In this Issue:
- An Overview of Welding and Fabrication .....1....
- DAE in Parliament.....15....
- DAE is proud of it’s Padma Awardees....23....
- Public Awareness Activities....23....
### PERFORMANCE OF NUCLEAR POWER STATIONS 2013-2014

 Operated by The Nuclear Power Corporation of India Limited (NPCIL)

<table>
<thead>
<tr>
<th>Station</th>
<th>Unit</th>
<th>Availability Factor (%)</th>
<th>Capacity Factor (%)</th>
<th>Generation Million Units</th>
</tr>
</thead>
<tbody>
<tr>
<td>Tarapur Atomic Power Station</td>
<td>1*</td>
<td>96</td>
<td>94</td>
<td>1322</td>
</tr>
<tr>
<td>(2 x160 MWe + 2 x 540 MWe)</td>
<td>2</td>
<td>58</td>
<td>58</td>
<td>806</td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>81</td>
<td>79</td>
<td>3739</td>
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<tr>
<td></td>
<td>4</td>
<td>88</td>
<td>85</td>
<td>4017</td>
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<tr>
<td>(* TAPS-1 was under refueling from July 2012 to January 2013)</td>
<td>2</td>
<td>95</td>
<td>96</td>
<td>1688</td>
</tr>
<tr>
<td>Rajasthan Atomic Power Station</td>
<td>3</td>
<td>99</td>
<td>101</td>
<td>1946</td>
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<tr>
<td>(1 x 200 MWe + 4 x 220 MWe)</td>
<td>4</td>
<td>89</td>
<td>92</td>
<td>1772</td>
</tr>
<tr>
<td></td>
<td>5</td>
<td>100</td>
<td>106</td>
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<td></td>
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<td>93</td>
<td>70</td>
<td>1354</td>
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<td>(2 x 220 MWe)</td>
<td>2</td>
<td>51</td>
<td>40</td>
<td>761</td>
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<td>Narora Atomic Power Station</td>
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<td>77</td>
<td>1490</td>
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<td>(2 x 220 MWe)</td>
<td>2</td>
<td>88</td>
<td>62</td>
<td>1214</td>
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<tr>
<td>Kakrapara Atomic Power Station</td>
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<td>97</td>
<td>1662</td>
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<td>1896</td>
</tr>
<tr>
<td>Kaiga Atomic Power Station</td>
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<td>82</td>
<td>1587</td>
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<tr>
<td>(4 x 220 MWe)</td>
<td>2</td>
<td>99</td>
<td>90</td>
<td>1740</td>
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<td></td>
<td>3</td>
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<td>91</td>
<td>1759</td>
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<td></td>
<td>4</td>
<td>79</td>
<td>75</td>
<td>1454</td>
</tr>
<tr>
<td><strong>Total (Capacity 4680 MWe)</strong></td>
<td></td>
<td><strong>88</strong></td>
<td><strong>83</strong></td>
<td><strong>34235</strong></td>
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</table>

NPCIL recorded an estimated operating profit of ₹ 2117 Crore during 2013-14

**Cover page Photo:**
Flag Hoisting Ceremony on the occasion of Republic Day of India, January 26, 2014 by Dr. C. B. S. Venkataramana AS, DAE Head Quarters Mumbai, India
AN OVERVIEW OF WELDING AND FABRICATION ASPECTS DURING MANUFACTURE OF NUCLEAR REACTOR COMPONENTS FOR 500MWe PROTOTYPE FAST BREEDER REACTOR

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Distinguished Scientist and Chairman and Managing Director Bharatiya Nabhikiya Vidyut Nigam Limited (BHAVINI)

Prototype Fast Breeder Reactor (PFBR) is a 500MWe pool type, sodium-cooled nuclear reactor, which is presently in an advanced stage of construction on the southeastern Indian coastline at Kalpakkam with physical progress of 96.5% at the end of December 2013 (Figure-1). BHAVINI is constructing this new technology nuclear reactor which puts India to higher pedestal of technological excellence. On completion, this reactor will not only produce 500MWe power but also pave way for abundant power production using indigenous materials & technology and will open up the gateway for energy security to the country for future. Indira Gandhi Centre of Atomic Research (IGCAR) has developed the technology for the Indian Fast Breeder Reactors. Indigenous PFBR is the outcome of several years of rigorous R&D effort by scientists, metallurgists and engineers of IGCAR.

The boundaries of sodium systems of Prototype Fast Breeder Reactor is designed so as to have an extremely low probability of-leakage, rapidly propagating failure and rupture under the static & dynamic loads expected during various operating conditions. The degradation of material properties (e.g. effect of sodium, temperature and irradiation), transients, residual stresses, flaw size etc. are the important considerations, which were taken into account. Austenitic stainless steels of grade 316LN were used as major structural materials for the primary and secondary sodium systems of PFBR. Versatile types of systems and varieties of components with complex constructional features require diversities in welding and fabrication processes for the PFBR components. The quantum of welding and fabrication too are fairly large for the large reactor equipment of PFBR. High operating temperature of various systems causing high stresses are minimized by designing thin walled structure. Most of the Nuclear Steam Supply System (NSSS) components are thin walled and require manufacturing in separate nuclear clean halls to achieve high levels of quality. High distortion in stainless steels due to high thermal expansion and low thermal conductivity offers a challenges to the fabricators to achieve stringent tolerance in large size PFBR components. The welding standards and acceptance criteria of PFBR equipment are more stringent compared to ASME/other International standards. Various control measures and quality assurance are instituted for reactor equipment during each and every stage of raw material procurement, welding, fabrication, non-destructive examinations, testing, handling, erection and post erection preservation to ensure high degree of reliability against failure for the design service life of 40 years. This paper highlights the challenges involved in welding and fabrication of few critical nuclear reactor equipment/systems of the 500MWe Prototype Fast Breeder Reactor.

Introduction

Prototype Fast Breeder Reactor consists of Primary Sodium Circuit (PSC), Secondary Sodium Circuits (SSC), Safety Grade Heat Removal Circuits (SGDHRC) and Steam-Water Circuit (SWC). The Figure-2 shows the PFBR flow chart and Figure-3 shows the reactor assembly. The primary sodium circuit removes the nuclear heat generated in the core and transfers it to the secondary sodium circuits through Intermediate Heat Exchangers (IHXs). The secondary sodium circuits, in turn, transfers the heat to steam/water circuit through Steam Generators (Sgs).
The primary liquid sodium is radioactive. Therefore, radioactive primary sodium is not used directly to produce the steam. In addition, the secondary sodium circuit in between primary sodium circuit and steam-water circuit is envisaged to prevent carryover of hydrogenous materials and reaction products (water, steam, hydrogen, sodium hydroxide) into the core, in case of a sodium-water/steam reaction incident in the Steam Generators.

The primary sodium circuit consisting of core, Primary Sodium Pumps (PSP), Intermediate Heat Exchangers (IHXs), primary pipe connecting the pumps and the grid plate, is contained in a single large diameter vessel called Main Vessel (MV). The main vessel has no penetration and is welded at the top to the Roof Slab (RF). The main function of Roof Slab (RS) is to provide thermal and biological shielding in the upper axial direction from the hot sodium pool and acts as a part of primary containment boundary supporting various reactor components. The core subassemblies are supported on the Grid Plate (GP), which in turn is supported on the core support structure (CSS). As a matter of abundant precaution, a Core Catcher (CC) is provided just below the core support structure. The Core Catcher is designed to prevent the core debris reaching the main vessel when seven fuel assemblies melt and ensures the cooling of the debris by natural convection of sodium.

The Main Vessel (MV) is surrounded by the Safety Vessel (SV) closely following the shape of the Main Vessel, with a nominal gap of 300mm which is large enough to permit ultrasonic inspection of the vessels using a robotic inspection vehicle. The space between the Main and Safety Vessels is small enough to keep the sodium level above the inlet windows of IHX ensuring continued cooling of the core in case of a leak of main vessel. Liquid sodium at 397°C is circulated by two primary sodium pumps through the core and in turn gets heated to 547°C.

The non-radioactive secondary sodium is circulated through two independent secondary loops, each having a secondary sodium pump, two IHX’s and four Steam Generators (SG’s). The primary and secondary sodium pumps are vertical, single stage and single suction centrifugal type with variable speed AC drives. The Steam Generators (SG) are vertical, once through, shell and tube type heat exchangers with liquid sodium flowing in the shell side and water/steam flowing in the tube side.

Diversity of activities, diversity of environment at different places of works, ever changing scenarios, difficult field conditions to work and perform, diversity of people and skills at different periods of project activities are only a few of the complexities that a mega project like PFBR encounters. Large size thin wall reactor vessel fabrication at site, handling and erection, complex geometry of grid plate, roof slab, control plug, control rods, stringent form, position and dimensional tolerances during fabrication and erection and very stringent weld specification make PFBR construction extremely exciting. The quality requirements in every arena of PFBR are far in excess of conventional engineering projects. BHAVINI management has gone beyond the specification requirements whenever required irrespective of the cost involved for high quality steel production for reactor equipment with stage by stage inspection from ladle to
product for raw materials, deployment of high quality welding consumables, deployment of highly skilled & qualified welders and creation of Nuclear Clean Halls for equipment manufacture. Highest level of quality control measures are imparted during every stage of welding, fabrication, Non-Destructive Examinations (NDE), testing, handling, erection and post erection preservation of reactor components. The positive experience of achievements in the field of new technology is the matter of pride to the nation. The PFBR has overcome many high technology manufacturing challenges by successful fabrication of critical, over dimensional reactor equipment with close tolerances. Many new and innovative procedures/techniques were developed for erection of reactor equipment meeting all the stringent specification requirements. The project has moved with robust demonstration of Indian technological capability. The following paragraphs highlight the selection of materials, welding processes and challenges involved in fabrication of few critical nuclear reactor equipment/systems of 500MWe Prototype Fast Breeder Reactor.

