Sodium Testing of Fast Reactor Components

P. Selvaraj, P. Puthiyavinayagam, A. Venkatesan, A.K. Bhaduri

Indira Gandhi Centre for Atomic Research, Department of Atomic Energy, Kalpakkam-603102, Tamil Nadu, India

E-mail contact of main author: pselva@igcar.gov.in

Abstract. Fast reactor components are required to work in hostile atmosphere of sodium, high temperature and radiation. Sodium testing of components before installation in reactor is essential for design validation and safe operation of the reactor. Components like control and safety rod drive mechanism, diverse safety rod drive mechanism, fuel handling machines and under sodium ultrasonic scanner, etc. were sodium tested to make it worth to use in reactor systems. Integrated testing of safety grade decay heat removal system was performed in a reduced power scale to ensure its functionality on demand and its response during transients. The sodium to air heat exchanger in this system was tested separately for its performance. A scaled down model of steam generator scaled in tube numbers was extensively tested for its heat transfer and other thermal hydraulic characteristics in steam generator test facility. An under sodium ultrasonic scanner which has a primary function to provide clearance for plug rotation for fuel handling operation has been developed for scanning the above core plenum to ensure safe start-up of fuel handling operation was also tested in sodium for checking its functionality. Functionality of leak collection tray and fusible plug provided in the reactor system for collecting the sodium leaks from piping and components was tested with 100 kg sodium leak ensure the collection and safe handling of leaked sodium in case of pressure boundary failure. The above sodium tests have provided valuable feedback in developing more robust components for future FBR. This paper brings out the details of testing performed for reactor components and systems for prototype fast breeder reactor.

Key Words: Prototype fast breeder reactor, Sodium testing, Reactor components, Experiments

1. Introduction

Prototype fast breeder reactor (PFBR) [1] is under advanced stage of commissioning at Kalpakkam, India. The reactor is designed with the experience gained by the successful operation of fast breeder test reactor (FBTR) [2] which is operating for last 32 years and the international reported experience of fast breeder power reactors. In addition to this modification and optimisation has been done in many reactor components and systems based on the indigenous research and development programs. Many of the reactor components were first of this kind, with major modification from the existing system or the data required for design validation and qualification was not available to the designers. Many reactor components were tested in air and in water for their performance due to the adequacy of testing in water / air, convenience and economic considerations. But for many reactor components and systems the functionality in water / air will be quite different with the functionality in high temperature sodium. Qualification of reactor components with top safety classification in actual working environment is one of the mandatory requirements for safe and reliable operation of the reactor. Hence many critical reactor components were required to be tested in sodium and qualified for the usage in reactor and safety related systems and components to characterise their performance during anticipated reactor conditions. According to this a detailed sodium testing program was made to cater the requirement of PFBR.

For testing of reactor components in sodium many experimental facilities were utilised at IGCAR. Around sixteen sodium facilities each with sodium hold up varying from 1 ton to 100 ton and sodium flow capacity up to 120 m^3/h are functional at IGCAR. Each of these sodium facilities are with specific functional features to carry out the testing of components. Presently these facilities are used for the development and testing of improved and optimised reactor components and systems for future fast reactors in the country. This paper describes the completed testing program for PFBR components and systems.

2. Testing of sodium components and systems

2.1. Design validation of Shutdown Systems

PFBR is equipped with two independent, fast acting and diverse shutdown systems. The absorber rod of the first system is called Control & Safety Rod (CSR) and that of the second system is called Diverse Safety Rod (DSR). The respective drive mechanisms are called Control & Safety Rod Drive mechanism (CSRDM) and Diverse Safety Rod Drive mechanism (DSRDM). CSRDM has the dual function of reactor control and safe shutdown, however, DSRDM has a single function of reactor safe shutdown. They are safety class-1 components and their qualification is one of the mandatory requirements for safe and reliable operation of the reactor.

