Thermal Stratification Analysis for Sodium Fast Reactors

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The sodium fast reactor (SFR) is the most mature reactor concept of all the generation-IV nuclear systems and is a promising reactor design that is currently under development by several organizations. The majority of sodium fast reactor designs utilize a pool type arrangement which incorporates the primary coolant pumps and intermediate heat exchangers within the sodium pool. These components typically protrude into the pool thus reducing the risk and severity of a loss of coolant accidents. To further ensure safe operation under even the most severe transients a more comprehensive understanding of key thermal hydraulic phenomena in this pool is desired. One of the key technology gaps identified for SFR safety is determining the extent and the effects of thermal stratification developing in the pool during postulated accident scenarios such as a protected or unprotected loss of flow incident. In an effort to address these issues, detailed flow models of transient stratification in the pool during an accident can be developed. However, to develop the calculation models, and ensure they can reproduce the underlying physics, highly spatially resolved data is needed. This data can be used in conjunction with advanced computational fluid dynamic calculations to aid in the development of simple reduced dimensional models for systems codes such as SAM and SAS4A/SASSYS-1.

I. THERMAL STRATIFICATION OVERVIEW

The thermal-hydraulic properties of liquid sodium make it an attractive candidate for use as a coolant in Generation-IV sodium fast reactors (SFR). With a high thermal conductivity and low Prandtl number, liquid sodium provides efficient heat transfer in the core and intermediate heat exchangers of the reactor[1].

One phenomenon which is of particular interest in a SFR is the occurrence of thermal stratification in the large volume of the hot pool. During accident scenarios it may be possible for a stratified layer between hot and cold sodium to develop an oscillating temperature field on mechanical structures. This temperature oscillation could reduce the life of effected components. Thermal stratification in a reactor pool occurs when coolant from the reactor core is at a significantly different temperature than the coolant in the reactor pool during a loss of flow scenario[1]. In one scenario hot sodium exits the core and converges with the buoyancy forces of the colder, and denser, sodium as it flows downwards in the plenum. This creates transient currents of non-isothermal sodium which can induce thermal stress in proximal mechanical structures[1] (Figure 1).

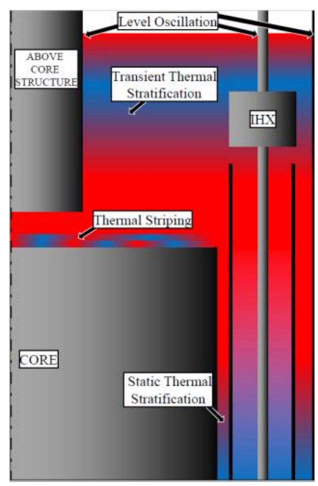


Fig. 1. Illustration depicting various thermal hydraulic effects which cause thermal fatigue in pool type sodium fast reactors.

Thermal stratification is primarily seen during loss of flow scenarios. Loss of flow scenarios are initiated during a reactor power trip. The two distinct incidents are the protected loss of flow (PLOF) and the unprotected loss of flow (ULOF)[2]. During the PLOF, the reactor successfully scrams which significantly decreases the temperature of the reactor core (Fig. 2)[2]. This releases relatively cold coolant from the core into the hotter reactor pool. Conversely, during a ULOF, the reactor is not scrammed and the core releases hot sodium from the core into the colder reactor pool (Fig 3)[2]. The decay heat removal systems then help to cool the reactor slowly. However, during the time it takes the reactor to safely shutdown and remove decay heat the reactor is subjected to temperature oscillations and high temperature gradients associated with thermal stratification[3].

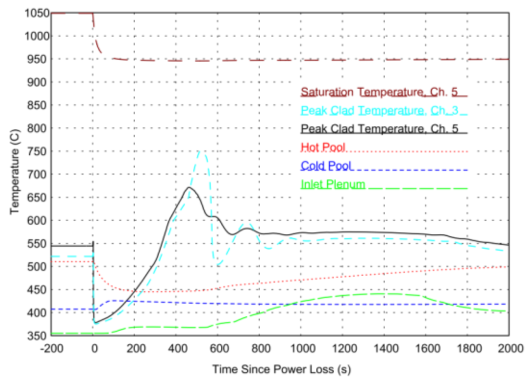


Fig. 2 Temperature fluctuation over time during a PLOF transient in the Argonne National Laboratory ABTR test simulations[2].

