Thermal Hydraulic Analysis of Small Modular Reactors (SMRs)

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1-INTRODUCTION

Small Modular Reactors (SMRs) represent a significant advancement in nuclear reactor technology, currently undergoing development across multiple nations. These reactors offer a versatile solution for power generation, with capacities ranging from tens to hundreds of megawatts, suitable for various applications, including remote, residential, and military areas where electricity is required. One of the most notable features of SMRs is their compact size, distinguishing them from conventional nuclear power plants (NPPs). Their small footprint allows for factory fabrication and transportability, enabling easier deployment and reducing construction costs. This flexibility in deployment makes SMRs an attractive option for regions with limited infrastructure or unique energy demands.

This study conducted the thermal hydraulic analysis for the SMR design, using NuScale's (NuScale Power, LLC, United States of America) reference design as a guide. However, certain parameters that were required for completing the design project were found to be not present; therefore, suitable judgments were made from thermal hydraulic literature to complete the design analysis. Table 1(a) provides details about the technical parameters involved, including the dimensions and operating conditions of the SMR. We specifically designed the SMR to operate optimally under full-power conditions, utilizing natural circulation as the primary method of core coolant flow, thereby eliminating the need for reactor coolant pumps. The small modular reactor (SMR)'s reactor pressure vessel (RPV) contains a helical coil heat exchanger and the reactor's core. The cylindrical containment vessel (CNV) houses the RPV. The Shroud, a pool of water, surrounds the CNV. This pool of water is useful for maintaining and regulating high temperatures caused by nuclear reactions inside the reactor core of the pressure vessel. The heat generated inside the core is carried by coolant, which is present in the primary loop of natural circulation. The hot fluid that continues to flow to HX aids in producing steam at elevated temperatures. We establish a dedicated turbine system to convert the hot steam into power generation.

Parameters	***Values***
Pressure in primary loop(bar)	138
Core Inlet (C)	265
Outlet Coolant Temperature (C)	321
RPV height(m)	17.7
Diameter of RPV (m)	2.7
Reactor Type	Integral PWR
Coolant Type	Light Water
Electrical Power (MW)	60
Fuel Type	UO2 pellet
Fuel Assembly Array	HEX
Number of Fuel Rods	331

Table 1(a): Technical parameters of SMR.

2-Single Phase Natural Circulation inside RPV

2.1-Reactor Core

The pressure vessel's core has a hexagonal geometry with a lateral spacing of 240 mm, as shown in figure 2.1(a). Each pin's diameter is 9.4 mm, and the core's height is 2.5 m. The fuel pellet is covered by 0.75 mm-thick Zircaloy cladding. In the fuel assembly, there are a total of 331 fuel pins. The reactor is designed to operate for 60 megawatts. The hydraulic diameter for the core should therefore be calculated before starting other numeric calculations and applying fundamental physics laws. The associated fluid volume for the core is therefore found to be 0.06728 by using the geometry shown in figure 2.1(a), and similarly, the wetted surface area can be calculated for the same to be 26.515. Using equation 2.1(a), we can now calculate the hydraulic diameter for the core. The hydraulic diameter comes out to be 10.15 mm.

$$Hydaulic\ Diameter = 4 * \frac{Fluid\ Volume}{Wetted\ Surface\ Area} \tag{2.1(a)}$$

