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Numerical investigation of turbulent natural convection in the lower plenum of sodium cooled fast reactor under core relocation scenario

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HIGHLIGHTS

- Turbulent natural convection in lower plenum of SFR under core relocation scenario is numerically simulated.
- Heat transfer coefficients are estimated from core catcher under PAHR condition due to TIB and ULOFA scenarios.
- Correlations for *Nu* are developed for wide range of Bo^* and are useful for the design of SFR core catcher.
- Heat transfer coefficients during PAHR condition under ULOFA are lower than that under TIB.

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ABSTRACT

Turbulent natural convection inside the lower pool of pool-type sodium cooled fast reactor (SFR) is numerically studied for the modified Boussinesq number (Bo^*) range of 5×10^9 to 2×10^{11}. The enclosure considered is a geometrical model of the lower plenum of a typical pool type SFR main vessel with the in-vessel type core catcher assembly. Aim of this study is to analyze the decay heat removal rate from horizontally spread core debris on the core catcher during post accident heat removal (PAHR) condition under core meltdown scenario in SFR and correlate the same with the associated non-dimensional parameters. The mass, momentum and energy conservation equations have been numerically solved in cylindrical co-ordinates using finite volume method and using SIMPLE algorithm for pressure-velocity coupling. Turbulence has been modelled using k-ε model and the computational model is validated against benchmark numerical and experimental studies on natural convection found in literature. PAHR has been simulated under two scenarios of decay heat removal (DHR), viz., under pump driven DHR condition above the lower plenum (i.e., under Total Instantaneous coolant Blockage (TIB) scenario) and under completely passive DHR condition (i.e., under Unprotected Loss Of Flow Accident (ULOFA) scenario). Nusselt number (*Nu*) is correlated as a function of Bo^* for natural convection above the debris bed on the top surface of the core catcher and also below the bottom surface of the core catcher. These conservative correlations can be used in the design of in-vessel core catcher in pool-type SFR in order to ensure safe retention of radioactivity within the primary containment system even under whole core accident scenarios (WCA).

1. Introduction

Core meltdown accidents in nuclear reactors are regarded as highly improbable (probability < 10^{-6} per reactor year) due to the high reliability of shutdown systems and thus they are categorised as beyond design basis accidents (BDBA). Nevertheless they have received global attention since radioactive core debris formed as a result of core meltdown can settle at the bottom of the reactor main vessel and can affect its integrity in safe containment of radioactivity. One of the engineered passive safety features to mitigate this in pool type Sodium cooled Fast Reactors (SFR) is the installation of core catcher in the lower plenum of the main vessel for collecting the relocated and disintegrated core materials. Heat generated by the radioactive decay of these particles settled on the core catcher will be removed by the coolant (liquid sodium) in the lower plenum by means of natural convection heat transfer.

The primary containment system of a typical medium size pool type SFR contains liquid sodium in three partially isolated enclosures known as the hot pool (i.e., the top pool), side pool (i.e., cold pool) and the lower pool as shown in Fig. 1. During normal

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Nomenclature

Bo	Boussinesq number ($Gr.Pr^2$)
*Bo**	Modified Boussinesq number ($Gr^*.Pr^2$)
C_p	Isobaric specific heat of the fluid, J/kg K
C_μ, C_1, C_2, C_3	Constants of the turbulence model
g	Acceleration due to gravity, 9.81 m/s^2
G_k	Buoyant production of turbulent kinetic energy, kg/m s^3
Gr	Grashoff number ($g.\beta.\Delta T.L^3/\nu^2$)
Gr*	Modified Grashoff number ($g.\beta.q''.L^4/\nu^2K$)
h	Heat transfer coefficient, W/m^2 K
K	Thermal conductivity, W/m K
k	Turbulent kinetic energy, m^2/s^2
L	Characteristic length = (Outer radius- Inner radius)$_{cc}$, m
Nu	Nusselt number (h.L/K)
P_k	Shear production of turbulent kinetic energy, kg/m s^3
Pr	Prandtl number (ν/α)
q″	Heat flux (W/m^2)
R^2	Coefficient of determination
r	Radial coordinate, m
T	Temperature, K
t	Time, s
u	Radial velocity component, m/s
v	Axial velocity component, m/s
z	Axial coordinate, m

