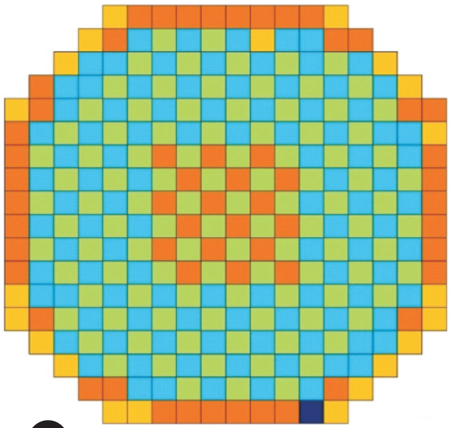
## Core loading pattern optimization for TAPS Units # 1&2 for maximizing the energy output

Fuel management of boiling water reactors TAPS #1 and 2 at Tarapur is very challenging. These units have been successfully operated for the past 50 years with an intense collaboration between NPCIL and BARC. It is worth mentioning that BARC is the sole agency responsible for the optimization of core loading patterns and fuel management of these units.

The operational history is followed and the energy output of every operating cycle is maximised. The refueling frequency is about 18 months. A third of the core is replaced at each refueling cycle. The core loading pattern for each operating cycle is evolved after a detailed optimization study with respect to several operational parameters. The complete fuel management entails design of the new reload pattern for the next operating cycle, core follow-up for the current operating cycle, and monthly updates for control rod sequences. The core loading optimization is more challenging if leakers i.e. failed fuel assemblies are identified and have to be removed. The optimization is done based on the Haling principle. The underlying principle is based on the fact that the core power distribution is maintained constant over the cycle. The Haling power distribution is the flattest power distribution possible over the entire cycle. In BWR, power distribution is bottom peaked due to reduced water density at top.




1 **Core loading pattern for TAPS BWR unit-1,cycle-27**

At Beginning-of-Cycle (BOC), the power peak at the reactor bottom part can be controlled by control rods. The control rods positions are changed over the operating cycle to obtain a flat power distribution and hence maximum energy output for the cycle.

TAPS BWR Unit-2 Cycle-26 was made critical on 24/10/2018 and is capable of delivering energy 280,000 MWD (~6.0 GWD/ST). This cycle was operated upto June 2019 with the optimised parameters and it required control rod sequence change. Control rod patterns at mid cycle for sequence change were worked out and the shut down margin was estimated.

TAPS Unit#1 was operated till January 8, 2020 with the operating cycle-26 and it has subsequently undergone refueling shut down. The reload pattern for cycle-27 of the Unit # 1 was worked out using the End-Of-Cycle (EOC) exposure of cycle-26. Haling calculation, described above was done for cycle-27 to achieve maximum power throughout the cycle. One of the fuel assemblies had failed and was identified at EOC. The average core exposure of Unit-1 cycle-26 at EOC was ~15 GWD/ST ( Burnup in TAPS BWR is estimated as Gigawatt-day /Short ton where 1 shot ton = 90 kg). The average discharge burnup at the unit-1 cycle-26 at EOC was ~20 GWD/ST with highest discharge burnup at ~26 GWD/ST. the core loading pattern worked out for cycle 27 for Unit#1 is shown in Figure 1. The core average exposure at Unit #1 cycle-27 at BOC is ~8 GWD/ST. The estimated value of cycle energy is 6.0 GWD/ST.

The TAPS BWR units have successfully operated with the inputs from BARC and are a fine example to continuous coordinated effort between the operators and reactor physics experts for safe day-to-day operation of the aging boiling water reactors.

This article was contributed by ***K.P. Singh, Rashmi Rai, Arun Kumar Singh*** and ***Umasankari Kannan***

Reactor Physics Design Division

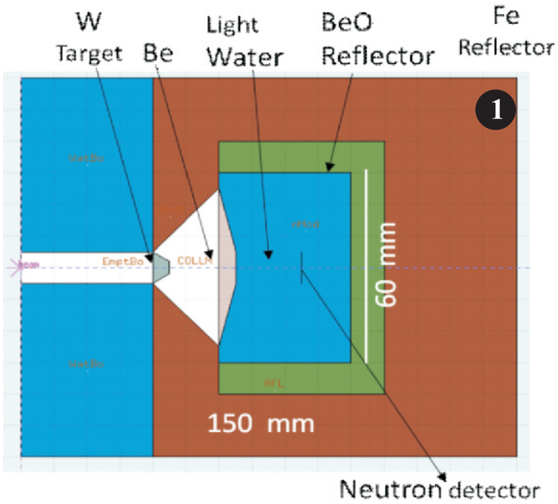
## Production of Molybdenum-99 from linear accelerators

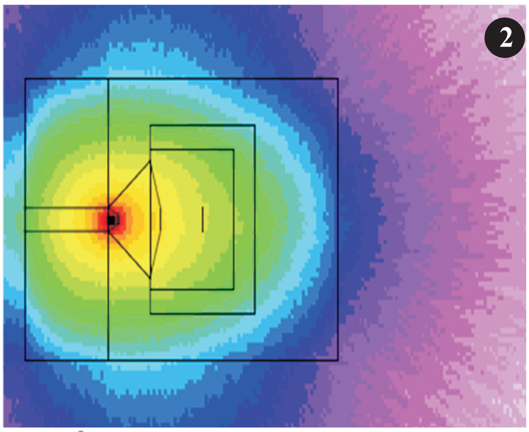
Tecnitium-99m (Tc-99m) is the decay product of Molybdenum-99 (Mo-99) and is widely used in nuclear medicine for diagnostic purposes. Mo-99 is primarily produced in research reactors using neutron induced reactions i.e. capture or fission. Alternate routes for producing Mo-99 are being explored worldwide. The neutrons from electron based linear accelerators (e-LINAC) can be another source to produce radioisotopes. A feasibility study for the production of Mo-99 with the electron LINAC facility at the Electron Beam centre (EBC), Kharghar was taken up. The aim will be to design a facility for producing low specific activity Mo-99 which is a precursor to Tc-99m which is extensively used as a radiotracer for diagnosis.