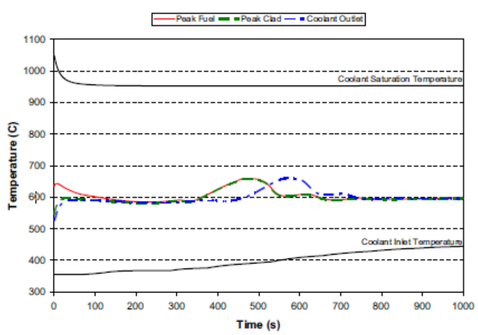


Fig. 3 Temperature fluctuation over time during a ULOF transient in the Argonne National Laboratory ABTR test simulations[2].

Thermal stratification has been a key point of interest in both commercial reactors, such as the MONJU reactor in India[1], and design reactors, such as the ABTR developed by Argonne National Laboratory[2]. Possessing high fidelity thermal stratification experimental data to validate computational models drastically increases the safety of reactors. Furthermore, understanding possible issues related to thermal stratification during the design process of a reactor is extremely important to reactor licensing. Accurate computational models will be required to aid in the safety analysis of a commercial reactor design[2]. These computational models include both systems and computational fluid dynamic (CFD) codes. In order to improve the CFD codes' ability to predict subtle and complex thermal phenomena, it is important to obtain high fidelity experimental data to use as a comparison and validation tool. The goal of this project is to study thermal stratification issues with the aid of experiments customized for comparison with CFD analysis, and then to use this information to formulate models for input into systems codes.

II. EXPERIMENTAL DESIGN

Thermal stratification in an SFR occurs when buoyancy forces of the sodium flow become significant with respect to the inertial forces of the fluid. This phenomenon is characterized by the Richardson number (Ri). The Richardson number is the ratio of the buoyancy to inertial forces. It can be used to scale and quantify the extent and existence of thermal stratification. (Equation 1)[4].

$$Ri = \frac{g\beta \Delta TL}{V^2} = \frac{Buoyancy}{Inertia} \tag{1}$$

where g is gravitational acceleration, ß is the thermal expansion coefficient, ΔT is the average axial temperature gradient of a sodium volume, L is the characteristic length, and V is the characteristic sodium velocity.

Three other key dimensionless numbers of importance when examining stratified volumes are the Peclet (Eq. 2), Grashof (Eq. 4), and Reynolds (Eq. 3) numbers[4].

$$Pe = \frac{V_c L}{\alpha^2} \tag{2}$$

$$\mathrm{Re} = \frac{V_c L}{\nu} \tag{3}$$

$$Gr = \frac{g(\rho_p - \rho_c)\Delta TL^3}{\rho_c \nu^2} \tag{4}$$

For the Peclet number, L is the characteristic length, V_c is the characteristic velocity jetted from the core, and α is the thermal diffusivity. For the Grashof and Reynolds numbers, ρ_p is the density of the ambient fluid, ρ_c is the density of the fluid in the core, ΔT is the temperature difference between the hot and cold sodium, L is the characteristic length, ν is the dynamic viscosity, and V_c is the velocity of the sodium jetted from the core. The Richardson number is further defined by the relationship of the Grashof number divided by the square of the Reynolds number.

The Peclet number is used to describe the natural phenomena of heat transport along a specific length. The Grashof number is a ratio of the buoyancy to viscous forces. When the jet from the reactor core impinges on the upper internal structure (UIS) of the reactor the fluid begins to mix. When analyzing stratification, it is important that flow out of the core be turbulent as it enters the reactor pool to ensure thorough mixing of the hot and cold fluid to cause thermal stratification[4]. These are the parameters of primary interest during the design of the thermal stratification experimental facility.

II.A System Design

In order to develop a relevant experimental facility it was important to incorporate key thermal hydraulic parameters and design features of a well-documented reactor. Thorough computational studies were conducted for the ABTR[2]. Therefore, the ABTR design was used as benchmark for the scaled design of the present thermal stratification experiment. Proper scaling of pool-type sodium fast reactors requires analysis of dimensionless numbers specific to an experimental campaign. Experimental modeling of thermal stratification relies most heavily on the Richardson, Peclet, and Reynolds numbers. This requires that the length scale be defined specifically in each dimensionless number. In literature, the length scale associated with pool-type reactor experiments varies[5]. University of Wisconsin-Madison has developed a primary length scales for use in designing a scaled model of the ABTR to analyze thermal stratification phenomena.

