APPLICATION OF THE 1995 CHF TABLE TO NUCLEAR FUEL ELEMENTS

Marcelo Antônio Veloso
Centro de Desenvolvimento da Tecnologia Nuclear, CDTN
Comissão Nacional de Energia Nuclear, CNEN
Rua Prof. Mário Werneck, s/n, Caixa Postal 941, Pampulha
31120-970, Belo Horizonte, Brasil

ABSTRACT

The 1995 CHF table for uniformly heated round tubes was developed jointly by Canadian and Russian researchers. The present work concerns the use of this table for the prediction of critical heat flux (CHF) in two 5x5 test sections simulating the reload fuel assembly of Angra-I PWR Nuclear Power Plant. Comparisons between measured and calculated CHF indicate that the 1995 CHF table with an appropriate diameter correction factor can also be applied to rod bundles of the type considered in this study. The relation for the diameter correction factor was derived from the CHF data. The tolerance limits associated with the departure from nucleate boiling ratio (DNBR) are evaluated by using statistical analysis.

I. INTRODUCTION

Critical heat flux (CHF) is one of the most important quantities when considering the safety limits of nuclear reactors, steam generators, and other thermal units. If a heated surface is cooled by a fluid under nucleate boiling condition, the heat transfer coefficient is relatively high and a large amount of heat can be removed with small temperature differences between the surface and the fluid. However, this excellent characteristic of heat transfer is not boundless, that is, the heat flux cannot be increased indefinitely. At some critical heat flux, the steam produced leads to the formation of a continuous vapor film over the surface which may cause the destruction of the heater due to a sharp increase in the surface temperature.

Hundreds of different models and correlations for the prediction of CHF exist in the literature and an enormous amount of experimental data are available at present. Most of the studies carried out on CHF have been reviewed in various publications [1-4]. The first reference is particularly interesting as it deals specifically with the CHF phenomenon in both internal and external flows and presents a vast bibliography on this subject.

In general, CHF correlations are applicable to specific geometries and cover specific ranges of flow parameters. They cannot be extrapolated to conditions far beyond the ranges of their data bases. To overcome this difficulty, Doroschuk et al. [5,6] proposed around 1975 the first standard table for calculating CHF in uniformly heated round tubes cooled by boiling water. The model developed by Doroschuk et al. is based on the local crisis hypothesis. This hypothesis suggests that the critical heat flux is solely a function of flow parameters at the point where CHF occurs. Furthermore, it is assumed that the history of the flow has no effect on the critical heat flux.

Since the publication of the table of Doroschuk et al., development works of CHF tables have been continued in Canada and Russia. In 1986, Groeneveld, Cheng and Doan [7] published the 1986 AECL-UO CHF table. This table, based on about 15000 tube data, covers wider ranges of flow conditions. Kirillov et al. [8] improved the table of Doroschuk et al. by using a data base with 7620 data points. Very recently, an updated table was developed jointly by Atomic Energy of Canada Ltd. (AECL, Chalk River, Canada) and Institute of Physics and Power Engineering (IPPE, Obninsk, Russia). This table, named 1995 CHF table [9], was derived from a combined AECL-IPPE world data bank consisting of about 23000 data points. It covers the following ranges of parameters: pressure, 0.1 to 20.0 MPa; mass flux, 0 to 8000 kg/m$^2$s; quality, -0.5 to 1.0; diameter, 0.003 to 0.04 m; and length to diameter ratio, 80 to 2485.

The standard CHF tables are presented for discrete range of pressure, mass flux, and steam quality. They are normalized for a fixed tube diameter of 8 mm. The critical heat flux for conditions between the tabulated values are obtained by linear interpolation, and an empirical correction factor is used to extend the table to tube diameters other than 8 mm.

Assuming that the critical heat flux is determined by the flow conditions where the phenomenon occurs, the CHF tables, although derived for upward flow in a uniformly heated 8-mm tube, can also in principle be extended to rod bundles. Correction factors to the tables taking into account the effects of grid spacers, cold wall, nonuniform heat flux
on CHF might be necessary when using them for rod bundles. Rod bundle correction factors applicable to the 1986 AECL-UO CHF table are reported by Groeneveld et al. [7,10].