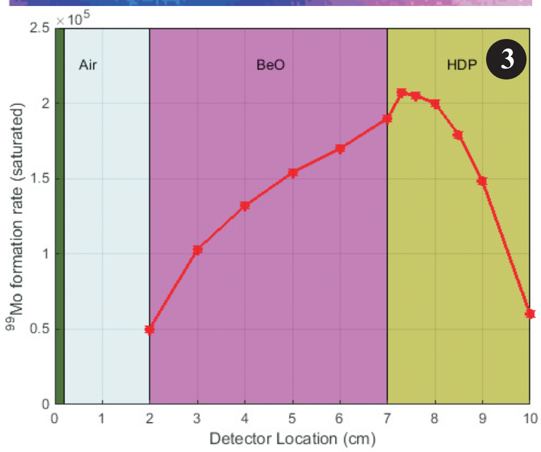
The basic principle of this alternate route is briefly outlined here. The beam of electrons emerging from electron accelerator is made to strike a suitable target like Tungsten or Tantallum. The Bremsstrahlung gammas thus produced are then allowed to fall on a Be target. Photoneutrons are then produced from Be target. These neutrons when captured by Mo-98 produce Mo-99. This reaction is basically $(e\text{-}\gamma\text{-}\eta)$ cascade reaction. Radio-isotope tracers are characterized by their specific activity which is the amount of radiations emitted per second per unit mass of the substance. The maximum activity that can be induced in the sample during irradiation is termed as saturation activity and is an important parameter for such irradiations.

As a first step, it was planned to measure the effective cross section of the capture reaction of Mo-98 in the e-LINAC set-up. Theoretical studies were done to design an experimental set-up for the same. Simulations of photoneutron source from 10 MeV electron beam striking a Tungsten (W) or Tantalum(Ta) target were performed. A set-up was designed and modeled using Monte Carlo particle transport code, FLUKA is shown in Figure 1. The neutron flux can be enhanced by optimizing the moderating medium surrounding the target. The design studies included optimization of the moderator with Be, Water, Carbon and Polyethylene. With the use of Be as target assembly surrounded with water, the saturation activity was found to be maximum. The effective cross section for production of Mo-99 was calculated to be 0.23 barns where the thermal neutron flux was nearly 1.0E+7 n/cm$^2$/s. The neutron flux profile obtained is shown in Figure 2. The specific activity for water moderated configuration was estimated to be 39 µci/g which can be enhanced to 250 µci/g by a suitable conical geometry of the target a shown in Figure1.

In order to perform the measurements for the effective cross section of 98Mo(n,$\gamma$) at EBC, Kharghar, the experimental set-up was re-optimized with Be target and High Density Polyethylene (HDP) moderator. Pre experimental analysis for measuring the 98Mo(n,$\gamma$) 99Mo cross section using 10 MeV LINAC was performed with Monte Carlo particle transport code, FLUKA and in-house code PATMOC.




**1. Experimental set-up designed using e-LINAC for Mo-99 production**

**2. Neutron flux profile in experimental set-up with Be as target and light water as moderator**

**3. Saturated rate of formation of Mo-99 at different locations of the set-up**

The experiment was performed on 7th October 2020 at EBC Kharghar. The electron accelerator was operated for 4 hours at 0.24 mA current. Natural Molybdenum metal foil along with other thermal and resonance flux monitors were placed on the emergent surface of the HDP block in the setup. The activation foils used were thermal flux monitors i.e. Gold, Copper, Manganese, Indium and Cobalt and resonance (epithermal) flux monitors namely, Scandium, Silver, Lutetium, Tantalum and Tungsten. Estimate of the effective cross section from the analysis of experimental data from different activation foils and methods range from 0.28 barns to 0.4 barns, which is reasonably higher than the 0.13 barns in a reactor environment. The spatial variation of saturation reaction rate for formation of 99Mo estimated from FLUKA for the experimental set-up is shown in Figure 3.

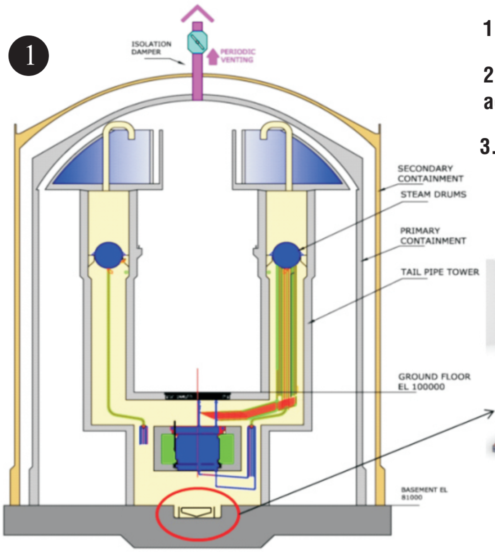
This article was contributed by ***Dr. Umasankari Kannan***

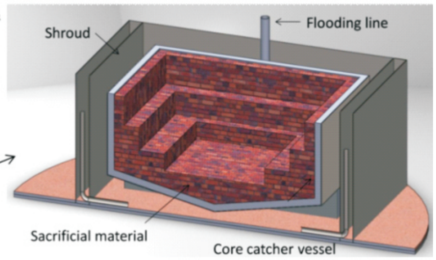
Reactor Physics Design Division

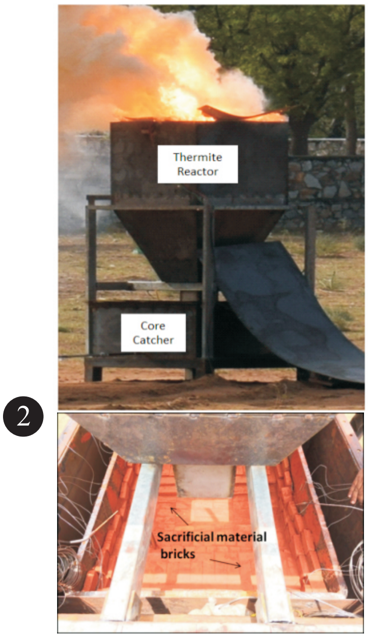
# Technology Development of Core Catcher for Indian Advanced Nuclear Reactors

Post Fukushima, to manage low probability core melt accidents, dedicated core catchers are being developed worldwide. Since these technologies are proprietary, for Indian AHWR and light water cooled reactors, technology for a unique core catcher has been developed to contain and cool the core melt for extended period and reduce the radioactivity release to public domain substantially. This core catcher can be deployed in other advanced light water reactors also. Cooling of more than 100 tons of this mixture of nuclear fuels, structural material, control rod materials, having temperature more than 2800 °C and generating decay heat continuously, to very low temperatures for prolonged period is a technologically challenging and scientifically complex task.