Literature advises that the Richardson number is the primary indicator for the existence of thermal stratification in a pool-type reactor[5]. Thus similitude in the Richardson number between the experiment and the actual reactor is important when analyzing thermal stratification. Keeping similitude with the Richardson number ensures that similar flow characteristics are being created in the experiment as are present in the reactor design.

Literature further suggests that the ratio between model and experiment for the Peclet and Reynolds numbers be as close to similitude as possible, barring experimental constraints[5]. This work is being conducted to scale the ABTR for experimental analysis of thermal stratification during loss of flow events, both protected and unprotected (PLOF and ULOF). In order to do so, the correct length scale had to be defined and validated for thermal stratification analysis.

In 1994 Peterson wrote an extensive paper surrounding analyzing thermal stratification as a result of buoyancy driven jets in a large volume. Peterson defines the length scale for the Richardson number as follows[6]:

$$L(Ri) = H_{sf}\left(\frac{H_{sf}}{d_{bjo}}\right)^2\left(1+\frac{d_{bjo}}{4\sqrt{2}\alpha_T H_{sf}}\right)^2, \qquad (5)$$

where H_{sf} is the stratified height on the volume, d_{bjo} is the buoyant jet diameter, and α_T is the entrainment constant. This is representative of a large stratified volume. However, in the case of a pool type reactor the buoyant jet immediately impinges on the UIS. This allows the term associated with the entrainment coefficient to be neglected and the equation for the characteristic length simplifies to:

$$L(Ri) = H_{sf}\left(\frac{H_{sf}}{d_{bjo}}\right)^2 . \qquad (6)$$

Equation 6 represents the length scale used to define the Richardson number for the ABTR and the experimental design in computational models. This length scale implies that the height of the pool is used to define the characteristic length of the Grashof number while the diameter of the core is used as the characteristic length to define the Reynolds number. Keeping similitude in the Richardson number between the experimental design and the ABTR was done by solving for the Richardson number of the ABTR with the parameters given in the 2008 ANL ABTR design report[2]. The parameters for the experimental test section were designed to keep similitude with the ABTR while still creating a facility large enough to produce higher ratios between the Peclet and Reynolds numbers. This involved analyzing the ABTR using a volumetric flow rate through the core that was seen during a PLOF or ULOF scenario in the ABTR[2]. This scaling analysis yielded the results provided in Table I.

TABLE I. Experimental design and ABTR characteristics. Ratio is defined as the particular experimental design parameter divided by the associated ABTR parameter.

Parameter	Experimental Design	ABTR	Ratio
H_{pool} [m]	1.2	8.02	0.15
$H_{outlet,1}$ [m]	0.4	2.673	0.15
$H_{outlet,2}$ [m]	0.8	5.347	0.15
D_{pool} [m]	0.3147	4.91	0.065
D_{UIS} [m]	0.1455	2.27	0.065
D_{core} [m]	0.1455	2.27	0.065
Q_{ULOF} [m^3/s]	0.001429	0.3775	0.0038
V_{core} [m/s]	3.759	0.2960	12.7
T_{hot} [C]	300	575	0.52
T_{cold} [C]	250	525	0.48
Peclet [-]	1616	12520	0.13
Reynolds [-]	28349	788237	0.04
Richardson [-]	1616	1616	1

The data shown in Table I were used to design the experiment (Fig. 4). These data represent the top end of the flow spectrum that is to be analyzed. This is at a 25% of full flow in the ABTR. The 25% flow ratio was indicative of the beginning of stratification according to the ABTR flow reports[2]. Initial flow coast down from full power to 25% happens very rapidly in both the ULOF and the PLOF scenarios[2].

Fig. 4. Experimental design based on 1:1 similitude with the ABTR's Richardson number.

Two different outlet heights are used to examine the effects of stratification when the coolant is released at different heights. In a reactor pool the coolant flows out of the pool into an intermediate heat exchanger (IHX). Different reactor designs use different IHXs at different heights inside of the pool[2]. This experimental design uses outlets at 1/3 and 2/3 of the overall height to examine this feature of a nuclear reactor.

This design was then analyzed utilizing un-steady Reynolds-averaged Navier-Stokes (URANS) with cubic anisotropy to account for 3D buoyancy effects. This analysis demonstrated that stratification would exist for the 25% flow scenario (Fig. 5). Figure 5 shows the temperature distribution in the y-z plane at interval times for the high flow rate (20 [GPM]) cold sodium (250[°C]) injection into hot sodium (300[°C]) case.