In this paper, the 1995 CHF table is compared with measurements of critical heat flux taken from two 5x5 test sections consisting of 24 electrically heated rods and an unheated rod. The test sections, namely TS10 and DTS35, are concerned with the development of the Siemens/KWU grid spacers for the reload fuel assemblies of Angra-1 nuclear power plant. The paper consists basically of two parts: the first part concerns the determination of the diameter correction factor for the 1995 CHF table by using the CHF data obtained from TS10 and TS35; subsequently, the predicted local subchannel conditions are substituted into the proposed CHF model to obtain a distribution of the departure from nucleate boiling ratios (DNBR) as a function of local steam quality; and the upper one-sided tolerance limit for the DNBR is estimated by applying statistical methods to the DNBR data.

II. DESCRIPTION OF EXPERIMENTS

A cross sectional view of the TS10 and DTS35 test sections is shown in Fig. 1. Inconel tubes electrically heated were used to simulate fuel rods. All the heater rods had an active length of 3000 mm and an outer diameter of 9.50 mm. The unheated rod (rod no. 19) simulating a control rod guide tube had an outer diameter of 11.96 mm. The rod pitch was 12.32 mm. The rod-to-wall gap was 4.32 mm for TS10 and 2.13 mm for DTS35.

Two levels of radial power were obtained by using tubes of the same outer diameter and two different wall thicknesses. Rods nos. 20 to 25 in TS10 rated 15% higher power than the other heater rods, while in DTS35 these six rods were heated at a 20% higher power. The test sections had uniform axial power distributions.

The change in the surface temperature associated with CHF was detected by thermocouples mounted inside the heater rods near the end of the active length. Rods nos. 20 to 25 in both the test sections were instrumented with four thermocouples per rod in the same axial plane, equally disposed in azimuth within the rod. Rods nos. 1 to 18 in TS10 had two thermocouples per rod at the same axial position. Each of rods nos. 1 to 18 in DTS35 had only one thermocouple.

For each CHF run, the independent test parameters (pressure, inlet temperature and flow rate) were preset to the desired values and the electric power to the bundle was then gradually increased until a rapid increase in surface temperature was shown by one or more of the CHF detectors. Once this happened, test section power was immediately reduced by the operator or automatically tripped. Data including voltage, current, inlet temperature, inlet flow rate, outlet pressure, and the identification numbers of rods and thermocouples indicating CHF were continuously recorded. The set of measurements taken immediately before the power reduction constituted the recorded CHF conditions.

The TS10 test series was carried out in the 6 MW High Pressure Water Rig situated at Atomic Energy Establishment Winfrith (AEZW), United Kingdom. The DTS35 test series was conducted at the Heat Transfer Research Facility of Columbia University, New York. A total of 203 CHF data points were obtained from these experiments (101 from TS10 and 102 from DTS35). The experiments covered the following parameter ranges: pressure, 7 to 17 MPa; inlet temperature, 200 to 330°C for TS10 and 150 to 330°C for DTS35; mass flux, 600 to 3600 kg/m²s.

III. DIAMETER CORRECTION

When using the standard CHF table for tube diameters other than 8 mm, the critical heat flux is given by the approximate relationship

\[ q_{\text{crit,D}} = q_{\text{crit,8mm}} \left( \frac{D}{8} \right)^k \]  

where \( q_{\text{crit,8mm}} \) is the critical heat flux for an 8 mm tube obtained from the CHF table, D is the tube diameter in millimeters, and k is an empirical parameter.