Selection of Materials and Welding Process

The major factors considered for the selection of materials include operating environment, availability of design data in nuclear codes, International experience, and quality & safety parameters. The principal material of construction is austenitic stainless steel grade 316LN (Cr: 17-18%, Ni: 12-12.5%, Mo: 2.3-2.7%, Mn: 1.6-2.0%, C: 0.024-0.03, N: 0.06-0.08%). This low carbon nitrogen alloyed stainless steel provides required material properties, ensures freedom from sensitization during welding and inter-granular corrosion of the components. In addition, this steel also possesses excellent high temperature mechanical properties. SS316LN plates were procured in solution annealed, pickled and passivated condition. During material procurement, specimens of the materials were subjected to chemical examination, metallographic examination, test for delta ferrite, inclusion content test, intergranular corrosion test as per ASTM A262, Practice E. During material procurement, plates were subjected to thorough Visual Examination/Liquid Penetrant Examination (LPE) and 100% Ultrasonic Examination (UE) with minimum 10% overlap of previous scan to ensure soundness of the plate. Grain size and chemical composition of materials have been precisely specified with upper and lower values to optimize the mechanical and creep properties. During material procurement, high temperature tensile test is also carried out in addition to tensile test at ambient temperature on the specimens to evaluate and ascertain the properties for service conditions. The raw materials like plates, forgings, tubes, pipes, bars etc. are procured as per approved quality assurance plan with stage by stage inspection to assure the quality event hough failure probability for raw material is low.

<table>
<thead>
<tr>
<th>SR</th>
<th>Defects</th>
<th>Tolerance</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Mismatch (For Both Sides weld)</td>
<td>For t &lt; 5mm, t/4mm max.</td>
</tr>
<tr>
<td></td>
<td>Mismatch (For single Side weld)</td>
<td>t = 5mm, t/10 + 1mm max.</td>
</tr>
<tr>
<td>2</td>
<td>Slope on welding materials of different thickness</td>
<td>Slope = t/4</td>
</tr>
<tr>
<td>3</td>
<td>Reinforcement (Face side)</td>
<td>Reinforcement = W/10 + 1mm</td>
</tr>
<tr>
<td>4</td>
<td>Reinforcement (root side) without back gouging</td>
<td>Reinforcement = t/20 +0.5mm OR 1.5mm max.</td>
</tr>
<tr>
<td></td>
<td>t= thickness of thinner part</td>
<td></td>
</tr>
<tr>
<td>5</td>
<td>Unfilled groove/root concavity</td>
<td>NIL</td>
</tr>
<tr>
<td>6</td>
<td>Undercut</td>
<td>NIL</td>
</tr>
<tr>
<td>7</td>
<td>Arc bite</td>
<td>NIL</td>
</tr>
<tr>
<td>8</td>
<td>Lack of penetration</td>
<td>NIL</td>
</tr>
<tr>
<td>9</td>
<td>Lack of fusion</td>
<td>NIL</td>
</tr>
<tr>
<td>10</td>
<td>Any type of crack</td>
<td>NIL</td>
</tr>
<tr>
<td>11</td>
<td>Arc spatter</td>
<td>NIL</td>
</tr>
</tbody>
</table>

Table 1: Acceptance criteria for weld joints

The welding and fabrication of PFBR equipment are carried out by combination of Gas Tungsten Arc Welding (GTAW) and Shielded Metal Arc Welding (SMAW) processes. The welding is carried out using 16-8-2 filler wires and E 316-15 electrodes with controlled heat input to minimize the distortion and dimensional deviations. The welding
procedure is qualified with stringent destructive and non-destructive examinations & testing before executing welding on the actual job. The acceptance limits for the joints are indicated in table-1. The qualification test coupons were subjected to all the non-destructive examinations applied in fabrication of actual job. During qualification, weld joints were subjected to thorough visual examination, Liquid Penetration Examination (LPE), Radiography Examination (RT), longitudinal tensile test at room temperature, transverse tensile test at room temperature and high temperature (550°C), bend tests, Charpy impact test, delta ferrite content test, Inter Granular Corrosion (IGC) test and metallographic examination for the complete transverse section of the weld. The QA, QC and inspection stages are covered 100% on all welds at various stages of manufacture. Root and final pass LPE and 100% radiography examination are done for all the job weld joints. In case radiography for the job weld joint is not possible due to practical limitations, the volumetric examination of weld joint are carried out by ultrasonic examination.

Production test coupons are welded for every 20m of production weld length for each type of weld joints adapting same process parameters of job welds for controlling and monitoring the weld quality during fabrication. The production test coupon undergoes all the destructive and non-destructive testing carried during procedure qualification. No re-rolling is permitted after welding on the components which may induce un-quantified stresses on the weld joints.

Manufacture of over dimensional Vessels and Equipment

The manufacture of reactor equipment is carried out in separate nuclear clean hall conditions as per PFBR specifications. The reactor assembly components and sodium system equipment fabrication has been a great challenge. The welding with stringent tolerances along with high distortion tendency of stainless steels makes the fabrication extremely challenging.

Manufacture of over dimensional components such as Main Vessel (12.9m diameter, 12.8m height, thickness varying from 25mm to 40mm, weight 135t), Safety Vessel (13.5m diameter, 12.5m height, thickness varying from 15 to 20mm, weight 110t), Inner Vessel (12.2m/6.35m diameter, 9.1m height, thickness varying from 15 to 20mm, weight 60t) involve die pressing of large size dished end and conical petals. The solution annealing of cold worked petals is a mandatory requirement if strain exceeds 10%. Innovative welding techniques were deployed for defect reduction, distortion control and reduction of heat affected zone. The forming techniques and bending methods were qualified with various examinations and testing and many trials were conducted on the mock-up for establishing the process parameters prior to taking up the actual job. Profile measurement of large sized petals requires a special technique of inspection employing non contact type Swing Arm Gauge.

Main Vessel has three cylindrical shells with outer radius of 6450 mm & 25mm thickness and having dished end at bottom with three different radius and curvatures. The design and fabrication of Main Vessel has the typical weld joints of single 'V', Double 'V', Intricate 'K' Type and Triple point weld joints. Due to large diameter to thickness ratio, utmost care is required during welding to avoid distortion and for maintaining the dimensions within the tolerance. Distortion during welding is minimized by ensuring minimum heat input/unit length of weld, welding on both sides wherever possible, sequential welding, simultaneous welding by two or more welders in case of large length welds, use of minimum bevel angles etc. The tolerance achieved during manufacture of above vessels is within specified 10mm for the 12 meters diameter, which is one of the great accomplishments in PFBR. This was possible only due to special welding techniques and various step by step controls during fabrication.

In addition to radiography examination of Main Vessel weld joints, ultrasonic examination is specified as an additional requirement for generation of Pre-Service Inspection (PSI) data, which could be later used as base-line data for In-Service Inspection (ISI). After completion of fabrication, the weld joints of Main Vessel were subjected to Helium Leak Test (HLT) during which the local leak rate has not exceeded 1X10^{-8} Pa-m³/Sec. Figure-4 shows the fabricated Main Vessel and Safety Vessel at specially constructed site assembly shop at PFBR site at Kalpakkam.

Thermal Baffle (Figure-5) in PFBR has two large concentric cylindrical shells, inner and outer shells of diameters 12.4m (thickness 20, 25mm) and 12.6m (thickness 20, 25 and 50mm) respectively and fabrication is one of the difficult and challenging task.
Roof Slab (RS) is a box type massive structure of ~230 tones made mostly from special carbon steel plates confirming to AFNOR- A48P2 (modified) material. The RS acts like a top shield above the main vessel and supports components such as intermediate heat exchangers, decay heat exchangers, large rotatable plugs, small rotatable plugs, control plug etc. PFBR also involves dissimilar joint welding between the carbon steel (A48P2) and austenitic stainless steel (316LN) at integration location of the Roof Slab and Main Vessel. This welding is carried out by combination of GTAW and SMAW processes using ER 309L and E 309-16 welding consumables with controlled heat input to minimize the dilution of carbon. As about 2500 tones of load would act on the critical integration weld joint between the main vessel and roof slab, qualified welding procedure with foolproof quality is inevitable.

The weld between sodium component i.e. primary pipe and grid plate (Figure-6) that supports the fuel subassemblies cannot be accessed for In-Service Inspection and therefore requires extra-ordinary skilled welders. Any failure of above weld during reactor operation would lead to inefficient decay heat removal from the reactor core, which must be avoided. Space constraints and lack of accessibility make the welding and inspection challenging. Highly skilled welders were deployed for welding between the primary pipes to grid plate meeting arduous specification requirements.