To achieve this, a unique core catcher has been developed by BARC which consists of inverted pyramid shaped thick steel vessel as shown in Figure 1. Special sacrificial material was developed in house, which absorbs the heat of this corium by melting and mixing in it. When corium mixes with this sacrificial material, it becomes lighter and moves to the top and the metallic components sink at bottom which is termed as density inversion. This has two advantages: (i) there is no metal at top thereby eliminating chance of hydrogen generation by metal water reaction and (ii) the light weight ceramic melt forms a stable crust enveloping the heat generating high temperature melt like a "capsule" so that when water is added to



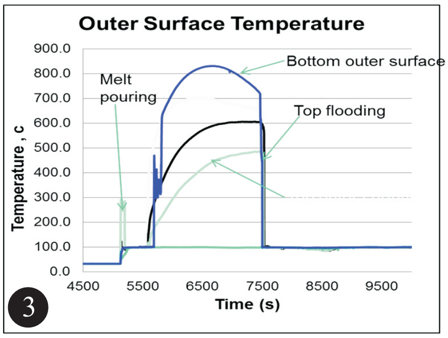




**1.AHWR Core catcher**

**2.Design validation at prototypic condition using actual sacrificial material**

**3.Vessel surface temperatures**




the top of melt to cool it, the stable crust prevents water seeping into the bottom of core catcher and avoids metal water interaction. The core catcher contains and cools corium for prolonged period. The core catcher has design life same as that of reactors.

The core catcher geometry and flooding strategy was optimized by conducting several experiments in scaled facilities [1-3]. In addition, integral experiments were conducted at prototypic condition and using actual sacrificial material in which about 550 kg simulant melt at more than 2500 °C [4] was poured in the core catcher and was cooled with water as per the actual flooding strategy (Figure 2). It was observed that, the inner vessel temperature remains always below 900 °C and the outer vessel temperature never exceeded water saturation temperature when water is present outside (Figure 3). A stable ceramic crust was obtained at the top. The experiment was repeated with 300 kg melt which showed good repeatability by obtaining similar results. The decay heat removal capability was also demonstrated in integral experiment which demonstrated stable crust formation and decay heat removal for prolonged period.

Together, all these tests demonstrated the efficacy of core catcher for cooling and stabilization of molten corium for prolonged period in case of severe accidents.

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[3] V., Ganesh., Kulkarni, P.P., and Nayak, A.K. (2020). "Study on Melt Coolability and Water Ingression Through Melt Pool in Scaled Down Core Catcher-The Influence of Vessel Angle." ASME. ASME J of Nuclear Rad Sci. January 2021; **7(1)**: 011301. https://doi.org/10.1115/1.4048235

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This article was contributed by
***P. P. Kulkarni, A. K. Nayak, S. K. Sinha***
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# Integrated Test Station for C&I Systems of AHWR

Control and instrumentation (C&I) systems play crucial role in safe, secure and reliable operation of Nuclear Power Plants (NPPs). Owing to their pivotal role, high demands are always placed on functional, performance and qualification requirements of such systems. This calls for critical testing and validation of each and every component of the C&I systems prior to their deployment in NPPs. Traditionally, these systems and components are tested in stand-alone mode, with limited simulated interface with other relevant systems. To stretch the testing beyond this, it was envisaged to design a facility that supports integrated testing of all the important C&I systems along with a plant simulator.

In this context, a facility named AHWR Integrated Test station (ITS) has been designed, developed & commissioned at Reactor Control Division, BARC. It is a configurable test bed for integrated testing

C&I systems of ITS

and validation of functional, performance and qualification needs of proposed C&I systems and architectures of NPPs. This first of its kind facility is configured with all the major control and protection systems and other components of the C&I architecture for Advanced Heavy Water Reactor (AHWR)[1]. It comprises of prototypes of C&I systems built using the in-house developed Trombay Programmable Logic Controller (TPLC), network switches, data servers, display workstations, test panels and a real time engineering simulator. These interconnected components form the test bed.

Prototypes of safety critical systems including Reactor Protection System with Test and Monitoring System, Containment Isolation System, and safety related systems including Reactor Regulating System, Reactor Process Control System, Core Monitoring System, Alarm Annunciation System etc are part of the ITS. Operator workstations for this facility, developed using an indigenous Linux based SCADA, are provided in a centralized control room. The safety related system prototypes are interfaced with the real time engineering simulator facilitating close loop integrated testing.

By virtue of the mathematical models of all the major systems and components including reactor kinetics, thermal hydraulics and process systems, the engineering simulator [2] allows creating realistic scenarios of normal and off-normal operating conditions of the plant, thus making it possible to test the C&I systems in a realistic, virtual reactor environment.

C&I Security is an important aspect which needs to be considered right from the conceptual design stage of C&I architecture. An indigenous trusted computing module named ANU-NISHTA has been deployed in servers and workstations to overcome the security issues posed by commercial Operating Systems used therein. Likewise, an in-house developed secure Network Time Protocol time server module has been used in this facility for time synchronization of C&I systems.

Engineering simulator of AHWR

Various components of the C&I architecture of AHWR are tested and validated in this facility. The ITS is being extensively used in verification and validation (V&V) of plant control algorithms and safety interlocks, optimization of control gains of reactor and process control systems, optimization of detector locations in the reactor, design of control room interfaces etc. ITS is also used as a platform to carry out dynamic safety analysis of AHWR.

