Ph. D. Research Seminar II

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**Thermal Hydraulic Safety Criteria and Limits**

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The main objective of the thermohydraulic design of the reactor is to ensure that the reactor core meet the requirements for steady state and transient operation without breaking the design basis.

To perform thermal hydraulic analyses of the nuclear reactor core it is important to keep in mind the thermal safety parameters. According to the international regulations for fluid systems the safety margins to design a nuclear reactor must be:

Cooling water.

A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink, shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Reactor coolant makeup.

A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided.

These design criteria could be not satisfying completely during emergency operations conditions. When this abnormal situation happened, other criteria play an important role.

The acceptance criteria for emergency core cooling to avoid cladding damage are:

* Peak cladding temperature.

The calculated maximum fuel element cladding temperature shall not exceed 2200° F.

* *Maximum cladding oxidation*.

The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

* Maximum hydrogen generation.

The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

* *Coolable geometry*.

Calculated changes in core geometry shall be such that the core remains amenable to cooling.

* Long-term cooling.

After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Thermal limits are provided for normal operation and transient events to maintain the integrity of the fuel cladding. This objective is achieved by limiting fuel rod power density to avoid overstressing the fuel cladding because of fuel pellet-cladding differential expansion and by maintaining nucleate boiling around the fuel rods so that the transition to film boiling is avoided.

LHGR

The heat generation rate per unit length of the fuel rod, commonly expressed in kilowatts per foot (kw/ft) of the fuel rod. LHGR is heat flux integrated over every square centimeter of the cladding surface for one linear foot of the fuel rod.

CPR Critical Power Ratio

It protects against fuel damage resulting from the loss of nucleate boiling by limiting the total power that a given fuel bundle is allowed to produce.

To achieve those conditions another concept, have to be commitment in the refuel safety plan and the operations, like:

Critical Heat Flux.

Is the physical phenomena where the liquid phase change happens due to a certain quantity of heat forming bubbles on the clad metal surface and heating water surround it.

Some correlations developed from appropriate steady-state and transient-state experimental data are acceptable for use in predicting the critical heat flux (CHF).

In BWR technology the term critical heat flux characterizes the rapid local deterioration of the heat transfer coefficient.

Some of the code development considerations must apply. To detect critical heat flux conditions (CHF) only the single phase and nucleate boiling regions of the flow boiling curve need to be considered. The thermal coupling between fuel and coolant is accomplished by using the concept of heat flux, as well. These can be accurately modeled by a correlation. Some of them, most widely used in the analyses are:

* Hench-Levy
* Barnet correlation

The Hench-Levy correlation was developed by GE based on single-, four- and nine-rod bundle experiments with uniform axial heat flux. This correlation is in the British System of Units and covers the range of system parameters. It is divided into two ranges one for 1000 psia of pressure and other for pressure other than 1000 psia. For 100 psia. The correlation is in terms of three limit lines as a function of the position and the mass flow.

The computer programs in which these correlations are used shall contain suitable checks to assure that the physical parameters are within the range of parameters specified for use of the correlations during the code development.