After detailed design and analysis, the CSR & CSRDM were qualified in two stages by extensive testing [3]. In the first stage, the critical assemblies / components of the mechanism were tested and qualified individually and the design parameters were fine tuned. The critical assemblies / components identified for qualification in CSRDM & CSR are – scram release electromagnet (EM), hydraulic dashpot & dynamic seals. Prototypes of these components were manufactured and tested in dedicated test facilities. Testing included the performance testing and endurance testing in simulated operating conditions. Design of these components were frozen based on these testings. A series of experiments were carried out to find out the load carrying capacity, temperature rise and response time of different electromagnet configurations. The development work of dashpot proceeded in three campaigns, incorporating the necessary modifications to improve the performance of dashpot after each successive campaign. The major parameters that characterise the performance of dashpot are the peak deceleration of mobile assembly and piston, and the peak dynamic fluid pressure developed. These parameters were measured during experiments. Testing of different types of seals were carried out and based on the test data, labyrinth type V-ring seals were selected. Sodium vapour deposition experiments were also carried out to estimate the severity of sodium deposits in annular spaces. The magnitude of sodium deposition rate is observed as very low.

In the second stage, Prototype CSR & Prototype CSRDM (Fig. 1) were manufactured and subjected to all integrated functional tests in air, in hot argon and subsequently in sodium at temperatures starting from 200°C up to 550°C. Sodium testing was carried out in a test vessel with provision for keeping the mechanism and absorber rod in aligned condition or in misaligned condition. Functional tests were carried out in aligned as well as in misaligned condition. The performance of the mechanism is found satisfactory. After successful completion of functional testing, endurance testing was carried out with pre calculated operation cycles. Root cause analyses were carried out on the problems encountered during testing and necessary modifications were incorporated. Based on the preliminary tests carried out, the design of CSR was modified to CSR - Mark-II. The results of all the tests show that the performance of CSRDM along with CSR is satisfactory throughout the endurance testing and there is no significant change in performance. The CSRDM along with CSR has been tested for 500 cycles keeping temperature of sodium at 530°C in aligned condition and 1093 cycles at 550°C with CSR - Mark-II with 30 mm misalignment. The CSRDM has been qualified for 14 years of life in the reactor and CSR is qualified for its life cycles.

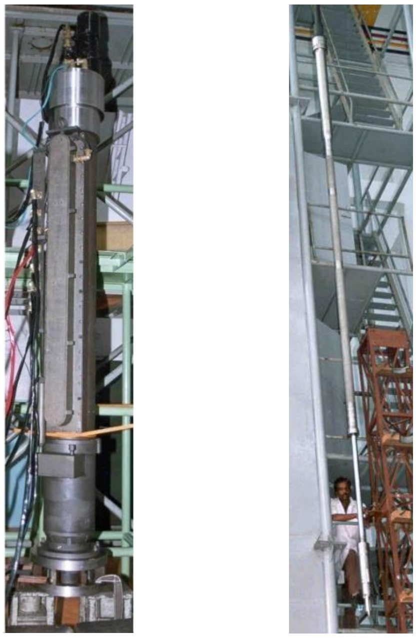


Fig. 1.Upper and lower parts of Control & Safety Rod Drive Mechanism

Similar to CSRDM, the design validation of DSRDM & DSR was also carried out in two stages [4]. In the first stage, development of scram release electromagnet was carried out in the stages from material selection to testing in sodium at high temperature.Soft iron with superior magnetic properties at elevated temperatures was used for manufacturing inner & outer cores. The testing carried out in sodium up to 550˚C demonstrated that EM meets all the design requirements. Endurance test was also carried out in sodium for a duration of 30 days. Variation of lifting capacity & response time over a period of time was found to be negligible. Studies were also carried out for evaluating the in-sodium self-welding susceptibility of EM and its armature of DSRDM.

In the second stage, Prototype DSR and Prototype DSRDM were manufactured and tested for their performance in air, in hot argon at 200 °C and then in sodium at temperatures starting from 200°C up to 400°C. Sodium testing was carried out in the same test vessel in which CSR was tested. Subsequently, the mechanism was subjected to endurance testing with respect to SCRAM operation and translation operations. During endurance testing, proper functioning of DSRDM & DSR was monitored by measuring the free fall & braking times and frictional force was measured. With these tests the DSRDM is qualified for 10 years of reactor operation, while DSR is qualified for full life of 2 years in reactor operation.

2.2. Testing of safety grade decay heat removal system

The transport of decay heat from the fuel pins to the ultimate heat sink, atmosphere for PFBR was experimentally studied in different facilities, means and mediums. The thermal hydraulics of the core to hot pool is studied with a slab model of core in water medium and the thermal hydraulic interactions of core- hot pool - DHX is studied again in water medium with 1/4th scaled down model of the reactor. The characteristic of decay heat transport from hot pool to the atmospheric air is studied in the sodium facility called SAfety Decay Heat removAl loop in NAtrium (SADHANA). The performance of the sodium to air heat exchanger is separately studied in sodium and air under forced circulation. The heat transport medium in the SADHANA experimental facility is sodium as in the case of PFBR. The schematic of passive decay heat removal system for experimental studies is given in Fig. 2.

