# BARCE HIGHLIGHTS











भाभा परमाणु अनुसंधान केंद्र BHABHA ATOMIC RESEARCH CENTRE

> स्वर्ण जयंती वर्ष GOLDEN JUBILEE YEAR 2006-2007

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### FOREWORD

Bhabha Atomic Research Centre (BARC) is celebrating its golden jubilee year during 2006-07. On the 20<sup>th</sup> January, 1956, Pandit Jawaharlal Nehru formally inaugurated the Atomic Energy Establishment Trombay (AEET), which is renamed as Bhabha Atomic Research Centre (BARC) on January 22, 1967. As a premier R&D centre of the Department of Atomic Energy (DAE), BARC has a mandate to provide R&D support to the nuclear power programme, to pursue all activities related to nuclear fuel cycle, to operate research reactors for supporting neutron beam research and supplying radioisotopes for various applications, to conduct frontline basic research in physical, chemical, biological and engineering sciences all of which lead towards improving quality of life of our people. The achievements BARC has made over the last 50 years are well known not only to the scientific community in the country but also to our people at large. Scientific achievements made by this premier research centre are well documented in various publications of DAE including the series "BARC Highlights". During this golden jubilee year, we have made an effort to bring out some of recent research and development accomplishments in the form of 8 volumes, highlighting the following areas:

- 1. Nuclear Fuel Cycle
- 2. Physical Sciences
- 3. Chemical Science and Engineering
- 4. Materials Science and Engineering
- 5. Life Sciences
- 6. Reactor Technology and Engineering
- 7. Electronics, Instrumentation and Computers
- 8. Environmental Science and Engineering

These volumes will showcase the latest work in the aforementioned areas and will demonstrate how each of these is directed towards achieving the overall goal of using nuclear energy for the benefit of our people.

Nuclear energy programme in India has now reached a level of maturity. Today, India is self-sufficient in building nuclear power stations of 540 MWe capacities and has gained sufficient mastery over the entire fuel cycle. We are at the threshold of entering the second stage of nuclear power programme, in which a rapid growth in installed capacity is expected through the fast reactor programme. In the area of basic research in science and engineering, BARC has been maintaining a leading position both in national and international scenario. One of the strongest points of basic research in BARC lies in its capability in building in-house sophisticated research facilities. The core competence of the scientists and engineers in our centre covers a very wide range as is reflected in the 8 companion volumes released on the occasion of the golden jubilee year.

Over the years a large infrastructure has been created in BARC for research and development work related to PHWRs. As a number of units are growing older, they need ageing management and performance enhancing back-fits. Extensive work is in progress in this area specially in the area of life management of coolant channels. It is worth mentioning here that BARC took up the challenge imposed by the technology denial regime, resulting in indigenous development of critical components and achievement of self-reliance in almost every aspect of PHWR technology.

For sustainable development of nuclear energy, besides fuel resources a number of important issues are required to be addressed. They should be economically competitive as compared to other sources of energy. The issue of heightened public concern about nuclear safety, waste management and many other issues need to be addressed. To address these issues a number of advanced reactor designs as well as fuel cycle technologies are being pursued worldwide. BARC has designed the Advanced Heavy Water Reactor (AHWR) which is under pre-licensing safety review by the Atomic Energy Regulatory Body. The main objective behind the development of this reactor is to develop and demonstrate technologies related to thorium-based systems well in advance and to develop a number of passive technologies. BARC has also developed a desalination technology, which effectively integrates with the AHWR technology. Considering our very small petroleum reserves, and increasing worldwide oil prices, it is prudent that India find an alternative to oil for its transport applications. The process heat from high temperature reactors provides this alternative in the form of high efficiency thermochemical hydrogen generation processes. In addition to this, the temperature reactor concept can also be applied for the development of power pack for electricity generation in remote areas. To cater to these requirements a Compact High Temperature Reactor (CHTR) is under design at BARC.

For breeding fissile uranium-233 from thorium, development of Accelerator Driven Sub-Critical System s(ADS) is the latest addition to the Indian nuclear programme. This system promises shorter doubling time and incineration of long-lived actinides and fission products thus effectively addressing the sustainability issues of availability of fissile material and of waste management. The current interest in ADS at BARC has led to research in wide ranging areas, which include, spallation target studies, thermal and structural analysis of the radio-frequency quadrupoles in the proton linear accelerators, among others.

BARC is operating three research reactors namely, Apsara, Cirus and Dhruva at Trombay. After satisfactory operation for nearly four decades, Cirus was refurbished and put back in service in October 2002. The activities at these research reactors have further enhanced the immense knowledge base in the field of design, development, operation and refurbishment of nuclear reactors. These reactors have provided an ideal platform to the engineers and physicists to perform a number of experiments and simulation studies.

BARC has made significant contributions towards the fast reactor programme. Based on the valuable experience gained with liquid metal fast breeder reactor technology, the construction of a 500 MWe Prototype Fast Breeder Reactor (PFBR) was started in October, 2004. Major contributions made by BARC include activities related to radiation shielding and design of inclined fuel transfer machines.

India is on a rapid economic growth path. A recent study has revealed that we will need to augment our electricity generation ten fold in the next four to five decades. To meet this target of high installed capacity in a short time, two 1000 MWe VVERs which are pressurised water reactors imported from Russian Federation are now under construction at Kudankulam. Our programme on reactor technology has been broadened to encompass light water reactors. Indigenous capabilities of in-core fuel management of these reactors are being developed.

BARC is pursuing activities for enhancing the safety of nuclear power plants. Studies related to seismic analysis, design of energy absorption devices, development of methodologies for meeting leak before break criteria are some of the important activities in this regard.

The present volume gives an outlines of the programmes in the area of Reactor Technology and Engineering, while highlighting the core competence of the scientists and engineers working in this area.

Srikumar Banerjee Director

### PREFACE

The Indian nuclear power programme today comprises our existing reactors, reactors under construction and planning, and future designs to provide long term energy security to the country. The different segments of this programme require associated R&D support to cater to diverse needs of ageing management, on-going design and design support, R&D for innovative designs, safety analysis and allied disciplines, and nuclear applications including desalination and non-grid based electricity. The intensity of such R&D activities has been progressively on a path of growth, commensurate with the growth of our nuclear power programme. BARC has the mandate to carry out multi-faceted research and engineering activities needed to support such wide range of tasks, in particular, focusing on thermal reactors and new nuclear energy systems.

This volume brings out the spectrum of some recent activities pertaining to diverse reactor technologies, research reactors, safety studies, material research, seismic and structural studies, integrity monitoring systems, in-service inspection, non-destructive testing, life management and extension of reactor components, component manufacturing and testing and desalination. The range of reactor technologies covered in this volume includes the indigenous Pressurised Heavy Water Reactors, Fast Breeders, the forthcoming Advanced Heavy Water Reactor, Compact High Temperature Reactor, and Accelerator Driven System. Work related to Boiling Water Reactors at Tarapur and 1000 MWe VVER Reactors at Kudankulam has also been included in the compilation.

The entire gamut of activities under Reactor Technology and Engineering encompasses a large number of scientific and engineering disciplines. Consequently, this volume attracted an overwhelming response from more than 300 scientist & engineers working in the field. The articles covering the activities in this vast technological arena, performed in BARC in the last few years, are compiled in a reader friendly format. For the benefit of the readers the E mail addreses of the contributors have also been given in the respective articles. This should facilitate the reader to obtain in-depth information, if desired.

I believe that the readers of this volume would be able to gain a reasonable insight into the wide spectrum of activities pursued at BARC, in the field of reactor technology and engineering. I thank all the authors for their unfailing enthusiasm and persistent cooperation.

R. K. Sinha



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### **1. ADVANCED HEAVY WATER REACTOR**

India's three-stage nuclear power programme is chalked out based on the domestic resource position of uranium and thorium. The first stage started with setting up of the Pressurised Heavy Water Reactors (PHWR) based on natural uranium and pressure tube technology. In the second phase the fissile material base will be multiplied in Fast Breeder Reactors using the plutonium obtained from the PHWRs. The third stage is focused on reactors designed to utilise the large thorium reserves based on thorium-<sup>233</sup>U fuel cycle. The Advanced Heavy Water Reactor (AHWR) has been designed to fulfill the need for the timely development of thorium-based technologies for the entire thorium fuel cycle. This chapter highlights the recent activities carried out in the design and development of the AHWR, such as the core & process system design, nuclear data, fuel design, fuel handling systems, safety analyses, analytical studies and experimental validation.

Nuclear power employing a closed fuel cycle is the only long term, sustainable option for meeting a major part of the Indian energy demand. Indian resources of thorium are larger than those of uranium. Thorium, therefore, is widely viewed as the 'fuel of the future'. Thorium-based nuclear fuel cycle possesses several well-known characteristics as indicated below.

### **Advantages of Thorium Fuel Cycle**

- Thorium can sustain a thermal breeding cycle using external fissile materials like uranium-235, plutonium or an accelerator driven neutron source.
- The cycle produces virtually no plutonium.
- The waste products contain low amounts of long-lived alpha-emitters

The Indian **Advanced Heavy Water Reactor (AHWR)** is designed and developed to achieve large-scale use of thorium for the generation of commercial nuclear power. This reactor will produce most of its power from thorium, with no external input of uranium 233, in the equilibrium cycle.

AHWR is a 300 MWe, vertical, pressure-tube type, boiling light water cooled, and heavy water moderated reactor. The reactor incorporates a number of passive safety features and is associated with a fuel cycle having reduced environmental impact. At the same time, the reactor possesses several features, which are likely to reduce its capital and operating costs.

### Important Safety Features of AHWR

- Slightly negative void coefficient of reactivity.
- Passive safety systems working on natural laws.
- Large heat sink in the form of Gravity Driven Water Pool with an inventory of 6000 m<sup>3</sup> of water, located near the top of the Reactor Building.
- Removal of heat from core by natural circulation.
- Emergency Core Cooling System injection directly inside the fuel.
- Two independent shutdown systems.

AHWR employs natural circulation for cooling the reactor core under operating and shutdown conditions. All event scenarios initiating from non-availability of main pumps are, therefore, excluded. The Main Heat Transport (MHT) System transports heat from fuel pins to steam drum using boiling light water as the coolant. The MHT system consists of a common circular inlet header from which feeders branch out to the coolant channels in the core. The outlets from the coolant channels are connected to tail pipes carrying steam-water mixture from the individual coolant channels to four steam drums. Steam is separated from the steam-water mixture in steam drums, and is supplied to the turbine. The condensate is heated in moderator heat exchangers and heaters and is returned to steam drums by feed pumps. Four down comers connect each steam drum to the inlet header.

Emergency Core Cooling System (ECCS) is designed to remove the core heat by passive means in case of a postulated Loss of Coolant Accident (LOCA). In the event of a rupture in the primary coolant pressure boundary, the cooling is initially achieved by a large flow of water from the accumulators. Later, cooling of the core is achieved by the injection of cold water from a Gravity Driven Water Pool (GDWP) located near the top of the reactor building.



In AHWR, subsequent to energy absorption in GDWP in vapour suppression mode, the Passive Containment Cooling System (PCCS) provides long term containment cooling following a postulated LOCA. GDWP serves as a passive heat sink yielding a grace period of three days. The core gets submerged in water long before the end of this period.

Consistent with the approach used in standardised Indian PHWRs, AHWR is provided with a double containment. For containment isolation, a passive system has been provided in the AHWR. The reactor building air supply and exhaust ducts are shaped in the form of U-bends of sufficient height. In the event of LOCA, the containment pressure acts on the water pool surface and drives water, by swift establishment of siphon into the U-bends of the ventilation ducts. Water in the U-bends acts as a seal between the containment and the external environment, providing necessary isolation between the two.



### Some Distinctive Features of AHWR

- Elimination of high-pressure heavy water coolant resulting in reduction of heavy water leakage losses and eliminating heavy water recovery system.
- Recovery of heat generated in the moderator for feed water heating.
- Elimination of major components and equipment such as primary coolant pump and drive motors, associated control and power supply equipment and corresponding saving of electrical power required to run these pumps.
- Shop assembled coolant channels, with features to enable quick replacement of pressure tube alone, without affecting other installed channel components.
- Replacement of steam generators by simpler steam drums.
- Higher steam pressure than in PHWRs.
- Production of 500 m<sup>3</sup>/day of demineralised water in Multi Effect Desalination Plant by using steam from LP Turbine.
- Hundred years design life of the reactor.
- A design objective of requiring no exclusion zone on account of its advanced safety features.

The AHWR fuel contains 54 fuel pins arranged in three concentric circles surrounding a central displacer rod. The inner two circles contain 30 (Th-<sup>233</sup>U)O<sub>2</sub> fuel pins and the outer circle contains 24 (Th-Pu)O<sub>2</sub> fuel pins. The central rod contains dysprosia in zirconia matrix. The fuel also incorporates a water tube for the spraying of ECCS water directly on fuel pins during a postulated LOCA. AHWR fuel is currently designed for an average burn-up of 24 GWd/t. Its design makes it amenable for reconstitution, if desired to facilitate a further extension of burn-up in the (Th-<sup>233</sup>U)O<sub>2</sub> fuel pins in future.

The AHWR fuel cycle will be self-sufficient in <sup>233</sup>U after initial loading. The spent fuel streams will be reprocessed and thorium and <sup>233</sup>U will then be recycled and reused. The AHWR fuel cycle has enough flexibility to accommodate a large variety of fuelling options. Incidentally, the thorium fuel cycle also presents low proliferation risks, a factor considered significant by several nations for export of nuclear technology. A quantitative analysis of the AHWR fuel cycle substantiates this feature.



Reactor power	:	920 MW t, 300 MW e
Core configuration	:	Vertical, pressure tube type design
Coolant	:	Boiling light water
Number of coolant channels		452
Pressure tube ID	:	120 mm
Lattice pitch	:	245 mm (square pitch)
No. of pins in fuel cluster	:	54 (Th-Pu)O <sub>2</sub> - 24 pins (Th- $^{233}$ U)O <sub>2</sub> - 30 pins
Active fuel length	:	3.5 m
Total core flow rate	:	2230 kg/s
Coolant inlet temperature	:	259 °C (nominal)
Feed water temperature	1	130 °C
Average steam quality	:	18.6 %
Steam generation rate	;	414.4 kg/s
Steam drum pressure	:	70 bar
MHT loop height	:	39 m
Primary shut down system	:	40 shut off rods
Secondary shut down system	:	Liquid poison injection in moderator
No. of control rods	:	13

### Important Design Parameters of AHWR



Scaled Model (1:50) of AHWR in Acrylic Material



### 1.1 REACTOR PHYSICS DESIGN & NUCLEAR DATA

### Reactor Physics Design of AHWR

The major design mandates are the production of more than 60% of the total power from  $Th/^{233}U$  and incorporation of inherent safety features such as the negative coolant void coefficient. A composite cluster with (Th, <sup>233</sup>U) MOX fuel in the inner region and (Th, Pu) MOX fuel in the outer region is designed to meet this desired fuel performance characteristics. Coolant void coefficient of reactivity can be made negative in a slightly under-moderated heavy water system at a lattice pitch of 220 mm. For the design pitch of 245 mm, the coolant void coefficient is significantly positive. Hence, a burnable absorber, dysprosium, is used in a centrally located multi-purpose displacer rod, to obtain the negative coolant void coefficient. Other inherent safety features include negative fuel temperature coefficient, negative power coefficient and low power density. In addition, core excess reactivity is low on account of on-power refuelling.

The basic lattice and composite fuel cluster has been designed for 245 mm pitch. The reactor has two independent, functionally diverse, fast acting shut down systems, namely, shutdown system#1 consisting of 40 mechanical boron carbide shut off rods and shutdown system#2 based on liquid poison injection. There are thirteen boron carbide control rods grouped into four regulating rods for fine reactivity control, four absorber rods for xenon override and five shim rods for power setback.

Reactor control and protection systems are designed in a fashion similar to that in currently operating PHWRs except that the reactivity devices occupy lattice locations in the AHWR. A notable feature of AHWR is the xenon override capability to restart reactor anytime following shutdown. Equilibrium core is fuelled to three burn-up zones to achieve a flattened power distribution. Core average discharge burn-up is 24,000 MWd/t.

The cluster design allows use of burnable absorber in the non-fuel multi-purpose displacer rod depending on the fuel type. Use of heavy water as moderator improves neutron economy and relatively soft neutron spectrum imparts flexibility to run it on different fuel-types. During the initial phase plutonium bearing fuel is used to generate <sup>233</sup>U required for composite cluster of the equilibrium core. Equilibrium core will be self-sustaining in <sup>233</sup>U and runs on composite cluster containing both <sup>233</sup>U and plutonium. Two different values of enrichment i.e., 2.5% in the top half and 4.0% in the lower half are used in plutonium bearing fuel pins such that the axial flux peak is shifted away from the high steam quality upper region in the coolant channel. This improves the thermal hydraulic characteristics of fuel channel, with fuel dry-out occurring at a considerably high channel power in comparison to that with a uniform axial enrichment.

Initial core design of the AHWR, approach to equilibrium and refuelling strategy studies along with the development of the flux mapping system, power distribution control system and analysis of various spatial instabilities are being pursued. Physics design to achieve higher fuel burn-ups of about 40,000 MWd/t by lattice pitch reduction and removal of burnable absorber from the fuel cluster are important ongoing activities.

Multi-group data library has been updated for elements of thorium chain and for certain elements specific to AHWR such as burnable absorber dysprosium. Various uncertainties due to thorium data have been identified and experiments are planned in the AHWR critical facility to resolve them. Physics design and preliminary safety analysis for experiments in critical facility have been completed. Core power distribution has been optimized, taking into account the neutronic thermal hydraulic coupling. Studies have been carried out on generation of <sup>233</sup>U required for equilibrium core, achievement of self-sustaining uranium cycle, recycling of uranium and its impact on overall fuel cycle including toxicity of spent fuel.





### Nuclear Data and Needs for Thorium Fuel Cycle Studies

The design of innovative reactor systems needs integral data validation studies to generate accurate nuclear data. The nuclear data from experimental measurements need to be evaluated, for arriving at basic point data libraries in the Evaluated Nuclear Data File (ENDF) format. The condensed nuclear data libraries are then used with Monte-Carlo and deterministic codes for reactor physics evaluations of reactor systems. In the context of thorium-based systems, the nuclear data for thorium fuel cycle studies of the AHWR and the CHTR are of current interest. The nuclear data for ADS systems, for neutron energies beyond 20 MeV are also important. Presently Indian nuclear data activities generically encompass the user-oriented approach starting from data files distributed by the IAEA.

India sponsors a regional IAEA-NDS nuclear data Mirror Site (http://www-nds.indcentre. org.in) for the Asian region.

### Nuclear data for Isotopes of Thorium Chain

The new concepts involving thorium systems require detailed basic nuclear data measurements and integral validation studies, as thorium has not received the required attention in the past. The nuclear data of the major and minor isotopes <sup>230</sup>Th, <sup>232</sup>Th, <sup>231</sup>Pa, <sup>233</sup>Pa, <sup>232</sup>U, <sup>233</sup>U and <sup>234</sup>U in the thorium fuel cycle needs

to be brought to the present level of quality that exists for the isotopes of the U-Pu fuel cycle. For instance, the self-shielded capture resonance integrals for <sup>232</sup>Th are higher in "jendl3.lib" as compared to those in "endfb6.lib" in the final stage of the IAEA WIMS-D update project, by several tens of percent as compared to 0.1% target accuracy. The design of advanced reactor systems, demands accurate nuclear data, in the resonance region that affect plant safety related feedback coefficients such as Doppler and coolant void reactivity effects as a function of burn up.



### Critical Facility (CF)

The AHWR simulations are based on assumptions and modeling approximations, which are sensitive to nuclear data uncertainties especially because of thorium fuel cycle. The evaluation of the lattice characteristics requires experimental validation to freeze the design and obtain regulatory clearance, before fuel fabrication. A multi-purpose Critical Facility (CF) has been designed and is in an advanced stage of construction. The sensitivity calculations illustrate that the critical height of the CF with AHWR representative core increases by 5 cm and 7 cm respectively when the "iaea.lib" and "jendl3.lib" libraries replace "endfb6.lib" WIMS-D library. For natural uranium core this is about 2 cm. The replacement of multigroup data of <sup>232</sup>Th alone in "jendl3.lib" by "endfb6.lib" changes the k-infinity by 10.24 mk, "jendl3.lib" yielding a higher calculated value of k-infinity.

#### **Compact High Temperature Reactor (CHTR)**

The design of CHTR, which exhibits intermediate neutron spectrum, was strongly influenced by considerations of nuclear data and associated uncertainties during its evolution. The cross sections for several new materials, such as Er, Bi and Ga that were considered for CHTR show large discrepancies in different cross section libraries. It is mandatory to have a negative Doppler feedback effect in the core design. The initial choice of pure<sup>233</sup>U as fuel had been revised due to the calculated positive Doppler reactivity feedback. Further, the calculated Doppler reactivity effect of <sup>233</sup>U has a large uncertainty as the nuclear data of resolved and unresolved resonance region are highly uncertain. As the spectrum covers regions above the thermal range, accurate knowledge of various transport and inelastic cross-sections of various constituents, such as <sup>233</sup>U, <sup>232</sup>Th, Be, Er, Th, which affect the design significantly at high temperatures are required. Experimental work to demonstrate these systems are underway.

### Thorium irradiation in FBTR (Kalpakkam)

In the 40 MWt Fast Breeder Test Reactor (FBTR), at Kalpakkam, fifty four thoria subassemblies (717 Kg) have been loaded in the 9<sup>th</sup> ring in the radial blanket after the nickel reflector. Plans are underway to consider loading additional 100 thoria assemblies in 7<sup>th</sup> and 8<sup>th</sup> rings as well. The <sup>233</sup>U produced in this reactor will have low content (5 ppm) of <sup>232</sup>U as compared to several hundreds of ppm in other situations such as a normal fast reactor core, ADSS core with thorium and thoria in PHWR. The reason for the expected low ppm of <sup>232</sup>U in FBTR is understood as due to three factors influenced by nuclear data and physics considerations: Nickel reflector brings the neutrons below the threshold of (n, 2n) reaction in <sup>232</sup>Th. The effective <sup>231</sup>Pa (n,  $\gamma$ ) cross section is much lower in a fast spectrum as the capture cross section falls rapidly with increasing energy. Thirdly, the accumulation potential of <sup>233</sup>U produced is more in saturation in a fast spectrum making the ppm content of <sup>232</sup>U in <sup>233</sup>U much smaller.

### PIE studies of irradiated thorium bundles in PHWRs

Identical loading of thorium bundles was used in KAPP-1 & 2, KAIGA-1 & 2 and RAPS-3 & 4 to attain flux flattening in the initial core. The thorium oxide used is about 400 kg in all the 35 bundles put together in a reactor. The bundles loaded in KAPP-1

& 2, KAIGA-1 & 2 and RAPS-3 & 4 have already been discharged from the core. Samples were obtained from one of the irradiated ThO, bundles and have been analyzed experimentally by alpha spectrometry for <sup>232</sup>U and by thermal ionization mass spectrometry for <sup>233</sup>U, <sup>234</sup>U, <sup>235</sup>U and <sup>236</sup>U by two different groups in BARC. The previous analyses by two teams in BARC gave a factor of six to eight under-predictions in the production of <sup>232</sup>U. The discrepancy was traced back to the fact that the effective one-group values of cross sections for isotopes of thorium fuel cycle and the use of assumptions in the ORIGEN code are not applicable to the irradiation of thorium in PHWRs. Simulation of the thorium experiment using the new WIMS-D libraries, has been successfully attempted. Sensitivity results of different modeling approaches such as single cell versus supercell model and treatment of (n, 2n) process (pseudo-fission versus explicit) to prediction of isotopic contents of urania have been obtained. The results are shown in Table.

Generation of integral data by gamma spectrometric analysis of the irradiated thorium fuel is also a part of this activity.

A direct consequence of <sup>232</sup>U concentration in bred <sup>233</sup>U from PHWRs, is its effect on radiation shielding modification in the AHWR critical facility. The outer thickness of the concrete has been increased by nearly 10% to compensate for the additional gamma dose emanating from the fuel clusters based upon experimental results of post irradiation analyses and using the new basic evaluated nuclear data files.

### IAEA-CRP on nuclear data for thorium cycle

The active participation in the IAEA-Co-ordinated Research Program, on the "Evaluated Nuclear data for Thorium- Uranium Fuel Cycle" enables to share information and to benefit from the developments related to the use of thorium around the world. An experimental benchmark based upon a 30 kW, <sup>233</sup>U fuelled research reactor KAMINI, at Kalpakkam, is under preparation. KAMINI is a low power research reactor designed and built by a joint venture of BARC and Indira Gandhi Centre for Atomic Research (IGCAR). KAMINI is the only reactor in the world operating with the <sup>233</sup>U fuel. Preparation of a benchmark on thorium irradiation experiments and burnup measurements in PHWRs is also underway.

	C/E	Ratio		
Nuclide	Single cell model		Super cell model	
	PFP	Explicit	PFP	Explicit
<sup>232</sup> U	1.10	1.15	0.79	0.85
<sup>233</sup> U*	0.95	0.95	0.99	1.00
<sup>232</sup> U/ <sup>233</sup> U(ppm)	1.16	1.22	0.80	0.86
<sup>234</sup> U	0.81	0.81	0.96	0.96
<sup>235</sup> U	0.73	0.73	0.92	0.92
<sup>236</sup> U	0.60	0.60	0.86	0.86
Total U (232, 233, 234, 235, 236)	0.93	0.93	0.98	0.98

Comparison of the Isotopic Contents in (J-11-9) Fuel Rod – 3<sup>rd</sup> Ring.

### Use of updated nuclear data for safety analyses and operation of existing reactors

The Fuel Temperature Coefficient (FTC) of PHWR fuel, calculated by the new 69-group "iaea.lib" library gives significantly different results at higher burn-ups and explains as a preliminary observation, the unexpected power rise that occurred in the KAPS-1 unit. In a PHWR, the FTC which is negative at low burnups becomes less and less negative and even turns positive at some burnup. The precise crossover point in burnup where the FTC becomes positive depends on many parameters such as the temperature range and 19 versus 37-rod cluster. The FTC is due to the combined effect of Doppler effect and fuel re-thermalization effect. Recent calculations of FTC of PHWR lattices, performed independently by several researchers, illustrate the following: The 27 group WIMS1981 library has a cross over point, for FTC at about 12000 MWd/t burnup; at about 9400 MWd/t with the same but 69-group library, at about 6000 MWd/t for a 19 rod cluster with the new "iaea.lib" library and at about 4500 MWd/t for 37 rod cluster of PHWR with the "iaea.lib" library. The crossover point of the FTC is not just the issue but how negative it should be, in order to overcome positive reactivity, that includes the positive xenon kill feedback whenever power transient occurs. The calculated coolant void reactivity using the new "iaea.lib" library is observed to be lower than the earlier results obtained using the 1971 library. The KAPS-1 overpower transient could be explained only with the use of new IAEA multi-group\_nuclear data libraries.

### International collaborations in experimental programmes

As multiple fuel cycles (e.g., U-Pu, Th-U), with the option of closing the fuel cycle are envisaged, the nuclear data requirements that are needed to develop the new systems with high burnup are demanding and include the entire range of actinides and fission products for multiple fuels. The neutron time of flight experiments to measure neutron induced reaction cross sections from 1eV to 250 MeV, in CERN (Geneva, Switzerland) and neutron transmission experiments using the 150 MeV electron linac at Pohang in South Korea, would help to reduce the existing uncertainties in simulation studies of new and advanced reactor concepts. The new measurements of nuclear data of these experiments and analyses including covariance error information are of importance. Experimental activation measurements to perform neutron source flux characterization in DHRUVA 3001 beam hole has been taken up in order to initiate neutron time.

### XnWlup software for inter-comparison of WIMS-D multigroup cross section

An upgraded version of the computer program 'XnWlup' has been developed in Visual C++ to produce readily, histogram plots of the multi group cross sections of a selected nuclide, as a function of neutron energy. It also provides the comparison of the nuclear data of different nuclides from different libraries. The upgraded version, Xnwlup3.0 has extra features as sought by the users and includes plots and inter-comparisons of selfshielded resonance integrals at various background dilutions and temperatures.

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#### Nuclear Data for AHWR Analysis

Nuclear data is one of the major inputs for the physics design of advanced reactor systems utilising thorium, such as the AHWR. The basic experimental data of the available Th-U cycle available requires to be established with the accuracy of current day standards. The requirements of accuracy for the basic data of the individual isotopes for thermal reactor systems, will have to be improved in order to predict, for instance, k-effective within 0.5%, feedback coefficients within 10 % and integral reaction rates within 1.0%. The aim of the task is to generate a processed multi-group data-set for AHWR analysis, which would entail benchmarking, data processing, integral testing, front-end and





back-end analysis and sensitivity studies of the systems using thorium. There are several ingredients in this task like, assessment of the energy group structure of thorium cycle evaluations, research towards indigenous data processing capabilities and development of experimental capabilities and nuclear data evaluation methodology.





The availability of the basic nuclear data for the thorium cycle has been assessed and the multi group processed library for isotopes has been updated. The thorium-fuelled lattices have been benchmarked to validate the available basic data. Further the nuclear data set with materials and material composites have also been updated for analysis of the advanced reactor systems. Experiments have been formulated to qualify the nuclear data, by validation through post-irradiation examination.

The review of the basic data isotopes of the Th-U cycle such as <sup>232</sup>Th, <sup>233</sup>Th, <sup>234</sup>Th, <sup>231</sup>Pa, <sup>232</sup>Pa, <sup>233</sup>Pa, <sup>232</sup>U, <sup>233</sup>U, <sup>234</sup>U etc. existing as of now exhibit large discrepancies. The thorium fuel cycle data in the WIMS data set has been extended with <sup>232</sup>U and <sup>231</sup>Pa. The <sup>232</sup>U content in the irradiated thoria bundles of PHWR has been estimated with the WIMS dataset and WIMSD code system. The ratio of <sup>232</sup>U/<sup>233</sup>U is calculated as ranging from 450 ppm to 590 ppm and is in good agreement with experimental results.

Sensitivity studies done for AHWR with different evaluated data sets like, ENDF-B/VI, JENDL 3.2 and JEF 2.2 show significant differences. For example, the <sup>233</sup>U absorption reaction rates differ by 5% in thermal energy range and about 10-15 % in higher energy ranges.

The void reactivity studies for AHWR lattice also show a scatter of about 25 %. WIMS data set has been updated with isotopes of dysprosium for AHWR analysis.

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### 1.2 ADVANCED COMPUTATIONAL TOOLS FOR PHYSICS DESIGN

### Space-time Analysis code for AHWR

The knowledge of the space and time dependent behaviour of the neutron flux is important for the reactor safety analysis under operational and accidental conditions.

A time-dependent diffusion theory code called ARK3 (**A**dvanced Heavy Water **R**eactor **K**inetics in **3**-D) is being developed for the AHWR. The code ARK3 has option to use advanced Krylov



subspace based solution techniques. The space-time analysis code simulates the transient, due to the disturbed reactor steady state, by numerically solving the time dependent diffusion equations. The code is coupled with a visualization tool to plot fluxes as a function of transient time, at any planar cross-section of the reactor core. It is validated against a HWR benchmark problem, which simulates power rise due to half-core coolant voiding and subsequent control action. Representative snapshot flux profiles in two central planar cross-sections when power is

The code ARK3 is currently being used to analyze AHWR transients such as LORA, operational transient with xenon and validation of reactor physics software in AHWR simulator. This code will be coupled with the thermal-hydraulic analysis code for studying the combined effects of neutronic and thermal hydraulic behaviour.

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at a maximum value, are illustrated.

### ATES3 – Anisotropic Transport Equation Solver in 3-D

An accurate prediction of the time dependent multi dimensional & multi energy group neutron flux at successive time instants, is one of the main aspects of reactor physics design. There are primarily two main approaches: deterministic (S<sub>n</sub>, Collision probability) and Stochastic (Monte Carlo). Often, the reactor core calculations are done with diffusion theory, which is an approximation of the neutron transport theory, a deterministic approach. But an exact transport theory treatment is necessary in several cases such as high leakage reactors, for fluxes at the boundary and beyond, shielding analysis, verification of the approximate methods etc.

Recently, a neutron-gamma transport theory code, called ATES3, has been developed in 3-D Cartesian geometry for steady state criticality and external source problems. Apart from conventional methods of solutions, the code makes use of a few advanced Krylov subspace based schemes. The code is written in Fortran-90 language and has modular structure. These features make it more understandable and comparatively easier to modify. The

code ATES3 has been validated against a few international benchmarks and is being subjected for more rigorous testing. Figure below gives the material layout and the corresponding thermal flux shape for an LWR benchmark. The flux dip at the Control Rod (CR) location can be seen clearly.

As is well known, transport problems are highly memory and CPU time intensive problems, a single PC or workstation is not sufficient. Hence, it is very important to adopt the present code to parallel computers. Efforts are being made to parallelize the code on BARC's ANUPAM parallel systems. Incorporation of methods of solutions and user-friendly advanced features like visualization tools etc. are being incorporated.



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### Monte Carlo Technique: Code Development and Reactor Physics Simulation

Monte Carlo, as a tool in numerical analysis has gained wide spread applicability over the past few decades. The advent of high speed computing machines has been mainly responsible for the continual development of Monte Carlo method. Used properly, Monte Carlo can give quick "first cuts" at difficult problems, that is problems which are intractable by the traditional analytical or numerical techniques.

The greatest advantage of the Monte Carlo method is the exact simulation of the geometry. In deterministic methods only some special geometry can be simulated exactly, for irregular geometry some approximations must be considered. Monte Carlo method does not take any approximations in defining geometry. For this reason Monte Carlo method is essential for reactor calculations which involves complicated geometry e.g. Secondary shutdown system of 500 MWe PHWR, hexagonal geometry of CHTR, Nuclear Power Pack, Pebble bed reactors etc. as well as for deep penetration problems.

The main objective is to develop a general geometry Monte Carlo code with burn up, which will be used for criticality calculations, safety evaluations, accelerator driven sub-critical system's calculations, shielding calculations etc. with greater confidence and wider flexibility.

### **Development of Random Number Generator**

Random number plays an important role in any Monte Carlo calculation. The accuracy of the results depends on the randomness of the random numbers, its uniformity and its cycle length.

To provide uniform random sequences having larger cycle length required for Monte Carlo calculations a Random Number Generator (RNG) with large cycle length (2<sup>57</sup>) has been developed using bit manipulation technique. Some of its properties namely uniformity, Expectation Value, Variance, Frequency distribution, Auto-Correlation, Chi-square test etc. have been performed. It was compared with RANDU of PC in FORTRAN, RAND of PC in Basic, RAND of Honeywell DPS-8 System and RAN of PDP-11/23 and found to be superior among

Types of RNG Used	Auto correlation coefficient between neighboring bits				
	N & N+1	N & N+2	N & N+3		
Current*	- 0.0001173	-0.0001446	0.0000424		
Randy of PC in Fortran	-0.0018439	0.0024644	-0.01003		
Randy of PC in Basic	-0.0006032	-0.0078416	0.0064602		
And of Honeywell DPS-8	0.0054235	-0.0029316	-0.0019246		
And of PDP-11/23	0.0009187	-0.0040208	-0.0103830		

Comparison of Auto correlation coefficient

all in respects to its randomness, cycle length, uniform distribution etc. Correlation Coefficient of this RNG has been compared with that of different RNGs in the Table. It is seen that current RNG has been much closer to the expected value (the correlation coefficient between neighboring bits of a random sequence is expected to be zero). This RNG is ready to be used in any code, which requires large cycle of uniform random sequences.

### Development of 69-group spherical geometry Criticality Code

Monte Carlo code for criticality calculation has been developed for spherical geometry with WIMS 69 group energy treatment. This code is being extended for AHWR/ PHWR lattice cell with WIMS 69 group cross-section data.

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### Simulations of Reactivity Induced Transients for Thermal and Fast Reactors and Stability Studies

An accurate prediction of the consequences of an accident in a nuclear reactor is vital from the reactor safety point of view. This in turn requires the solution of coupled time-dependent neutron diffusion equations, time-dependent heat conduction equations and single and two phase coolant dynamics equations. All of these require large computer memory and computational time. Present day large-sized power reactors are neutronically loosely coupled. The looseness of the coupling is further enhanced by the deliberate flattening of the power distribution. The study of the neutronic transient behaviour under accidental conditions in such reactors requires accurate methods of solution of system of coupled multidimensional multi energy group time dependent neutron diffusion equation. Two distinct approaches exist for this purpose namely; the direct (implicit time differencing) and Improved Quasistatic (IQS) approach. Both the approaches need solution of static space energy dependent neutron diffusion equations at successive time steps.

A three-dimensional computer code 3D-FAST was developed based on Incomplete LU (ILU) preconditioned Biconjugate Stabilized method. The code was parallelized on ANUPAM distributed memory parallel system. The domain decomposition technique was used to create parallelism. The parallel



computational scheme was tested by analyzing a well-known Canadian PHWR benchmark problem, which simulates a loss of coolant accident.

The transient was simulated using two energy groups and  $52 \times 52 \times 40$  meshes. Twenty-nine space and energy dependent calculations were done with time step of the order of 0.1 sec. Table presents the CPU gain due to parallelization. The code was

disassembly of the core, which introduces sufficient negative reactivity. The calculation of disassembly reactivity requires the solution of coupled neutronics and hydrodynamics equations. A computer code for pre-disassembly calculations, which calculates coolant voiding, fuel melting, fuel and clad deformation and molten fuel slumping, is being developed. For the disassembly phase a computer code DISA is developed. This solves point kinetics equations coupled with two

Parallel System	No. of Slave processors	Solver CPU time (sec)	Speed-up by parallelisation	Parallel efficiency
	Sequential	803	-	-
ANUPAM-PC	2	410	1.95	97.5 %
	4	246	3.26	81.5 %
	10	130	6.18	61.8 %

CPU times for Parallelised BiCGSTAB[ILU] for IQS Approach (e =  $10^{-6}$ )

used to analyze the inadvertent withdrawal of two control rods along with drainage of light water from the zone controller units (ZCUs) for 540 MWe PHWR. Figures show the variation of reactivity and power as a function of time for this transient.

The accident analysis of fast reactors is generally carried out in two phases. The first phase is generally called as pre-disassembly phase and the second one as disassembly phase. In pre-disassembly phase, the transient is analyzed up to coolant voiding and fuel melting. The disassembly phase calculations are carried out with reactivity rates estimated from coolant voiding and fuel slumping. These transients are terminated by the











code by replacing the point kinetics calculations by multidimensional, multi energy group neutron diffusion code 3D-FAST. It is also planned to couple both phase calculations.

New concepts have emerged in the dynamics of nonlinear systems in last two decades. As part of our nonlinear studies in reactor physics we have studied some of these issues for a typical PWR. Here the dynamics refer to a single-phase coolant using point kinetics and a feedback through fuel & coolant temperature coefficient of reactivity. Typical scenario of limit cycle reactor operation were observed as a function of coolant temperature coefficient. The temporal behaviors can be identified for certain values of this parameter.

dimensional hydrodynamics equations. Figures show the equation of state for fuel used in DISA. The variation of net reactivity and power as function of time for a hypothetical transient in a typical fast reactor are shown in figures. It is planned to improve the neutronics model of pre-disassembly For a specific value of coolant temperature coefficient the critical state becomes an oscillatory state [limit cycle]. This latter state constitutes a new operational regime for reactor dynamics. These studies contribute towards understanding safety and performance of reactors.



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### 'XnWlup' Software for Reactor Physics Applications

As a result of the IAEA coordinated research program entitled "Final Stage of the WIMS library Update Project" new and updated WIMS-D libraries are generated by processing evaluated nuclear data files such as ENDF/VI.6, JENDL-3.2 and JEF-2.2. These WIMS-D libraries provide knowledge about the various relevant neutron-nuclear cross sections data in the form of 69/172 neutron energy groups. In order to help the WIMS-D library users to quickly view the plots of the energy dependence of the multi-group cross sections of any nuclide of interest, a computer program 'XnWlup' is developed for MS-Win operating system

using Microsoft Visual C+ +. It is also possible for the WIMS-D library users to compare the energy dependence of cross section data of various nuclides, different WIMS-D libraries and different temperatures.

The first version of this software 'XnWlup1.0' helps to obtain the histogram plots of the values of cross section data of an element/isotope as a function of energy. The second version of this software 'XnWlup2.0' is serving as an exhaustive equivalent handbook of WIMS-D cross section libraries for thermal reactor applications and used for comparing different WIMS-D compatible nuclear data libraries originating from various countries. The next version of this software 'XnWlup3.0' was developed to plot the cross sections of a resonant nuclide using resonance integral tabulated data of WIMS-D library for the given background dilution cross section and temperature. Also the revised software 'XnWlup3.0' is now capable of plotting either the resonance integral data as a function of dilution cross section for a selected temperature grid point or as a function of temperature for a selected dilution cross section grid point for a given resonance energy group.



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### 1.3 REACTOR SHIELDING AND IRRADIATION EXPERIMENTS

#### AHWR Shielding Experiments at APSARA

Shielding experiments essentially study the neutron and gamma streaming through duct geometries of shield models, to optimize and validate the actual reactor shield designs. Streaming experiments have been carried out at the shielding



Shield model assembled in the shielding corner

corners of the Apsara research reactor to optimize the end shield design of the AHWR. These experiments have generated data on neutron and gamma streaming through ducts of various sizes and shapes, present in the AHWR end shield. These duct geometries represent various configurations of lattice tubes and ventilation ducts.

The experiments included reference model without any streaming path, model simulating ECCS duct, model simulating stepped ventilation duct, lattice tube simulation model-1 (empty



inside cavity, water filled cavity and inside cavity filled with shield plug with and without water), lattice tube simulation model-2 (similar to the model-1 except that it is surrounded with a layer of steel balls and water). Results of the lattice tube simulation model-2 experiments, in which lattice tube, end fittings and shield plug have been simulated are given in the illustrations.

Measurements have been carried out along the center of lattice tube filled with air, water and shield plug. Radiation streaming through 3mm air gap between lattice tube and end fitting has also been studied. In all these experiments shield models simulating only one lattice tube is used. Experiments with a more representative model which has four lattice tubes and inbetween space filled with steel balls and water is underway.





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### Material Irradiation in Cirus / Dhruva Reactors for AHWR Program

### **Carbon Irradiation in Dhruva Reactor**

Samples of pyrocarbon have been irradiated in the Dhruva Reactor, to study the fast neutron fluence effect on its properties like wigner energy. A pyrocarbon sample has been irradiated in Dhruva reactor core, at location TR-15, 04 A for 1105.359 MWd. During the irradiation the fast neutron fluence, seen by pyrocarbon sample has been measured by activation method. In the measurement threshold neutron detectors like Ti, Ni and Fe have been used. After irradiation gamma activities of these irradiated detectors have been measured in a standardized high purity germanium detector coupled to a multi-channel analyzer. The fast neutron fluence (>1MeV) seen by the pyrocarbon sample was measured to be 1.28E + 18 n/cm<sup>2</sup> at 100 MW power of the reactor. Further irradiations of pyrocarbon samples are planned at different fast neutron fluence levels.

### **Dysprosium irradiation**

Dysprosium is used as a burnable absorber in the AHWR, as  $Dy_2O_3$  in  $ZrO_2$ , to achieve desirable reactor physics characteristics. Natural dysprosium mainly consists of five isotopes with mass number 160 to164. Dy-164 has largest absorption cross-section for thermal neutrons, the others have much lower but significant absorption cross sections. It is a chain absorber and the end product Holmium is a stable isotope, which is also a thermal neutron absorber. Some of the isotopes are good resonance absorbers for epithermal neutrons. It's suitability for AHWR is due to its slow burning characteristics and high radiation stability.

The cross sections of Dysprosium isotopes have not been validated systematically. Hence it has been planned to carry out irradiations of varying dysprosium content in  $ZrO_2$  pellets in Dhruva or Cirus tray rods over a period of time. Tentatively it is desired to have three different dysprosium weight % enrichment of 3%, 5% and 7% in  $ZrO_2$  pellets. Currently, pellets of  $Al_2O_3$  (95%) and  $Dy_2O_3$  (5%) have been made for initial studies.

It has been planned to carry out irradiations at low power along with thermal, epithermal flux monitors for short duration and another one for a longer duration for measuring isotopic composition change by mass spectrograph. The characteristic gamma activity of the flux monitors will be measured by a HPGe detector and the data will be analyzed to get thermal and epithermal neutron flux distribution in and around the pellets.





### 1.4 REACTOR SAFETY ANALYSIS

### Preliminary Safety Analysis of AHWR

Safety analysis is a major part of the process and licensing requirements of a nuclear power plant. The passive features of the AHWR cover entire range of normal operating conditions, transient conditions, accident conditions and beyond design basis accidents. The safety analysis exercise for the AHWR has been carried out in two parts namely deterministic safety analysis and probabilistic safety analysis. This article gives overview of the first facet. The deterministic approach requires establishment of set of Postulated Initiating Events (PIEs) as a starting point. The PIEs are selected from different categories based on phenomena encountered and AHWR specific reactor characteristics. This is followed by finalizing the reactor trip parameters, ECCS acceptance, fuel failure criteria and parameters required to build up the simulation model namely geometric, thermal-hydraulic and reactor kinetics with feed back. Safety analysis code RELAP5/MOD 3.2 has been used extensively to carry out all the analyses along with limited usage of subchannel analysis computer code COBRA IV.

Analytical computer code specific to AHWR plant model has been developed to address the variation in thermal-hydraulic parameters in each system expected from the PIEs. The plant model involves simulation of various systems and their components. This includes Main Heat Transport System (MHT), Emergency Core Cooiling System (ECCS), Containment, Isolation Condensers (ICs), Moderator System and MHT purification system. Other important modeling for systems to be mentioned in the plant models are steam drum pressure- level controller and reactor regulation- protection system.



A set of 55 postulated initiating events have been analysed for 750 MWt and up rated 920 MWt designs to address plant behaviour to judge the adequacy of the design with respect to ECCS acceptance and fuel failure criteria. Contribution from the analyses were utilized for deciding

- Design pressure of the containment and main heat transport system
- Defuelling refuelling time to avoid boiling crisis,
- Necessity of compartmentalization of ECCS header to avoid core starvation on ECCS header break,
- Assessment on hot shut down condition with ICs submerged in GDWP,
- Assessment of relief devices capacity to ascertain the integrity of pressure boundary
- Assessment of adequacy of ECCS along with mode of in core safety injection.

Studies are also carried out for optimization of ECCS design parameters and evaluation of moderator as an ultimate heat sink.

This whole task has been achieved with active participation of the analysis group, reactor trip committee, task force committee, design review committee, preliminary safety analysis review committee and safety documentation committee. In the light of number of design changes in 920 MWt design and re-assessment of reactor trip parameters, a fresh evaluation of the proposed reactor will be carried out with respect to ECCS acceptance and fuel failure criteria covering pre-decided PIEs. Various optimization studies are underway. Limited uncertainty analyses has been done to support best estimate evaluation.

Clad surface temperature transients at different axial locations for maximum rated reactor channel for event like 200% Inlet Header Break is illustrated.

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### Level-1 PSA Study of the Advanced Heavy Water Reactor

Preliminary Level-1 Probabilistic Safety Assessment (PSA) study of the AHWR, has been carried out to obtain integrated statement of Core Damage Frequency (CDF). Important accident sequences suggest design modifications and to obtain insights about important dependencies and human interactions important to safety. Based on this study, design modifications have been suggested in emergency core cooling system, isolation condenser system, end shield cooling system and gravity-driven water pool recirculation system. This study has indicated that the core damage frequency of this reactor based on suggested modifications is of the order of 10<sup>-</sup> <sup>7</sup> per reactor year. The dominating initiating event contributing to this CDF value is small LOCA. In arriving at this figure, it has been assumed that the passive systems are completely reliable. However, this needs to be ensured by carrying out thermal hydraulic studies. Also, the contribution from small LOCA to CDF will reduce if Leak Before Break criteria is incorporated in the design. Revised PSA study of this reactor is being carried out using the latest design documents.

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### Assessment of Leak Rate from the Cracked Pipe

The Leak Before Break(LBB) concept is employed in defense in depth concepts of reactor design to avoid any unstable failure in the pressure boundary, namely carbon steel pipes and Zirconium made pressure tubes. The LBB in the pressure boundary can be ensured when detection, confirmation and location of the leak are carried out and the reactor is placed in a depressurized condition before the crack exceeds the critical crack length. The LBB concept has been applied for designing the AHWR reactor.



Comprehensive theoretical and experimental study is going on to support the LBB concept applied to Primary Heat Transport System and steam lines for the AHWR and the PHWR. The theoretical study includes elasto-plastic model development for crack opening area and thermal-hydraulic model development for estimating critical flow rate through crack and slits. Over the years various flow models are developed and validated against the published data specific to Pressurised Water Reactors (PWRs). Prediction of crack flow with crack angle variation for PHWR Steam Generator (SG) outlet pipe is furnished (Fig.). Influence of different flow models namely Henry's Homogeneous Non-Equilibrium Model (HHNM) and Homogeneous Frozen Model (HFM) on crack flow rate is also shown. A Comparison of the computed flow with flow from detectable Leakage Size Crack (LSC) gives an idea of the order of semi crack angle which gives a detectable crack flow.

An experimental setup is built at Mechanical Engineering Department, Jadavpur University, Kolkata to estimate the crack/ slit flow under high pressure and high temperature conditions simulating the pressure and temperature condition of primary



heat transport system of the PHWR and the AHWR. The maximum operating parameters are 90 bar and 250° C. The facility consists of three systems namely the heating system achieved through oil-fired thermic fluid heater, the pressurization system through nitrogen system and the high pressure and high temperature system which consist of buffer chamber and test section. The system is instrumented with pressure, flow and temperature sensors. Safety devices and control logic interlocks have been incorporated to ensure safety in the loop. Four close circuit cameras continuously monitor the health of the system. Hot commissioning followed by some experiments in slits has been carried out. Series of experiments are planned with slits and cracked pipe to generate a robust database with different pressure and different sub-cooling to validate the developed computer code.

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### Nuclear Containment safety Research for Beyond Design Basis Accidents

For the ultimate load capacity assessment of Indian reactors, in-house finite element codes ULCA for static inelastic analysis and ARCOS3D for transient dynamic inelastic analysis have been developed. These computer codes have been extensively benchmarked with experimental test results of Sandia Laboratory, NUPEC, Japan and USNRC-sponsored containment model test and NUPEC, Japan sponsored seismic shear wall tests. Indian containments have been shown to have a factor of safety of more than 2 over the design pressure with these codes.

Recently Sandia Laboratory, USNRC and NUPEC, Japan cosponsored a round robin analysis program to determine the ultimate load capacity of a 1:4 size steel lined Pre-Stressed Concrete Containment Vessel (PCCV) model which represents the Pressurized Water Reactor (PWR) containment of Ohi-3 nuclear power station in Japan. BARC ULCA code pre test predictions have been shown to be in excellent agreement with the experimental limit state test (LST) and subsequently the Structural Failure Mode Test (SFMT) results released in 2003 by the organizers. The PCCV model has the design pressure ( $P_d$ ) of 0.39 MPa and the best estimate of the ultimate pressure of 3.15  $P_d$ , the lower bound conservative estimate of the minimum pressure of 2.8  $P_d$ , which the test model would at least reach during the test and upper bound estimate of the maximum collapse pressure of 3.45  $P_d$ ; beyond which the test model is unlikely to remain in pressurized condition were predicted with the in-house code with 90% confidence level as per the requirements of the round robin analysis activity. The subsequent test carried out at Sandia showed that the model had



Sandia Containment Model

the limit state test pressure of 3.3  $P_d$  and the structural failure pressure of 3.6  $P_d$ . BARC computational results have been found to be among the best of all the predictions.



PCCV Model Burst Test View

The pre-test predictions with BARC code ULCA published in international journals before the release of the experimental data are in excellent agreement with the test results.