Manufacture of Core Catcher and Core Support Structure

PFBR has 181 fuel subassemblies. Core catcher is an in-vessel cooling device for post-accident heat removal of the core debris resulting from Beyond Design Basis Event (BDBE) of total instantaneous blockage to a single fuel sub-assembly. During a single sub assembly melt-down accident, the molten fuel along with structural material (core debris) in contact with liquid sodium will be fragmented and are expected to settle on the bottom surface of Main Vessel (MV) in the absence of core catcher. The decay heat generated within debris bed may lead to possible failure of main
vessel. Hence, the core catcher serves as an in vessel core debris retention device and provides Post Accident Heal Removal (PAHR) of debris by natural circulation. Considering its importance, high standard quality control is essential during fabrication.

The core support structure (CSS) is an important structure, provides support for the core. The maximum diameter of the core support structure is 7.83 meters. The CSS consists of top plate and bottom plate interconnected by number of vertical stiffeners. The vertical stiffeners are arranged in square grid pattern at the center and in radial pattern at the periphery of the structure.

A number of cutouts are provided in the top and bottom plates and in vertical stiffeners to enable access for welding to all the regions during manufacture and to reduce the total weight without affecting the overall stiffness of the structure. Figure-7 shows the welding and fabrication activities of core support structure and integration with MV in the site assembly shop at PFBR site. It is nearly impossible to repair any defect in the Core Catcher and Core Support Structure after their installation in the reactor. Hence, these components have to be fabricated with utmost care to a very elaborate quality control scheme.

The amount of welding involved during manufacture of CSS is too large. In case of distortion of CSS during reactor operation due to residual stresses, it may cause disturbance of the core assemblies and its control mechanisms. Therefore, the entire core support structure after rough machining of the flange underwent stabilization heat treatment at 530°C for 660 minutes for dimensional stability. The heat treatment was carried out with nitrogen purging to minimize the oxygen affects. After completion of heat treatment, the surfaces of austenitic stainless steel were subjected to pickling followed by passivation. Mixture of 70% concentration nitric acid (HNO3) solution (10-20% volume), 40% concentration hydrofluoric (HF) acid (1-3% by volume) and de-mineralized (DM) water (balance volume) is used for pickling operation. Mixture of 70% concentration nitric acid (HNO3) solution (10-20% volume) and DM water (balance volume) is used for passivation to get homogenous passive chromium oxide layer on the surfaces.

Challenges in fabrication of Intermediate Heat Exchangers

Intermediate heat exchanger (IHX) is a shell and tube type, counter current, sodium to sodium heat exchanger. IHX is a very important, massive, over dimensional (~42 tones in weight, ~2 meters in diameter and ~18 meters in length) and critical component of reactor, as it transfers heat from the radioactive primary circuit sodium to non-radioactive secondary circuit sodium forming the boundary between these two circuits.

The tube bundle of each IHX has 3600 nos. of straight seamless tubes (OD19mmX0.8WT). Each tube is supported with anti-vibration belts at 11 locations to minimize the flow induced vibration during reactor operation. The fabrication of PFBR Intermediate Heat Exchangers (IHX) involves various challenges. There are 49700 nos. of ferrule supports required to be welded on anti-vibration belt for each IHX maintaining the pitch with very tight tolerances, which is one of the challenging tasks.

The tubes are initially rolled and then seal welded to the either ends of top and bottom tubesheets by autogenous pulsed GTAW process. The rolling provides strength to the joint and seal weld is to ensure leak tightness for the joint. Even though conventional heat exchanger tube to
tubesheet joints are done first by welding and then rolling, PFBR tube to tubesheet joints are executed first by rolling using mechanical tube expanders and then welding to avoid stresses on the seal welds during tube expansion step. The Helium Leak Test (HLT) is done at tube bundle stage in addition to the final fabrication stage, during which the local leak rate shall not exceed 10-8 Pa·m3/s.

Subsequently, the shell welding is carried out which is extremely difficult and challenging task due to small gap between the tube bundle and surrounding shell. Special methodology and arrangements were made to avoid arc strike or fusion on the tube during shell welding around the tube bundle.

A mechanical hardfaced seal arrangement (Figure-10) at the interface of the IHX outer shell and the inner vessel (IV) stand pipe in the reactor is the chosen design concept to ensure leak tightness in the IHX penetrations. Based on the radiation dose rate and shielding considerations during maintenance, handling and decommissioning, nickel based RNiCr-B hardfacing alloy (Colmonoy-5) is chosen to replace the traditionally used cobalt based stellite alloys to improve the resistance to high temperature wear and to avoid galling of mating surfaces in sodium environment. Stellite has Cobalt-60 which emits hard gamma rays and has long half life of 5.3 years. The more versatile Plasma Transferred Arc Welding (PTAW) and Gas Tungsten Arc Welding (GTAW) were used for deposition of Colmonoy-5. The hardfacing on the seal ring which has hardfacing on all the mating surfaces is a challenging task. The diameter is too large and thickness & width is too small. Extensive varieties of trials were conducted on the mockup to optimize the hardfacing process along with heat treatment cycle to obtain minimum distortion deploying special tools and fixtures.

Intermediate Heat Exchanger involves 10mm and 28mm thick borated stainless steel components around the tube bundle and bottom portion confirming to A 887, Type 304 B4, Grade B classification. This helps in reducing neutron population reaching the secondary sodium and thereby reducing the activity in secondary sodium system while flowing through the IHX.

Various types of welding trials were conducted on the test coupons for welding procedure qualification of borated stainless steel components using versatile grades of welding consumables. More than hundred numbers of trial coupons welded using 10 varieties of welding consumables have shown fissures and cracks in the weld. The analysis indicated that commercially available welding consumables do not have good compatibility with SS 304 B4, Grade B borated steel. Special development work was taken-up for the first time in the country by developing a special grade ~1% borated welding consumable (GRINOX-308BRN electrode) specifically for welding of borated steel components of IHX.

Fuel sub-assemblies of PFBR

PFBR has 181 numbers of fuel sub assemblies arranged in triangular pitch. Each fuel sub-assembly consists of a foot and a handling head welded to the central hexagonal sheath, which houses 217 numbers of fuel pins arranged in a triangular pitch. Alloy D9 in 20% cold worked condition (20CW D9) has been chosen for clad and
wrapper tubes of PFBR due to its high resistance for swelling and irradiation creep. Various controls were exercised during manufacture of clad and wrapper tubes (Alloy D9) for achieving 20% cold worked condition.

Pilgering technology is adopted for fabrication of Hexagonal channels (Hexcans) for PFBR, as forming sheets into two halves of a hexagon and joining them by GTAW process involves certain inherent limitations like non-uniform microstructure, inferior mechanical properties and poor corrosion resistance at the weld and Heat Affected Zone (HAZ). Pass schedules were developed for pilgering Hexcans to meet the specification of 20% residual cold work condition and consequent enhanced mechanical properties. Seamless pilgering route has some distinct advantages over other fabrication routes like closer dimensional control, higher material recovery and elimination of weld ensures a uniform microstructure and therefore better mechanical properties.

**Welding and manufacture of Steam Generators**

The Steam Generator (SG) is a vertical, once through, shell and tube type heat exchanger with liquid sodium flowing in the shell side and water/steam flowing in the tube side. Steam Generator is 26 meters in length, 42 tones in weight and about 1.2 meters in diameter. The operating pressure and temperature of steam generator is very high (steam outlet pressure is 172 bars & temperature is 493°C). The operating experience of SG’s in other countries revealed that the tube leakage affects the availability of plant, as sodium and water/steam are separated by a single wall tube in the SG. In case of a crack/failure in tube, high pressure water/steam reacts with shell side sodium and results in an exothermic reaction with evolution of hydrogen, corrosive reaction products and intense local heat. The highly reactive nature of sodium with water/steam requires that the sodium to water/steam boundaries of the Steam Generators must possess a high degree of reliability against failure. This is achieved in design and manufacturing by maximizing the tube and tubesheet integrity and more importantly by proper selection of tube to tubesheet joint configuration.

Modified 9Cr-1Mo material is selected as major material of construction for PFBR Steam Generators. Resistance to loss of carbon to liquid sodium, resistance to stress corrosion cracking in caustic & chloride atmosphere and high creep strength properties made modified 9Cr-1Mo, a candidate material for selection. The tubes are seamless and produced by electric arc melting followed by Electro Slag Refining (ESR) with tight control on inclusion content to achieve sound weld during autogenous welding process between the tube and tubesheet. Ultrasonic examination and eddy current testing is done on the entire tube length in accordance with ASME SEC III Class I. Each tube is subjected to hydro testing as per PFBR specification to ensure the integrity. Long seamless tubes (each 23 meters) are used in order to reduce the number of tube to tubesheet welds and enhance the reliability.

