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SYMULATION OF TRANSIENT HEAT DIFFUSION OF NUCLEAR REACTOR SIMILAR TO NUSCALE SMALL MODULAR REACTOR DESIGN WITH A DECAY POWER SOURCE TERM

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ABSTRACT

The Hankel transform of the second kind was used to consider the diffusive heat transfer of a design similar to the Nuscale reactor. The diffusive heat transfer allows for about 17 hours of protection from the decay heat of the reactor before reaching the critical temperature of water, the point where higher level of reactor damage may occur. The model does not consider other important modes of heat transport such as convection due to natural circulation or radiative heat transport. Simplified thermal properties of the reactor were used, along with real dimensions and design components of the Nuscale reactor.

NOMENCLATURE

$\hat{C}_p$ – Isobaric heat capacity of reactor
ρ – Density of reactor
k – Thermal conductivity
a – Thermal diffusivity
u – Temperature of reactor
r – Radial dimension
t – Time dimension
S' – Power per volume of decay heat

INTRODUCTION

It is the purpose of this paper to determine the time scale of the loss of power transient behavior of the Nuscale SMR design. To determine the transient behavior of the cylindrical reactor vessel, the Finite Hankel Transform of the second kind was applied to resolve the partial differential equation. This was done to estimate the required intervention time to prevent reactor core damage.

The nuclear power community proposed improvements to the second generation nuclear powerplants that advance the safety and economic feasibility of nuclear power. Commercial nuclear power companies, in tandem with research groups, innovated several feasible reactor designs that capture these advances called the Small Modular Reactor (SMR). Nuscale, a leading company in the SMR design, is currently working with the Nuclear Regulatory Commission (NRC) to license their reactor design. The Nuscale SMR has a few design parameters significantly different from the standard second-generation design that change the reactor coolant system.

The reactor design must include, not only steady-state operation of the reactor, but a transient, accident scenario, analysis to pass the licensing process. One of the significant changes to the Nuscale design from second-generation reactors is a complete reliance of natural circulation phenomena to pump the reactor coolant. They argue, the reactor design is safer, because there are no pump failure transient mechanisms.

Natural circulation of fluid works based off the principle of density differential due to temperature gradients in a gravitational field. The hot fluid is less dense than the respectively cooler fluid, and will experience an upward buoyancy force opposite the pull of gravity. The reactor core's heat generation from fission supplies the temperature gradient (standard Light Water Reactor (LWR) nuclear fuel), and a steam turbine for power generation provides the heat sink and the return cold leg of coolant.

In the event of a turbine failure or cold leg blockage, the circulation of the coolant will stop either because the temperature differential will shrink in the first case, or because of a physical barrier to flow in the second case. In the case of these accidents, the fission reaction shut down and the surrounding water pool will become the primary coolant for the reactor as seen in Figure 1. Although the fission reaction shuts down, there is still considerable heat generation through fission decay products which follow the following relations [1]:

$$\frac{\boldsymbol{P(t_s)}}{\boldsymbol{P_0}} = \boldsymbol{-6.14575 \times 10^{-3} \times \ln(t_s) + 0.060157}$$
$$for\ 1.5 \le t_s \le 400\ s$$

$$\frac{P(t_s)}{P_0} = 1.40680 \times 10^{-1} \times t_s^{-0.286}\ for\ 400 < \boldsymbol{t_s} \le 4 \times 10^5\ \boldsymbol{s}$$

$$\frac{P(t_s)}{P_0} = 8.70300 \times 10^{-1} \times t_s^{-0.4255}\ for\ 4 \times 10^5 < \boldsymbol{t_s} \le 4 \times 10^6\ \boldsymbol{s}$$

$$\frac{P(t_s)}{P_0} = 1.28420 \times 10^{1} \times t_s^{-0.6014}\ for\ 4 \times 10^6 < \boldsymbol{t_s} \le 4 \times 10^7\ \boldsymbol{s}$$

(1)

where t_s is time after shut down, P_0 is the steady-state thermal power generation of the reactor, $P(t_s)$ is the decay power produced.