FEM Model Prediction of Failure



Displacement response comparison with experimental data released by Sandia Lab

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## Evaluation of Ultimate Load Capacity of AHWR Containment Structure

The nonlinear finite element analysis of the inner containment structure of Advanced Heavy Water Reactor has been carried out to determine its ultimate load capacity. The deformed shape of the containment at the ultimate pressure as shown in figure and the internal pressure versus deformation curve at the mid height of cylinder shown in figure obtained from the analysis illustrates a factor of safety of approximately 2.78 over the design pressure of 0.185 N/mm<sup>2</sup>. The major safety-related highlights from the analysis are:

- The crack initiation starts at internal pressure of 0.462 N/mm<sup>2</sup>
- The through thickness cracking is observed at an internal pressure of 0.467 N/mm<sup>2</sup>
- The reinforcement yielding starts at an internal pressure of 0.510 N/mm<sup>2</sup>
- The 0.2 mm crack width is observed at an internal pressure value of 0.515 N/mm<sup>2</sup>





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#### 1.5 DESIGN AND ANALYSIS OF AHWR FUEL

## Design of AHWR fuel

The AHWR fuel has been designed to meet the requirement of thermal hydraulics, reactor physics, fuel handling, and reconstitution (i.e., replacement of outer ring of irradiated (Th-Pu)O<sub>2</sub> fuel pins with fresh ones). The vertical pressure tube configuration has guided the structural design of the fuel assembly. The fuel assembly is 10.5 m in length and is suspended from the top in the coolant channel.

The AHWR fuel assembly consists of a fuel cluster sub-assembly and two shield sub-assemblies: shield 'A' and shield 'B'. These sub-assemblies are connected to each other through a collet joint. The AHWR fuel cluster contains (Th-<sup>233</sup>U)O<sub>2</sub> and (Th-Pu)O<sub>2</sub> fuel. The fuel cluster has 54 pins arranged in three rings –inner most ring of 12 pins of (Th-<sup>233</sup>U)O<sub>2</sub> with 3.0% <sup>233</sup>U enrichment, middle ring of 18 pins of (Th-<sup>233</sup>U)O<sub>2</sub> with 3.75% <sup>233</sup>U enrichment and outer most ring of 24 pins of (Th-Pu)O<sub>2</sub> with Pu enrichment of 2.5% in the upper half and 4.0% in the lower half of the pin. The fuel pins are assembled in the form of a cluster, with the help of the top and bottom tie-plates, with a central rod connecting the two tie-plates. The central rod has a burnable absorber Dy<sub>2</sub>O<sub>3</sub> in ZrO<sub>2</sub> matrix in the form of capsules. Each capsule consists of 12 pins. The central rod has provision for spraying of Emergency Core Cooling System (ECCS) water directly over the fuel pins in the event of Loss of Coolant Accident. Six spacers along the length of the cluster provide the intermediate pin spacing and stiffness to the cluster. Various components of the fuel cluster like dummy fuel pins, shields, top and bottom tie plates, spacers, central rod, collet joints have been designed and fabricated. A dummy fuel cluster has been assembled and pressure drop test, endurance test and vibration tests have been carried out for the cluster at Flow Test Facility and Integral Test Facility. A short length model of the fuel cluster showing different components of the fuel cluster has been made for display and for better understanding of the fuel cluster.

The injection of water from the holes of the central rod was visualised in a test set up and its flow versus pressure drop characteristics were obtained. A set up has been designed for studying the wetting of the fuel pins and effectiveness of spray



cooling of ECCS water using electrically heated pins. Leakage of water from the flexi collet joint has also been evaluated.

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#### Fuel cycle analysis for AHWR

The AHWR follows a closed fuel cycle with the objective of utilisation of plutonium to a minimum and self-sustaining in <sup>233</sup>U. The <sup>233</sup>U required for the equilibrium core is planned to be generated in-situ. Hence, the initial core of the AHWR would consist of all (Th-Pu)O<sub>2</sub> fuel clusters. There will be a gradual transition from initial core (Th-Pu)O<sub>2</sub> clusters to equilibrium core containing both (Th-Pu)O<sub>2</sub> and (Th-<sup>233</sup>U)O<sub>2</sub> fuel. The process losses of fissile and fertile material in the reprocessing and refabrication should be considered while carrying out the fuel cycle evaluations. The fuel cycle analysis detailing the movement of fissile and fertile materials for various combinations of design parameters – from initial core to transition core to equilibrium cored is being carried out.

#### Near term fuel cycle for the AHWR

The AHWR fuel cycle for the near term, is a closed fuel cycle, envisaging recycle of both fissile <sup>233</sup>U and fertile thoria. The fuel cycle time is eight years - four years of in-reactor residence time, two years of cooling, one year of reprocessing and one year of re-fabrication. For the initial few years, annual reload would consist of only (Th-Pu)O, clusters. The spent fuel cluster before reprocessing would undergo dis-assembly for segregation of (Th-Pu)O<sub>2</sub> pins, (Th-<sup>233</sup>U)O<sub>2</sub> pins, structural materials and burnable absorbers. The (Th-<sup>233</sup>U)O<sub>2</sub> pins will require a two stream reprocessing process i.e. separation of thorium and uranium whereas the (Th-Pu)O<sub>2</sub> pins will require a three stream reprocessing process i.e. separation of thorium, uranium and plutonium. Part of the recovered thoria will be recycled into the reactor immediately by using it for the fabrication of (Th-<sup>233</sup>U)O<sub>2</sub> pins. The rest will be stored for 17-20 years by which time it would be similar to freshly mined thorium in radioactivity and hence will be used to fabricate (Th-Pu)O<sub>2</sub> pins.

#### Long term fuel cycle for the AHWR

India's three-stage nuclear power programme, based on closed fuel cycle, comprises Pressurised Heavy Water Reactors (PHWRs), Fast Breeder Reactors (FBR) and systems for thorium utilisation.

Natural UO<sub>2</sub> from PHWRs produces plutonium which along with Depleted Uranium will be used in FBRs for power generation and fissile material multiplication. The plutonium required for the AHWR will come from the PHWRs. Over a long period of time, part of the plutonium from FBR may also be used in AHWR. The plutonium from reprocessing of AHWR fuel has low fissile content and will be sent to FBR.

Partitioning of waste is planned for AHWR. The minor actinides like Np, Am and Cm would be separated from the waste. Similarly higher decay heat materials like Strontium and Cesium and long-lived fission products like lodine and Technetium would be separated from the bulk of the waste. The R&D work on partitioning has already commenced. The partitioning technology developed will be equally applicable to both PHWR and AHWR. Transmutation of minor actinides like Neptunium, Americium and Curium and long-lived fission products like lodine and Technetium, segregated from bulk of the waste from AHWR and PHWR, is planned for the long-term fuel cycle for AHWR. The design of AHWR offers flexibility to incorporate a wide range of fuel cycles, including those based on (Pu-Th)O<sub>2</sub> alone.





## Fuel analysis for AHWR

The performance analysis essentially involves the thermo-mechanical analysis of fuel by modelling the various phenomena that take place during its irradiation. The analysis has been carried out by employing the methodology used for the well-established  $UO_2$  fuel using GAPCON-THERMAL code. The available database on the thoriabased fuels is quite small. However, even the most conservative estimates indicate that the thoria-based fuels have an inherent advantage over the  $UO_2$  fuel in terms of the two major performance bearing parameters thermal conductivity and fission gas release. Thermal conductivity is considerably higher and the diffusion coefficient, which is a measure of fission gas release, is lower for thoria-based fuel. The thermo-physical properties of (Th-Pu)O<sub>2</sub> fuel have been incorporated in the code for carrying

out the analysis. The salient results of the analysis for various limiting parameters are given in the Table.

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## **1.6 CRITICAL FACILITY**

A low power Critical Facility is under construction as part of over-all technology development program to validate the physics design of the thorium-based Advanced Heavy Water Reactor (AHWR) and for validation of safety parameters for 540/700 MWe PHWRs as well as advanced fuel designs for the PHWRs.

The nuclear design of the lattice and core of AHWR envisages the use of a novel 54-pin MOX cluster with different enrichment of <sup>233</sup>U and <sup>239</sup>Pu in Thoria fuel pins with a dysprosium displacer rod at the centre. The nuclear design of the lattice and core of AHWR envisages the use of a novel 54-pin MOX cluster with different enrichment of <sup>233</sup>U and <sup>239</sup>Pu in Thoria fuel pins with a dysprosium displacer rod at the centre. It is designed to have negative void coefficient of reactivity. Fuel assembly characteristics are evaluated by using transport theory codes WIMS or CLUB with 69 or 172 group library. The core calculations are done with few group diffusion theory methods. For new fuel assembly concepts, as in the AHWR, lattice and core characteristics have to be measured in Critical Facility to validate the above calculations.



The Critical Facility is designed to facilitate study of different core lattices based on various fuel types, moderator materials and reactivity control devices. The reactor is designed for a nominal fission power of 100 W with an average flux of 108 n/cm<sup>2</sup>/s. The design provides flexibility to arrange the fuel inside the core in a precise geometry at the desired pitch. Reactor criticality is achieved by the manual control of moderator level in the core.

Initially, three types of cores using heavy water as moderator and reflector will be studied. The three cores are based on different fuel types viz.

- 19 pin natural uranium metal fuel clusters to constitute the reference core.
- 54 pin (Th-Pu) MOX/ (Th-<sup>233</sup>U) MOX clusters to constitute the representative AHWR core.
- 37 pin natural uranium oxide fuel clusters to constitute the 540 MWe PHWR core.

## **Reactor Structure**

The reactor consists of a cylindrical Aluminum tank (Reactor Tank) of 330 cm ID and 500 cm height to accommodate the fuel, moderator and the shut-off rods. The reactor tank is adequately sized to provide a radial reflector thickness (heavy water) of about 40 cm around the core. The reactor tank, which is open at the top, is connected to a square box housing the lattice girders from which the fuel assemblies are suspended. Shut-off rod head-gears are also supported by these girders. The lattice girders are designed to provide the flexibility of arranging the fuel clusters in any desired square lattice pitch ≥206 mm. The reactivity addition/regulation of the reactor is carried out manually through controlled addition of moderator heavy water to the reactor tank at a pre-determined rate, to ensure safe limits of reactivity addition rate. Reactor protection system comprises fast gravity-driven solid shut-off rods. During long-term shutdown of the reactor, moderator dumping will maintain.

#### **Reactor Block**

The reactor is housed in a concrete reactor block. The concrete wall of the reactor block provides shielding in radial direction. Shielding in axial direction is provided by two motor-operated movable shield trolleys provided at the top. The reactor block is located inside a reactor building.

The reactor tank, square box and lattice girders which support fuel and control assemblies are housed inside a cavity in the reactor block. The reactor tank accommodates the fuel, moderator and shut off rods. The tank is open at the top to facilitate connection with a square box through an elastomer expansion joint.

The square box houses the lattice girder assemblies. The bottom plate of the square box has a 3300 mm diameter opening in the centre to provide full access to the reactor tank. The top



plate of square box has a 3350 mm diameter opening at the centre to allow access to the lattice girders. The opening is covered by a 25 mm thick revolving plate supported on a bearing, which in turn rests on the square box top plate. An oil seal is provided between the fixed top plate of the square box and revolving plate to confine the cover gas within the square box and to avoid ingress of atmospheric air and moisture into the heavy water contained in the reactor tank.

Lattice girder assemblies are designed to support the fuel

assemblies and reactivity control devices and offer flexibility of configuring the desired core lattice, at the required pitch. The lattice girder assemblies can be spaced as desired with the help of centre zero scale provided along the side of stainless steel rail beams.



Reactor Tank Under Fabrication



Square Box

#### Moderator and Cover gas system

For the initial set of experiments heavy water will be used as the moderator and the reflector. Nitrogen gas is used as the cover gas for the heavy water. The system is designed to supply the required inventory of heavy water in a controlled manner to the reactor tank to attain reactor criticality for various core configurations / reactor experiments. Since the reactor is designed to operate at a very low power level, no dedicated core cooling system has been provided. The small amount of heat generated in the core is transferred to the moderator and cover gas and gets eventually dissipated into atmosphere by natural convection.



### **Control Instrumentation**

For reliable neutronic power measurement and reactor protection, neutron detectors and associated electronics have been used with sufficient redundancy. The neutron detectors are located in the graphite fillers below the reactor tank as there is substantial difference in expected critical heights of AHWR and PHWR core configurations. For reactor start up from source range two pulse channels are provided. For power measurements and protection in intermediate & power range, six DC channels, consisting of three Log-linear channels (for providing Log-rate and Linear Power signals for manual regulation of the reactor power), two safety channels and a multi-range DC channel (having seven ranges with full scale values of 0.5 mW to 500 W), are provided. Adequate conventional instrumentation is provided to monitor and record the process parameters such as flows, pressures, temperatures, levels, etc. and to generate trips/alarms. An in-core flux mapping system consisting of an array of 25 LEU-based miniature fission counters, working in pulse mode, will provide three dimensional flux distribution in the core.



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Range Overlap Diagram of Nuclear channels
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## **Shutdown Devices**

The fast-acting shut off rods are used as primary shutdown system. Fast shut down of the reactor on a trip signal is achieved by gravity fall of six-cadmium (Cd encased in Al) shut off rods into the core. The shut-down system is designed to provide sufficient negative reactivity insertion reliably, for fast reduction of reactor power to render the reactor sub-critical following a reactor trip. The shut off rods are normally parked above the core region. The total worth of



Shut-Off-Rod Head Gear under Endurance

the shut off rods is about 68 mk and can vary depending on the core configuration. Shut off rods can be withdrawn from the core, one at a time in a predefined sequence, only when the trip is reset, and are parked in their normal parking positions. On a reactor trip signal, in addition to the insertion of the shut-off rods, moderator is also dumped from the reactor tank up to a predefined level by opening two fast-acting dump valves to provide adequate shut down margin, independent of the shut off rods, in the most reactive core.

#### **Experimental studies**

Initially, three types of cores using heavy water as moderator and reflector are planned to be studied.

The AHWR representative core of the Critical Facility is a variant of the reference core, wherein, initially the central nine natural uranium clusters of the reference core will be replaced by 54 pin (Th-Pu)/(Th-<sup>233</sup>U) oxide clusters to simulate the initial and equilibrium AHWR core. Eventually the core will be made critical with (Th-Pu)/ (Th-U<sup>233</sup>) oxide clusters alone. The central cluster in the representative core is expected to have the spectrum very close to actual AHWR spectrum.

In order to facilitate extensive measurements, the central cluster is designed for removal of fuel pins from the cluster, placement and retrieval of foil detectors from the removable fuel pins and features to fill/ drain the desired fluid into/ from the cluster for experiments with different coolants as required.

In critical facility, experiments are done in cold, clean conditions. Thus only simulation models and nuclear data are validated in these conditions directly. As the temperature of moderator in CF cannot be raised above 60°C, it will be difficult to measure fuel temperature coefficient. Possibility of carrying out these measurements by electrically heating central cluster is underway. Some typical measurements to be carried out are listed below:

#### **Experiments planned for AHWR**

 Critical height and level coefficient measurement for various lattice pitches involving both (Pu-Th) oxide and composite

Reference Lattice Core (RLC)		19 rod cluster of Natural Uranium 61 lattice locations (6 for SORs) 245 mm pitch
AHWR	·	54 pin AHWR clusters in Central 9 positions of RLC245 mm pitch
PHWR 540 MWe	•	Six bundles of 37 pin natural UO <sub>2</sub> fuel per channel 69 lattice locations (6 for SORs) 286 mm pitch

- Measurement of flux profile, reaction rates and the ring power factor inside the removable fuel pins of experimental fuel cluster.
- Measurement of Dynamic and static worth of reactivity devices
- Lattice cell parameters
- Assessment of coolant void reactivity
- Measurement of Worth of Dysprosium burnable poison in ThO<sub>2</sub> & (Th-Pu)O<sub>2</sub> and (Th-<sup>233</sup>U)O<sub>2</sub> in AHWR clusters.
- Reaction rate and initial conversion ratio measurements
- The flux distribution measurements with Cu/Au wires and foils
- On-line measurements by distributed set of Fission Chambers.
- Spectrum measurements by Lutetium and other activation detectors – High Purity foil materials (Cu, Au) for thermal, with Cd cover for epithermal and threshold detectors like In, Ni, Fe, Ti, Zn etc, <sup>237</sup>Np coated foils to be used with SSNTD for fast neutron measurements
- Special fuel foils for measuring various reaction rates.
- U foils of different <sup>235</sup>U/<sup>238</sup>U contents
- Pu foils with different <sup>239</sup>Pu/<sup>240</sup>Pu contents
- <sup>233</sup>U foils with different <sup>233</sup>U/<sup>234</sup>U content.

For experiments in the reference core, the metallic uranium cluster at the centre of the core has fuel pins with activation detector foils. Five fuel pins in this cluster are of removable type to facilitate insertion and removal of the activation detectors. Each removable fuel pin has six flux measuring foils. The flux measuring foil is clad in Al/Cd.



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#### Fuel design for Critical Facility

The lattice physics experiments will be carried out in three phases, i.e., Reference core, AHWR representative core and PHWR core.

## Natural metallic uranium cluster for reference core

The reference core will contain natural metallic uranium clusters at 55 lattice locations arranged at a square pitch of 245 mm. The metallic uranium cluster will be hung from the top of girders through an extension assembly consisting of a top adaptor and an extension rod. The cluster consists of 19 pins of uranium clad in aluminium arranged in two rings of six and twelve pins around a central pin. They are held between top and bottom tie plates, which are welded to the aluminium fuel tube. The



fuel cluster will not be in contact with coolant. The top tie plate has provision for attachment of the extension rod through pins. All the fuel pins required for natural metallic uranium clusters have been fabricated for reference core of critical facility. Thirty eight assemblies have been made.

## Experimental metallic uranium cluster

For reactor physics experiments in the reference core of critical facility, the metallic uranium cluster at the centre of the core will be of removable pin type for placing activation detectors in it. The cluster has provisions for the following:

- Five pins with provision for placing activation detector foils
- Foils wrapped around these pins at select locations
- Ø1.0 mm wire inside the cluster along its complete length
- Ø1.0 mm radial wire as chords, at three elevations.





## **Experimental Thoria Cluster**

One thoria cluster containing 19 thorium dioxide pins will be loaded at the central lattice position of the reference core. Reactor physics measurements, including reaction rate, flux measurement, level coefficients, void reactivity etc. will be made with this experimental thoria cluster. The cluster will have provisions for placing activation detectors.

## **AHWR Representative Core**

The AHWR representative core will be constituted by replacing central 3 x 3 array of natural metallic uranium clusters by AHWR fuel clusters. They will be hung from the top in a similar way to the reference core clusters. The experiments will be carried out with sets of clusters – 1<sup>st</sup> set consisting of nine all (Th-Pu) MOX cluster and 2<sup>nd</sup> set of nine clusters containing (Th-Pu)MOX and



(Th-<sup>233</sup>U)MOX pins. One cluster from each of the sets will be an experimental cluster with features of removable pins and removable dysprosium assembly (central rod).

The design of all (Th-Pu) MOX cluster for AHWR-CF (AHWR fuel cluster for Critical facility) has been finalised. The design has been made amenable for remote assembly. A dummy cluster of AHWR-CF fuel made up of SS components has been

fabricated. This cluster will be used to demonstrate the assembly and testing of the handling tools (for pins and for cluster as a whole). Short length solid S S bars are used to simulate fuel pins but other components like tie-plates, closure plug etc. are of full scale. Removal type of central rod has been made and will also be used for tool testing.

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## Design and Development of Shut-off Rod Drive Mechanism

The shut-off rod drive mechanism is designed with several advanced features. However, it is the requirement of designing drive mechanism within the given space constraints which makes its design a custom built. As the mechanism is custom built, its design is qualified and proven for reactor use through prototyping and subjecting it to life-cycle testing on full-scale test facility.

The prototype drive mechanism has been tested for more than 5000 cycles on full-scale test facility. Design basis report has been



Prototype headgear unit



completed and also reviewed for safety qualification. Presently, nine drive units are under advanced stage of manufacture.

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## Gamma scanner to validate design parameters of low irradiated fuel pin of advanced heavy water reactor in the critical facility

A fuel pin Gamma scanner has been developed to validate physics design parameters of fuel pin for the Advanced Heavy Water Reactor in the Critical Facility. It is a fully automated, computerinterfaced radiation measurement system for scanning the irradiated fuel pin by a pair of Sodium Iodide (NaI) detectors and one Germanium detector (HPGe), for collecting collimated gamma spectra in three directions. The data generated will facilitate the optimization of the fuel pin design.

The system is designed to impart cyclic linear pull and rotate the fuel pin through predetermined angle and supersedes other engineering techniques using lead screws, cams etc. which are not viable for use in conjunction with long fuel pins. The device technique developed and validated as shown in the figure confirms with a fitting solution in particular holding the fuel pin hollow cluster to a predetermined non-superfluous force.

The gamma scanner developed is primarily suitable for low irradiated fuel pins. At this stage conceptualization and fabrication aspects of basic devices of gamma scanner like development of twin chuck, it's up-gradation for adaptability to scan different fuel pins, computer interface, cyclic device sequence programming and its automated operations are validated and standardized. In future, this technology will be implemented in the following areas.

- Development of compact gamma scanning mechanisms in conjunction with counterweight techniques for underwater gamma scanning of irradiated fuel elements.
- Development of portable mechanisms to facilitate in situ
  Gamma Scanning of fuel pins encased in a heavy cast.
- Measurement of inside diameters of clusters of long lengths.
- Development of heavy-duty compact gamma scanning mechanisms in conjunction with counter weight techniques to scan fuel elements up to 100 mm in dia and 5 meters in length and weighing approximately 50 kg.



Gamma Scanner for low irradiated fuel pins



Actual photograph and components of innovative twin chuck concept

## 1.7 DESIGN OF COMPONENTS AND PROCESS SYSTEMS OF THE AHWR

## Design of Process & Safety Systems of the AHWR

Process design of major reactor systems in the AHWR nuclear island and steam cycle involves finalisation of process parameters, design of process equipment, analysis for system behaviour under various conditions (normal, upset, accident, faulted), preparation of flow sheets, Piping and Instrument Diagrams (P&ID), system & equipment layout.



Handing over of AHWR Detailed Project Report to Dr. Anil Kakodkar, Chairman, AEC by Mr R.K. Sinha, Director, RD&DG

The Design Basis Reports (DBRs) and Detailed Project Report (DPR) of process and safety systems of 920 MW<sub>t</sub> AHWR have been prepared. Peer review of detailed project report of AHWR has been completed by NPCIL and review comments have been incorporated in the design.

#### Main Heat Transport System

The Main Heat Transport (MHT) System of AHWR transports heat from fuel rods to steam drum. AHWR uses boiling light water as the coolant in the high-pressure MHT system and heavy water as the moderator in a low-pressure system. The coolant recirculation in the primary system is achieved by two-phase natural circulation, which depends on the density difference between the hot and cold legs of the primary loop. This mode of core cooling is adopted not only during normal operation but also during transients and accidental conditions.

The MHT system consists of a common reactor inlet header from which 452 inlet feeders branch out to an equal number of fuel channels in the core. The outlets from the fuel channels are connected to tail pipes, 113 of which are connected to each of the four steam drums. From each steam drum, four down comer pipes are connected to the common inlet header.

During normal operating conditions, the steam drum pressure is maintained at 7 MPa. The water level in the steam drum at nominal operating conditions is 2.2 m. The primary loop circulation rate maintained by the density difference is



approximately 2237 kg/s at nominal operating conditions. The average core exit quality is about 18.7 % for the rated reactor operating conditions.

The two-phase mixture leaving the core is separated into steam and water in the steam drum. The steam-water separation in the AHWR steam drum is achieved naturally by gravity without the use of mechanical separators. During hot shut down condition, the decay heat is removed using main condenser. However, in case of non-availability of main condenser, Isolation Condensers will remove the decay heat in passive mode.

#### **Emergency Core Cooling System**

The Emergency Core Cooling System (ECCS) in the AHWR provides cooling to the fuel in passive mode during first fifteen minutes of LOCA by high pressure injection from Advanced Accumulator and later for three days from the Gravity Driven Water Pool. The advanced accumulator is designed to provide a large amount of cold water directly into the core in the early stage of LOCA and then switches passively to inject small amount for a longer duration to quench the core. This passive switching is achieved by the Fluidic Flow Control Device of the advanced accumulator.

The system consists of four accumulators containing 240 m<sup>3</sup> of water pressurised by nitrogen at 5.0 MPa pressure. During the event of large pipe rupture i.e., the inlet header rupture, MHT pressure falls due to blow down, and cold water from the accumulators enters the reactor core. The high pressure injection lines are provided with rupture discs, check valves and isolation valves. The low pressure injection from GDWP initiates after isolation of accumulators by closing the valves.



#### Gravity Driven Water Pool

The Gravity Driven Water Pool (GDWP) functions as a heat sink for Passive Concrete Cooling System, condenses the steam flowing through the isolation condensers during reactor shutdown and sources low pressure Emergency Core Cooling (ECC) injection for removing decay heat for three days following LOCA. The GDWP also functions as a suppression pool to cool the steam and air mixture during LOCA.

The GDWP is located in the dome region of the reactor building and contains 6000 m<sup>3</sup> of water inventory. Stainless steel lining of 6 mm thickness is provided inside the GDWP. There are eight compartments in the GDWP which are interconnected to each other. Each compartment of GDWP contains one Isolation Condenser for core decay heat removal during shutdown and is provided with outlet nozzles at various elevations for flow of water into the core during LOCA. The Passive Containment Coolers are located below the GDWP for containment heat removal during and after LOCA.

The GDWP recirculation and cooling system consists of four loops. Each loop consists of a heat exchanger, a pump, a filter, an ion exchanger and a chemical addition tank for maintaining water chemistry. These loops operate in rotation with two of them operating at a time. A bypass purification line, consisting of a filter and an ion exchanger, is provided to remove the suspended and ionic impurities so as to maintain the crud concentration within limits.

To prevent biological growth in GDWP, its water will be continuously recirculated by the re-circulating pumps. The recirculation flow will be maintained by two pumps operating at a time on rotation basis. The GDWP re-circulation system takes care of re-circulating, filling and draining the water in each compartment. Each compartment has a separate inlet and outlet line for supply and drain of pool water respectively.



## Important Design Features of Process & Safety Systems to Improve Plant Performance & Safety

#### Moderator heat utilisation

Heat generated in heavy water moderator and heat transferred from fuel channel to moderator is generally transferred to process water in circulation loop, which goes as waste heat. In AHWR, the moderator system is designed to utilize the optimum quantity of heat to improve the thermal power output at Turbine Generator (TG). Out of 58 MWt moderator heat, 36 MWt is utilised to heat the condensate feed water before the LP heaters in steam cycle. With heat utilization the condensate is heated from 44°C to 68°C, resulting in saving of steam extraction requirement of 51.72 T/h from LP Turbine and thereby generating 3.42 MWe power.

The moderator outlet temperature is increased to utilize the maximum heat. The heat utilization is optimized keeping the sufficient time margin ( $\sim$ 3-4 h) for moderator to reach boiling temperature during station black out condition and also

based on the limitations like size of the heat exchanger and temperature cross.

The economic benefit in terms of electric power gain is considerable compared to capital cost involved in additional to and fro feed water piping from turbine building to reactor building, additional heat exchangers and valves. The efficiency of steam cycle is 33.4% after utilisation of moderator heat.

## Passive Moderator Cooling System during Station Black Out

Passive moderator cooling is designed to avoid boiling of moderator and to avoid release of tritium from moderator cover gas due to pressure rise in the event of Station Black Out (SBO). Feasibility study is completed and detailed design is under progress.

The passive moderator cooling system removes 2 MW moderator heat by a heat exchanger suitably elevated from the heat source center i.e. core. Both tube side heavy water

moderator and shell side Gravity Driven Water Pool (GDWP) water circulate in the heat exchanger by buoyancy force. GDWP with 6000 m<sup>3</sup> water inventory serves as heat sink. Heat exchanger and the connecting piping are designed to minimise the pressure drop to achieve the required flow by natural circulation.

During Station Black Out (SBO) condition due to unavailability of forced circulation & cooling, moderator temperature starts increasing due to heat generation in moderator and heat transfer from hot main heat transport system (MHT) at 285°C. MHT temperature remains hot during station blackout without any crash cooling provision while Isolation Condensers (ICs) removing core decay heat by passive valves in operation. Analytical estimation shows a moderator bulk mean temperature of 90°C in 1 h 45 min and cover gas pressure reaches relief valve set pressure of 1.5 kg/cm<sup>2</sup> (abs.). The temperature further rises to 100°C in three hours and continues to rise with time in the absence of cooling provision. With the passive cooling provision and 2 MW moderator heat removal, the cover gas pressure and moderator temperature of the system is maintained below the relief pressure setting and saturation temperature.



#### Passive Shutdown Device

The Passive Shutdown Device (PSD) is an additional provision to effect shutdown in the event of over pressure in the MHT system due to the failure of wired mechanical ShutDown System (SDS-1) and liquid poison injection in moderator (SDS-2) due to any malefficient activity. PSD adds liquid poison to moderator from a pressurized poison tank, by actuating a valve passively driven by steam pressure. A rupture disc is provided between the passive valve and the steam drum for releasing the steam to valve by rupture action at over pressure above the trip set pressures for SDS#1 or SDS#2.



#### Passive concrete cooling system

Passive concrete cooling system protects the reactor V1 volume concrete surface from high temperature MHT, by natural circulation of water in tubes near the concrete structure. This passive concrete cooling system has eliminated blowers and associated power supply failure events. The concrete temperature is maintained below 55°C by the optimum number of cooling pipelines, over the conventional cabinet insulation panel surrounding MHT The natural circulation of water from gravity driven water pool, in the pipes, maintains the concrete temperature. Corrugated plate outside the insulation panel gives a better heat transfer contact with the cooling pipes. Water circulation stabilizes due to density difference between the cold supply line connecting GDWP from outside V1 volume and cooling pipes, which picks up heat transferred through insulation panel.

Passive concrete cooling system design is completed and experimental facility is being planned to validate the analytical results.



## Waste heat utilisation from MHT purification circuit to produce D.M. water

Purification flow is normally cooled in regenerative mode and later by non-regenerative coolers before passing it to lon exchange columns. Non-regenerative cooling is usually provided at the last stages of cooling which otherwise requires uneconomical number of regenerative coolers in series with associated temperature cross in heat exchanger design. In AHWR, Purification flow 2000 lpm is cooled to 100°C in regenerative mode and further temperature reduction to 42°C is achieved in non-regenerative cooler, leading to transfer 8 MWth heat load to process water as waste heat.



The purification system cooling circuit is modified to utilise the optimum quantity of waste heat of 5 MWth to produce 300 m<sup>3</sup>/day Demineralised water by providing an isolation heat exchanger, which transfers heat from 100°C to 64 °C to Low Temperature Evaporation (LTE) Desalination Plant. This design feature lowers the heat load on process water besides producng 300 m<sup>3</sup>/day D.M. water.

## Thermal Analysis for Estimation of Rate of MHT Temperature Rise

The heating of Main Heat Transport (MHT) system from 35°C to 285°C during start up from cold shutdown condition has been studied by utilising core heat from 2% to 5% Full Power (FP), after attaining criticality. Analytical estimates shows that the heating time varies from 5 hrs to 26 hrs for core heat varying from 2% to 5% FP and purification flow varying from 0 to 100%. These analytical estimates consider heat capacities of structural, fuel and water inventory, losses to surrounding moderator, top and bottom end shield, air surrounding MHT piping in V1 area.



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## Design of Coolant Channel of the Advanced Heavy Water Reactor

Coolant channel forms the heart of pressure tube type nuclear reactor. Coolant channel houses the fuel assembly along with shields and seal plugs. The coolant flows past the fuel assembly removing the nuclear heat from the fuel pins. The AHWR coolant channel has many features which facilitates onpower fuelling, direct injection of cold water from emergency core cooling system to hot fuel pins in the event of LOCA, easy replaceability of pressure tube, suitable interfaces with Main Heat Transport (MHT) system, as well as annular gap to separate the hot pressure tube ( $285^{\circ}$ C) from surrounding cold moderator ( $65^{\circ}$ C).

## **Design Description**

Coolant channel consists of cold worked Zr-2.5Nb pressure tube in the reactor core region, which is extended by stainless steel end fittings at the top and the bottom. The coolant channel is located and locked in its position by anti-torque hardware and yoke assembly attached to top and bottom end fittings respectively. Bottom end fittings are provided with feeder couplings to connect it with feeders, which are welded to Inlet Header. Lower ends of tail pipes are welded to top end fittings and top ends of tail pipes are connected to steam drum.

Fuel assembly is located inside the pressure tube and is inserted and taken out of the channel through the top end fitting. The top end fitting has suitable features to support and locate the fuel assembly along with the shield plugs in its designed location and to interface with the fuelling machine to carry out fuelling operations. After fuelling is over, the top end fitting is closed using a seal plug which butts against a seal face provided in the bore of the end fitting.



Coolant Channel Assembly

## **Operational Details**

Light water coolant from MHT System enters the coolant channels through bottom end fittings, which are connected to feeders. Coolant at 259°C flows past the fuel assembly by removing the heat generated by nuclear reaction and flows out as steam-water mixture at 285°C through the tail pipe to the steam drums where steam is separated and sent to turbine.

## Easy Replaceability of Pressure Tube

The estimated life of pressure tube in this reactor is about 30 years due to various degrading mechanisms such as irradiation creep, corrosion and hydrogen pick up. Easier replaceability of Pressure tube is taken into consideration in the design to reduce the duration of shut down of the reactor, man-rem exposure and the cost of replacement. Certain features are provided in its design for the above purpose and are described below.

#### Pressure tube

Replacement of pressure tube has been planned by removing it along with bottom end fitting, through the bore of the top end fitting. This has been achieved by keeping the bore of the top end fitting more than the maximum outside diameter of pressure tube, after considering the irradiation induced diametrical creep of 4 % and the outside diameter of the bottom end fitting. In view of this

and the rolled joint requirements, pressure tube is designed to have an out side diameter of 133 mm and thickness of 6.1 mm at the top end and 90 mm outside diameter at the bottom end.

# THICKER END CORE REGION REDUCER

## **Rolled Joint**

Pressure tube is joined to top end fitting with a rolled joint, which can be assembled and tested remotely. The rolled joint can be detached by shock heat method. Top end fitting which is not planned to be removed during pressure tube replacement, is provided with an additional set of rolled joint grooves which could be used for making the second joint.



## Feeder Coupling

Feeders are connected to the bottom end fittings by feeder couplings. This is a special compact mechanical coupling with a self-energising metal seal as sealing element which requires lower tightening load than other mechanical couplings. The effectiveness of sealing increases with pressure in this coupling. The bottom end of the bottom end fitting is provided with an in-built hub and the feeder is also ending with a suitable hub. These hubs are provided with a taper on its outer portion and the inner side of clamp is provided with a matching taper. When the clamp is tightened over the hubs, they become closer and the required amount of tightening load is applied on the sealing element. The feeder coupling has been designed and developed on the basis of performance tests.

Pressure tube





calandria tube sufficiently in advance so that the reactor can be safely shut down before any catastrophic failure occurs. The annulus is closed at the top end and open at the bottom.

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## **Composite Sleeve rolled Joint**

The tail pipe is of SS304L and is required to be welded to top end fitting. Hence, it is preferable to have top end fitting of SS304L in order to avoid a dissimilar metal welding in the nozzle region. Due to large difference in thermal expansion coefficients of SS304L and pressure tube material, a rolled joint between pressure tube and SS304L end fitting operating at high temperature is not feasible. Hence, a composite sleeve is designed consisting of outer sleeve of SS403 material which is having lower thermal expansion coefficient, shrink fitted over inner sleeve of SS304L material. At higher temperatures, outer sleeve will introduce compressive stresses on the inner sleeve, which will partially compensate for the relaxation of contact pressure, between the pressure tube and the inner sleeve.

## Annulus Leak Monitoring System

The Annulus Leak Monitoring System (ALMS) is incorporated in the design of the coolant channel to ensure 'Leak Before Break Criterion'. It is designed to detect leaks from pressure tube, its joints and from moderator system into the annulus between pressure tube and

## Development of rolled joint detachment technology

One of the major design features of AHWR is the replaceability of pressure tube during the lifetime of the reactor. In order to demonstrate this aspect, a technique for detachment of rolled joint between end fitting and pressure tube is being pursued. The technique is based on radio frequency induction heating, in which the pressure tube in the rolled joint area is rapidly heated using an induction coil and then suddenly cooled. A schematic of the technique is shown in the figure. During heating, as the pressure tube temperature starts increasing and the stress developed reaches the yield stress, there will be no further increase of the contact pressure between pressure tube and end fitting. During subsequent cooling, a gap will be generated between the end fitting and pressure tube. After two to four cycles, the joint will become free. During subsequent heating, an axial load is applied, which will contract the pressure tube radially and finally it will come out of the rolling grooves. Variation of contact pressure with temperature

in the case of 220 MWe PHWR calandria tube rolled joint, as estimated analytically is given in the Figure.

In order to estimate the various parameters involved and to optimize their values, a mock up facility is being set up. To reduce transmission loss, coaxial cable is being purchased, which will be used to transfer power from the induction-heating machine to the work coil. In order to apply axial load during detachment work, a tool has been designed, as illustrated in figure.

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## Containment Isolation Devices for Nuclear Reactors

The peripheral RCC wall of the nuclear reactor is termed as containment and is the ultimate barrier between the reactor and the atmosphere. Utilities are fed in and out of the reactor through pipe penetrations. The largest such penetrations pertain to ventilation system. The dampers provided for closing these penetrations and bottling up of the containment are thus critical equipment of the dynamic containment system. Dampers provided in such penetrations are required to close in a perfectly leak-tight manner within few seconds in case of any abnormal situation. The safety and reliability of entire containment system depends mainly on these dampers.

Most of the existing systems utilize butterfly valves for this purpose. These butterfly valves are basically designed for fluid duty at a system pressure of 5 kg/ cm<sup>2</sup> whereas in reactors the pressure of not more than 2 kg/cm<sup>2</sup> is expected even under severe accidental condition like LOCA. When these valves are used for low pressure systems handling atmospheric air the performance does not remain satisfactory. Unequal seal compression, unequal wearing

of seal, different behavior of seal material in water and air, inadequate deflection and sealing force due to low pressure, accumulation of dust on sealing surface aggravating the seal compression etc. are the problems being faced in butterfly valves.

Hence there was a need to design these isolation devices specifically for the intended application instead of just trying to adapt available valves in a unsuitable manner. The isolation devices can be either active type i.e. through external actuation by means of an electro-pneumatic circuit or passive type i.e. self-actuating by the fluctuation in system parameters viz. pressure & velocity of air in the duct during LOCA. Work is being carried out on both the fronts.

On the active damper side an electro-pneumatically operating, lightweight, positive seal, airtight damper, with rotary cum axial motion was developed in TSD. The axial motion provides perfect sealing by overcoming inherent defects of butterfly valves. This damper has compressed air to open and spring to close on active signals. Thus it is termed as active damper. Initial design utilizes multi-four bar mechanism to achieve the rotary-cumaxial motion. Efforts are being made to reduce the number of linkages. The test results are highly encouraging.

On the passive damper side simple spring loaded disc critically designed to allow a particular range of flow is being developed. The disc automatically closes when there are instantaneous increases in air velocity in the duct.

The active damper along with passive dampers in series can provide a perfect solution to the need of positive and reliable isolation of any nuclear reactors. The major works involved are conception of design, development of prototype working model, development of testing set up, fabrication and testing prototype model, test data interpretations and subsequent design modifications. Presently work is being carried out to design, fabricate & test a damper suitable for the conditions specified in AHWR reactors and also to fabricate a damper to establish the design principles of Passive Dampers.





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## **1.8 FUEL HANDLING FOR AHWR**

## Development of Fuel Handling System of AHWR

Main requirement of fuelling system is to re-fuel the reactor by gaining access to the reactor top face during refuelling. The fuel handling system mainly consists of a fuelling machine, an inclined fuel transfer machine, a temporary fuel storage bay located inside the reactor building and a fuel storage bay located outside the reactor building. Main design challenges faced in the design and development of fuel handling system are related to the handling of 10.5 metre long fuel assembly, shielding requirement and seismic qualification of 19 meters tall and 600 tonne heavy fuelling machine.

#### Fuelling Machine

The function of fuelling machine is to remove and insert the fuel assembly in three pieces i.e., shield A, shield B and fuel cluster with a reliable operation. This is done with the help of separator assembly, magazine assembly and ram assembly of the fuelling machine. Magazine assembly temporarily stores the plugs and fuel and ram assembly operates the plugs for their removal, movement or installation. Ram assembly consists of three coaxial rams and outer ram travels up to 7.6 meters for removal of fuel assembly from the



channel. Considering the shielding requirement and long and heavy structure of fuelling machine it has been supported at higher elevation to achieve fine alignment with the channel. Fuelling machine interacts with coolant channel through snout assembly making leak tight connection for refuelling. 3-D modelling and design has been completed for the machine. Presently manufacturing of prototype fuelling machine is in progress.

Trolley and carriage assembly support the heavy shielding structure and locate the machine coarsely to within  $\pm 3$  mm of the particular channel to be refuelled.

## Fuel Storage Bay

There is a small temporary fuel storage bay (TFSBx) located inside reactor building through which fuelling machine interacts for charging new fuel and receiving spent fuel. The permanent fuel storage bay is located outside the reactor building for storage and cooling.

## Inclined Fuel Transfer Machine

Inclined Fuel Transfer Machine (IFTM) is used to transfer the fuel across the reactor building. IFTM is a tall machine connecting inside and outside fuel storage bays through containment walls. A water



filled pot containing fuel, guided in an inclined ramp, is hoisted up in the tilting leg and subsequently hoisted down to unload the fuel on the other side. The concept of IFTM for fuel transfer is most suitable because of less requirement of space inside reactor building, on line transfer, small containment penetrations, ensured cooling of fuel throughout the transfer and passive containment isolation features. Passive containment isolation is enabled by water head available in the inclined ramp in case of LOCA.

#### Fuel Handling System Hardware Components

Main hardware components include snout plug, seal plug and collet joint. Snout plug is a part of fuelling machine and provides leak tight closing end for the fuelling machine when it is not connected to any channel or fuel port. The necessary sealing is accomplished by a special mechanism using radial 'O' ring. Skinner seal, a metallic face seal, ensures a leak tight connection between end fitting and the fuelling machine during refuelling. Seal plug is a critical sub assembly of the coolant channel assembly and acts as a pressure boundary to prevent the escape of MHT steam/water. Collet joint has been used for joining different components of fuel assembly. All the above components have been tested in reactor-simulated conditions and design has been proven.





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## 1.9 ANALYTICAL STUDIES AND EXPERIMENTAL VALIDATION

## Investigations on the stability characteristics of the Advanced Heavy Water Reactor

The Advanced Heavy Water Reactor adopts natural circulation for removal of fission heat during start-up, power-raising and accidental conditions in addition to the rated full power operating condition. With several parallel boiling channels having different power and resistances connected between the header and the steam drum with very long feeder and tail pipes, the reactor may experience various types of instabilities during its operation from atmospheric to rated pressure and the power raising process.

Occurrence of the thermo-hydraulic instabilities may further induce power oscillations through the void reactivity coupling. Instabilities of any form are undesirable from the viewpoint of reactor operation, control and safety. It is required to predict the stable and unstable regions of the reactor operations during the design stage so that if instability is found, methods of suppressing or procedures to avoid them can be worked out. Further to this, it is also required to generate the stability maps considering the neutronic feed back effects at various conditions, which are useful for the design and operation of the reactor. Modelling the natural circulation static and dynamic instability characteristics of the AHWR have been carried out using two computer codes TINFLO-A and TINFLO-S. These codes have been developed to predict the static instabilities such as Ledinegg type and the density-wave instabilities of the reactor respectively. Computer code TINFLO-S can simulate the interactions of several parallel boiling channels of the reactor to induce the out-of-phase instability in the reactor when operated under natural circulation conditions. To analyse the flow pattern transition instability in the reactor, the model considers the flow pattern transition criteria and flow pattern specific pressure drop models both in single and twophase regions of the horizontal and vertical sections of the reactor.

Simplified models for coupling of neutron kinetics with thermal hydraulics in the AHWR have been developed. To study the interaction between different parts of the core through neutron diffusion, a coupled multipoint kinetics model has been applied in place of simple point kinetics model for the neutron dynamics. The model considers the reactor to contain 'N' number of sub-cores which are sub-critical, isolated by reflectors and influenced each other only through leakage neutrons number of which is proportional to

the average neutron flux over each subcore. Each subcore may contain one channel or group of channels having the same power and resistances.

In the recent past, the out-of-phase instability is explained as a phenomenon in which the neutron higher modes are excited by the thermal hydraulic feed back effects. The higher modes are all sub-



critical, which could result in out-of-phase oscillations depending on the sub-criticality of the harmonic mode and the void reactivity feed back. To simulate the out-of-phase instability in the AHWR core, a mathematical model has been developed based on the modal point kinetics model. The out-of-phase instability behaviour of the AHWR considering the coupled multi-point kinetics model and modal point kinetics model has been compared.

Influence of void reactivity feedback and fuel time constant on the thermal hydraulic stability behaviour of the AHWR has been analysed. Effect of delayed neutrons on the reactor stability has also been analysed. Constant decay ratio lines, which are indications of the stability margin of the reactor, were predicted at the rated pressure



conditions of the reactor.

The results shown in figure indicate that the out-of-phase mode oscillations are more dominant as compared to the in-phase mode oscillations in the reactor because of the extra single phase friction in the down comers which stabilise the in-phase mode oscillations.

Both Type I and Type II instabilities were found to occur depending on the operating conditions such as the heat generation rate and subcooling. Type I instabilities occur at low power with initiation of boiling in the core when the quality is low. Under low



quality conditions, a slight change in quality due to any disturbance can cause a large change in void fraction and consequently in the driving head to induce oscillations. Whereas Type II instabilities occur at high power conditions when the quality is high. The two-phase frictional pressure loss may be high owing to the smaller two-phase mixture density. Having a large void fraction will increase the void propagation time delay in the two-phase region of the system. Under these conditions, any small fluctuation in flow can cause a larger fluctuation of the two-phase frictional pressure loss due to fluctuation of density and flow to induce the oscillations in the system.



Boiling Channels of the AHWR (Numbers at threshold points indicate the frequency of oscillation)

The Type I instabilities is of concern for the AHWR operation, especially during the power raising process, start-up, set-back and shut down conditions. However, since the heat generation rate is less, it may not cause occurrence of CHF in the channels. However, the reactor operation at these conditions may not be possible and to avoid them, a suitable operation procedure needs to be worked out. On the other hand, the Type II instabilities occur at much higher power, beyond the reactor trip set point. Hence, it is not of concern for the AHWR operation.

The frequency of oscillations at different operating conditions are estimated and also shown in the same figure. It is observed that the frequency of Type II instabilities is much larger than that for Type I instability. Moreover, the frequency of oscillation is very less in the AHWR (<0.08 Hz) as compared to that normally observed in vessel type BWRs (>0.5 Hz) due to very long feeder pipes in the AHWR main heat transport system. Due to the low frequency of oscillations, the Type I and Type II thermal hydraulic instabilities can get suppressed with the negative void reactivity feed back inherent with the neutronic characteristics of the core. To study the neutron field dynamics, a coupled multipoint kinetics model is used.

The codes developed are currently validated with in-house data and other commercial codes like RELAP5. As a future course of this activity, four equation drift flux model of the AHWR configuration will be used to assess its stability and compared with the homogeneous model. Stability behaviour of the AHWR considering carry-under will form part of the simulation studies.

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## Simulations of AHWR start up procedure

The instabilities during the reactor start up from low-pressure, .can be overcome by starting the reactor at higher pressures, by externally pressurizing the system. This can be achieved by introducing steam from external boiler into the steam drum. The start-up procedure for AHWR consists of stage-wise system pressurization up to 70 bar, by means of an external boiler and system heat-up with 2% reactor power. This has been simulated using a two-fluid, nonhomogenous thermo-hydraulic computer code RELAP5/MOD3.2, widely used for reactor transient analysis.

Simulations have confirmed the stability of single-phase as well as two-phase natural circulation during the reactor start-up, with stage-wise external pressurization and low power heat up. This scheme requires an external boiler with a pressure rating of 70 bar. Further start-up case studies at lower system pressure have also been investigated. The results of these simulations are discussed below.

## Variation of core flow during reactor start-up with an external stage-wise pressurization

Pressurization is carried out in such a way that cold pressurization limits for structural components is not exceeded. In this scheme

the boiling inception takes place at 70 bar and stable two-phase natural circulation is achieved.

## Start-up procedure with onetime external pressurization of system at 10 bar

At lower pressure due to the static head reduction at the steam drum flashing takes place as indicated in the figure by void fraction in steam drum and at the core exit. It is then followed by the boiling inception in the core. Flashing at 10 bar however induces smaller amplitude oscillation but can be severe at further lower pressures. After boiling inception the system is closed by closing the boiler valve and with internal steam production system is pressurized up to 70 bar. Flow is found to be stable during two-phase natural circulation.





## Reactor start-up from atmospheric pressure

Flashing instability occurs with significant oscillation amplitudes. It is then observed to reduce at increasing pressures. Low-pressure (atmospheric) natural circulation start-up experiments conducted in High-pressure Natural Circulation Loop (HPNCL) indicated that the instabilities are associated with boiling inception. The heater power in this experiment has been kept at 20 kW. The natural circulation is found to become stable at higher pressures.





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#### Flow pattern transition instability loop

Instabilities due to flow pattern transition in natural circulation reactors are reported to occur while operating near the slug flow to annular flow transition regions. The primary reason for this instability is that the frictional pressure loss in slug flow is more than that in annular flow. Therefore, while operating near the slug to annular flow transition condition, a slight increase in power leads to annular flow. Due to the lower pressure drop in annular flow, the flow rate tends to increase. The increased flow rate reduces steam production and hence the flow reverts back to the slug flow regime. Now due to the larger frictional pressure drop, flow reduces causing more steam production and the flow switches to annular regime once again. The process repeats itself and is known as flow pattern transition instability. The main objectives of the experiments were to generate data for

- bubbly flow to slug flow transition,
- slug flow to annular flow transition,
- void fraction and
- flow pattern specific pressure drop.

Experiments on steady state and stability behavior of two-phase natural circulation have been completed in four loops differing in diameter. The experiments have also generated data on the void fraction, pressure drop and flow patterns. Also, it has helped in the development and testing of the Electrical Conductance Probe (ECP) which is developed in-house, its performance has been compared with neutron radiography.

The steady state analysis shows a close agreement with the theoretical results. Void fraction assessment with the various correlations has yielded in identifying few correlations which can be used for low mass flux systems such as natural circulation systems. A photograph of the facility as it was erected in the Apsara reactor hall and few typical experimental results are illustrated.

The two phase natural circulation experiments performed in the APSARA loop (1/2" diameter) have been simulated using the twofluid, non-homogeneous computer code RELAP5/MOD3.2 to find out the mass flow rate, void fraction, test section exit quality and pressure drop. The results have been compared with the experimental measurements. The comparison of experimental results with those that are calculated by RELAP5/MOD3.2.

The facility was also used for the investigation of pressurized startup procedure in natural circulation systems. This facility also generated a few data points for critical heat flux under oscillatory natural circulation flow.



(FPTIL) at APSARA reactor hall



Reactor Technology & Engineering BARC HIGHLIGHTS





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## Development of Generalized Flow Correlation for Two-Phase Natural Circulation Loops

It is essential to know the flow rate to establish the heat transport capability of natural circulation loops. A large number of scaling parameters are available in the literature. But practically it is very difficult to simulate all the given parameters between prototype and model. Another problem associated with the existing scaling laws are that they do not give the steady state flow rate directly whereas all of the parameters generally are dependent upon the flow rate. A generalized flow correlation is needed to simulate the steady state behavior with a single non-dimensional parameter.

A set of homogeneous Navier-Stoke equations have been solved to derive the correlation for Reynolds number ( inertia force/ viscous force) in terms of modified Grashof number,  $Gr_m$  and contribution of loop geometry towards Friction number (effective loss coefficient for the entire loop),  $N_G$  This correlation is valid for both uniform as well as non-uniform diameter loops.

To account for the density variation in the buoyancy term, a new parameter  $\beta_h$ , which is the volumetric thermal expansion co-efficient and defined as  $\beta_h = \frac{1}{v_{\pi}} \begin{pmatrix} \partial v_{\partial h} \\ \partial \rho_h \end{pmatrix}_p$  as been used, where,  $v_m$  is mean specific volume and h is the enthalpy at that pressure and quality.