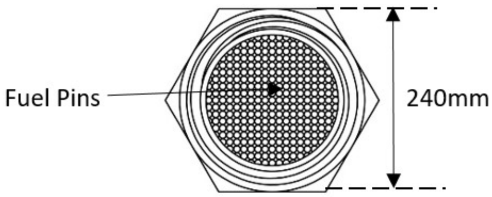

Fig 2.1(a): Fuel Assembly containing 331 fuel pins

2.2-Heat Exchanger

The heat exchanger consist of an inner shroud and an outer shell, as shown in figure 2.2. The HX used here is helical-coiled, with six rings concentric to one another. The number of tubes placed in each ring varies by two. There are a total of 6 rings. Ring 1 has 8 tubes, Ring 2 has 10 tubes, Ring 3 has 12 tubes, Ring 4 has 14 tubes, Ring 5 has 16 tubes, and Ring 6 has 18 tubes. Each tube has an outer diameter of 10 mm and a wall thickness of 1.2 mm. The tube's average length is 20 meters. The coil's length is 4.56 meters. The inner shell of HX has an outer diameter of 151 mm, and the outer shell's inner diameter is 425 mm. The 6 different concentric rings can be given geometrical pitch circle diameter based on longitudinal pitch by diameter ratio of 2.29.Therefore, the lateral pitch is 0.0229 m, and Δ PCD becomes 0.0458 m, which is twice the lateral pitch dimension. Therefore, for the purpose of analysis, we can draw pitch circles in place of 6 rings, with PCD 1 to be 173.9mm, PCD 2 to be 219.7mm, PCD 3 to be 265.5mm, PCD 4 to be 311.3mm, PCD 5 to be 357.1mm, and PCD 6 to be 402.9mm.

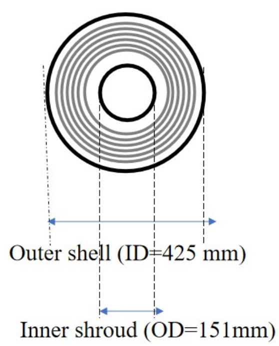

Fig 2.2: HX geometry showing helical coils.

The fluid volume for the HX can thus be calculated to be 0.4424, and the wetted surface area can be calculated based on proper consideration of the wetting area of contact due to the flow of coolant, which comes out to be 55.069. Therefore, we can use the flow area and wetted perimeter to determine the hydraulic diameter of the HX. By using equation 2.1 again, we get the hydraulic diameter for the HX configuration to be 32.13 mm, which is roughly 3 times the hydraulic diameter of the core.

2.3-Natural Circulation Calculation using Integral Momentum Balance using Light Water as Coolant

At 265 °C, the coolant enters the reactor core. Equation 2.3(a) allows us to calculate the temperature difference between the core inlet and outlet based on the reactor's full power of 60 MW.

$$\dot{Q} = \mathrm{m}C_P\Delta\mathrm{T} \qquad (2.3(a))$$

Let the distance between the thermal centers (h) be the center-to-center difference in height between the core and HX. The thermal center height is an important parameter for designing the SMR for a given coolant mass flow rate and temperature difference between the core inlet and outlet. Due to the flow of coolant inside the primary loop of natural circulation, there is some frictional pressure drop in the reactor core and the HX as well. There may be a loop pressure drop, but the current analysis only considers losses inside the reactor core and HX. We can calculate the frictional pressure losses using equation 2.3(b).

$$\Delta\mathrm{P} = \frac{fL}{D_H}\ \frac{\rho u^2}{2} \qquad (2.3(b))$$

In calculating the frictional pressure losses in the core, the thermodynamic properties are determined at the core inlet temperature; conversely, for the frictional losses in the heat exchanger, the thermodynamic parameters are assessed at the core outlet temperature. Given that the Reynolds number within the core and the heat exchanger exceeds 2X, equation 2.3(c) is applicable for determining the friction factor for both the core and the heat exchanger.

$$f = 0.184Re^{-0.2} \qquad (2.3(c))$$

The frictional pressure drop calculation can be calculated based on certain coolant mass flow rate ($\dot{m}$). Therefore Integral Momentum Balance is used by assuming zero minor pressure losses and major losses inside the reactor core and the HX only. The equation 2.3(d) is the governing equation for Integral Momentum Balance.

$$\frac{d\dot{m}}{dt}\sum_{all\ links}\frac{L_i}{A_i} + \frac{4\dot{m}^2}{2\rho}\sum_{all\ links}\frac{1}{A^2{}_i}\left(\frac{f_i\,L_i}{d_{h-i}} + \frac{K_i}{4}\right) - \oint_{loop}\rho_R\, g\beta T dH = 0 \qquad (2.3(d))$$

