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FEATURES OF THERMAL-HYDRAULIC CALCULATION OF SUBCOOLED WATER REACTOR VVER-1200

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ОСОБЕННОСТИ ТЕПЛОГИДРАВЛИЧЕСКОГО РАСЧЕТА ВОДООХЛАЖДАЕМОГО РЕАКТОРА ВВЭР-1200

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Аннотация. *На основе установленных корреляций для труб круглого сечения и модифицированных для пучков тепловыделяющих стержней с поправочными коэффициентами проведен анализ теплогидравлических характеристик активной зоны реактора ВВЭР-1200 при нормальном режиме эксплуатации. Сначала проведен анализ геометрии топливных стержней, массового расхода и скорости теплоносителя в тепловыделяющих сборках, плотности теплового потока в наиболее горячей части активной зоны, затем теплогидравлический анализ характеристик теплоносителя, таких как энтальпия, температура и запас до параметров насыщения. Эмпирические корреляции Суботина были использованы в расчете числа Нуссельта и конвективного коэффициента теплоотдачи для установления температур оболочки стержней и топлива. Критический тепловой поток (КТП) рассчитан с использованием эмпирических формул Гидропресса и Левитана-Ланцмана для пучков стержней в диапазоне режимных параметров теплоносителя, подходящих для реактора ВВЭР-1200. Используемое термодинамическое качество основано на оценочных параметрах термодинамического равновесия для соответствующих давлений теплоносителя с использованием таблицы водяного пара IAPWS Excel. Эмпирические формулы для расчета КТП показали хорошее соответствие, в сравнении с существующими экспериментальными и табличными данными.*

Introduction. The number of empirical CHF correlations has increased over the past 50 years with more than 100 correlations for only sub-cooled water flow in tubes [1]. This cumbersome complexity of CHF phenomenon led to the collation of experimental data points for ranges of thermal and hydraulic parameters to develop a lookup table for CHF phenomena [2, 3]. For the VVER rod bundle, the CHF value obtained from Bobkov table is multiplied by four factors to be applicable to VVER reactor [4].

Research methods. We modeled the thermal-hydraulics analysis of VVER-1200 using the initial design parameters and calculated the thermal-physical parameters of the coolant and the geometrical and thermal parameters of the fuel elements. The Subbotin Nusselt correlation [5] was used to determine the Nusselt number and hence the heat transfer coefficient of the coolant. Then we employed the OKB Gidropress [5] and Levitan-Lantsman [6] correlations to predict the critical heat flux considering that they are well established correlations for triangular lattice with relevant correction factors and application ranges for a VVER reactor. The OKB Gidropress CHF empirical correlation is developed with a standard deviation of about 13.1 %, while the Levitan-Lantsman correlation has a standard deviation of about ±15 %.

Thermal-Hydraulics Analysis. The Design of thermal-hydraulic calculation of a pressurized water reactor (VVER-1200) was carried out with the computed and given geometrical and thermal parameters of the fuel pellet, cladding and the entire core in Tables 1 and 2, and the thermal parameters of the coolant were computed with the aid of Magnus Holmgren's IAPWS Excel Steam Table.

The heat flux density $q_f(z)$ for each value of the coordinates z

$$q_f(z)\cdot\pi\cdot d_{clad}\cdot H_0 = q_l(z)\cdot H_0 \tag{1}$$

- OKB Gidropress Critical heat flux correlation,

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

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

$n = 0.105\cdot p - 0.5 = 0.105\cdot 16.2 - 0.5 = 1.201$

- Levitan-Lantsman Critical heat flux correlation

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

$29.4 \le p \le 196$; $750 \le (\rho\omega) \le 5000$

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

where p is coolant pressure, MPa; x is relative enthalpy of the coolant; (ρw) is mass velocity, kg/(m²·s); d_{clad} is the outer diameter of the fuel cladding, mm

- Departure from Nucleate Boiling Ratio (DNBR)

$$DNBR(z) = \frac{q''_{cr}(z)}{q_f(z)} \tag{5}$$

Analysis of Result. The result of our calculation for the selected thermal hydraulic parameters is presented as distribution graphs in Fig. 1 (a-d), respectively. In the analysis, the computed equilibrium quality distribution depicts a typical highly subcooled flow, with an inlet subcooled of about 50 °C and an outlet subcooled of about 19 °C. the inlet outlet equilibrium qualities are -0.36 and -0.163. This clearly shows that no onset of significant void (OSV) took place within the reactor core under normal operation.