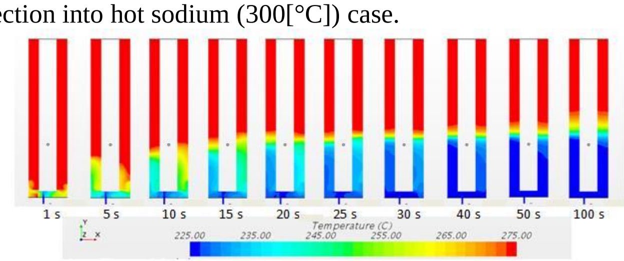


Fig. 5 Temperature distribution in the y-z plane at different times for the experimental design cold injection into hot sodium case.

The aforementioned CFD case shows that thermal stratification has occurred at the lower outlet height during higher flow rate scenarios. This is considered the least likely case for stratification to occur in a smaller design. Thus, the CFD model demonstrates the proposed experimental facility is appropriate for attaining high fidelity stratification data in liquid sodium.

II.B Thermal Stratification Experimental Facility

This system was designed to coincide with the parameters determined through dimensional analysis. The following design is currently under construction at the University of Wisconsin – Madison (Fig. 6). The function of the loop is shown in Figure 7.

Fig. 6 Current state of the thermal stratification experimental facility at the University of Wisconsin – Madison.

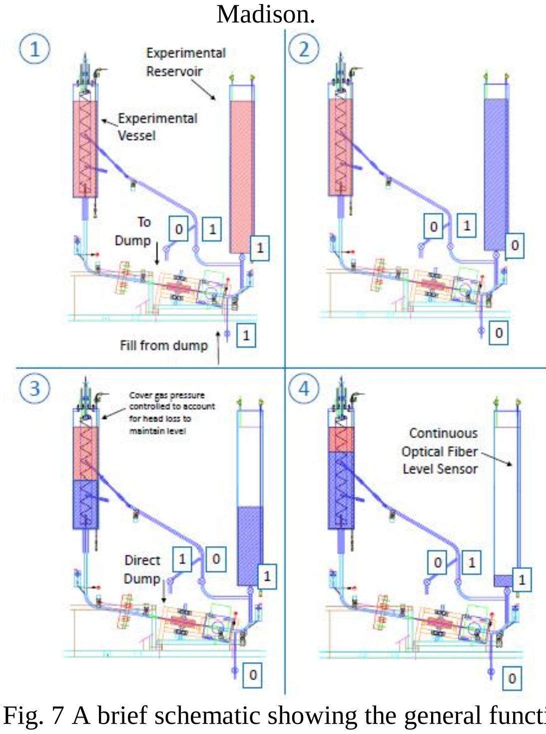


Fig. 7 A brief schematic showing the general function of the liquid sodium thermal stratification experimental facility broken into 4 steps[7]. Note that each valve is labeled with either a 0 signifying fully closed or a 1 signifying fully open.

The facility will be filled with isothermal sodium from a 300 gallon sodium reservoir (Step 1). The experimental reservoir will then be isolated from the experimental vessel and the desired temperature difference between both vessels will be reached (Step 2). During Step 2, a moving magnet pump (MMP) will be used to circulate sodium in the primary loop at the desired flow rate for a specific experimental run[8]. The primary loop will then be closed and sodium from the experimental reservoir will be allowed to flow through the experimental vessel (Step 3). From the experimental vessel, the sodium is released back into the 300 gallon sodium fill/dump tank. The experiment will then be stopped by turning off the MMP and shutting of the valve isolating the 300 gallon tank from the experimental facility (Step 4). The facility can then be refilled with sodium of an isothermal temperature and a new experiment can begin.

II.C Instrumentation

The experimental facility is equipped with type K thermocouples, optical fiber temperature sensors, and optical fiber level sensors. This instrumentation has been specifically selected to help obtain high fidelity data as well has have precise control of the facility. Deployment of optical fiber temperature sensors in sodium has been a thoroughly researched topic by Weathered et al[9]. Weathered et al. also developed a use for the optical fibers as a precise level indicator for sodium in a uniform cylinder[10]. Distributed optical fiber temperature sensors will be used throughout the course of the stratification experiments to acquire high spatial resolution temperature profiles. Utilizing optical frequency domain reflectometry of Rayleigh backscatter from a near infrared laser scan, the fiber optical sensors can acquire 1D temperature readings every 0.653mm over 10 meters.