Experiments were conducted in three loops of inside diameter 10.21 mm, 15.74 mm and 19.86 mm respectively in a facility having the geometry as in Figure. The steam separator, the condenser and the associated piping (the portion inside the rectangular box in Figure. were the same for all the loops. In addition, experimental data were generated in a 49.3 mm inside diameter High Pressure Natural Circulation Loop (HPNCL) shown in Figure. The developed correlation was tested against the data (generated with different loop diameter: 9.6-49.3 mm) and it was seen that there is a reasonable agreement (with an error bound of  $\pm$  40%) with the proposed correlation.



Schematic of the experimental loop





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## Integral Test Loop (ITL)

ITL is a scaled facility which simulates the main heat transport system (MHTS), Emergency Core cooling System (ECCS), Isolation Condenser System (ICS) along with the associated controls of the AHWR. The scaling philosophy for the ITL facility is based on the 3-level approach in which integral, local phenomena and boundary flow (mass and energy) effects are given due importance.

The integral (or global) scaling is based on the power-to-volume scaling philosophy. Care has been taken to preserve important local phenomena, which can significantly influence the integral behaviour. Typical examples are CHF (critical heat flux) in the core simulator, steam-water separation in the steam drum etc. Important boundary flow effects simulated are that due to emergency coolant injection, feed water injection, etc. The facility is a single channel test facility simulating the full elevation, pressure and temperature of the AHWR.

## **Objectives of the Integral Test Loop**

- Generation of database for the performance evaluation of the following systems in the plant environment
- Natural circulation in the MHT loop
- Steam separation process in the steam drum
- Fluidic device in the advanced accumulator
- Gravity driven cooling system
- Isolation condenser system and
- Active shutdown cooling system
- Evolution of a start-up procedure, generation of database for plant transients and accident scenarios like LOCA


#### **Scaling Philosophy**

To check the adequacy of the scaling philosophy, simulation calculations were performed on the prototype and model for the same operating conditions and the results are graphically presented.



Comparison of predicted flow rates for AHWR and ITL



#### **Pre-Test analysis**

To gain an insight in to system thermal hydraulics a pre-test analysis has been carried out using non-homogeneous, non-equilibrium two fluid thermal hydraulic code RELAP5/MOD3.2. The steady state and transient (e.g. startup and power raising) performance of ITL



Variation of flow rate with power for different pressure



has been predicted. The predicted steady state mass flow rate was found to increase with power. With reduction in pressure, void fraction and buoyancy force increases, which increase the mass flow rate. The start up can be associated with following types of instabilities e.g. geysering, flashing and low quality density wave instability. Typical start-up transient at 2% power (40 kW) with an initial pressurization of 20 bar is shown in the Figure. As soon as the boiling starts at 25000 s the low quality density wave instability sets in. This instability dies out as the system pressure rises. The power raising from 2% to 100% (2035 kW) at 0.5 % per second shows stable operation.



#### Simulation of Steady state Behaviour

Few natural circulation experiments were carried out at various powers and pressures to simulate steady state and normal operating behaviour of the system.

#### **Simulation of Station Blackout**

During class III and class IV power supply failure the reactor is shutdown following a Secondary Feed Pump (SFP) trip and Isolation





condensers (ICs) are valved in to remove the decay heat in the reactor under natural circulation mode. To simulate such a scenario few experiments (station blackout simulation) have been performed. In simulation experiment following SFP trip the system power is tripped and set to follow a programmed decay power curve through a ramp generator. During this the system pressure rises due to decay heat and at set pressure the isolation condenser valve (ICQOV-4) opens and lets the steam condense and allows it to flow to steam drum. Thus IC acting as cold leg for the system is able to remove decay heat effectively and maintains the system under natural circulation mode. Few typical results of the experiments performed are illustrated.

The facility has been installed and commissioned in all respects and experiments that are planned are listed as follows

- Station blackout at various system pressures and powers
- Loss Of Coolant Accident (LOCA) simulation with various header break sizes at various system pressures and powers
- Simulation and evolution of the start-up procedure for various system pressures and sub cooling
- Natural circulation experiments at different powers, pres sures and subcooling
- Performance evaluation of the Active ShutDown Cooling System (ASDCS)
- Performance evaluation of the Fluidic Flow Control Device (FFCD) of advanced accumulator.

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### Investigation on Critical Power of AHWR Fuel Bundle

The Critical Heat Flux (CHF) is an important parameter, which limits the power that can be extracted from a nuclear fuel bundle. The critical power of a fuel bundle is the bundle power corresponding to the CHF condition for a given operating conditions like axial power profile, pressure, mass flux and inlet subcooling. In natural circulation based reactors like AHWR, the other constraining parameter is the thermal hydraulic instability. These two parameters are important design consideration for the safe operation of the AHWR. Investigation shows that no upper instability threshold exists for high inlet subcooling. Hence, the maximum operating power is still limited by CHF. Therefore CHF in the bundle needs to be reliably ascertained to determine the available safety margin. Such a prediction needs to be carried out so that necessary corrective action/design modification in the bundle can be worked out to achieve acceptable thermal margin.

Suitable prediction methods such as empirical correlations, CHF look-up table and mechanistic approaches for the CHF have been



Comparison between measured and predicted CHF by LUT Using HBM with modified bundle correction factor

assessed. A computer code has been developed to predict the critical power of AHWR bundle with CHF Look-Up Table (LUT). The Heat Balance Method (HBM) was adopted for the evaluation of critical power by LUT approach. The bundle correction factor used with LUT was modified to improve the prediction accuracy. The assessment of this approach was carried out based on the experimental data, on the rod bundle CHF data available in the



literature. The comparison between the experimental data and the prediction showed that around 88.3% data was predicted with an accuracy of  $\pm 20\%$  as shown in Figure.

The variation of the critical power with the subcooling is illustrated. The maximum operating channel power of 2.6 MW is also shown in this figure. Apart from this, the Janssen-Levy model has also been adopted to predict the Minimum Critical Heat Flux Ratio (MCHFR). A simplified mechanistic model has been employed in conjunction with the subchannel code for the prediction of CHF and the variation of film flow rate, deposition and entrainment rates are depicted. The Critical Power Ratio (critical power to the operating power) has been estimated to be 1.51 and 1.54 using mechanistic and LUT approaches respectively. The prediction reveals that there is an adequate thermal margin available in the bundle design of the uprated AHWR (920 MWth).



However, due to uncertainty in the prediction of the complex phenomena of CHF there is a need to substantiate the prediction of the thermal margin by experimental data in simulated geometrical and operating conditions of AHWR. In view of this, a CHF program has been formulated to generate experimental data on rod bundle CHF. A tubular test section for the CHF experimental set up has been installed and experiments are in progress. The fuel rod simulator for the CHF experiment has also been designed and is being fabricated. The Freon CHF data is planned to be generated and equivalent water data can be evaluated using the fluid to fluid modeling approach.

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# Experimental Measurement of Pressure Drop Across Various Components of AHWR Fuel Bundle

The pressure drop is an important parameter for design and analysis of many systems and components. The driving head and the mass flux being lower in the natural circulation system than those in forced circulation systems, accurate measurement of pressure drop is essential. The fuel cluster design of AHWR with 52 rods and 54 rods were evaluated analytically as well as experimentally. The various components of the fuel bundle are the fuel rods, the bottom and top tie plates and grid type spacers. In view of the complex geometry of the flow cross section through the fuel channel it becomes necessary to generate data on pressure drop across the fuel bundle and its components experimentally.

Single phase water and two phase air-water experiments were carried out in flow test facility with 52 rods cluster (with five and six spacers) to generate pressure drop data for the bundle and its various components. The total pressure drop across the bundle with five spacers was found be 3-6% less than that with six spacers.

The single phase variation of fuel friction factor i.e., the resistance offered by the wall surface to the flow, the variation of loss coefficient i.e., resistance to flow due to shape or flow geometry change across tie-plates and spacer are shown in the illustration. The experiments were also carried out for Reynolds Numbers in the range of 7900 to 79000 with 54 rods cluster with six spacers for single phase and two-phase.

A two-phase multiplier ( $\Phi_{LO}^2$ ) for 54 rods fuel cluster was calculated by taking the ratio of two phase pressure drop and single phase pressure drop across the bundle. A correlation is derived using the experimental two phase multiplier data. The present correlation was able to predict the experimental data with an error band of 20%.

Single-phase pressure drop experiments have been carried out in the 3 MW Boiling Water Loop (BWL) for 52 rods cluster. The components like shield plug, collet and rod bundle were fabricated and installed in the coolant channel test section in 3 MW BWL. Single phase pressure data have been generated and analysed. A typical result for the pressure drop across shield plug of AHWR coolant channel is presented below. It can be seen that the shield plug loss coefficients reduces exponentially with the Reynolds number.







#### AHWR fuel cluster vibration studies.

The fuel bundle vibration induced by high velocity coolant flow is an important issue that needs to be addressed for ensuring its satisfactory performance in the reactor. Normally the coolant flow has the potential to excite the bundle and the fuel elements inside it. This excitation can give rise to low amplitude vibrations that are enough to cause inter element rubbing. There are many reported incidences of fret related damages to fuel bundles due to rubbing in all type of reactors. The vibration studies on AHWR fuel cluster was taken up in an experimental facility to assess the fuel cluster/element vibration.

AHWR fuel cluster is a 4.3 meter long slender structure made up of 52 fuel rods of 11.5 mm diameter. The bottom and the top of the cluster have collets and the fuel tubes are held by spacers along its length to maintain gap between the tubes and to provide guide support to the fuel rods. The cluster is housed inside

a coolant channel through which the coolant flows from bottom to top. The coolant flow, which is driven by natural circulation, induces vibration in the fuel rods. In order to characterise the dynamic behaviour of the cluster, study has been carried out to assess the fuel rod and cluster vibration in a simulated flow test facility.

The set-up shows the optical and laser devices used for direct measurement of fuel vibration through an optical window. The effect of single and two-phase (air and water) flow on cluster vibration has been studied. The frequency band of flow excitation and the level of cluster and tube vibrations have been identified. The experimental study has led to the conclusion that AHWR cluster vibrations are below five microns and tube vibrations are insignificantly low. Due to these observed level of vibrations, the possibility of putting just five spacers instead of original six spacers is under review. Such a design change is expected to reduce the pressure drop in the channel, which is a desirable feature for the natural circulation loops in AHWR.

The illustration shows the vibration spectrum of the cluster and the frequency band of flow excitation for five and six spacers in the cluster. The vibration spectrums of five spacers and six spacers cluster show similarity in its contents. The cluster modes around 5 Hz and 11 Hz can be clearly seen in both the spectra. There are no other indications that could be attributed to significant fuel element vibrations in both the cases.

The fluid fluctuation spectra for both the cases also show similarity except for minor variations of little consequence.



**Advanced Heavy Water Reactor** 



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# Development of Indirectly Heated Fuel Rods

The safety of nuclear reactor is to be ensured not only under normal operating conditions, but also under transient and accident conditions. Loss Of Coolant Accident (LOCA) is one of the postulated accidents, the course of which is strongly dependent on thermal hydraulic characteristics of the reactor core, comprising of fuel rods. The complex nature of the phenomena occurring during accidents calls for extensive experimental investigations. Electrically heated Fuel Rod Simulators (FRSs) are extensively used to simulate nuclear heating (particularly decay heat) in out-of-pile experiments. The designs of FRS are mainly categorized into directly heated fuel rod simulators and indirectly heated fuel rod simulators. In directly heated type FRS, current is passed directly through the tube which geometrically simulates the cladding of a nuclear fuel rod. This type of FRS finds application in experiments related to steady state heat transfer in single and two-phase flow of fluids, which are nonconductors of electricity. Very high heat flux can be achieved in this type of FRS. However, for unsteady state tests, simulation of stored heat is extremely difficult in such a design. For safety related experiments, fast transients are involved, the stored energy and transfer of stored energy in fuel play an important role. Indirectly heated FRS, which simulates stored energy better, is preferred for such applications. In indirectly heated FRS, the heating element is kept inside the clad tube and the gap is filled up with the ceramic powders compacted to a certain density to achieve high thermal conductivity of the powders.

Indirectly heated Fuel Rod Simulators (FRSs) have been developed to perform out-of-pile thermo-hydraulic experiments. Two types of FRS (one with and the other without gas gap) have been fabricated and tested. The FRS with gas gap has been tested in 3 MW Boiling Water Loop (BWL) up to 35% of AHWR fuel rod rated power while the one without gas gap has been tested up to 130% of rated power.



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# Design and Development of Directly heated Fuel Rod Cluster Simulator (FRCS)

Fuel Rod Cluster Simulator is a key component of thermal hydraulic test facilities to generate data on CHF, pressure drop and subchannel mixing. In addition, they are also required in simulated integral test facilities simulating nuclear reactor systems. In view of this, a direct electrically heated 54 rod Fuel Rod Cluster Simulator (FRCS) simulating AHWR fuel bundle has been developed. In the fuel cluster electrical heating of the clad simulates nuclear heat generation. The special safety feature of AHWR, like direct in-bundle emergency coolant injection has also been simulated in this FRCS.

The FRCS requires precise fabrication of spacers and fabrication of heater tubes by joining of nickel rod and inconel tubes. A special technique has been developed to install the thermocouples and to take out lead wires penetrating the pressure boundary. Subsequent to hydro-test and thermal cycling tests, the FRCS has been installed and operated in the integral test loop at operating pressure and temperature successfully.



Fuel Rod Cluster Simulator: a) Different Components b) Installed at Integral Test Facility

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# Studies on steam drum thermal hydraulics in an Air Water Loop

In the normal operating conditions of AHWR the average core exit quality is about 17.6%, which corresponds to a void fraction of 81.36%. This causes a swelling in the steam drum i.e., an increase in the steam drum level and if the reactor is tripped suddenly the water level in the steam drum may fall below the top of the baffle plates, due to sudden collapse of voids. This may disrupt the natural circulation in the main heat transport system and can lead to an increase in the clad surface temperature of the fuel. Hence, studies are required to know the exact swelling in the steam drum. The other objectives of this loop are :



Variation of void fraction with air flow rate

- measurement of bubble and droplet distribution and investigation of (a) carry-over phenomena i.e., entrainment of water droplets in the air going out of the pool and (b)
  - carry under phenomena i.e., entrainment of air bubbles along with the water flowing to the down comer

Experiments are carried out in an Air Water Loop (AWL) to measure the void fraction (swell) in a pool of water by the swell level. Void fraction was estimated from the measured swell level and was compared with the calculated values from Kataoka Ishii correlation and good agreement is observed between these two.

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# Natural Circulation Flow Distribution Studies Set-up

This experimental set-up has been established to study flow distribution in a natural circulation driven system like Main Heat Transport (MHT) system of the Advanced Heavy Water Reactor (AHWR). This set-up consists of ten parallel flow channels, each provided with individually controlled heat sources. Largely transparent construction of this set-up facilitates visualization of phenomenon like thermal hydraulic Instability, flow pattern transition etc., at atmospheric pressure and temperature up to 100 °C.

Although this set-up is built to study sensitivity to the process parameters of flow sharing between parallel coolant channels, modular construction makes it a versatile tool useful for a wide variety of other studies in the area like reactor start up procedures, possible channel flow reversal, effectiveness of flow pattern control devices and effect of change in system configuration.



Natural Circulation Flow Distribution Studies Set-up

For providing insight in to the flow distribution at different power levels flow, temperatures and power of individual channels are monitored. The channel flows are measured using a specially designed low-flow, low-loss venturi meters. The process parameters are displayed on-line in the mimic flow diagram by a PC-based data acquisition system.



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# Investigations on parallel channel instability and simulation of void reactivity feedback in the Parallel Channel Loop

Parallel channel instability is the controlling instability (i.e. having the least stable region) and is an important design consideration for the AHWR. Apart from this, two-phase natural circulation flow exhibits phenomena like flashing and geysering which occur predominantly at low pressure and hence is a concern during the start-up. To study these phenomena an experimental facility – Parallel Channel Loop (PCL) is being set-up. This facility will also simulate the void reactivity feedback effects of the AHWR. This is very important in the context of a boiling-water-reactor where power oscillation due to void reactivity feedback may either reinforce or suppress the flow instability. The main objectives of the Parallel Channel Loop are:

- steady state natural circulation behaviour with equally as well as unequally heated channels,
- generation of out-of-phase (regional) and in-phase (global) instability maps,
- void fraction measurement using conductance probe,
- simulation of neutronic feed back on thermal hydraulic oscillations,



- study of carryover and carryunder: demonstration using transparent sections in riser and downcomer,
- study of low-pressure two-phase instabilities like flashing and geysering,
- Effect of nano-particles on natural circulation and stability.

The pre-test analysis are carried out using RELAP5/Mod3.2 and the illustration represents a typical instability obtained using this code.

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# Experimental studies on Scaled Model of Advanced Accumulator with Fluidic Flow Control Device

A 1:5 scaled model of advanced accumulator with fluid flow control device has been designed, fabricated and installed at BARC. The fluid flow control device is a simple passive device, which allows initially large amount of flow from the accumulator and later reduces the flow automatically due to formation of vortex.

When the water level in the accumulators is above the standpipe the water enters the fluid flow control device through both the inlets i.e., stand pipe and side connection, and since the flow is smooth, a large flow of water is discharged from the accumulators. When the water level in the accumulators drops below the top of the stand pipe, the water enters the chamber through the side connection only, which is tangential to the chamber. This increases the flow resistance, due to formation of vortex, resulting in reduction of flow.

Three devices of different dimensions have been fabricated to evaluate their performance. The objectives of the scaled model are performance evaluation, verification of low characteristics, studies on water level transient in stand pipe at transition and expansion behaviour of nitrogen, Two sets of experiments have been completed for initial pressure up to 1 MPa. The graph shows variation of accumulator discharge flow with time.





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#### Studies on Thermal Stratification

Thermal stratification denotes the formation of horizontal layers of fluid of varying temperature with the warmer layers of fluid placed above the cooler ones. Thermal stratification is encountered in large pools of water increasingly being used as heat sink in new generation of advanced reactors like GDWP of AHWR in which Isolation Condensers are immersed.

Stratification influences heat transfer to pool to a great extent and heat storage capacity of the pool in the form of sensible heat is significantly reduced. It can also threaten the structural integrity if the pool is made of concrete. Hence thermal stratification is not desirable and we have to minimize its effect.

Experiments were carried out in a rectangular glass tank (440 x 100 x 300 mm) with an immersed strip heater for visualization of thermal stratification phenomena. Theoretical analysis was also carried out for this case using CFD (Computational Fluid Dynamics) codes TRIO\_U and PHOENICS. For further investigation of this phenomenon, a case study has been carried out with side heated cavity containing water. This 2-D problem is solved to obtain the velocity and temperature profiles using CFD codes TRIO\_U and

PHOENICS. Simulation of whole isolation condensers will be carried out to study the thermal stratification phenomenon in GDWP pool. Based on these simulations, the configuration of isolation condensers will be proposed such that the effect of thermal stratification is minimized.



Measured and predicted temperatures with 3125 W/m<sup>2</sup> heat flux for all nine thermocouples after 1800 s







Velocity vectors after 15000 s for constant heat flux of 500  $W/m^2$ 

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# Studies on Passive Containment Cooling System (PCCS) of AHWR

Containment is a key component of a reactor system, since it is the last barrier designed to prevent large radioactivity release to the environment under accidental conditions. The Passive Containment Cooling System (PCCS) removes the energy released into the containment during a postulated Loss Of Coolant Accident (LOCA). Two alternative designs for PCCS are under consideration. One of the alternatives being considered is a system in which the containment steam condenses inside the vertical tubes immersed in a pool of water. In the second alternative, the containment steam condenses on the Passive External Condenser (PEC) which comprises of cooling tubes connected to a water pool above it via headers as shown in Figure. The containment steam condenses on the outer surfaces of the tubes and water from the pool circulates through



these tubes by natural circulation. An important aspect of the working of PCCS is the potential degradation of the performance due to the presence of noncondensable gas in the vapor.

# First design concept (steam condensation inside vertical tube)

Experiments have been carried out for steam condensation inside vertical tube in presence of noncondensable gas. Figure shows the comparison of the predicted local heat transfer coefficients inside a vertical tube with experimental data.



# Second design concept (steam condensation outside inclined tube)

Design of the Passive External Condenser (PEC) for PCCS has been carried out. A computer code CONISA (<u>CON</u>densation In Steam-<u>Air</u> mixture) has been developed to estimate the heat transfer coefficient, for steady-state free convective film condensation in presence of noncondensable gas where the steam condenses on the cooling surface. The model has been validated against experimental data available in literature. The results have also been compared with the correlations developed by Uchida and Dehbi. The variation of heat transfer coefficient with change in air (noncondensable gas) mass fraction in the steam-air mixture is illustrated. To validate this model, an experimental setup has been fabricated, installed and commissioned.





In severe accidents, other noncondensable gases like hydrogen can get mixed with the steam-air mixture making the treatment of multicomponent noncondensable gases having different mass diffusion coefficient important. The computer code CONISA has been modified to account for the multicomponent noncondensable gases mixed with vapour. A computer code CONFIN has also been developed for the passive external condenser to account for finned tubes.



Passive external condenser experimental set-up

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# Measurement of Reactor Power and Flow Using <sup>16</sup>N gamma signals

The power and flow measurement technique using <sup>16</sup>N signals is an on-line and non-intrusive method, independent of the flow regimes. Hence this measurement technique is of importance to the AHWR. The thermal power & flow measurement experiments have been carried out in Dhruva reactor. The results are given in the illustrations.

<sup>16</sup>N is produced in the core by fast neutron activation of <sup>16</sup>O of the coolant and decays with a half-life of 7.35 s emitting 7 MeV gamma

rays. The <sup>16</sup>N activity i.e., the radionuclide production and the thermal power are directly proportional to the neutron flux and hence <sup>16</sup>N activity is used for on line power measurement. Coolant flow measurement is obtained from the transit time measurement, i.e., by cross correlating <sup>16</sup>N noise signals from two detectors, placed along the flow path of the coolant.



Typical <sup>16</sup>N noise signals from a pair of detectors installed near the pipeline of Coolant loop # 3 of Dhruva at 40 MWt.

Experiments have been carried out in the coolant loop-3 of the Dhruva reactor, by placing two Nal scintillation detectors (with out any shielding), 90 cm apart, below the coolant pipe line. The detectors were located 2 cm away from the surface of the coolant pipeline, positioned by mild steel platforms. The signals were captured and analysed in the FFT analyzer from the reactor hall. The power and the flow measurements have been carried out at different power levels from start up to 40 MWt. The dc output signals were found to be proportional to the power level. The measured transit time was 156 ms and the corresponding calculated velocity was  $5.76\pm0.17$  m/s and the corresponding flow rate worked out to be  $2.34\pm0.07x10^4$  l/min, which agreed well with the actual flow rate of  $2.18x10^4$  l/min, indicated by the venturi flow meter. The experiments have been repeated with <sup>6</sup>Li glass scintillation detector and similar results were observed, (BARC Report/2002/l/023).



Linear fit of cross power spectral density-phase vs. frequency data obtained from N-16 noise signals at a reactor power of 40 MWt.

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#### Two phase flow measurement

The important addressed problems in thermal hydraulics of the Advanced Heavy Water Reactor (AHWR) are channel thermal power measurement and detection of critical heat flux (CHF). The process in Main Heat Transport (MHT) System of AHWR is partly single phase and balance is in two-phase. Direct measurement of twophase parameters such as void fraction and its distribution, twophase flow and finally thermal power output of the reactor is essential and a complex task. Research and development activities in these essential and difficult measurement fields have been taken up. Following are the instruments and measuring systems under development.

- Capacitance type on-line void fraction measuring system
- Void Fraction Measurement by Conductivity Probe
- Two Phase Mass Flux Measurement by Pitot Tubes Assembly and Gamma Densitometer
- Rotating Electric Field Admittance Probe

#### **Development of Two-phase flow Instrumentation**

Thermal power produced by fuel can be measured by measuring the void fraction in tail pipe and other parameters of interest along with suitable correlation. In order to fulfill this requirement, development of capacitance type void fraction sensor has been taken up as no suitable instrument as available for in-situ measurement.

Variation in void fraction, within the volume enclosed by two parallel plates of an electrical capacitor, leads to variations in the value of the capacitance accordingly. This phenomenon has been used for in-situ measurement of steam/water fraction in a metallic pipe, carrying high pressure and high temperature fluid medium. Capacitor elements are installed with enough electrical insulation such that the volume between the parallel plates only contributes to impedance under measurement.

In order to verify the principle of operation and performance of the sensor, an experimental setup was designed, fabricated and installed in the Heat Transfer Laboratory.





Design of industrial grade instrument for measurement of void fraction inside the metallic pipes, has been done as a spool piece of the straight piping with flanged end connections with high pressure and high temperature rating as per ANSI code. Ceramic pipe sheath is provided inside the metallic spool piece and four parallel plates of SS316 are installed on inner surface of the ceramic pipe, thereby ensuring the electrical insulation and with out any disturbances to the fluid flowing in the pipe. Hardware components for assembling the instrument are under advanced stage of procurement.

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Two sets of sensors with angular aperture of 30° and 150°, 68 mm ID & 68 mm length have been installed. Output signal range for 30° sensor plates was found to be 2.98 pF to 66.41 pF and signal range for 150° sensor plates was found to be 4.40 pF to 110.02 pF with 100% air to 100% water. A maximum percentage variation in normalized output signal because of constant air bubble flow of 5 lpm at various (X-Y) positions was found to be 1.5%.



#### Void Fraction Measurement by Conductivity Probe

The knowledge of void fraction (fraction by volume of gas phase to a total given volume) forms an important part of two-phase analysis. In nuclear reactors, pipelines and other industrial operations, accidental rupture of a pipe produces a two-phase flow of gas and liquid. In nuclear reactor design, measurement of various twophase flow parameters is important to verify analytical procedures for predicting reactor behavior in a loss of coolant accident. Crosssectionally averaged mixture density (or void fraction) and mass flux are the most difficult parameters to measure.

Most of the void fraction measuring techniques which are based on the effects of nuclear reactions, such as gamma attenuation, beta attenuation, X-ray attenuation, neutron diffusion, are not applicable inside the reactor cores where intense fields of all these nuclear radiation is predominant.

Among the non-nuclear methods of void fraction measurement, one of the most important methods is the electrical impedance method. The electrical impedance of a two-phase flow depends on the concentration and distribution of the phases.

Depending on the system, the impedance will be governed by conductance or capacitance. Based on conductance principle to measure local void fraction the Single point conductivity probe and Five point conductivity probes were developed.

#### Principle of Measurement

- The principle of measurement is based on the difference in conductivity between liquid and gaseous phase.
- When probe tip in contact with liquid phase, the circuit between two electrodes is closed, whereas the circuit is opened as soon as probe tip touches the bubble.
- The time averaged local void fraction is given by
  - α Σ i=1,2,3.....N

Where N is total number of samples and (i) is the number of samples indicating presence of gas at the probe tip.

#### Construction of single point Conductivity Probe

This type of probe is constructed by using S.S wire of 1 mm diameter and insulated by using teflon sleeve as shown in Figure, for measurement of bubble time, one end of the wire is shaped conically and kept un-insulated and the other end of the wire is connected with supply. The metal pipe line through which two phase flow is flowing acts like a second electrode.



# Five point Conductivity Probe with Teflon sleeve as insulation

This type of probe is constructed by using S.S wire of 0.5 mm diameter and insulated by using Teflon sleeve. The probe has five pointed electrodes which face into the flow. The tip of wires form the bubble detecting electrodes. One electrode is at the center of the pipe and two pairs are opposite each other at radii of 0.707 and 0.913 times the inside radius of the pipe. The cross-section

of the pipe can be regarded as consisting of a circle and two annuli all of equal area, with one electrode monitoring the circle and two electrodes monitoring each of the annuli.



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# Two Phase Mass Flux Measurement by Pitot Tubes Assembly and Gamma Densitometer

Mass velocity and void fraction measurements are required in transient two-phase steam – water flow experiments related to thermal hydraulic studies on reactor safety. Thus experimental measurements play a key role in providing information for design, analysis and predicting system behaviour. Hence, design and development of simple, rugged and inexpensive pitot tubes based flow sensor for cross section averaged mass velocity measurement and measurement of chordal void fraction and average mixture density by gamma densitometer has been carried out. The design criteria followed are: (a) Local measurement near the centre region of the horizontal chord gives good result to predict the chordal mass flux in most flow regimes due to the uniformity of the local mass fluxes along the chord. (b) The cross section averaged mass flux calculations using pitot tubes along the chord were in good agreement if horizontal mixture density was used in place of local mixture density and (c) it is relatively easy to measure chordal average mixture densities by gamma ray attenuation.

#### Sensor Design

The present design consists of five pitot tubes located across the pipe section and positioned in a vertical line, one at the centre of



the pipe, one each at 0.286 diameter and 0.429 diameter above and below the centre of the pipe to ensure that all the flow regimes are covered. Five differential pressure transmitters measure the velocity heads for the pitot tubes. The pitot tubes assembly is installed in the two-phase horizontal section of High Pressure Natural Circulation Loop as shown in figure for sensor validation. The twophase velocity profiles obtained across vertical chord are shown in the graph. The gamma densitometer consists of Cs-137 source with principal photon energy of 0.662 MeV, NaI(TI) scintillation detectors, single channel analysers with associated electronics for chordal average void fraction measurements. The present device is developed for simple operation and data interpretation. The Cs-137 gamma source is ordered through BRIT and commissioning of gamma densitometer is in progress.

This pitot tubes-based sensor is rugged; simple, reliable and easy to operate. It is a sturdy sensor for adverse high temperature and high-pressure steam water applications. The sensor introduces only



very little disturbance in flow path, which is very important for two phase natural circulation studies. Flow regime identification is possible using chordal void fluctuations.

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### Development of Rotating Electric Field Admittance Probe

Electrical impedance techniques have proven attractive for many applications because of their generally fast response to void variations, high signal output and simplicity of operation. The important design criteria are the sensor shall not introduce disturbance to the flow, it shall not cause pressure drop in the system and the measurements data shall predict various flow regimes in two-phase mixtures. The present novel electrical probe, which is being developed to measure void fraction in a pipe of circular cross section, meets these requisites. In the present design the electrodes are mounted coaxially inside and form part of the pipe wall to eliminate the disturbance to the flow. Further the electric field, which is perpendicular to the flow, is rotated electronically to distribute it throughout the sensor volume.

#### Sensor Design

The sensor consists of six stainless steel electrodes separated by ceramic insulators. These electrodes and insulators form the inside perimeter of the sensor flow area. The three electrode pairs are excited such that the signal for each pair is 120 degrees out of phase with the other. A rotating electric field is thus generated within the sensor volume. Three identical admittance measurement circuits are connected to the electrode pairs and the absolute values of the signals from these circuits are summed. The resulting signal



is proportional to the admittance between the sensor electrodes. A reference sensor, which produces a signal proportional to the admittance of the single-phase liquid, is used to compensate for the changes in the admittance of the liquid due to variations in the temperature and concentration of impurities. The relative admittance of these two sensors gives the void fraction of the two-phase mixture. The sensor details are given in the figure. The probe electronics measures magnitude and phase of the admittance. By measuring relative variations in conductive and capacitive components and fluid impedance particularly the phase angle, the flow distributions can be identified. The phase velocities can be predicted from the transit time signals obtained from the pair of probes by random signal analysis methods.

This technique is simple and relative simplicity of operation/ data interpretation makes it feasible for real time measurements. Also the probability density functions of relative admittance fluctuations can provide flow pattern discrimination. Since it is a volume average technique it will give better results and because of faster response, can be used for transient studies.

The fabrication of low temperature sensor for air water calibration is completed and fabrications of signal conditioning modules are in progress.

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# Super Critical Water Natural Circulation Loop (SCW-NCL)

Super critical fluids have the advantage that there is no phase change above the critical point (CP) eliminating the occurrence of the critical heat flux phenomenon. The large variation in thermal expansion coefficient near the CP can be exploited for designing natural circulation-based Super Critical Water Reactors (SCWRs). From the viewpoint of design, it is essential to identify the operating parameters near the CP such that high circulation rate and hence high heat transfer rates are achievable during natural circulation. In addition, it is desirable to operate such loops in a stable condition, which requires identification of the stable and unstable zones by a stability analysis. Experimental determination of stable and unstable zone in loops of different diameters is planned. A computer code has been developed to carry out steady state and stability analysis of a super critical water Natural Circulation Loop (NCL). Using the computer code, steady state analysis has been carried out to obtain the parametric effects on the natural circulation



Schematic of (SCW-NCL)



flow rate in a supercritical loop. The loop has been designed for 300 bar pressure and  $400^{\circ}$ C temperature.

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# 1.10 EVALUATION OF DESIGN BY INPRO METHODOLOGY

BARC participated in the IAEA initiated International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) which involved evolving a methodology for evaluation of innovative nuclear reactors and fuel cycles. The AHWR has been accepted as a case study to test the adequacy of INPRO methodology, to suggest any changes, if needed and to evaluate AHWR using INPRO methodology. The methodology consists of application of basic principles, user requirements and criteria in six areas of economics, environment, & sustainability, safety of nuclear installations including fuel cycle facility, waste management, proliferation resistance and cross cutting issues.

The basic principles, user requirements and the criteria in all the above-mentioned six areas were reviewed and modifications were

suggested. The evaluation of INPRO methodology for the AHWR comprised of:

- identifying the salient design features of the reactor,
- listing the attributes affecting each of the design features and
- evaluating effect of INPRO parameters on each of the attributes.

This procedure was well appreciated by the agency. The case study has been completed and a final report has been sent to IAEA containing the description of the plant, near and long-term fuel cycle of AHWR, review of the Basic Principles, User Requirements and the Criteria, salient design features of AHWR and case study evaluation.

The following table is part of the evaluation of INPRO methodology for AHWR in the area of fuel cycle. The different design features of AHWR related to fuel cycle are listed and their effects on different areas of INPRO parameters are indicated.

Sr. No.	Main design Features	Areas of Case Study								
		Economics	Environment & Sustainability	Safety	Waste Management	Proliferatio n Resistance	Cross- Cutting Issues			
1	Thorium fuel	V	$\checkmark$		$\checkmark$	$\checkmark$				
2	Closed fuel cycle	~	$\checkmark$		$\checkmark$	$\checkmark$				
3	High burnup fuel	~	$\checkmark$							
4	High conversion Fuel	~	$\checkmark$							
5	Partitioning & Transmutation	$\checkmark$	$\checkmark$		$\checkmark$					
6	On power refuelling		$\checkmark$			$\checkmark$				
7	Critical facility	V	$\checkmark$							
8	Fuel cycle facility co-location	$\checkmark$				$\checkmark$				
9	Reprocessed fuel composition					$\checkmark$				

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# 2. R & D FOR PHWR

#### INTRODUCTION

The present nuclear power programme in India is based mainly on a series of 220 MWe pressurised heavy water reactors. These reactors use pressure tube technology with natural uranium as fuel and heavy water as moderator and coolant. Currently, there are fifteen PHWRs in operation including 540 MWe TAPS 4, the first of its kind in India. This chapter highlights the current R&D activities being carried out in the field of physics design & safety assessment, fuel management studies, on-line flux mapping systems, fuelling machine testing, design of control and shut off rod mechanisms, experimental studies and severe accident programmes.

# 2.1 FUEL MANAGEMENT STUDIES TO OPTIMIZE FUEL UTILIZATION IN INDIAN PHWRs

The optimised fuel management makes the best use of the fuel and therefore reduces the unit energy cost. Hence different possible alternatives have been studied to optimise the fuel utilisation in Indian PHWRs. A few of them have been successfully implemented in the power reactors. Fuel management code TRIVENI is being used in all the power stations. The equilibrium core optimisation studies have been performed by TAQUIL and the snapshot core calculations were performed by TRIVENI.

The following burnup optimisation studies have been carried out to improve the fuel utilisation in Indian PHWRs.

- Full power operation with corner adjusters fully withdrawn from the core.
- Reactor operation at reduced power with peaked flux distribution.
- Regular use of Depleted Uranium (DU) fuel at full power operation.

- Use of depleted uranium in the initial fuel loading of the new reactors and the reactors started after en-mass coolant channel replacement activity.
- Use of depleted uranium and deeply depleted uranium (about 0.3% w/w) fuel in the initial fuel loading of TAPS-4 reactor (540 MWe).

Lattice calculations for MOX-7,MOX-97, MOX-888 clusters and equilibrium core burnup optimisation studies have been carried out for the PHWRs. Feasibility studies have been carried out for the use of thorium MOX cluster such as MOX-Th24, MOX-Th20 in PHWRs. The core calculations with regard to the utilisation of MOX-7 cluster as well as MOX-888 cluster along with natural uranium at large scale in PHWRs have also been carried out.

The operation in the peaked flux distribution scheme has been implemented at all the PHWRs and more than 30% saving in natural uranium has been achieved. Usage of depleted uranium in PHWRs has been implemented at RAPP-3&4 & KGS-1&2 and 30% saving in natural uranium has been realised. Test irradiation of 50 bundles of MOX-7 is under progress at KAPS-1. Existing PHWRs can use 40% of MOX-7 or MOX-888 along with natural uranium.





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## 2.2 REACTOR PHYSICS DESIGN AND SAFETY ASSESSMENT OF PHWRS

#### Criticality of 540 MWe PHWR

Safety of the physics design and commissioning procedures have been evaluated for TAPS 3 & 4 units, of the 540 MWe PHWR design. Significant changes in design of control and safety systems have been incorporated for the first time in these reactors, due to their larger size, compared to the standard 220 MWe PHWRs. Technical specifications for operation has been evolved for first approach to criticality and Phase – B commissioning of TAPS Unit # 4, the first reactor of this kind in the country.

#### Maximisation of discharge burn-up of PHWR fuels

The design support and safety analysis for the changes in the fuelling strategies for smooth transition from two burnup zone pattern (Flat flux) to one zone pattern (Peak flux) for maximising the discharge burnup have been carried out. The evaluation of trial irradiation of Pu MOX bundles in KAPS-1 reactor has been completed. Presently, the design discharge burnup of PHWR bundle is 15,000 MWd/t U and few bundles were irradiated beyond this limit up to 20000 MWd/t U, to obtain physics data such as the fall in the power production, changes in the reactivity after refueling and also metallurgical details through post irradiation examination (PIE). The extended irradiation of natural uranium bundles in two specified channels have been analysed.

# Evolution of criticality procedures for re-tubed reactors and upgradation of Computer code "Triveni"

Criticality procedures have been evolved for startup of reactors after long shutdowns like en-mass coolant channel replacement and these procedures have been approved by safety and regulatory bodies (SARCOP and AERB) and have been implemented in the approach to criticality of MAPS, Unit-2. Modification of the fuel management code TRIVENI as and when required and inclusion of special features such as the supply of latest nuclear data files for better estimations of reactivity coefficients is one of the important ongoing activities.

# Performance evaluation of Self Powered Neutron Detectors

Data collection and analysis of the linear response with power changes and dynamic response of SPNDs(Self Powered Neutron Detectors) following transient change of power in KGS – 1 and RAPS-3 was another valuable exercise taken up to validate and gain confidence in the usage of SPNDs for regulation and protection in 540 MWe units. Usage of SPNDs for incore monitoring of neutron flux during power operation would be a necessity in large size reactors and hence has a lot of relevance for the AHWR as well as 700 MWe PHWR.

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# 2.3 ON-LINE FLUX MAPPING SYSTEM (OFMS) DEVELOPMENT FOR TAPS-3&4

Large power reactor systems, such as 540 MW(e) PHWRs, are known to sustain power tilts following minor local reactivity perturbations. If these tilts are not controlled, they can lead to unacceptable power distributions. Therefore, it is essential to have in-core neutron flux measurements and also to process them to get knowledge of the operating state of the power reactor with the help of the On-Line Flux Mapping System. This system is a virtually continuous power regulating scheme to maintain the core power distribution closer to the design intent.

Software for obtaining the power distribution from the in-core neutron flux measurements has been developed.

#### The physical OFMS system

#### Flux Mapping (FM) detectors

A total number of 102 Vanadium type in-core Self Powered Neutron Detectors (SPNDs) are provided in 26 vertical assemblies.

These are called the Flux Mapping (FM) detectors. They are one pitch (28.6 cms) in length and made to measure the point thermal flux at their locations. The electrical signal generated in the FM detectors is due to the b decay of V. The relevant reaction that takes place is as follows.



The response of FM detector lags behind the flux by about 5 minutes due to  $\beta\,$  decay of  $\,\,V^{*52.}$ 

#### Flux Mapping (FM) Software

In OFMS software, the instantaneous neutron flux is expanded as combination of pre-determined flux shapes whose combining coefficients are determined from the on online flux measurements by employing the Least Square method. The LS procedure reduces the flux mapping software in just two matrix multiplication. The first multiplication gives the instantaneous values of the combining coefficients for modes corresponding to the FM detector readings and the other multiplication gives the flux map from the combining coefficients. Flux map thus generated is used for calibrating the Cobalt SPNDs, used for controlling the reactor.

#### Validation of OFMS

Detailed programs for validation of OFMS software includes the following;

#### Static configuration validation

The flux map generated is tested against large number of theoretical simulation of possible reactor operating states. In this validation, the detector readings are estimated from the flux shape itself with random errors sampled from Gaussian distribution superimposed on detector readings. Such detector readings are fed to OFMS software to get the flux which is then compared with the actual simulation. In this simulation no dynamic feed-backs such as xenon, temperature etc. are considered. Gross global feed-backs (as determined by the crosssections) are considered. This validation is used for determining the size of basis set for expansion.

#### Dynamic configuration validation

Under this scheme, the OFMS software is extensively tested in dynamic close loop environment. A space-time kinetics code with dynamic RRS feed-back, xenon feed-back and temperature feedback is developed for this purpose. A detector model for both vanadium and cobalt SPNDs is also included to give realistic estimate of detector readings. This validation gave the insight to the adequacy of the FM system.

#### Experimental validation

As part of validating the OFMS software, flux mapping experiments have been carried out by simultaneous irradiation of copper wires in five pre-selected in-core lattice positions of CIRUS at low power (~230 kW). Some salient features of OFMS software were validated during the course of these experiments.

#### The present design of OFMS configuration is arrived at from the

above validation. There will be a total number of 29 flux shapes (Fundamental + 18 lambda modes + 8 shapes of adjuster bank withdrawal + 2 shapes for control rod insertion) in the expansion set. The reactor regulating system will supply the adjuster rod and control rod bank status to OFMS. OFMS will then select required number of flux shapes from these 29 shapes depending upon the devices' status. OFMS has also built in algorithm to consider the effect of failed detectors.

The errors in FM are sensitive to the difference between the fundamental mode in which the reactor is operating and the actual mode set used in OFMS software. Therefore the complete set of 29 flux shapes will be changed depending upon the core irradiation status (at 0 FPDs, 40 FPDs, 100 FPDs ,400 FPDs and the equilibrium core). This is done to minimize the prediction errors in flux map and to get better knowledge of the operating state of the reactor.

The Hardware implementation and testing of OFMS was carried out at ECIL using generated detector signals from the dynamic core simulation code. The onsite commissioning of the OFMS is completed at TAPP 4 site, Tarapur. The OFMS is loaded at site with initial core (0 FPDs) mode set. The in situ testing and fine tuning the response of OFMS will be undertaken as and when the full power operation of the reactor commences.

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#### 2.4 FUEL ANALYSIS

#### PHWR fuel

In the context of the round-robin exercise carried out in BARC, analysis for the PHWR fuel bundles irradiated in KAPS has been carried out using the modified GAPCON code. The peak center line temperature evaluated was about 1600°C (for 51 kW/m linear heat rating) and cumulative fission gas release of about 3% has been calculated for a fuel burnup of 15 GWd/t. The analysis results have been compared with the post irradiation examination results.

Parameter		Case Stu	ldy	PHWR design power envelope outer pin			
	Outer Ring	Middle Ring	Central pin	100 %	110 %	115 %	
Peak pellet temp (°C)	1528	1169	1070	1592	1781	1870	
F.G.R (%)	2.34	.049	.019	1.69	4.07	5.73	
FGR (c.c)	8.13	0.13	.046	5.42	13.0	18.22	
Gas press. (Bar)	8.34	2.62	2.55	6.44	11.90	15.73	

#### MOX fuel

A 19-rod MOX bundle has been designed for loading in PHWRs and 50 of these bundles have been loaded in the KAPS reactor. MOX fuel for TAPS has been designed and twelve bundles have been successfully irradiated.

An analysis has been carried out using ANSYS to evaluate the fuel temperatures due to hot spot formed by Pu-agglomeration, for various Pu particle sizes at different heat generation rates for TAPS-BWR and PHWR (220MW<sub>e</sub>) MOX fuel. The meshed finite



Meshed finite element model for pellet with the particle

element model for pellet with the particle is shown in Figure. The fabrication specification for the maximum Pu-rich particle size is in the range of 200-400 microns. The analysis shows that the agglomerates can cause spikes in fuel temperatures of about 300-500°C depending upon the size and concentration of Pu in the agglomerate. Variation of particle maximum temperature for various heat generation ratios and for different particle sizes for TAPS-BWR fuel are illustrated. The presence of the Pu-rich particle on the surface of pellet will result in clad temperature rise of only about 100-200°C.





An analysis has been carried out for MOX fuel pins irradiated at PWL CIRUS, as part of round robin exercise, using modified GAPCON code. The peak centreline temperature evaluated was in the range of 1300°C (for 42kW/m linear heat rating) and cumulative fission gas release of about 2.0% for fuel burnup of 16 GWd/t.

# 2.5 TESTING OF FUELLING MACHINE OF 540 MWe PHWR

The Fuelling Machine of 540 MWe PHWR Reactor is an important intricate equipment. For satisfactory performance of the assembled Fuelling Machine, Performance & acceptance testing of Fuelling Machine is carried out under the simulated reactor conditions.

#### Fuelling Machine Test Facility (FMTF)

Fuelling Machine Test Facility has been set up and fuelling machine head is installed on the test carriage and the oil hydraulic water hydraulic systems have been commissioned. The fuelling machine head has been calibrated & commissioned. Necessary rectification and fine-tuning of fuelling machine head have been incorporated by suitable modification /changes of components.



The following tests were done in reactor-simulated condition:

- Performance testing:- 6 times bundles loading & 6 times bundles receiving.
- Acceptance testing : 10 times bundles loading & 10 times bundles receiving.
- Cycling of Ram extension:- 10 times loading and 10 times receiving.

The fuelling machine head has been qualified with successful completion of above tests. Full testing has been completed in very short period of 5 months.

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# 2.6 TESTING OF RAM ASSEMBLY OF 540 MWe PHWR

For satisfactory performance of the assembled fuelling machine, elaborate testing of various sub-assemblies of fuelling machine is carried out separately. Later, the assembled fuelling machine has been again completely tested before delivery to the site.

RAM assembly is one of the important subassembly of the fuelling machine and it is the first of its kind, to be used in 540 MWe PHWRs. It comprises of three telescopic rams namely B Ram, C Ram & Latch Ram. All these rams are operated through oil hydraulic motors. B Ram & C Ram are provided with water hydraulic back up. These rams are used for refuelling operations & various plug operations.

Ram assembly and the Oil & water Hydraulic system has been installed and commissioned. The calibration & commissioning of the Ram assembly in the Ram assembly test facility has been completed.



Ram assembly test facility(RATF)

Necessary rectification and fine tuning of Ram assembly have been incorporated by suitable modification /changes of components. The B Ram Cycling equivalent to 300 channel refueling operation & Plug operations equivalent to 100 channel refueling operation have been successfully carried out to qualify the Ram assembly.

# 2.7 DESIGN AND DEVELOPMENT OF ADJUSTOR ROD, CONTROL ROD AND SHUT-OFF ROD DRIVE MECHANISM FOR TAPS 3 & 4

This development work has been taken up under an MoU signed between BARC and NPCIL. The control mechanisms for TAPS units 3&4 are designed with a number of advanced features like modular construction giving ease of maintenance, 90% free fall for shut-off rod/control rods giving high reliability and consistent rod drop performance, on-line test facility for shut-off rods to ensure rod availability on demand while reactor is under operation and partial release & re-arresting of control rods for reactor step back function.

Design, prototyping, testing & qualification on full-scale test set-up have been completed. Manufacture, assembly and testing of drive units for reactor use have been completed. They have been installed and commissioned at TAPS-4.



Full-scale test set-up at BARC



Typical rod drop characteristics for shut-off rod for TAPS, units 3 & 4



Reactivity control mechanisms on deck plate of TAPS unit 4

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# 2.8 FISBE-EXPERIMENTAL STUDIES AND AUGMENTATION OF FACILITY FOR ACCIDENT SCENARIOS AND OPERATIONAL TRANSIENTS

FISBE is an integral test facility to simulate PHWR Primary Heat Transport System including Emergency Core Cooling System as well as secondary heat removal circuit. Maintaining the elevations same as in the reactor, power to volume scaling philosophy is followed in the design of the facility. Volume scaling in respect of 220 MWe PHWR units is 1:76.5. Time scale is preserved. The test facility is extensively instrumented to measure temperature, flow rate, pressure/differential pressure, level, void fraction etc., at various points in the loop. This facility will be used for LOCA and non-LOCA transient experiments. These experiments will help in the creation of a data base for the assessment and validation of thermal hydraulic computer codes used for predicting the transient behaviour during postulated accidents and operational transients.

Experiments have been conducted for single-phase and two-phase natural circulation and station blackout studies in the FISBE facility. Some of the results based on the data generated in the facility are given below. The facility is being augmented with simulated PHT Pumps, Fuel Cluster Simulators, Fast Acting Valves, Bleed Condenser, Regenerative Cooler, Jet Condenser, Pool Boiling Coolers, Low Pressure Accumulator and Secondary Feed Pump. The loop is instrumented with state of the art smart sensors and a PLC based control system. The loop is further equipped with a Fast Data Acquisition System (FDAS) for studying transient phenomenon.







Primary Circulating Pump



Low Pressure Accumulator



Quick Opening Valve



PC Based Control



Fast Data Acquisition System



Jet Condenser

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# 2.9 SEVERE ACCIDENT STUDY PROGRAM

Severe core damage phenomena can occur for a Beyond Design Basis Accident where multiple failure of safety systems has been postulated. A high degree of consequences is expected for a very low probability case. The world research covers mainly two parts namely severe core damage and containment loading with respect to energy release and radioactivity. In the field of severe core damage laboratory scale to full scale accident simulation are conducted to understand the complex phenomena arising from high temperature. Thermalhydraulics, thermo-mechanical, high temperature chemical reactions, material relocation, etc. are the major fields studied extensively.



Efforts are made to understand different facets of the phenomena for Light Water Reactors (LWRs) as well as Pressurised Heavy Water Reactors (PHWRs) to assess the pressure boundary integrity for



both the reactors and calandria vessel for the PHWRs. For the LWRs the understanding has been achieved with well experimentally



validated severe accident analysis computer codes namely ICARE/ CATHARE, ASTEC and RELAP/SCADAP and participation in the experimental validation program of PHEBUS-FPT0 experiment with code ICARE2. Some of the computational prediction are shown below for FPT0 experiment and fuel temperatutre transient for small break Loss of Coolant Accident with station Black out of VVER-1000.

For PHWR theoretical model developments namely Channel Analysis in Severe Accident (CHASA), Fuel Coolant Interaction (FCI) and Molten Pool (MPOOL) code are being developed. At present CHASA features in-channel thermal-hydraulic models, more emphasis on radiation among fuel pins and from fuel pins to pressure tube(PT), thermo-mechanical deformation model of pressure tube, calandriatube(CT) outer surface boiling crisis and high temperature chemical reaction model. Melting of the fuel pins, relocation and oxidation of the molten material, are the areas on which work is to be carried forward. Some predictions during the development period are cited below,



Two dimensional Computational Fluid Dynamics (CFD) Code development for Fuel Coolant Interaction (FCI) and behaviour of Molten Pool (MPOOL) arising from corium as a result of Beyond Design Basis Accident (BDBA) for PHWR has been carried out. The beginning of the FCI code development plant has been done with study on film stability followed by full scale code development. Effects of various fluid temperatures (degree of subcooling) on film growth are shown in the illustration.





To support the code development, experiments for studying pipe blowdown and vapor pull through in the reactor headers are planned for the initial phase of the accident. The next phase includes the experiments of channel heat up leading to thermo-mechanical phenomena in the pressure tube and boiling crisis of calandria tube with sagging and ballooning. Release of molten material from reactor channels and its distribution due to its neighbouring channels will be studied through the fuel coolant interaction experiments

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# 3. COMPACT HIGH TEMPERATURE REACTOR (CHTR)

#### INTRODUCTION

It is generally agreed that in the long term, nuclear energy would emerge as the primary source of energy replacing fossil fuels. It would be expected to satisfy all energy related needs of mankind. Thus, In addition to producing electricity, it would provide necessary energy for producing alternate fuel or energy carrier for transport applications, facilitating production of potable water and satisfy various heating needs of the populations living in colder parts of the country. Small and compact nuclear power packs with very long refuelling periods would supply electricity in areas not connected to the electrical grid of the country. A high temperature reactor has a large potential to satisfy all the abovementioned needs.