The welding process for tube to tubesheet selected is inside bore, autogenous pulsed GTAW process (Figure-12). Each tube to tubesheet joint is preheated to 200-250°C and subjected to Post Weld Heat Treatment (PWHT) at 760±100°C for 30 minutes. Each tube to tubesheet joint is subjected to thorough visual examination, LPE, radiography using micro-focal rod anode x ray, profile check (concavity, convexity and wall thickness), pressure testing and helium leak testing. Specially designed dial gauge is used to check the weld profile from inside for all the joints. Replica technique is also used to crosscheck the internal weld profile for 1% of the total joints in addition to dial gauge measurement. External profile measurement is done by replica technique for all the weld joints. Acceptance limits of tube to tubesheet weld joints are kept stringent as given in Table 2. After completion of tube bundle, the row wise hydro test is done to check the integrity. Subsequently, shell assembly is carried out around the tube bundle.
Table 2: Acceptance limits of tube to tubesheet weld joints

<table>
<thead>
<tr>
<th>Test</th>
<th>Parameter</th>
<th>Acceptance Standard</th>
</tr>
</thead>
<tbody>
<tr>
<td>Visual/Replica</td>
<td>a) Weld concavity on outside/inside</td>
<td>0.2 mm max</td>
</tr>
<tr>
<td></td>
<td>b) Weld reinforcement on outside / inside surfaces</td>
<td>0.35 mm max.</td>
</tr>
<tr>
<td></td>
<td>c) Weld thinning</td>
<td>0.2 mm max.</td>
</tr>
<tr>
<td>Micro-local rod node x-ray</td>
<td>a) Total pore count (sum of all visible pore diameters) in the entire weld.</td>
<td>2.7 mm max.</td>
</tr>
<tr>
<td></td>
<td>b) Total pore count in 3mm circle anywhere in the weld</td>
<td>0.6 mm max.</td>
</tr>
<tr>
<td></td>
<td>c) Individual pore diameter</td>
<td>0.46 mm max.</td>
</tr>
<tr>
<td></td>
<td>d) Lack of fusion</td>
<td>NIL</td>
</tr>
<tr>
<td></td>
<td>e) Lack of penetration</td>
<td>NIL</td>
</tr>
<tr>
<td></td>
<td>f) Cracks</td>
<td>NIL</td>
</tr>
<tr>
<td></td>
<td>g) Undercut</td>
<td>NIL</td>
</tr>
<tr>
<td>Helium Leak Testing</td>
<td>a) Local leak</td>
<td>$2.66 \times 10^4$ Pa\cdot m$^{-2}$s$^{-1}$</td>
</tr>
<tr>
<td></td>
<td>b) Global leak</td>
<td>$6.66 \times 10^4$ Pa\cdot m$^{-2}$s$^{-1}$</td>
</tr>
</tbody>
</table>

Even though Shielded Metal Arc Welding (SMAW) process is permitted as per PFBR specification, 100% GTAW process alone is executed for the shell welds to meet impact properties of the welds. Hot wire and cold wire Gas Tungsten Arc Welding (GTAW) process is adapted for welding based on the practical possibility and feasibility. The SG fabrication involves welding of 12mm, 30mm and 90mm thick metal weld joints. The 12mm thick shell welding around the tube bundle is carried out by 100% cold wire GTAW process. The 30mm thick shell welding is carried out by combination of cold and hot wire GTAW process. As amount of weld metal to be deposited during fabrication is high, it was decided to carry out hot wire GTAW process on 30mm thick shells wherever possible to improve the weld deposition rate. The most important benefit from the use of a hot wire system is the virtual reduction of porosities from the weld deposits. Literature reveals that I2R heating of the filler metal wire as it approaches the weld puddle drives off most of the volatile surface contamination. Since hydrogen or hydrogen containing compounds entrapped on the filler metal surface are a primary cause of porosity, use of hot wire system is beneficial to remove the major source of defect. The use of hot wire current not only increases the deposition efficiency but also considerably reduces dilution, which depends on hot wire amperage and the amount of filler material melted off. The root pass and subsequent pass welding for all 12mm and 30mm thick shells were carried out by manual cold wire GTAW process. There is no much added advantage of carrying hot wire GTAW process on 12mm thick shells, as root and subsequent pass welding is carried out for 4-5mm thickness by manual GTAW process and remaining weld deposition thickness left would be only around 7-8mm.

The welding of 12mm thick slender shells around the tube bundle with outer diameter of 855mm is extremely challenging task due to vicinity of tubes (i.e. sodium/water boundary). Utmost care is required during root pass welding to avoid arc strike/fusion on tubes, as gap between the tube bundle and shell inside diameter (ID) is too small. Eddy current testing is developed and demonstrated on the peripheral tubes to ensure no arc strike/fusion has occurred on the tubes during shell welding.

Due to existence of tube bundle, the internal fixtures/spiders cannot be used for distortion control during welding. As each shell is assembled in 2 halves around the tube bundle, shape correction cannot be carried out by re-rolling, as a conventional industrial practice. In addition, as per PFBR specification re-rolling after welding is not permitted which may induce stresses on weld joints and may cause crack/failure of weld joints during transient reactor operating conditions.

As the gap between tube bundle and shell ID is too small, maximum 0.5% ovality is specified for PFBR SG shell assemblies against maximum 1% permissible ovality as per ASME, section III, NB-4221.1 and ASME section VIII.1.
Division-1, UG-80. Minimum ovality is desirable with a view to reduce the bending stresses on the shell assembly during service conditions. Many attempts were made to continually improve processes, techniques and practices to reduce shell ovality after welding. Tremendous efforts were put on 12mm thick shells to achieve less than 0.5% ovality after welding.

The plates were rolled to form the complete cylindrical shell and were tacked along the long seam locations before parting into two halves for assembly around the tube bundle. As 12mm thick shells after parting at component stage were opening up (Figure-13) due to forming stresses present in the shell before parting, it was extremely difficult to have the set-up of these parted shells around the tube bundle before welding. Enormous numbers of trials were conducted to understand the behavior of shells to meet the final dimensions.

It was noticed that in many of the cases, reason of ovality more than the required after welding was due to peak-in of shells. Shells were peaking in along the long seams due to weld shrinkage and distortion. After many welding trials, finally it was decided to part the shells, pre-camber the half shells with the help of fixtures and carry out Intermediate Stress Relieving at 760±10°C for 1 hour soaking time of these half shells in pre-cambered condition in the range of 8 to 12mm. The above exercise has delivered outstanding results in meeting the required ovality of shell assemblies after welding.

After completion of shell assembly, entire 26 meter length SG is subjected to Post Weld Heat Treatment (PWHT) in a single charge at 760±10°C for 4 hours soaking time to relieve the welding stresses and to get homogenous tempered martensite structure. A dedicated sliding type calibrated electrical furnace of about 30 meters length with thyristered controls is used for PWHT. Special high temperature sustaining rollers were kept beneath the saddle supports of SG during heat treatment to take care of about 240mm thermal expansion during heat treatment. Due to very high as-welded hardness of the modified 9Cr-1Mo welds (typically more than 440 VHN), the handling of Steam Generators before PWHT is done without direct lifting using belts. Saddle supports were made for shifting of SG from fabrication area to furnace area.

Austenitic stainless steel is used as principal constructional material for all sodium systems in the reactor. Ferritic steel of type modified 9Cr-1Mo (Grade 91) has been selected as Steam Generator material, as this steel has excellent resistance for stress corrosion attack in caustic and chloride environment. Therefore, a dissimilar joint between ferritic steel of Steam Generator and austenitic stainless steel of rest of the sodium system is inevitable.

The operating experience of direct austenitic/ferritic junction revealed poor performance in power plants. The use of direct bimetallic joint between austenitic stainless steel and ferritic steel has led to several problems in service like thermally induced cyclic stresses resulting from the difference in coefficients of thermal expansion between the ferritic and austenitic steels (difference in coefficient of thermal expansion of these two materials is ~30-40 percent). With enhanced thermal fatigue damage in fast reactors due to excellent heat transfer of sodium and serious effects of sodium leaks, design and manufacture of transition joint becomes all the more important. Therefore, Alloy 800H material is selected as transition piece between SS316LN and modified 9Cr-1Mo in Steam Generators, as the coefficient of thermal expansion of Alloy 800H lies between coefficient of thermal expansion of modified 9Cr-1Mo & SS316LN materials. After completion of transition joint welding, the inside and outside diameters are machined to the entire length with tight tolerances to remove the root defects for ensuring transition joint is free of mechanical stress concentration for enhancing the resistance to damage resulting by thermal cycling of joint during service.

After completion of fabrication of SG, the shell side and tube side were subjected to hydro test to ensure integrity of the equipment. The hydro test pressure during tube side is very high, i.e. 300 bars. The shell side is subjected
well as shell side with very stringent acceptance criteria. The local leak rate shall not be more than $2.66 \times 10^{-9}$ Pa m$^{-3}$/s and global leak rate shall not be more than $6.66 \times 10^{-9}$ Pa m$^{-3}$/s. In addition to leak tightness, HLT also ensures complete dryness of the component, as the test is done in vacuum mode.

The straightness of 26 meters length fabricated Steam Generator is achieved within 3mm for all the 8 nos. of steam generators, which is one of the engineering excellences in welding technology.

Fabrication of Sodium to Air Heat Exchangers

Nuclear decay heat generated in the core after reactor shutdown has to be removed to avoid temperature rise to maintain the structural integrity of the reactor components and core melt down due to temperature increase. The heat removal during shutdown stage is possible by Safety Grade Decay Heat Removal (SGDHR) system. As per the safety guidelines, failure probability of decay heat removal function shall be less than $10^{-7}$/ry. In order to achieve this value, highly reliable Safety Grade Decay Heat Removal (SGDHR) system is provided in PFBR. Sodium to Air Heat
Exchanger (AHX) in SGDHR loop transfers heat from the intermediate circuit sodium to atmospheric air by natural convection.