II.D Experimental Vessel Design and Fiber Deployment

The top cap of the experimental vessel is built such that there are 12 ports for deployment of fiber optic instrumentation (Fig. 8 & 9). One port will be used to deploy a fiber optic level sensor. The other ports will be used to analyze the axial temperature distribution along the length of the vessel. A removable flange is used to hold the UIS in place during an experimental run. The UIS will be helically wrapped with an optical fiber temperature sensor for a more informative thermal map of the experimental vessel. The flange can be rotated between experimental runs to gain an exceptional understanding of thermal stratification inside the vessel pool.

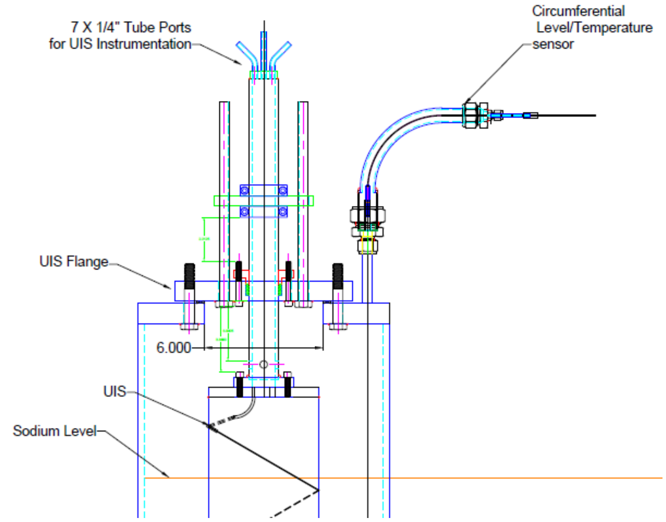


Fig. 8 The upper portion of the experimental vessel.

Sodium will be introduced into the experimental vessel through three 0.5” inlets. This is designed to be representative of coolant jetting from a reactor core into the reactor pool. The three inlets together generate a slug of sodium that will impinge on the bottom of the UIS. Sodium flow from each inlets may be independently restricted or shut off, depending on experimental requirements. The inlet configuration is shown in Figure 9.

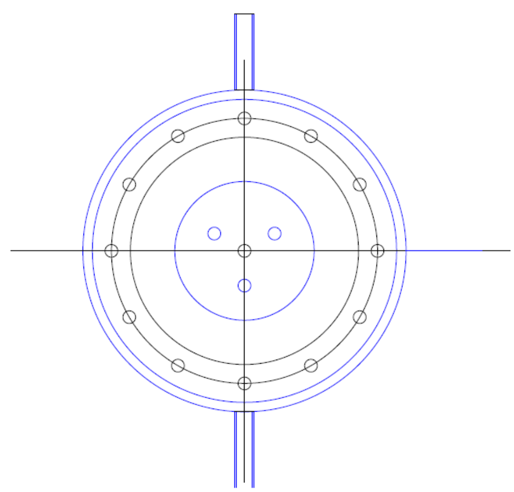


Fig.9 Bottom view of the experimental vessel. Three inlet pipes are surrounded by twelve optical fiber pass through locations.

V.A. Experimental testing procedure

The system will first be filled entirely with liquid sodium fed from the 300 gallon sodium dump tank at approximately 120 [°C]. Once the system has been entirely filled it will then be isolated from the dump tank. The sodium will then be pumped continuously through

Loop 1 using the moving magnet pump. This sodium will be brought up to a constant desired temperature. Meanwhile, the sodium in the reservoir tank will be brought to a separate temperature desired for testing. Once the experiment has reached its desired temperatures, the MMP will be changed to the experiments designated flow rate. Once the flow rate is achieved the bypass valve will be closed. Then the lines from the experimental vessel to the dump tank and from the feed tank to the experimental vessel will be opened. This will allow sodium to be purged from the feed tank through the experimental vessel at a constant temperature difference. The experimental runtime is dependent on the flow rate as there is a limited amount of sodium in the feed tank. Once the feed tank is void of sodium the experiment is over. Residual sodium left within the system will be drained back into the dump tank.