The **Compact High Temperature Reactor (CHTR)**, under development at BARC, is mainly <sup>233</sup>U -Thorium fuelled, lead-bismuth cooled and beryllium oxide moderated reactor. This reactor, initially being developed to generate about 100 kW<sub>Th</sub> power will have a core life of 15 years and will have several advanced passive safety features to enable its operation as compact power pack in remote areas not connected to the electrical grid system. The reactor is being designed to operate at 1000°C, to also facilitate demonstration of technologies for high temperature process heat applications such as hydrogen production from water. Hydrogen is being considered world wide as future energy carrier for transport applications. The design guidelines for the reactor includes utilisation of thorium-based fuel, passive reactor safety and passive heat removal features under all operating conditions so as to have very low demands in operator skill and availability.


#### Description of the CHTR

The reactor core consists of nineteen prismatic beryllium oxide (BeO) moderator blocks. These 19 blocks contain centrally located graphite fuel tubes. Each fuel tube carries fuel inside 12 equi-spaced longitudinal bores made in its wall. The fuel tube also serves as coolant channel. The fuel is based on TRISO coated particle fuel, which can withstand very high temperature (1600°C). A cross section of the particle fuel is shown below;

These particles are mixed with graphite powder as a matrix and made into cylindrical fuel compacts. The fuel compacts are packed in fuel bores in the walls of each of the nineteen fuel tubes. Eighteen blocks of beryllium oxide reflector surround the moderator blocks. These eighteen blocks have central holes to accommodate passive power regulation system. This system works on temperature feedback and in case of rise of coolant outlet temperature beyond design value, inserts negative reactivity inside the core. Graphite reflector blocks surround these beryllium oxide reflector blocks.



This part of the reactor is contained in a shell of a material resistant to corrosion against Pb-Bi eutectic alloy coolant and suitable for high temperature applications. Top and bottom closure plates of similar material close this reactor shell. Schematic of a single fuel bed and cross-sectional layout of the reactor core are shown in the figure.

## Cross-sectional schematic of TRISO coated particle fuel for CHTR





Above the top cover plate and below the bottom cover plate, plenums provide for core-outgoing and core-incoming coolant respectively. Nuclear heat from the reactor core is removed passively by the lead-bismuth liquid metal coolant which flows due to natural circulation between the plenums, upward through the fuel tubes and returning through the down comer tubes. These plenums have graphite flow guiding blocks to increase the velocity of the coolant between the coolant channel exit and the entry to the down comer tubes of the reactor. The flow-guiding blocks have passages for the coolant to flow from the inner to outer region of the plenum. The reactor shell is surrounded by two gas gaps that act as insulators during normal reactor operation and reduce heat loss in the radial direction. There is an outer steel shell, surrounded by heat sink. This shell has been provided with fins to improve its heat transfer capabilities. A passive system has been provided to fill the gas gaps with molten metal in case of abnormal rises in coolant outlet temperature thereby providing a conduction path of heat

transfer from reactor core to outside heat sink. On top of the upper plenum, the reactor has multi-layer heat utilisation vessels to provide an interface to systems for high temperature heat applications. A set of sodium heat pipes is in the upper plenum of the reactor to passively transfer heat from the upper plenum to the heat utilisation vessels with a minimum drop of temperature. Another set of heat pipes transfers heat from the upper plenum to the atmospheric air in the case of a postulated accident. To shut down the reactor, a set of seven shut-off rods has been provided, which fall by gravity in the central seven coolant channels. Appropriate instrumentation like neutron detectors, fission/ ion chambers, various sensors and auxiliary systems such as a cover gas system; purification systems, active interventions etc. are being incorporated in the design as necessary. If temperature of coolant increases beyond its design value, a set of seven shut-off rods fall by gravity in the central seven coolant channels to shutdown the reactor.

#### IMPORTANT DESIGN PARAMETERS OF CHTR

Reactor power	:	100 kWt <sub>th</sub>
Core configuration	:	Vertical, natural circulation type
Coolant	:	Molten lead-bismuth eutectic alloy
Number of fuel tubes	:	19 with 35 mm ID and 75 mm OD
Fuel	:	<sup>233</sup> UC <sub>2</sub> + ThC <sub>2</sub> based TRISO (TRi-ISOtropic) coated particles made fuel compacts.
Refuelling period	:	15 Effective Full Power Years (EFPYs)
Fuel burnup	:	68000 MWd/Tonne
Fuel enrichment	:	33.75% (2.7 kg 233U + 5.3 kg Th)
Moderator material	:	BeO
Reflector material	:	Partially BeO and graphite
Lattice pitch	:	135 mm (triangular pitch)
Active fuel length	:	0.7 m
Total core flow rate	;	6.7 kg/s
Coolant inlet temperature	:	900 °C
Coolant outlet temperature	;	1000 °C
Loop height	:	1.4 m
Core height	;	1.0 m
Core diameter	:	1.27 m
Passive power regulation system/ primary shutdown system	:	18 floating annular B <sub>4</sub> C elements of passive power regulation system
Secondary shut down system		7 mechanical tungsten made shut off rods

#### IMPORTANT SAFETY FEATURES OF CHTR

- A strong negative Doppler coefficient of the fuel for any operating condition;
- High thermal inertia of the all-ceramic core and low core power density;
- A negative moderator temperature coefficient;
- A large margin between the normal operating temperature of the fuel (around 1100 °C) and the leak tightness limit of the TRISO coated particle fuel (1600 °C) to retain fission products and gases;
- There is a very large thermal margin to Pb-Bi (Boiling point 1670 °C) boiling;
- Use of the low pressure Pb-Bi coolant no over pressurization and no chance of reactor thermal explosion due to coolant emergency overheating;
- The high temperature Pb-Bi coolant is chemically inert. Even in the eventuality
  of contact with air or water, it does not react violently with explosions or fires;
- There is a negligible thermal energy stored in the coolant and available for release in the event of a leak or accident;
- For coolant, the reactivity effects (void, power, temperature, etc.) are negative.
- A low induced long-lived gamma activity of the coolant; in case of a leakage, the coolant retains iodine and other radionucleides.
- No pressure in the coolant allows the use of a graphite coolant channel, improving neutronic of the reactor.



At the current stage, a feasible configuration of the reactor from reactor physics, heat removal and reactor control considerations have been worked out. R & D and technology development for most of the technology development areas have been initiated. For carrying out system studies, verifying the developed computer codes and for material compatibility studies; many experimental facilities are under development. Developmental work related to materials and fuels are being carried out. Analytical studies related to safety and effects of external events are underway.

#### Future Work and Design of New Reactors

- Experimental facility for CHTR (Initial operation at low temperature – subsequently increase in power to demonstrate high temperature capability)
- Design and development related activities for 600 MW<sub>Th</sub> HTR for hydrogen production as well as a low power compact power pack for supplying electricity in areas not con nected to electrical grid.

#### 3.1 PHYSICS DESIGN OF CHTR

The physics design of the CHTR has been carried out with 2.7 Kg <sup>233</sup>U in 8 Kg of fuel (U+Th). A core life of nearly 15 full power years could be achieved with the above configuration. The variation of K-eff with burnup is shown in the figure. Since the initial reactivity (>100mk) is quite large, it had been decided to add a burnable poison to the fuel. Two options have been analysed respectively, with 180 g Erbium homogeneously mixed in fuel and 40 g Gadolinium only in the central fuel assembly. The effect of both types of burnable poisons can be seen in Figure. The Tables present the total worth of primary and secondary shut-down systems, fuel temperature coefficient and maximum worth of a single control rod when the reactor is in critical state.

When reactor is in critical state, if there is an inadvertent withdrawal of a control rod, a positive reactivity is introduced, resulting in power rise and rise in fuel and coolant temperatures. A negative reactivity feedback is introduced by the rise in fuel temperature and the power stabilizes at about 2 to 3 times the initial power. The variation of relative power, fuel and coolant temperatures with transient time for two types of burnable absorbers are illustrated.



Reactor State	Keff at operating temp (1000 °C)
All CRs IN	0.8501961
All CRs OUT	1.0488827
Total Worth of Control Rods	222.8 mk

Worth of control rods in primary shut-down system

Shut off rods in coolant channel	K <sub>eff</sub> Tungston without Clad (D=2.0 cm)
All 19 channel	0.6599
Inner 7 channel	0.8634
Outer 12 channels	0.8810

Worth of Shut-off rods in secondary shut-down system

	Gadolinium	Erbium
Max. Control Rod Worth (mk) at Critical Height	2.28	1.74
Fuel Temp. Coeff. (Δk/k / °C)	- 1.1 × 10 <sup>-5</sup>	- 3 × 10 <sup>-5</sup>

Control rod worth and fuel temp. coeff. for different burnable absorbers



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## 3.2 MODELING OF PERFORMANCE OF TRISO PARTICLES

CHTR uses TRISO coated fuel particles. This study aims to model the behavior of the TRISO particle under irradiation. The TRISO fuel particles consist of a central fuel containing kernel surrounded by four layers, viz. low density pyrocarbon buffer layer, inner high-density pyrocarbon layer, silicon carbide layer and outer high-density pyrocarbon layer. The overall dimension of this particle is ~900  $\mu$ m. The silicon carbide layer acts as the retaining layer for the fission products and no external cladding is needed. There are numerous mechanisms for failure of such particles viz. kernel migration, fission

product attack on structural layer, etc. One such mode is by fracture of the silicon carbide layer (pressure vessel failure). Some preliminary modelling of TRISO particles has been initiated, by using FEM to model this pressure vessel failure mechanism. For these calculations, an axisymmetric element capable of modelling creep and irradiation induced swelling was formulated and programmed into a general purpose FEM code. A 0.5-degree sector (to maintain aspect ratio of one) of the TRISO particle was analysed. The dimensions of the particle and the meshing used are illustrated and the results along with comparison with available theoretical solutions for a perfectly spherical particle are presented in the Table.

	Analytical (MPa)	Soln. Results of developed FEM code (MPa)
Radial stress at ID of SiC layer	-7.94	-7.60
Radial stress at OD of SiC layer	-14.11	-13.01
Tangential stress at ID of OPyC	51.34	44.87
Tangential stress at ID of SiC layer	-47.69	-47.84

Comparison of stresses obtained in the present analysis with the analytical solution



The stresses induced have a statistical distribution due to the variations in kernel diameter, buffer layer thickness, etc. Additionally the coating strengths also exhibit a Weibull distribution. All these variations need to be combined to estimate the failure fraction as a function of burn-up, using statistical tools, based on Monte Carlo techniques. This study will lead to evaluation of acceptability of the fabrication tolerances for TRISO particles and modeling of other degradation mechanisms.

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# 3.3 PASSIVE POWER REGULATION SYSTEM (PPRS)

The concept of passive power regulation is one of the important guidelines in the design of CHTR. The coolant temperature is used to passively drive the control rods, to achieve power regulation. The system performance analysis and its response to various anticipated scenarios are discussed.



The PPRS, has a gas header connected to a driver tube, control tube, floating absorber and lead bismuth eutectic. The control tube is concentric and surrounds the driver tube. The gas header, attached to a niobium driver tube acts as a temperature sensor.



The driver tube is housed within a niobium control tube and is filled with Pb-Bi eutectic. The control rod, which is housed in the control tube, is made up of boron carbide and is clad in niobium. The annular space between the driver and control tube is also filled with Pb-Bi eutectic and the control rod is designed to float in it. The system is thus designed to have two free liquid surfaces, one in the driver tube and the other in the control tube. The space above the liquid level is filled with an inert gas like argon. The PPRS gas header, located in the top plenum, is submerged in the coolant. The current location of the gas header ensures that it senses the coolant temperature immediately downstream of the heat removal devices (a set of heat pipes). Under normal operating conditions, the gas header is surrounded by coolant at a temperature of approx. 900 °C. Any condition, which causes the coolant to return at a temperature higher than the normal temperature will cause the gas, in the gas header, to also heat up. Due to this pressure will rise in the driver tube, causing a pressure imbalance between the driver and the control tube. This, in turn, will cause the level of Pb-Bi eutectic in the driver tube to go down and that in the control tube to go up. Since the absorber rod floats on Pb-Bi eutectic, it will also

rise with the Pb-Bi eutectic level in the core, thus inserting negative reactivity. Depending on the temperature rise that has been sensed, the system will stabilise at a particular value of insertion. The PPRS was analysed using TRAPPOR (TRAnsient Passive POwer Regulation), a computer code developed in-house.

Simultaneously, means to actively control the CHTR are also being explored within the same setup. Analysis has been carried out for two scenarios, one in which the driver tube is pressurised and the other in which the control tube is depressurised.



Pressurisation of the driver tube (Effect of different pressure input rates on insertion on control rods.)



In addition, work has been done to analyse the upper plenum design of the CHTR. Since the PPRS depends, for its operation, on the temperature signal carried by the coolant, it is important to know the time frame in which coolant will carry the signal to the PPRS header. The analysis showed that the coolant velocities are very low.



Following this analysis the design of the upper plenum has been changed to have channels for flow of the coolant. These channels serve as preferential flow paths and thus reduce the time interval in which the change is sensed. The headers are now submerged in pockets constructed along these channels.

Design of a PPRS test setup, for carrying out experimental analysis under various transients and validation of computer codes, is also underway.

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#### 3.4 THERMAL ANALYSIS OF CHTR CORE UNDER NORMAL OPERATING AND POSTULATED **ACCIDENT CONDITIONS**

A detailed thermal analysis is essential in design, to ensure fuel integrity under all foreseeable scenarios, without operator intervention. A steady state analysis of the reactor core has been carried out to determine the prevalent temperatures under normal operating conditions, to estimate the thermal stresses. A steady state parametric analysis has also been carried out to estimate and minimise the heat loss from the reactor core, through the gas gaps by all modes, under normal operating conditions.

In the analyses It has been assumed that under postulated accident conditions, the reactor power stabilizes at a peak neutronically limited value of, twice the normal operating power and the entire heat is transferred to an outer heat sink by conduction, through the reactor vessel wall. The maximum fuel temperature has been calculated by a FEM analysis under steady state conditions. This has been followed by a transient analysis, to estimate the available time for safety devices to act. In this analysis it has been assumed that all heat sinks have been lost along with a sudden rise in power. It has also been assumed that an adiabatic boundary condition prevails for all surfaces. The initial temperature distribution has been applied from a steady state analysis and a step increase of twice the reactor power is assumed.

In all the above analysis the thermal contact resistances at all interfaces were assumed negligible. The heat generation in nuclear fuel and moderator has been considered. Appropriate material properties have also been assumed.











The temperature distribution given in the figure, at planes through the top, mid and bottom of the active length shows that the maximum temperature gradient occurs just outside the outermost location of the fuel channel and hence becomes the prime candidate for failure due to thermal stresses.

The variation of total heat loss with respect to inner gas gap shows that increasing this gap beyond 20 mm will not affect any significant economy in core heat.



Temperature distribution under postulated accident conditions with radial conduction path been established to outer heat sink



Variation of radial heat loss with respect to inner gas gap dimension. Outer gas gap is kept constant at 10 mm.

The temperatures distribution obtained under postulated accident conditions with heat rejection to the outer heat sinks have been obtained by considering aluminium, indium and tin in the gas gaps which yielded maximum fuel temperatures well below the maximum allowable value for the fuel ( $1600^{\circ}$ C).

The results from the transient analysis show that even after fifty minutes have elapsed after overpower, the fuel temperature does not exceed  $1510^{\circ}$ C, well below the limiting temperature of the fuel.



Temperature distribution in central fuel channel after 3000 seconds (50 minutes) after overpower and loss of all heat sinks

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## 3.5 PRELIMINARY DESIGN OF GAS-GAP FILLING SYSTEM

One of the basic guidelines of CHTR design is the rejection of the entire core heat to a surrounding heat sink under a condition of heat sink loss and a neutronically limited power of 200 kW. This study aims to design such a system.

The preliminary design of the gas-gap filling system has been completed. A brief description of the system is given in the following paragraphs.

This system consists of a bulb, which is immersed in the liquid metal in the upper plenum, downstream to the heat pipes. This bulb



communicates with a reservoir, which contains the liquid metal. The reservoir is connected to the inner gas gap by means of siphon tubes.

The bulb senses the temperature increase of the coolant and forces the liquid metal to move up the siphon tube and siphon is started. The liquid metal is conveyed to the inner gas gap by means of the siphon tube. Holes have been provided on the shell separating the two gas gaps at its bottom end so that liquid metal flows into the outer gas gap also. The gas displaced by the liquid metal is pushed into a gas tank.

As the level in the reservoir starts decreasing, the pressure of the gas above the liquid in the reservoir also decreases. This may prevent the flow of the liquid metal after certain time. To avoid such a situation, a vent tube is provided which is connected at one end to the gas tank and the other end is dipped into the reservoir upto a certain depth. When the liquid in the reservoir goes below this level,

pressure on both sides becomes equal and this ensures that the siphon process is continued without any break. This system has an advantage that once started this system will stop only when the entire liquid metal has been poured down. To reduce the chances of common mode failure, the liquid metal reservoir is subdivided into four compartments, the capacity of each compartment catering to one quarter of the gas gap volume.

Three probable liquid metals, which can be used to fill the gas gaps, have been considered. These liquids are Indium, Tin and Aluminum which were chosen, due to their ease of availability, low melting points and good thermal conductivities.

Using this principle, the height of the siphon above the liquid level was calculated to be 135 mm for Indium and Tin and 380 mm for Aluminium for a set point of 1000°C. The gas bulb encounters this temperature in the event of a loss of heat sink.

A computer programme FD (Finite Difference) was written for estimating the time taken for filling the gas gaps by solving the onedimensional momentum conservation equation by using finite difference methodology.



The figure also shows the effect of the vent tube. The line A shows the liquid level in the reservoir, without considering the effect of the vent tube, wherein the pressure in the reservoir keeps decreasing, preventing the complete flow of liquid metal. The case in which the vent tube is incorporated is shown as B. The level in the reservoir falls below that of the excess liquid metal (indicated by the blue line), which implies that the gas-gaps have been filled.

Cases were analysed to get the gas-gap filling time for each of the three liquid metals considered. Different numbers of siphon tubes were also considered. The results are summarised.

No. of siphon tubes per reservoir	Time taken to fill gas-gaps (seconds)		
	Indium	Tin	Aluminium
5	26.0	25.6	22.8
6	23.2	23.2	20.4
7	21.2	21.3	18.6
9	18.45	18.5	16.1

Estimated time taken to fill the gas gaps

A parametric analysis to fix the size of the bulb has also been carried out. The results indicate that as the volume of the gas and metal (of the bulb) decreases, the response also becomes faster. This may be accounted by the high heat transfer coefficient of the liquid metal, which makes the effect of decrease in surface area less apparent. A bulb of 5 cm diameter and 5cm height gives a response time of 2 seconds and has been tentatively selected for use in the present system.

Estimates of the fuel temperatures, by using each of the above metals in the gas gaps, have been obtained by finite element analyses. The maximum fuel temperatures calculated are 1304°C, 1240°C and 1321°C for Indium, Aluminium and Tin respectively, which are much below the maximum allowable temperature (1600°C) of TRISO type fuel.

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## 3.6 APPROACH FOR GRAPHITE COMPONENT DESIGN

A large proportion of CHTR is composed of graphite, a brittle material. Two draft codes, (which are yet to be finalized) are available for graphite component design of HTRs, namely the ASME Section III Division 2 Subsection CE and the KTA 3232, Ceramic internals for HTR pressure vessels, 1992.

In view of the absence of authoritative design rules for nuclear components, a literature survey was carried out to identify the



Graphite fuel tube

design issues involved. In the design of brittle materials, the usual deterministic design procedures are invalidated by two behavioral characteristics, i.e., the statistical nature of its strength and its large variation. Hence the survival probability in a given stress distribution, is thereby calculated using Weibull statistical design technique. The accurate determination of statistical parameters, a large number of samples need to be tested. The fatigue curves for graphite follow the same pattern as that of metals, when drawn with homologous stress (peak stress divided by mean tensile strength) vs. number of cycles, but with a larger scatter in data.

Following this assessment a code Brittle Design.f90 was written. The failure probability of a given structure was estimated by utilising the stress distribution obtained from a FEM analysis. The effect of multiaxial stresses was incorporated by using the principle of superposition. The failure probability of each element is given by,

$P_s^e = \exp\left\{-\int_V \left[\left(\frac{\sigma_1}{\sigma_0}\right)^m + \left(\frac{\sigma_2}{\sigma_0}\right)^m + \left(\frac{\sigma_3}{\sigma_0}\right)^m\right]dV\right\}$	
--	--

Where	
$\sigma_1, \sigma_2, \sigma_3$	Principal stresses (assumed uniform within each element)
$\sigma_0$	Characteristic strength
m	Weibull modulus
v	Volume of component
e	Element number
$P_s^{e}$	Survival probability of the eth element

The failure probability (Ps) of the entire structure is then calculated by,



This code has been used to estimate the failure probabilities of graphite reflector and the upper plenum blocks.

The thermal stresses in the upper plenum block and the graphite reflector block was calculated and used for estimating the failure probabilities. Tentative values for the statistical parameters have been assumed for this analysis. It has been found that if the upper plenum block is fully restrained in radial direction, the probability of failure is approximately one, implying an almost certain failure. However, this value dropped to 0.0040 if free expansion is allowed, representative of the more favourable stress levels. The probability of failure of the graphite reflector blocks has been estimated to be 1.24x10<sup>-5</sup> and 1.87x10<sup>-6</sup> for the cases where it is free to expand in axial direction and fully constrained in the axial direction respectively. It is interesting to note that, for this case, an axial constraint gives lower value of failure probability. An examination of the stress distribution indicates that the tensile stresses are more predominant in the first case, which is reflected in the higher value of failure probability. A similar approach is applicable for the design of beryllia blocks.

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## Compact High Temperature Reactor (CHTR)

## 3.7 MODELING, DESIGN, FABRICATION AND TESTING OF HIGH TEMPERATURE HEAT PIPES

The heat pipe is a very effective device for transmitting heat at high transfer rates over considerable distances with extremely small temperature drops. They are simple in construction, easy to control, passive in operation and can be used in any orientation. Heat pipes are used in CHTR to remove heat, under normal operating and accidental conditions, from the primary coolant.

#### Development of Computer Codes

A computer code "HPDATA" has been developed to simulate and carry out heat pipe design and analysis under steady state conditions. The heat pipe operation is dependent on various limits to its operation i.e. viscous limit, sonic limit, capillary limit, boiling limit and entrainment limit. This code can calculate the various operating limits for heat pipe operation and the parametric variation of these limits for various heat pipe diameters and wick configurations, by making use of empirical correlations. The code also calculates, the temperature drop between the heat pipe ends and the vapour pressure profile inside the heat pipe.





Variation of entrainment, capillary and boiling limitations with internal diameters of heat pipes



Vapour pressure variation inside heat pipe

## Development of High Temperature Heat Pipe Test Facility

The high temperature heat pipe test set up aims to

- study performance of heat pipes,
- study startup behaviour of frozen heat pipes,
- study operation under various transients and
- evaluate impact of corrosion by liquid metals under high temperature on heat pipe lifetime and performance
- Serve as a test bed for validating computer codes

The working temperature range for the setup has been selected as 400°C to 1200°C, so as to enable testing of moderate to high temperature heat pipes. With minor modifications, this may also be used for testing of low temperature heat pipes. The heat pipe has three regions - evaporator, adiabatic and condenser sections, which serve as heat input, heat transport and heat output areas respectively. Hence the facility has a means to input heat (by means of a heater), reject heat (to a calorimeter) and means to minimise heat losses (by means of insulation). The setup consists of a heat pipe enclosed in a metallic vessel. To prevent oxidation of the heat pipe material at high temperatures, the vessel is purged with a mixture of argon and helium gas. The portion of the vessel enclosing the condenser section of the heat pipe is cooled by means of silicone oil flowing in a helical coil wrapped around its outer diameter.



The purging gases also play an additional role in regulating the amount of heat removed. Heat losses from the adiabatic region is minimised by alumina based high temperature insulation surrounding the vessel. The evaporator portion of the heat pipe is surrounded by the heater arrangement, which is positioned inside the vessel. The heat pipe test setup is designed to test heat pipes at a maximum power input of 20 kW. In view of the large heat input, high temperatures involved and rapid heating requirement, a RF induction heater is used.

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## 3.8 THERMAL HYDRAULIC STUDIES ON THE COMPACT HIGH TEMPERATURE REACTOR (CHTR)

The reactor mainly consists of fuel tubes, solid moderator and reflector, coolant, Primary Heat Transfer (PHT) loop and safety & control systems. The flow of the coolant in the loop is maintained by natural circulation. The loop and the core are contained in a high temperature metallic vessel. Annular gas gaps are provided to reduce the radial heat loss during normal operating condition. The solid surfaces of the gaps are coated with low emissive material to reduce the heat transfer rate, further. These two gaps are followed by a high conducting solid wall and then by the outermost shell. External fins are provided to enhance the heat transfer during accidental conditions.

#### Thermal Hydraulic Analysis

A computer code, HTR-NC, is developed for thermal hydraulic analysis of primary loop of 100 kW<sub>th</sub> CHTR. Parametric studies on coolant flow rate; temperature variation, pressure drop and power in the loop have been carried out at several stages of development. The results from the analysis have been applied to optimize the down comer tube size and orifice size of the loop. Variation of size of the orifices, which are located at the down comers, with the core height are illustrated. Analysis has also been carried out for 5 MW<sub>th</sub> CHTR, to optimize the geometry of the primary coolant loop. The variation of core outlet temperature at different chimney height for different fuel tube diameters is also illustrated in the figure.





#### Heat Transfer Analysis

Steady state thermal analyses have been carried out to find temperature distribution in the geometry of the reactor. The analyses have been carried out for the normal operating and accidental conditions of the reactor. In the normal operating condition the reactor operates at 100 kWth and the coolant temperature rises from 900° C to 1000° C, by taking heat from the fuel tube. Analyses have been carried out for different filling liquid metals and reactor structure materials. It has been found that the maximum fuel temperature is below the maximum allowable temperature (~1600°C). The temperature distribution at the mid cross section of the reactor at normal operating temperature is illustrated in the figure.



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SHIELD BLOCKS-DEPLETED CONVERTER ASSLY (0.65 % U-235)-

## 4. PROTOTYPE FAST BREEDER REACTOR

#### INTRODUCTION

The second stage of Indian nuclear power programme involves establishing Fast Breeder Reactors for power generation. The Prototype Fast Breeder Reactor is being developed to demonstrate the techno-economic viability of Fast Breeder Reactor technology. This chapter highlights the recent activities carried out in the fields of radiation shielding and design of inclined fuel transfer machine. The PFBR is a 500 MWe, sodium cooled, pool type, mixed oxide (MOX) fuelled reactor having two secondary loops. The primary objective of the PFBR is to demonstrate techno-economic viability of fast breeder reactors on an industrial scale. The entire primary sodium circuit is contained in a large diameter vessel (Æ 12900 mm) called main vessel and consists of core, primary pumps, intermediate heat exchanger and primary pipe connecting the pumps and the grid plate. The vessel has no penetrations and is welded at the top to the roof slab. The main vessel is cooled by cold sodium to enhance its structural integrity. The core subassemblies are supported on the grid plate, which in turn is supported on the core support structure.



The main vessel is surrounded by the safety vessel, closely following the shape of the main vessel, with a nominal gap of 300 mm to permit robotic and ultrasonic inspection of the vessels. The safety vessel also helps to keep the sodium level above the inlet windows of the intermediate heat exchanger ensuring continued cooling of the core in case of a leak of main vessel. The inter space between main & safety vessel is filled with inert nitrogen. The main vessel is closed at its top by a top shield, which includes roof slab, large & small rotary plugs and control plug. The top shield is a box structure made from special carbon steel plates and is filled with heavy density concrete ( $r = 3500 \text{ kg/m}^3$ ) and provides thermal and biological shielding in the top axial direction. The principal material of construction is SS 316 LN for the vessels and boiler quality carbon steel for top shield. The reactor vault concrete provides the biological shielding in the radial and bottom axial direction outside the main vessel.

Design features		
Thermal power, MWt	1250	
Electric output, MWe	500	
Core height, mm	1000	
Core Diameter, mm	1900	
Fuel	PuO <sub>2</sub> -UO <sub>2</sub>	
Fuel pin outer diameter, mm	6.6	
Pins per fuel subassembly	217	
Fuel clad material	20 % CW D9	
Diameter of main vessel, mm	12900	
Primary circuit layout	Pool	
Primary inlet / outlet temp, °C	397 / 547	
Steam temperature , °C	490	
Steam pressure , MPa	16.6	
Reactor containment	Rectangular	
Plant life, y	40	
No of shutdown systems	2	
No. of decay heat removal systems	2	

## 4.1 DESIGN AND DEVELOPMENT OF INCLINED FUEL TRANSFER MACHINE (IFTM) FOR FUEL HANDLING SYSTEM OF PFBR-500 MWe



#### Introduction

For better utilization of fuel & available natural resources in India and based on the successful operation of 40 MWt Fast Breeder Test Reactor (FBTR) at the IGCAR, Kalpakkam, a 500 MWe Prototype Fast Breeder Reactor (PFBR) is being built at Kalpakkam, which will be first reactor of its kind in India. PFBR will be a 500 MWe (1250 MWt), 2-loop, sodium cooled, pool type reactor. It will utilize the MoX fuel ( $PuO_2 + UO_2$ ) and depleted Uranium oxide as blanket. PFBR is designed vertical in configuration. Off load refuelling is envisaged for PFBR. It is designed to do refuelling after every 185 Effective Full Power Days (EFPD) of the reactor. In one refuelling campaign 62 fuel SA, 25 blanket SA and 5 absorber SA will be replaced. Inclined Fuel Transfer Machine (IFTM) is one of the fuel handling machine of PFBR which transfers spent SA from reactor vessel to fuel building and fresh SA from fuel building to reactor vessel.

#### Description

During spent sub-assembly (SA) handling, IFTM receives the spent SA from Transfer Arm (TA) inside the reactor vessel and delivers it to Cell Transfer Machine (CTM) in fuel building to transfer it to storage bay and during fresh SA handling it receives fresh SA from CTM and delivers it to TA to place inside the reactor core. IFTM transfers SA in 17<sup>o</sup> inclined position to the vertical. Design of IFTM is totally indigenous. The irradiated SA after being cooled at storage location inside reactor vessel is put in a sodium filled Transfer Pot (TP) of IFTM by Transfer Arm (TA) at In-vessel Transfer Position (IVTP). TP is then hoisted up inside Rotatable Shielded Leg (RSL) by hoisting mechanism through Primary Ramp (PR) & Primary Tilting Mechanism (PTM). RSL is rotated by 180° by rotation mechanism and aligned on secondary side and the TP is lowered in Ex-Vessel Transfer Position (EVTP) through Secondary Ramp (SR) Secondary Tilting Mechanism (STM) from where the irradiated SA is replaced by fresh SA using Cell Transfer Machine (CTM). The fresh SA will be transferred from EVTP to IVTP in the reverse manner. A shield plug has been provided in the primary ramp for attenuating primary sodium gamma rays during reactor operation. Gate valves have been provided on both primary and secondary ramps, which act as a barrier between the radioactive argon cover gas of MV & fresh argon gas of Fuel Transfer Cell (FTC), which is required for containment isolation. Bellows are provided on both sides to absorb the differential thermal expansion. Fuel handling takes place within a leaktight cell. Adequate shielding and sealing arrangement has been provided in IFTM. During fuel handling IFTM internals are maintained at high temperature (423-473 K) by hot argon flushing during fuel handling operation in order to maintain sodium in liquid form filled in transfer pot. Also during reactor some of the components of IFTM viz. PR & PTM see very high temperatures (823 K).

#### Design and analysis

IFTM design has been finalized. Detailed design of all the components of IFTM has been completed. Design of all the components of IFTM is based on all the types of loadings viz. dead weight, thermal loading and seismic loading. Detailed stress analysis has been performed for static, thermal and seismic loadings. Design of PR & PTM has been analyzed for creep and fatigue loadings also as they see high temperature and thermal cycling. Profile of ramps and tilting mechanism has been finalized by making 3-D models and simulating the pot movement through them, which was done using VIZ software. Various design documents and reports (45) have been prepared and issued.

#### Development activities

Functioning of IFTM has been demonstrated by commissioning of small scale (1:10) acrylic working model of IFTM. Design of critical components has been validated by making small scale models viz. secondary ramp, tilting mechanism, siphoning arrangement, safety brake, etc.

Test setup of double chain hoisting mechanism has been commissioned to see the behavior of chain-sprocket system and chain sensing arrangement. Various functional and design requirements viz. compatibility of chain & sprocket, possibility of chain coming out of sprocket grooves, entanglement of two chains while pot movement, effect of unequal length of chains, behavior of chain with load, performance of chain sensing arrangement, etc. have been checked. Load testing of operation and emergency handling have been demonstrated successfully. Endurance test is in progress.

#### Manufacturing

After completion of detailed design various detailed drawings (165) and technical specification for manufacture of IFTM have been prepared and handed over to BHAVINI to float the tender. Bhavini had floated tender for manufacture of IFTM and quotations were received. Technical evaluation of bidders is in progress.

## Performance test

In future it is planned to manufacture an IFTM and test it in shop for functional requirements. Then IFTM will be installed at Large Component Test Rig (LCTR), Kalpakkam for performance test in reactor simulated conditions. First it will be tested in air at ambient conditions and then at reactor simulated conditions along with control system before making it reactor worthy.



Test setup of hoisting mechanism

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## 4.2 PROTOTYPE FAST BREEDER REACTOR SHIELDING EXPERIMENTS AT APSARA

PFBR is a pool type sodium cooled fast reactor. Core is surrounded by fertile blankets and in-vessel shielding. The Intermediate Heat eXchanger (IHX) and sodium pumps are in the pool. The in-vessel shield is provided to reduce radiation damage to inner vessel, secondary sodium activation, activation of IHX, sodium pumps, axial leakages through bottom FP gas plenum. Mockup shielding experiments in Apsara shielding corner were carried out to optimize design of this in-vessel shield.

Apsara is a 1MW<sub>th</sub> swimming pool reactor with HEU Al alloy fuel. It is presently operated up to 400 kW<sub>th</sub>. As Apsara is a thermal reactor, PFBR blanket leakage neutron spectrum in shielding corner was simulated using depleted uranium fuel assemblies.

#### Summary of experimental programmes

- Six radial bulk shielding experiments were planned to evaluate secondary sodium activation.
- One axial bulk shielding experiment was planned to estimate attenuation of neutrons reaching neutron monitoring system (fission counters).
- Three radiation streaming experiments were planned to
- (i) Simulate plenum region of fuel subassembly, an empty space in each fuel assembly pin to accumulate fission product

gases and study the damaging effect of leaking fast neutrons on the grid plate,

- (ii) Simulate transfer arm model to study gamma streaming from the active top sodium layer for different configurations and
- (iii) Simulate gaps in the top deck plate to study radiation streaming of mainly sodium gammas.

In any shielding calculation scheme the uncertainties due to modeling, method and nuclear data are absorbed in bias factors obtained from experimental measurements. The ratios of measured to calculated reaction rates corresponding to parameters of interest (sodium activation in IHX, fission rate at fission counter detector location, fast neutron leaking from bottom plenum, exit gamma for transfer arm etc.) are called Bias Factors (BF). These BFs are used as multipliers in the exit flux from a shielding set up. These experiments were used by reactor physicists of IGCAR in arriving at proper bias factors.

## Summary of bias factors arrived at by designers of PFBR shield

- A bias factor of 4 is necessary for transport through shields of SS/Na/ B<sub>4</sub>C.
- Bias factor has come down to a factor of 2.5 in case of secondary sodium activity in IHX.
- Neutron flux at detector location on the lattice plate is under-predicted by a factor of 2.

A round robin Benchmark inter-comparison was also performed by various shielding experts of BARC, IGCAR and AERB using codes and nuclear data available with them. Eight sets of experiments to study neutron transport through various single materials of interest to fast reactor shield design, which would provide integral benchmark data for nuclear data evaluation are also completed. A typical experimental lay out for these experiments is shown in figure. Activation detectors sensitive to thermal, epithermal and fast neutrons were irradiated at various depths of shield models consisting of single shield materials. Some typical results of neutron flux attenuation in various shield materials are illustrated.







Top View of a typical converter assembly and shield model arrangement in Apsara Reactor Shielding Corner





Attenuation of epithermal neutron flux



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## 5. R & D FOR BOILING WATER REACTORS

## INTRODUCTION

The twin units of Boiling Water Reactors at Tarapur were the first nuclear power stations established in the Indian sub continent. These reactors, commissioned in 1969, are the longest serving Boiling Water Reactors in the world. This chapter on Boiling Water Reactors highlights the work done on simulated optimisation scheme for control rod withdrawal sequence, the core shroud acoustic load evaluation and the transient dynamic response.

Tarapur Atomic Power Station (TAPS) has an installed capacity of  $2 \times 210$  MWe. This reactor is a forced circulation boiling water reactor, producing steam for direct use in the steam turbine. The fuel consists of uranium dioxide pellets contained in Zircaloy-2 tubes. Water serves as both the moderator and coolant.

The bottom entry cruciform control rods provide Safety Control Rods Accelerated Movement (SCRAM) as well as reactor control function. Each drive has its own separate control and scram, devices. A control rod reactivity-limiting device called Rod Worth Minimiser (RWM) limits the reactivity addition in the reactor and assures that any reactivity addition is as per predetermined program. In addition to the control rod drive system, a stand by Liquid Poison System (LPS) is provided to manually initiate injection of a neutron poison (liquid sodium pentaborate solution) into the reactor to make the reactor sub-critical.

An Emergency Core Cooling System (ECCS) is designed to pump water directly from the suppression pool into the reactor vessel and drywell under the postulated loss of coolant accident conditions. The system design is such that there is minimal fuel damage and containment structures are not challenged even in the worst accident scenario. This system has inbuilt redundancy to take care of any component failure.



CC	DRE DESIGN	ΙΟΑΤΑ
Thermal Out put		660 MM/t (Dual Cycla) / 210 MM/a
Perseta dala seconda su tauta	-	
Re-rated thermal output		530 MVVT (Single Cycle) / 160 MVVe
(Operating on single cycle since 1984 af	ter isolation of	secondary steam generators)
Operating pressure	-	6.89 x 10 <sup>6</sup> Pa(1000 psig)
Total core flow	-	2897.78 kg/s (23 X 10 <sup>6</sup> lbs/h)
Primary steam flow	-	261.11 kg/s
F	uel Assembl	ies
Number of fuel assemblies	-	284
Fuel rod array	-	6 X 6
Active Fuel length	-	3.66 m (12 feet)
Fuel material	-	UO <sub>2</sub> (Pellets)
Average fuel enrichment	-	2.44 % <sup>235</sup> U
Fuel clad	-	Zircaloy-2
Discharge Burn-up/fuel bundle	-	21600 MWd/t
C	Control Syste	em
Number of movable control rods	-	69
Shape of control blade	-	Cruciform
Control rod poison material	-	B <sub>4</sub> C granules in SS tubes.
React	tor Pressure	Vessel
Reactor vessel inside diameter	-	3.66 m (12 feet)
Reactor vessel overall length	-	16.41 m (53 feet - 10 inch.)
Reactor	Recirculatio	on System
Number of Loops	-	02
Pump Capacity	-	7404.26 m³/h(32,600 gpm each)
Pump head (Total)	-	50.29 m (165 feet)
	Turbine	
Туре	-	TC Dual Admission Single flow

## 5.1 SIMULATED ANNEALING OPTIMIZATION SCHEME FOR OBTAINING CONTROL ROD WITHDRAWAL SEQUENCE IN TAPS REACTORS

Tarapur Atomic Power Station (TAPS) consists of two Boiling Water Reactors (BWRs). The monitoring of BWRs has traditionally been accomplished by a combination of analytical and plant measurement techniques. Earlier methods of core analysis were composed of curve fits or discrete factors applied to core lattice analytical results obtained from more sophisticated nuclear models. The high speed and large memory of latest computer systems have provided an ideal capability for enhancement of core analysis methods.

The key to the accurate core analysis is the rapid and accurate determination of the three-dimensional core power distribution. The more accurately the power distribution is known, the more closely the operating limits may be set to real thermal design limits of the fuel. This permits a higher design power density, with consequential fuel cost reductions. The core power distribution calculation is performed in two steps - (i) the calculations of basic lattice parameters and (ii) three dimensional core simulations.

In TAPS, 69 cruciform type boron carbide control rods are provided for control and shut down purpose. Control rods are withdrawn to maintain the criticality and to control the core power distribution during the reactor power operation. Shifts in core power distribution can be controlled (within certain limits) by choice of control rod withdrawal sequence. A computer code COMETG developed, is routinely employed for the purpose of obtaining the effect of control rod moment on core power distribution and determination of criticality. There can be many ways in which control rods could be withdrawn. To consider all of them is manually difficult and not possible. Therefore optimal control rod withdrawal sequence is obtained by well-known optimization method of Adaptive Simulated Annealing scheme developed for this purpose. In this scheme, random control rod patterns are generated as test cases and given as input to COMETG. The output of COMETG includes core operating state, thermal limits and 3-D core power distribution. The core power distribution thus obtained is compared with the desired Haling power distribution (power distribution at EOC) with the

objective to minimize the difference between them. This code is made operational in parallel computing environment for obtaining quicker results.

## The power escalation program requirements

TAPS follows hybrid PCIOMR guidelines for power raise after each BOC / Mid cycle sequence change startup. These guidelines are designed in such a way that, during the power raise operation, the fuel bundles would receive lesser thermal shock. This procedure normally takes 13-14 days to reach reactor power level to full power. The figure gives the graphical representation for the typical power escalation program. In order to sail smoothly through the power escalation program, guideline partial power patterns at 90 MW(e)



(point 3), 140 MW(e) (point 4), at 150 MW(e) (point 4) and 160 MW(e) (point 6 : Full Power) are determined using above optimized code. Accordingly TAPS chalk out their programme for the power escalation. The off gas activity trends of the past cycles are shown for different cycles as point values of BOC and EOC for unit number 2.



Le	gends
Un	it 2: Cycle 18,
Un	it 1 : Cycle 19
Un	it 2 : Cycle 19 and in further cycles

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#### The burn-up compensation requirements

The off gas activities are also shown in details for both units during entire cycle operation.





## 5.2 TAPS CORE SHROUD ACOUSTIC LOAD EVALUATION AND TRANSIENT DYNAMIC RESPONSE

Potential safety concerns have been raised by regulatory bodies regarding the 360 degrees circumferential separation of TAPS-BWR core shroud following LOCA. Material degradation accelerated by crevices, residual stress, cold work, sensitisation, and corrosive environment are detrimental for impulsive acoustic load due to pipe break. This might either prevent full insertion of the control rods or open a gap in the shroud large enough to preclude adequate core cooling. Validation of in-house 3D finite element code FLUSHEL for the coupled fluid-structure interaction transient analysis of light water reactor components in case of sub cooled and saturated blowdown accidents has been made with the simulation of German HDR (Heiss-Dampf Reaktor) v.32 LOCA experiment on a full scale PWR model for single and two phase blowdown problems. Implementation of unified sub cooled and saturated critical flow models, non-equilibrium effects due to flashing for the rarefaction wave propagation have been made in the code. Acoustic load evaluation and structural safety assessment of core shroud of TAPS-BWR - postulated Recirculation Line Break (RLB) was subsequently carried out with this code.





Comparison of Acoustic Pressure Time History and % Error with Experiment for 2 phase blow down within down comer of German HDR experiment.



Acoustic Pressure Time History in TAPS Down comer



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## 6. REACTOR TECHNOLOGY: KUDANKULAM 1000 MWe VVER

## INTRODUCTION

Considering the growing energy demands and the necessity to increase the energy potential, a second line of light water reactors has been added to the current indigenous programme of Pressurised Heavy Water Reactors. Two Light Water Reactors of 1000 MWe VVER units are being installed at Kudankulam in collaboration with the Russian Federation. These reactors in addition to accelerating the nuclear energy potential would also help in expanding the knowledge pool by broadening the research activities in reactor technology. This chapter on Kudankulam VVERs, highlights the recent work in the areas of reactor analysis, code development with visual interfaces for physics computation and pin-by-pin simulation of hexagonal lattice cores. V V E R is an acronym for "Voda Voda Energo Reactor" meaning water-cooled, water moderated energy reactor. The VVER reactors belong to the family of the Pressurised Water Reactors (PWRs). The KK-VVER has a three-year fuel cycle. This reactor requires annual refueling of one third of the core i.e., approximately 55 fuel assemblies.

The reactor plant consists of four circulating loops and

a pressurising system connected to the reactor with each loop containing a horizontal steam generator, a main circulating pump and passive part of emergency core cooling system (accumulators). The loops are connected with the reactor pressure vessel assembly by interconnected piping. The reactor also consists of a reactor protection and regulation system, engineered safety features, auxiliary system, fuel handling and storage system.



Reactor Thermal Power	3000 MW
Electrical.	1000 MWe
Number of circulating loops.	4
Working Pressure in Primary Circuit	15.7 MPa
Rated Coolant Temperature	
at Reactor Inlet	291° C
at Reactor Outlet	321° C
Coolant Flow Rate through reactor	86.000 m <sup>3</sup> /h
Reactor Pressure Vessel	SS clad low alloy steel.
Diameter (inside)	4134 mm
Total height	11185 mm
Number Hexagonal Fuel Assemblies	163
Reactor internals. (Core barrel, Core baffle), and Protective tube assembly).	Austenitic SS
Nos. of control rods	121
Life time	40 years
Containment	Double with primary steel lined.
Turbo-Generator,	1000 MWe (3000 rpm)
Specification	Magnitude
Number of fuel elements in fuel assembly	311
Spacing of fuel elements	12.75 mm
Number of non-fuel tubes in fuel assembly	20
No of absorber rods	18
Number of spacer grids in fuel assembly within the core	15
Material of non-fuel tubes for absorber	Zr + 1% Nb
Material of central tube of fuel assembly	Zr + 1% Nb
Material of burnable absorber rod	CrB2 in <u>Aluminium</u> matrix with boron content of 0.20, 0.036, 0.05 g/cm <sup>3</sup> .
Number of fuel assemblies	163
Number of fuel assemblies with BAR in	
1st cycle	42
Stationary Cycle	18
Number of fuel assemblies containing absorbing rods of the control and protection system. 1 <sup>st</sup> cycle	
Stationary cycle	85 103

#### 6.1 VVER-1000 MWe REACTOR ANALYSIS

The VVER-1000 MWe reactor core of Kudankulam (KK) Project is a Pressurized Water Reactor (PWR) of Russian design. It is necessary to develop indigenous capability of in-core fuel management of these reactors. This capability is also essential for an in-depth review of the PSAR documents submitted by Russian Federation for KK Project. The detailed analysis and comparison of results with the Russian design project reports giving the physical characteristics under various steady state conditions has revealed that it is essential to develop capability for analysing some of the slow (xenon) and fast transients.

Indigenous lattice burnup code EXCEL and core diffusion analysis codes TRIHEX-FA and pin-by-pin simulation code HEXPIN have been developed and are used to analyze the KK core. After generating the complete lattice database with EXCEL code for 11 fuel types, the VVER-1000 Mwe reactor core of KK Project was followed up for 8 fuel cycles. Each hexagonal assembly cell was divided into 54 triangular meshes. The results like critical soluble boron, radial and axial power distribution, 2-D and 3-D peaking factors were compared with Russian data. The calculated critical boron with a uniform  $k_{eff}$  normalization agreed well with Russian data for all eight fuel cycles. The deviation was slightly more for first fuel cycle, possibly due to non-equilibrium Sm load. Power dependent feedback is being implemented in TRIHEX-FA code to reduce the deviations observed in power distribution. The modeling of reflector region is also being fine tuned.

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Typical Comparison of Radial Power and Burnup Distribution Core Follow-up Simulation for KK Cycle-8

## 6.2 DEVELOPMENT OF CODE SYSTEM FOR THERMAL REACTOR DESIGN WITH VISUAL AID SOFTWARE PACKAGES 'VISWAM'-A COMPUTER CODE PACKAGE FOR THERMAL REACTOR PHYSICS COMPUTATIONS

The nuclear cross section data and reactor physics design methods developed over the past three decades have attained a high degree of reliability for thermal power reactor design and analysis. This is borne out from the analysis of physics commissioning experiments and several reactoryears of operational experience of two types of Indian thermal power reactors, viz. BWR and PHWR. Our computational tools were also developed and tested against a large number of IAEA CRP benchmarks on in-core fuel management code package validation for the modern BWR, PWR, VVER and PHWR. Though the computational algorithms are well tested, their mode of use has remained rather obsolete since the codes were developed when the modern high-speed large memory computers were not available. The use of Fortran language limits their potential use for varied applications. A specific Visual Interface Software as the Work Aid support for effective Man-Machine interface (VISWAM) is being developed. The VISWAM package when fully developed and tested will enable handling of the input description of complex fuel assembly and the reactor core geometry with immaculate ease. Selective display of the three dimensional distribution of multi-group fluxes, power distribution and hot spots will provide a good insight into the analysis and also enable intercomparison of different nuclear datasets and methods. Since the new package will be user-friendly, training of requisite human resource for the expanding Indian nuclear power programme will be rendered easier and the gap between an expert and any new entrant will be greatly reduced.



VISWAM Code Package for Reactor Physics & Shielding Computations



VISWAM: Present Status of Code development

Typical viewing of multigroup cross section by the 'XnWlup' code is shown in figure. One of the combo boxes for creating EXCEL input file is also shown. There are separate boxes for entering/editing the input set for pincell, supercell, assembly diffusion module etc.





The visual mode of creating input to typical VVER fuel assembly description is also given. The visual input can be internally transferred to the other digital form of input.


The 3-D flux profile of the complex core simulation can be viewed for any axial plane and each energy group. Figures give typical 3D flux display at selected plane for different types of reactor analysis. These were plotted using the program 'RealPlot3D' or the 'Display' program. These 3D plots give the clear depiction of flux profiles in large 3D cores. Inadvertent input error, if any, can easily be identified and corrected.



Typical Thermal Flux Profile in a Thorium Loaded Reactor – by Plot3D



Typical Epithermal Flux Profile in TAPS BWR – by Plot3D



Typical Thermal Flux Profile in a LWR

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## 6.3 HEXPIN CODE FOR PIN BY PIN SIMULATION OF HEXAGONAL LATTICE CORES

The code 'HEXPIN' has been developed for core follow-up analysis for first time with a pin-by-pin cell description of the entire core and reflector regions up to pressure vessel. The input to HEXPIN code consists only of fuel assembly type disposition. The geometrical specifications within each fuel type are directly derived from the output of hexagonal lattice cell burnup code EXCEL. The core external regions are alternate ring layers of steel and water up to pressure vessel. The hexagonal cells within a given radius are automatically identified by the code. The HEXPIN code has been used for small PWR cores with 7/13/19 assemblies and also for the VVER-1000 MWe reactor core of KK Project with 163 assemblies. With HEXPIN code the deviation in power distribution is within 2% of the Russian values in core interior regions.





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## 7. OPERATING RESEARCH REACTORS AT TROMBAY

## INTRODUCTION

BARC is currently operating three research reactors namely Apsara, CIRUS and Dhruva at Trombay. These reactors serve as basic research platforms for nuclear scientists and engineers. Apsara is the first research reactor built in India and commissioned in 1956 ushering in the nuclear era of the country. It is a 1MWt pool type reactor using high-enriched uranium as fuel with light water as coolant and moderator. This reactor is extensively used for basic research, radioisotope production, neutron radiography, detector testing and shielding experiments. CIRUS is a 40 MWt vertical tank type research reactor, commissioned in 1960. It has been serving in the frontier research areas of materials testing and neutron beam research. After satisfactory operation for nearly four decades, Cirus was refurbished and put back in service in October 2002. The reactor power has been raised in steps to the design power level and production of radioisotopes & experimental irradiations have been resumed. In the year 1985, indigeniously designed & built 100 MWt. tank type reaserch reactor DHRUVA was commissioned. Dhruva is a high power research reactor with a maximum flux level of 1.8 x 10<sup>14</sup> n/cm<sup>2</sup>/s, heavy water cooled with natural uranium as fuel. The reactor system is adaptable to varying core configurations due to provisions for interchangeability of fuel, shut off rods and other in-pile irradiation assemblies. This reactor has been extensively utilised for neutron beam research, materials analysis, activation analysis, solid-state physics and studies on magnetic materials.