Reliable values for exponent k are not available yet. Doroschuk et al. [5,6] corrected their tabulated critical heat fluxes with a value of \( k = -1/2 \). Groeneveld et al. [7] recommended an exponential value of \( -1/3 \) to be used in conjunction with their 1986 AECL-UO CHF table. For the
boundary is dispersed flow to dispersed flow is \( X \)
The left boundary of the transition region from annular-dispersed flow to dispersed flow, and (5) dispersed flow. The reader is referred to Ref. [8] for the description of these flow patterns.

According to Kirillov et al., the quality at the onset point of the annular-dispersed flow region is given by

\[
X_1 = 2.7 \left( \frac{\rho_f \sigma}{G^2 D} \right)^{0.25} \left( \frac{\rho_g}{\rho_f} \right)^{0.333}
\]

where \( G \) is the mass flux, \( D \) is the tube diameter, \( \sigma \) is the surface tension, and \( \rho_f \) and \( \rho_g \) are the densities of saturated liquid and saturated vapor, respectively. The parameters in Eq. (2) are all expressed in SI units. The right side boundary of this region is

\[
X_2 = 0.188 - 0.068 - 0.266
\]

when \( 1 \leq P \leq 6 \) MPa

\[
X_2 = 0.52 \left( \frac{10^3 \rho_f \sigma}{G^2 D} \right)^{0.280} \left( \frac{\rho_g}{\rho_f} \right)^{0.0.119}
\]

when \( 6 \leq P \leq 20 \) MPa

The left boundary of the transition region from annular-dispersed flow to dispersed flow is \( X_2 \) and the right boundary is

\[
X_3 = 1.18 \left( \frac{10^3 \rho_f \sigma}{G^2 D} \right)^{0.238} \left( \frac{\rho_g}{\rho_f} \right)^{0.204}
\]

when \( 1 \leq P \leq 6 \) MPa

\[
X_3 = 0.57 \left( \frac{10^3 \rho_f \sigma}{G^2 D} \right)^{0.30} \left( \frac{\rho_g}{\rho_f} \right)^{-0.0367}
\]

when \( 6 \leq P \leq 20 \) MPa

The values of \( k \) estimated by Kirillov et al. are listed in Table 1. The CHF values in transition regions 2 and 4 are obtained by linear interpolation between the boundary values. For pressures above 14 MPa, the correction factors are defined only for regions 1 and 5 since the other regions are degenerate.

When the calculated subchannel conditions at the location of critical heat flux for a particular run are substituted into a CHF correlation, a predicted CHF is produced. The ratio of the predicted CHF to the measured CHF at a given location is the departure from nucleate boiling ratio (DNBR), that is,

\[
DNBR = \frac{q_{\text{crit,predicted}}}{q_{\text{crit,measured}}}
\]

(7)

where \( q_{\text{crit}} \) denotes the critical heat flux. The diameter correction factor for a given CHF run may be expressed as

\[
f_D = \frac{q_{\text{crit,measured}}}{q_{\text{crit,8mm}}} = \frac{1}{DNBR_{8\text{mm}}}
\]

(8)

where \( q_{\text{crit,8mm}} \) is the critical heat flux obtained from the CHF table without the diameter correction.

Subchannel mass flux and steam quality at the CHF location for each TS10 and DTS35 CHF run were computed with the PANTERA-1P code [11], a CDTN version of the COBRA-IIIC [12], from the observed values of pressure, inlet temperature, bundle average mass flux, and bundle average heat flux. Determinations with the PANTERA-1P were carried out by using the input parameters and input correlations summarized in Table 2. The parameters given in this table have the same definitions as those ones used in the COBRA-IIIC. The extent of turbulent mixing among subchannels was determined using a turbulent mixing coefficient (\( \beta \)) of 0.050 [13]. It should be emphasized that the CHF subchannel was assumed to be that subchannel adjacent to the CHF rod presenting in calculation the highest exit steam quality.