The initially desired test will be to inject cold sodium into a hot pool as this produces the greatest stratification. This will be done at a flow rate of approximately 5 [gpm]. The sodium will then be allowed to exit at the 2/3 height outlets. A UIS of diameter 5.5" will rest 2" above the core inlet. All of the aforementioned parameters can be changed depending on the desired experimental campaign. Initial testing will comprise primarily of varying the inlet flow rate. Then running similar experiments with one of the inlets plugged to asses CFD accuracy for model prediction. Eventually a loss of flow scenario will be tested to model the flow coast down of pumps inside actual reactors. The pump coast-down will be modeled according to the ABTR PLOF and ULOF flow data. Using a variable frequency drive, the MMP can be finely controlled to simulate both the PLOF and ULOF scenarios. Shown in Figure 10, are the experimental capabilities of the stratification facility, given the amount of sodium in the experimental reservoir and the achievable flow rates.

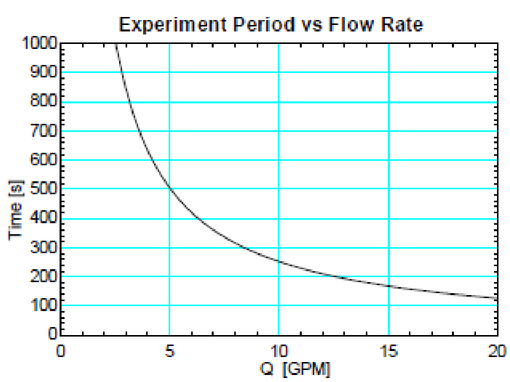


Fig. 10 The experimenatal time allauded given a specific flow rate.

In parallel with the acquisition of experimental data, CFD analysis will be conducted to further assess the computational capabaialities of CFD codes. The experimental data will serve as validation to the CFD codes.

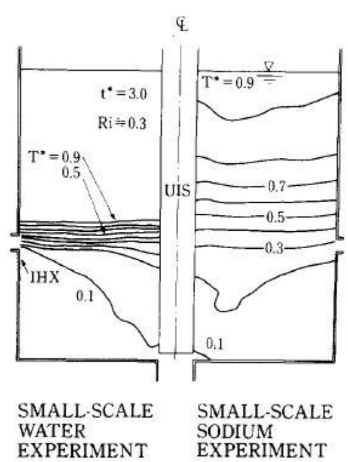
III. COMPUTATIONAL FLUID DYANMICS

Computational Fluid Dynamics (CFD) serves a dual role in the project. Initially, CFD will be used to support the design of experiment. Later, CFD will be used to advance the understanding of the three dimensional thermal mixing and support the progression of thermal stratification modeling, a key requirement in advancing the safety of nuclear reactors.

In this first phase of the project, the key contribution from CFD analyses has been to provide insights to the experimentalists in order to drive the design of the stratification facility. The effort has included three successive steps: (1) deliver confidence that CFD can predict stratification phenomena consistently with experiments and provide a useful design tool; (2) provide insights on the evolution of the thermal transient for a representative ABTR geometry; (3) support the evaluation and refinement of the final test facility configuration.

In order to assess the applicability of the CFD methodology, simulations have been performed for the case of thermal stratification in a cylindrical plenum with sodium, in accordance with a published experiment conducted by Tanaka et al. in 1990[4]. While the coarseness of the measurement data does not allow a fully quantitative benchmarking, the flow configuration is particularly valuable and provides an excellent platform to judge the applicability of the CFD method to the design of experiment.

In the experiments performed by Tanaka, a cylindrical plenum was used (Fig. 11) with sodium flow entering from the bottom, and leaving the plenum through a circular outlet slit. Thermally stratified layers were generated by suddenly decreasing the inlet temperature while keeping the flow rate constant.



g. 4. Comparison of temperature distribution obtained in water and sodium tests.

Fig. 11. Tanaka's experimental findings

Consistent with the experiment, a CFD model was built for the same geometry with sodium as the working fluid (Fig. 12). Unsteady RANS with cubic anisotropy consideration was used for the transient simulation, to include an accurate description of the 3-dimensional buoyancy effects[11].

