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METHODS OF CRITICAL HEAT FLUX PREDICTION IN SUB-COOLED WATER FLOW IN VVER-1200 FUEL RODS

Abstract. The prediction methods for critical heat flux (CHF) in sub-cooled boiling is presented with the aim of finding suitable model to use in the prediction of CHF in VVER-1200. Various models are available in literature, including; experimental data collected over the past 40 year for rod bundles

and circular tubes, look up table (LUT) developed for circular tubes with correction factors for rod bundles of different geometrical configuration, mechanistic models developed to predict CHF independent of empirical terms and phenomenological simulation using state of the art codes and computational fluid dynamics (CFD). The relevant models include; mechanistic model (bubble crowding model and liquid sub-layer dry-out model), Groeneveld and Bobkov LUT and empirical correlation methods applicable to high pressure (Bowring, Hall et al, Becker et al, Griffel et al, Katto, Tong-75, Mod-Tong, Levy, W-2, W-3, OKB Gidropress and Levitan-Lantsman methods). A demonstration of prediction accuracy of one of the correlation models was carried out using the Levitan-Lantsman correlation and validated by results of similar analysis on a VVER reactor. The Levitan-Lantsman correlation performed well when compared with LUT.

1. Introduction

The critical heat flux is an essential parameter in the operation of nuclear reactors, considering that it describes the heat distribution per unit area of the heated channel. One of the greatest thermal-hydraulic challenges is the prediction of the point of departure from nucleate boiling in a reactor core. The critical importance of CHF in reactor operation and the lack of reliable mechanistic models, prompted experts rely on semi-empirical and empirical correlation models and look up table [1]. One of the important reviews of CHF are the reviews by Yang B. W. et al [2], on the progress made in rod bundle CHF during the past 40 years and AbdulHameed M. et al [3] on empirical correlations of CHF in rod bundles, they noted the independence of subcooled CHF on axial heat flux non-uniformity. They observed that the heated length is accountable for the decaying of axial CHF and therefore should be considered during CHF computation using either the look up table or the empirical correlations. The work done by Liu W. et al [4], noted that subcooled nucleate boiling typically occur towards the outlet of a PWR reactor under operational state, and as such, departure from nucleate boiling is the CHF regime that is likely to occur because of the low equilibrium quality in the heated rod bundle.

2. Prediction of rod bundle CHF

CHF in rod bundles can be evaluated using mechanistic model, empirical correlation, LUT, simulation and experimental data. In this work, we applied the Levitan-Lantsman correlation to evaluate CHF for VVER-1200 rod bundles.

2.1 Calculation using Rod Bundle Correlations Validated by Groeneveld and Bobkov Look up tables

The evaluation of critical heat flux, largely depends channel geometrical configuration and thermal properties of the coolant. The specific thermal-hydraulic parameters include; pressure, equilibrium steam quality, mass flux, diameter of the fuel rod, pitch to diameter ratio, geometrical configuration of the fuel bundle, power density distribution and spacer (grid or wire). In the evaluation of DNB, the MDNBR plays a crucial role in limiting the operating power of PWR for safety purposes. This point was emphasized in the work of M. Amin Mozafari, F. Faghihi [5], where they employed three (3) methods to evaluate MDNBR OF VVER-1200 fuel rod. The three (3) methods applied include; the Westinghouse W-3 correlation, the OKB Gidropress correlation [6], and the Bobkov [7]; Groeneveld [8], look up tables. The result of their MDNBR evaluation using the Bushehr and Temelin nuclear power plants model ranged from 16.1 to 2.62. The W-3 correlation is written as;

$$\begin{aligned} q''_{cr} = &\{(2.022-0.06238p)+(0.1722-0.01427p)\cdot\exp(18.177-0.5987p)x_e\}\cdot \\ &\{(0.1484-1.596x_e+0.1729x_e|x_e|)\cdot 2.326G+3271\}\cdot\{1.157-0.869x_e\}\cdot \\ &\{0.2664+0.8357\exp(-124.1D_h)\}\cdot\{0.8258+0.0003413(h_f-h_{in})\} \end{aligned} \quad (1)$$

Where q''_{cr} is the critical heat flux for uniformly heated channel in kW/m^2, x_e is the local equilibrium quality, p is the pressure in (MPa) and G is the mass flux in kg/m^2s D_h is the hydraulic diameter, h_f is the liquid saturated enthalpy and h_{in} is the inlet enthalpy.