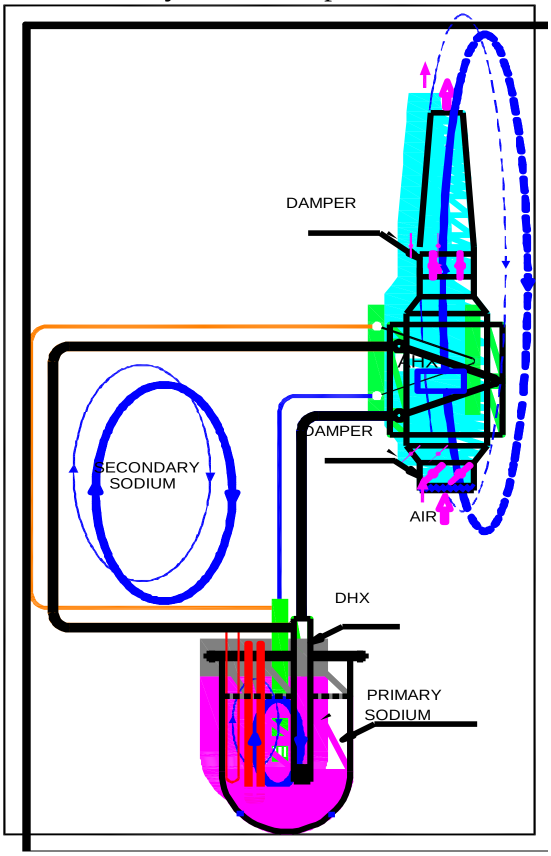


Fig. 2: Schematic of passive decay heat removal system for experimental studies

While scaling down the Prototype system to the experimental facility the ratio of Ri and Eu in the prototype and model is maintained as unity. The ratio of Pe and Re in the model is maintained in same order as in the prototype system [5]. This ensures the reproduction of most of the physical phenomenon present in the prototype during steady state operation to the scaled down experimental model. The capacity of SADHANA loop is 355 kW and the height difference between the thermal centres of DHX and AHX is 19.5m. In SADHANA the sodium in test vessel which simulates hot pool of PFBR is heated by immersion type electrical heaters. This heat is transferred to the intermediate sodium through the model DHX. The intermediate sodium gets circulated in the intermediate loop by the buoyancy head developed in the loop due to the temperature difference in hot and cold legs of the loop. The heat from intermediate sodium circuit is rejected to the atmosphere through the AHX. A 20m high chimney develops the air flow required to transfer the heat from secondary sodium to the atmosphere through AHX.

The facility was commissioned and has completed around 5000 hr of high temperature operation. The transport of decay heat from the reactor hot pool to atmosphere through passive decay heat removal system is demonstrated and the design of the system was validated by experiments in the SADHANA facility. The results obtained from various experiments verified the adequacy of the Safety Grade Decay Heat Removal (SGDHR) system to remove the anticipated amount of decay heat from reactor hot pool. With a sodium pool temperature of 550°C, the system transported heat at the rate of 425kW from the sodium pool to atmosphere which is 19.4% more than its nominal capacity [6]. The availability of the SGDHR system followed by an unprotected reactor trip depends on the dynamic response of the system after the sudden opening of the AHX dampers and found that the system will be fully functional in around 510 seconds after the initiation of opening the dampers [7].

A finned tube cross flow sodium to air heat exchanger model with 2MW heat transfer capacity with sodium on tube side and air on shell side was tested in the Steam Generator Test Facility for its design validation. Heat transfer experiments were carried out with forced circulation of sodium and air. The AHX was operated for more than 10,000 hours at rated temperature. The heat transfer capability of the heat exchanger is as expected and the performance of the heat exchanger is as intended in the design. The testing of AHX confirmed the adequacy of heat removal capacity of the heat exchanger at nominal temperature and flow conditions. During the test AHX transferred 2.34 MW of heat power from sodium to air with rated flow and temperature conditions [8].