Assuming Steady State in-compressible fluid flow through the loop and by neglecting minor pressure losses term the equation 2.3(d) can be simplified into equation 2.3(e) where ΔP refers the major frictional losses in the loop.

$$\Delta P_{Core} + \Delta P_{HX} = \rho\mathrm{gh} \qquad (2.3(e))$$

The parametric analysis can be done based on 'h' and '$\dot{m}$'.Assuming certain 'h' and '$\dot{m}$' and then doing the iterative calculation using Goal Seek command of EXCEL gives ΔT and $\dot{m}$ corresponding to given 'h'. Therefore one can change the thermal center height i.e. 'h' and arrive at certain configuration of RPV. Since the height of RPV is 17.7 m therefore the inside the RPV the NCL height can be allowed to those values for which it does not exceeds the RPV height. Also one must keep certain spacing from the wall of NCL. The result for the parametric analysis is shown in table 2.3(a).The mass flow rate of coolant is increasing with increase in the distance between the thermal center (h).Corresponding to h= 12 m the ΔT = 71.35 i.e. the outlet core temperature is $T_{Outlet} = T_{inlet} + \Delta T$ that gives 336.35 C which is closer to the temperature mentioned for core outlet in table 1(a).

h(m)	*ΔT*	*ṁ(Kg/sec)*
2	187.91	65.15
4	128.76	95.07
6	103.34	118.46
8	88.503	138.32
10	78.55	155.84
12	71.35	171.57

Table 2.3(a): Distance between the thermal centers is parametric with assumed mass flow rate and corresponding ΔT is calculated for light water as coolant.

3-Steady State Decay Heat Removal System

Emergency core cooling systems (ECCS) consist of reactor vessel valves (RVVs). To ensure the overall safety of the reactor during loss of coolant accidents (LOCAs) inside containment, the reactor removes the decay heat. This guarantees the covering of the core and the removal of decay heat. The steam released from the RPV increases the containment vessel pressure, and the RPV depressurizes. We assume that the decay heat within the CNV at steady state represents 2% of the full power of the SMR. The ECCS removes the decay heat with steam.

The CNV undergoes condensation and convective heat transfer. The steam released from the pressure vessel is at saturation temperature, which corresponds to CNV pressure. Within the containment, the saturated vapors from the pressure vessel are condensing and forming a specific boundary layer, allowing the Nu Condensation equation to calculate the heat transfer coefficient due to the laminar flow near the wall. We also presume the absence of non-condensable elements in our design analysis.Assuming that there are negligible shear stress at liquid-vapor interface following Nu Equation can be used which is written in equation 3(a).

$$\overline{Nu_L} = \frac{4}{3} Nu_L = 0.943\left(\frac{\rho_l(\rho_l-\rho_v)h'_{fg}gL^3}{k_l\,\mu_l(T_{sat}-T_W)}\right)^{1/4} \quad (3(a))$$

There is a pool of water on the outer side of the CNV that heats up and boils; it could even evaporate completely, leaving the shroud empty. The present analysis is based on an open NCL outside the CNV. This occurs when all the water in the pool boils off. The air dampers located at the bottom of the shroud will open, allowing air to circulate. As a result, the Natural Circulation Loop will be open. Equation 3(b), which expresses the Churchill-Chu correlation quantitatively, will establish a turbulent natural convection heat transfer.

$$\overline{Nu_L} = \left[0.825 + \frac{0.387\,Ra_L^{1/6}}{(1+\left(\frac{0.492}{Pr}\right)^{\frac{9}{16}})^{8/27}}\right]^2 \quad (3(b))$$

The heat energy balance for the CNV is written in equation 3(c). The heat transfer coefficient can be calculated based on equation 3(a) and 3(b) for inside and outside CNV.

$$h_{inside}(T_{sat} - T_{wall}) = h_{outside}(T_{wall} - T_{shroud}) \quad (3(c))$$

We perform the parametric analysis by taking into account various CNV pressures. We iteratively calculate the wall temperature using Excel's Goal Seek function, and validate it using a MATLAB code that applies the Secant Rule, obtaining the wall temperature using equation 3(c).

