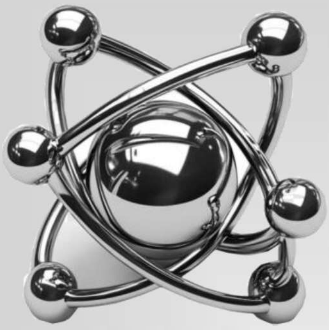

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FUNDAMENTALS OF THERMAL-HYDRAULIC CALCULATIONS OF HEAT EXCHANGE APPARATUSES OF NUCLEAR REACTORS

In this article, the fundamentals of thermal-hydraulic calculations of heat exchange apparatuses of nuclear reactors are going to be analyzed. Thermal and hydraulic calculations of heat exchangers are divided into constructive and calibration calculations. The task of constructive (design) calculations is to determine the required geometric dimensions of the surface of the heat exchanger.

The following parameters are set:

- Thermal power №, kW;
- Mass flow rates of coolants G, kg/s;
- Specific enthalpies (temperatures) of coolants at input and output, in kJ/kg (T, K);
- Coolant pressure P at the inlet and output of a heat exchanger, in Pascals, or the limit values of hydraulic resistance;
- Dependencies for thermophysical parameters of heat transfer: density ρ, enthalpy J, dynamic viscosity Pa·s, thermal conductivity W/m·K, specific heat capacity J/kg·K, saturation temperature T, heat of evaporation kJ/kg, and thermal conductivity (W/m·K) of the wall – specified in tubular form or in the form of polynomials;
- Ambient temperature and permissible temperature on the outer surface of the heat exchange body;
- Maximum overall dimensions and weight of the equipment.

The final goal of the calculation, taking into account all the requirements, is to select and calculate the optimal design of the heat exchanger. This calculation is carried out for the nominal operating mode of the unit. The tasks of the verification calculation include:

Detailed calculation of the heat transfer unit in the nominal mode for the selected geometric relationships of the heat transfer surface;

Calculation of the heat transfer unit with the selected (known) geometry of the heat transfer surface in partial modes in modes in order to determine the enthalpies (temperatures) of the coolants and thermal power;

Calculation of the dynamics of a specific heating element in unsteady modes to determine the parameters of coolants and the wall under conditions of transient processes, emergency modes, and hydraulic instability.

Analysis of the fundamental thermal-hydraulic principles and equations combines Thermodynamics, Heat Transfer, and Fluid Mechanics.

A. Conservation Laws (The Foundation). These are typically applied to a control volume of a heat exchange equipment.

1. Conservation of Mass (Continuity Equation):
   - What it does: Tracks the flow of coolant into and out of a region.
   - Simple Form (Steady State): $\dot{m}_{in}$ = $\dot{m}_{out}$ (mass flow rate in equals mass flow rate out).
2. Conservation of Energy (First Law of Thermodynamics):
   - What it does: Balances the heat added to the coolant with its change in enthalpy (internal energy + flow work).
   - Form: Q = ṁ*(h_{out}= h_{in})
     - Q = Heat transfer rate (W)
     - ṁ = Mass flow rate (kg/s)
     - h = Specific enthalpy (J/kg)
3. Conservation of Momentum:
   - What it does: Relates pressure drop to flow velocity, friction, and changes in elevation. It's crucial for determining pumping power.
   - Simplified Steady-State Form (for a pipe): ΔP_{total}= $\Delta P_{friction}$ + $\Delta P_{elevation}$ + $\Delta P_{acceleration}$

B. Modes of Heat Transfer

1. Conduction: Governs heat transfer *through* the fuel pellet and cladding.
   - Fourier's Law: q" = -k*dT/dx
     - q" = Heat flux (W/m²)
     - k = Thermal conductivity (W/m·K)
     - dT/dx = Temperature gradient
2. Convection: Governs heat transfer from the cladding surface to the coolant. This is the most critical mode.
   - Newton's Law of Cooling: q" = h*($T_{surface}$- T_{bulk})
     - h = Convective heat transfer coefficient (W/m²·K) - This is the key parameter to calculate.
3. Radiation: Generally negligible in a reactor core due to high coolant flow rates, but can be significant in severe accident scenarios.

C. Critical Thermal-Hydraulic Phenomena

1. Boiling Heat Transfer:
   - Subcooled Boiling: Boiling occurs at the hot cladding surface even though the bulk coolant is below saturation temperature. This is normal in BWRs and PWRs.
   - Nucleate Boiling: Efficient heat transfer with bubbles forming. Desired regime.
   - Critical Heat Flux (CHF) / Departure from Nucleate Boiling (DNB): A crisis point where bubbles form so rapidly they blanket the surface, insulating it and causing a rapid, dangerous temperature spike. Avoiding DNB is a primary design constraint for PWRs.
2. Two-Phase Flow:
   - Void Fraction (α): The fraction of a channel's volume that is occupied by vapor. It drastically affects flow density, pressure drop, and neutron moderation.

- Flow Regimes: The pattern of liquid/vapor mixture (bubbly, slug, churn, annular, mist). Each regime has different heat transfer characteristics.
  - BWR Analysis: Centered entirely around modeling two-phase flow and void fraction.

3. Pressure Drop Components:
   - Frictional: Due to shear stress between the fluid and the channel walls. Calculated using correlations like Martinelli-Nelson (for two-phase) or Darcy-Weisbach (for single-phase).
   - Gravitational: Due to changes in elevation ($\rho * g * \Delta h$).
   - Accelerational: Due to changes in flow velocity (significant in boiling channels where fluid expands).

In conclusion, the fundamentals of thermal-hydraulic calculations of heat exchange apparatuses of nuclear reactors involves analytical workflow. The definition of the geometry of the apparatus has to always be done corresponding to the nominal intended operation of an equipment, including fuel rod diameter, pitch, and channel length.

Definition of operating conditions which include pressure, temperature, mass flow rate, and power. Applying the conservation laws to find bulk fluid temperatures and pressures. Calculating the heat transfer coefficients by using appropriate correlations for the flow regime (single-phase, subcooled boiling, nucleate boiling). Checking critical limits by analyzing the cladding and fuel temperatures data within limits. Calculating the system parameters by determining total pressure drop and the required pumping power.

This process ensures that nuclear heat exchange apparatuses can reliably and safely perform their function under all envisioned operating conditions.

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