2.3. Performance Testing of Transfer Arm

Transfer Arm (TA), the in vessel fuel handling machine of PFBR, is used for transferring Fuel Sub-Assemblies (FSA) within the reactor i.e. inside the main vessel of the rector. It transfers FSA within the core and also between the core and Transfer Pot (TP), of ex-vessel fuel handling machine, located in the periphery of the core. Fig. 3 shows a view of transfer arm when it was undergoing testing.

Fig. 3: A view of transfer arm under testing

The machine was first tested in air at room temperature. These tests were useful in identifying and rectifying problems that hampered the functioning of the machine. The machine was then re-assembled on the sodium test vessel and hot air testing with 60°C to 150°C for 30 cycles followed by hot argon testing with 60°C to 150°C for another 30 cycles was carried out. These tests were carried out because it was easier to identify and rectify problems resulting from high temperature operation in air/argon compared to that in sodium. This was followed by in sodium testing simulating the operating environment in the reactor.

In sodium testing was carried out for 10% of the operating cycles that TA undergoes during 40 years of reactor life. Performance testing consisted of cyclic testing in sodium at 200°C interspaced with two dwell periods in sodium at 400°C and 547°C. Sodium testing was carried out in a specially erected test vessel. A portion of the core, consisting of a central sub assembly surrounded by the adjacent ring of seven sub-assemblies, was simulated. The SA being handled was positioned at the centre of the seven SA cluster and each cycle consisted of removal of the sub assembly from the centre of the core, insertion of the sub assembly in the Transfer Pot, removal of the sub assembly from the Transfer Pot and insertion of sub assembly back into the core centre. The cycles of testing was also carried out using a SA with 50 mm bowing. After resumption of initial testing at 200^{o}C for 61 cycles, difficulty was experienced with operation of gripper fingers and raising guide tube. Investigation was carried out after dismantling of machine from the vessel. Then problem was identified and suitable modification was carried out. After carrying out the required modifications, testing of TA was resumed and 600 cycles of sodium testing was carried out. In this campaign, after completion of the initial 300 cycles, 100 hours of dwell period at 547^{o}C was done to study the performance of the machine after high temperature reactor operating conditions.

During testing, it was found that overall performance of the machine was satisfactory. No scoring marks were observed on the rubbing surface of the relatively moving components. No noise was observed throughout the testing. Argon pressure in the space between IT and OT was measured intermittently. Negligible drop in the pressure was ensured the healthiness of the bellows provided between IT and OT. Load cell reading for hoisting the gripper assembly with FSA and torque values of different motors were well within the acceptable limits. After completion of testing, sodium cleaning of TA was done in-situ using CO_2-water vapour process using a sodium removal system that was made exclusively for this purpose. TA was then dismantled completely from the test vessel and water cleaning of the individual components was done. Visual inspection of all components was done and absence of scoring marks or rubbing marks on the working surfaces confirmed.

2.4. Testing and Qualification of Inclined Fuel Transfer Machine

Inclined Fuel Transfer Machine (IFTM) is the ex-vessel fuel handling machine of PFBR, which is used for the transfer of sub-assemblies between the in vessel transfer position (IVTP) located inside the reactor and the ex-vessel transfer position (EVTP) located in the fuel building. It consists of sodium filled transfer pot to carry the fuel sub assembly, hoisting mechanism for transferring the transfer pot with FSA and rails assembly for guiding the transfer pot during its movement. Once spent FSA is kept inside the transfer pot at IVTP location, it will be hoisted by chain and sprocket arrangement of hoist mechanism. During hoisting transfer pot travels through Primary Tilting Mechanism and primary Ramp into the rotatable shielded leg located above the roof slab of the reactor. The transfer pot is then lowered to EVTP through secondary ramp and secondary tilting mechanism. Fig. 4 shows the schematic sketch of IFTM.

The testing plan was for 10% of envisaged life in reactor, which is 600 cycles and is divided into two stages, viz. Stage 1 in which the primary ramp and primary tilting mechanism along with transfer pot were tested in sodium, followed by Stage 2 where the integrated machine was tested in hot air. In tests in Stage 1, performance testing of primary ramp and primary tilting mechanism was carried out in Test Vessel along with non-reactor grade components which are necessary for qualification testing. The effect of operation at rated temperature of 547^{o}C on aerosol deposition in relatively cold regions was also simulated by means of two dwell period tests each of 150 h duration at 547^{o}C. The dwell periods were interspersed between the cyclic tests. The system was tested for a total of 510 cycles in sodium. The performance of the system was satisfactory. After sodium cleaning, visual examination and liquid penetrant examination was done to confirm absence of any defects on the components.