This chapter highlights the shielding experiments in APSARA, ageing management activities in CIRUS and installation of 2-MW inpile loop in Dhruva, carried out recently.

#### 7.1 APSARA

Apsara has been extensively used for basic research as well as in areas like beam tube research, neutron radiography, forensic science, shielding experiments, testing of reactor components & neutron detectors and trace element analysis. The average availability factor of the reactor is more than 80%.

## Prototype Fast Breeder Reactor (PFBR) bulk shielding experiments

Ten sets of shielding experiments have been carried out in three phases, with mock-up radial and axial shield configurations of PFBR shielding. The data collected has been used by the designers for understanding the uncertainties in the calculation methods and neutron cross section data set used in the PFBR shielding design. This has also provided bias factors for detailed design calculations and optimization of the shield design.

The shield models consisted of combination of shield materials of sodium, steel, boron carbide and graphite for studying neutron

attenuation in the axial and radial configuration of the proposed PFBR design. Shielding experiments involved the following major activities:

- Design, fabrication and installation of an assembly to enhance the neutron flux of desired neutron spectrum in the shielding corner cavity.
- Design, fabrication and installation of bulk and additional shielding assemblies in and around the shielding corner.
- Modification of the shield model trolley.
- Design of trolley for installation and handling of Uranium converter assemblies.
- Modification of handling tools for the converter and irradiation foils.

Prior to the experiments core re-shuffling and required mock-ups at low power operation were carried out. The experiments were successfully carried out with simulated shield models.



## Advanced Heavy Water Reactor (AHWR) shielding experiments

Shielding experiments were carried out with concrete models inside the shielding corner cave, by operating reactor in [C-dash] position at a power level of 40 kW. The converter assemblies used for the PFBR shielding experiments were removed to provide the required thermal neutron spectrum. Gold and Indium foils with and without Cd cover were irradiated at four designated locations for flux measurements. The experimental data collected would provide valuable information for AHWR shielding design.

## Reactivity worth estimation of Burnable Poison Rods

Small size pins of two types, containing Vibro-compacted powder of Gadolinium-Aluminate with different Gadolinium concentration (Burnable Poison Rods) were tested for estimation of their relative worths and uniform distribution of absorber material. Relative worth measurements were done by sub-critical method in Apsara core central location D-4 by placing a hollow Graphite bock in place of the solid Graphite rod. Reactor was made sub-critical by 1 mK by partial unloading of the core and the testing was done in two parts (straight and inverted) covering the full length of the BPR which was more than the fuel height. After initial reactivity calibration of 9 BPRs by sub-critical measurements, 30 more pins of two types were tested for the measurement of their relative reactivity worth.

#### Neutron detector testing

Thermal column facility and core position dry tube G-7 have been extensively used for testing various types of neutron detectors like Fission Counters, LPRMs, IRMs, Self Powered Neutron Detectors and Ionization Chambers for performance evaluation. These detectors are subsequently used in nuclear power stations and other research reactors.

#### Utilization of Neutron Radiography facility

The neutron radiography facility in Apsara has been extensively used for non-destructive testing of low Z materials, light elements encased in heavy elements, nuclear fuel, reactor control assemblies, etc. by radiography to assess the integrity, homogeneity, composition and assembly defects. The facility has been recently used to study the flow transition instability in a natural circulation loop relevant to the geometry of AHWR primary system. An experimental loop was set up in reactor hall & a heater test section of dimensions  $\frac{3}{8}$ ",  $\frac{1}{4}$ ",  $\frac{3}{4}$ " and 1" were tested using the real time neutron radiography. The data obtained were compared with the output of electrical conductance probes. Experiments were conducted to generate data for (i) Bubbly flow to slug flow transition, (ii) Slug flow to annular flow transition, (iii) Void fraction & (iv) Flow pattern specific pressure drop. Measurements were carried out at different pressures ranging from 1 to 85 Bar & heater power of 1 to 10 kW.

## Experimental validation of Reactivity meter based on Kalman filtering technique

Performance of reactivity meter based on Kalman filtering technique was evaluated under different operating conditions such as steady state operation, power variations, shutdown and trip conditions. For this, signals were tapped from Log, Linear channel outputs and also signals corresponding to control rod positions. Reactivity meter could be successfully used for estimation of instantaneous reactivity in critical steady state, sub-critical steady state and in transient conditions. It could also be used to monitor the sub-criticality of the system.

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#### 7.2 CIRUS

Cirus attained its first criticality in July, 1960 and operated with an availability factor of more than 70%. The reactor has been refurbished recently with a view to achieve a trouble-free and safe life extension of 10-15 years.

After refurbishment, the reactor has been re-commissioned. A detailed review of the performance of reactor systems at the design power level has been carried out and the reactor has been made available in 2004 for research and isotope production. Production of radioisotopes has commenced and various experiments are being planned. Facilities for scientific experiments & research, engineering experiments and other neutron applications, which were removed to enable

refurbishment of the reactor, are being progressively installed back.

#### Aging Management & Refurbishing Of Cirus

Detailed ageing studies on reactor Systems, Structures and Components (SSC) were undertaken to examine in detail the technical viability of extending the life of the reactor by another 10 to 15 years. Based on the studies, refurbishing requirements



of critical components were identified and a comprehensive action plan was drawn up. The plan encompassed several safety up-grades to fulfill the current safety standards. The refurbishment outage was utilised for incorporation of safety upgrades.

## Repairs & Seismic Retrofitting to the Ball Tank

Strengthening of central shaft and cupola joints of the emergency storage tank (Ball tank) by additional reinforcement with steel plates and epoxy grouting making it a monolithic structure to meet Safe Shutdown Earthquake (SSE) qualification criteria as per the present standards.

Rectification of seepages from a few locations on the concrete pour joints of spherical concrete surface of Ball Tank by lining of the entire internal surface exposed to water with eight layers of epoxy coating impregnated with two layers of fiber-glass cloth.



Ball Tank (Inset- seismic retrofitting)

## Refurbishing Work on Underground Primary Coolant Pipelines

Non-isolatable segments of the underground carbon steel primary coolant pipelines (four meters below ground) were pressure tested after core unloading. The leaky segments were cleaned using high-pressure water jets and reconditioned with special weather protection measures.



Wave-guides using acoustic emission technique were installed for monitoring any future leakages.

# Replacement of Failed Fuel Detection System

The new failed fuel detection system, is based on gross gamma monitoring of the sample stream tapped off from each of the 17 coolant channel outlets cross headers. A suitable detector



tank and necessary instrumentation were designed and delay coils were incorporated in the sample flow path to allow decay of short lived activity. This system is designed for improved performance and easy maintenance leading to less man-rem consumption.

## Remote Installation of Split Sealing Clamps on RV Helium Line Flange Joints

The repairing of the tongue and groove flanged joints between the reactor vessel and cover gas system piping is an involved job, due to site constraints and handling of heavy shields in high radiation fields. A technique for remote repairs of the joint was developed in-house and qualified in a mock-up facility. This included installation of split sealing clamps by maneuvering with help of nylon strings and a camera & CCTV system.



leaky flanged joints

#### Other important jobs :

- Detailed inspection and metallurgical studies undertaken towards re-qualification of reactor vessel and calandria lattice tubes for many more years of operation.
- Rectification of leak from the weld joint of coolant inlet line to the upper aluminum thermal shield by installation of hollow plug using remote handling techniques.
- Physical separation of ball tank make up pumps, a safety related equipment, to guard against common cause failures.
- Design, installation & commissioning of an improved iodine removal system with combine HEPA and activated charcoal filters in place of obsolete alkali scrubber and silver coated copper wire mesh iodine filters.
- Detailed studies and theoretical analysis towards assessment of thermal safety of the Graphite reflectors to assess the need for graphite reflector annealing to rule out the possibility of stored energy release to be beyond the cooling capability of pile block ventilation system under any postulated initiating events. Design and installation of an on line data acquisition system for continuous multipoint temperature monitoring of graphite reflector.
- In-house design and installation of an automatic ground fault detection system for the 125V DC power supply system.
- Repair/replacement of plant equipment based on performance review, availability of spares & obsolescence.
  Replacement of kilometers of old piping, cable & process

## Cirus Start up & power operation after refurbishing

During the initial stages of startup, significant mismatch between neutron power and thermal power has been observed. This was attributed to attenuation of neutron flux at the detector location due to wetness of graphite reflector. The mismatch was corrected by suitable repositioning of the ion chambers of Reactor Regulating System (RRS) and gain adjustment of amplifiers. Similar exercise was repeated at higher power levels. The reactivity anomaly gradually decreased with progressive operation of reactor at higher powers. Parametric data at 30 MW reactor operations have been reviewed thoroughly. Special attention was given to graphite temperature data. The observed graphite temperatures, thermal analysis data and the measurements of rate of release of Wigner energy on samples of graphite removed from the thermal column location have been reviewed. Accordingly, the limit on graphite temperature was raised from 130°C to 150 °C. Thereafter, the reactor power was raised in steps to design power level of 40 MW during November, 2004. Utilization of various sample irradiation facilities of the reactor has commenced. Installation of experimental set up by the researchers at the beam tube locations, which were dismantled during refurbishment, are being installed back.

## Technology demonstration of reactor waste heat utilization for sea water desalination

During the refurbishing outage, a desalination unit of 30 tonne /day capacity, based on low temperature vacuum evaporation process had been integrated with primary coolant system of the reactor towards demonstration program for utilization of waste heat from nuclear reactor. An intermediate demineralised water system has been provided to transfer heat



from the primary coolant of the reactor to the unit in order to minimize the probability of radioactive primary coolant ingress into the sea water.

The potable water produced by the unit is passed through a mixed bed ion exchange cartridge and is utilized to augment the capacity of demineralised water make up plant at Cirus.



In-situ 30 tonne/day desalination unit

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## 7.3 DHRUVA

As a national facility Dhruva continues to spearhead the neutron beam research programme. It also remains the major source of supply of radioisotopes for applications in the field of medicine, industry and agriculture. A number of research scholars from various academic institutions in the country, utilized the reactor under the aegis of Inter University Consortium for DAE facilities for basic research.

#### Utilization :

 On-power tray rod for production of I<sup>125</sup> by irradiation of natural xenon gas.

- Carrier assembly for irradiation and safe handling of gamma uncompensated ion chamber to enable its accelerated life testing in a radial beam hole.
- A pneumatic carrier facility for short time irradiation of samples for Neutron Activation Analysis (NAA). With a flux of ~5x10<sup>13</sup> n/cm<sup>2</sup>/s the detection limits have improved and scope for studying short lived isotopes has also enhanced.
- Prompt Gamma ray Neutron Activation Analysis (PGNAA) system using the guided neutron beam facility for the on-line analysis of various materials.
- A controlled temperature irradiation facility for carrying out irradiation studies on reactor structural materials at controlled elevated temperatures was designed and installed in one of the beam-holes.
- A neutron beam line shielded tunnel (100 t weight) was installed at neutron beam tube TT-1015 providing multi-ports for neutron scattering experiments. This further expanded the scope of neutron scattering experiments by students under Inter University Consortium for DAE facilities (IUC-DAE).

## Improvement in On-Line Monitoring System for Emergency Cooling System Logic circuit

#### The FIT system

Three Auxiliary Coolant Pumps (ACPs) are provided to remove the core decay heat during reactor shutdown state whenever the main coolant pumps are not in operation. Each ACP is provided with two prime movers one an electrical motor and other a water driven turbine. The operation of the ACPs is controlled by a three channel solid state ECS logic. Finite Impulse Testing (FIT) system has been provided for online testing of the ECS logic.

This system has been upgraded to a new three channel FIT system. The system is designed in a manner that the FIT can be done only for one channel at a time. The system also monitors inputs and outputs of ECS logic and generates discordance in case of any mismatch amongst the three channels. The new system also has self diagnostic features. An operator console is provided which stores the FIT test data, input/output monitoring data, self-diagnostics data and suitably displays the



Set up of Multi-port for neutron scattering experiments

results and alarms. The system identifies faults in the ECS logic up to the card level.

The three channels of FIT-ECS were integrated and tested using the simulator for more than 5000 hours. The final integrated test of all the three FIT-ECS channels as per approved test plan was completed and one unit was installed & commissioned for channel-A. The performance of the system has been satisfactory. The other two will channels be commissioned after obtaining the clearance from the safety authorities.

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Commissioned FIT-ECS Channel-A

#### Installation of 2 MW In-Pile Loop in DHRUVA

To test the 37 pin fuel bundles of 540MWe PHWRs and the AHWR fuel an in-pile loop is being installed at Dhruva. Fabrication and installation of test section for the flow test station has been completed for out of pile flow testing of fuel specimen. All the out of pile equipment & piping have been installed, tested and cold commissioned. The system is being operated bypassing the in-core test section.

## Safety analysis of 2 MW in-pile loop

This loop is an engineering test loop to carry out irradiation experiments of fuel bundles under simulated conditions of pressure and temperature of 540 MWe PHWR has been installed at the G-13 position of DHRUVA reactor core. The test loop is a high pressure (110 kg/cm<sup>2</sup>) and high temperature (275° C) system coupled to a low pressure (7 kg/cm<sup>2</sup>), low temperature (65° C) and low power reactor, through the core neutronics. The neutron flux available in the reactor core generates 1.6 MW of heat in the test fuel, which is removed by the circulating coolant. The vertical test section running through the reactor core houses three test fuel bundles. Prior to commissioning of the loop safety analysis needs have been carried out.



Installation of IPL test section at Dhruva flow test station



Set up for Rolled joint of IPL test section

The loop was modeled using two fluid thermo-hydraulic code RELAP5/MOD3.2. Numerical model was tested to calculate the steady state loop parameters and was compared with the outof-pile commissioning test data. The results were found to be in good agreement with the experiment. The following safety analyses were carried out.

- Double-ended guillotine rupture in the test section pressure tube
- Loss of secondary coolant flow i.e. loss of heat sink
- Loss of pressure control i.e. operational failure of surge tank
- Flow coast down analysis
- Loss of flow

The aforementioned accidental scenarios were considered along with various reactor trips for shut down system actuation and interlocks. For all safety analysis first reactor trip is neglected. The results of the analysis for the case of sudden guillotine rupture of the pressure tube in the test section just before the fuel bundles are presented here. Depressurization of the system followed by the rupture is observed to be dependent on the resistance of the restrictions in the flow path. The gaps get filled and pressurized first and then depressurize in stepwise manner. The transient variation in power and pressure of the system are illustrated.





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## 8. ACCELERATOR DRIVEN SYSTEMS (ADS)

## INTRODUCTION

The Accelerator Driven System (ADS) is an innovative concept of employing a sub critical reactor coupled to a high power proton beam accelerator, through a spallation neutron source. This system gains immense importance due to its ability to transmute radioactive waste, inherent safety, efficient fuel usage and its direct applicability in the large-scale thorium utilisation under the three-stage power programme. The important ongoing activities highlighted in this challenging research area are the spallation target studies, experimental programmes and thermal and structural analysis of the radio frequency quadrupoles in the high intensity proton linear accelerators.

#### 8.1 ACCELERATOR DRIVEN REACTOR SYSTEMS

The Accelerator Driven System (ADS) is a new type of reactor which produces power even though it remains sub-critical throughout its life. All operating reactors in the world are " critical" reactors - which means that the number of neutrons produced by fission is exactly balanced by the number lost by leakage and absorption by various materials in the reactor. This balance is responsible for maintaining a constant reactor power at any desired level. Sub-critical reactors produce fewer neutrons by fission than are lost by absorption and leakage, and require an external supply of neutrons to maintain a constant reactor power. This external neutron supply comes from the interaction of a high energy proton beam with a heavy atom nucleus such as lead through what is known in nuclear physics as spallation. The power level in an ADS is greater for stronger external sources and for reactors which are closer to "critical".

Such reactors were conceived by the Nobel laureate physicist, Carlo Rubbia (Report CERN/AT 95-53) and his team at CERN, among others, for power generation, but have caught the attention of the world for an equally important role - that of burning nuclear waste. It is well known that nuclear reactors generate radioactive waste which retains its radio-toxicity for millions of years and disposal of this waste has been a major source of public concern. The new reactor is designed to safely transmute the waste into stable elements or those whose radioactivity is relatively short lived, while producing useful power.

Indian interest in ADS has an additional dimension, which is related to the planned utilisation of its large thorium reserves for future nuclear energy generation. Thorium has the added advantage that it produces much less quantities of long lived radioactive wastes as compared to uranium. However, thorium by itself is not fissile and must be first converted to fissile U-233 by neutron irradiation, a process called breeding. In ADS, the accelerator delivers additional neutrons over and above those coming from fission. Moreover long term reactivity changes due to burnup are not controlled using parasitic absorber rods. The ADS is, therefore, expected to possess superior breeding characteristics as compared to critical reactors. Since ADS reactors are not required to maintain criticality, it is possible to increase burnup i.e. to extract more energy from a given mass of fuel. This effect is rather large for thorium based fuel. Being a new type of reactor, the ADS requires development of several

technologies related to high power accelerators, removal of the intense heat generated by the interaction of the high power proton beam with the target, and associated materials development.

Accurate computer simulations play a very important role in determining the performance of the ADS reactor. The studies are geared to develop accurate computer simulation codes for ADS, compile necessary nuclear data for this purpose, carry out experimental and numerical tests regarding the adequacy of the simulations and finally to use these simulations to evaluate ADS performance with regard to the design objectives. The "stateof-the-art" codes have been developed for carrying out fuel burnup simulations based on the exact Monte Carlo method and the (accurate and quicker) multigroup transport theory method for this purpose. The codes are functional for fixed fuel ADS and are being put to use for evaluating some of the interesting ideas conceived for applications of ADS. Further development of these codes is being carried out to include fueling operations (insertion, removal, or shuffling) and it is expected to be completed within a year.

A facility for carrying out experiments on the Physics of ADS is being set up at Purnima labs, BARC. A report on the experimental program has been prepared and experiments will commence once the facility becomes operational. The experiments will serve the equally important purpose of testing the simulation methods under development. Measurement of the degree of sub-criticality is one such important experiment, as monitoring this parameter for ADS will be an important safety requirement. This can be done by pulsing the accelerator or by studying tiny fluctuations in the reactor power, called "noise". A new theory of Reactor Noise in ADS has been developed and is gaining international acceptance.

The Advanced Heavy Water Reactor (AHWR) is being designed and developed at BARC for the purpose of thorium utilization. In view of the remarks on thorium utilization in ADS made earlier, the following questions assume importance. What is the reduction in the annual fuel requirement of a thorium fuelled heavy water reactor if operated in the ADS mode? Is a self sustaining cycle possible? How much extra energy can be extracted from a given mass of fuel before it is discharged? What would be the accelerator power required to drive such a reactor? Figures illustrate the use of the simulation codes for answering such questions. Figure shows the maximum multiplication factor that is possible in a heavy water reactor with thorium fuel in a once through cycle (i.e. one which does not require reprocessing of thorium for recycling in the reactor). The initial fissile seed could be natural uranium or even spent fuel from Pressurised Heavy Water Reactors (PHWRs).







The scheme allows 10% thorium (1 GWd/t is equivalent to fission of 1% fuel mass) to be burnt before the fuel is discharged. However, the energy gain is small and hence, a large fraction of the reactor power (about 30%) would have to fed back to the accelerator. If light water is used as the coolant, the multiplication factor is lower and the ADS would require greater accelerator power.

Figure shows the effect of recycling U-233 produced in the reactor after mixing it with thorium in different proportions. Since the discharged fuel contains somewhat larger U-233 than what is required for recycling at 1.25% level, it is clear that a self sustaining cycle is comfortably possible with a multiplication factor of 0.95 at a discharge burnup of 4% fuel mass. Such a scheme requires about 15% of the produced energy to be fed back to the accelerator. Though this is a small fraction the corresponding accelerator power is still rather large for a commercial sized power producing reactor. As noted before, use of light water instead of heavy water as coolant requires more accelerator power. Clearly it is desirable to have some means to effect a further reduction.

The one-way coupled fast-thermal ADS reactor conceived at BARC can be used for this purpose. This is illustrated in Figure. The neutron source produced by the interaction of the proton beam with the target is first boosted in a small fast region (surrounding the target) having Pu as fissile material and a liquid metal coolant such as lead. These neutrons then enter the main thermal reactor region where most of the power is produced. The outer region will be a heavy water moderated reactor for which the technology is well established. Such an arrangement can considerably bring down the accelerator power requirements. It has the added advantage that the inner booster region can be used for burning long lived waste produced in our first and second generation reactors based on uranium and plutonium fuels.

Many such studies are required to evolve a suitable ADS design and the associated fuel cycle strategies for thorium utilization. The R&D program of the ThPD group on ADS is geared to provide the necessary simulation tools and the theoretical direction for this activity.

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#### 8.2 SPALLATION TARGET STUDIES

One of the key components of the ADS is the spallation target. Based on neutron yield, thermal-hydraulics, radiation damage issues, liquid Lead-Bismuth-Eutectic (LBE) has been chosen as spallation target. An R&D programme has been initiated to address various technology issues. Under this programme, generated in the riser gives rise to the liquid metal circulation. The nitrogen is separated in the separator and mercury alone flows through windowless simulation region, downcomer pipe and enters the riser pipe and window simulation region. The maximum mercury flow rate of 6kg/s can be achieved in this facility.

a mercury and LBE experimental facilities are presently being set up. In view of many similarities of mercury with LBE, a mercury experimental facility is being setup.

## Mercury experimental loop

This loop is being setup primarily to study and develop diagnostics for target development. The loop consists of mixer, riser, downcomer, separator, window windowless target and simulation regions, dump tank. In addition to normal instrumentation, various special diagnostics like Ultrasonic Velocity Profile (UVP) monitor system for velocity field mapping, free surface level measurement based on laser triangulation technique, <sup>60</sup>Co based gamma-ray measurement system for void fraction distribution have been setup. Both window and windowless target flow simulation corresponding to around one-fifth the actual target geometry but without heat input is being simulated in this facility. The circulation of the liquid metal is achieved by injecting nitrogen in to the loop through mixer located above window simulation region of the riser. The two-phase that is



## Plasma torch as heat source for ADS target thermo hydraulic simulation

In accelerator driven sub critical reactors operating in window configuration, heat deposition in the window material and thermo hydraulic profiling of the liquid metal flow near the window are parameters of extreme significance. The actual experiment requires 10 GeV, 10 mA proton beams, the thermal effect could very easily be simulated by a 100 kW plasma beam tailored to deliver powers up to 1-2 kW/cm<sup>2</sup> uniformly on the entire target

window surface. A transferred arc plasma torch operating in argon up to 100 kW has been tested by delivering powers up to 1.6 kW/ cm<sup>2</sup> on a water cooled substrate of 80 mm diameter and shaped as the actual window. A rotating magnetic field has been used to increase the width of the heat flux profile to make the delivery more uniform.

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Plasma torch based ADS thermohydraulic Simulator

## 8.3 THERMAL AND STRUCTURAL ANALYSIS OF CW-RFQ FOR ACCELERATOR DRIVEN SYSTEM

A high intensity proton linear accelerator (linac) is required for typical ADS applications. It will produce a continuous wave (CW) proton beam of current of about 10 mA and energy of 1 GeV. Its low-energy section will consist of a number of components including a high intensity Radio-Frequency Quadrupole (RFQ).

The integrated thermal/structural analysis of a RFQ is crucial to its design and operation. The heat generated due to resistive losses in the copper walls of RFQ is required to be dissipated. If the RFQ is not properly cooled, this heating can cause large temperature excursions in the RFQ structure along with distortions in the vane geometry, leading to thermal strains. The thermal distortions lead to significant deformation and detuning of the structure.

The RFQ consists of four vanes and individual vanes will be made of Oxygen Free High Conductivity (OFHC) copper. A four-vane type RFQ structure and its cross section are shown in Figure.



The RFQ is 3.62 m long and it is planned to fabricate into 4 segments of 0.91 m each because single module RFQ would be very unstable due to the longitudinal Higher Order Modes (HOM) being very close in frequency to the accelerating mode frequency. Due to RF heating about 92 kW/m heat is dissipated on the inner surfaces of the RFQ and this heat is to be removed to limit thermal deformation of the structure.

The acceptance criterion is that the total frequency shift due to thermal deformation within the entire cavity should be less a  $\pm$ 50 kHz. The mechanical fabrication tolerance of the structure is very tight (1/100th of mm) from beam dynamics point of view and these tolerances have to be maintained during high power operation. A cooling scheme of 24 cooling channels using water as the coolant was designed to extract the heat.

Frequency shift depends upon the thermal deformation of the vanes and is very sensitive to radial deflection of the vane tip. Hence, temperature rise of the vane tip should be minimum. The location of the channels specially the vane channel-1 should be as close to the tip as possible for efficient removal of heat. The heat transfer can be increased by higher flow rate of cooling fluid but it is limited by material erosion. Fluid temperature should be such that the over all temperature rise and the resulting detuning should be within limit.

Analysis of the RFQ has been done in two stages. In the first stage 2-D analysis has been carried out and frequency shift has been evaluated at inlet and outlet plane for various channel inlet temperatures. For 3-D analysis, the channel inlet temperature has been chosen on the basis of the minimum average frequency shift as obtained from the 2-D analysis. After doing a number of iterations, temperature distributions, subsequent deflection and frequency shift have been obtained.

#### Results and discussion

#### (a) 2-D analysis

For a typical case (channel inlet temperature of 289 K (16 °C)), temperature contour plots are shown at inlet and exit planes maximum temperature is obtained at vane tip.

The variation of frequency shift at inlet and exit plane for respective cases are shown in the Figure. There is a difference of 42 kHz between inlet and exit plane. Average frequency shift is zero for channel inlet temperature of 16.2 °C. Frequency shift is observed to closely follow the tip deflection i.e. higher tip deflection leads to higher frequency shift.

#### (b) 3-D analysis

Fluid temperature of 16.2 °C at channel inlet has been considered for 3-D analysis which is expected to give minimum frequency shift. Temperature contour plot is shown in Figure which shows that maximum temperature is at the vane tip.

Subsequent deflection has been obtained for reference temperature of 303 K (30 °C). Deflection at the vane tip for axial



(Uz), radial (Uy) and resultant (Usum) along axial direction is shown in Figure. The maximum deflection in Y (radial) direction is 8.1 micron at one end of the RFQ.

The variation of average frequency along axial direction is shown in Figure. It shows that the frequency shift goes to 180 kHz at the ends. The overall average frequency shift is 63.755 kHz., which is more then the prescribed limit. The average frequency shift obtained by 3-D analysis is quite different than 2-D analysis for same fluid channel inlet temperature. In order to get frequency shift very close to zero, one more case has been analyzed with considering the fluid channel inlet temperature of





Temperature contour plot (inlet temperature :16.2°C)



16.5 °C for which average frequency shift is 1.736 kHz. Hence in 3-D analysis with changing fluid channel inlet temperature by 0.3 °C (16.5-16.2) the change in average frequency shift is 62 kHz. Hence to maintain average frequency below 50 kHz the fluid channel inlet temperature can be varied from 16.25 to 16.75 °C.



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## 9. NEW REACTOR CONCEPTS

## INTRODUCTION

The Indian nuclear power programme is envisaged in three stages focusing on the judicious utilisation of our fuel resources, especially the vast thorium reserves to ensure long-term energy sustenance. The multifaceted research activities have also widened ensuring synchronous advancement. In this chapter the concepts of a thorium breeder reactor and a multi purpose research reactor have been highlighted.

## 9.1 THORIUM REACTOR DESIGN WITH PuO<sub>2</sub> SEED (TWO YEAR FUEL CYCLE – MINIMUM CONTROL MANOEUVRES)

The study of 'A Thorium Breeder Reactor' (ATBR) core with  $PuO_2$  seeded core was carried out with the latest 172 group WIMS data library and PHANTOM-TRISUL code system. It was reaffirmed that the core is capable of operating at 1875 MWt or 600 MWe with no refueling up to 720 days. The burnup reactivity variation is as low as  $\pm$  5mk. The peaking factors intrinsically decrease with cycle burnup. No significant control maneuvers are needed for flux shaping and reactivity control. Some of these salient features are shown below. The void coefficient remains negative in the operating range of steam fraction and for entire lifetime.







ATBR – 600 Equilibrium Core with  $PuO_2$  Seed In ThO<sub>2</sub> - Epithermal Flux Distribution at BOC – 0 FPD



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## 9.2 MULTIPURPOSE RESEARCH REACTOR

Considering the present and future requirements of research reactor based facilities design of a new "Multi Purpose Research Reactor" (MPRR) has been undertaken. The MPRR with its high neutron flux and irradiation volume will provide the platform for research in reactor fuels, reactor materials, condensed matter research for study of structure and dynamics of materials, stress analysis of engineering components especially the reactor materials, neutron radiography, time of flight refractrometry, small sample investigations for the study of new and novel materials. The MPRR will supplement the isotope production capacity of Dhruva and Cirus research reactors to meet the projected requirements of various isotopes beyond the year 2015.

	Salient features of MPRR			
Reactor Type	Pool Type With Fixed Core			
Thermal Power	20 MW			
Maximum Neutron Flux	5.0 X 10 <sup>14</sup> n/cm <sup>2</sup> /sec (Th) 1.0 X 10 <sup>14</sup> n/Cm <sup>2</sup> /sec (Fast)			
Reactor Core	37 lattice positions > 28 standard fuel assemblies. > 6 control fuel assemblies. > 3 irradiation positions			
Lattice pitch	81.7 mm.			
Coolant/Moderator	Light Water			
Reflector	Heavy Water			
Shut Down System	Fast Acting Solid Shut-Off Rods / Reflector Dumping			
Shut-off rod material	Hafnium			
Fuel	Plate Type, LEU (19.75% w/w U235) (U <sub>3</sub> Si <sub>2</sub> dispersed in Al)			
Fuel loading density	3 gm/cc			

The proposed reactor is a 20 MW (thermal) research reactor with a maximum thermal neutron flux of 5.0 x 10<sup>14</sup> n/cm<sup>2</sup>/sec. It will be fuelled with Low Enriched Uranium (LEU) (19.75 wt % U235) dispersion type fuel (U3Si2-Al) and will use de-mineralised water as coolant and moderator. The reactor core will be surrounded by an annular heavy water reflector tank to obtain a large irradiation volume to maximize the number of irradiation positions available for isotope production and material irradiation. Most of the irradiation positions will be accommodated in the heavy water reflector tank surrounding the core. The heat from primary coolant system is ultimately dissipated to atmosphere through a cooling tower system, which acts as a secondary coolant system. The reactor block will be housed in a confinement building.

The reactor core is constituted by 37 lattice positions laid in 5 x 5 square array at a square pitch with three positions added on each side. The nominal core consists of 28 standard fuel assemblies and 6 control fuel assemblies. The central and two peripheral positions will be used for high irradiations requiring high neutron flux. Graphite/beryllium fillers occupy the space between the peripheral core lattice assemblies and the inner surface of the reflector tank. As presence of water in this region reduces the thermal neutron flux levels in the reflector region, maximum water gap between filler and reflector tank will be restricted.

The reactor block consists of two isolatable water filled bays, reactor bay and fuel storage bay. The bays are lined with SS 304L plates. For the beam tube research a separate area is provided in the reactor building. The researchers' area and the reactor pool are sized such that adequate core submergence is always assured even in the event of a rupture of beam tube.







Experimental facilities are provided both in the reactor core and reflector tank. Three lattice positions are provided for in-core irradiation of samples requiring high neutron flux. The positions can be used for material irradiation or isotope production. Seven



Experimental positions

tangential beam tubes are provided for beam tube research, neutron radiography and detector testing & calibration. A total of 15 vertical tubes of assorted sizes are provided in the reflector tank. These positions can be used for the production of isotopes, silicon doping, etc. Most of these irradiation positions are located in the region of highest neutron flux.

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## **10. NUCLEAR SAFETY**

## INTRODUCTION

This chapter deals with work done in connection with safety of reactor components. The areas addressed in this connection are postulated pressure tube failure accidents inside the calandria, analysis of aircraft impact on reactor building, fire modeling, thermal analysis of transportation cask for transportation of nuclear materials etc. Several computational codes like FLUSOL & FLUSHELL, IMPACT, PFIRE-M, FAIR, risk monitor etc., have been developed and used. Methodology and testing for qualification approval and ageing studies on hardware systems/components/materials used in nuclear instrumentation have also been dealt with.

## 10.1 FLUID-STRUCTURE INTERACTION STUDIES FOR PHWR CALANDRIA AND IN-CORE COMPONENTS

Postulated pressure tube failure accident inside the calandria is one of the important design basis accidents that needed to be addressed for ensuring the integrity of calandria and in core safety related components. Earlier studies were carried out to obtain the limiting pressure on the calandria shell and the influence of pressurization of calandria shell due to volume addition of the flashing fluid, bubble growth, shock wave propagation, bubble condensation and bubble collapse were investigated with a one dimensional model. This pioneer study helped to identify the influence of above mentioned various parameters.

For detailed investigation of the problem, transient finite element two-dimensional code FLUSOL and three-dimensional code FLUSHEL were developed to analyze this class of reactor safety problems. In these studies the influence of shock wave propagation on a local six channels model was studied with coupled fluid and shell model. This model was used to analyze the shock wave loading of the neighboring channels due to postulated failure of channel C1 shown in the figure. Fish mouth opening and double ended rupture of the channel leading to generation of loadings due to the surface wave front and line wave front in case of axial cracking of the pressure tube were simulated in this study for TAPS 3 & 4 540 MWe PHWRs. This local model included the wave reflection effects from the neighboring channels while radiation boundary was used at the moderator boundaries to avoid any spurious wave reflections.



It was concluded that the neighboring channels would meet the design requirements though the local shock pressures were higher than the calandria shell pressure as shown in the table. The predictions made from these studies have been verified with the results of the simulated pressure tube failure accident experiments carried out for PHWRs in Whiteshell laboratory.

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## 10.2 DAMAGE EVALUATION IN 540 MWE INDIAN PHWR NUCLEAR CONTAINMENT FOR AIR CRAFT IMPACT

The loading time history for Boeing and Airbus categories of aircrafts has been generated with in-house code "IMPACT" developed for soft and hard missile impacting on rigid and deformable targets for penetration and perforation simulation. Similitude relations were developed using Riera's model as reference aircraft Boeing 707-320 (with known crushing strength and mass distribution along the aircraft length) and load time histories were developed for the modern aircrafts. The transient analysis assumes the impact due to smaller domestic planes of Boeing and Airbus families with high velocity which could be possibly maneuvered at lower heights (typically ~50m for a nuclear containment) easily compared to heavier aircrafts of larger sizes, which is a realistic postulation.

Nonlinear transient dynamic analysis of 540 MWe PHWR containment structure has been carried out for Boeing 707-320 and Airbus 300B4-200 impact for cracking, crushing and rebar yielding evaluation. Computational simulation of scabbing, spalling, penetration, perforation and damage evaluation of safety structures for internal / external hard / soft missiles with multiple barriers has been illustrated. The conclusions made from the analysis are:

- Outer Containment Wall [OWC] would suffer local perforation with a peak local deformation of 117mm at 0.19 sec.
- The stress and strain values at impact location are within the limits till 0.19 seconds subsequently the load gets transferred to inner containment wall due to local perforation of outer containment wall.

- Overall integrity of OCW structure would be maintained as the global displacement at points away from impact is of the order of ~ 2 to 5 mm Inner Containment Wall [ICW].
- Local cracking and rebar yielding in ICW observed with maximum displacement of 115 mm.
- The global displacements at points away from impact

is of the order of  $\sim 5$  to 10 mm.

No perforation of inner containment wall is observed.

The conclusions made in this analysis are (with total available wall thickness of 1.36 m for combined outer and inner containment wall) similar to the recently published report by United States Nuclear Regulatory Commission (USNRC) for US NPP stations.

Aircraft model	Length of the aircraft (m)	Total Weight (Kg)	Engine Weight (Kg)	Peak Load Including Engine (MN)	Duration Of Impact (sec)	Crushed Length (m)
B 707-320	39.6	90520	14012	72.244	0.39	29.927
B 707-720	43.525	99492	-	71.66	0.41	32.142
B 727-200	46.7	95238	7258	62.945	0.42	33.073
B 737-200	30.5	53297	7258	54.996	0.30	23.001
A300B2-200	53.57	142900	23140	83.151	0.50	39.071
A300B2-100	53.57	158400	23140	92.838	0.52	40.042
A300B4-200	53.57	165900	23802.24	97.443	0.53	40.501
A*300B2-100	53.57	34585	23140	16.14	0.40	28.416
A310-202	43.9	132000	22222	95.738	0.45	34.653











Displacement at Point A of inner containment wall in Combined Model





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## 10.3 COMPUTATIONAL FLUID DYNAMIC FIRE MODELLING FOR NUCLEAR POWER PLANTS

Conventionally, the fire propagation and consequence analysis is carried out by simplified methods, flow models and zone model based codes. Availability of advanced computational machines has now facilitated the use of CFD codes for such studies. In recent years a framework has been established for modeling fire in NPPs and allied facilities which uses the CFD codes apart from the conventional codes. The Cost-Effective Simple Fire Modelling techniques (CESFM), which involved the use of simple, empirical equation have been developed and utilized for solving the international benchmark problem of emergency switch gear room of a PWR and a practical problem of diesel generator room of Madras Atomic Power Station power plant. Various zone model based codes like CFAST, BRANZFIRE and OZONE have been acquired and utilized in the studies on ventilated fires, visibility studies apart from applications in NPPs. An in-house two layer single room zone model based code PFIRE-M has also been developed. Neural network based methodology has been used to predict the sprinkler actuation time and time to flashover occurrence in event of fire. The data required for training the network was generated using the zone model based code BRANZFIRE.

An acquired CFD code FDS (Fire Dynamics Simulator), which is an advanced and dedicated large eddy simulation based code has been extensively validated by modelling natural convection, forced convection, mixed convection, tunnel fire, shopping complex fire, ventilated fire, Steckler experiment, buoyant plume, plume in presence of a cross wind and other reported experiments and benchmarked against international cable fire benchmark problem for emergency switchgear room of a PWR.

Apart from the applied validation studies the FDS has been used for basic research and separate effects studies on plume entrainment, pulsating pool fire, plume puffing, plume flow structures, flame exhaust, bi-directional flow in presence of a large opening and oscillatory flow behaviour in ceiling opening. CFD Modelling is being extensively used in solving the applied problems from nuclear industry i.e., evaluation of various possible fire fighting strategies for NPPs, a hypothetical fire near Indian PHWR containment building due to a plane crash and fire risk assessment for the Cobalt Teletherapy System.





An innovative procedure has been established wherein the input from the deterministic fire modeling in terms of fire propagation, fire detection parameter and equipment survival goes to the Probabilistic Fire modeling which in turn estimates conditional probability of damage to the safe-shutdown system (target) during a postulated fire. The fire growth time has been determined using the CFD based code for various rooms with various situations like ventilated / unventilated/ Door open/

Door Close/ Trash fire/ Power cable Fire etc. These fire growth times have been used to calculate the non suppression frequency by probabilistic modelling. Significant improvements were achieved in the PSA conclusions when aided by the CFD based deterministic analysis as compared to conventional zone model analysis.



## 10.4 THERMAL ANALYSIS OF TRANSPORTATION CASK

The transportation cask/packages are used to transport of radioactive material from one place to other. Different types of packages are used for the transport of radioactive material. Packages are classified to various categories based on activity and physical form of material contained in the package which are specified in the regulatory guides. The package should demonstrate its compliance to various tests under normal and accident conditions and to general design requirement as required by regulatory authority. The overall objectives of all these tests are to demonstrate that the loss of radioactive contents or dose to the public do not exceed the respective limits specified in the regulatory guides.

For thermal test, package should be tested on fully engulf fire for at least 30 minutes. A fire should be controlled to an extent that it sufficiently engulfs a test package and develops at least the required minimum heat flux (based on the temperature of 800°C to ambient temperature) to the package. A package may be thermally tested in a furnace if acceptable conditions in the furnace can be achieved. Alternately, the same can be achieved by carrying out requisite thermal analysis. The thermal analyses of a number of casks such as cask for transportation of Thorium fuel, marine product irradiator, exposure device etc. have been carried out.

## Spent Fuel Transportation Cask

Fire test analysis of spent fuel cask has been carried out. This cask is 2026 mm long 1176 mm width and 1346 mm in height. Due to symmetry 1/8<sup>th</sup> model has been modeled. Finite element mesh considered for this geometry is shown below.

Design basis: Type-B(M), Shape: Cuboidal, Shielding thickness (lead): 150 mm, Cavity size: 1600 mm x 750 mm x 920 mm

#### Analysis for normal condition

For this case maximum surface temperature was obtained for the cask with ambient temperature of 42 °C with a specified insulation to account for solar heat flux incident on outer surface and heat transfer coefficient. The temperature obtained was within the limit.

## Fire test analysis During fire

Fire test has been simulated by specifying boundary condition of 800 °C for 30 minutes with no solar heat flux, surface absorptivity of 0.8 and flame emissivity of 1.0. Initial condition for temperature distribution for fire test has been taken from normal condition. Suitable heat transfer coefficient on outer surface with temperature dependent material properties has been considered. Melting is modeled as enthalpy formulation. The contours plot at the end of the 30 minutes are shown in Figure. It is found that maximum temperature occurs at the corners and melt front penetrates up to about 63 % thickness at corner of lead at 30 minute fire test.

#### Post fire

Analysis has been continued after 30 minute fire until all temperature start dropping. The boundary conditions of 42 °C as ambient temperature, 0.8 as surface emissivity and suitable heat transfer coefficient were applied for post fire analysis. The melt front continues to penetrate post fire due to the stored heat. It was found that corner region completely melted after ~48 minutes approximately. The temperature variation at



the corner location of various materials across thickness is shown in figure.

Similarly, analysis has been carried out for the cask used for transportation of exposure device and marine product irradiator. A finite element mesh and contour plot for the exposure device are shown in the in figures. These analyses help in design improvement and reduce the number of tests needed for demonstration of the test compliance.

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## 10.5 FUEL MODELLING UNDER NORMAL OPERATION AND ACCIDENTAL CONDITIONS

The prediction of fuel pin behavior under different reactor transients is an important requirement for safety analysis. The fuel pin including clad forms the first barrier against release of radioactive material to public domain. Hence, the assurance of integrity of clad under all reactor scenarios will improve reactor safety considerably.

A finite element based code FAIR has been developed over the years for this purpose. This code has modeling capability of complex thermo-mechanical and chemical processes occurring in a nuclear reactor fuel pin.

Code addresses following issues related to high burn-up of fuel

- > Conductivity Degradation
- Radial Flux Redistribution
- Fission Gas Release from HBS
- > Pellet-Clad Mechanical Interaction
- High Burnup Grain Structure
- > Burnup Dependent Grain Boundary Saturation Limit
- > Burnup Dependent Mechanical Properties
- Burnup Dependent Failure Mechanism
- Fuel Matrix Saturation Limit
- He Adsorption and Release
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#### **10.6 RISK MONITOR – AN APPLICATION OF PSA**

In nuclear power plants, safety is an important issue. Probabilistic Safety Analysis study provides insights into plant processes, mechanisms and possible interaction between plant systems, both for existing plants with operating histories and for plants still in the design stage. In view of this, on line safety has received lot of attention from operation and maintenance personnel.

Risk Monitoring can be defined as being the process whereby a complex technical facility is continuously monitored as regards the functioning or non-functioning of its different subsystems and the risk emanating from the facility is evaluated on the basis of this information. In the widest sense it can be regarded as being part of the risk management of a plant. Risk Monitoring provides safety status information for a plant and thus aids in decision making about whether continued plant operation is tolerable under certain system function outages. It may also support operations and be of help in deciding on maintenance strategies allowing immediate assessment of different plant configurations.

Risk Monitor, a PC based software which can assess the risk profile has been developed. This software can be used to optimise the operation in Nuclear Power Plants with respect to risk over the operating time. Risk Monitor is user friendly and can reevaluate Core Damage Frequency (CDF) for changes in component status, test interval, initiating event frequency etc. Plant restoration advice, when the plant is in high risk configuration, current status of all plant equipment and equipment maintenance prioritization are also provided in the package. Using this software, 'What-If' analysis can also be done.

## Software Developmental Aspects of Risk Monitor

The Software has been developed in Visual Basic. The various modules developed in the package are as follows and main screen is shown in the following figure.

- a) System Modelling Options
- b) Main Summary & On-Line Risk
- c) Component data base
- d) Component Out-of-Service & Restore
- e) What-If Analysis



#### Applications

Various applications of Risk Monitor software are given below:

#### Decision-making in operations

If CDF value exceeds the prescribed probabilistic safety criterion, efforts should always be made to lower the CDF through different tests and maintenance policies.

#### Maintenance strategies

Risk Achievement Worth (RAW), which is the ratio of risk when a component is down to the nominal risk, is the well suited indicator for deciding maintenance policies. Maintenance actions need to be planned according to the order in which RAWs of components are ranked, i.e. components having higher RAWs need to be maintained with higher priority. Similarly, component having higher Risk Reduction Worth (RRW), which is defined as ratio of nominal risk to the risk when a component is completely available, should be given attention from the design point of view, since these can enhance the reliability of the system. This type of decision is less sensitive to the absolute values of the component failure parameters; however, the relative values of failure parameters influence the values of RAW and RRW.

#### **Risk Informed In-Service Inspection**

The Risk Informed In-Service Inspection (RI-ISI) program aims at integrating traditional engineering evaluations with insights gained from PSA. The prime use of PSA is to obtain an estimate of risk and relegate it to various systems and down to components to obtain an idea of their importance in terms of contribution to the risk. Risk Monitor can be effectively employed for analysing the change in CDF whenever there is a change in Inspection plans and thereby analyse for an optimum scheduling plan. Risk importance measures like Fussel-Vesely and Birnbaum Importance are evaluated for various components and systems in the Risk Monitor for risk informed inspection planning.

#### **Review of Technical Specifications**

Technical Specifications are usually based on deterministic assessment and engineering judgement. Technical specifications based on probabilistic considerations can be evolved to optimise the Allowable Outage Times (AOT) and Surveillance Test Intervals (STI) for various Systems and components.

#### **Emergency Operating Procedures and Risk Management**

The Emergency Operating Procedures (EOPs) are generally based on the considerations of failures in process systems. From the event tree relevant to a particular process system failure, safety systems can be identified and their availability can be ensured so as to maintain the plant in a safe domain. EOPs for the operation of certain safety systems (e.g. fire water injection to steam generator in station blackout situation) based on dominating accident sequences as identified in PSA can be effectively used in risk management.

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## 10.7 LOCA QUALIFICATION AND THERMAL AND RADIATION AGEING STUDIES OF THE COMPONENTS USED IN NUCLEAR INSTRUMENTATION

#### Introduction

Components in various systems in Nuclear Power Plants may be subjected to harsh environmental conditions like high humidity, temperature and radiation during the normal operation as well as during the accident condition such as LOCA. Hence, it is essential to ensure reliable operation of these components during the above conditions. Towards this objective, qualification approval and ageing studies on hardware systems/components/ materials, to provide reasonable assurance regarding their survival capability under simulated environment even at the end of specified service life, is needed. Facilities like PANBIT and LOCA simulator have been set up within BARC. A similar facility to test bigger components like pump motors, motorised valves, etc has been setup at Electrical Research and Development Association (ERDA), Vadodara.

Technical services are being regularly provided to upcoming and operating Nuclear Power Stations which have significantly helped in taking appropriate decisions in the areas such as (i) standardisation of new engineering hardware and their procurement, (ii) estimation of residual life, (iii) failure analysis and reliability improvement and (iv) import substitution.



Ageing research studies are being carried out in respect of electronic, electrical and process instruments. These include studies on cables and elastomeric materials etc. for various NPPs and other nuclear facilities.

#### Test and Measurement Facilities

#### Thermal ageing

Thermal chambers of various ranges and different dimensions are available for carrying out thermal ageing studies of components and equipment. Temperature range varies from room ambient to 300°C as desired. Provision for on-line performance monitoring of the items being tested has also been provided in all thermal chambers. Process air connection can also be made for the testing of process instruments.



#### LOCA environment simulator

This is a cylindrical vessel made of 6 mm thick stainless steel sheet. Internal diameter is 100 cm. and straight length is 120 cm. Maximum steam temperature and pressure achievable are 150°C and 3.4 kg(g) (50 psig) respectively. Oil free air compressor has been connected to the simulator

for performance evaluation of pneumatic devices during LOCA test. Provision has been made for recording and scanning of steam temperature, monitoring of pressure, water spray and on-line measurement of performance parameters. Two safety devices, a pressure relief valve set at 35 psig and a rupture disc with 50 psig rupture pressure, have been mounted on the simulator. Provision has also been made for manual release of steam and draining of condensate.

#### Synergism Simulator

In order to study interaction effects of combined environments, prevailing simultaneously in NPP

containment, synergism simulator has been set up in BARC in collaboration with BRIT. This facility consists of temperature humidity chamber, gamma radiation source along with a provision for applying electrical stresses. Internal dimensions of the chamber are 84 x 84 x 90 cm<sup>3</sup>. It is possible to vary magnitudes of these stresses as per design of experiment. Temperature can be varied from room ambient to 80°C with relative humidity up to  $95\pm5\%$ . However, temperature can be varied from room ambient to 150° C when used as temperature chamber alone. Dose rates can be varied from 2 to 30 krad/hr. by using 3 lead shields for the attenuation of gamma field. It is also possible to

study dose rate effects. Provision has been made for on-line measurement of performance parameters. Dose rate outside the synergism simulator (original existing PANBIT facility) can be varied from 1 to 900 krad/hr depending upon the distance of test items with respect to source.


Sr No.	Cable type	<b>IR</b> , Ω		
		Initial	During LOCA	After LOCA
1	PVC/FRLS PVC	>10 <sup>12</sup>	10 <sup>7</sup> -10 <sup>8</sup>	10 <sup>9</sup>
2	HR PVC/FRLS PVC	>10 <sup>12</sup>	10 <sup>6</sup> -10 <sup>7</sup>	10 <sup>9</sup> -10 <sup>10</sup>
3	EPR/Neoprene rubber	>10 <sup>12</sup>	10 <sup>8</sup> -10 <sup>9</sup>	10 <sup>9</sup>
4	SiR/EVA	>10 <sup>12</sup>	10 <sup>8</sup> -10 <sup>9</sup>	10 <sup>9</sup>
5	XLPE/FRLS PVC	10 <sup>11</sup>	10′	10 <sup>11</sup>
6	XLPE/PVC	> <b>10</b> <sup>12</sup>	10 <sup>7</sup>	10′
7	PE-Coaxial	>10 <sup>12</sup>	0	0

Effect of LOCA test (120°C steam) on insulation resistance (IR) of unaged cables

Sr. No.	Cable type	<b>ΙR</b> , Ω		
		Initial	During LOCA	After LOCA
1	PVC/FRLS PVC	> <b>10</b> <sup>12</sup>	10 <sup>9</sup>	> <b>10</b> <sup>12</sup>
2	HR PVC/FRLS PVC	>10 <sup>12</sup>	10′	>10 <sup>12</sup>
3	EPR/Neoprene rubber	>10 <sup>12</sup>	10 <sup>8</sup> -10 <sup>9</sup>	>10 <sup>12</sup>
4	SIR/EVA	>10 <sup>12</sup>	10 <sup>8</sup>	>10 <sup>12</sup>
5	XLPE/FRLS PVC	10 <sup>11</sup>	<10 <sup>6</sup>	<b>10</b> <sup>11</sup>
6	XLPE/PVC	>10 <sup>12</sup>	10 <sup>8</sup>	>10 <sup>12</sup>

Effect of LOCA test (120°C steam) on insulation resistance (IR) of thermally aged (100°C for 60 days) cables

Sr. No.	Cable type	<b>ΙR</b> , Ω		
		Initial	During LOCA	After LOCA
1	PVC/FRLS PVC	> <b>10</b> <sup>12</sup>	10 <sup>7</sup>	>10 <sup>12</sup>
2	HR PVC/FRLS PVC	>10 <sup>12</sup>	10 <sup>6</sup>	> <b>10</b> <sup>12</sup>
3	EPR/Neoprene rubber	>10 <sup>12</sup>	10 <sup>7</sup> -10 <sup>8</sup>	> <b>10</b> <sup>12</sup>
4	SIR/EVA	> <b>10</b> <sup>12</sup>	10 <sup>8</sup> -10 <sup>9</sup>	> <b>10</b> <sup>12</sup>
5	XLPE/FRLS PVC	<b>10</b> <sup>11</sup>	10′	10 <sup>11</sup>
6	XLPE/PVC	>10 <sup>12</sup>	<10 <sup>6</sup>	>10 <sup>12</sup>

Effect of LOCA test (120°C steam) on insulation resistance (IR) of radiation aged (50 Mrad) cables

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### 11. ENGINEERING - MATERIAL RESEARCH

#### INTRODUCTION

This chapter covers the research and development activities on the materials of importance for the development of nuclear reactor technology. These activities cover various aspects of materials research ranging from the development of processes for extraction, fabrication routes for optimisation of material properties, characterisation of materials for newer alloys by micro-structural studies etc.. Along with this, various studies related to degradations and failure mechanisms under the application environment are also being carried out to characterise as well as optimise the performance of these materials.