The fabrication of Sodium to Air Heat Exchangers (AHX), which is a part of Safety Grade Decay Heat Removal (SGDHR) system, involves many difficulties and challenges. The major material of construction of AHX is Modified 9Cr-1Mo steel (Normalized & Tempered). AHX has sodium inlet and outlet headers with many hot formed pullouts on its surface having OD38.1mmX2.6mm thickness which are connected to a finned tube bundle.

Due to many pullouts, the forming of pullouts by conventional method by heating inside the furnace is not recommended, as the material undergoes repeated heating and cooling cycles during individual forming of pullouts.

Due to many pullouts, the forming of pullouts by conventional method by heating inside the furnace is not recommended, as the material undergoes repeated heating and cooling cycles during individual forming of pullouts.

The forming of pullouts in a group is practically not possible as angular orientation is not same for all the pullouts and varies depending on the profile. The above metallurgical and practical limitations necessitated development of special die, punching tools and heating arrangement for individual forming of the header pullouts. The pullouts were made by heating locally at 950-1100°C using induction heaters followed by local die & punch pressing. Due to many hot formed pullouts, complex geometry, the welding sequence, heat treatment and non-destructive examinations of the welds is an exigent task.

After header pullout forming & heat treatment, the header segments are joined together by circumferential seam welding. Due to limitation in the area for circumferential seam welding of headers (due to existence of pullouts in the circumferential weld area), welding is required to be carried out within narrow area available between the pullouts by zig-zag method (Figure-19), which is one of the critical activity and challenge for welders. After completing welding of pullout header segments, the same is positioned at tube bundle area for welding of header pullout to bend tubes.

![Figure-18: Zig Zag welding of circumferential seam of pullout header segments](image)

![Figure-19: Header pullout to bend tube welding and non-pullout header segment to pullout header segment welding of AHX](image)
Figure-19 shows the skill and difficulty involved in welding of bend tube to header pullout and welding between the header segments, which has very less accessibility for welding.

The tube to tube weld joint is carried out by autogenous pulsed GTAW process in 5G position. The development program for tube to tube welding (Figure-20) was a concentrated effort of almost 18 months. Various different types of trials were conducted for the welding of enormous nos. of joints. As there is no past experience of welding on higher thickness by autogenous welding process, the technology development for tube to tube welding was an exigent task. Welding on higher thickness (as received tube thickness was in the range of 2.8-2.9mm which was within the positive tolerance of specification requirement) by autogenous welding process specifically on modified 9Cr-1Mo material is carried out successfully for the first time in the country; no open literatures could be traced out on the above topic.

**Glimpses of fabrication of sodium piping circuits**

The scope of piping fabrication work is too large due to versatile types of system with varieties of pipes, fittings, valves involved. The piping is connected to large sodium tanks as well as small sodium components such as exchanger economizers, flow meters, thermal mixers, EM pumps, catch pots and relief pots etc. which forms the terminal joints of the sodium piping system. All sodium pipelines inside Reactor Containment Building (RCB) of PFBR are provided with hot guard pipe and are inerted with nitrogen. The guard piping and the containment penetrations require sequential welding and NDT which are unique to PFBR. Limited space at site for the erection of sodium piping along with welding at difficult to access and confined areas makes the work all the more challenging.

Manufacturing of thin and big bore piping with tight tolerances makes the fabrication extremely difficult. With strict rules of sloping to be given to the piping to make conducive for full draining of the sodium loops, the fabrication challenges becomes multifold. The stringent 3 stage helium leak testing-spool stage, after spool erection and after integration with the terminal components is adopted to ensure leak tightness of the system.

Utmost care is to be taken to avoid mixing of welding consumables, tools/tackles maintaining the nuclear clean environment. Various constraints due to limited space, congested layout, cable trays and panels & ventilation systems rooting etc. posed several challenges to sodium pipe line erection which were successfully accomplished in PFBR.
Rigorous Foreign Material Exclusion (FME) practices were adopted during each and every stage of fabrication to ensure absence of foreign material inside the piping. Boroscopic/ Fibroscopic inspection of pipe lines is carried out in the inaccessible areas to ensure FME in the complete piping systems.

Concluding Remarks

Manufacture of reactor assembly and sodium circuit equipment has been a great challenge. The specification requirements, dimensional tolerances and acceptance criteria for PFBR are far more stringent than ASME or many other international standards. It is heartening that the design features have been correctly translated into welding and manufacture in PFBR. The achievements in welding science and technology during construction of Prototype Fast Breeder Reactor is a matter of pride to the nation. Very high standard quality control & quality assurance during welding and fabrication has given adequate confidence on trouble free service from Prototype Fast Breeder Reactor for the designed service life of 40 years.

Acknowledgement

PFBR is outcome of national effort of technology development. Besides IGCAR which is the architect of PFBR, several scientific and educational institutes and industries have participated in development of technologies and successful translation in PFBR. Various units Department of Atomic Energy including Indira Gandhi Centre for Atomic Research (IGCAR), Bhabha Atomic Research Centre (BARC), Nuclear Fuel Complex (NFC), Electronic Corporation of India (ECIL) have contributed enormously in design and special material production. Indian industries too played very significant role in realizing PFBR. The author sincerely acknowledges the contribution made by each and every team member of BHAVINI and industry partners for successful accomplishments of welding and fabrication challenges which were never tried and attained in the past.

About the author:

Dr. Prabhat Kumar joined Bharatiya Nabhikiya Vidyut Nigam Limited (BHAVINI), India, on the day of incorporation of the company. With vast experience of design and manufacturing, QA, R&D and construction and commissioning of different nuclear reactors in the country, he was chosen to launch and steer the programme of breeder nuclear reactors operating on high energy neutrons (Fast Breeder Reactors) which produce more plutonium than consumed as fuel. He was designated as the ‘Project Director’ of Prototype Fast Breeder Reactor (PFBR) from the day BHAVINI was incorporated and was inducted in the Board of Directors of BHAVINI. The author is currently the Chairman and Managing Director on the Board of BHAVINI and a Distinguished Scientist of Department of Atomic Energy of India. He can be contacted at prabhatbhavini@gmail.com
LOK SABHA STARRED QUESTION NO. 369 ANSWERED ON 19.02.2014
SETTING UP OF NUCLEAR POWER PLANTS SHRI HANSRAJ G. AHIR : SHRI M.B. RAJESH :

Question
Will the PRIME MINISTER be pleased to state:
(a) whether the Government proposes to set up Nuclear Power Plants (NPPs) in various States of the country during the 12th Five Year Plan;
(b) if so, the details and the present status thereof along with their capacity, State and location-wise;
(c) the details of the amount released and expenditure incurred on all the under construction NPPs during the last three years and the current year along with the financial assistance, if sought, from any national or international financial institutions for these projects;
(d) whether there is cost escalation in any of these projects and if so, the details thereof and the reasons therefor; and
(e) the steps taken or being taken by the Government to ensure adequate fuel supply for nuclear power reactors in the country?

ANSWER

THE MINISTER OF STATE FOR PERSONNEL, PUBLIC GRIEVANCES & PENSIONS AND PRIME MINISTER’S OFFICE (SHRI V. NARAYANASAMY):

(a) to (e) A statement is laid on the Table of the House.

STATEMENT REFERRED TO IN REPLY TO LOK SABHA STARRED QUESTION NO. 369 FOR ANSWER ON 19.02.2014 BY SHRI HANSRAJ G. AHIR AND SHRI M.B. RAJESH REGARDING SETTING UP OF NUCLEAR POWER PLANTS.

(a) & (b) Yes, Sir. The XII Five Year Plan envisages commencement of work on 19 new Nuclear Power Plants with a total installed capacity of 17400 MW. The details are given in Annexure-1.

(c) The details of the Revised Estimate (RE) and Expenditure incurred for the under-construction Nuclear Power Plants during the last three financial years and the current financial year upto December 2013 is as below:

<table>
<thead>
<tr>
<th>Sr. No.</th>
<th>Project</th>
<th>2010-11</th>
<th>2011-12</th>
<th>2012-13</th>
<th>2013-14</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>RE</td>
<td>Exp</td>
<td>RE</td>
<td>Exp</td>
<td>RE</td>
</tr>
<tr>
<td>1</td>
<td>KGS-3&amp;4</td>
<td>165</td>
<td>139.40</td>
<td></td>
<td></td>
</tr>
<tr>
<td>2</td>
<td>KKNPP 1&amp;2</td>
<td>900</td>
<td>803.67</td>
<td>1000</td>
<td>933.58</td>
</tr>
<tr>
<td>3</td>
<td>KAPP 3&amp;4</td>
<td>550</td>
<td>352.89</td>
<td>1218</td>
<td>1077.38</td>
</tr>
<tr>
<td>4</td>
<td>RAPP 7&amp;8</td>
<td>325</td>
<td>287.71</td>
<td>862</td>
<td>545.73</td>
</tr>
<tr>
<td>5</td>
<td>PFBR</td>
<td>720</td>
<td>605.32</td>
<td>905</td>
<td>631.33</td>
</tr>
</tbody>
</table>

All the values are ₹ in crores.
*Actual expenditure incurred up to December 2013. RE indicates approved budget by concerned PSU.