In the tests in Stage 2, integral testing of machine in hot air was carried out under nuclear clean condition. During hot air testing of the IFTM, non-reactor grade primary ramp & primary tilting mechanism were heated by tape heaters such that primary tilting mechanism is maintained at 200°C and primary ramp temperature varies from 200°C to 140°C from bottom to top. Hot air was used for heating Rotatable Shield Leg and sprocket in argon chamber to 150ºC simulating hot argon heating in reactor. Hot air testing of IFTM for 600 cycles was carried out after this. During testing the movement of transfer pot was smooth. The motor current and torque recorded were found to be within the specified limits for both hoisting and rotatable shield leg rotation. The leak tightness of the machine was good throughout the testing.

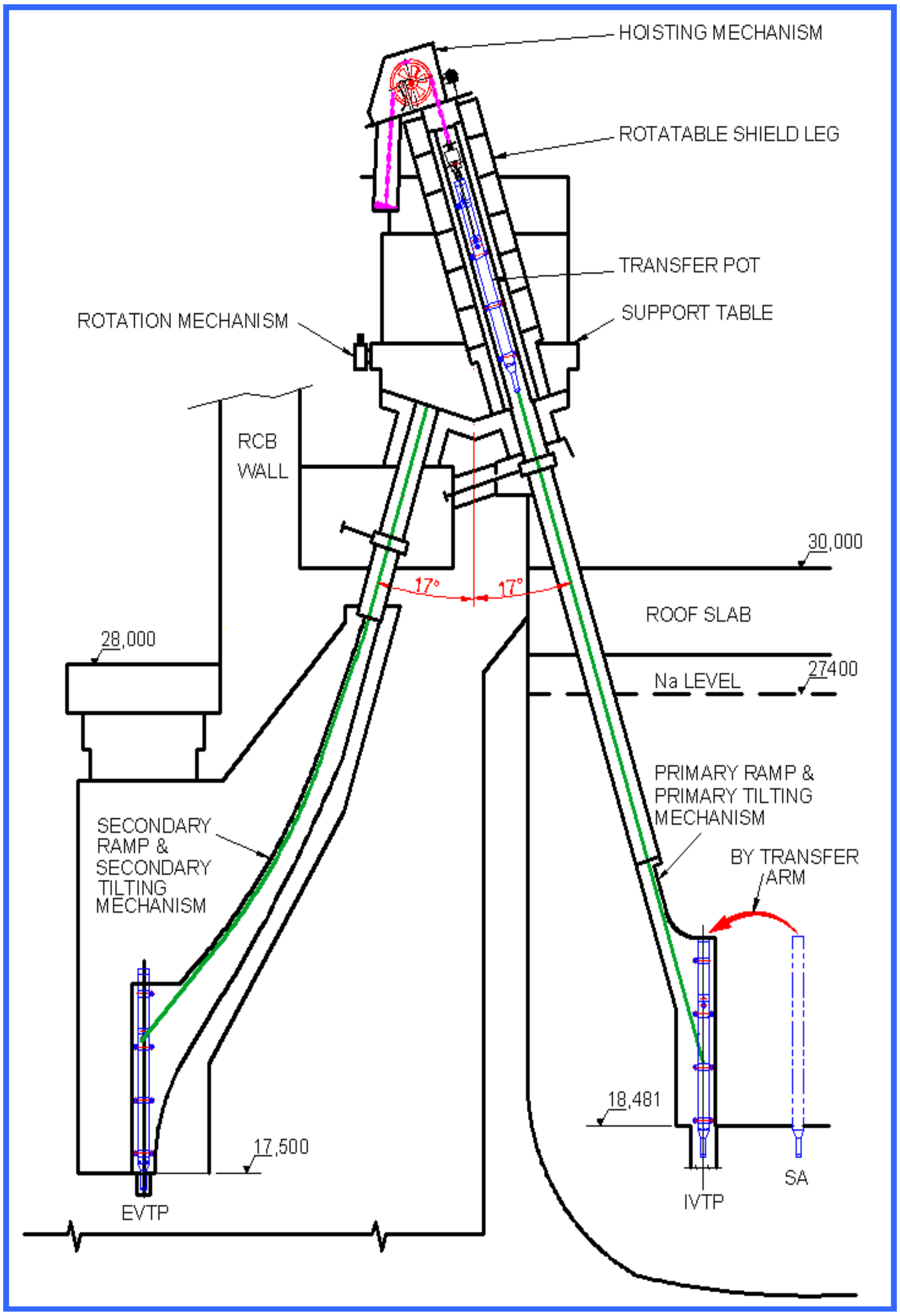


Fig. 4: Inclined fuel transfer machine

2.5. Development of Under Sodium Ultrasonic Scanner (USUSS) for PFBR

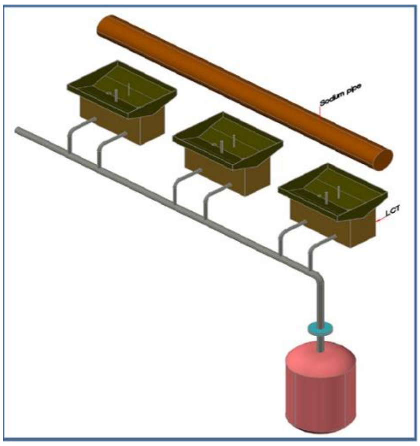
An Under Sodium Ultrasonic Scanner (USUSS) has been developed for PFBR to detect protrusion and growth of the fuel sub assemblies before every fuel handling operation. For qualifying the scanner to use in the reactor, the scanner along with the ultrasonic transducers, control and drive system of automation and ultrasonic imaging system were tested in water and in sodium. USUSS consists of upper and lower parts. The upper part consists of two AC servo motors for translational movement and rotational movement of the spinner tube. The lower part consists of 7.5 metres long spinner tube with transducer housing at the bottom with ultrasonic transducers. A USUSS assembly is shown in Fig. 5.

Fig.5: Scanner Assembly

In order to test the functionality of the scanner in water and sodium, a target assembly consisting of 19 Sub-Assemblies (SAs) simulating the central SA, first and second rings of PFBR core configuration and 5 SAs simulating the protrusion of various heights and orientation. Imaging of SA top surface was carried out in water and sodium using ultrasonic imaging system. Transducer holder was rotated by 360° in steps of 1° and echo signals were acquired from top surface of the SAs. The USUSS has been qualified for use in PFBR by satisfactory performance during testing in water and in sodium.