The OKB Gidropress Critical heat flux correlation is written as;

$$q''_{cr} = 0.795\cdot(1-x)^n\cdot(G)^m\cdot(1-0.0185\cdot p) \quad (2)$$

$$m = 0.311\cdot(1-x)-0.127;$$

$$n = 0.105\cdot p-0.5;$$

$$P = pressure(MPa); G = mass-flux(kg/m^2s); x = steam-quality;$$

$$D = tube-diameter(mm); q''_{cr} = critical-heat-flux(W/m^2)$$

The Levitan-Lantsman Critical heat flux correlation [9] is also a compactible correlation for VVER reactors. Its prediction of MDNBR for VVER-1200 is comparable to the work of M. Amin Mozafari, F. Faghihi. The correlation is given below as;

$$\begin{aligned} q''_{cr}(8mm,G,p,x) = &\left[10.3-7.8\left(\frac{p}{98}\right)+1.6\left(\frac{p}{98}\right)^2\right] \\ &\left(\frac{G}{1000}\right)^{1.2\{[0.25(p-98)/98]-x\}}\cdot e^{-1.5\cdot x} \end{aligned} \quad (3)$$

$29.4 \le p \le 196$; $750 \le G \le 5000$

$$q''_{cr}\left(d_{clad}, G, p, x\right) = q''_{cr}\left(8mm, G, p, x\right)\left(\frac{8}{d_{clad}}\right)^{0.5} \quad (4)$$

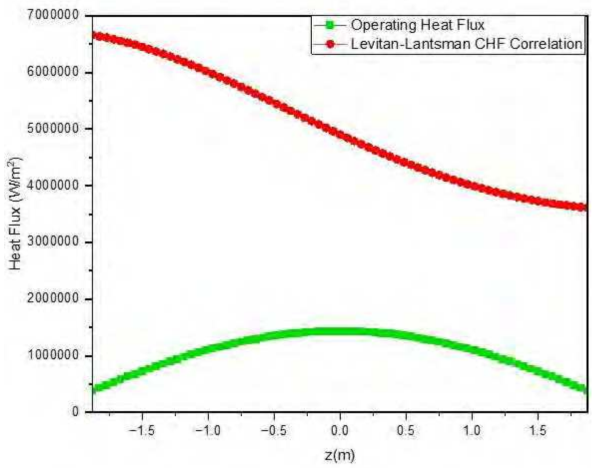
$$P = pressure(bar); G = mass-flux\left(kg/m^2 s\right); x = steam-quality;$$

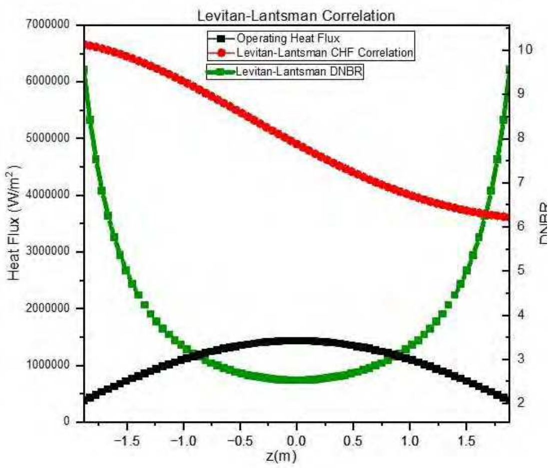
$$D = tube-diameter(mm); q''_{cr} = critical-heat-flux\left(W/m^2\right)$$

Other correlations such as Bowring, Hall et al, Becker et al, Griffel et al, Katto, Tong-75, Mod-Tong, Levy, W-2, that are applicable at high pressure condition were reviewed by Liu P. et al [10]

2.2 Result of Calculation using Levitan-Lantsman Correlation

The computed result is presented in Figures 1 and 2 (heat flux/Levitan-Lantsman CHF vs fuel rod height; heat flux/Levitan-Lantsman CHF and DNBR vs fuel rod height). The Levitan-Lantsman Correlation, predicted the MDNBR for VVER-1200 to be 2.53 as seen on the graph in figure 2. The result is comparable to the results of various MDNBR for VVER reactors found in literature. The thermal properties of VVER-1200 reactor used in this work were evaluated using the Magnus Holmgren's IAPWS Excel Steam Table and the plot were obtained using ORIGIN-2023.





3. Conclusion

The W-3, OKB Gidropress and Levitan-Lantsman correlation were presented. The Levitan-Lantsman correlation was selected for the CHF evaluation due to its good predictive power in evaluating CHF for subcooled boiling at high pressure and mass flux. With correction factors, it can be used to predict CHF in VVER-1200 rod bundles.

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