Legend : KGS : Kaiga Generating Station KKNPP : Kudankulam Nuclear Power Project KAPP : Kakrapar Atomic Power Project RAPP : Rajasthan Atomic Power Project PFBR : Prototype Fast Breeder Reactor
Projects at sr. no. 1 to 4 above, being implemented by Nuclear Power Corporation of India Limited (NPCIL), a public sector undertaking of the Department are funded by a mix of equity and debt. The equity in the last three years and current year has been mobilised from internal resources of NPCIL only and no equity has been released from Government. The debt portion was mobilised from domestic market borrowings and external credit/loan. The PFBR, being implemented by Bharatiya Nabhikiya Vidyut Nigam Limited (BAVINI), is funded through Government equity besides 5% equity contribution from NPCIL. Debt portion (20%) is to be raised after entire equity is utilised.

(d) There is no cost escalation with respect to the approved completion cost of KGS Units–3&4, KAPP Units–3&4 and RAPP Units–7&8 projects. In respect of KKNPP Units–1&2, the initial approved completion cost was 13,171 crore, which was revised to 17,270 crore in May, 2013. The cost escalation has been mainly due to delay in the project completion. There were local protests at site, during September 2011 to March 2012 which severely impeded work and momentum of the project.

In respect of PFBR, the original approved cost envisaged was 3492 crore which was revised to 5677 Crore. The increase of cost was mainly due to the following reasons:

(i) Construction of PFBR was originally planned to be done departmentally. Later the responsibility was entrusted to BAVINI, a public sector undertaking of the Department of Atomic Energy leading to change in tax incidence. Furthermore, Service Tax was introduced after the financial sanction for the project was accorded.

(ii) Being first of its kind reactor in the country, several modifications were carried out during execution of the project and new items were necessitated compared to those envisaged in the original design.

(e) Government has made efforts to augment indigenous uranium supply by opening of new mines and processing facilities. Fuel supply contracts have been signed for import of fuel for reactors under the International Atomic Energy Agency (IAEA) Safeguards. In respect of future projects to be set up with international cooperation, lifetime fuel supply guarantees are being incorporated in the commercial contracts. Fuel linkages for future indigenous projects will be ensured at an appropriate time. PFBR is totally indigenous project and also government has already allocated fuel for PFBR.

Annexure-1

<table>
<thead>
<tr>
<th>Project</th>
<th>Location</th>
<th>Type</th>
<th>Capacity (MW)</th>
<th>Status</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ghavip Units 1 &amp; 2</td>
<td>Gorakhpur, Haryana</td>
<td>PHWR</td>
<td>2 x 700</td>
<td>Administrative approval &amp; financial sanction for the project accorded. Foundation stone laid by Hon. Prime Minister on 13.01.2014. Site infrastructure works are in progress. Start of construction (First Pour Concrete) is scheduled in June 2015.</td>
</tr>
<tr>
<td>CMAPP Units 1&amp;2</td>
<td>Chutka, Madhya Pradesh</td>
<td>PHWR</td>
<td>2 x 700</td>
<td>Pre-project activities (Land acquisition, obtaining statutory clearances, site investigations) in progress.</td>
</tr>
<tr>
<td>Mahi Sardar, Units 1&amp;2</td>
<td>Mahi Sardar, Rajasthan</td>
<td>PHWR</td>
<td>2 x 700</td>
<td>Pre-project activities (Land acquisition, obtaining statutory clearances, site investigations) in progress.</td>
</tr>
<tr>
<td>Kaiga Units 5&amp;8</td>
<td>Kaiga, Karnataka</td>
<td>PHWR</td>
<td>2 x 700</td>
<td>Land available, other pre-project activities initiated.</td>
</tr>
<tr>
<td>FBR Units 1&amp;2</td>
<td>Kalpakkam, Tamilnadu</td>
<td>FBR</td>
<td>2 x 500</td>
<td>It is envisaged to start construction of two more fast breeder reactors of 500 MWe capacity each at Kalpakkam, Tamilnadu, in XII Five Year Plan.</td>
</tr>
<tr>
<td>AHWR</td>
<td>To be decided</td>
<td>AHWR</td>
<td>300</td>
<td>AHWR will be a 500 MWe Nuclear Power Plant. Standing Site Selection Committee of Department of Atomic Energy is examining the issues associated with siting of AHWR at candidate sites. The design of the reactor, at present has been carried out taking site independent input for a coastal area to facilitate sea water based cooling.</td>
</tr>
<tr>
<td>KKNPP Units 3&amp;4</td>
<td>Kudankulam, Tamilnadu</td>
<td>PHWR</td>
<td>2 x 1000</td>
<td>Project was accorded. Administrative &amp; financial sanction. Discussions on General Framework Agreement with Atomexport of Russia in progress.</td>
</tr>
<tr>
<td>JNPP Units 1&amp;2</td>
<td>Jeitsapur, Maharashatra</td>
<td>LWR</td>
<td>2 x 1650</td>
<td>Land acquired. Environmental and CRZ clearances obtained. Site infrastructure and investigation works in progress. Discussions with M/s Areva, France to arrive at project proposal in progress.</td>
</tr>
<tr>
<td>Kovada, Units 1&amp;2</td>
<td>Kovada, Andhra Pradesh</td>
<td>LWR</td>
<td>2 x 1500</td>
<td>Pre-project activities (Land acquisition, obtaining statutory clearances, site investigations) in progress. Discussions with GE Hitachi Nuclear Energy (GEH) to arrive at project proposal are in progress.</td>
</tr>
<tr>
<td>Chhayamithi Virdi Units 1&amp;2</td>
<td>Chhayamithi Virdi, Gujarat</td>
<td>LWR</td>
<td>2 x 1100</td>
<td>Pre-project activities (Land acquisition, obtaining statutory clearances, site investigations) in progress. Preliminary contract for sharing technology details signed with Westinghouse Electric Company (WEC), discussions to arrive at project proposal are in progress.</td>
</tr>
</tbody>
</table>

Legend : PHWR – Pressurized Heavy Water Reactor FBR – Fast Breeder Reactor AHWR – Advanced Heavy Water Reactor LWR – Light Water Reactor
NUCLEAR POWER PLANTS IN THE COUNTRY SHRI BASAWARAJ PATIL:

Question:

Will the PRIME MINISTER be pleased to state:

(a) the total number of nuclear plants in India stating the locations thereof;
(b) the power generation capacity of these nuclear plants; and
(c) keeping in view the nuclear disaster in Japan two years ago, the fresh steps taken by Government for the safety of nuclear plants?

ANSWER

THE MINISTER OF STATE FOR PERSONNEL, PUBLIC GRIEVANCES & PENSIONS AND PRIME MINISTER'S OFFICE (SHRI V. NARAYANASAMY):

(a) to (c) A statement is laid on the Table of the House. *****

STATEMENT REFERRED TO IN REPLY TO RAJYA SABHA STARRED QUESTION NO. 397 FOR ANSWER ON 20.02.2014 BY SHRI BASAWARAJ PATIL REGARDING NUCLEAR POWER PLANTS IN THE COUNTRY.

(a) & (b) There are 20 nuclear power plants with installed capacity of 4780 MW. The details are given below:

<table>
<thead>
<tr>
<th>Location &amp; State</th>
<th>UNITS</th>
<th>Capacity MW</th>
</tr>
</thead>
<tbody>
<tr>
<td>Tarapur, Maharashtra</td>
<td>TAPS-1</td>
<td>160</td>
</tr>
<tr>
<td></td>
<td>TAPS-2</td>
<td>160</td>
</tr>
<tr>
<td></td>
<td>TAPS-3</td>
<td>540</td>
</tr>
<tr>
<td></td>
<td>TAPS-4</td>
<td>540</td>
</tr>
<tr>
<td></td>
<td>RAPS-1*</td>
<td>100</td>
</tr>
<tr>
<td></td>
<td>RAPS-2</td>
<td>200</td>
</tr>
<tr>
<td></td>
<td>RAPS-3</td>
<td>220</td>
</tr>
<tr>
<td></td>
<td>RAPS-4</td>
<td>220</td>
</tr>
<tr>
<td></td>
<td>RAPS-5</td>
<td>220</td>
</tr>
<tr>
<td></td>
<td>RAPS-6</td>
<td>220</td>
</tr>
<tr>
<td>Rawatbhata, Rajasthan</td>
<td>MAPS-1</td>
<td>220</td>
</tr>
<tr>
<td></td>
<td>MAPS-2</td>
<td>220</td>
</tr>
<tr>
<td></td>
<td>NAPS-1</td>
<td>220</td>
</tr>
<tr>
<td></td>
<td>NAPS-2</td>
<td>220</td>
</tr>
<tr>
<td>Kalpakkam, Tamil Nadu</td>
<td>KAPS-1</td>
<td>220</td>
</tr>
<tr>
<td></td>
<td>KAPS-2</td>
<td>220</td>
</tr>
<tr>
<td>Naraora, Uttar Pradesh</td>
<td>KAQA-1</td>
<td>220</td>
</tr>
<tr>
<td></td>
<td>KAQA-2</td>
<td>220</td>
</tr>
<tr>
<td></td>
<td>KAQA-3</td>
<td>220</td>
</tr>
<tr>
<td></td>
<td>KAQA-4</td>
<td>220</td>
</tr>
</tbody>
</table>

* RAPS-1 under extended shutdown since October 2004.