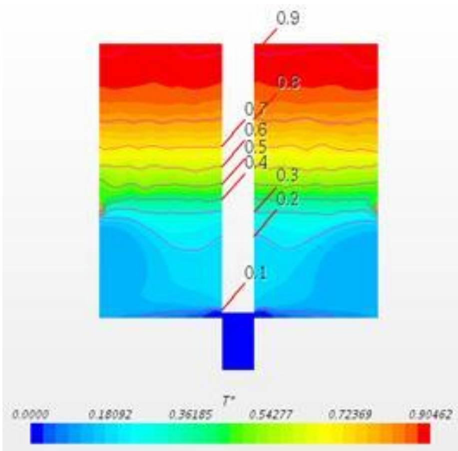


Fig. 12 CFD results from Tanaka experiment replication.

To support the design of the stratification experiment, CFD has been leveraged as it provides the ability to predict complex 3-dimensional flow and thermal distributions with high accuracy. In order to provide confidence on the applicability of the CFD method, it has firstly been compared to existing experimental measurements for thermal stratification in a cylindrical plenum with sodium. Good comparison with published experimental data has validated our CFD tool and model set up. In parallel with the acquisition of experimental data CFD simulations will be run. Once CFD simulations have been validated by experimental data, CFD simulations can then be used to inform models built for systems codes.

IV. SYSTEMS CODE DEVELOPMENT

This project focuses on two particular system codes, both of which are under development at Argonne National Laboratory: NEAMS System Analysis Module (SAM) [6] and SAS4A/SASSYS-1[12].

The SAS4A/SASSYS-1 computer code is a system level code developed by Argonne National Lab (ANL) for thermal, hydraulic, and neutronic analysis for power and flow transients in liquid-metal-cooled fast reactors[12]. In more recent years, several modeling additions and enhancements have been made to meet U.S. DOE programmatic needs, including efforts to couple the code with external CFD simulations to resolve flow distribution and thermal stratification[13].

Once fully developed, SAM is expected to provide fast-running, improved-fidelity, whole-plant transient analyses capabilities. SAM utilizes the object-oriented application framework MOOSE, and its underlying meshing and finite-element library libMesh, as well as linear and non-linear solvers PETSc, to leverage modern advanced software environments and numerical methods. SAM also takes advantage of advances in physical and empirical models and seeks closure models based on information from high-fidelity simulations and experiments. SAM employs a 1-dimensional transient model for single-phase incompressible but thermally expandable flow. A 3-dimensional module is also under development to model the multi-dimensional flow and thermal stratification in the upper plenum or the cold pool in an SFR reactor vessel.

The SAS4A/SASSYS-1 code performs thermal, hydraulic, and neutronic analysis of power and flow transients in liquid-metal-cooled nuclear reactors. SAS4A/SASSYS-1 can be used to perform deterministic analysis of anticipated events as well as design basis and beyond design basis accidents. With its origin in the late 1960s, the SAS series of codes has been under continuous use and development for over forty-five years. SAS4A/SASSYS-1 currently includes a legacy thermal stratification model that divides the problem into a small number of distinct temperature regions separated by horizontal interfaces[12]. The model also includes the concept of different stages in the calculation, a plume height correlation, and another correlation for the interface rise due to the entrainment of a hot layer into a cooler plume. This model does not provide results consistent with experimental data and has been known to cause non-physical oscillations in the primary sodium flow rate.

First steps in developing improved system code models include re-examining the current model developed by Howard and Lorentz and incorporating the Peclet number within this model. Once this is achieved and a better understanding of the shortcomings of the current model is gained, other techniques will be explored. The system code will then be used to run simulations on the designed thermal stratification experiment. The system code will then be used to validate the thermal stratification experiment.

VI. CONCLUSIONS

Sodium fast reactors are currently under commercial development. As the new age of SFR's are ushered in it will be extremely important to have computational models to enhance the safety reactors prior to commercial deployment. A thermal stratification experimental facility is being constructed to obtain high fidelity temperature and velocity data. This data will be used to validate detailed 3D computational fluid dynamic models. The CFD codes will then be used to inform models that are used in systems code analysis for nuclear reactors. The combined effort of this team will yield significant advancements in systems code development, CFD analysis capabilities, as well as experimental techniques.

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NOMENCLATURE

Ri = Richardson number
Pe = Peclet number
Re = Reynolds number
Gr = Grashof number
L = Characteristic length
V_c = Velocity jetted from core
α = Thermal diffusivity
α_T = Entrainment coefficient
ν = Kinematic viscosity
g = Gravitational constant
ρ_c = Density of fluid in the core
ρ_p = Density of fluid in the pool
T = Temperature
H = Height
d = Diameter

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