This chapter gives a brief description of all such activities mentioned above and provides the readers with an option to probe further into the topic of their interest by contacting respective author(s).

## 11.1 CHARACTERISATION AND DEVELOPMENT OF NEW CLADDING ALLOYS

The future PHWR reactors with higher coolant temperatures, higher burn ups and partial boiling condition operations will require development of newer cladding alloys.

# Study of the influence of controlled additions of interstitial and substitutional alloying elements for optimum microstructure and properties of the alloy

Several studies on the binary Zr-1Nb and the quaternary Zr-1Nb-1Sn-0.1Fe alloys have shown superior corrosion resistance and irradiation creep for the cladding applications in comparison to existing alloys. Based on these observations a series of Zr-Nb (Nb varying from 0.5 to 1.5) alloys with systematic addition of Sn and Fe have been prepared with varying oxygen content. Microstructural characterization of these alloys has been accomplished using optical, scanning and transmission electron microscopy.

# Study of the range of possible metallurgical treatments for chosen alloy systems

The microstructures of Zr-based alloys are generally very sensitive to thermo-mechanical treatments. The guenched and tempered microstructures of these alloys have been studied. The annealing study of a binary and quaternary alloy has been completed. The microstructures of cast specimens of the two new alloys, Zr-1Nb-0.8Sn and Zr-1Nb-1Sn, showed a needle-like Widmanstatten structure. The needles of the  $\alpha$ -phase were found randomly oriented within the grains and a second phase was seen segregated at the plate boundaries. The hot rolled microstructure showed evidences of dynamic recovery occurred during the process of deformation at a temperature above the recrystallization temperature. Presence of smaller, defect-free and equiaxed grains formed during recrystallization process were observed in TEM micrographs. However, evidences of partial recrystallization were noticed as the needle like Widmanstatten structure was retained in various parts of the samples used for metallographic examination. The process of  $\beta$  quenching

resulted in the martensitic transformation and as a result of this the  $\beta$  phase present in the specimen was transformed to martensite. The microstructure showed sharper and longer laths. Also, the stress associated with the martensitic transformation resulted in the formation of twin related laths.

The performance of these optimized alloys in terms of corrosion resistance, hydriding, mechanical properties and irradiation creep is planned to be studied.

Hydriding studies of binary Zr-Nb alloys have been completed. Hydriding studies were carried out on Zr-2.5Nb and Zr-1Nb alloys to understand the mechanism of hydride formation in dilute Zr-Nb alloys. TEM studies have shown that in both the alloys  $\delta$ -hydride is formed in the slowly cooled samples and  $\gamma$ -hydride is formed in the rapidly cooled samples. In both the alloys acicular morphology of hydride plates was predominantly observed. In case of the Zr-1Nb alloy a zigzag morphology of the  $\delta$  hydride was seen in samples containing higher concentration of hydrogen. Occasionally internally twinned hydride plates were observed only in the case of Zr-2.5 Nb. In this study it has been seen that the hydride plates disregard the  $\alpha/\alpha$  as well as  $\alpha/\beta$  interfaces. The orientation relationships between the  $\alpha$  and the  $\delta$  hydride phases is {(0001) $\alpha$ // (111) $\delta$  and  $[1120]\alpha // [110]\delta$  and between the  $\alpha$  and the  $\gamma$  hydride phases is  $\{(0001)\alpha/((111)\gamma \text{ and } [1120]\alpha/([110]\gamma)\}$ . These orientation relationships were found to be identical in both alloys. In the case of  $\alpha$  to  $\gamma$  hydride formation, it was seen that the habit plane predicted by the phenomenological theory of martensitic transformation matches exactly with the experimentally determined value.

#### Development of fabrication flow sheet

A collaborative work with NFC is being pursued where evolution of microstructure during different stages of fabrication of cladding tubes will be assessed and correlated with the properties. Based on these studies an optimized fabrication flow sheet will be developed for cladding tubes with reproducible microstructure and properties.



Microstructure of as extruded Zr-1Nb alloy



Microstructure of as extruded Zr-1Nb-1Sn-0.1Fe alloy



Microstructure of hydride in Zr-1Nb

# Texture and Microtextural study in Zr-base alloys:

#### Texture development in two phase Zr-2.5Nb alloy

Development of the deformation texture was noticeable in single-phase (hcp  $\alpha'$ ) regions, but insignificant in two-phase (hcp  $\alpha$  and bcc  $\beta$ ) regions. Taylor type deformation texture models could predict the texture changes in single-phase microstructure, but cannot explain lack of change in two-phase Zr-2.5Nb alloy.

The majority of the two-phase structure consists of Widmanstätten  $\alpha$  in a  $\beta$  matrix. Deformation in such regions was restricted to softer  $\beta$ . Uni-directional rolling aligned the  $\alpha$  plates along rolling direction and only on rolling plane, but individual  $\alpha$  plates remained nearly single crystalline. No macroscopic strain was present in  $\alpha$  plates and significant hardening of  $\beta$  was observed. In-Plane Rigid Body Rotation of  $\alpha$  plates in an apparently continuous  $\beta$  matrix possibly explains absence of quantitative texture development in rolled two-phase alloy.

#### Effect of Texture on Hydriding

Precipitation of Hydrides may lead to reduced ductility, transition from ductile to brittle fracture and premature failure due to Hydrogen Induced Delayed Cracking (HIDC). The harmful effects of these hydride precipitates depend on many factors - such as relative amount, distribution and morphology of the hydride phase. The presence or relative formation of hydrides is expected to depend strongly on the structural parameters - crystallographic texture, microstructure/ microtexture and residual stress anisotropy. The relative role of such structural parameters on the formation of hydrides is being studied. Different Zircoloy-2 samples (obtained from different stages of Zry-2 tube fabrication) with different bulk crystallographic textures and anisotropy in residual stresses were hydrided to 200 ppm of hydrogen. These formulated the materials for detailed structural characterization. The hydrides as observed in OIM (Orientation Imaging Microscopy) scans, in general, were along the grain boundaries – possibly ruling out a 'direct' role for bulk crystallographic texture. Identification of hydride forming boundaries was attempted to determine the possible role of grain boundary nature on the hydride formation. The study of the elastic stiffness from the orientation data, on the other hand, was aimed at bringing out the role of elastic deformation and to link the hydride formation with macroscopic values of anisotropic distribution in residual stresses.

# Microstructural studies of Irradiated RAPS Pressure Tube Material

It is known that microstructure controls the material properties, thus, it is important to understand the structure –property relationship during irradiation of the material. Unlike cubic material there are not many investigations available for hcp metal and alloys like zircaloys. In this study, samples from the pressure tube exposed for 6.7 effective full year from the high flux region of the RAPP-I reactor were obtained. These tubes were exposed to the total cumulative damage in the range of 2.5-3 dpa.

The microstructure of the irradiated samples was different in many respects in comparison to the microstructure of the unirradiated samples. The presence of defect structures in the form of dislocations-loops, dislocations array etc. could be seen in the irradiated sample. These loops were mostly c type loops lying in the basal plane. The refinement and alignment of the precipitates in a particular crystallographic direction were indicative of this fact that the dissolution and redistribution of the precipitates had occurred in the irradiated samples. In the present case amorphisation of any precipitate could not be confirmed. The irradiated structure did not suggest extensive recovery of the structure as reported in other investigations. The hydride needles were also observed in the irradiated samples, however, in this case the morphology of the 'a` matrix had changed. They did not show any sharp interface as normally seen in the unirradiated samples.

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#### 11.2 STUDIES ON PH 13-8 Mo STEEL

Alloy PH13-8Mo is a precipitation hardenable martensitic stainless steel, which has strength and toughness superior to 17-4PH steel and is used in various components such as 'ball-screw' and 'seal disc' of 500 MWe PHWR. The mechanical properties, e.g. strength and toughness of this alloy are derived,

to a great extent, through the development of microstructure and specifically through the precipitation of fine, coherent  $\beta$ -NiAl ordered phase on aging of solution-quenched structure at temperatures ranging from 783 K to 893 K. Studies on the microstructure of solution-quenched PH13-8Mo steel, aging kinetics of  $\beta$ -NiAl formation and the effect of grain size on the mechanical properties are conducted to find out the structure-property correlation in this steel.

The solution-quenched structure of PH13-8Mo is important, since, this structure is further aged to obtain various combinations of strength and toughness in the material. In this study solution-quenched microstructure of wrought PH13-8Mo steel was investigated by Transmission Electron Microscope (TEM) and Small-Angle Neutron Scattering (SANS). It was found that the needled shaped austenite  $(\gamma)$ , as shown in figure was retained in the quenched structure when the alloy was subjected to fast quenching after austenitizing at high temperature. The retained austenite was seen to maintain Kurdjumov-Sachs type relationship with the martensite ( $\alpha'$ ). The blocky primary carbides, as shown in figure was confirmed, with the help of Energy Dispersive Spectroscopy (EDS), to be rich in Mo and Cr. These carbides were found to contribute strongly to the SANS signal and their shape and size were modeled with the help of software figure. While presence of retained austenite softens the solution-quenched stainless steel, the primary carbide may affect the precipitation of NiAl ordered phase and hence the property of the alloy.



a) Martensite (α') and retained austenite
(γ) in the solution-quenched structure;
b) Primary Carbide



Nucleation and growth kinetics of  $\beta$ –NiAl precipitation in wrought PH13-8Mo stainless steel was evaluated using Avrami formalism. For this, different solution quenched samples were aged at 783 K, 808 K, 840 K and 868 K, respectively, for durations ranging from 180 s to 86400 (24 h). Activation energies of  $\beta$ –NiAl precipitation at 75 % and at 90 % transformation in wrought PH13-8Mo, respectively, was calculated to be 126.7 kJ/mol and 188.17 kJ/mol. This indicates that the nucleation is faster than the growth and helps to explain the highly-resistant-to coarsening nature of  $\beta$ –NiAl precipitates. The analysis further indicates nucleation of  $\beta$ –NiAl precipitates along or near dislocations, followed by needle or plate like growth helped by dislocation-aided diffusion.



To study the effect of grain size, PH 13-8 Mo stainless steel samples from three different batches of melt were subjected to solutionizing at 1223 K for 30 minutes followed by quenching. The quenched samples were aged at 783 K (950 °F). While all samples achieved specified strength, % elongation and hardness, only one sample that had smallest and uniform grain size in the longitudinal and transverse direction could achieve the specified toughness value. Composition analysis indicated that all samples conformed to that of PH 13-8 Mo stainless steel. The correlation of toughness with the grain size is shown in figure. Hence, small and uniform grain size appears to be an important factor to achieve the right combination of strength and toughness in this steel.

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### 11.3 REFRACTORY METALS AND ALLOYS FOR ADSS, AHWR AND CHTR

Refractory metals and alloys such as Nb-1Zr, Nb-1Zr-0.1C, TZM, Mo-30W, Ta-10W, etc. are capable of withstanding an aggressive environment with respect to radiation, temperature, corrosion (gaseous and liquid metal) and stress for prolonged period. These alloys are potential candidate materials for new generation reactors, viz. ADSS, CHTR and AHWR. The process adopted for preparation of these alloys consists of aluminothermic co-reduction of the mixed oxides in the presence of judicial proportion of additives to result in a consolidated alloy product well separated from the slag. As prepared alloy is further subjected to homogenization in an arc melting furnace and finally refined by electron beam melting. Charge composition (0.1kg scale) used for the preparation of a

Targeted Alloys	Charge Composition (wt%)	% Yield	Composition attained (wt%)
Mo-0.5Ti-	MoO2-72.8,	94	Mo-99.1Ti-
0.1Zr	TiO2-2.3,		0.4Zr-0.08,
(TZM)	ZrO2-0.6,		Al-0.2
Mo-30W	MoO2-54.4.	93.5	Mo-71W-
	WO2-22.3,		28.3Al-0.6
	Al-20.4,		
	CaO-2.9		

Data and results on preparation of refractory metals alloys

selected few refractory metal-based alloys and the results are summarised in table. An optical microstructure of Mo-30W alloy revealing large grains of about 200-300  $\mu$ m is presented in Figure.



Optical microstructure of Mo-30W alloy.

Refractory metals and alloys have low oxidation resistance which restricts their application at elevated temperatures. Proper oxidation resistant coatings need to be developed to circumvent this limitation. Silicide, aluminide and or alumino-silicide type of coating are reported to be well suited for high temperature applications.

Amongst a large number of coating techniques, Halide Activated Pack Cementation (HAPC) process has been taken up to form single and multilayer coatings. The HAPC method is a diffusion coating process that involves embedding the substrate into a sealed or vented refractory container with a powder mixture called the pack (Al or Si +  $NH_4F$  or  $NH_4Cl + Al_2O_3$ ),



SEM image of Al-Si coating on TZM

which is then heated at 1173-1473K in an inert surrounding atmosphere for 8-24 hours. In the process halide activator ( $NH_4F/NH_4CI$ ) decomposes at high temperature to produce volatile halide vapours of the master alloy (AI/SI) elements. The chemical potential gradient between the master alloy and the substrate drives gas phase diffusion of the metallic halides to result in surface deposition.



the coating and TZM matrix.

50 to 60  $\mu$ m thick coating could be obtained on different refractory metal-based alloys. The SEM image of coated surface of TZM alloy reveals that the coating is smooth and granular. In the microstructure of coating cross section on TZM, no cracks were observed in the coating layer. The two types of layers, present in the coating, the outer layer consists of the ternary intermetallic phase of Mo(Si, Al)<sub>2</sub> and the inner layer of MoSi<sub>2</sub> phase.





The distribution profiles of AI, Si, Mo and W at the interface between the coating and the matrix of Mo-30W alloy, indicate that the outer layer contains very high amount of aluminium with slight variation in percentage and the inner layer consists of (Mo, W)Si<sub>2</sub> type of intermetallic phase.

Coated samples were tested for their oxidation resistance by heating initially to a temperature of around 1000 K and finally to 1300 K for 24 hrs. in dry air. The samples were found to withstand tested temperature without suffering from oxidation.

Preparation of refractory lining, its regular maintenance and likely contamination of the product from the refractory lining are the limitations of refractory lined reactor used in the reduction campaign. It has been planned to replace refractory lined reactor with a refractory less reactor having provision to operate under an inert atmosphere to overcome the above cited limitations as well as to prevent pick-up of the impurities from the open atmosphere. Mixed oxide reduction campaign on kilogram scale would be initiated using open top refractory lined reactor by the beginning of 2007.

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# 11.4 DEVELOPMENT AND MICROSTRUCTURAL STUDIES OF Cu-1Wt%Cr-0.1%Wt Zr ALLOY FOR THE FIRST WALL OF TOKAMAK

Dilute Cu-Cr-Zr alloys and their minor modifications, because of their high thermal conductivity, high strength and high resistance to fatigue at 723 K, find applications in high technology areas like yet-to-be-commercialised *tokamak*. Cu-1 wt % Cr-0.1 wt%Zr is used to fabricate diverter plates and heat transfer elements and is also likely to be employed as a heat sink in the first wall of future torroids. Elsewhere in the world, this alloy has been developed and studied quite exhaustively but some aspects like orientation relationship of the nucleating precipitates with the matrix and role of zirconium in imparting fatigue resistance have remained controversial. At BARC, an attempt was made to develop and study this alloy and carry out its electron beam welding to stainless steel and its diffusion bonding to tungsten as well as stainless steel – all this with new interesting results that are described below.



Micrographs and Selected Area Electron Diffraction Patterns from Cu-1Cr-0.1Zr Alloy in the Initial Stages of Decomposition

The alloy was made by melting together small pieces of oxygen free high conductivity copper, electrolytic chromium and iodide zirconium in an induction furnace in a yttria-lined graphite crucible. High purity argon was used as a protective atmosphere in the induction unit. The study of this alloy, after proper thermo-mechanical treatments, revealed that significant gains in tensile properties accrue from small additions of chromium and zirconium. Also, two kinds of second phase particles were observed: (a) coarse particles that formed during solidification



Diffusion Bonded Tensile Assembly (LHS) and Back Scattered Electron Images of the Interfaces (RHS) Between Cu-Alloy and Tungsten (Top) Between Cu-Alloy and SS 316 (Bottom)

of the alloy and dissolved only partially during the solutionising treatment and (b) chromium rich ordered *bcc* precipitates resulting from the decomposition of the supersaturated solid solution. The precipitation was, however, seen to take place through the formation of solute rich clusters that evolve, on further aging, into metastable ordered *fcc* precipitates before attaining their equilibrium status as ordered *bcc* precipitates. A comparison of these results with those on dilute binary Cu-Cr alloys revealed that the addition of Zr to the binary alloy strongly influenced the sequence of precipitation. Also, the stacking fault energy got lowered from 0.074 J/m<sup>2</sup> for pure Cu to 0.037 J/m<sup>2</sup> for the ternary alloy. Because of this, an improvement in the fatigue resistance of the alloy results due to a change in the nature of slip from *'wavy'* in the case of pure Cu to *'planar'* in the case of the ternary alloy.

The first wall of a commercial tokamak is to consist of the plasma facing tungsten bonded to Cu-Cr-Zr alloy (as an intermediate heat sink material) and stainless steel 316, which will handle the structural loading. For this purpose, the process of diffusion bonding using nickel as an intermediate layer could obtain metallurgically sound joints between the three. The whole assembly could be bonded in a single step and the tensile strength of the assembly was found to be 325 MPa with fracture always occurring in tungsten.

Also, joining of high conductivity Cu-1wt%Cr-0.1wt%Zr alloy to the austenitic stainless steel 316 appears difficult by conventional welding techniques due to some physical and metallurgical factors. With wide differences in melting points, coefficients of thermal expansion and thermal conductivities, along with formation of hard brittle phases, leave no chance except going for low but concentrated heat input by electron beam welding. Welding carried out by selecting appropriate weld parameters and offset for 1mm thick plates revealed that complete mixing of the melts from the alloy and the steel is difficult to achieve if liquation cracking is to be avoided. However, despite the macro-inhomogeneity in the weld pool, the welds were found to be crack-free and stronger than the alloy, irrespective of the thermal history of the welds. These results are relevant for the commercial scale electron beam

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## 11.5 ZIRCONIUM ALLOYS AND MODIFIED 9Cr-1Mo FERRITIC STEEL

## Fabricability of a Modified 9Cr-1Mo Ferritic Steel

The modified 9Cr - 1Mo (T 91 grade) steel was developed over the years by steel makers with an aim to meet requirements of SG pipes and headers in boilers as a substitute to austenitic stainless steels and was found to have good void swelling resistance under fast neutron environment. Though, the available data suggests that the modified 9Cr-1Mo steel has excellent fabrication properties and adequate long term strength, it is still necessary to get a few field trials before it can be accepted as the substitute material for high temperature use. The aim of the present investigation will be to evaluate the fabricability aspects of this steel at room and elevated temperatures. This evaluation is done by estimating the mechanical properties under different microstructural conditions. Deformation behavior was studied from 77 to 1273 K under these microstructural conditions. The results were analyzed using existing deformation models and the deformation mechanism is classified in strain-rate, strain and temperature space.

#### Studies on Zirconium alloys

Two series of alloys, which have found extensive applications as in-core structural materials of nuclear reactors, are zircaloys and Zr-Nb alloys. Though both the alloys have given reasonably good performance, efforts are being made to modify composition and microstructure and to understand in-service degradation mechanisms so as to obtain enhanced performance. Studies related to safety issues of concern viz., creep deformation, deformation behavior under accidental conditions and hydrogen/hydride induced degradation of mechanical properties are being presently pursued at the Materials Science Division.

#### Deformation behavior of Zr-alloys

In Pressurized Heavy Water Reactor, Zr-2.5Nb tubes serve as miniature pressure vessels operating at about 573 K, with a coolant pressure of  $\sim$  10 MPa. In order to understand and predict the deformation behavior of this alloy under accidental

conditions, deformation behavior was characterized by uni-axial tension tests at temperatures between room temperature and 1073 K as a function of specimen orientation, test temperature and strain-rate. Results showed that both yield and ultimate tensile strengths of this alloy decreased with increasing test temperatures, with a rapid fall in strengths above a temperature of 623 K. The alloy exhibited extensive superplasticity (ductility exceeding 100 %) in the temperature range of 923 – 1073 K and under optimum conditions it exhibited a ductility value of approximately 950 %.

#### Hydride embrittlement of Zr-alloys:

The hydrogen absorbed by Zr-alloy tubes during their service precipitates as brittle hydrides once their Terminal Solid Solubility (TSS) is exceeded and results in deterioration in their mechanical properties. TSS of hydrogen in both zircaloy-2 and Zr-2.5Nb alloy was determined using dilatometry technique as a function of temperature, direction of approach of temperature and microstructural condition. Further, it was demonstrated that the tensile and fracture toughness properties of Zr-alloys, studied in the temperature range of 300 to 573 K using a direct current potential drop technique are a strong function of the orientation of the hydrides. A benignly oriented hydride precipitate, under the influence of external stress, can undergo a change to a less favorable orientation at stress levels beyond a threshold value under suitable conditions. Threshold stress for reorientation of hydrides was determined for Zr-2.5 wt. % Nb pressure tube material in the temperature of 423 – 723 K and the effect of the reoriented hydrides on the tensile properties was determined in the temperature range of 298-573 K. A theoretical model of reorientation, based on the packing density difference between the grain-boundary and the grain, was proposed to explain the basis of reorientation of hydrides in this alloy.

The dissolved hydrogen migrates up the hydrostatic stress gradient and can accelerate the fracture process (Delayed Hydride Cracking, DHC). Based on a comparative study on the DHC velocity in CWSR (Indian, Canadian) and quenched and aged (RBMK) Zr-2.5Nb pressure tube alloy, the concept of threshold hydrogen concentration for DHC initiation and propagation was proposed. The variation in the DHC velocity for initiation and propagation with temperature was computed and it was suggested that DHC couldn't take place below a certain limiting hydrogen concentration.

Another manifestation of localized embrittlement, caused by hydrogen migration down the temperature gradient, is the formation of hydride blisters at cold spots. The morphology of blisters and their microstructural details were systematically studied in a Zr-2.5Nb pressure tube material. The computed stress field around a hydride blister, grown under controlled thermal boundary conditions could explain the hydride platelet orientation in the matrix surrounding the blister. Experiments are under way to determine the cracking strength of blisters and influence of presence of blisters on mechanical properties.

# Influence of hydrogen on creep deformation in Zr-alloys

The creep rate and the applied stress can be related through empirical relations involving many microstructural parameters. The objective of this work was to bring out the importance of empirical relations in distinct stress ranges and to show the failure of simple linear extrapolation of the creep rate from one stress range to another while predicting creep rate. The creep data of Zr-2.5Nb and modified 9Cr-1Mo steel were used as illustrative examples to demonstrate the use of single specimen short term tests (at a fixed temperature) as supplementary to conventional creep tests using multi specimen long term tests. The limitations of extrapolation of creep data from one stress range or metallurgical condition to another were also demonstrated. Based on these results it was further possible to demonstrate that the creep rate increased with hydrogen addition while the stress exponent and mechanism of creep deformation remained unaffected.

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# 11.6 LOW TEMPERATURE SENSITISATION AND LOW TEMPERATURE EMBRITTLEMENT OF AUSTENITIC STAINLESS STEEL AT REACTOR OPERATING CONDITIONS: EFFECT OF RESIDUAL STRAIN

The AHWR is being designed for an operating life of 100 years with a provision for replacement of components that cannot last for that long. Major issues in determining the design life are the material selection and life prediction of stainless steel recirculation pipelines for the AHWR. Since AHWR will have water chemistry similar to BWRs, the issue of Stress Corrosion Cracking (SCC) of these components has to be addressed. Ensuring absence of sensitisation and Low Temperature Sensitisation (LTS) over the complete design life of the components will ensure freedom from early onset of IGSCC. However, there is no data to assure that LTS will not take place in 100 years of operation at around 300°C.

All the IGSCC failures in recirculation pipelines of BWRs have occurred at the heat affected regions of weldment and not elsewhere in the base material. It has been shown that there is a residual strain of 15-20% in the region between the weld fusion line and the heat-affected zone (HAZ). Since all the LTS related cracking have been in this region, it is pertinent to study the LTS behaviour of stainless steels which are in cold-worked condition. In a recent work completed last year, it has been shown that cold work has an important influence on LTS behaviour of stainless steels. Types 304/304L/304LN/316L/316LN were used for LTS studies in solution annealed/fabricated conditions with or without different degrees of cold rolling. Modes of deformation/working encountered during fabrication of stainless steel components were also investigated viz. cold

rolling, bending, machining/grinding, warm working. The Degree Of Sensitization (DOS) was measured using electrochemical potentiodynamic reactivation (EPR) technique. It was shown that cold worked materials heat-treated at 500°C for 11 days do not always result in LTS but the mechanism changes to sensitisation with newer chromium rich carbides precipitating at grain boundaries. Therefore, extrapolation of the results obtained at 500°C to the behaviour at 300°C (500°C for 11 days as equivalent to 300°C for 100 years) is not valid. It was also shown that type 304 L and 304 stainless steels have a tendency to transform into martensite upon cold rolling. The martensite phase sensitised very fast (in a few minutes) between 300 to 500°C. This changed the LTS behaviour completely from that of an all-austenitic stainless steel and is shown in Figure. Similarly, on samples of type 304 that were bent by 90° showed accelerated LTS kinetics due to retained strain in the material.

The influence of Low Temperature Sensitisation (LTS) on the Degree Of Sensitisation (DOS) in two grades of 304LN stainless steels with 0.12 and 0.15 wt.% nitrogen were evaluated. A heat treatment at 500°C for 11 days was used to simulate LTS for 100 years at 300°C. The constant strain samples in annealed, sensitised



extent of low temperature sensitization (after 500°C heat treatment for 11 days) for various austenitic stainless steels. SS 304LN1 has 0.12% nitrogen and SS 304LN2 has 0.15% nitrogen.



Complete intergranular cracking of the sensitized type 304 stainless steel in 400 hours of exposure to high purity water at 280°C. The figure shows (a) intergranular facets of the fracture surface and (b) secondary intergranular branches emanating from the main intergranular crack.

and LTS treated conditions were used in the SCC tests. In the first SCC test the specimens were exposed in a boiling solution of acidified NaCl as per G 123, ASTM. In the second test, the specimens were exposed in a recirculating loop in oxygenated water at 280°C and 8 MPa for 1000 hours. A similar test was done with high purity water (specific conductivity 0.055  $\mu\text{S/cm}).$  It was shown that 304LN with 0.15 wt.% nitrogen is more susceptible to sensitisation and LTS. For comparison, sensitised 304 samples were also exposed in the test using high purity water. While the sensitised 304 showed intergranular cracking (shown in the figure), the DOS developed in the annealed materials after the LTS treatment was not sufficient to make it susceptible to inter granular stress corrosion cracking. However, type 316LN stainless steel was found to be the most resistant grade of stainless steel to LTS. Even after 20% cold rolling, the 500°C heating for 11 days could not increase its DOS.

Complete intergranular cracking of the sensitised type 304 stainless steel in 400 hours of exposure to high purity water at 280° C. The figure shows (a) intergranular facets of the fracture surface and (b) secondary intergranular branches emanating from the main intergranular crack.

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# 11.7 STUDIES ON STRESS CORROSION CRACKING AND CORROSION FATIGUE BEHAVIOUR OF STAINLESS STEEL

Duplex Stainless Steels (DSS) exhibit higher resistance to localised corrosion as compared to the single phase austenitic stainless steel but under certain conditions it may become susceptible to hydrogen embrittlement. Susceptibility of DSS to hydrogen embrittlement was studied using the Slow Strain Rate Test



under different applied cathodic potentials showing the effect of hydrogen embrittlement and the lower figure shows the crack arrest at the austenite phase.

Technique (SSRT) technique in 0.1N sulphuric acid under cathodic charging conditions. Results show that depending on the magnitude of cathodic potentials DSS can be severely embrittled. Cracks were mostly seen to originate from the ferritic phase and were seen to terminate on certain occasions when it encountered the austenite phase as shown in the figure.

Data from SSRT tests reveal only the susceptibility to SCC qualitatively and has little importance in design or in residual life assessment. But fracture mechanics approach to corrosion yields useful results which help in determining the residual life of in-service components in various environments. Effect of hydrogen on the fracture toughness of DSS was investigated using this approach. Cathodically hydrogen precharged fatigue precracked CT specimens were used to determine the fracture toughness as per the ASTM standard E 1820. Results indicated a decrease in the initiation CTOD values and also a decrease in the crack growth resistance  $\left(\frac{dI}{da}, \frac{dS}{da}\right)$ .

Duplex stainless steels have widespread marine applications. Thus corrosion resistance in chlorides becomes critical. Corrosion fatigue tests were thus carried out at room temperature in 3.5% NaCl solutions of varying pH, for different specimen orientations – TL and LT to study this behaviour. Testing was done as per ASTM standard E 647. Results indicated an increase in the Fatigue Crack Growth Rates (FCGR) by a factor of 2-3 in low applied  $\Delta$ K regimes due to the presence of chloride ions. This indicates an appreciable effect of the chloride ions on the FCGR.



IGSCC of sensitized SS 304 in thiosulfate solution tested at a strain rate of  $10^{-6}$ /sec.

At high applied  $\Delta K$  regimes, the FCGR values didn't have any appreciable difference indicating mechanical failure to be dominant with very little influence of the environment. Electrochemical tests on solution annealed DSS indicated that no pitting occurred even at very high applied potentials. Hence pitting was ruled out as a precursor to fatigue crack growth.

Though SS 304 is a workhorse in the stainless steel family, its application in sea water is limited to a great extent due to its proneness to localized corrosion. In the sensitized condition very dilute quantities of thiosulphate are sufficient to cause SCC in presence of residual/applied strain. It is prone to SCC in the presence of very small quantities of thiosulphate which may be formed due to the presence of sulphate reducing bacteria. Tests were carried using the Slow Strain Rate Test (SSRT) technique. Results of SSRT tests at room temperature indicated that sensitization and thiosulphate concentration greatly increased the susceptibility to SCC. Intergranular SCC was observed in solutions containing as low as 5 ppm thiosulphate. Experiments were also carried out at various applied potentials and current transients were observed to occur which corresponded to crack nucleation. Results indicated that film rupture and dissolution was the most likely mechanism of SCC.

Initial studies on FCGR/SCC have been carried out at room temperature. However, most of the engineering materials used in plants experience high temperatures and pressures in various aqueous environments. FCGR data available under such conditions will be of great significance. An experimental facility to carry out fatigue/SCC studies in environments at elevated temperature and pressure is planned to be set up. Such fatigue/SCC studies can be carried out in various conditions of applied potentials and water chemistry. Various reactor components operating under such conditions will be studied in such a setup. Crack growth will be monitored using the reversible DCPD technique. Testing facility for carrying out the tests under flowing conditions is also planned to be setup to simulate the conditions experienced by the materials in actual service.

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## 11.8 FRACTURE BEHAVIOUR OF ZIRCONIUM PRESSURE TUBE ALLOYS: EFFECT OF HYDROGEN AND IMPURITIES

The fracture toughness of the fabricated cold worked or annealed pressure tube materials is influenced primarily by the microstructure, texture and flow strength, as fabricated alloys show typical ductile fracture over temperature range of interest. However, severe degradation in the fracture toughness is observed due to irradiation hardening and hydrogen/ deuterium absorption during service mainly from the corrosion processes. The effect of hydrogen on the performance of the component is reflected in the change in fracture behaviour and/ or by its role in the delayed hydride cracking behaviour. Hydrogen embrittlement in zirconium alloys depends primarily upon the amount of hydrogen present in the matrix, morphology, orientation and distribution of hydrides and the texture, strength and fracture toughness of the zirconium matrix itself.

This work, using unirradiated samples, has generated a database, which can be referred for assessing the toughness level of the pressure tube materials (Zircaloy-2 and Zr-2.5Nb) as a function of the hydrogen content. Through a systematic study, it was also possible to suggest a test procedure for generating the entire J-resistance curve from a single sample at temperatures greater than transition temperature. This minimizes the number of tests required for fracture toughness measurement on samples of irradiated pressure tubes at elevated temperatures. In the sections to follow the salient features of



J-R curves for Zircaloy-2 pressure tube alloy charged with 50 ppm of hydrogen at different test temperatures using ASTM method (open symbols) and Load Normalization method (filled symbols).

the new test method adopted for this data generation and some important experimental observations on the effect of hydrogen and impurity segregation are given.



A new methodology for characterizing fracture behaviour by load normalization method, which is based on the Key Curve approach, has been established. This method does not need any on-line crack-monitoring unit. It uses single specimen load – Load Line Displacement (LLD) record for the evaluation of





J-R curve. This new technique has been used to characterize fracture behaviour of both Zircaloy-2 and Zr-2.5Nb pressure tube materials over a range of temperature and hydrogen content. This method compares well with the conventional ASTM test methods. Figure compares J-R curves obtained by this





technique with that obtained from conventional ASTM test method especially when small amount of plastic deformation precedes the fracture process.

Hydride embrittlement studies in the Zr-2.5Nb pressure tube have been carried out over a range of temperature and hydrogen content ranging from 25 ppm to 60 ppm, as fabricated pressure tubes obtained from double melted (DM) ingot, when charged with different levels of hydrogen show sharp reduction in fracture toughness values at ambient temperature. It is seen that the fracture



Fracture surface of quadruple melted Zr-2.5Nb alloy showing fully ductile fracture with large shear areas between dimples.



Fracture surface of double melted Zr-2.5Nb alloy showing fissure like features and shallow dimples.

toughness of hydrided material slowly increases with increase in the test temperature up to 150°C and above 150 to 170°C a sharp increase in the toughness value is observed. The Quadruple Melted (QM) material containing lower amount of chlorine showed better fracture properties than DM material.

It is seen that the fracture toughness of Zr-2.5Nb pressure tube material can be improved substantially by reducing the level of trace element impurities like carbon, phosphorus and chlorine. Chlorine content reduces from 2.5-5 ppm in DM condition to around 0.5 ppm in QM condition. It has been demonstrated at Nuclear Fuel Complex, Hyderabad that such a purity can be routinely achieved by guadruple vacuum arc remelting practice. Fractographic investigation revealed the presence of linear low energy fracture channels and aligned second phase precipitates in double melted alloys (Cl  $\approx$  2.5 ppm), which were absent in quadruple melted alloys (Cl ≈0.5 ppm). These results show that control of trace impurities is essential as it results in substantial increase in the crack growth toughness (dJ/da values) coupled with improved initiation toughness and reduced scatter in toughness levels. An additional benefit from guadruple melting is reduced hydrogen level as in fabricated tubes, which come down to 4 to 5 ppm in quadruple melted tubes from the 10 to 15 ppm level in the double melted alloy. Considering the critical role of hydrogen in pressure tube material, this may give a substantial increase in the pressure tube service life.

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## 11.9 RATCHETING BEHAVIOUR OF PRIMARY HEAT TRANSPORT PIPING STEELS

Ratcheting has been defined as accumulation of strain due to asymmetrical stress cycling in materials. The phenomenon is potentially dangerous, because for a suitable combination of mean and cyclic loads, there can be continuous cycle-by-cycle plastic deformation. This contributes to the development of unacceptably large strains in the component. PHWR and AHWR have extensive piping layout to provide for heat transport. Ratcheting experiments carried out on piping components used in Pressurized Heavy Water Reactor (PHWR) have shown that during seismic excitation the cross section of a pipe subjected to internal pressure and cyclic bending undergoes progressive ovalization. There is, therefore, a need to understand and characterize the ratcheting behaviour in these piping steels. In this study the uniaxial cyclic deformation experiments have been carried out on SA 333 Gr 6. It is a low carbon steel used as primary heat transport piping and forged header material in PHWRs. This study is a part of an ongoing comprehensive program for characterization of the ratcheting behaviour of various piping steels used in PHWR and AHWR.

The ratcheting tests were carried out at  $\sigma_{_{max}}$  = 350 MPa with five different stress ratios ( $R_{\sigma} = (\sigma_{min}/\sigma_{max})$ ;  $R_{\sigma} = 0$ , -0.25, -0.5, -0.75, -1) and five levels of stress rate (= 5, 10, 100, 1000, 10000 MPa/s). The stress controlled tests, have been carried out by varying  $R_{\sigma}$ . Figure shows the combinations of  $R_{\sigma}$  and which have shown ratcheting (represented as filled points) as well as those that have not displayed accumulation of strain (represented as open points). The trends of the stress – strain hysteresis loops have also been shown for comparison in the case of a few points in the ratcheting domain. The plot shows that at  $R_{r} = 0$ and -0.25 there has been no observed ratcheting. However, for the stress rates between 5-1000 MPa/s, employed for the tests at  $R_{-} = -0.5$  and -0.75 there has been significant ratcheting evident by the accumulation of strain. This suggests that in the present quasi-reversed tests, stress ratio has to be sufficiently negative for ratcheting to occur. Comparing the hysteresis curves in terms of the  $\sigma$ - $\epsilon$  plots for stress ratios R<sub>g</sub> = -0.5 and -0.75, at slow and fast stress rates it has been found that at stress rate of 1000 MPa/s the hysteresis loop that develops during ratcheting is smaller than in the case of stress rate of 10 MPa/s (inset C and D in Figure). Further it has been seen from the evolution of hysterisis loops within the ratcheting domain that as the stress

ratio becomes more negative the hysteresis loop opens up during ratcheting. This suggests that a negative R<sub>2</sub> ratio promotes the development of a hysteresis loop during ratcheting. The plot also shows that the accumulation of strain in the first few cycles is more rapid at slow loading conditions than at a stress rate of 100 MPa/s and 1000 MPa/s. This is more clearly seen in figure, where the effect of two stress ratios on the evolution of strain has been shown. It is clearly seen that at the slow stress rates (5 and 10 MPa/s) the strain accumulation is much faster as compared to stress rates of 100 and 1000 MPa/s. Further at these faster stress rates there is an incubation period before the ratcheting commences. The conditions observed for the two types of accumulation, one with an incubation period and one without have been shown in figure, as filled diamond and circle symbols respectively. The appearance of incubation period is also seen for  $R_{2} = -1$ , where there is no mean stress. It is also seen from figure that at stress rate of 10000 MPa/s there was no ratcheting for both the asymmetrical and symmetrical stress cycling. The ratcheting boundary has been defined on the basis that no significant strain accumulation has been observed for 10<sup>3</sup> cycles.

In the present tests it has been observed that the difference in the ratcheting rates due to a change in  $R_{\sigma}$  from -0.5 to -0.75 increases as the applied stress rate decreases. This may be due to an interaction between stress rate and stress ratio in the slower stress rate regime where there is a possibility of rapid dislocation multiplication. At a combination of low stress rate and negative stress ratio the significant increase in strain accumulation rate may be due to dynamic Lüders band formation and larger dislocation annihilation rates. This leads to a synergistic interaction between stress rate and stress ratio, which may accelerate the onset and increase the extent of cyclic softening in the ratcheting tests.



The plot also shows insets for the evolution of hysterisis loops for  $R_{\sigma} = -0.5 \& -0.75$  at (A) stress rate = 100 MPa/s (B) stress rate = 10 MPa/s and the hystesrsis loop obtained at  $e_{pl} \approx 0.38$  for (C) stress rate = 10 MPa/s and (D) stress rate = 1000 MPa/s.



## 11.10 CORROSION BEHAVIOUR OF COLD WORKED ALLOY 800 IN 673 K STEAM

Cold worked (~ 50 %) Alloy 800 containing a small volume fraction of hexagonal  $\varepsilon$ -martensite in the austenite matrix was exposed in 673 K steam of initial pH either in the nearly neutral or alkaline region for a period of 264 hours. The alloy indicated very low corrosion rates. Scanning Electron Microscope (SEM) studies of the alloy exposed in 673 K steam of initial pH in the nearly neutral region revealed a few small oxide particles whereas, a number of small oxide particles was noticed on the surface exposed in steam of initial pH in the alkaline region. Energy



Hexagonal  $\epsilon$ -martensite in cold-worked Alloy 8001

dispersive X-ray analyses indicated an enrichment of Fe, followed by Ni and Cr on the surfaces exposed in steam similar to bulk composition of the alloy. X-ray Photoelecron Spectroscopy (XPS) studies revealed that the surface films formed on the alloy after an exposure in 673 K steam of initial pH in the nearly neutral region contained mixed oxides of chromium, iron and nickel along with their elemental forms whereas, the presence of chromium oxide could not be detected in the surface films that formed in 673 K steam of initial pH in the alkaline region.



Small oxide particles in the alloy exposed in steam of initial pH 10. (A large, faceted TiN particle is visible.)



chromium for the sample of cold worked Alloy 800 exposed in steam of initial pH 6.8.



XPS spectrum recorded for (a) iron and (b) oxygen for the sample of cold worked Alloy 800 exposed in steam of initial pH 6.8



# 11.11 MAKING STAINLESS STEEL MORE RESISTANT TO IRRADIATION DAMAGE (RADIATION-INDUCED SEGREGATION)

Austenitic stainless steels are used extensively in the core of the nuclear power reactors. The main cause of failure of in-core components in light water reactors is Irradiation Assisted Stress Corrosion Cracking (IASCC) in oxygenated, high temperature (at  $\sim 290^{\circ}$  C), high purity water. IASCC is the life limiting degradation mode for all the in-core components made of austenitic stainless steels and it has been attributed to the formation of a narrow but deep chromium depletion zone at the grain boundaries due to irradiation without the formation of chromium carbides. Neutron irradiation of energy higher than  $\sim$  1 MeV in nuclear reactors results in variation of microstructure and microchemistry in austenitic stainless steels. Variation of microchemistry in austenitic stainless steels along grain boundary is termed as Radiation Induced Segregation (RIS) that results in Cr depletion at grain boundaries and segregation of Ni, S, and P. IASCC is promoted in austenitic stainless steels when a threshold fluence is reached - cracking was observed in Boiling Water Reactor (BWR) oxygenated water at a fluence above 5  $x10^{20}$  n/cm<sup>2</sup> (E > 1 MeV) that corresponds to  $\sim 0.7$  dpa (displacement per atom) for BWR and  $\sim 1$  to  $2 \times 10^{21}$  n/cm<sup>2</sup> (E > 1 MeV) for Pressurized Water Reactor (PWR). Any enhancement of the resistance of austenitic stainless steel to RIS and IASCC will be highly useful in designing and fabricating nuclear power reactors with a long service life. This ongoing study is aimed at three main strategies for improving the resistance of austenitic stainless steels to RIS.

The first approach is to alloy the stainless steels with the dilute quantities of oversized elements such as Pt, Hf, Ce, Zr, or Nb. Austenitic stainless steels 304 and 316L with the different addition of Ce (0.01 to 0.1 wt%) have been obtained to study the effect of oversized solute addition on the RIS. The irradiation work and characterisation of chromium depletion at grain boundary as a result of irradiation by Electrochemical Potentiodynamic Reactivation (EPR) test on unalloyed stainless steel has been started. The second approach is to modify the structure at the grain boundaries (to increase the fraction of the special grain boundaries) so as to reduce the tendency for the segregation/ depletion of elements at the grain boundaries. In this approach, higher fraction of special boundaries ( $\Sigma < 29$ ) will be introduced in the material using cold rolling procedure and solution annealing. This will be subsequently subjected to proton irradiation and characterization of RIS. The characterisation of each type of grain boundary will be done using Orientation Imaging Microscopy (OIM). The third approach is to create a presegregation of Cr at the grain boundaries in the solution-annealed condition of austenitic stainless steels using different temperature and cooling rates during solution annealing.





As received SS 304 sample with DL-EPR value of 0.05 and the sample of SS 304 irradiated at 4.2 x 10<sup>16</sup> protons/cm<sup>2</sup> of energy 4 MeV (0.2 dpa) at FOTIA showing radiation induced segregation (RIS) and a DL-EPR value of 0.25.

In this just initiated study, accelerated irradiation using proton beam of high energy (up to 10 MeV) is used to simulate the irradiation damage due to high energy neutrons. Towards this, a high temperature experimental setup to be used for the proton irradiation at 300° C was designed and fabricated; and experiments were carried out using proton irradiation at FOlded Tandem Ion Accelerator (FOTIA) and Pelletron at room temperature and at 300° C in separate experiments. Samples were irradiated to different levels of dpa and subjected to electrochemical tests after irradiation. Characterisation of irradiated samples revealed that electrochemical techniques like EPR have measured the extent of Cr depletion due to irradiation caused by different levels of irradiation damage in austenitic stainless steels. Cr depletion was not observed in the samples irradiated at room temperature up to 0.2 dpa. Cr depletion zones were noticed in the sample irradiated at  $300^\circ$  C up to 0.2 dpa and corresponding EPR value was 1.12. This study also confirms the fact that maximum damage is not on the surface - EPR values were found to increase up to a maximum value and then decrease with the depth from the sample surface, e.g. 1.12, 0.4 and 0.2 after repeated EPR testing and metallographic polishing. Further experiments are planned at Pelletron to evaluate the effect of the nature of grain boundary and oversize solute element addition on RIS in the austenitic stainless steels.

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# 11.12 QUALIFICATION OF AUSTENITIC STAINLESS STEEL FOR 100 YEARS DESIGN LIFE OF AHWR

Austenitic Stainless Steel AISI 304LN has been selected for primary system components of AHWR. Design life of AHWR is 100 years. Low Temperature Sensitisation (LTS) has been identified as one of the material degradation

mechanisms which can limit the life of component by making it susceptible to localised corrosion viz. IGC/IGSCC. Even though stainless steel may be non-sensitized to start with, welding can result in formation of chromium carbide nuclei at grain boundaries in Heat Affected Zone (HAZ) which can grow at 300° C and over long term service may result in completely sensitised microstructure. This necessitated the qualification of austenitic stainless steel material with respect to LTS susceptibility.

The Kinetics of LTS is controlled by diffusion of Cr through the

lattice. Using Arrhenius equation and appropriate activation energy for Cr diffusion, it was established that 100 years service at 300° C can be simulated by 11 days thermal exposure at 500° C. Austenitic stainless steel 304 LN in solution was annealed, as well as welded conditions were given this simulation heat treatment and evaluated for LTS susceptibility as per ASTM A262 Pr. A & E. Solution annealed material did not show sensitized microstructure whereas welded material showed sensitised microstructure in ASTM A 262 Pr. A. However, in both the conditions material passed ASTM A262 Pr. E test. This observation indicates AISI 304LN grade material will not become susceptible to LTS –induced IGC/IGSCC during service at 300° C for 100 years.

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# 11.13 MATERIAL DEVELOPMENT AND CORROSION MONITORING FOR SAFETY AND RELIABILITY OF REPROCESSING AND WASTE MANAGEMENT FACILITIES

The processing of spent fuel and waste and its storage form an integral part of nuclear program. The program envisages reprocessing, long term storage in metallic tanks, vitrification of high level waste and also a closed end fuel cycle. Most of these activities involve highly corrosive process fluids - nitric acids and molten glass. A lot of work is already done for materials in nitric acid environments. The mechanism of corrosion resistance of the Nitric Acid Grade (NAG) of type 304L has been studied in detail. The strict control on chemical composition, especially on carbon, nickel, chromium, silicon and phosphorus together with controlled microstructure (fine grain size and retained cold work) helped improve resistance to intergranular corrosion and resistance to general corrosion in concentrated nitric acid solutions. The lower general corrosion rates in nitric acid (5-10 mpy) allow longer life of stainless steel components as the oxidizing higher valence ions that enter the process fluids as (general) corrosion products increase the corrosiveness of the process fluid with time. This is especially beneficial in applications where once flow conditions do not exist e.g. in acidic waste storage tanks.

The role of inclusions in corrosion of stainless steels in nitric acid environment has been brought out clearly in our earlier study. The role of type of inclusions (oxides/nitrides/sulfides/silicates etc) on initiation of corrosion in nitric acid medium will be studied separately. Recent work on the mechanism of end grain pitting has shown that processing of the stainless steels may also result in chromium depletion along the flow lines in highly worked products e.g. tubular products. EPMA analysis had shown the presence of chromium depletion in regions adjacent to the flow bands and the end grain corrosion attack was along the direction of these flow bands. Earlier this type of segregation of Nb/ Ti along flow bands and end grain corrosion was reported for stabilized stainless steels.

Emerging newer materials promise to be much more resistant to nitric acid environments and would stretch the limits of concentration of nitric acid that can be handled without appreciable corrosion. Fuels for newer types of reactors require use of halides in nitric acid streams. Hence, the materials of construction have to be resistant to oxidizing acids containing halides at elevated temperatures. Type 310L NAG, alloy 33 and titanium based alloys are some of the most promising alloys suitable for such environments. Many of these materials are also candidate materials for underground disposal of spent fuel. The passivity and pitting behaviour of alloy 33 has been evaluated in as received and also in welded condition in HCl solution at room temperature as well as at 65°C. The advantage of addition of 0.4% nitrogen in alloy 33 in improving the passivity and pitting resistance was brought out by comparing the results with an experimental alloy similar to alloy 33 that was prepared in the lab and did not contain nitrogen. Electrochemical corrosion testing in NaCl and HCl solutions (different concentration and pH) and immersion corrosion testing in other standard test solutions showed that alloy 33 is highly resistant to localized corrosion. Studies on welded samples showed that molybdenum rich phases do form in the heat affected zone of the weldment of alloy 33. The characterization of these phases by SEM-EDS and TEM is being done.

Initial corrosion studies on the Ti-5%Ta-1.8% Nb alloy manufactured at NFC, Hyderabad for IGCAR have shown that this alloy and its welded structure are highly resistant to corrosion in potential regions which cause transpassive corrosion for stainless steels. In addition, these were found to be

highly resistant to end grain corrosion also. The general corrosion rate was very low, < 2 mpy in concentrated boiling nitric acid solutions (e.g. practice C, A 262, ASTM). The influence of beta phase that gets precipitated during welding/processing was not found to be harmful possibly because of absence of depletion of the alloying elements around the beta phase.

concentration of acidic waste stream during vitrification and use of other corrosive additives will be accounted for to arrive at a safe material-process combination for a specified duration.

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behaviour of alloy 33 in mill annealed condition and welded condition with that of alloy 22 in 1 N HCl solution at 65°C

Vitrification of high level liquid waste involves molten glass and the materials of construction have to be resistant to degradation in it for a sufficiently long timeframe. The interaction of materials of the process pot (alloy 690) with molten glass needs to be studied in detail and mechanism of degradation established. It is of utmost importance to be able to predict the "safe" operating regimes in terms of temperature and time for the process pots used in the vitrificaion process.

In future, different materials, including alloy 690 and its variants will be exposed in molten glass at defined temperatures between 900 to 1200°C for different time durations. This would help in choosing the most optimum material and also to schedule replacement of components ahead of its anticipated operation life to ensure safe and reliable operation. Other factors like



# **12. HEALTH PHYSICS & ENVIRONMENT**

## INTRODUCTION

In the siting of a nuclear power plant, three zones are defined for control of population. The innermost zone, called the Exclusion Zone (EZ), surrounds the plant and defines an area directly under the control of the plant. The second zone, an annulus around the exclusion zone defines the Sterilised Zone or the Low Population Zone, where the growth of population is limited by administrative control. The outer-most zone defines the minimum distance to a high population centre This chapter talks about the evolution of exclusion zones, siting practices and factors determining the extent of exclusion zone for the current and future nuclear facilities.