Table 1. Values of \( k \) in Eq. (1) [8]

<table>
<thead>
<tr>
<th>Region</th>
<th>( k )</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>1.18</td>
</tr>
<tr>
<td>3</td>
<td>0.57</td>
</tr>
<tr>
<td>5</td>
<td>0.188</td>
</tr>
</tbody>
</table>

Table 2. Input Parameters and Correlations

<table>
<thead>
<tr>
<th>Correlation</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Crossflow resistance factor, ( K_{ij} )</td>
<td>0.5</td>
</tr>
<tr>
<td>Transverse momentum parameter, ( \beta )</td>
<td>0.5</td>
</tr>
<tr>
<td>Turbulent momentum factor, ( f_t )</td>
<td>0.0</td>
</tr>
<tr>
<td>Friction factor correlation</td>
<td>Blasius [3]</td>
</tr>
<tr>
<td>Subcooled void correlation</td>
<td>Levy [14]</td>
</tr>
<tr>
<td>Bulk void correlation</td>
<td>Smith [15]</td>
</tr>
<tr>
<td>Two-phase friction multiplier correlation</td>
<td>EPRI [16]</td>
</tr>
</tbody>
</table>

The diameter correction factors obtained for both the test sections are plotted in Fig. 2 as a function of local steam quality. Despite the spread of points, this figure indicates that \( f_D \) tends to decrease with increasing quality. Because of this trend, it was decided to express the
diameter correction factor as
\[ f_D = \left( \frac{D_w}{8} \right)^{k(x)} \] (9)

where \( D_w \) is the wetted equivalent diameter of the CHF subchannel. Exponent \( k(x) \) as a function of local steam quality was obtained by considering the flow patterns reported by Kirillov et al. [8]. Tentatively, using conventional regression analysis, the following relationships were derived for exponent \( k(x) \):

\[ k(x) = -0.263 - 47.1x^2 \quad \text{for} \quad x \leq X_0 \] (10)
\[ k(x) = -0.604 - 6.65x \quad \text{for} \quad x \geq X_1 \] (11)

where \( X_0 \) is the upper boundary of bubble flow, and \( X_1 \) is the quality at the onset of annular-dispersed flow which is given by Eq. (2) with diameter \( D \) replaced by \( D_w \). Quality \( X_0 \) was approximated by the relation
\[ X_0 = \frac{1}{1 + (1/\alpha - 1) \rho_f / \rho_g} \] (12)

where \( \alpha = 0.3 \) is the transition void fraction value reported by Hewitt [17].

In the transition region from bubble flow to annular-dispersed flow, exponent \( k(x) \) may be evaluated by linear interpolation, that is,
\[ k(x) = \frac{x - X_0}{X_1 - X_0} \left[ k(X_1) - k(X_0) \right] + k(X_0) \] (13)

Due to reduced number of CHF data in the regions that follows the annular-dispersed flow – transition region from annular-dispersed flow to dispersed flow, and dispersed flow – no attempt was made to derive equations for \( k(x) \) in such regions. Comparisons between predicted and measured values of critical heat flux have indicated that Eq. (11) may also be applied to those flow regions.

**IV. DNBR STATISTICS**

The DNBR values should be one at all CHF test conditions if experimental errors were zero, if the flow parameters at the CHF location were perfectly determined by a subchannel code, and if the critical heat fluxes were exactly predicted by the CHF correlation. As this can never be the case, the DNBR data are found to produce a probability distribution.

When a sample \( \text{DNBR}_1, \text{DNBR}_2, \ldots, \text{DNBR}_n \) is taken, the mean value is given by
\[ \overline{\text{DNBR}} = \frac{1}{n} \sum_{i=1}^{n} \text{DNBR}_i \] (14)

and the standard deviation may be evaluated from
\[ S = \left( \frac{1}{n-1} \sum_{i=1}^{n} \left( \text{DNBR}_i - \overline{\text{DNBR}} \right)^2 \right)^{1/2} \] (15)

where \( n \) is the total number of data. The average error and the root-mean-square error are defined as
\[ \epsilon_{\text{avg}} = \frac{1}{n} \sum_{i=1}^{n} \epsilon_i \] (16)
\[ \epsilon_{\text{rms}} = \left( \frac{1}{n} \sum_{i=1}^{n} \epsilon_i^2 \right)^{1/2} \] (17)

where
\[ \epsilon_i = \frac{q_{\text{crit.predicted}} - q_{\text{crit.measured}}}{q_{\text{crit.measured}}} \]

A typical Pressurized Water Reactor (PWR) design criterion is that CHF will not occur in the core at a 95% probability with a 95% confidence level. In order to meet this criterion, a limiting value of DNBR is determined by applying statistical methods to random samples of DNBR data taken from tests simulating reactor conditions. For historical reasons, the minimum DNBR predicted in the core cannot be less than 1.30.