**Further Reading**:

[1] "Preliminary Safety Analysis Report of AHWR", Report no. AHWR/PSAR/USI-**01591**

[2] "Engineering Simulator for Advanced Heavy Water Reactor", BARC Newsletter, pp. 1-5, **May-June** 2018

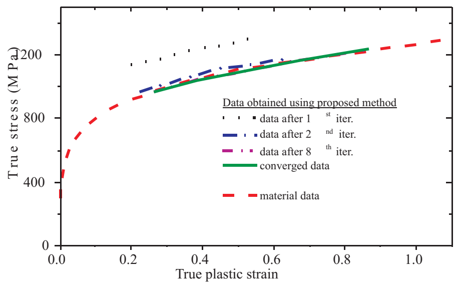
This article was contributed by ***S.R. Shimjith, D.A. Roy and Uday Vaidya***

Reactor Control Division

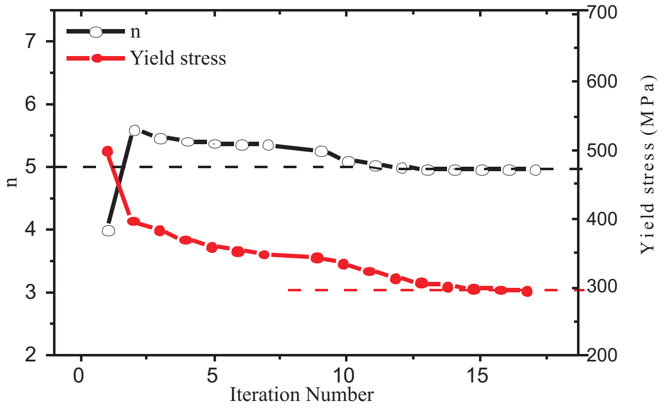
# Evaluation of material stress-strain curve using load-indentation data from ball-indentation tests

For evaluation of mechanical properties such as yield stress, ultimate tensile strength and strain hardening exponent of service exposed materials, ball-indentation technique is handy as it can be used in the field using a suitable portable device. However, the expressions used in literature to evaluate true stress and strain contain empirical constants which are not valid for multiaxial nature of the indentation process. Moreover, these constants are dependent upon the type of material, microstructure etc. which are difficult to estimate a-priori. In this work, these limitations are removed through use of a novel method where multiaxial nature of stress and strain are incorporated in the formulation itself.

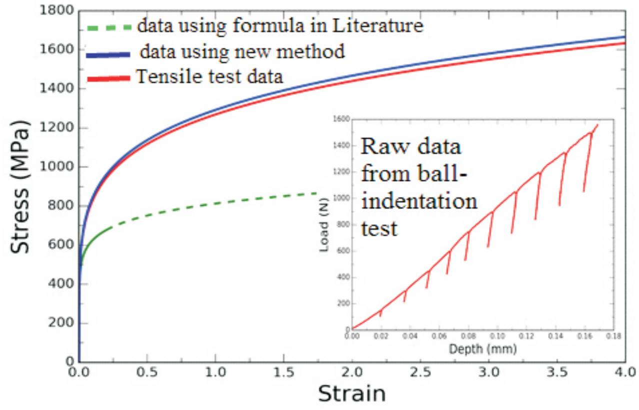
A new set of stress and strain multiaxial parameters have been evaluated from 3D finite element analysis and these have been expressed as functions of load, yield stress and strain hardening exponent (n). To evaluate the true stress-strain curve, the raw data in terms of load-indentation curve from experiment is taken. Using an initial guess value for yield stress and 'n', the multiaxial stress and strain parameters are evaluated. These data are used further to evaluate stress-strain curve. This algorithm is invoked in an iterative manner by updating the multiaxial parameter and the stress-strain curve is updated after each iteration as shown in Fig. 1. The stress-strain curve, yield stress as well as the hardening exponent 'n' converge quickly as shown in Fig. 1 and 2 respectively.




1. Stress-strain curve obtained from raw data




2. Convergence characteristics of new method




3. Comparison of stress-strain data for SA516Gr.70 steel among tensile, new, old methods

This method is valid for a wide range of strain data unlike the method in literature where strain is limited to 20%. This method doesn't use any empirical constant and the expressions for true stress and true strain are valid for multiaxial state of deformation as observed in ball-indentation tests. Moreover, no material-specific constants are required in the algorithm.

The algorithm has been applied to evaluate stress-strain curve of SA516Gr.70 steel. The load vs. depth of indentation data as obtained from experiment is shown in Fig. 3. The stress-strain curve as obtained using the method in literature and the current technique is shown in Fig. 3. The results of new method are vastly superior when compared to method currently used in literature and it is also close to tensile test data. Hence, it can be concluded that this new technique is more accurate and versatile as the multiaxial effect of state of stress in the indentation region of ball is taken care of inherently in the formulation.

This article was contributed by ***M.K. Samal, A. Syed & J. Chattopadhyay,*** Reactor Safety Division

# Nuclear system for PHWR Start-Up after En-Mass Coolant Channel Replacement

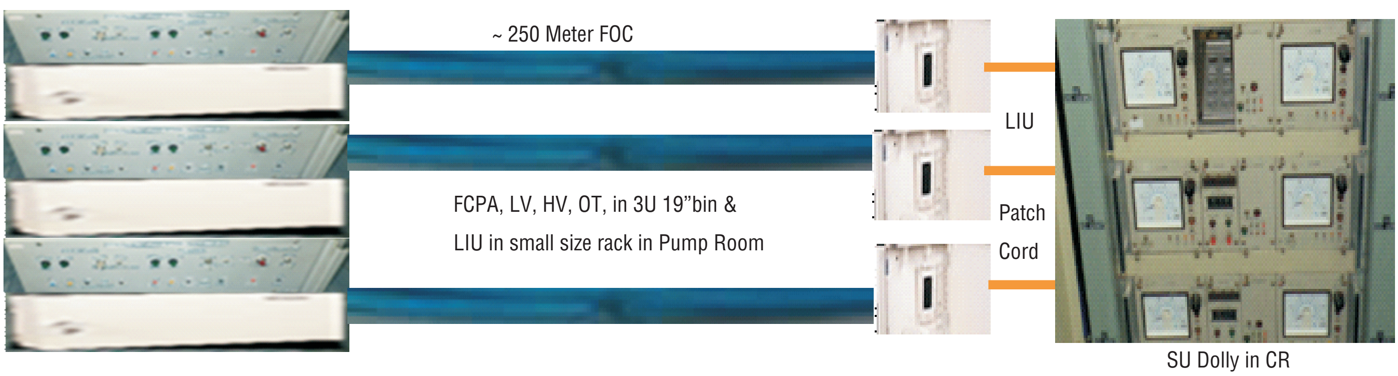
Indian PHWRs conventionally use $^{10}$B lined Proportional Counters (BLPC) based in-core start-up channels during Initial Fuel Loading (IFL) and First Approach to Criticality (FAC).