2.6. Testing of Leak Collection System of PFBR

Leak collection system is a passive sodium fire mitigation system erected in PFBR for handling large scale sodium leak. The system consists of Leak Collection Trays (LCT), drain lines, fusible plug and dump tank. Fig. 6 shows the configuration and details of leak collection trays used in PFBR. Experiments were conducted with water based leak collection system to study the possible spillage of water from the LCT. It was observed that a maximum of 49% spillage was observed at 4 bar pressure inside the pipe. Arrangements with four layers of square type wire mesh to arrest the spillage was introduced in LCT and the arrangement was found to be very effective in suppressing the spillage below 2%. The dynamic viscosity of sodium at operating condition is almost similar to that of water at test condition, which dictates the spillage behavior. Repeating many tests with sodium is really a cumbersome task. Hence for studying the spillage behavior water is used as test fluid. When sodium leaks into the leak collection tray sodium burning takes place only at the top surface. The reaction products dissolves in sodium which may increase its viscosity marginally. Otherwise the effect of sodium burning on the operation of leak collection tray is very little.



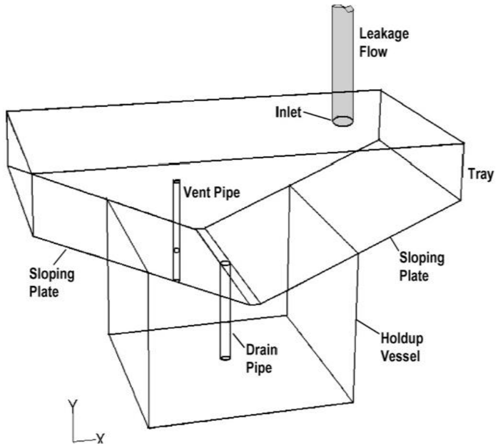


Fig. 6:Configuration of Leak Collection Tray

Another set of experiment was conducted to test melting of fusible plug. Woods metal is the preliminary choice for the fusible plug with 72ºC as melting point. About 10 kg of sodium was loaded in the sodium tank fitted with rupture disk at the bottom and heated to 300 ^{o}C using surface heaters. The sodium tank was pressurised by using argon gas to rupture the disc and hot sodium was drained through 100 NB pipe line into fusible plug assembly. The woods metal disc in the fusible plug melted by absorbing heat from hot sodium and the sodium drained to the bottom holdup vessel. The melting time of fusible plug was around 10 s during gravity flow of sodium and less than 2 s at forced sodium flow of 1.7 bar pressure.