The OKB Gidropress correlation predicted the minimum departure from nucleate boiling ratio (MDNBR) to be 4.69, ±13.1 % of 4.69 amounts to 4.03 and 5.35. The Levitan-Lantsman correlation predicted the minimum departure from nucleate boiling ratio (MDNBR) to be 3.51. ±15 % of 3.51 amounts to 2.98 and 4.03. The results of the CHF showed that the Levitan-Lantsman correlation predicts CHF and MDNBR fairly better with an estimated value of about 2.98 which is quite comparable to the values found in literature for instance the works of Mozafari M. A., Faghihi F [7].

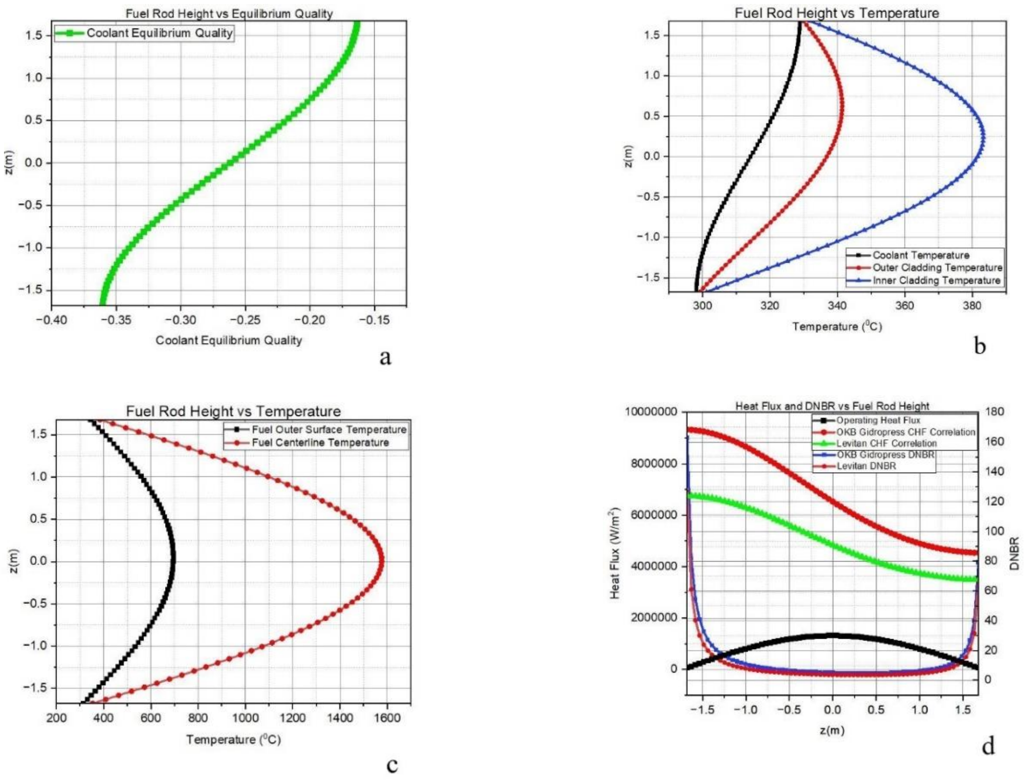

Fig. 1. Coolant equilibrium quality axial distribution (a); Coolant, outer and inner cladding temperature distribution (b); Fuel outer surface and centerline temperature distribution (c); Heat flux, OKB, Levitan CHF and DNBR distribution

Conclusion. In conclusion, a thermal hydraulic design calculation was performed for VVER-1200 reactor under normal operating condition. The selected thermal parameters that were analyzed did not deviate from established values applicable within the range of VVER-1200 reactor.

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