In fresh core, in-core low gamma dose rate allows to measure neutron flux during IFL & FAC using BLPC with Pre-Amp located close to detector & pulse processing channel at around 50-meter distance, interfaced to Start-Up (SU) dolly in Control Room (CR) at around 200-meter distance. SU dolly houses triplicate Scalar / Timer, analog electronics, trip module, display & meters.

During start-up after EMCCR (En-Mass Coolant Channel Replacement), in-core gamma dose rate is high which precludes use of BLPC due to its poor gamma tolerance & count rate limitation of the detector & instrumentation system. Hence it requires specialized nuclear system based on Fission Counter (FC: IEC 60515 & ORNL reports) having higher gamma tolerance & higher count rate capability.

Nuclear System comprising Fast Current Pulse Amplifier (FCPA) with Optical Transmitter (OT) in 19", 3U bin & Optical Receiver (OR) was developed as shown in Figure 1. Triplicate bins & Light-guide Interface Unit (LIU) are located in Reactor Building (RB). FCPA amplifies FC signal, discriminates against noise and gamma pulses & drives OT which launches optical pulses on three sets of Armoured Fibre Optic Cable (FOC) that interface the pulse outputs from bins to SU dolly in CR through LIU and Optical Receiver (OR).

The system noise performance is enhanced by adequate power line & EMI-EMC filters, low noise LV, HV supply. The optical pulse transmission scheme makes the system immune to field noise,




1. Triplicate Startup channel of KAPS-1 used after EMCCR

maintains pulse characteristics & results in accurate high count rate capability up to 100 Kcps.

The system, first of its own kind in Indian NPPs, using optical pulse transmission scheme was used for KAPS-1 start up after EMCCR. The system with about 5% accuracy, linearity and acceptable statistics obtained though settable counting time performed satisfactorily for low neutron flux in high gamma background under challenging reactor noisy environment.

The system recorded counts during all phases of reactor start-up & up to criticality, matching the theoretically estimated counts within 10%. The system was also employed for execution of subsequent physics experiments.

The system can be used for FAC & start up after long shut down for any Nuclear Power Plant.

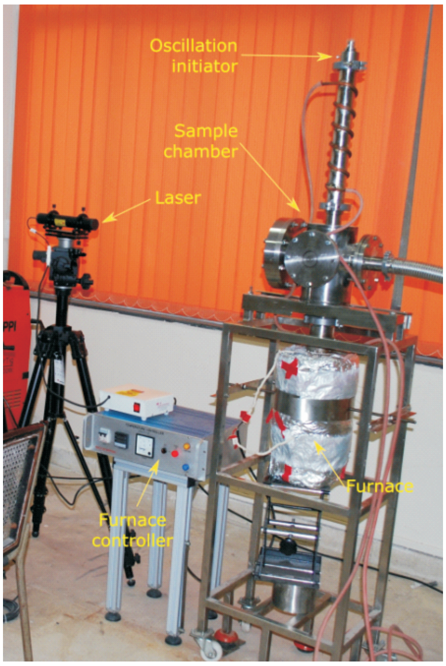
This article was contributed by ***Dr. P.V. Bhatnagar***, Electronics Division

## Indigenous instrument development for molten fluoride salts

Molten fluoride based salts are to be utilised in both the primary and the secondary circuits of Indian Molten Salt Breeder Reactor (IMSBR) and Innovative High Temperature Reactor (IHTR). Instruments for measurement of molten salt density, impurities and its purification are not available commercially. High temperature viscometers need to be imported and the molten salt sample is susceptible to contamination from the cover gas. Further, special precautions need to be taken while handling salts containing uranium and thorium. In view of this, these instruments have been developed in-house.

### Densitometer and viscometer for molten salts

The densitometer is based on Archimedes principle with compensation for thermal expansion. The viscometer is based on the oscillating cup technique in which the sample is sealed inside a capsule and the viscosity is determined by the change in damping characteristics of the filled capsule vis-à-vis the empty capsule. As the sample is sealed inside the capsule, it is not exposed to the cover gas and can be used for low level active salts.




**Oscillating cup viscometer**

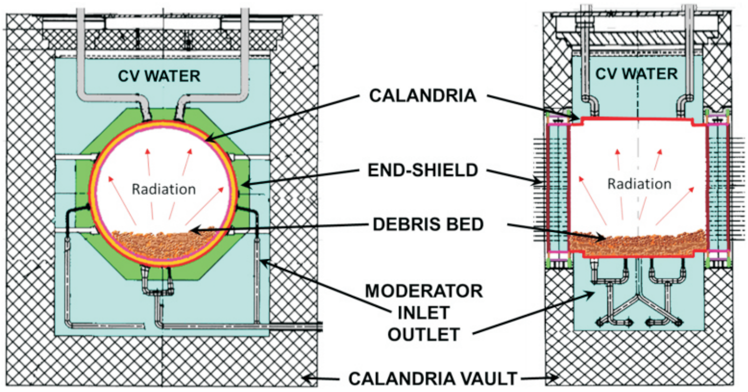
### Impurity measurement and purification systems for molten salts

The electrochemical measurement technique has been developed for online measurement of impurities at high temperatures which can be utilised in a high radiation environment. In order to remove impurities from the salts, an electrolysis based purification method has been developed. These facilities were utilised to find out impurities and to purify FLiNaK, the secondary side coolant salt for both IMSBR and IHTR.