The decay power generation will raise the temperature of the reactor potentially causing damage to the reactor core. The water pool containment is a heat sink in this accident scenario, a barrier to attenuate the response time needed to prevent structural damage to the reactor. A key point in time in this scenario is when the primary reactor coolant begins to boil. If this occurs, the pressure in the reactor core increases sharply and the thermal properties of the coolant change to be less favorable and the probability of reactor damage increases sharply. So it is important to know exactly how much time the reactor will remain below the critical temperature after a loss of power accident.

This situation can be modeled by an infinitely long cylinder surrounded by a water pool of constant temperature as represented in the following PDE:

$$\rho \hat{C}_p \frac{\partial u}{\partial t} = k \times \frac{1}{r}\frac{\partial}{\partial r}\left(r\frac{\partial u}{\partial r}\right) + S'(t)\ for\ r_1 \le r \le r_2\ and\ t \ge 0$$

$$u(r,t)|_{t=0} = f_0(r) = 583\ K$$

$$\left.\frac{\partial u(r,t)}{\partial r}\right|_{r=r1} = f_1(t) = 0\ K/m$$

$$u(r_2,t) = f_2(t) = 373.15\ K$$

(2)

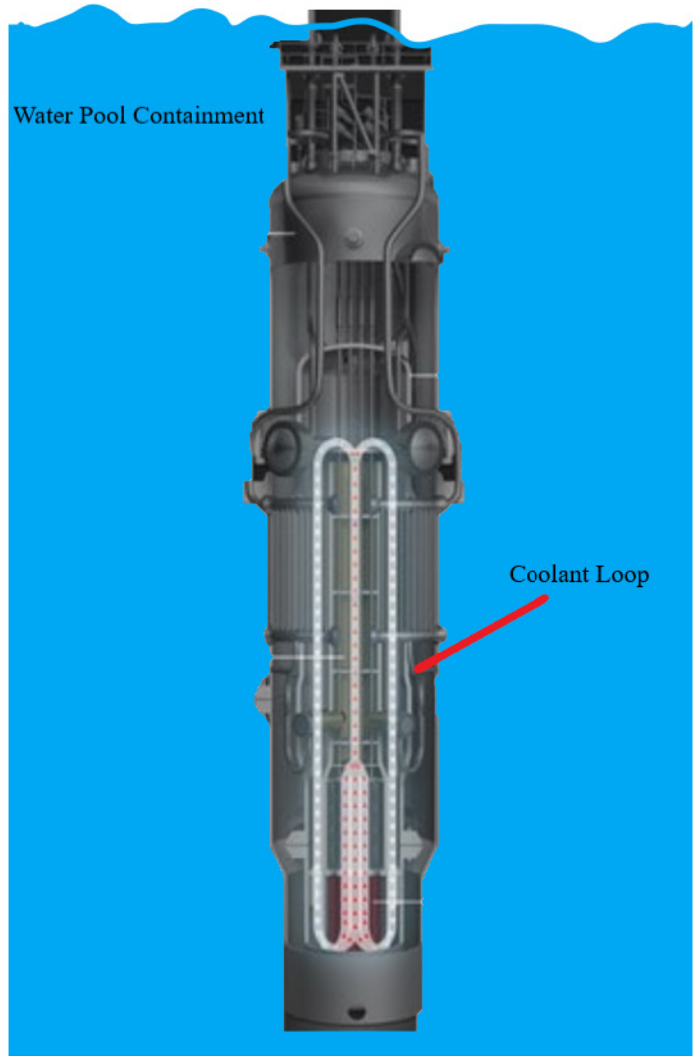

Figure 1: Reactor vessel is contained in a water pool. The water pool is assumed to be at atmospheric pressure and 100 °C in loss of power accident. Picture adapted from [2].

Equation (2) assumes that the cylinder is infinitely long so that the temperature profile is only a function of the radius, and the main source of heat transport is diffusion. The boundary conditions assume that the reactor vessel is at a uniform temperature initially and the water pool is kept at the boiling temperature of water at atmospheric pressure because the pool temperature will not exceed this level. The source term is simplified by assuming the entire body of the reactor even generates the decay heat, while, in reality, most of the heat will be generated at the bottom of the reactor where the core is located.