2.7. Testing of steam generator

Steam generator (SG) is a crucial component in a nuclear power plant because its availability is directly linked to the availability of heat transport system and thus the plant availability. A model steam generator with 19 tubes and of 5.5 MWt nominal power which simulates the 547 tube, 156 MW in PFBR was tested in Steam Generator Test Facility (SGTF) for validating the thermal hydraulic and mechanical design of the steam generator. The length of steam generator is 23 m and it produces super-heated steam at 17.2 MPa pressure and 493°C temperature. The material of construction of the model steam generator is Mod 9 Cr 1 Mo ferritic grade steel. The sodium and water / steam side pressure and temperature conditions of the model SG are same as that of PFBR.

Heat transfer experiments were conducted with the model steam generator to evaluate the heat transport capability of once through sodium heated steam generator. From the experimental data it is estimated that the steam generator has 8.3% more tube surface area than that required to produce steam at nominal conditions [9].Characteristics of two phase flow pressure drop through steam generator tubes were established using the model steam generator. Thermal hydraulic stability of steam generator for steady state and plant transient cases were studied and established that the steam generator is free from the instabilities for all its normal operating regimes. Flow induced vibration studies were carried out in the same SG model and established that the tube vibration due to sodium flow is well within the design values. In service inspection tools developed for testing the integrity of SG tubes were tested in the model SG and their usefulness is verified.

Rate of diffusion of hydrogen through steam generator tubes has been established by experiments. Hydrogen leak detection sensors were tested in SGTF as part of their qualification for reactor usage. Cold traps used in the secondary sodium system of PFBR are required to be regenerated after every 56 months of operation due to excess hydrogen loading. The procedure for cold trap regeneration was tested in SGTF with a model cold trap and the process parameters were optimized.

3. Summary

Sodium testing is one of the important steps for the design validation and qualification of reactor components. Sodium testing of PFBR shutdown systems and fuel handling mechanisms were useful for confirming their operation in the reactor. In addition the performance of safety graded decay heat removal system is validated using scaled down experiments in sodium. Number of tests and experiments were carried out to validate thermal and mechanical design of steam generator in a dedicated steam generator test facility. Various experiments and testing of components helps in perfecting the components and systems and ensures their trouble free operation throughout their design life of 40 years.

References

[1] Chetal S.C., et al., "The design of prototype fast breeder reactor", Nuclear Engineering and Design 236 (2006) 852-860.

[2] Srinivasan G., et al., "The Fast Breeder Test Reactor—Design and operating experiences", Nuclear Engineering and Design 236 (2006) 796–811.

[3] Rajan Babu, V., et al., "Testing and Qualification of Control & Safety Rod And Its Drive Mechanism Of Fast Breeder Reactor", Nuclear Engineering And Design 240 (2010) 1728-1738.

[4] Vijayashree R., et al., "Design, development, Testing & qualification of Diverse Safety Rod & Drive Mechanism for a prototype fast breeder reactor", Journal of engineering for gas turbines & power132(2010) 102921-9.

[5] Padmakumar G., et al., "SADHANA Facility for Simulation of Natural Convection in the SGDHR System of PFBR", Journal of Progress in Nuclear Energy 66 (2013) 99-107.

[6] Vinod V., et al., "Experimental evaluation of safety grade decay heat removal in prototype fast breeder reactor", Nuclear Engineering and Design 265 (2013) 1057–1065.

[7] Vinod V., et al., "Experimental study on the transient response of passive decay heat removal system", Nuclear Engineering and Design 280 (2014) 564–569.

[8] Vinod V., et al., "Experimental evaluation of sodium to air heat exchanger performance", Annals of Nuclear Energy 58 (2013) 6-11.

[9] Vinod V., et al., "Experimental evaluation of the heat transfer performance of sodium heated once through steam generator", Nuclear Engineering and Design 273 (2014) 412–420.