## 12.1 EXCLUSION BOUNDARY FOR NUCLEAR FACILITIES

Being adjacent to the plant fence, the exclusion zone is the area of greatest importance. It essentially defines a buffer zone where the public has no access. It also helps to define the fenced plant area, the site area and the public area. A regulatory dose control structure can now be put in place so that planned releases from the plant during normal operation and inadvertent releases during abnormal conditions can be controlled to within acceptable limits.

An interesting use of the exclusion area is that several nuclear power stations can share the same site and take support from the common auxiliary facilities, such as fuel fabrication plants, Fuel Reprocessing PSlants [FRPs], waste management facilities. As many as four twin-unit stations have been co-located at several places around the world. The Rawatbhata site in Rajasthan is an example.

St erilised

Zone (5 km)

Exclusion

Emergency

Zone (1.6 km)



The concept of the Exclusion zone originated in the USA in the early 1950s, when there was an acute awareness of the potential effects of nuclear accidents on the surrounding population. This idea was mooted primarily to insulate the public from the harmful effects of low probability, high consequence accidents.

The earliest attempt at sizing the EZ was made by the US Safeguards Committee in USAEC Document WASH-3, wherein exclusion distance was numerically specified as a circle of radius R (miles) =  $0.01 \sqrt{P}$ , where, P is reactor thermal power (kW). This formula obviously would not yield practical sizes for medium-sized or large power reactors: for a typical 3000 MWt reactor, this formulation gives an exclusion radius of 17.3 miles (27.9 km).

The US siting practice as embodied in 10 CFR 100 for the determination of the exclusion boundary and the low population

zone around a reactor defines these radial distances as follows:

An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.



The methodology for implementing this in the US Context is coded in the USAEC document TID-14844 . When implemented, the exclusion distances for most US reactors fall in the range 0.5–1.6 km.

#### Exclusion zone in the Indian Context

The Indian siting code defines the exclusion area as follows:

An exclusion area of at least 1.5 km radius around the plant shall be established. This area shall be at the exclusive control of the station and no public habitation shall be allowed in the area. Under design basis accident conditions, a member of the public shall not receive a dose equivalent more than 0.1 Sv for whole body and 0.5 Sv for thyroid of children.

A Sterilised area upto 5 km around the plant shall be established by administrative measures. In this area, natural growth is permitted but planned expansion of activities which will lead to an enhanced population growth are not allowed by administrative measures. On the lines of the above, all NPP sites in India have an exclusion boundary of radius 1.6 km, except the Kaiga site, where the radius is 2.3 km. Where more than one twin station is sited, each must have an exclusion radius of 1.6 km.

## Siting Practice in Other Countries

The practice in some countries is illustrated in the table:

#### Factors Determining Exclusion Boundary

The factors determining the exclusion boundary are: reactor type and power, engineered safety features, containment design and characteristics of the site. The US code of practice assumes a severe beyond design basis accident and does not give credit to design features save the containment. In the Indian regulatory practice, a design basis accident is considered at the reactor site and the distance is found out at which both the reference levels are met defines the exclusion zone. However, a minimum exclusion radius of 1.5 km is always applicable.

Country	Exclusion	Restricted Zone	Remarks
	Zone (radius)	(radius)	
Canada	~1 km	-	Limits are placed on individual and
			collective doses
Czechoslovakia	500 m	-	Typical values
India	1.6 km	5 km	No population centre greater than
			10000 within 16 km in the main wind
			direction.
Italy	0.8 -1 km	-	Typical values adopted
USA	~0.65 km	Low population	The values have been found to be
		zone of $\sim 5 \text{ km}$	acceptable for plants licensed in the USA
			in the 1960s and early 1970s
USSR	~3km	-	Typical values

Typical National Practice

The first Indian power reactors were the two 210-MWe BWRs sited at Tarapur. A site radius of 1.6 km was selected. The distance was believed to be conservative but to what extent could not be ascertained. Recently an interesting application of the 10 CFR 100 methodology was made to the Tarapur BWRs. The results indicate an exclusion radius of about 950 m. An adaptation of the US methodology to Indian PHWRs is shown in Tables. For 220 MWe double containment PHWRs, an exclusion boundary of about 625 m is calclulated.

Similar results are also indicated from recent DBA calculations made on the Indian PHWRs for submission to the AERB's NPP Zones Review Committee. An exclusion radius of about 800m was indicated. It is on the basis of these and other calculations, that the committee has recommended an exclusion zone of 1.0 km for all future Indian reactors with the same reference levels.

#### **Exclusion Zone for Other Nuclear Facilities**

Auxiliary nuclear facilities like fuel reprocessing plants are sited in the Exclusion Zone of an NPP, thereby implying that they pose a relatively small hazard to the public which can be accommodated within the buffer zone. This has been recently confirmed for the Tarapur site for two FRPs, one handling the irradiated PHWR fuel and the other the high burnup AHWR fuel, when it was demonstrated that for four Upper Limit accidents in the FRPs, the consequence could be limited to the fenced area of each plant.

In view of the involatile waste forms and relatively small inventories, the same can be said to be valid for waste management facilities at power reactor sites.

#### Exclusion Zone for Advanced Reactors

As stated earlier, an exclusion zone reflects the hazard to the public from the normal and abnormal operation of a power plant. It represents a passive safety area that cushions the impact on public of abnormal operation of a nuclear plant. Since releases for normal operation through the liquid and air routes cannot be eliminated altogether and similarly, since high consequence accidents cannot be entirely ruled out, exclusion zones can be optimized but perhaps not done away with altogether for the present reactors.

It is now accepted that while the second generation nuclear plants aimed for improvement of reactor safety through active features and the third generation plants aim at the same through passive safety features, the Fourth Generation reactors aim to positively eliminate accidents through Safety-by-Design approach. The last approach can obviate the need for off-site emergency preparedness, which in turn might make it possible to eliminate the need for exclusion zone in power plant siting.

Reactor Power	10 CFR 100	Modified
(MVVt)	Assumptions	Assumptions
	Containment	Containment
	Leak Rate :	Leak Rate:
	0.1 %/day	0.1 %/day
500	709	550
800	974	620
1000	1,132	649
2000	1,794	730
3000	2,361	784

N.B.: Whole body dose controlling for Proposed Assumptions Thyroid dose controlling for 10 CFR 100 Assumptions Exclusion Boundary Distance (m) for Light Water Reactors

Reactor Power (MWt)	Containment Leak Rate: 0.1 %/h	Containment Leak Rate: 0.2 %/h	Containment Leak Rate: 0.3 %/h
500	781	1,234	1,564
800	1,076	1,624	2,016
1000	1,236	1,835	2,277
2000	1,837	2,627	3,215
3000	2,283	3,200	3,856

N.B.: All exclusion distances above based on thyroid dose which is controlling.

Exclusion Boundary Distance (m) for Single-Containment PHWRs

Reactor Power	Partial -Double	Full - Double
(MWt)	Containment	Containment
	Leak Rate :	Leak Rate:
	0.3 %/h PC	0.3 %/h PC
	30.0 %/h SC	3.0 %/h SC
500	619	573
800	891	626
1000	1,034	649
2000	1,564	736
3000	1,949	790

N.B.: Partial double-containment exclusion distances controlled by thyroid dose. Full double-containment exclusion distances controlled by whole body dose Exclusion Boundary Distance (m) for Double Containment PHWRs

#### 13. R&D ON LIFE MANAGEMENT OF REACTOR COMPONENTS

### INTRODUCTION

This chapter covers various research and developmental activities pertaining to the life management of ageing components of Indian nuclear reactors. During the course of operation in the reactor environment, components age and degrade which affect their intended performance and limit their service life also. Sometimes, unintended material degradations take place as well, for which no consideration was given at the time of design. These unintended degradations not only affect the performance of the components but also shorten their intended service live. Such a situation has among others, an economic penalty as well.

Several R&D activities are being carried out to meet the challenges posed by the situation. These activities range from the basic experimental studies under the simulated environment to understand the mechanism of degradations, development of numerical models for predicting such degradations in advance and design, development and commissioning of repair tools and technologies for life management and life extension of these components.

#### **13.1 LIFE EXTENSION OF COOLANT CHANNELS**

The loose fit garter springs of coolant channel assemblies, used in the first seven Indian PHWRs, are found to be susceptible to displacement from their initially installed locations due to vibrations caused by a number of construction and commissioning activities such as hot commissioning. This displacement could result in occurrence of premature cold contact between pressure tube and calandria tube due to creep deformation of both the tubes in that channel, where such displacement of GS is significantly large. This may lead to formation of hydride blister over a period of reactor operation. The embrittlement and cracks caused by the hydride blisters may result in rupture and eventual failure of the pressure tube. There was a need to design and develop a remotely operable system which can precisely detect and relocate these displaced garter spring spacers in the coolant channel assemblies of new as well as operating reactor for extension of service life as part of the programme for the life management of coolant channels of Indian PHWRs.

Pressure Tube Flexing Tool was developed to reposition the displaced loose fit garter springs by creating artificial "Walking Mechanism" in fresh reactors. This tool uses the same technique to reposition the displaced garter spring to the new desired location, which is responsible for its displacement during the hot conditioning of the reactor. This tool was successfully employed in Narora-1 and 2 and Kakrapar-1.

To accomplish similar task in operating reactors, the INtegrated Garter spring REpositioning System (INGRES) was developed for repositioning of garter springs in highly radioactive coolant channels. The INGRES system incorporates sophisticated electrical, instrumentation, pneumatic, hydraulic and mechanical sub systems. The system is operated through computer interface in a special control room outside the reactor building. The various versions of INGRES were developed and deployed at different reactor sites.



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# 13.2 DEVELOPMENT OF ANALYTICAL TOOLS FOR ASSESSING SERVICE LIFE OF PHWR COOLANT CHANNELS

A considerable amount of research and development work for the past 20 years, that has been directed towards understanding the various degradation mechanisms of Zircaloy-2 pressure tubes has led to the development of a number of numerical models relevant to these degradation mechanisms. These models, developed and synthesised in the form of computer codes, have undergone extensive validation through internationally available information and those derived from indigenous ISI and PIE programmes related to pressure tubes of Indian PHWRs. These computer codes are being used on a regular basis for estimating the residual life of the coolant channel components and taking various safety-related decisions.

As the pressure tube material in the new and the re-tubed Indian PHWRs is Zr-2.5%Nb, the era of zircaloy-2 as the pressure tube material will come to an end in near future. The in-pile degradation mechanisms of both the materials are similar except for the rates and their severity. The critical parameters determining the life of the Zr-2.5%Nb coolant channel are different from those of zircaloy-2 pressure tubes. Some of the degradation models developed earlier for zircaloy-2 pressure tubes like those of irradiation-induced creep and growth model and hydride blister model can be used for Zr-2.5%Nb pressure tubes as well, provided the constants relevant to Zr-2.5%Nb pressure tube material are used. These constants will have to be determined from information generated from ISI of these channels.

As the hydrogen-induced degradation mechanisms are amongst the critical life limiting mechanisms, work related to hydrogen uptake model and the Delayed Hydride Cracking mechanism (DHC -predominant mode of crack propagation in Zr-2.5%Nb material) has been undertaken. Besides, analytical models for optimisation of the pressure tube design and the parameters for the pressure tube–end fitting rolling are also under development.

The subsequent sections deal with each of the models developed.

## Computer Code to Study the Crack Propagation by Delayed Hydride Cracking (DELHYC) Mechanism

The model 'DELHYC' uses Finite Difference numerical technique to solve the differential equation of hydrogen diffusion under stress and concentration gradients. The basic concept of the model has been derived from published work. The model is based on the assumption that crack tip hydrostatic stress field has cylindrical symmetry and it remains constant inside the plastic zone and decays as a function of r-<sup>1/2</sup> outside the plastic zone. Based on this assumption, the complex differential equation of 3-D hydrogen diffusion degenerates to a 1-D problem, which is then solved by the finite difference numerical technique. The model has the capability to simulate (a) delayed hydride cracking under isothermal condition where the test temperature is achieved either by heating or cooling (b) effect of direction of approach to test temperature (c) effect of thermal cycling on delayed hydride cracking mechanism.



The model first evaluates the critical length of hydride using the inputs of applied stress intensity factor and material properties like Poisson ratio, yield strength, fracture strength of hydride at a given temperature. Thereafter, rate of diffusion of hydrogen (hydrogen flux) to the crack tip under stress gradient and concentration gradient is evaluated at the end of each small time interval, which together constitutes the total experimental period. The hydrogen diffused to the crack tip is checked for its potential to grow to hydride of critical length. When the hydrogen diffused to the crack tip is sufficient for the growth of hydride of critical length, the crack front is assumed to advance by an amount equal to the critical length of hydride.

Accurate analysis of the hydrogen diffusion at the crack tip requires finite element analysis tools. Analysis based on finite difference method along with cylindrical approximation to the stress field at the crack tip gives results, which are in reasonable agreement with experiment. Some typical studies carried out using this code for the published experimental work have shown an excellent match.

#### Computer code for design optimisation of Pressure Tube thickness for AHWR

A computer program has been developed to optimise pressure tube thickness from strength and creep/growth considerations. The diameter of pressure tube had been fixed based on thermal hydraulic and fuel bundle dimension considerations. The minimum required thickness of pressure tube based on strength consideration changes as per pressure and temperature conditions at different points along the length of the tube. The program calculates the pressure and safe working stress based on temperature at different points spaced at an interval of 10 mm from inlet for design, normal operating and hot shutdown conditions. For safe working stress at a particular temperature, it linearly interpolates the values at defined temperatures. Similarly for operating pressure and temperature at a particular elevation, it linearly interpolates the defined values at previous and next data points defined by thermal hydraulics considerations. For all possible combinations of loads from tail pipes and feeder pipes in addition to pressure loads, it compares the stress components with allowable limits as per ASME requirements. Based on this analysis, it arrives at a final



strength consideration

minimum thickness required from strength point of view for any predetermined loading condition. Based on the final thickness for all loading conditions, it calculates creep and growth of the channel under normal operating conditions and estimates the probable life of the pressure tube for predetermined extreme values of design parameters.

## Computer Code for In-Service Simulation of Composite Sleeve Rolled Joint

In this code, a composite sleeve (made of SS304L and SS403) rolled joint with pressure tube has been simulated as a compound



Reduction in residual contact pressure with increasing interface radius.

cylinder interference joint. In this simulation, the rolled joint between pressure tube and intermediate sleeve (SS304L) is simulated as an equivalent interference fit joint of same dimensions. The shrink fit joint between the intermediate sleeve and outer sleeve (SS403) has also been simulated. As the composite sleeve is subjected to thermal cycle from room temperature to the operating temperature, a reduction in the contact pressure between pressure tube and intermediate sleeve takes place. This code quantifies the reduction in contact pressure in subsequent thermal cycles.

The code has been utilized to arrive at a feasible interface radius between intermediate and outer sleeve to have a reasonable contact pressure even after subsequent thermal cycles during service.



## 13.3 DEVELOPMENT OF SLIVER SAMPLE SCRAPING TECHNIQUE (SSST)

Sliver Sample Scraping Technique (SSST), has been developed to take samples from the inside surface of the operating pressure tubes of Indian PHWRs. These samples are later analysed for assessing safe operating life of the pressure tubes. The SSST is a non-destructive technique as it does not affect the remaining service life of pressure tube and is remotely operable. The sample,



Wet Scraping Tool (WEST)



weighing approximately 90 mg, is analysed for estimating the hydrogen content of the sampled pressure tube. It is being used frequently for estimation of hydrogen content of pressure tubes of operating Indian PHWRs. Dry as well as wet versions of scraping tools, based on SSST, have been developed. The wet version tool, known as WEt Scraping Tool (WEST), is operated through fuelling machine for obtaining the samples.

A Memorandum of Understanding (MoU) between BARC and NPCIL was signed in March 2003 for fabrication of WEST tools and their accessories, including training of manpower to carry out scraping operation in operating reactors.

Scraping tools have been used extensively for life management activities of pressure tubes since 1998. Till date, approximately 637 samples have been obtained from 170 pressure tubes in various campaigns and from various reactor sites.



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#### 13.4 BOAT SAMPLING TECHNIQUE

Boat Sampling Technique (BST) has been developed for obtaining samples from the surface of any operating component. The technique is a nondestructive surface sampling technique as it does not cause any plastic deformation or thermal degradation of the operating component. BST can be used, remotely and in water-submerged condition, with the help of a handling mechanism. The samples are boat-shaped, having 3 mm maximum thickness and require 180 minutes for getting scooped from a location. The samples are used for metallurgical analysis to confirm the integrity of the operating component. BST incorporates mainly sampling module, handling mechanism and electric and pneumatic sub-systems.

Based on specific requirements, two types of sampling modules have been developed. Sampling Module-1 has been developed



Sample from SS Plate

for obtaining samples from the Heat Affected Zone (HAZ) of weld H4A of the core shroud of Tarapur Atomic Power Station (TAPS). Integration of Sampling Module-1 with its



Sampling Module-1



Handling Mechanism is being done and qualification of the whole system is under progress. Boat Sample from soft material has been obtained as preliminary qualification of the system. Full scale mock-up is being planned and in near future the system shall be deployed at TAPS.

Sampling Module-2 has been developed for obtaining in-situ samples from flat plate and from any cylindrical vessel, having a diameter of more than 9 inches. This technique is proposed to be used at the Heavy Water Plant at Baroda in near future, for obtaining boat samples from the Ammonia Converter.

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# 13.5 HYDROGEN CHARGING AND STUDIES OF HYDROGEN INDUCED DEGRADATION MECHANISMS IN ZIRCONIUM ALLOY PRESSURE TUBE SPOOL PIECE

It is well known that hydrogen, when present in zirconium alloys manifests into problems like hydride blistering, hydrogen embrittlement and propagation of flaw introduced either during service or in the manufacturing process by Delayed Hydride Cracking (DHC) mechanism. These hydrogen-induced degradations limit the safe operating life of Zirconium alloy pressure tubes and hence are required to be studied for (i) understanding their mechanisms and (ii) quantification of the extent of damage by each of them on periodic basis under in-reactor conditions. Several experimental and analytical studies related to these phenomena have been carried out world wide and are still being done. Most of the published research work is limited to small specimens and very few of them have been carried out on actual component. Difficulty in charging hydrogen in a large length of Pressure tube could be the prime reason for lesser number of above studies on the actual component.

The technology for accelerated but controlled charging of hydrogen in large length of pressure tube piece has been developed over the past few years. In this, zirconium alloy pressure tube piece filled with 1.0 M (or higher) aqueous solution of lithium hydroxide and hydraulically sealed at both the ends is heated to 300°C for several hours ranging from 50 hrs – 300 hrs depending upon the amount of hydrogen to be charged. Typically, 50 hrs of experiment will charge nearly 30 ppm –



40 ppm of hydrogen. The schematic of the experimental set-up is shown below.

Several other studies like growth of hydride blisters, crack propagation by DHC mechanism under the simulated in-reactor condition of pressure and temperature and hydride embrittlement etc., have been planned. Development of technique for in-situ measurement of hydrogen is being carried out.

These studies will not only help in understanding the mechanisms in a better way but also help in strengthening the in-house developed numerical models for these damaging mechanisms.

## Growth and Characterisation Studies of Hydride Blister

The experiment of growing multiple blisters in a zirconium alloy pressure tube piece under the simulated in-reactor pressure and temperature condition was carried out in an experimental facility called 'High Temperature Loop' (HTL). A total of 10 blisters were planned to grow on the hydrogen charged Zr-2.5%Nb pressure tube spool piece. These were symmetrically located with respect to tube ends in two groups of five each. The axial separation between the two groups was 50 mm. Within the group, the blisters were located at different angular positions.



#### **Experimental Set-up**

The experimental set-up consisted of a 160 mm long Zr-2.5%Nb alloy pressure tube piece hydraulically sealed at its both ends by SS flanges with the help of SS304 spiral wound graphite gasket of 4 mm thickness and 5 mm radial width and 6 high strength Carbon Steel studs.

Hollow brass cones of different tip diameters (3 mm – 8 mm) were used for simulation of cold spots on the outside surface of pressure tube piece. The temperature of the cold spots was maintained between  $30 - 40^{\circ}$  C by circulating cold water through



Experimental set-up connected High Temperature Loop



Zoomed View showing the arrangement of Brass Cooling Cones

them. The photographs presented in figures show the actual assembly of the experimental set-up.

#### **Experimental Conditions**

The temperature and pressure of  $280^{\circ}$  C and  $90 \text{ Kg/cm}^2$  respectively were maintained during the period of the experiment. The pH was maintained around 9-10.5. The flow rate was maintained around 10-15 lpm.

The experiment was conducted for 30 days. During this period, all the above parameters defining the experimental conditions were meticulously maintained. After the experiment was over, the piece of the pressure tube was removed from the assembly for visual observations and other examinations.

#### Characterisation of the blisters

Experimentally grown blisters were characterised by various nondestructive and destructive techniques to find the size, shape, integrity, depth etc. The techniques used included neutron radiography, video-microscopy, 3D surface profiling, scanning electron microscopy and optical microscopy. Examination showed that blisters in the tube were of different sizes. The smaller hydride blisters were free from cracks but larger ones having depth of more than 0.4 mm were cracked.





The figure gives video-microscope image on the tube surface, the 3D surface profile of the blister, the neutron radiograph of blister and the cross section of the hydride blister. The blister is very small in size Its diameter is about 1.1mm and depth about








#### R&D on Life Management of Reactor Components





0.3 mm. The image of the blister cross section shows that blister is free from any cracks even though the tube was internally pressurized during the experiment.

All the blisters present in the tube had lenticular shape, which was similar to those observed in irradiated tubes.

#### Burst Test of the Blistered Tube

The blistered tube was sealed at both the ends with SS Flanges and SS304 serrated metallic gaskets. The sealing load was applied by 6 high strength carbon steel studs. The schematic of the assembly is shown below.

The sealed assembly was connected to a reciprocating type hydraulic pump. Before the start of the pressurising process, the assembly was kept inside a chamber made of 10 mm thick and 8" carbon steel pipe. The tube was pressurised in the step of 20 Kg/cm<sup>2</sup>. It was held for 10 minutes at each pressure. In this way, the pressure was raised up to 380 Kg/Cm<sup>2</sup> at which it burst

with a thudding sound. The unique feature of this bursting was that the crack length was extending from one end to another and it had passed through the middle of both the blisters falling in one line. The picture of the cracked pressure tube is shown below.



The yield strength and ultimate tensile strength of the material of test pressure tube was not been measured at that moment. In these circumstances, the strength revealed by the burst test was compared with that available in the literature. In the transverse direction, UTS of a CW Zr-2.5% Nb tube produced by NFC, India, was found to be 641 MPa whereas the burst test of the blistered tube revealed the value to be around 430 MPa. This conclusively proves that the blistered tube is likely to fail at a pressure much lower than that actually given by UTS. The extent of decrease in the burst pressure will depend upon the size of hydride blister(s). In the present case depths of the cracked blisters have been observed to be around 600 microns.

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#### 13.6 SLUDGE LANCING EQUIPMENT (SLE)

Periodic sludge lancing enhances steam generator life by mitigating corrosion between its tubes and tube sheet. The present modular sludge lancing equipment has been designed to dislodge and remove sludge from secondary side of tube sheet of mushroom type steam generators of PHWRs. This equipment comprises of

- Closed loop de-mineralised water circulation system
- High-pressure remote lancing Jet Manipulator Assembly (JMA)
- Remote Visual Inspection System (RVIS)
- Instrumentation & control system.

The equipment is basically divided into six process modules, specially designed JMA, control panels, electrical power supply panel and RVIS. All dimensions of modules are within 2.8 m. The modules are interconnected with suitable flexible hoses with cam type quick release couplings. This entire system including JMA & RVIS is monitored and manouvered by electronic instrumentation and PLC-based supervisory control units. Successful working of sludge lancing equipment is demonstrated on a steam generator mock up. The schematic flow diagram of the sludge lancing equipment is shown below.



The six process modules comprise:

- Air-operated double diaphragm pumps for removing the sludge water mixture,
- Sludge tank and Rotary Cleaner for removing sludge particles above 100 μ;
- 30 μ, 2 μ & 0.5 μ filter assemblies for removal of dislodged sludge,
- Storage tank with 4 cubic meter capacity and centrifugal pumps for circulation
- High pressure Triplex Plunger pump for generating high-pressure water at 250 bar at a flow of 250 lpm.

The high pressure lancing water in the nozzle head assembly generates multiple high velocity jets along the 3 mm wide inter tube lanes of the steam generator to clean the tube sheet. This high-pressure water jet dislodges the sludge from the secondary side of steam generator tubes (of limited heights ~ 150 mm) and tube sheet face, which gets collected on the bottom tube sheet. This is extracted to Sludge Tank (ST-1) by self-priming air operated double diaphragm pump (EP-1 or EP-2) capable of handling 0 to 18.0 m<sup>3</sup> / hour of sludge water. This sludge water is passed through the rotary cleaner and further filtered using basket and cartridge filter stages before being circulated to the high-pressure triplex pump. The discharge of the triplex pump is connected to the nozzle head of the jet manipulator assembly.

The specially designed compact pneumatically-operated manipulator carrying the nozzle head assembly (Jet manipulator assembly) moves in forward and reverse directions, in the no tube lane by gripping to the steam generator tubes. The nozzle head is also moved vertically up/down for effective lancing. The camera positioning, control and lighting module of the Remote Visual Inspection System (RVIS) are connected to another pneumatically operated manipulator, for visual inspection of the tube sheet. The camera module of the RVIS houses four CCD cameras with 75,50,25 & 4 mm focal lengths and light source. The pan & tilt mechanism gives angular orientation to the camera module. The process modules are located in the ground floor and the manipulators & valve control stations are located close to the steam generator in the SG floor.

The equipment is maneuvered remotely from two control consoles. The interlocked equipment and manipulator controls are implemented using networked PLC-based control system.





Air operated diaphragm pumps (EP-1 and EP-2),



Sludge tank (ST-1), Centrifugal Pump (BP-1), Rotary Cleaner (RC)



Main storage tank (ST-2)



High pressure Triplex plunger pump (TP)



Main control panel



Important equipment of the sludge lancing system

-	17:	An Elizarda LENA		An Example in the set			
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Performance evaluation of lancing nozzles



Snapshots of the ceremony of handing over of sludge lancing system to NPCIL, attended by Chairman, AEC, Director, BARC, CMD, NPCIL

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# 14. SEISMIC, STRUCTURAL AND LBB STUDIES

#### INTRODUCTION

This chapter contains articles on analytical and experimental simulation for structural qualification of reactor components and systems under normal, upset and accident conditions. Development of methodologies for meeting Leak before Break (LBB) criteria for high pressure and temperature piping systems under all conditions, fracture studies on various piping components, generation of valuable information by testing large piping components, analytical and experimental studies of bimetallic weld joints and development and testing of energy absorbing devices like Elasto Plastic Dampers (EPDs) and Lead Extrusion Dampers (LEDs), are some of the important research activities highlighted in this chapter.

# 14.1 CYCLIC FRACTURE INVESTIGATION ON STRAIGHT PIPES UNDER REVERSIBLE LOADING

The current Leak Before Break (LBB) assessment is based primarily on the monotonic fracture tearing instability. The maximum design accident load is compared with the fracture-tearing resistance load. The effect of cyclic loading has generally not been considered in the fracture assessment of nuclear power plant piping. Indian nuclear power reactors consider Operational-Basis-Earthquake (OBE) and Safe-Shutdown-Earthquake (SSE) events in the design of various structures, systems and components. It is a well-known fact that the reversible cyclic loading decreases the fracture resistance of the material, which leads to increased crack growth. Unlike monotonic fracture, in cyclic fracture, the instability depends on the full load history and on parameters such as loading ratio, loading range and number of load cycles. A cracked component, which is safe for monotonic load, may fail in limited number of cycles when subjected to fully reversible cyclic load of same amplitude. Keeping this in view a series 23 tests have been carried out on circumferentially through wall cracked seamless and circumferential seam welded straight pipes under reversible cyclic bending loading. Out of the 23 tests, 15 were on SA333 Gr.6 Carbon Steel Pipes and the remaining 8 on AISI Type 304 LN Stainless Steel Pipes. The carbon steel grade is same as that of the Indian 500 MWe Pressurised Heavy Water Reactor's (PHWR) Primary Heat Transport (PHT) system and the stainless steel grade is the proposed material for Advanced Heavy Water Reactor's AHWR PHT system. The investigations have been carried out for both load and displacement controlled loadings. All the tests have been conducted under quasi-static i.e. slow loading rates simulating only the cyclic nature of the loading, which has substantial effect on the fracture resistance of the material. The cyclic test results have been compared with the corresponding monotonic pipe fracture test results.

The comparison of the displacement controlled quasi-static cyclic test and the corresponding quasi-static monotonic test results shows that the cyclic loading has less influence on the maximum load carrying capacity but there is significant loss in the energy absorbing capability of the piping during the cyclic loading. Comparision of the J-R curves from cyclic fracture tests calculated using the envelope curve and  $\eta$  factor procedure, with the corresponding monotonic test J-R curve shows that there is

significant reduction in the fracture resistance under cyclic loading conditions.









Comparison of the load versus load line displacement curves of displacement controlled cyclic and corresponding monotonic test results.





The load controlled quasi-static cyclic tests shows the importance of the number of the load cycles in the fracture assessment of piping subjected to cyclic loading event. The cyclic fracture tests have shown that the pipes may fail in limited number of load cycles with the load amplitude sufficiently below the monotonic fracture/collapse load . A simplified material specific (for SA333Gr6 carbon steel) master curve has been generated directly from these pipe cyclic tearing tests. The master curve is the plot of the cyclic load amplitude (given as % of maximum load recorded in corresponding monotonic fracture test) versus number of load cycles to fail (N<sub>f</sub>). This curve can readily be used in the current practice for LBB qualification for evaluating the critical load (which accounts for cyclic damage and number of



Normalised Load Amplitude versus Number of load cycles to fail  $(N_i)$  curves Load Controlled cyclic tests having Load ratio -1.

load cycles) in term of the percent of monotonic critical load. For the postulated 50 cycles of high amplitude loading, the failure load reduces to 78% of the monotonic failure load for the base material and to 74% of the monotonic failure load for weld material.

The crack growth in cyclic tearing consists two parts, one is crack growth due to low cycle fatigue and the other is due to static tearing. Few load-controlled tests have been investigated in detail with the objective to establishing a methodology for prediction of instability and failure life under fully reversible cyclic loads. In the study crack growth by both, fatigue (under large scale yielding i.e. Low Cycle Fatigue Crack Growth) and static fracture have been considered. The cycle-by-cycle crack growth contribution by both the modes has been calculated and then accumulated predicted crack growth has been plotted against number of cycles.



These experiments highlight the need of modifying the current LBB design rules (i.e. allowable load limits for peak dynamic loads at various postulated crack locations) to make the LBB design more realistic.

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#### 14.2 FRACTURE EXPERIMENTS ON PIPING COMPONENTS

Leak-Before-Break (LBB) concept is now universally used to design the Primary Heat Transport (PHT) piping of the Pressurized Heavy Water Reactors (PHWR). The same is being followed to design the newly built Indian PHWRs. This concept aims at the application of fracture mechanics principle to demonstrate that pipes are unlikely to experience catastrophic break without prior indication of leakage. This requires detailed fracture mechanics analysis of different piping components e.g. straight pipes, elbows and branch tees. There are several issues in fracture mechanics analysis that requires experimental validation, for example, transferability of specimen fracture resistance data to component level, method of extrapolation of fracture resistance curve beyond test range, lack of sufficient elbow fracture test data to validate various new developments. Against this background, a comprehensive Component Integrity Test Program was initiated to address these unresolved issues.

Total 45 fracture tests were carried out at SERC, Chennai on 27 pipes and 18 elbows made of PHT carbon steel material SA333Gr6 of 8-16 inch diameter with various crack configurations subjected to bending moment loading. During the fracture experiments, load, load line displacement, crack growth, crack opening displacement at various locations of the notch were measured continuously during experiments. A new image processing system SPIAS was developed to measure crack growth during fracture experiments.

Figures show the experimental set up for fracture tests and also some typical out-of-plane crack growth in carbon steel pipes and elbows. Figures also show the role of stress tri-axiality in the transfer of J-R curve from small laboratory specimen to component. These test results form a valuable database for experimental validation of any new theoretical developments in future related to the integrity assessment methodology of piping components.



Typical out-of-plane crack growth in carbon steel pipe



Photograph of pipe fracture test



Crack growth in a surface cracked pipe in carbon steel pipe



Elbow fracture test set up



Crack opening of a throughwall-cracked elbow in carbon steel pipe



Variation of stress-triaxiality (A<sub>nq</sub>) near crack tip with crack driving force (J-integral)



Identical J-R curves of specimens, pipes and elbows having identical stress-triaxiality



## 14.3 PROPOSING NEW EQUATIONS TO EVALUATE FRACTURE PARAMETERS OF PIPING COMPONENTS

Simple analytical formula to evaluate fracture parameters of components often simplifies the integrity assessment. While many formulae are available for pipes, very few are available for one of the important piping components, namely, elbows. These equations are especially useful for Leak-Before-Break calculations of PHT piping components of Indian PHWR.



New closed-form equations have been proposed for the following cases :

- To evaluate stress intensity factors of throughwall cracked elbow under combined internal pressure and bending moment. These equations were recommended by European Structural Integrity code SINTAP for integrity assessment of cracked elbows.
- To evaluate limit moment of throughwall circumferentially cracked elbows under in-plane closing and opening bending moments.
- To estimate elastic-plastic *J*-integral and Crack Opening Displacement (COD) of throughwall circumferentially cracked elbow under closing moment.
- To evaluate limit moment of defect-free elbows under combined internal pressure and bending moment.

 Limit load based general expressions of 'η' and 'γ' functions to evaluate fracture resistance (*J-R*) curve from test data.

All these newly proposed equations have been validated with the experimental data, generated under *Component Integrity Test Program* and also available in the literature. Large number of research papers have been published in international journals based on these works.

Figure shows the surface plot of weakening factor ( $M_L/M_o$ ), crack angle (2 $\theta$ ) and radius to thickness (R/t) ratio for throughwall circumferentially cracked elbows subjected to closing bending moment. Table and figure show the comparison of test data with the predictions of the newly proposed closed-form equations.



Comparison of test data with predictions of the newly proposed Crack Opening Displacement (COD) estimation scheme for throughwall circumferentially cracked elbow under closing moment

Test no.	Description of elbow test specimens	Expt. Collapse moment (kN-m)	Predicted by the newly proposed eqn. (kN-m)	% difference <sup>*</sup> -2.7 5	
ELTWIN8-1	8 inch diameter, opening moment, crack angle (2θ)= 95°	98.4	101.1		
ELTWIN8-2	8 inch diameter, opening moment, crack angle (2θ)=125°	80.0	76.0		
ELTWEX8-4	8 inch diameter, closing moment, crack angle (2θ)= 98°	108.7	100.2	7.8	
ELTWIN16-1	16 inch diameter, opening moment, crack angle (2θ)= 96∘	857.0	847.1	1.2	
ELTWIN16-2	16 inch diameter, opening moment, crack angle (2θ)=123°	699.1	678.8	2.9	
ELTWEX16- 3	WEX16- 16 inch diameter, closing moment, crack angle (2θ)= 65°		1092.3	5.9	
ELTWEX16- 4	/EX16- 16 inch diameter, closing moment, crack angle (2θ)= 94°		962.1	2.4	
ELTWEX16- 5	TWEX16- 16 inch diameter, closing moment, crack angle (20)=124°		819.4	-3.4	

Comparison of experimental collapse moments with theoretical predictions (The bracketed numbers indicate the percentage difference with respect to experimental values)

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## 14.4 DEMONSTRATION OF LEAK BEFORE BREAK DESIGN CRITERIA FOR PIPES OF PHT SYSTEM PHWR

Fail-safe design criteria such as Leak-Before-Break (LBB) based on fracture mechanics concepts requires the demonstration of integrity of the piping system by showing that unstable crack growth will not occur before a crack penetrates the wall thickness, nor will it occur for a through-wall leakage size flaw. LBB evaluation is divided in three levels. In level-1, it is shown that in view of the stringent specifications in material, design, fabrication, inspection and testing, there will be no crack initiation, thus avoiding the possibility of crack propagation. In level-2, it is postulated that a crack of certain length and depth has escaped





Test set-up



Maximum crack depth Vs number of cycles for different initial crack depth and R=0.5

detection. But, it can be shown that for the duration of plant life this crack will not grow enough to penetrate the wall, let alone cause catastrophic failure. In level-3, it is postulated that the crack has penetrated the wall and showed that the resultant through-wall crack is stable, produces leakage in sufficient quantity to enable detection and corrective action can be taken before it becomes critical.

Studies have been carried out on 28 carbon steel pipes and pipe elbows to demonstrate the leak before break design criterion and validate the analytical procedures. Summary of the typical results for 200 NB pipes and elbows having part through notch is given below.

In case of pipes, number of cycles to crack initiation can be predicted well by evaluating local stress based on a fracture mechanics approach. For the typical stress range expected in the



piping of PHWR, the number of cycles to crack initiation is very large compared to the expected number of cycles. Paris constants obtained from the standard specimens can be used directly for crack growth rate analysis of pipes. The use of the fatigue crack growth curve given in ASME Section XI will produce a conservative result. Even after crack initiation, the number of cycles required for the crack to grow through-wall is enormously large thus satisfying LBB level-2 criteria. The ratio of moment required to cause instability to the moment expected during SSE is more



than  $\sqrt{2}$ , thus satisfying the LBB level-3 criterion.

In case of elbows, crack initiation has been observed from the inside as well as outside surface of the crown region of the elbow irrespective of the nature of stress (i.e. tensile or compressive). Crack growth from the outer and inner surface in thickness direction is shown in figure. The two cracks join to produce through wall crack. There is no crack under growth monotonic loading for the elbow having through wall crack at the crown. Failure of the elbow has been observed by net section collapse.

Cracked Elbow

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## 14.5 FATIGUE AND FRACTURE RESISTANCE OF AUSTENITIC STAINLESS STEEL PIPE AND PIPE WELDS

A component integrity test program was started to evaluate the fatigue and fracture behaviour of the main heat transport system piping of the Advanced Heavy Water Reactors. Under this program, extensive studies have been carried out on pipe and pipe welds of austenitic stainless steel material (type 304LN). Tests on full-scale pipe welds using different welding processes are rare. The effects of crack shape evolution during crack growth for different initial aspect ratio on fatigue life prediction have also been studied.

Results of the studies on crack resistance behaviour of the austenitic stainless steel pipe and pipe welds along with CT specimens can be summarized as: For the typical stress range expected in the piping of AHWR, the number of cycles to crack initiation is very large as compared to the expected number of cycles. Fatigue crack growth also depends on the aspect ratio. Aspect ratio (2C/a) at the point of through thickness lies in the range of 3 to 4 irrespective of the initial notch aspect ratio. This provides justification for the usual assumption in LBB that for a reasonable part through crack the length at break out (leakage size crack) is not likely to be more than that is normally assumed. Crack growth in surface direction is more for the aspect ratio greater than 4 as compared to thickness direction. Notch of semi circular front (2C/a=2) maintains its shape till through thickness. Number of cycles required for crack to grow through thickness is very large compared to expected during plant life. The use of the fatigue crack growth curve given in ASME Section XI will produce a conservative result. Fracture resistance properties of the pipe and pipe welds prepared by GTAW are comparable whereas that of pipe welds prepared by GTAW+SMAW is on lower side. Initiation of crack growth starts prior to the maximum moment attained by the pipe welds. For the base material the growth is negligible or small. For GTA weld the growth is marginally higher than base. There is substantial growth of crack in case of pipe welds (GTAW+SMAW) shown in figure. Because of crack growth limit load expression available for the pipes gives non-conservative results. This non-conservatism gets enhanced in pipe welds of GTAW+SMAW. Suitable multiplication factor of 0.85 for GTAW and 0.7 for SMAW has been suggested for conservative prediction of limit load based on flow stress of the base material.





Crack growth in base (300NB)



Crack growth in base (150NB)



Crack growth in weld (GTAW+SMAW,300NB)



Crack growth in weld (GTAW 150NB)

## 14.6 FATIGUE-RATCHETING INVESTIGATIONS ON PRESSURISED PIPES, ELBOWS AND PIPING LOOP

The Nuclear Power Plant (NPP) piping components, which are under high pressure, are subjected to large amplitude reversible cyclic loading during the earthquake event. During this event the stresses may exceed the elastic limit of piping material. In this situation there is a strong possibility of accumulation of plastic strain by ratchet action in addition to the low cycle fatigue damage. Ratcheting can substantially reduce the fatigue life and lead to failure due to excessive plastic strain accumulation.

To understand the failure phenomena, 3 the fatigue-ratcheting experiment on 200 NB Sch 100 size pipes were conducted at IIT, Madras. The phenomenon of ratcheting was clearly observed. The pipe bulged under the application of constant internal pressure and cyclic bending moment applied through actuators. The circumferential strain increased with number of cycles before the final rupture at bulged portion.



Bulging of pipe due to ratchetting

Similar studies were carried out on 90° elbows at IIT, Bombay. These tests have been carried out under constant internal pressure  $(0.3\sigma_{ys} - 0.7\sigma_{ys}$  hoop stress) and large amplitude (approximately 3-4Sm) cyclic displacement controlled loading. In all the tests, the elbows have failed at crown location (axial through wall crack formed at crown location), in limited number of cycles (100- 400 cycles).



Circumferential Strain Accumulation



Schematic of Elbow Ratcheting Test Setup



Failure of elbow ratcheting test

Significant ballooning (8-12%) has taken place with simultaneous fatigue damage, which is followed by crack initiation and growth till it becomes through wall thickness. Figures show the accumulation of local hoop strain at crown and gross hoop strain i.e. dilation/ballooning of the elbow cross section (increase in crown-crown and extrados-intrados diameter). For detecting the crack initiation, various techniques such as Acoustic Emission Technique (AET), Ultrasound Technique (UT) and Magnetic Barkhausen Noise Technique (MBT) have been used and consistent readings obtained.

After performing the above component tests, a piping loop was subjected to ratcheting test on a shake table, thus simulating the earthquake loading. Again there was strain accumulation followed by failure due to fatigue-ratchetting.







Piping loop on shaketable



On the analytical front, biaxial ratcheting analysis has been carried out on the pressurized pipe tubes subjected to reversible cyclic axial strain loading. The radial dilation after 5, 10, 20, ... load cycles has been evaluated for various combinations of primary and secondary stresses. Based on these the Ratchet Assessment Diagrams (RADs) have been developed for SA333Gr6 carbon steel material. The RADs are the plot of iso-strain (i.e. accumulated plastic strain after 'N' number of cycles) curves on 'stress amplitude (secondary load)' and 'hoop stress (primary load)' plane and provide a simplified way of ratchetting assessment.

The ballooning in the elbow tests was predicted from these RADs. The comparison of the predicted and experimental values is reasonably good.







- The phenomenon of ratcheting has been observed in piping components when loaded by actuators as well as when loaded on shaketable.
- In general, the number of cycles required to produce significant ratcheting is quite large. Detailed Ratcheting Assessment can help in reducing the conservatism in the seismic design of piping.
- Further studies are required to formulate design rules to account for this phenomenon.

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## 14.7 RATCHETING STUDIES OF PIPING MATERIALS FOR NUCLEAR POWER PLANTS

Ratcheting is defined as the accumulation of strain during asymmetrical stress cycling in a material. This is quite significant as for a suitable combination of mean and cyclic loads there can be large deformations in the structure. In piping materials, even at room temperature significant ratcheting strain accumulation has been found both at specimen level and in actual piping component. The seismic design aspect of piping components receives high importance and is met through a conservative design practice, which leads to increase in cost of building nuclear facilities. It has therefore been realized that in order to develop realistic design code rules ensuring safety against seismic loads experimental ratcheting data needs to be determined. PHWR and the proposed AHWR have an elaborate piping network to cater to the PHT requirements of the respective nuclear reactors. They are primarily made up of SA 106 B and SA 333 Gr 6 for the old and new PHWR respectively and 304L for AHWR. Ratcheting studies are being carried out on the above mentioned materials both at the lab specimen and component level. The objective of the studies is to determine the most important factors leading to ratcheting, derive design criteria based on experiments and finally validation of design in component level tests. Finite Element Analysis and modeling based on various coupled kinematic hardening rules will be also used to predict the ratcheting behavior.

An experiment has been conducted on a straight pipe subjected to internal pressure and cyclic bending load. Cyclic bending load has been applied to the pipe in three point and four point bend test configurations. Finite Element Analysis also has been carried out to simulate ratcheting phenomenon. The pipe is modeled using large strain plastic shell element and analyzed for constant internal pressure. Chaboche non-linear kinematic hardening model is used for predicting ratcheting response.



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At system level, a 3 D piping system has been tested to study ratcheting phenomenon under earthquake loading. The piping system is made of carbon steel grade SA 106 Gr B, Schedule 40. The piping system is supported on a single U-clamp and its both ends are anchored to shake table. A tri-axial shake table system with six degrees of freedom and ten ton payload capacity has been used for the tests. The piping system is pressurized with water and is subjected to a series of seismic waves. Strains are recorded at different locations of the piping system. Ratcheting has been observed at various locations of anchors and elbows. The strain histories at different locations of the piping are shown in figures.



Test setup of the piping system



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## 14.8 INVESTIGATIONS ON IN-ELASTIC BEHAVIOUR AND EFFECTIVENESS OF RE-STRENGTHENING METHOD OF REINFORCED CONCRETE BEAM COLUMN JOINTS

Reinforced concrete structure undergoes deterioration under various environmental and loading conditions during its lifetime. Beam-column joints are critical sub-assemblages in reinforced concrete structure. Under a severe earthquake, inelastic behavior of these joints plays an important role in absorbing the energy by the hysteretic behavior. Hysteretic behavior of joint primarily depends upon its ductility, which in turn depends upon reinforcement detailing. Ductility is defined as capacity of member to sustain inelastic deformations without failure. In seismic re-evaluation of old structures and evaluation of collapse load for new structure, information related to ductility ratio and degradation characteristics is vital. The collapse load of a structure is that ultimate load pattern beyond which it will form a mechanism leading to instability and thereby collapse. It is required to ascertain that non-safety related structure would not fall on the safety related structures. The main aim of this project is to quantify the ductility factors available as prescribed by Indian standards. This information will be helpful in the seismic reevaluation and seismic retrofitting of the reinforced concrete structure. This experimental scheme is focused on detailing aspects and their influence on inelastic behavior of joints. About 48 joints representing our existing structures with varying cross-sections and reinforcement details are being tested. Till now, testing of 16 joints has been successfully completed.



Test set-up at SERC, Chennai



Crack pattern at ultimate stage for joint tested under monotonic loading



An analytical model is developed which is validated with experiment that can be used for nonlinear analysis of reinforced concrete structures. This data is useful to our DAE for seismic reevaluation of old structures, formulation of new codes and development of strong database in this frontier.



Old structures built as per standard available then; or the structure, which has seen some damage, may need some strengthening to sustain earthquakes of greater intensity. Fiber reinforced composites is one of the emerging material used for repairing and re-strengthening of the reinforced concrete structures. This activity deals with experimental and analytical investigation of the inelastic behavior of reinforced concrete beam-column joints under monotonic and quasi-cyclic loading. The tested joints are being repaired using fiber reinforced composites and again tested to check their re-strengthening capacity.



Experimental Results of joint under quasi cyclic loading conducted at IIT, Mumbai

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## 14.9 ENERGY ABSORBING DEVICES FOR SEISMIC RESPONSE REDUCTION

One of the ways of reducing seismic response is by using snubbers, which permit low thermal movement but arrest the fast seismic movement. However, snubbers are costly and require periodic maintenance. Therefore there is thinking in favour of using energy absorbers in place of snubbers to reduce the seismic response in PHWR equipment and piping systems. Energy absorbers like Elasto-Plastic Dampers (EPDs) and Lead Extrusion Dampers (LEDs) absorb large part of vibration energy and lesser energy is transmitted to piping systems and equipment.







stable yielding deformation, resulting in high levels of energy dissipation or damping. The area of this force displacement relationship gives the energy dissipated by damper plate.



Elasto-plastic dampers were fabricated and tested for its characteristics. Shake table tests were performed on piping



systems with EPDs. Subsequently the EPD was incorporated in a piping loop at SERC (Chennai) and the response of the loop with and without EPD was measured. It is seen that there is significant reduction in the response of the piping system. Nonlinear dynamic analysis was also carried out on the piping and the methodology was validated by comparison with test results . A complex piping with and without EPDs was also tested on shake table at CPRI (Bangalore) and the reduction in the response of the piping is observed.



Complex piping system on shake table showing location of Damper attached



Lead Extrusion Dampers work on the principle of extrusion of lead. LED absorbs vibration energy by plastic deformation of lead and thereby mechanical energy is converted to heat. On being extruded lead re-crystallizes immediately and recovers its



Lead Extrusion Damper.



Test carried out on Lead Extrusion

original mechanical properties before next extrusion. The area of force displacement relationship of the damper gives the energy dissipated in the extrusion of lead. Lead extrusion Damper of 15 tonnes capacity was designed and fabricated. Static and dynamic tests are carried out on the LED at SERC Madras. The desired hysteretic characteristics were demonstrated by this test. One possible application of the LED is in the seismic response reduction of coolant channel when it is coupled to the fuelling machine. The load transmitted to the coolant channel is significantly reduced when the LEDs are attached to the F/Ms. Shake table



tests are planned on vessels restrained by Lead Extrusion Dampers.

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#### 14.10 STRUCTURAL INTEGRITY ASSESSMENT OF BIMETALLIC WELDED JOINTS

Structural integrity assessment of Bi-Metallic Welds (BMW) is an important issue for the power plant and process industry. Bi-metallic joints are often used to connect ferritic pressure vessel nozzles or ferritic piping with austenitic piping. Recent international surveys of such welded joints have shown that there are several cracking problems, due to fabrication induced defects, ageing, corrosion or thermal fatigue caused by temperature changes during service life of plant. Defects in structural components often occur within or near welds across which tensile properties may vary significantly. This mismatch in tensile properties can affect the plastic deformation pattern of the defective component and hence the crack driving force such as the J-integral or the crack tip opening displacement. Structural integrity assessment methods for homogeneous structures can be applied to welded structures if the tensile properties of the weakest material are used. However such a simplified approach can lead to an unduly conservative result. Conventional defect assessment methods such as R6 and GE/EPRI etc. have been modified to incorporate strength mismatch effects. All these defect assessment procedures require an accurate estimate of the limit load.

In BARC, comprehensive studies were performed at the specimen level. The classical upper bound approach of limit analysis was modified to include the presence of compressive zones in specimens subjected to predominant bending load. Theoretical plane-strain solutions were obtained for standard SENB, TPB and CT specimens and were compared with the classical slip-line field solutions. In order to determine the fracture properties, experiments were performed on cracked specimen to obtain their load-deflection behaviour. This load deflection curve is then converted into J-R curve with the help of plastic eta  $(\eta_n)$  functions. For the case of homogeneous specimens eta functions are available in ASTM (E 813-88) standards. The theoretical solutions obtained from the developed modified upper bound approach reveals that compared with the  $\eta$ -factor for a homogeneous specimen a soft under-matched weld will increase it while a hard over-matched weld will decrease its value. This observation is in accordance with the findings of other researchers working in this area. This means that the current J-integral estimation equations may underestimate the J-integral for under-matched welds but will over-estimate them for over-matched welds. The modified upper bound approach was also used to obtain accurate plastic rotation factors (r\_) for bi-metallic specimens like SENB, TPB and CT specimens. These plastic rotation factors would be used to obtain CMOD based eta functions ( $\eta_{\mbox{\tiny CMOD}})$  as they provide a more robust and accurate J-estimation. The developed limit load



solutions of bi-metallic specimens were used to obtain the equivalent homogeneous model. This equivalent stress-strain curve can then be used in the available defect assessment procedures or Finite Element Analysis to evaluate crack driving force like J-integral or crack tip opening displacement. A typical



comparison of the J-integral obtained from the equivalent stress-strain relationship to that obtained by using the conventional approach is indicated in the above figure.

area of integrity assessment of cracked welded fracture mechanics specimens. However, no such comprehensive work

assessment of cracked welded pipe joints. Conventional welding procedures designed to achieve over-match that provides a shielding effect on the base metal. This is important in so far as the weld properties can be less well controlled as compared to the base metal. In order to understand the mechanics and development of plasticity pattern in cracked welded pipes priority was given to over-matched welded pipes. Since limit load solutions are already available for various

In past two decades considerable work has been done in the has yet been done regarding the

are

From the experience gained at the specimen level it can be concluded that for an over-matched weld plasticity development is more in the base material. Since the adjacent base material dissipate a large amount of energy during plastic deformation the standard ASTM (E 813-88) procedure is likely to give much higher fracture toughness. This higher "apparent fracture toughness" does not represent the real loading which the crack tip is subjected to. As a

result, there is a need to develop new solutions of plastic eta functions  $(\eta_n)$  for cracked bi-metallic pipe so that correct fracture toughness can be obtained experimentally. In an under matched weld plastic strains get concentrated within the weld and thus the crack driving force is high. Narrow gap weld can reduce this mismatch effect if under matching is mild. However if under matching is too severe than a very high constraint would be at the crack tip due to the adjacent harder material and thin welds



specimens, efforts were made to develop the limit load solution of cracked welded pipe joints from those of bi-metallic specimens. The basic issue in developing the limit load solution is to make equivalence between the geometrical characteristics of a pipe and those of a specimen. Using an analogy with three point bend specimen, the limit load of a welded pipe having a through wall circumferential crack has been obtained.

would be more detrimental and may show a brittle failure.