The wall temperature of the containment depends on the CNV pressure. As the CNV pressure increases, so does the wall temperature. Figure 3(a) displays the parametric data obtained for a specific height of CNV where heat transfer occurs (H = 30 m).

The heat flux corresponding to CNV pressure at the same situation discussed in figure 3(a) has been analyzed and the result obtained has been plotted in figure 3(b).

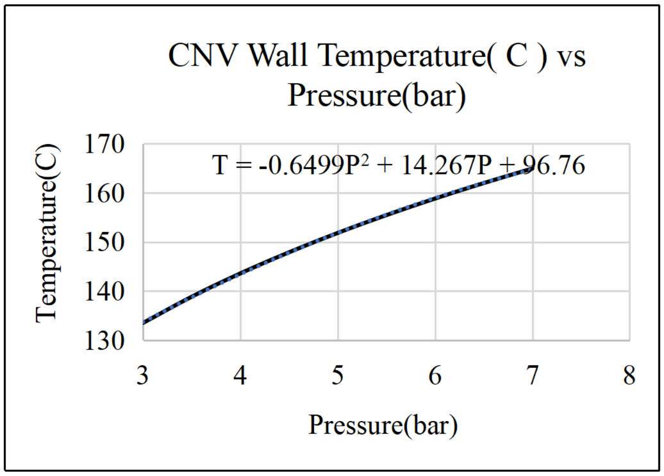

Figure 3(a): CNV pressure variation for decay heat 2% of full power for H= 30m.

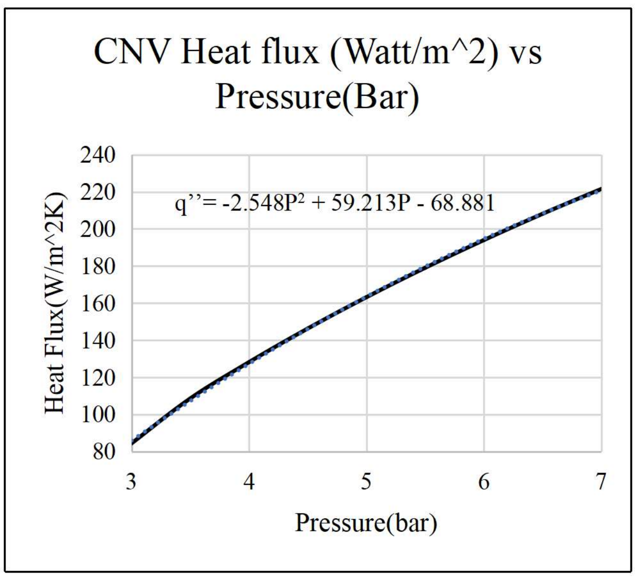

Figure 3(b): Heat Flux through CNV wall vs Steam issued pressure due to choking of valve of RPV.

4-Transient analysis of Decay Heat Removal System during Depressurization inside RPV.

Here as the choking is initiated from Reactor Pressure Vessel (RPV), two-phase conditions will be encountered. The model considered here for analysis is *Homogeneous Fluid Model* (HFM) for two phase depressurization therefore the fluid is treated as homogeneous. The discussion has considered the homogeneous model, which is valid when the depressurization from RPV is rapid. We have developed a model based on mass and energy balances that can predict the variation of pressure and the quality of steam in the containment with time. The rate of change of pressure is the function of specific volumes & specific enthalpies.