This article was contributed by ***A. Basak***, High Temperature Reactor Section

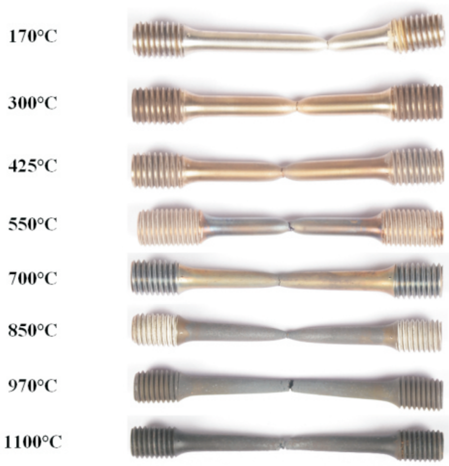
## Time Estimation for In-Calandria Retention of Corium under Total Loss of Heat Sink

Severe core damage accidents in Indian PHWRs are rarest of the rare scenario that involve cascading failure of multiple systems provided under defence-in-depth design philosophy. Severe accident analysis requires postulation of accident scenario that involves multi-physics aspects such as reactor physics, thermal-hydraulics, transport behaviour, structural degradation etc. Calandria vessel of PHWR acts as a physical barrier in limiting the progression of such accident. It is submerged in a large pool of water in Calandria vault and is expected to contain and cool the core debris/ corium for reasonable duration, delaying/ preventing the adverse event of molten corium-concrete interaction. Thus, the in-Calandria retention of core debris/ corium for a larger time frame is of great importance to operator and regulator for planning of accident management action.

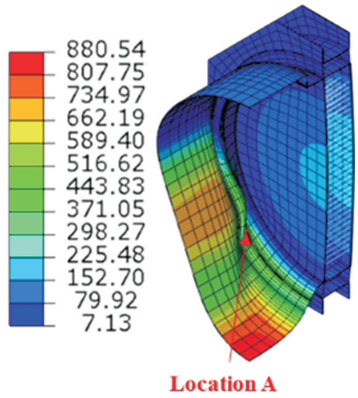



**1. Calandria located inside vault under core-collapse conditions**

In view of above, structural integrity of Calandria has been assessed for in-vessel retention of core debris/ molten corium for a postulated accident scenario of unmitigated total loss of heat sink (no active means are available to remove the decay/ stored heat) for standard 220 MWe and 540 MWe Indian PHWRs. In absence of any management action, the fuel channels would degrade gradually and collapse in the form of debris on to the Calandria bottom (Fig. 1). Here onwards the Calandria is responsible for cooling and retention of core debris/ corium which has been analysed in detail using sequential thermal-hydraulic and structural analyses. Thermal-hydraulic analysis considered all modes of heat transfer–conduction, convection and radiation; two phase heat transfer; decay heat; heat from metal–water reaction; melting and ablation; and was conducted using the ASTEC code with suitable PHWR specific adaptations. Structural response was evaluated using finite element analysis where geometric and material non-linearities; temperature dependent material properties and deformation models were considered. Here the domain consisting of complete Calandria-end shield assembly was modelled to account for the boundary conditions and flexibility provided by structural elements such as annular plates and diaphragms. High temperature material data of Calandria material up to




2. Failed tensile specimens of 304L at different temperatures




3. Displacement (mm) in 540 MWe PHWR Calandria at 74.1 hrs from core collapse

1100°C were obtained from systematic tensile and creep tests programme (Fig. 2). Structural failure time of Calandria was assessed considering various likely failure modes that are plastic instability, excessive inelastic strains and creep-stress rupture criteria.

The study revealed that in 540 MWe PHWR, Calandria remained fully submerged till ~ 36 h after core collapse. The debris bed didn't undergo melting till 74 h and Calandria failure is observed roughly 3 days after the core collapse (Fig. 3). In standard 220 MWe PHWR, Calandria failure is observed at about 83 h after the core collapse. It demonstrated that the Calandria, aided by inherent design features of PHWR viz., several heat sinks and low power density, delays the accident progression extensively and provide a large time frame available to the operators for appropriate SAMG actions. The details of study are available in BARC external reports (BARC/2017/E/005, BARC/2019/E/008, BARC/2019/E/011, BARC/2020/E/015 and journal papers (NED, Vol.367, Article ID – 110791, NED, Vol.368, Article ID-110801).

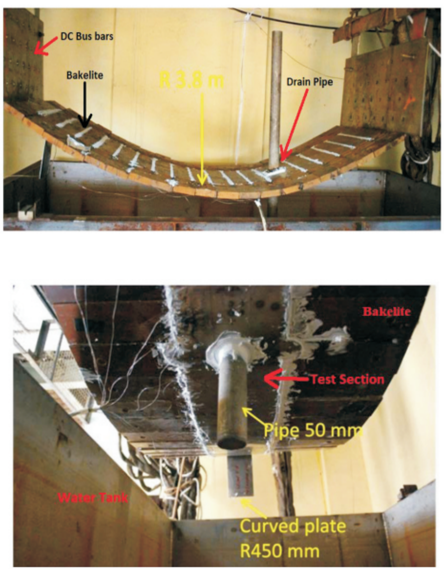
This article was contributed by ***Keshav Mohta, Onkar S. Gokhale, Suneel K. Gupta, Deb Mukhopadhyay, J. Chattopadhyay***

Reactor Safety Division

## Demonstration of Critical Heat Flux Limits in calandria vessel of PHWRs under severe accident condition

Pressurized Heavy Water Reactors (PHWRs) are workhorses of Indian Nuclear Power Program with or 19 PHWRs currently in operation having installed capacity of 5160 MWe. In –general, the safety record of these reactors has been excellent over the years. However, the recent Fukushima accident has compelled the nuclear engineers to have a re-look into the safety systems of these reactors. In PHWRs, a very low probability accident can be postulated involving extreme events accompanied with failure of multiple safety systems which can lead to the collapse of coolant channels which may ultimately relocate to the lower portion of the calandria vessel (CV) forming a particulate bed. Due to the continuous generation of decay heat in the debris, it may melt and form a molten pool (also called as corium) at the bottom of the CV. If not cooled, this will ultimately fail the vessel, release into containment leading to basemat melt through and generation of large amount of gases including hydrogen which pose a threat to containment failure.