The highest temperature in the core will be located at the center of the cylinder. The solution to the equation (2) will us to track the peak temperature as a function of time. The method used to solve equation as stated previously is the Hankel Transform type 2. This renders the following solution:

$$u(r,t) = \sum_{n=1}^{\infty} \bar{u}_n(t) \frac{X_n(r)}{\|X_n(r)\|^2}$$

$$\bar{u}_n(t) = \exp(-a\,\lambda_n^2\,t)\left[\bar{u}_{0n} + \int_0^t \exp(-a\,\lambda_n^2\,\tau)\,\bar{Q}(\tau)d\tau\right]$$

$$\bar{u}_{0n} = \int_{r_1}^{r_2} u_0(r)\,X_n(r)\,r\,dr$$

$$\bar{Q}(t) = \bar{S}(t) - a\,r_2\,X'_n(r_2)f_2(t)$$

$$\bar{S}(t) = \int_{r_1}^{r_2} S(t)\,X_n(r)\,r\,dr$$

$$S(t) = \frac{S'(t)}{\rho\hat{C}_p}$$

$$X_n(r) = J_0(\lambda_n r)$$

$$X'_n(r) = -\lambda_n J_1(\lambda_n r)$$

$$\lambda_n\ is\ defined\ by\ J_0(\lambda_n r_2) = 0\ \ where\ n = 1, 2, \ldots \tag{3}$$

The temperature at the center of the core at the initial condition is subject to instability in the model. The center temperature will only be slightly different from the temperature 0.1 meters from the center of the reactor. This position is, therefore, used instead of the core temperature.

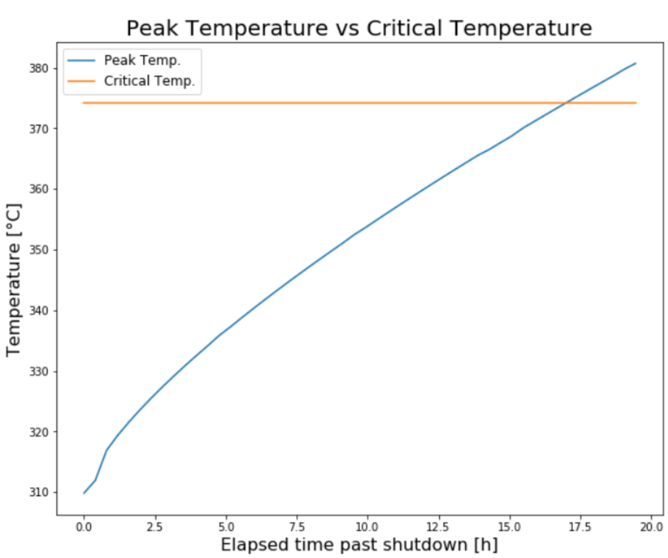

Figure 2: Temperature of the reactor 0.1 m from the center

Further, the physical properties of the reactor were assumed to be close to that of water, and constant, as the temperature range under consideration is not wide. The coolant will be a significant potion of the reactor vessel and the main component in contact with the reactor core, so these assumption should be fair.

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The temperature rise in the reactor to the critical temperature occurs in a matter of hours after shutdown under the imposed conditions.

CONCLUSIONS

In traditional LWRs the reactor is equipped with cold water spray nozzles powered by pumps. Because the Nuscale model explicitly avoided the use of pumps and back up systems, the cooling strategy needs to be significantly different. The water pool cooling model presented in this work however demonstrates that this cooling method is only sufficient for the first several hours after a loss of power accident.

There are several simplifications, however, that may extend the time to reach the critical temperature not considered in this model. Firstly, there are other modes of heat transport not considered here, the most significant being convective heat transport by the natural circulation at the boundary. In the model it was considered to be held at constant temperature, when it could have been represented as a convective heat transport on the outside surface of the cylinder wall increasing the heat flow from the reactor to the water pool. In addition, the minor heat transport mode of radiation would also increase heat rejection from the reactor.

In addition to simplifications, there are several assumptions I needed to make about the design of the reactor and the physical properties that may not be an accurate representation of the real design, but are still the fair assumptions under the limited knowledge of the system available on the website. Assumption such as the physical properties: density, heat conductivity, heat capacity, water pool temperature operating conditions, etc.

Deriving conclusion from the diffusion only model, as presented in this work, suggests that the relying solely on diffusive heat transfer would on allow for about 17 hours of protection before outside interference to avoid the temperature in the core to reach a critical level.

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