(c) Post-Fukushima, review of safety of all nuclear power plants in operation in the country and those under construction was undertaken by task forces of Nuclear Power Corporation of India Limited (NPCIL) and a committee of the Atomic Energy Regulatory Board (AERB). These reviews have found that Indian nuclear power plants are safe and have margins and features in design to withstand extreme natural events. Recommendations were made in these reviews to take the safety to a higher level, which have mostly been implemented.
LOK SABHA UNSTARRED QUESTION NO.4027 - 19.02.2014
KUDANKULAM NUCLEAR POWER PROJECT- by SHRI P.L. PUNIA; SHRI RAJAIH SIRICILLA; SHRI M. KRISHNASSWAMY:

Will the PRIME MINISTER be pleased to state:

(a) whether the Government has taken note of peoples' protest against Kudankulam Nuclear Power Plant;
(b) if so, the action taken or proposed to be taken by the Government in this regard;
(c) whether India and Russia have recently held discussions on various projects including Kudankulam project; and
(d) if so, the details and the outcome thereof?

ANSWER

THE UNION MINISTER OF STATE FOR PERSONNEL, PUBLIC GRIEVANCES & PENSIONS AND PRIME MINISTER'S OFFICE (SHRI V. NARAYANASAMY):

(a)&(b) Yes, Sir. Public Outreach has been enhanced to spread awareness among the people about the project, nuclear power and its related aspects in order to allay the apprehensions of the people, particularly about issues of safety of the plant and impact on livelihood. A multi-pronged approach has been adopted in this regard. Neighbourhood Welfare Programmes have also been taken up in consultation with the State Government around the site.

(c)&(d) Discussions are held periodically and at various levels with the Russian side on the entire spectrum of the India-Russia civil nuclear cooperation programme, including the Kudankulam Nuclear Power Project. Both the sides have reaffirmed their commitment to the Agreement concluded on December 5, 2008 between the Government of the Republic of India and the Government of the Russian Federation on cooperation in the construction of additional nuclear power plant units at Kudankulam site as well as in the construction of Russian designed Nuclear Power Plants at new sites in the Republic of India and the Agreement between the Government of Republic of India and the Government of the Russian Federation on Cooperation in the use of atomic energy for peaceful purposes.

LOK SABHA UNSTARRED QUESTION NO.4082-19.02.2014
CIVIL LIABILITY FOR NUCLEAR DAMAGE By SHRI M.I. SHANAVAS:

Will the PRIME MINISTER be pleased to state:

(a) whether as per the Civil Liability for Nuclear Damage Act, 2010, nuclear suppliers are not liable to pay more than the cost of supplied equipment as damage in case of nuclear accidents and if so, the details thereof;
(b) whether the said Act has effectively addressed the issue of expeditious payment of compensation to victims in the event of an accident and if so, the details thereof along with the compensation paid in this regard during the last three years and the current year;
(c) whether there exists an ambiguity over the definition of 'Supplier' under the civil nuclear liability regime and if so, the steps taken to clear this ambiguity;
(d) whether it is a fact that there is no instrument available in the country to cover recourse risks to claim for the damages caused by suppliers; and
(e) if so, the details thereof?

ANSWER

By- THE UNION MINISTER OF STATE FOR PERSONNEL, PUBLIC GRIEVANCES & PENSIONS AND PRIME MINISTER'S OFFICE (SHRI V. NARAYANASAMY):

(a) Section 4 of the Civil Liability for Nuclear Damage Act, 2010 channels the liability for nuclear damage to the operator of the nuclear installation. The operator of the nuclear installation, after paying the compensation for nuclear damage, shall have a right of recourse against the supplier in accordance with Section 17 of the said Act.
(b) The objective of the Civil Liability for Nuclear Damage Act, 2010 is to ensure prompt compensation for the victims in
the unlikely event of a nuclear incident. No nuclear incident occurred during the last three years and, therefore, the question of compensation does not arise.

c) Rule 24 of the Civil Liability for Nuclear Damage Rules, 2011 defines the term 'Supplier'.

d) & e) Under the Civil Liability for Nuclear Damage Act, 2010, only the Operator is required to furnish insurance policy or such other financial security or combination of both, covering his liability. Some suppliers have expressed concern regarding non-availability of cover for their risks.

**LOK SABHA UNSTARRED QUESTION NO.4075- 19.02.2014**

**MANAGEMENT OF NUCLEAR WASTE. By- SHRI NISHIKANT DUBEY: SHRI M.I. SHANAVAS:**

Will the PRIME MINISTER be pleased to state:

(a) whether the Government has made any assessment of the quantity of nuclear waste generated by the Nuclear Power Plants (NPPs) in the country;

(b) if so, the details and the outcome thereof along with the technology being used in the country for the management of nuclear waste;

(c) whether the Government proposes to launch a programme for development of a process for high level waste management; and

(d) if so, the details thereof along with the action taken in this regard?

**ANSWER**

**By- THE UNION MINISTER OF STATE FOR PERSONNEL, PUBLIC GRIEVANCES & PENSIONS AND PRIME MINISTER’S OFFICE (SHRI V. NARAYANASAMY):**

(a) Yes, Sir.

(b) The Government is using latest technology for management of nuclear waste generated during operation of nuclear power plants. The details are as follows:

   (i) The low and intermediate level radioactive waste generated during operation and maintenance of nuclear power plants is segregated, its volume reduced using various technologies and solidified. This solid/solidified waste is packaged in suitable containers to facilitate handling, transport and disposal.

   (ii) Disposal of low and intermediate level waste is carried out in specially constructed structures such as stone lined trenches, reinforced concrete trenches and tile holes. These disposal structures are located both above and underground in access-controlled areas. Disposal system is designed based on multi barrier principle for ensuring effective containment of the radioactivity. The areas where the disposal structures are located are kept under constant surveillance with the help of bore-wells laid out in a planned manner. The underground soil and water samples from these bore wells are routinely monitored to confirm effective confinement of radioactivity present in the disposed waste.

   (iii) Gaseous waste is treated at the source of generation. The techniques used are adsorption on activated charcoal and filtration by high efficiency particulate air filters. The treated gases are then diluted with exhaust air and discharged through a tall stack with monitoring.

   (iv) Liquid waste streams are treated by various techniques, such as filtration, adsorption, chemical treatment, thermal and solar evaporation, ion exchange, reverse osmosis etc. The concentrate from treatment of liquid waste are immobilised in inert materials like cement, polymer etc. The nuclear waste handling, treatment, storage and disposal is carried out as per the well laid down procedures and guidelines stipulated by the Atomic Energy Regulatory Board (AERB).

(c) High level waste is managed in the country by a well-established process called vitrification. Vitrification plants are in operation at Trombay, Tarapur & Kalpakkam for more than two decades.

(d) During reprocessing of spent fuel, 2-3 percent of spent fuel becomes waste and the rest is recycled. This 2-3 percent waste is called high level waste (HLW). A three step strategy is adopted in India for management of HLW which involves:

   (i) Immobilising high level liquid waste into inert solid glass matrix. This process of converting high level liquid waste into solidified glass matrix is called ‘vitrification’.

   (ii) Interim storage & cooling of these vitrified waste products in specially designed storage vaults for a period of 30-40 years. This is to dissipate the heat generated on account of decay of fission products associated with these waste products.

   (iii) Disposal of vitrified waste products in well-engineered disposal facilities after this storage period of 30-40 years.
Will the PRIME MINISTER be pleased to state:

(a) whether the Government is considering to carry out a safety audit of all atomic power plants in the country;
(b) if so, the details thereof;
(c) whether the Government is also considering to conduct a survey in the areas around atomic plants to understand the health and environmental implications; and (d) if so, the details thereof?

ANSWER

By- THE UNION MINISTER OF STATE FOR PERSONNEL, PUBLIC GRIEVANCES & PENSIONS AND PRIME MINISTER'S OFFICE (SHRI V. NARAYANASAMY):

(a) &(b) All the nuclear power plants in India are under continuous regulatory surveillance by Atomic Energy Regulatory Board (AERB). Periodic safety audit of all atomic power plants in India is carried out by the AERB. All nuclear power projects undergo an elaborate in-depth safety review during the consenting stages, viz. siting, construction, commissioning, etc. After satisfactory review during project stage, AERB issues operating licence to a nuclear power plant for period of upto five years. During the licence period, nuclear power plants are under regulatory surveillance and their safety performance is monitored in compliance with prescribed guidelines. A minimum of two regulatory inspections of each nuclear power plant is also carried out in a year to verify compliance with various safety requirements. A consolidated safety assessment of the plant is undertaken while renewing the operating licence. In addition to the prescribed safety review assessments, comprehensive safety audits of all Indian nuclear power plants against external events were undertaken by AERB and the Nuclear Power Corporation of India Ltd. following the Fukushima accident.
(c)&(d) The environmental matrices like air, water, soil, vegetation, crops, milk, fish, etc. around each of the nuclear power plant site are regularly monitored by an independent Environmental Survey Laboratory (ESL) set up at each of the site prior to operation of the plant. The samples for analysis are selected on the basis of potential pathways of exposure and an area upto a distance of 30 km is covered. The data collected has not indicated any measurable change in radiation levels in the environment and radioactivity in the environmental matrices compared to the base line data. Epidemiological survey for health assessment in respect of employees and their families staying in the nearby township and villages of each of the nuclear power plant have been carried out by reputed local medical colleges and analysis has been carried out by Tata Memorial Hospital (TMH), Mumbai. In addition, annual medical check-ups are carried out for all workers regularly. The examinations/studies have found that the morbidity pattern of all ailments is lower than the national average of the corresponding ailments. There has also not been any rise in cancer morbidity compared to national average.

