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Experimental and numerical study of phwr specific suspended debris

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ABSTRACT

In 220 MWe Indian Pressurized Heavy Water Reactor (IPHWR), under a postulated scenario of unmitigated Station Black Out (SBO) or a large break Loss of Coolant Accident (LOCA) along with the failure of all heat sinks, the fuel channels are likely to heat up due to un-availability of coolant. These events are called Beyond Design Basis Events (BDBEs) where designed engineered safety feature to be not available. The accident sequence leads to moderator boil-off in Calandria. The postulated boil-off leads to slow un-covering of the channels leading to fuel bundle heat up. The exposed channels in the steam environment are known as suspended debris. The present investigation aims to study the heat up behaviour of a single fuel channel of the Calandria for the 220 MWe PHWR. During moderator boil-off phase, it is postulated that decay heat of submerged fuel channels produce steam and acts as a coolant for the suspended channels forming debris like configuration. Experimental study and numerical simulation have been carried out for one-meter, fuel channel at 0.25%–1% of the decay heat. Numerically predicted temperature profile of the outer surface of CT, PT and fuel pins are validated with experimental results. Numerical results are obtained for total heat flux, radiation heat transfer and heat transfer coefficient of the outer surface of CT. Radiation is found to be the dominant mode of heat transfer and for 0.25–1.0% it contributes between 72 and 79% of total heat transfer. From the numerical study, it is concluded that the fuel bundles of the channel are heated maximum up to 804 °C if the decay heat 1% of the rated power. Radiation and convective steam cooling helps to limit the temperature rise of the bundle.

1. Introduction

IPHWR of 220 MWe consists of 306 fuel channels, each having twelve fuel bundles. Each fuel bundle consists of 19 fuel pins (Gera et al., 2010; Dutt et al., 2015). Each fuel pin consists of uranium dioxide pellets kept inside a clad tube made of Zircaloy-4. The fuel bundles are housed in a tube, called Pressure Tube (PT), which is the pressure retaining component. Further, the PT is placed concentrically in another bigger tube called Calandria Tube (CT). The whole assembly, called a fuel channel, is placed horizontally inside the Calandria (Mukhopadhyay et al., 2002; Yadav et al., 2013). The annulus gap between PT and CT is filled with CO_2 which serves as thermal insulation between high temperature coolant inside PT and low temperature moderator outside CT in the Calandria. Fig. 1 depicts the plant layout of the IPHWR. Coolant flows in closed circuit named Primary Heat Transport System (PHTS). Coolant passes through the gap of the PT and fuel bundle with the help of Primary Circulating Pumps of PHTS. The hot water exchanges heat in the steam generators. Steam generator produces steam to run the turbines (Bajaj and Gore, 2006). The PHWR possess various engineering safety measures to overcome any accidental scenario (Nitheanandan et al., 2017). However, accidents have very low probability to occur. The fuel channel under postulated accidental scenario has drawn attention to the community of scientist and engineers as PT integrity is to be maintained during abnormal operating conditions of the plant. In PHWR, accidents that damage the reactor core are categorized in two types, namely Limited Core Damage Accident (LCDA) and Severe Core Damage Accident (SCDA) (Majumdar et al., 2014). The postulated LCDA scenario may take place with the Loss of Coolant Accident (LOCA). Normally nuclear power plant is shut down immediately after getting accident signal. Even after nuclear fission chain is suspended, heat is continuously generated by the decay of the radioactive isotopes in the fuel channels. After shut down, the energy released inside the fuel bundles of reactor is referred to as decay heat. After nuclear power plant shut down, reactor channels continuously produces decay heat at a rate of 2% thermal output at 1 h, 1% at 5 h, 0.5% at one day and 0.1% at 10 days by the radioactive fission products (Bopche & Sridharan, 2010).

A feeder break or header break leads to a postulated LOCA scenario which results in a decrease in coolant flow rate. This leads to system depressurization. Under such circumstances, the Emergency Core

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Nomenclature

a	absorption coefficient (m^{-1})
$\mathbf{b}$	body force vector
C	specific heat (J/kg.K)
d	oxide layer thickness (m)
D_{st}	characteristic length for steam flow outside CT
D^{CT}	diameter of CT (m)
D^{PT}	diameter of PT (m)
e	total specific energy $[h + u^2/2 + gz]$ (J/kg)
E	experimental
Gr	Grashof number
h	heat transfer coefficient ($W/m^2.K$)
I	radiation intensity, which depends upon position ($\mathbf{r}$) and direction ($\mathbf{s}$) ($J/m^2.rad$)
k	thermal conductivity (W/m.K)
L	characteristic length for air flow in CT-PT annulus
n	refractive index
Q	volumetric heat generation (W/m^3)
p	pressure (Pa)
Pr	Prandtl number
Ra	Rayleigh number
Re	Reynolds number
Ri	Richardson number
$\mathbf{q}$	heat flux vector
$\mathbf{r}$	position vector
$\mathbf{s}$	direction vector
$\mathbf{s}'$	scattering direction vector
S	simulation
s	path length (m)
t	time (s)
T	local temperature (K)
Te	temperature (°C)
T^{amb}	ambient temperature of steam (°C)
T^{CT}	temperature of the CT (°C)
T^{PT}	temperature of the PT (°C)
$\mathbf{V}$	velocity vector
a	absorption coefficient (m^{-1})