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## INTRODUCTION

This chapter covers various research and developmental activities pertaining to the online integrity monitoring of the components of Indian nuclear reactors. These systems help in advance planning of programmes for in-service inspection, maintenance and life extension of the ageing components. Besides, such systems also help in taking timely action against any abnormal behaviours by detecting them in advance.

Several systems for online monitoring have been developed and deployed in field. A real time creep and fatigue monitoring computational code BOSSES has also been developed.

## 15.1 ON-LINE HEALTH MONITORING OF TURBINE BLADES

The rotating blades in a steam turbine are one of the most critical components working round the clock in hostile steam environment. Due to the demand of high duty cycle and adverse working environment, there are number of



blade failures reported all over the world. Such incidences are a safety hazard as well as economic burden on industry. Presently, there is no commercial diagnostic package available for monitoring the health of the turbine blades during operation. With a view to fulfill this industrial requirement, BARC in collaboration with NPCIL and thermal power plants in private sector have developed a blade diagnostic system. The system is non-intrusive, i.e. it can be implemented in an operating plant and is cost effective. The system is tuned to detect blade vibration caused by internal excitation during off design working condition or due to other design deviation that affect the blade dynamics. The technique has been implemented and validated in nuclear and thermal power plants.

## 15.2 ONLINE CONDITION MONITORING SYSTEM FOR THE MAIN COOLANT PUMP NO-3 OF DHRUVA REACTOR

Main coolant pumps of Dhruva research reactor in Trombay, are in continuous use since 1985 when the reactor was made critical. Over these years, these pumps have served the plant



very well. However, in the recent times, there have been a couple of reactor outages due to non-availability of these pumps. Pump breakdown has largely been due to ageing related deteriorations. One of the early indicators of ageing in any machinery is its vibrations history. Realising the importance of the continuous availability of the pumps for the reactor operation, an online vibration monitoring system has been installed on pump No.3 in September 2004. The mounted accelerometers on the bearings of the motor and the flywheel are connected to a PC-based data acquisition and analysis system. The system keeps a trend of the key frequency components in the vibration signal that help in monitoring the health of the bearing and the shaft. On two occasions, the system has detected premature failure in the bearing. The plant has made good use of the system in the way it takes advance action to replace the faulty bearing. The system is operating satisfactorily without any breakdown so far. Based on the good feedback, action has been taken to install similar system on all the three pumps.

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## 15.3 REAL TIME CREEP-FATIGUE MONITORING SYSTEM

Real time creep-fatigue monitoring system is a regulatory requirement in power plants of many countries. Such a system helps during in-service inspection, maintenance and life extension program to generate database for future design.

A finite element based in-house system BOSSES has been developed for such purpose. The system is in operation in heavy water plants and NTPC Dadri plant for many years. The system has three main modules

- data acquisition,
- finite element solver and
- visualization.







Above figures show some of the computed results by this system for HRH pipe bend at NTPC plant over the period of monitoring.



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# 16. ENGINEERING IN-SERVICE INSPECTION & REMOTE AUTOMATED TOOLS

#### INTRODUCTION

This chapter summarises several technological developments that have taken place in the field of in-service inspection of the components, repair of the faulty components and handling of emergency situations in nuclear reactors and other nuclear facilities. All these tools have been developed to meet the challenging requirements of remote operation under the harsh environment of radiation, temperature and pressure. In addition, fail safe design and easy installation and operation have also been the guiding philosophies to meet the requirements of inherent safety and keeping the reactor outages and the manrem consumption to a minimum. Some of the important developments are channel inspection tool "BARCIS", radiation-resistant camera, In-situ Property Measurement tool (IProM), CHannel Isolation Plug (CHIP), remotely operated hydraulic trolley with manipulator (ROHYTAM) and water hydraulic manipulators.

## 16.1 BARC CHANNEL INSPECTION SYSTEM (BARCIS) FOR KAPS

BARC Channel Inspection System (BARCIS) is an indigenous tool developed for in service inspection of coolant channels of Pressurised Heavy Water Reactors (PHWR). BARCIS system for KAPS was developed and supplied to KAPS in August 2003. This system is being used for in-service inspection in units 1 & 2 of KAPS.



BARCIS drive mechanism in fuelling machine vault of KAPS

BARCIS-KAPS system has been designed to be lightweight for ease of installation and hence considerably reducing total Man Rem expenditure. The new system will also facilitate accessibility to top row channels by the BARCIS. Till date four BARCIS systems have been supplied to NPCIL and more than 900 coolant channels have been inspected.

Technique for inspection of rolled joint has been developed and is being used to qualify the rolled joint spools in unit 1 of MAPS before re-tubing. This technique is also being integrated in the BARCIS system and will be available as an inspection option in the near future.

#### Improved contingency handling equipment for BARCIS

A device for handling leakages for full coolant channel bore has been developed and integrated as part of BARCIS contingency handling equipment. The device is very easy to handle and can arrest a full bore leak in 30 seconds.



Full bore leak arresting device

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#### 16.2 RADIATION-RESISTANT CCTV CAMERA

CCTV cameras are used extensively in inaccessible areas for monitoring as well as for visual inspection. Commonly available CCD cameras have limited radiation life (in the order of few thousand rads). Hence for application in highly radioactive areas, radiation-resistant CCTV cameras are needed.

To fulfill this requirement, radiation-resistant CCTV camera has been developed indigenously. This camera has been designed for inspection of coolant channels of PHWRs. The camera has a radiation life of 100 Mega-Rads. This camera system has been successfully used in calandria tube inspection (during EMCCR) of units 1&2 of MAPS. The camera was also deployed for visual inspection of channel K11 in the unit 1 of Kaiga atomic power station.



Use of this camera in various waste management facilities is being considered. The camera is also being adopted for calandria vault inspection of unit 1 of KAPS.

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#### 16.3 IN-SITU PROPERTY MEASUREMENT SYSTEM (IProMS)

A hydraulically-operated system called In-situ Property Measurement System (IProMS) has been developed to estimate the mechanical properties of structural components. The technique is based on ball indentation, in which, multiple indentation cycles are made on the component using a spherical indenter. A minimum of eight cycles is used and each cycle consists of indentation, partial unload and reload sequences. The load and corresponding deformation are recorded during the test. An analysis of the data recorded gives an estimate of the yield stress, true stress-true plastic strain curve, UTS, strain hardening exponent and Brinell hardness of the material.

User-friendly software has been developed for the purpose of data acquisition, analysis and presentation of results. The use of hydraulic method for application of load and LVDT for measurement of deformation has enabled reduction in the size of tool head such that it can be used for measurement of mechanical properties of pressure tube by doing indentation on the inside surface. The same tool head can also be used for indentation on the external surface of pipes or on specimens, with suitable clamping.

A large number of experimental trials have been carried out on specimens prepared from material with known mechanical properties to establish the feasibility of using hydraulic application of load for ball indentation trials. Close comparison has been observed between the mechanical properties estimated from conventional test and ball indentation tests. IProMS has been designed based on feedback from these trials.





IProMS tool head.

The load deformation curve and the flow curve, estimated for a typical material during the feasibility studies are shown in figures.



#### 16.4 HYDRAULIC INSIDE DIAMETER MEASUREMENT SYSTEM

Inside diameter of pressure tube of coolant channel of operating Indian Pressurized Heavy Water Reactors (PHWRs) increases due to irradiation and creep under operating conditions. A hydraulically-operated three legged micrometer type inside diameter measuring instrument, named HYRIM, was developed for measurement of inside diameter of pressure tubes in wet channel condition during in-service inspection (ISI) activities. It has a measurement accuracy of 0.1 mm and an inspection range of 82 to 92 mm.

The system consists of an inspection head and a measuring

station hydraulically connected by a long hose. Measuring station consists of piston, cylinder, lead screw and hand wheel. As the hand wheel is rotated, the piston of measuring station moves forward. The volume of liquid moved by this piston enters and creates equivalent space in the cylinder of the inspection head. The tapered piston rod in the inspection head push out the radial legs until the legs are in contact with the inside diameter of the tube. The final contact point can be established by a sudden rise in force i.e., pressure in closed volume. The linear scale at the measuring station is graduated in terms of diameter with respect to axial movement. The linear scale is fixable in such a way that final position of hand wheel gives the direct reading in terms of inside diameter.

Qualification was done in two stages. In the 1<sup>st</sup> stage, qualification trials were carried out in dry tube without seal plug and its handling system. In 2<sup>nd</sup> stage, qualification trials were carried out in water filled pressure tube at the coolant channel mock-up facility at KAPS, NPCIL. Feasibility of operating procedure, compatibility with the existing ISI seal plug and drive tubes and accuracy of measurement was checked and found to be satisfactory during qualification trials.

After successful qualification trials, the system was found to be suitable for its use in the reactor. As part of further development work, automation of the operating system is being done to increase speed of operation and to reduce radiation dose.





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### 16.5 DESIGN AND EVOLUTION OF MAN-REM SAVING TOOLS FOR OPERATING PHWRs

In 220 MW(e) PHWR there are 306 coolant channels. Pressure tube material for these coolant channels is zircaloy-2 / Zr-2.5 Nb. Coolant channel is the important component of the Primary Heat Transport (PHT) System. Heavy Water (D<sub>2</sub>O) circulates at high pressure (100 kg/cm<sup>2</sup>) and high temperature (565 K) through PHT as coolant. Presence of radiation deteriorates the material properties. Each coolant channel has seal plug on either side to facilitate refuelling. To enhance the performance of the PHWRs, life management programme for coolant channels plays a major role. For satisfactory service life of the coolant channels it is required to carry out periodic inspection and maintenance. In the past, technique/tools have been used for this purpose which need channel in dry condition. Various techniques have been developed to monitor different parameters. For these techniques existing FMs are used which makes it possible to do the job remotely and quickly. These techniques have been described in the following paragraphs.

## Axial Creep Measurement of coolant channels

A technique is being perfected for non-contact axial creep measurement of coolant channel. Laser-based sensor and ultrasonic sensor have been tested for this and later has been

found suitable. A tool is developed which can hold the sensor in the FM. Software has been developed for acquisition and processing the data and to give creep value in lattice shape of the reactor. The scheme has been tried in one of the units on experimental basis. Further refinement is in progress.

## Special Sealing Plug for BARC Channel Inspection System (BARCIS)

Periodic In-Service Inspection (ISI) of the coolant channels is carried out to monitor the health of the coolant channels in PHWRs. BARCIS has been developed to carry out the ISI of the coolant channels. The special seal plug enables to install the inspection head remotely in the channel, to be inspected, by FM and also allows the entry of drive tube to be connected to inspection head for transferring the collected data by inspection





head to outside. Thus special seal plug plays a major role in ISI operation. Different versions of plug have been evolved and used for ISI of more than 700 channels in 220 MW (e) PHWRs.

The entire scraping operation of one channel can be completed in about 4-5 hours. This tool has become standard tool for taking sliver samples. WEST has been successfully used for scraping channels in different units of Indian PHWRs.

Recently Mark IV design has been developed having advanced features for better reliability and safety. This design is mature, sturdy, operator friendly and saves further man-rem expenditure and time. Maintenance frequency required is also less. MK-IV design of the plug has been used for ISI of 35 channels at MAPS successfully.

#### WEt Scraping Tool (WEST)

Hydrogen concentration is an important parameter that must be assessed to evaluate the fitness for

service of pressure tubes. Earlier methods required removal of channel or required channel in dry condition. To increase the productivity, to minimize man-rem expenditure and reactor shut down time a scheme was worked out to convert the existing tool such that it can be used to take the sliver samples from water filled channels using fuelling machines. This has been accomplished by incorporating in the tool some additional features like miniature, special bearings for self orientation at 12'O clock position, compatibility with fuel, provision of a piston to transmit the fuelling machine ram force and motion to the carriage of the tool etc.. The technique was named as WEt Scraping Tool (WEST). 'C' Ram force of FM is applied to move the carriage at desired speed taking proper sliver cut. Scheme is such that three WEST tools are used which enables taking of three sliver samples from three axial locations from each channel, in a single visit of fuelling machine.



#### Leaky Seal Plug identification by using Acoustic Emission Technique

After installation of seal plug, fuelling machine ensures that leakage is within acceptable limit of 20 gms/min. After some time this leakage is expected to reduce to less than 10 gms/day. Considering large number of seal plug and rise in trium activity due to small leakage also routine monitoring of plugs is necessary. For this purpose an acoustic emission based leakage monitoring system has been developed in collaboration with other groups within BARC. The leakage through seal defect gives rise to acoustic signal, which is picked up by an acoustic sensor and signal is processed to assess the leakage. A tool mounted with sensors is held by fuelling machine. Fuelling machine takes this tool to reactor face and contact is established for monitoring the signal. The data acquisition and analysis system is kept in



CHIP is installed by FM in the end fitting. Some of the components are removed to make the closure seal face approachable. After repair components are installed back and CHIP is removed by FM. 'O' ring seals are used for sealing. Safety features have been incorporated. CHIP has become a standard tool for all operating power stations. This along with EFBA were very helpful to repair the heavily damage face in C-13 channel of KAPS-2. Job could be completed in very short time.

accessible area. Laboratory experiments indicated that a leakage of 100 cc/hour might be possible to measure. The technique was used once in KAPS unit –2 and in NAPS unit -2 successfully. Attempt is being made to develop the scheme for confirming the leakage from the detected leaky seal plugs.

# Pressure tube I.D. Measurement using Three Point Micrometer

A scheme for measuring coolant channel inside diameter in water filled condition for 220MW(e) PHWR was



e v o l v e d jointly with Nuclear Power Corporation of India. The tool was qualified and used for measuring

## Channel Isolation Plug (CHIP)

During operation of closure seal face gets damaged. To enable this repair in water filled channel, a concept of Channel Isolation Plug was developed for isolating the closure seal face and make it accessible for repair. This plug has been supplied to almost all power stations. the reactor, some time

the inside diameter of some of the channels in KAPS unit -2 in the year 2002. The measurement did not indicate ballooning of pressure tubes. This tool was recently used again in KAPS.



#### End Fitting Blanking Assembly (EFBA)

EFBA has been developed to block the excessive leakage from the seal plug temporarily till the next opportunity for the repair. This works as a secondary sealing device. An 'O' ring seal is used to be located, behind the seal plug on an adapter. This is retained by split rings and retainer sleeve. It has got the provision for releasing the pressure before removal. This has been qualified by analysis as well as experiments. The dimensions are optimised such that neighboring channels can be refueled. Installation takes about 1 minute time. This is being used by the operating PHWRs.

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# 16.6 REMOTELY OPERATED HYDRAULIC TROLLEY ALONG WITH MANIPULATOR AND DECOMMISSIONING TECHNOLOGY DEVELOPMENT

Remotely operated Hydraulic Trolley along with 6 DOF Hydraulic Manipulator (ROHYTAM) has been developed for handling some of the emergencies, repair and decommissioning work in PHWRs and other nuclear facilities. ROHYTAM is electro hydraulic servo controlled master-slave manipulator mounted on an

autonomous remotely operated mobile trolley. ROHYTAM is operated remotely with CCTV video feedback. Wireless PLC-based control system of ROHYTAM is with in built safety features. Manipulator has master-slave controls where as trolley is controlled through joystick. Two micro controllers, one at trolley & other at control cabin are interconnected as well as connected to computer & control console. ROHYTAM can be controlled through control console or soft master manipulator and soft panel.

Payload capacity of ROHYTAM at gripper is 50 Kg and at fore arm is 500kg. The reach of the ROHYTAM is up to 3 meters in both horizontal and vertical directions. ROHYTAM has nine electro hydraulic servo linear/rotary drives. ROHYTAM and various remote operations demonstrated by the ROHYTAM are shown in photographs. The various types of end effectors of ROHYTAM provide following remote working capabilities.

- Handling of fuel bundles of PHWR
- Manual override operation of direction control valves of MAPS fuelling machine
- Nut splitting operation with integrating nut splitter
- Dismantling greylock joint nut using hydraulic torque wrench
- Cleaning floors in an inaccessible area using vacuum cleaner
- Operations on concrete structures by integrating concrete rotary saw
- Operations on concrete structures by integrating concrete hammer
- Operations on concrete structures by integrating concrete drill
- Operations on concrete structures by integrating concrete splitter.
- Operations on concrete structures by integrating concrete spreader
- Operations by integrating tube cutter



ROHYTAM

#### **Remote Working Capabilities**

Operations by integrating crusher

Modified ROHYTAM, which does not have any external hoses and cables, is under development. The new design is lightweight and with improved work envelope. The modular design provides flexibility and fully enveloped and streamlined construction facilitates easy decontamination of ROHYTAM. The laser guidance system will be used for precise positioning of the manipulator arms, detecting obstacle and mapping the working area. The modified ROHYTAM can be further extended for under water operations.



Fuel bundle handling



Nut splitting operation



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## 16.7 DEVELOPMENT OF WATER HYDRAULIC SERVO ACTUATORS AND UNDER WATER MANIPULATORS

Water hydraulic servo actuators are being developed in BARC for applications like under water manipulators used for tele-operations and repair work in underwater radioactive areas like Calandria vault, fuel storage bay, reactor vessel, deep sea applications etc. Water hydraulics provides superior power transmission and control components to other power technologies. In such applications it is required to use precise position, velocity and force control system. Water hydraulic servo linear and rotary actuators have been developed for such application. Using these actuators Water Hydraulic Under Water Manipulators are being developed.

Water hydraulic test facility has been established in fluid power lab BARC for development and testing of Water hydraulic servo actuators. Various standard experimental tests for evaluating a servo linear actuator have been performed.
Engineering - in - service inspection & remote automated tools



Water hydraulic servo linear actuator in test



Combined Double Axis Servo Rotary Actuator & its Cross Sectional View developed for Under Water Manipulator



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## **17. ROLE OF NON-DESTRUCTIVE EXAMINATION IN NUCLEAR INDUSTRY**

## INTRODUCTION

Non-Destructive Examination (NDE) methods are traditionally employed for detection, location and characterisation of flaws in engineering components. These three factors form vital inputs for structural integrity assessment of engineering components, their residual life estimation and extension beyond the design life.

This chapter deals with the development of NDT methods for in-service inspection of reactor core components and characterisation of material properties.

The past few years have witnessed a surge in the application of NDE methods for characterization of material properties as well. Extensive research is being carried out worldwide to develop NDE method(s) for measurement of residual stress. In Department of Atomic Energy, NDE methods are employed for Quality Control during fabrication of nuclear fuel and reactor core components and Ageing Management of nuclear facilities like Power and Research Reactors, Heavy Water Plants and Waste Management Plants. In future there will be demands on development of NDE methods for in-service inspection of new generation reactors like AHWR, VVER (PWR) and PFBR. The NDE technology in the Department has evolved over the years from the conventional techniques like penetrant testing, radiography, ultrasonic, etc. to more advanced techniques like ultrasonic imaging, acoustic emission, barkaussen noise, magnetic flux leakage, neutron radiography and so on. The impetus for this development has come from the stringent quality requirement of reactor core components and the need to monitor their degradation periodically by NDE. The past decadeor so has witnessed the development of two very successful indigenous in-service inspection tools, one for the coolant channels of PHWRs and the other for core shroud of BWR.

There exist numerous challenges during in-service inspection of critical components, which are required to be addressed. Suitable NDE methods like replication, optical profilometry and ultrasonic spectroscopy need to be developed to classify the flaws in PHWR pressure tubes into sharp or blunt. Existing NDE methods for detection of hydride blisters in coolant channels will be fine-tuned and new methods like critical angle measurement and lamb waves will be studied. One of the most challenging tasks would be the measurement of hydrogen in the pressure tube by NDE. Ultrasonic and Eddy Current based methods hold good promise in this regard. In Pressure Vessel type reactors (BWRs, PWRs), the emphasis will be on development of testing techniques for pressure vessels, nozzles and pipelines. Advanced technology like ultrasonic phased array needs to be developed and employed for faster and reliable inspection. Ultrasonic Guided Wave technology will play a crucial role in examination of inaccessible pipelines in these reactors. Extensive analytical and experimental work using magnetic and ultrasonic methods will be required for assessment of irradiation damage in core components.

In future, the key research areas in the field of NDE will be

towards improvements in flaw and material characterization capabilities and broadening the scope of NDE so that it can be applied on inaccessible components operating in hostile environments. The key drivers in achieving these objectives will be acoustic-based techniques like ultrasonics and acoustic emission, electro-magnetic based techniques like eddy current and magnetic-based techniques like barkaussen noise, magnetic flux leakage, 3MA approach, etc. The key goals will be to develop and implement NDE methods and techniques that are more sensitive, accurate, reliable, informative, cost-effective and fast so that the inspection can be carried out effectively in the shortest possible time.

## 17.1 DEVELOPMENT OF NDT METHODS FOR IN-SERVICE INSPECTION OF REACTOR CORE COMPONENTS

The coolant channel, comprising of a pressure tube, surrounding calandria tube and a pair or two of garter spring spacers, is at the core of Pressurized Heavy Water Reactors (PHWRs). The integrity of zirconium alloy pressure tubes (Zircaloy-2 / Zr-2.5% Nb) is central to the safety of PHWR. Non-destructive Examination (NDE) of coolant channels during periodic in-service inspection plays a crucial role in this regard. It provides information on presence or absence of flaws in pressure tube, the location of garter spring spacers and the gap between the pressure tube and calandria tube. These inputs are vital to the plant operators and the regulatory authorities for taking decision regarding the continued operation of coolant channels. The non-destructive examination of coolant channels employ UlTrasonic (UT) and Eddy Current Testing (ECT) methods. These sensors are mounted on an inspection head of a semi-automated tool, BARCIS (BARC Channel Inspection System). Hundreds of coolant channels in all the operating PHWRs of our country have been examined by BARCIS.

The International Atomic Energy Agency (IAEA) initiated a Coordinated Research Programme (CRP) on Intercomparison of Techniques for Pressure Tube Inspection and Diagnostics involving countries, which include Canada, Korea, China, Argentina, Romania and India. The objective of this CRP was to intercompare inspection and diagnostics techniques for pressure tubes during their service, as being used and developed by participating countries. Phase 1 of this CRP, which deals with 'Flaw Characterization in Pressure Tubes by in-situ



Indian Pressure Tube Sample for IAEA



Non-Destructive Examination (NDE) techniques' involved round-robin transfer of pressure tube samples containing artificial flaws amongst participating laboratories. The flaws are machined in such a manner that they closely resemble the real defects of concern, like delayed hydride cracking and fretting damage due to debris and bearing pads. The flaws are also hidden by an outside cover so that the inspection personnel are unaware about the number, location and type of flaws in the sample. As part of this CRP, pressure tube samples from all the participating countries were examined. Apart from the conventional ultrasonic angle beam pulse-echo technique, new methodologies such as angle beam pitch-catch technique and normal beam pulse-echo technique, are used for flaw detection. This approach ensured that flaws in all orientations (axial, circumferential, equiaxed, inclined) on inside and outside surface of pressure tube are detected with high reliability. Advanced techniques like ultrasonic imaging and time-of-flight based sizing techniques are used to accurately characterise the flaws. At the end of Phase 1, the results of NDE examination by participating countries will be compared with the true dimensions of the flaws (found out by profilometry or

destructive means) for all the samples. This will help to identify most accurate and reliable methods of characterization for different kinds of flaws in pressure tubes and also define areas of future R&D to fully meet the flaw characterization requirements of PHWRs.







#### Ultrasonic B-scan Images of Flaws in Pressure Tube Samples

One of the most challenging tasks during in-service inspection of pressure tubes is the detection of zirconium hydride blisters. The conventional ultrasonic testing technique based on reflection principle cannot detect the blister as there is no significant acoustic impedance (product of sound velocity and density) difference between zircaloy and zirconium hydride. Ultrasonic examination techniques based on measurement of longitudinal to shear velocity ratio and time-of-flight based B & C-scan imaging were developed in the laboratory for detection of uncracked zirconium hydride blisters. When the pressure tube is examined from the ID surface using normal beam longitudinal wave, there is a reduction in time-of-travel of OD signal at the blister location as the velocity of longitudinal wave is higher in zirconium hydride as compared to zircaloy. The shift in OD signal is clearly seen in the B-scan image. The reverse happens while using shear wave. This technique was very reliable for blisters of the order of a millimeter. In order to detect blisters of much finer depth (0.2 mm), existing techniques are being refined. New techniques based on critical angle and spectral analysis are also being attempted in this regard.



Development of Non-Destructive Examination Methods for Core Shroud of Boiling Water Reactors (BWR)

In the early 1990s, several boiling water reactors all over the world revealed InterGranular Stress Corrosion Cracking (IGSCC) in the heat affected zones of core shroud weld. The core shroud is a cylindrical vessel (inside the pressure vessel) that surrounds the core of BWR. Its primary function is to separate the upward flowing primary coolant from the downward flowing feed-water. It also provides structural support to the core and maintains its geometry. The core shroud in BWRs at Tarapur is made of AISI 304 grade Stainless Steel. This material, if sensitized during welding, is prone to IGSCC attack due to the presence of oxygenated water chemistry under BWR operating conditions. In view of the above, it was decided to carry out in-service inspection of core shroud welds. The core shroud at Tarapur has nine circumferential welds (H1 to H10), out of which only the



top four are accessible for non-destructive examination. The examination was required to be carried out remotely and under-water from inside surface of the shroud using special manipulators from top of the reactor.

Non-destructive examination methods based on visual and ultrasonic examination were developed, standardized and implemented for carrying out in-service inspection of core shroud welds. While the objective of visual examination is

to detect IGSCC starting from ID surface, ultrasonic examination was employed to detect cracks starting from OD as well as ID and then characterize them with respect to their length and depth. In order to qualify the effectiveness of visual examination, Sensitivity Resolution Contrast Standard (SRCS) Panels were fabricated. The panel consists of a stainless steel plate with fine wires (15 to 70 microns diameter) running across its length and width. The SRCS panel is suspended in the reactor pools and the camera is kept at a distance and angle that is encountered during the examination of core shroud. If the finest wire on the SRCS panel is seen, then the visual examination system is qualified for examination of core shroud. For ultrasonic examination, probe holding mechanisms



IGSCC in Test Plate

were designed and fabricated for different weld geometry. One such mechanism is the Carriage for Retracting and



were examined in a phased manner. No IGSCC was revealed in any of the four welds of core shroud at Tarapur.

give the feedback to the operator on the radial direction of

ultrasonic beam. The top four welds of both the BWRs at Tarapur

The phenomenon of IGSCC in BWRs is not just limited to Core Shroud, but several other systems like primary pipelines. Ultrasonic testing is the most commonly employed NDE method for detection and sizing of IGSCC. Since IGSCC is characterized by extensive branching, has rough surface and is filled with corrosion products, the conventional amplitude-based ultrasonic testing techniques undersizes IGSCC. In order to overcome this limitation, extensive study based on advanced time-offlight based ultrasonic testing techniques was carried out. IGSCC of varying depths were generated in the laboratory on 25 mm thick stainless

CART Assembly for Core Shroud

steel plates by accelerated tests. The cracks were subsequently sized by conventional and advanced methods.

Advancing of Transducer (CART). The CART carries three ultrasonic transducers, one each for angle beam examination of top and bottom heat affected zone and one normal beam to

Actual depth of IGSCC (mm)	Depth by Amplitude Comparison (mm)	Depth by Time-af-Flight (mm)
6.7	2.1	6.6
7.0	1.5	7.0
4.3	1.9	4.9
5.8	1.2	5.8

Comparison of Conventional (Amplitude based) and Advanced (Time-of Flight based) Technique for IGSCC sizing

The study indicated that the time-of-flight based techniques are far accurate as compared to the conventional amplitude based sizing techniques.

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## 17.2 DEVELOPMENT OF NDT METHODS FOR CHARACTERISATION OF MATERIAL PROPERTIES

Non-destructive examination (NDE) methods hold good potential to characterize microstructural and mechanical properties of components. They can be employed to (i) determine various metallurgical properties like grain size, inclusion content, elastic modulus, tensile strength, fracture toughness, etc., (ii) qualify various thermo-mechanical processing treatments during fabrication and (iii) assess the service-induced damage on components due to various degradation mechanisms like fatigue, creep, corrosion, thermal and neutron embrittelement, etc. Numerous studies have been carried out in the Department to characterize material properties on zirconium alloys, stainless steels, metallic and oxide fuel, by employing NDE methods based on ultrasonic, eddy current, magnetic and acoustic emission technique. Two of the significant studies in this field are described below.

### Qualification of β heat treatment of uranium rods

Uranium rods are used as fuel elements in nuclear research reactors. One of the critical steps in the fabrication route of these elements is the  $\beta$  heat treatment of uranium rods after hot rolling. The rolling operation introduces a [010] texture along

the length of fuel rod, which is responsible for thermal cycling and irradiation growth during reactor operation. The purpose of  $\beta$  heat treatment is to randomize the preferential crystallographic texture and provide dimensional stability to the fuel element in the reactor. Conventionally,  $\beta$  heat treatment of uranium rods is gualified by thermal cyclic growth test, during which the sample is subjected to 1000 cycles from 0°C to 550°C and back. The test is very time consuming and can be performed only on limited number of samples from a particular batch. Hence, there was a need to develop an NDE technique by means of which  $\beta$  heat treatment can be qualified nondestructively. Ultrasonic testing technique based on velocity, attenuation and backscatter measurement was developed to qualify the  $\beta$  heat treatment. The study carried out on samples corresponding to different conditions indicated that the ultarsonic velocity in axial and radial direction for properly heat treated rod is almost identical, while significant difference was observed in velocity in these two directions, if the heat treatment is not satisfactory. This technique has been employed to qualify the heat treatment of few batches of uranium rods as well as to qualify any changes made in the heat treatment process.



Ultrasonic test parameters for different conditions of uranium rod

### Ageing degradation in alloy 625

Nickel base alloy 625 is extensively used in chemical industries, like heavy water plants, for its high temperature strength and corrosion resistance. Ageing degradation during long term exposure at high temperature due to precipitation of  $Ni_3Nb / Ni_2Mo$  phase causes reduction in ductility and fracture toughness. Study carried out to correlate ultrasonic testing parameters viz. velocity and attenuation with the degree of age hardening in Alloy 625 has shown excellent results. The

measurements were performed on samples with high temperature service exposures (approx. 500°C to 600°C) for 25,000, 50,000 and 1,00,000 hrs. The results indicated that in the service aged samples, both velocity and attenuation was higher as compared to virgin sample.

Ultrasonic Test Parameters for Virgin & Service Degraded Alloy 625

The samples were then subjected to heat treatment (700°C, 6 hrs.) in order to regain the mechanical properties. Consequent to this heat treatment the velocity and attenuation reduces and approaches that of the virgin sample. The study established that the onset of age hardening in this alloy as well as the recovery of its mechanical properties after heat treatment can be detected by monitoring ultrasonic velocity and attenuation.

The importance of non-destructive characterization of material properties cannot be overemphasized. This approach overcomes the limitation of the conventional 'coupon based method' and can be employed during fabrication of components as well as to get a realistic information of material property degradation during service. Extensive analytical and experimental studies are being carried out in the Department to develop NDE methods for assessment of service-induced degradation of material properties in nuclear components.

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# 18. ENGINEERING - COMPONENT MANUFACTURING, TESTING & TECHNOLOGY TRANSFER

### INTRODUCTION

This chapter summarises some of the work done related to component manufacturing, development of facility for testing of the reactor components, development of components for import substitution and development of several spin-off technologies to be transferred to public / private enterprises. Manufacture of components for both power reactors and research reactors are being done in the manufacturing facility built in BARC over a period of time. Some of the important components manufactured are fuelling machine heads, sensor stop & pusher assemblies for fuelling machine, seal discs, components for control rod guide tubes of BWR (TAPS). A high temperature and Pressure and Temperature Cycling Facility (PTCF) have been developed in BARC for testing of reactor components like valves, pipe and pipe fittings, couplings, thermal and pressure sensors which are occasionally subjected to high temperature and pressure.

Special purpose machines for reactor maintenance, fast acting valve and other special purpose valves have also been developed. The know how of several technologies has been transferred to outside parties which include a hydraulic circuit with acceleration deceleration valve, some water lubricated bearings etc.

## 18.1 MANUFACTURING OF COMPONENTS OF POWER / RESEARCH REACTOR

Manufacture of components for both power reactors and research reactors are being done in the manufacturing facility built over a period in BARC. This research centre pioneered in the field of process development and manufacture of the several critical components of the nuclear reactors like fuelling machine heads, sensor stop & pusher assemblies for fuelling machine, seal discs, components for control rod guide tubes of BWR (TAPS) etc.

### Fuelling machine components for 540 MWe PHWR

Technology for the manufacture of ram housing assemblies, pressure housings and end covers for fuelling machines of 540 MWe PHWRs has been developed. In addition, technology for electrolysing sealing plug jaws for use on these reactors has been developed and matured to a level of batch production of this component.



Ram Housing assembly



Pressure Housing & End cover

### End fitting body and liner tube for 540 MWe PHWR

End Fitting Bodies and Liner Tubes (1700 each) required for TAPS 3 & 4 have been manufactured. The tubular shaped end fitting body, with a maximum outer diameter of 188.5 mm and length of 2516 mm is made out of solid AISI 403 type stainless steel forging. The raw material in the form of solid forging undergoes 20 different stages of machining operations. The components are subjected to thermal stress relief operation for relieving residual stresses induced during the manufacturing operation. Ultrasonic flaw detection, magnetic particle testing and hydrostatic testing are being carried out at appropriate stages to qualify these components.

The liner tube is made out of a seamless stainless steel tube AISI SS 410 grade. It has 111.86 mm outer diameter, 104 mm bore and 2135 mm length, demanding IT7/ IT8 grade dimensional tolerances, with stringent geometrical features like cylindricity and straightness of the order of 0.01 mm / 100 mm.

All the tooling, the manufacturing process and the qualification procedures required for these components have been developed in-house.



End Fitting Bodies



### Control Rod & Shut off Rod Mechanisms

Indigenous development of manufacturing of control rod and shut-off rod drive mechanisms involved development of procedures to weld dissimilar alloys, procedures for machining of intricate, high precision components and rigorous destructive and non-destructive quality assurance plans. Many important sub-systems of this equipment were manufactured and delivered.



Control rod drive mechanism housing

Critical components for other reactors

250 Seal Shield Plug Assemblies of improved design, which will replace the existing assemblies were manufactured for the DHRUVA research reactor. Among the other critical components

related to power/ research reactors manufactured in BARC include bearing sleeve for sodium pump for Fast Breeder Test Reactor (FBTR) and sleeve for Prototype Fast Breeder Reactor (PFBR).

### Components for AHWR critical facility

#### Reactor tank

The Reactor tank is made of aluminium of grade ASME SB-209 / Al-1060. The outside diameter of tank is 3320 mm. The shell thickness is 10 mm and the height is 5000 mm. The thickness of the base plate is 25 mm which is fabricated from 40 mm thick Al-1060 plate. The welding is carried out as per the requirements of ASME section III. About 50-meter length of welding of 10 mm thick



Reactor tank for AHWR critical facility (Matl. Al. 1060)

p l a t e s (including both the longitudinal and circumferential welds) and 4-meter length of welding of 40 mm thick plate are involved. All the welds were qualified by 100% radiography. Exhaustive preparations Jigs / fixtures / pre-heating arrangements etc. are made to execute this large vessel. For this purpose, Vertical Welding (3G-position) with Twin Welders Technique was developed, established and used.

#### **AHWR Critical Facility Square Box**

The square box, made of Austenitic SS 304L material, is in direct communication with reactor tank and forms the outer





boundary of the Critical Facility. The lattice girder assembly, placed inside the square box, carries and positions the fuel assembly. The square box is a fabricated leak tight enclosure of overall size 4610 mm sq x 1550 mm height. The top and bottom plates are 20 mm thick and the side plates are 6 mm thick. Neoprene gasket is used to achieve leak-tightness between the top plate and the square box. The bottom plate has a circular opening of 3300 mm diameter. A ring of 3470 mm diameter welded concentric to this opening facilitates its connection to the reactor tank through an elastomer seal ring.

The top plate has a circular opening of 3350 mm diameter to allow access to the reactor tank. A revolving floor supported on a bearing closes this opening. An ingenuously designed oil seal comprising of two metallic ring shells of appropriate diameter and height welded to the top plate of the square box and another similar ring shell welded to the revolving floor, isolates the atmosphere from the environment inside the box. The lattice girder system carries AHWR and PHWR test fuel assemblies, with flexibility of configuring the reactor core at any desired square pitch between 206 mm and 286 mm. All lattice positions within the variable square pitch are accessible though a set of 4 flanged, oblong openings on the revolving floor. These openings are provided with appropriate closures to ensure leak tightness and are positioned and sized in such a way that rotating the revolving floor to the required extent accesses all lattice positions in the Reactor Core.

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## 18.2 DEVELOPMENT OF PRESSURE AND TEMPERATURE CYCLING FACILITY (PTCF)

Reactor components like valves, pipe & tube fittings, couplings, thermal & pressure sensors are occasionally subjected to high pressure and high temperature cycling tests to satisfy qualification standards such as BS 4368. It was observed that even reputed Indian manufacturers and test agencies were not



equipped for testing these items at high temperature and pressure conditions specified for reactor applications. Therefore the need was felt to develop an in-house high pressure & temperature test facility.

The Pressure and Temperature Cycling Facility (PTCF), which has been developed for this purpose, is a mobile, rig-type, compact, high pressure and temperature test facility. It provides a test platform for evaluation of process instruments, components and small equipment designed for use at PHWR/AHWR process



conditions. The facility can be operated either in high pressure or in high temperature cycling mode depending upon the test requirements.

The process flow sheet of the facility is shown above. The overall size of the facility is 2(L)x2(W)x2.5(H) meters. The facility mainly consists of a stainless steel water storage tank and a high pressure simplex reciprocating pump, Interchanger, Loop Heater, Mixing chamber, Test Section and a Loop Cooler. The operating conditions of PTCF are indicated in the table below.



The Pressure and Temperature Cycling Facility (PTCF) has been successfully developed & installed at BARC. The temperature cycling tests to the maximum temperature of 200 deg C with 4 hours cycle period and pressure cycling tests for 10 to 100 Kg/sq.cm. amplitude at 1 Hz, have been carried out.



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### 18.3 SPECIAL PURPOSE MACHINES FOR REACTOR MAINTENANCE

### Guide Sleeve and Sensor Tip Cutting Machine for 220 MWe PHWR

A special purpose machine tool and machining process have been developed to cut the guide sleeve of the fuelling machine of 220 MWe PHWR, in case guide sleeve can't be retracted back into the fuelling machine after a refueling operation. This will be an emergency situation for a PHWR.



Guide Sleeve Cutting Machine

Material of the guide sleeve is 17-4 PH, having core hardness of 35 HRC. The surface is nitrided which gives the hardness of 60 HRC. The thickness of guide sleeve is 5.75 mm. Approach for cutting this tube is only from the other end of the reactor channel, which is at a distance of about 10 meters. The machine is in modular form for ease of its assembly and handling.

#### De-canning Tool (TAPS)

The De-canning tool has been designed and developed at CDM and successfully used to de-channel the irradiated TAPS fuel assemblies.

### Seal cutting Machine

This special purpose machine has been designed, manufactured and tested to cut the Omega seal of the steam generator, which is located in an inaccessible region. This tele-operated machine has 3 cutting heads, 3 feed drives and 3 chip suction heads. The operation of this machine is fully computerised.

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The maximum working distance of this machine is 2.7 meter and minimum diameter that can be turned is 110 mm. This is operated by central control console, which can be kept away from the working area. All the chips generated during cutting are sucked, and collected in chip strainer-cum-collector.



### Boring And Grooving Machine

Boring and grooving machine was developed to increase the bore diameter and groove diameter of some of the existing bores and grooves in the calandria tube sheet of MAPS unit-2. The boring and grooving operations were done in-situ. These operations involved enlarging the existing bore size from 120 mm to 122 mm and making of three grooves in the bores.



Boring and Grooving Machine

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#### **18.4 DEVELOPMENT OF FAST-ACTING VALVES**

Fast-acting valves are required for conducting LOCA simulation experiments in several test facilities. This valve has potential application in Secondary Shutdown Systems of AHWR, 220 MWe & 500 MWe PHWRs and specially wherever fast opening/closing applications are involved. Design of this valve is based on stored energy principle.



Since these valves are generally imported and are not customized for the experimental requirement clubbed with very high price. Hence this fast-acting valve has been developed indigenously as an import substitute.

#### Design features

- a) Pressure: 125 Bars
- b) Temperature: 300° C
- c) Opening Time: <10 m-sec
- d) Closing Time : Not important
- e) Upstream Size: 65NB
- f) Downstream Size: 150/400NB
- g) Max. upstream pressure : 116 Bar.

The valve disc is kept pressed against the seat by several disc springs over a shaft or stem. The shaft/ stem is so designed that it is a combination of elbow levers which are stretched and spring loaded when the valve is closed. By bending the elbow levers with an air actuated piston, the valve disc gets suddenly unloaded from the spring power and can be opened by the fluid pressure. The time of opening of the valve is dependent on the upstream pressure of the valve. This opening time is also dependent on several other design parameters such as:

- disc diameter,
- weight of the valve stem and associated parts and
- disc lift.



The manufacturing of this valve has been successfully completed and has been subjected to 500 open-close cycles. The valve seat is found to be perfectly leak tight at 110 bar up stream pressure. The open and closing time for this valve is less than as specified by users. The opening time is around 3 to 5 milli-second and closing time is 2.5 to 3.0 seconds.

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## 18.5 DEVELOPMENT OF WATER HYDRAULIC VALVES

These special purpose valves have been developed for controlling the water hydraulic actuators of fuelling machine.

## Differential Pressure Reducing Valve for high pressure drop (DPRV)

The DPRV is used for controlling the force developed by the fuelling machine water hydraulic actuators. A constant differential pressure is required to be maintained between pressure housing and actuators supply line for proper operation of actuators, which are located inside it. The differential pressure should remain constant even if there are fluctuations in fuelling machine pressure, pump supply pressure, actuator flow etc. This critical requirement is achieved by fast response DPRV.

In-house developed DPRVs are being used at all new reactors. Anti cavitation DPRV has also been developed. It can handle large pressure drop across it and give excellent performance. These valves are both of 30 lpm (220MWe PHWR) and 70 lpm (540MWe PHWR) capacity.



### Auto Differential Pressure control Valve (ADPCV)

This valve has been developed to control the actuator force as well as direction of motion of the actuator. It works on the principle similar to DPRV except the differential pressure for advance and retract direction of actuator can be varied by fuelling machine control program remotely. This single valve replaces large number of imported components in Ram-B and Ram-C water hydraulic circuit of fuelling machine.



### Pressure Compensated Flow Control Valve (PCFCV)

PCFCV is designed for pump pressure as well as flow control in a hydraulic circuit. It maintains a constant flow in the circuit irrespective of pump pressure or load and maintains the pump pressure equal to the load. This way it saves lot of energy as well as many imported components like control valves and controllers in water hydraulic supply system for fuelling machine. The valve is completely indigenous.



Pressure Compensated Flow Control Valve

### High Flow Servo Valve (HFSV)

High Flow Servo Valve has been designed and developed to meet the requirement of high flow (1100 lpm) with precise control (2% accuracy). The current design consists of two modules, high flow control valve and a pressure compensator. It doesn't contain any dynamic elastomeric seal.

It is under manufacture and testing of the valve has been planned at the Integral Thermal Facility (ITF), BARC.

### 3-Way Ball Valve

A passive 3-way ball valve has been designed for Emergency Core Cooling System (ECCS) of AHWR. It will divert ECCS supply to either inlet header or ECCS header depending upon which header has failed. The ball valve actuates with the help of differential pressure between two headers and same working fluid is being used to actuate the ball valve. It also has motorized actuator attachment, which is normally kept detachable. Engineering - Component Manufacturing, Testing & Technology Transfer



## 18.6 ACCELERATION DECELERATION CONTROL VALVE AND CIRCUIT FOR HYDRAULIC LIFT





The hydraulic circuit with acceleration deceleration valve is a simple and cost-effective alternative for controlling the acceleration and deceleration of hydraulic actuator while starting and stopping. The circuit can be used for getting a jerk free motion of any hydraulic actuator. Since it is compact and low cost, it finds great application in hydraulic material handling equipment where expensive proportional valves are used just to control the initial acceleration and final deceleration of the actuator.

The know-how for this technology has been transferred to M/s. Expert Equipments Pvt. Ltd., Thane, (Maharashtra).

The party has manufactured the control valve & trial run is in progress. The valve will be shortly available in the market.

18.7 ELECTROLYSING OF REACTOR CHANNEL SEALING PLUG JAWS





It is a thin dense chrome plating process and consistently gives a hardness value of RC 70 – 72 on a plating thickness of 7 – 12 microns. With Electrolizing no grinding is necessary after plating because the process lends itself to flat even distribution all over the part. This process is extensively used for elect rolizing (hard chrome plating) the reactor channel sealing plug jaws for 220 MWe and 540 MWe nuclear reactors.

The know-how of "Electrolyzing of reactor channel sealing plug jaws" has been transferred to M/s. Avasarala Automation Ltd., Bangalore.

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### 18.8 DEVELOPMENT OF INDIGENOUS WATER LUBRICATED BEARINGS

Stainless steel (AISI 440C) bearings are used in Fuelling machines of Indian PHWRs. As they have to work under chemistry controlled water' conventional lubricants cannot be used. Poor lubrication property along with the corrosive nature of water medium limits the life of these bearings. Availability of these bearings is always of concern, as they have to be imported. In view of the above, it was envisaged to develop these bearings indigenously which involved understating of material behaviour, heat treatment, load rating under water lubricated condition and wear.

With permutation and combination of parameters like ball material, cage material, clearances etc., a set of bearings has been fabricated. In-house expertise has been used in finalisation of procedures of heat treatment of the materials and the testing of bearings. Testing of these bearings in a test set-up developed in-house for one million revolutions at the operating load did not reveal any catastrophic failure. Imported bearing tested in the similar condition had failed catastrophically much before the desired life at operating load due to improper material and heat treatment of the bearing races.

Bearings have been developed for Spent Fuel Chopper at Tarapur. Installation of these bearings in the chopper has improved its availability significantly. Development of coated water lubricated bearings, where a standard carbon steel bearing is coated with suitable coating to prevent its corrosion while working under rolling contact, is under way. After extensive studies on various coating materials and testing for rolling contact fatigue in Four-Ball tester, composite coating of Electroless Nickel (EN) was selected as the candidate material. Prototype bearing with this coating has been developed and is being tested in the laboratory.



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Water lubricated bearing test set-up



# **19. NUCLEAR DESALINATION**

### INTRODUCTION

This chapter briefly covers the work recently done in the field of nuclear desalination. The 300 MWe Advanced Heavy Water Reactor (AHWR) being developed in BARC has a make up water requirement of about 360m<sup>3</sup>/day. This make up water has to be DeMineralised (DM) water. Conventional technology exists for producing DM water from raw water. But, to meet this high requirement of AHWR, the conventional method will not be cost effective. As such, there is a need to develop new technologies for DM water production to satisfy the make up water need of the AHWR. The Desalination Division of BARC has come up with a Low Temperature Multi Effect Desalination Plant (LT-MED) using low temperature steam, which would produce 500m<sup>3</sup> of DM water every day.

BARC is setting up 300 MWe Advanced Heavy Water Reactor (AHWR). Make up requirement of DeMineralized (DM) water for this reactor is about 360m<sup>3</sup>/day. The conventional DM system raw water is used to produce DM water at an exorbitant cost. Desalination of seawater is one of the options that can be used to meet the requirements of DM water for the reactor. AHWR uses seawater for condenser cooling. A small fraction of this seawater can be taken as feed for desalination plant.

## 19.1 LT-MED DESALINATION PLANT UTILIZING LOW PRESSURE STEAM AVAILABLE FROM THE LP TURBINE OF AHWR

A Low Temperature Multi Effect Distillation (LT -MED) desalination plant, utilizing low pressure steam @ 0.95 bar available from the LP turbine of AHWR, is proposed to be integrated with the Advanced Heavy Water Reactor (AHWR) for

producing 500 m<sup>3</sup>/day DeMineralised (DM) water for in-house requirement as shown below.

One important feature of the LT -MED desalination technology is that it can produce high purity water from highly saline seawater. This process requires minimum pretreatment and is eco-friendly. Another important factor is that it operates with low pressure steam extracted from the LP turbine of power plant which results in negligible loss of power. In addition to utilization of low pressure steam, sharing of common facilities for seawater intake and reject and operation and maintenance will drastically reduce the production cost of DM water. This high quality water can be used by AHWR for DM water make up with minor polishing and reduce load on the ion-exchangers bed (demineraliser) thereby cutting down the use of regeneration chemicals which pollute to the environment. An other use of this high quality water is that it can be mixed with the existing





brackish water in suitable proportion, thereby augmenting the availability of potable water.

The **LT-MED** plant is a 4-effect low temperature desalination plant having Horizontal Tube Thin Film (HTTF) type of evaporators. Low-pressure steam is used in the tube side as heating medium. Feed seawater is sprayed on the outside of horizontal tubes by spray nozzles forming a thin film of seawater. Nucleate boiling takes place on the outside of the tubes. This type of boiling is more efficient than pool boiling due to heat transfer through, thin film of seawater and absence of any hydrostatic head over the boiling liquid. This results in high heat transfer coefficients and heat transfer is possible with low temperature differences. An intermediate heat exchanger is used as an isolation barrier to eliminate any probability of radioactive contamination and generate low pressure steam for use in the desalination plant. Low-pressure steam from the isolation heat exchanger is used in the first effect as heating medium. Vapors generated in the effects are reused in the succeeding effects as heating medium. The effects are maintained at a lower pressure than the preceding effect. The vapors generated in the last effect are condensed in the final condenser. Condensate from all the effects is collected as product water. Figure below shows the schematic flow diagram of the LT-MED Desalination Plant for AHWR.

## 19.2 LTE DESALINATION PLANT UTILIZING WASTE HEAT FROM MHT PURIFICATION CIRCUIT OF AHWR.

The design of AHWR incorporates several features to simplify the design and to eliminate certain systems and components, so as to make it economically competitive with other available options for power generation. Utilization of waste heat by using desalination is one such feature of AHWR. A proposal to utilize waste heat from Main Heat Transport (MHT) purification circuit



to produce high quality desalinated water by Low Temperature Evaporation (LTE) process has been envisaged.

In AHWR Main Heat Transport (MHT) purification system process water @ 33.3 kg/s at 259  $^{\circ}\text{C}$  and at 70 bar is cooled to 42 °C and at 10 bar before passing it through the purification filters and ion exchange columns. Purification flow is cooled up to 100°C in regenerative heat exchangers and 23 MW<sub>+b</sub> heat is recovered by purification return flow to MHT. Further cooling up to 42°C is done by transferring heat to process water in non-regenerative cooler in the present design. This cooling results in loss of 8 MW<sub>th</sub> heat to process water. Considering this, it is felt that a part of this waste heat can be utilized for producing the desalted water. It is possible to utilize this heat for desalination purpose in AHWR for producing about 250 m<sup>3</sup>/day desalted water for the reactor make up and plant utilization. Around 5 MWth heat (from 100°C to 64°C) can be utilized from MHT purification flow for desalination purpose. Figure shows the schematic process flow diagram of the proposed LTE desalination plant for AHWR.

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