Will the PRIME MINISTER be pleased to state:

(a) whether the Government proposes to conduct public awareness programme for the local community about nuclear and radiation safety;
(b) if so, the details thereof;
(c) whether these kind of programmes will be conducted throughout the country by the concerned agencies; and (d) if so, the details thereof and if not, the reasons therefor?

ANSWER

THE UNION MINISTER OF STATE FOR PERSONNEL, PUBLIC GRIEVANCES & PENSIONS AND PRIME MINISTER'S OFFICE (SHRI V. NARAYANASAMY):
(a) to (d) Yes Sir, Public Awareness programmes are undertaken on a regular basis by Department of Atomic Energy (DAE). Workshops and seminars are being conducted throughout the country by scientific bodies and associations to enhance public awareness about nuclear science, applications of radiation and radio-isotopes. Target audience for such programmes are university faculty, college lecturers and students, while subjects covered include nuclear and radiation safety, application of radio-isotopes for societal benefits, health physics aspects, safe handling of radioactive substances and radioactivity etc., Educational lectures are followed by demonstration experiments. Further, public awareness programmes, to spread awareness and allay the fears and apprehensions about nuclear power and related aspects among the local community are also conducted by Nuclear Power Corporation of India Limited (NPCIL) around nuclear power plant sites.

RAJYA SABHA UNSTARRED QUESTION NO. 2804-20.02.2014
EXEMPTION OF NUCLEAR PLANTS FROM ENVIRONMENT PROTECTION ACT- By SHRI AJAY SANCHETI:

Will the PRIME MINISTER be pleased to state:

(a) whether the nuclear power plants in the country are not subject to Environment Protection Act that covers other energy generation plants, if so, the reasons therefor; and
(b) the details of the mega N-Power plants finalised without paying due consideration to environment?

ANSWER

By- THE UNION MINISTER OF STATE FOR PERSONNEL, PUBLIC GRIEVANCES & PENSIONS AND PRIME MINISTER'S OFFICE (SHRI V. NARAYANASAMY):

(a) No, Sir. All nuclear power plants are subject to the Environmental Protection Act.
(b) Does not arise

RAJYA SABHA UNSTARRED QUESTION NO. 2333-13.02.2014
DEVELOP SAFETY POLICY - SHRI MOHD. ALI KHAN: SHRIMATI T. RATNA BAI:

Will the PRIME MINISTER be pleased to state:

(a) whether the Atomic Energy Regulatory Board (AERB) has failed to develop safety policy, standards, codes and guides; and
(b) if so, the details thereof and reasons there for?

ANSWER

THE UNION MINISTER OF STATE FOR PERSONNEL, PUBLIC GRIEVANCES & PENSIONS AND PRIME MINISTER'S OFFICE (SHRI V. NARAYANASAMY):

(a) No, Sir.
(b) The Atomic Energy Regulatory Board (AERB) has issued Safety Codes, Safety Standards and Safety Guides for regulation of nuclear and radiation facilities/activities in India. AERB has issued a total of 143 regulatory documents, which provide the requirements as well as guidance for various nuclear and radiological facilities/activities regulated by AERB. These regulatory documents, together with various rules promulgated under the Atomic Energy Act, 1962; and the Atomic Energy (Radiation Protection) Rules, 2004, Atomic Energy (Safe Disposal of Radioactive Wastes) Rules, 1987; Atomic Energy (Factories) Rules, 1996 and AERB's Mission Statement, provide for the safety policies with respect to radiation, industrial and nuclear safety, consistent with the functions of AERB stipulated in the Presidential Order dated 15th November 1983 constituting the AERB.
RAJYA SABHA UNSTARRED QUESTION NO. 2800- 20.02.2014
SHORTAGE OF FUEL/URANIUM by SHRIMATI SMRITI ZUBIN IRANI:

Will the PRIME MINISTER be pleased to state:

(a) Whether it is a fact that some of our Country's nuclear power plants are facing shortage of fuel/uranium;
(b) If so, the details thereof and the reasons therefor along with the action taken/ being taken by Government in this regard;
(c) Whether Government proposes to explore an alternative fuel or technology for generation of nuclear power in the country, if so the details thereof; and
(d) the steps taken/being taken by Government in this regard?

ANSWER

THE UNION MINISTER OF STATE FOR PERSONNEL, PUBLIC GRIEVANCES & PENSIONS AND PRIME MINISTER'S OFFICE (SHRI V. NARAYANASAMY):

(a)&(b) Yes, Sir. The country has 20 nuclear power reactors under operation with an installed generating capacity of 4780 MWe. Of these, one reactor, RAPS-1 located at Rawatbhata, Rajasthan (100 MW) is under extended shutdown for techno-economic assessment. Under separation plan, ten of our reactors are currently placed under IAEA safeguards and are eligible for imported fuel. These reactors are RAPS 2 to 6 located at Rawatbhata, Rajasthan: KAPS 1&2 at Kakrapar, Gujarat and TAPS 1&2 at Tarapur, Maharashtra. These reactors normally operate at their full capacity. Ten nuclear power reactors viz., KGS 1 to 4 located at Kaiga, Karnataka; NAPS 1 & 2 at Narora, Uttar Pradesh; MAPS 1 & 2 at Kalpakkam, Tamil Nadu; and TAPS 3 & 4 at Tarapur, Maharashtra continue to use uranium sourced within the country. Due to a mismatch between demand and supply of domestic Uranium, the total power generated by these reactors is generally lower than their gross installed capacity of 2,840 MWe. Following extensive work for exploration of Uranium in the country, however, the supply of Uranium from Indian mines is progressively improving and accordingly, capacity utilisation of these 10 reactors has increased during last three years.

(c&d) Yes, Sir. The Indian Nuclear power programme has been, right from its inception, formulated to make maximum use of limited domestic Uranium resources and large Thorium resources in a scientifically viable sequential manner. Accordingly, India's nuclear power programme is formulated in three stages. In the first stage, electricity is generated using natural uranium fuel in Pressurised Heavy Water Reactors (PHWRs). In the second stage, spent fuel from PHWRs after further processing is used in Fast Breeder Reactors (FBRs). Thorium in itself cannot produce electricity and, in the later part of the second stage when enough nuclear installed capacity has been reached, Thorium has to be first converted to Uranium-233 in a FBR, which is then to be used to launch the third stage for generating electricity using Uranium-233 and Thorium based fuel. As of now, India has entered into the second stage of the nuclear power programme and several components of an extensive research, development and demonstration programme related to various aspects of Thorium based nuclear fuel cycle, have been completed. Based on this work the design and development of an Advanced Heavy Water Reactor (AHWR) that will demonstrate a range of thorium fuel cycle technologies, along with advanced passive safety features, has been carried out. Initial activities towards the construction of AHWR are included in the 12th Five Year Plan.
DAE is proud of its Padma Awardees for the year 2014

PADMA SHRI

Shri Sekhar Basu,
Director, BARC, Mumbai

Dr. Ravi Bhushan Grover,
Director, HBNI, Mumbai &
Homi Bhabha Chair Professor

PUBLIC AWARENESS ACTIVITIES

The 101st Indian Science Congress was held at Jammu University, Jammu during February 3 – 7, 2014. The theme of the exhibition was India Vision 2020, with focal theme as “Innovations in Science & Technology for Inclusive Development”. DAE participated in the exhibition displaying all its activities.

DAE also organised a Public Outreach session where senior Scientists spoke about the various activities and S & T programmes being pursued by DAE.

DAE participated in ‘Anu Week – 2014’ at Nagercoil (close to Kudankulam), Tamil Nadu during January 21 – 25, 2014. The event comprised an exhibition, a seminar for high school/college teachers on the ‘Societal Applications of Atomic Energy’, quiz contest for school students etc. The event also included a visit of the plant by all the participants in the seminar. The visit proved immensely fruitful in removing misconceptions about the plant. The exhibition was a concerted effort by DAE, BARC, NFC, Palayakayal Unit, Heavy Water Plant, Tuticorin and KKNPP, NPCIL.

The Bangalore India Bio 2014 organised by the Government of Karnataka, was held at Bangalore during February 10 – 12, 2014. DAE exhibited its achievements in the area of healthcare, biotechnology, agriculture and food. Jonaki Laboratory, BRIT, Hyderabad also participated in the event. Several technologists, farmers and scientists visited the DAE pavilion.
In its endeavour towards enhancing public awareness and acceptance of nuclear energy programs, GCNEP organized a program at the Mata Raj Kaur Institute of Engineering & Technology (MRKIET) Rewari, Haryana on 27th & 28th February 2014. The particular focus of the program was students and faculty of engineering, degree, pharmacy, law and management colleges.

The program comprised a two day seminar and an exhibition, where experts from BARC and DAE spoke on various popular topics like Applications of radioisotopes in: Healthcare & Environment, Agriculture and Food Processing, Hydrology, Nuclear Energy requirement, Misconceptions & Facts etc.

The exhibition was open to all and many school children from the nearby schools were facilitated to visit the exhibition. The students found the program to be very informative and appreciated GCNEP’s efforts in reaching out to the rural institutes. The students desired that more of such outreach programs be conducted at the level of schools & colleges to remove the misconceptions about nuclear energy.
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