Fortunately, the Calandria Vessel is surrounded by tons of vault water, which can act as a core catcher in this scenario. This is termed as in-vessel retention of molten corium. Under these conditions, it is necessary to ensure that the heat flux imposed on the inner wall of calandria due to the melt is less than the maximum heat that can be removed by outer wall (also called critical heat flux (CHF)) at the bottom of the calandria vessel wall. The calandria vessel being 8 meter in diameter with 6 meter in length, it is very difficult to predict

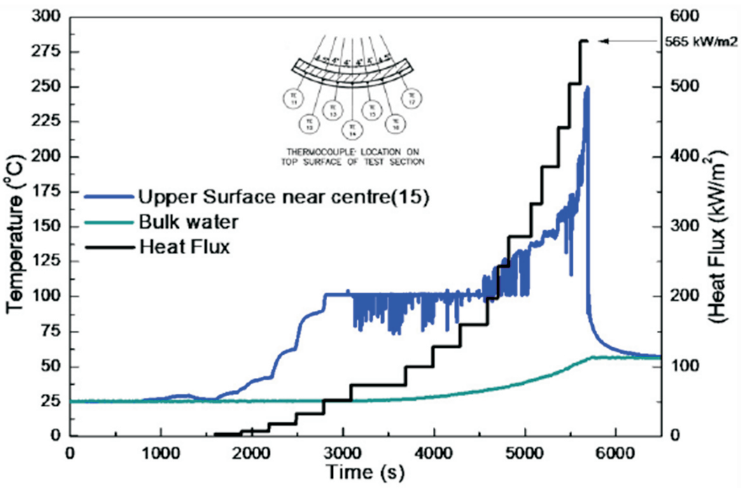



1 Experimental Facility with prototypic curvature

the CHF occurring on the external surface with downward boiling with any heat transfer correlations. Hence, conducting experiments is a necessity. To determine the CHF, experiments were performed on a 25° curved section with actual curvature [1] as that of calandria vessel (Figure 1). The test section was connected to DC electrical supply. The upper side of the test section was insulated so that all the heat flux was directed downward as in actual case. The test section was submerged in vault water. During the experiment, electrical power to the test section was increased in small steps. Temperatures at various locations on test section were monitored. At a particular heat flux, there was a sudden jump in the test section temperature at bottommost point which marks the CHF (Figure 2). The CHF is highly dependent on the vault (bulk) water temperature. Experiments were conducted with different vault water temperatures. It was observed that, at around 75°C vault water, CHF occurs at 565 kW/m². As vault water temperature increases the CHF decreases to 425 kW/m² at saturated conditions (Figure 3). Our earlier experiments on in-vessel retention have determined that, the maximum heat flux imparted on the inner wall of the calandria vessel is around 200 kW/m² [2].This ensures that sufficient thermal margin is available during severe accidents.

There was a concern that, in actual calandria vessel, moderator drain pipes are located at the bottom. This may pose obstruction to sliding of bubbles and can have effect on CHF. The effect of moderator drainpipe on CHF was also investigated by placing drain pipes at the bottom and conducting experiments in similar ways. As the drainpipes are away from the center (6 ° and 13° location), no effect on CHF was observed. CHF was obtained at bottom of the calandria vessel as in earlier case. These experiments established that, sufficient thermal margin is available for in-vessel retention of corium in calandria vessel during severe accident conditions.

2

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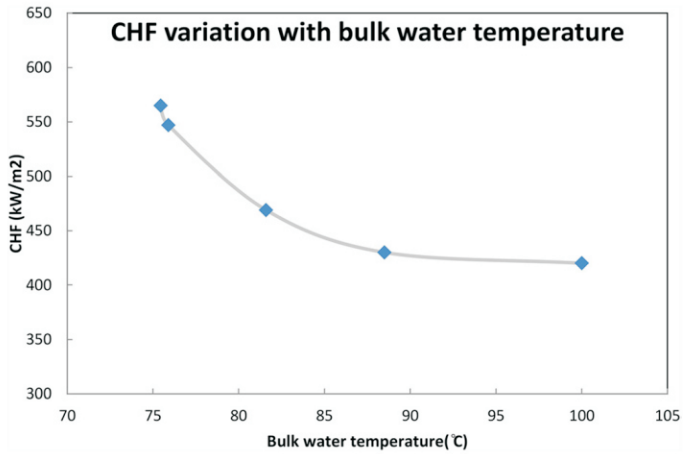


**2. Sudden Temperature rise indicating CHF**

**3. CHF variation with bulk water temperature**

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[1] Verma, P.K., Kulkarni, P.P., Pandey, P., Prasad, S.V., and Nayak, A.K. (November 16, 2020). "Critical Heat Flux on Curved Calandria Vessel of Indian PHWRs During Severe Accident Condition." ASME. J. Heat Transfer. February 2021; **143(2)**: 022101. https://doi.org/10.1115/1.4048823

[2] Prasad, S. V., Kulkarni, P. P., Yadav, D. C., Verma, P. K., and Nayak, A. K. (November 29, 2019). "In-Vessel Retention of PHWRs: Experiments at Prototypic Temperatures." ASME. ASME J of Nuclear Rad Sci. January 2020; **6(1)**: 011601. https://doi.org/10.1115/1.4043999

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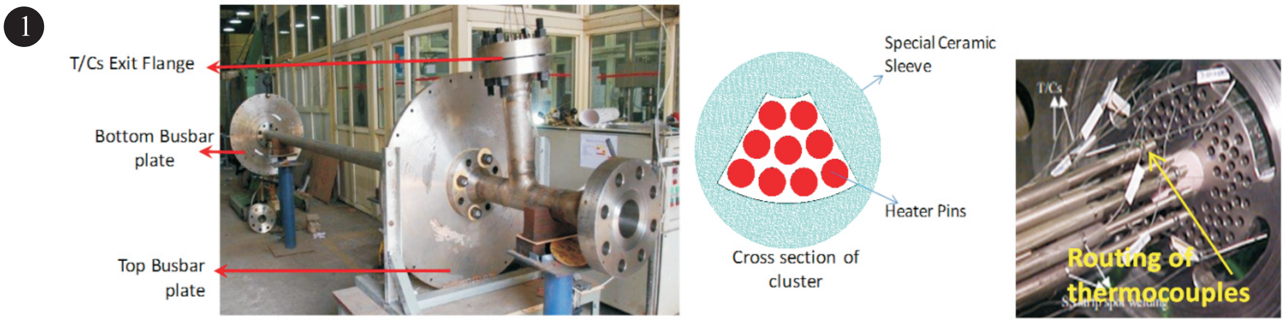
## Full Scale Demonstration of Thermal Margin of AHWR and Development of AHWR-Critical Power Correlation

