chapter and the Metal Sciences Division of The Indian Institute of Metals. The conference is sponsored by Board of Research in Nuclear Sciences, India and Co-Sponsored by The Japan Society of Mechanical Engineers, The Society of Materials Science, Japan, The Korean Society of Mechanical Engineers and The Korean Institute of Metals and Materials. Eminent experts in the areas of creep, fatigue, creep-fatigue interaction, materials development and high temperature design will be invited to share their knowledge and perspective on these topics during this conference

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Best Paper and Poster Awards

Dr. Baldev Raj, Dr. T. Jayakumar, Dr. John Philip, Shri K Arunmuthu, Shri P Arun Kumar and Shri T Saravanan were awarded the "Ron Halmshaw Award" for paper titled "Automatic detection and sizing of pores from radiographic images using the Hough transform" for the year 2010 by British Institute of Non-Destructive testing as the Best Paper that appeared in Insight 2010.

Shri K. Krishna Prasad, Smt R. Vijayashree, Shri V. Rajan Babu, Shri M. Krishnamoorthy, Shri S. Raghupathy and Dr. P. Chellapandi NSEG received the Best Paper award for paper titled "Routing of Tubes in Control Plug to Facilitate Manufacture and Assembly" presented in the National Conference on Design and Manufacturing (NaConDM2011) held at Indian Institute of Technology, Madras during May 2011.

Awards & Honours

Dr. S. Venugopal, GRIP, MMG has been elected as a "Fellow of American Society of Materials (ASM)" for his significant contributions in the area of "Deformation processing of materials"

Dr.John Philip, NDED, MMG was awarded the MRSI medal-2011 by Materials Research Society of India and he has been given Membership in the American Nano Society.

Shri Sumantra Mandal, MTD, MMG has been awarded the "Sudharshan Bhat Memorial Prize" for best Ph.D thesis in Metallurgical and Materials Engineering from IIT-Madras, Chennai.

Our New Director

The then Director Dr. Baldev Raj, greets Shri S.C.Chetal on his assumption of charge as Director, IGCAR

Shri S.C.Chetal, Director, Reactor Engineering Group has taken over as Director, Indira Gandhi Centre for Atomic Research, Kalpakkam, with effect from 1st May, 2011.

Shri S.C.Chetal, a graduate in Mechanical Engineering, is from the 14th Batch of Training School of Bhabha Atomic Research Centre. Since his joining the Indira Gandhi Centre for Atomic Research in 1971, he has been engaged in the field of fast reactor engineering and has played a key role in Indian fast reactor programme. He has made valuable contributions to the design and technical support for operation of Fast Breeder Test Reactor. He is the principal design engineer of indigenously developed 500 MWe Prototype Fast Breeder Reactor (PFBR), now under advanced stage of construction by BHAVINI at Kalpakkam. His contributions to PFBR include design, research & development, materials and manufacturing technology and enabling and enhancing the capabilities of Indian industries in the manufacturing of nuclear components. He is Fellow of the Indian National Academy of Engineering. He has won a number of awards and honours, notable among them include; INS Award (2003) from Indian Nuclear Society, VASVIK Award (2007) and National Design Award (2007), from Institution of Engineers (India).

The editorial responsibility of the IGC Newsletter has been entrusted to a new committee. The committee under the convenorship of Dr. M. Sai Baba has worked relentlessly in bringing out the news letter in time and with improved quality for each of the issues. The contributions of the members of the outgoing editorial committee: Shri Utpal Borah, Dr. K. K. Satpathy, Shri N. Desigan, Shri S. Varadharajan, Shri C. Jayakumar and Shri J. Daniel Chellappa is greatly acknowledged.

The new committee with the following members is taking over the responsibility from the next issue: Dr. M. Sai Baba, Chairman, Editorial Committee Members: Dr. K. Ananthasivan, Shri M.S.Chandrasekar, Dr.N.V. Chandra Shekar, Dr.C.Mallika, Dr. K.S. Narayanan, Shri V.Rajendran, Dr. Saroja Saibaba and Dr. Vidya Sundararajan.

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