Greek symbols

β	Isobaric thermal expansion coefficient of fluid, K^{-1}
ε	Dissipation rate of turbulent kinetic energy, m^2/s^3
μ	Dynamic viscosity of the fluid, N s/m^2
μ_t	Turbulent viscosity, N s/m^2
ν	Kinematic viscosity of fluid, m^2/s
ρ	Density, kg/m^3
σ_k, σ_T, σ_ε	Turbulent Prandtl numbers of k, T and ε

Subscripts

eff	effective

operation of the reactor, the primary sodium pumps drive coolant flow from the side pool to the fuel subassemblies through the grid plate (GP). The coolant which comes out of the subassemblies reaches the top pool and then enters the Intermediate Heat Exchangers (IHX) where heat is transferred to the secondary sodium. The cooled primary sodium eventually returns to the side pool to complete the cycle. Any significant variation in coolant flow with respect to the operating power will normally result in safe shut down of the reactor by the shutdown systems. However, there are certain very low probability events which can potentially result in core meltdown and relocation scenarios in SFR as described below.

Scenario 1: Core relocation due to Total Instantaneous coolant Blockage (TIB): In case of TIB in one of the fuel subassemblies (SA), the coolant temperature around the fuel pins inside that SA increases due to localised loss of flow even though the coolant which leaves the core may not show appreciable temperature change so as to trigger the shut down signal. Due to this there is risk of fuel melting inside the fuel pins and it can result in pin failures. Multiple pin failures like this can lead to the meltdown of a maximum of seven fuel subassemblies by the time the reactor shuts down upon receiving a signal from DND (Delayed Neutron Detection system) (Moxon et al., 1986; Morcan et al., 1991). Under this scenario, the primary coolant pumps are available and they provide flow through the grid plate and reactor core for necessary decay heat removal. The molten corium formed due to TIB can leak to the lower plenum and come in contact with the coolant to get fragmented and quenched due to hydrodynamic and thermal

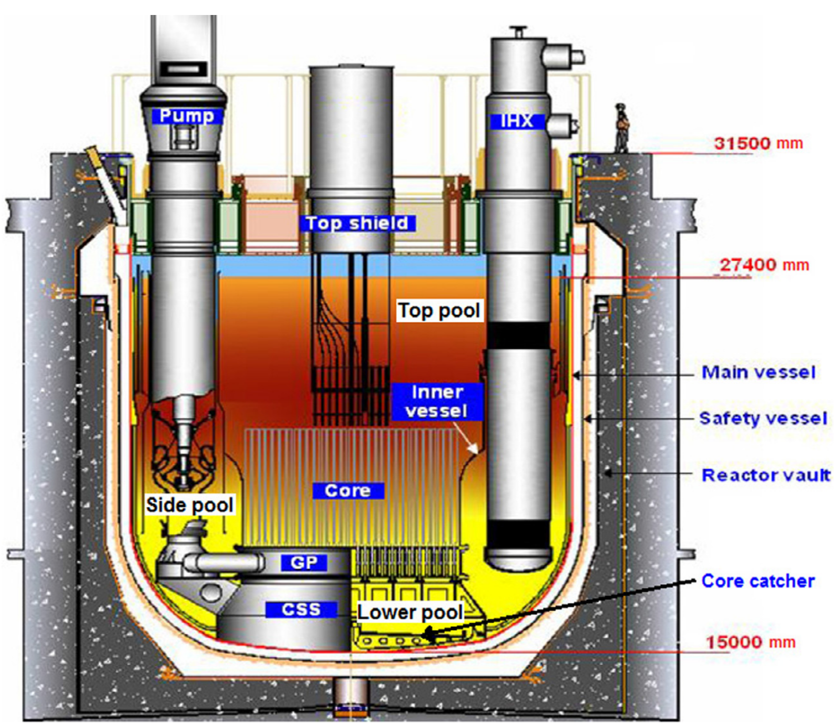

Fig. 1. Layout of the primary pool of pool-type SFR.