Theoretically, there is no limit to the power which can be generated in a nuclear reactor core. Practical limit however exists due to the ability to carry away the heat generated and the related phenomenon is termed as the Critical Heat Flux (CHF). The margin available against the CHF is known as thermal margin. Thermal margin is thus an important parameter which limits the reactor power. In AHWR, CHF is one of the important design parameters. Adequate thermal margin must be maintained under normal operation and anticipated transients and, this needs to be demonstrated experimentally in view of the uncertainties in the theoretical predictions of CHF. The fuel cluster power corresponding to CHF




condition is termed as the critical power. The ratio of Critical power to operating power serves as the figure of merit for thermal margin. This is known as Critical Power Ratio (CPR).

CHF phenomena in rod bundles are highly dependent on geometry. Thus, the existing models/correlations of CHF are not applicable to AHWR. Moreover, the CHF correlations for the rod bundle are proprietary in nature. The Jansen-Levy model of CHF was adopted at the design stage of AHWR which was found to be highly conservative when compared with the other approaches like the CHF look-up table and experimental data. Thus, a full scale demonstration of CHF in the rod bundle is necessary for the design validation of AHWR as an important regulatory input. Simulation of nuclear heating using electrical simulator under high pressure and high temperature, mimicking reactor conditions, and detection of CHF occurrence are the challenging tasks for CHF experiments in a rod bundle.

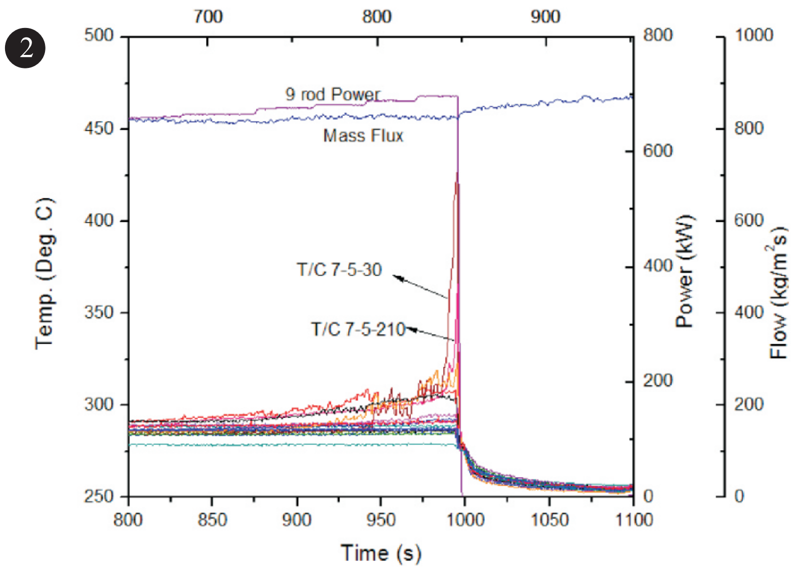
In view of the above, CHF experiments were conducted with simulated fuel rod cluster in 3 MW Boiling Water Loop (BWL) at one to one pressure, temperature and flow conditions of AHWR. Fig. 1 shows assembled fuel rod simulator of AHWR and its sectional view. Fig. 2 depicts the temperature transients during CHF experiments indicating temperature excursion at CHF. The experimental data has led to development of a dedicated AHWR Critical Power Correlation (AHWR-CPC). The form of this correlation is shown below.

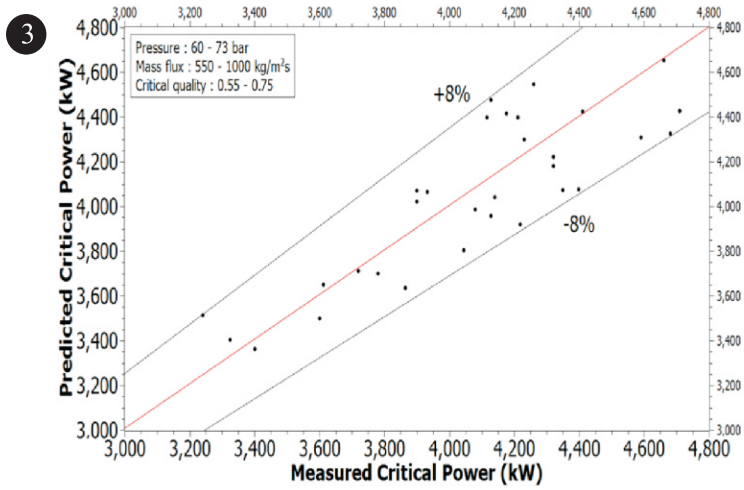
$$q''_{CHF} = Ap^{*2} + Bp^{*} + C$$

Where, $p^{*} = p/70\,;\ A = f_1(G,x)\,;$

$B = f_2(G,x)\,;\ C = f_3(G,x)$

$p$ is the pressure in bar, $G$ is the mass flux in kg/m$^2$s, $q''$ is the heat flux in kW/m$^2$.




1. Assembled Fuel Rod Cluster Simulator of AHWR with 1/6$^{th}$ symmetry sector and mounting of the thermocouples on the fuel pins
2. Typical critical power experiment of AHWR showing temperature rise at CHF
3. Thermal margin evaluation using AHWR-CPC

The AHWR-CPC correlation was found to predict the CHF within ±8%. The exact nature of the correlation is not revealed due to its proprietary nature. A statistical analysis of correlation data indicates minimum required thermal margin of 1.15 for AHWR-CPC. The experiments established that adequate thermal margin exists in the AHWR design. This is the first time that such rod bundle CHF experiments have been conducted in India under reactor conditions.

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