Modelling of Depressurization:

1-Mass Balance

$$\left(\frac{dm}{dt}\right)_{system} = \left(\frac{dm}{dt}\right)_{CV} + \dot{m}_e - \dot{m}_i$$

$$\left(\frac{dm}{dt}\right)_{system} = 0 \;\; and \; \dot{m}_e = 0$$

$$\left(\frac{dm}{dt}\right) = \dot{m}$$

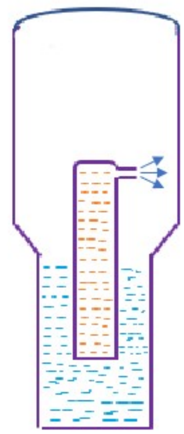

Figure 4(a): Steam breaching out of RPV

$$\frac{Vd\rho}{dt} = \dot{m} \qquad (4(a))$$

2-Energy Balance

$$\dot{Q} - \dot{W} = \frac{dE}{dt}_{CV} + \dot{m}_e\left(h_e + \frac{u_e^2}{2} + gz_e\right) - \dot{m}_i\left(h_i + \frac{u_i^2}{2} + gz_i\right)$$

$$\dot{W} = 0 \; and \; \dot{m}_e = 0$$

$$\dot{Q} = \frac{d}{dt}(\bar{\rho}Ve) - \dot{m}\bar{h}$$

$$\dot{Q} + \dot{m}\bar{h} = V\frac{d}{dt}\left(\bar{\rho}\left(ie + \frac{u^2}{2} + gz\right)\right)$$

$$\dot{Q} + \dot{m}\bar{h} = V\frac{d}{dt}(\bar{\rho}\bar{h} - p) \qquad (4(b))$$

$$\dot{Q} + \dot{m}\bar{h} = V\bar{\rho}\frac{d\bar{h}}{dt} + V\bar{h}\frac{d\bar{\rho}}{dt} - V\frac{dp}{dt}$$

$$\frac{d\bar{h}}{dt} = \frac{\dot{Q}}{\bar{\rho}V} + \frac{1}{\bar{\rho}}\frac{dp}{dt} \tag{4(c)}$$

$$V\frac{d}{dt}\left(\frac{1}{\bar{v}}\right) = m^{\cdot} = \frac{1}{\bar{v}^2}\frac{d\bar{v}}{dt} = \frac{\dot{m}}{V} \qquad \text{(Using 4(a))}$$

$$\frac{d}{dt}\left(v_f + xv_{fg}\right) = \dot{m}\frac{\bar{v}^2}{V}$$

$$\frac{dp}{dt}\left(\frac{dv_f}{dp} + x\frac{dv_{fg}}{dp}\right) + v_{fg}\frac{dx}{dt} = \frac{\dot{m}\bar{v}^2}{V} \tag{4(d)}$$

$$\frac{dp}{dt}\left(\frac{dh_f}{dp} + x\frac{dh_{fg}}{dp} - \bar{v}\right) + \frac{dx}{dt}h_{fg} = \frac{\dot{Q}\bar{v}}{V} \tag{4(e)}$$

5-Conclusions

The utilization of a Nu Scale reference design facilitates the performance of a thermal hydraulic study on Small Modular Reactors (SMRs). The analysis computes hydraulic diameters, temperature differentials, and coolant flow rates for light water. The study demonstrates that the cooling of the SMR core may be maintained through natural circulation, eliminating the need for reactor coolant pumps. Parametric calculations demonstrate that a thermal centre height, coolant mass flow rate and core outlet temperature. The inclusion of helical coils and specific geometrical elements in the design of the heat exchanger enables the achievement of optimal heat transfer and effective steam generation, resulting in increased power output. The study also investigates the transient and steady-state characteristics of the decay heat removal system, with a focus on the mechanisms of heat transfer and emergency core cooling systems. Small modular reactors (SMRs) show great promise for nuclear power applications due to their inherent safety features and operational flexibility. This comprehensive analysis lays the foundation for the optimization and implementation of SMR technology.