Greek symbols

ρ	density (kg/m^3)
μ	dynamic viscosity ($N.s/m^2$)
υ	kinematic viscosity (m^2/s)
β	coefficient of volumetric expansion (1/K)
ε	hemispherical emissivity
σ	Stefan-Boltzmann constant ($5.67 \times 10^{-8}\ W/m^2.\ K^4$)
σs	scattering coefficient (m^{-1})
θ	angular position in degree
Φ	phase function
Ω′	solid angle (rad)
τ	stress tensor

Subscripts

i	inner surface
o	outer surface
st	steam
a	air

Cooling System (ECCS) provides water injection into the system to fill the PHTS, thus mitigates the consequences of LOCA.

By postulating further failure of ECCS injection, fuel channels temperatures are likely to rise and stabilize at an elevated temperature as moderator acts as a heat sink to the fuel channels (Yadav et al., 2014). At high temperature, a PT balloons or sags depending on its internal pressure. If the pressure is low, PT sags due to self-weight and weight of the fuel bundles and contacts the CT (Nandan et al., 2012; Nandan et al., 2010). At high internal pressure, PT balloons and contact CT circumferentially (Nandan et al., 2010). Either ballooning or sagging of PT establishes contact with CT, which increases heat transfer from fuel channel to the moderator (Gupta et al., 1996, Negi et al., 2017a,b). The moderator in the Calandria acts as a heat sink and prevents core degradation. By further postulating failure of the moderator cooling system, the moderator is expected to boil off continuously by absorbing decay heat from the heated fuel channels leading to gradual exposure of fuel channels from the top. A pressure builds up from moderator boil-off is expected in Calandria leading to rupture of the Over Pressure Rupture Disk (OPRD). A part of moderator expulsion is expected to follow by depressurization. This may lead to uncovering of few rows of fuel channels at the top of the Calandria. These channels are in steam environment. A channels in steam environment are termed as suspended debris. With time as moderator boils off, more layers get exposed to steam environment. In suspended debris, heat transfer to the steam is lower than the decay heat generated within the debris. As a result, exposed channels gets heated with the decay heat of the fuel bundles. Depending on internal pressure, PT sag or balloons and touches CT. PT imparts load to the CT. This leads to sagging of the fuel channel. Excessive sagging may lead to disassemble of a fuel channel. Channel sagging and disassembly depends on the CT temperature. Hence, it is important to study the temperature profile of the CT of suspended debris. Considering the time frame of the accident progression, the reactor will have a maximum of 1% decay heat at the time of such an accident.

The objective of this work is to study the temperature and heat transfer behaviour of a fuel channel in a steam environment corresponding to a given decay heat. In this work, a postulated SCDA scenario is considered and experiment is conducted for suspended debries with the assumptions that half of the Calandria is in steam environment.

At the initial stage, only top three rows of fuel channels get exposed and remaining fuel channels remain submerged in moderator. Steam velocity for top three rows of exposed fuel channels would be more in comparison to the case when the Calandria is half filled with moderator. This is due to higher steam generation rate as more channels are submerged and the interface area of moderator and steam is less than the half-filled Calandria. In our experimental facility, steam generation for the case of top three rows of exposed fuel channels was difficult due to limited power supply in the laboratory. However, in experiments, steam velocity is achieved in scaling ratio 1:1 for the scenario in which the Calandria is half submerged with moderator.

Fig. 2(a) depicts a non-scaled 3D view of exposed fuel channels and submerged fuel channels inside the Calandria. At 1% decay heat, the fuel channel, having maximum power among the top three rows, generates decay heat of 4.03 kW/m. This is less than the core average of 4.16 kW/m at 1% decay heat. Steam generated by the decay heat of submerged fuel channels in the moderator would rise upward and interact with the exposed fuel channels inside the Calandria as shown in Fig. 2(b). Experiment has been conducted at heating rate of 4.16 kW/m which corresponds to 1% decay heat. Conducting experiment at higher heating power and lower steam velocity than that would occur when the top three channels are exposed, the resulting temperature of the simulated channel will be higher than the actual case. If this temperature is within the limit, then the channels in the top three rows will be well within the limit.

Experimental study and numerical simulations have been carried