For a normal random variable DNBR with unknown mean \( \mu \) and unknown standard deviation \( \sigma \), the upper one-sided tolerance limit for the DNBR is given by [18]
\[ \overline{\text{DNBR}} + K.S \] (19)

where \( \overline{\text{DNBR}} \) is an estimate of \( \mu \) and \( S \) is an estimate of \( \sigma \). Values of \( K \) such that the probability is \( \gamma \) that at least a proportion \( p \) of a normal distribution is below the minimum DNBR (MDNBR) may be obtained from tables given in textbooks of statistics.
The ratios of predicted to measured critical heat fluxes obtained by using the 1995 CHF table along with the diameter correction factor given by Eq. (9) are plotted in Fig. 3 as a function of local steam quality. The points on the line DNBR = 1.0 indicates that the predictions match the measurements. The points above this line mean that the proposed model overestimates the critical heat fluxes. Conservative predictions are indicated by the points falling below the line DNBR = 1.0.

![Figure 3. Ratio of Predicted CHF to Measured CHF as a Function of Local Steam Quality](image1)

It can be inferred from Fig. 3 that the DNBR data points distribute around DNBR = 1.0 and that most of the data fall within the range $0.8 \leq \text{DNBR} \leq 1.2$. The 1995 CHF table with Eq. (9) predicts close to 70% of the data within the ±15% error range. This is shown in Fig. 4, where the error histogram is plotted using 10% error intervals. The average and rms errors are 14.5% and 19.2%, respectively.

![Figure 4. Histogram of the DNBR Data Points](image2)

DNBR data shown in Fig. 5 correspond to those runs for which calculated CHF subchannel exit steam qualities were less than 0.20. This upper quality is still above the maximum quality (about 0.1) expected to occur in typical PWR fuel elements under normal operation. Four test runs on TS10 presenting DNBR greater than 1.5 were excluded from the sample, since they were believed to contain experimental deviations. The average and rms errors associated with the data plotted in Fig. 5 are 11.3% and 14.5%, respectively.

![Figure 5. Ratio of Predicted CHF to Measured CHF for Local Steam Quality less than 0.2](image3)

Assuming that the DNBR data points are normally distributed, Eq. (19) may be used for determining the upper DNBR tolerance limit which contains at least 95% of the distribution with probability 95%. DNBR tolerance limits for several quality ranges are summarized in Table 3. Values of $K$ corresponding to $\gamma = p = 0.95$ and sample size $n$ (number of DNBR data points) were obtained from tables prepared by Owen [18]. It can be observed that for steam quality up to 0.20 the minimum DNBR is less than 1.30 and satisfies the design criterion that critical heat flux will not occur in the geometries under consideration at a 95% probability with a 95% confidence level.

<table>
<thead>
<tr>
<th>Quality $x$</th>
<th>$\text{DNBR}$</th>
<th>$S$</th>
<th>$n$</th>
<th>$K$</th>
<th>$\text{MDNBR}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>$x \leq 0.10$</td>
<td>1.021</td>
<td>0.100</td>
<td>70</td>
<td>1.990</td>
<td>1.219</td>
</tr>
<tr>
<td>$x \leq 0.15$</td>
<td>1.020</td>
<td>0.129</td>
<td>100</td>
<td>1.927</td>
<td>1.268</td>
</tr>
<tr>
<td>$x \leq 0.20$</td>
<td>1.021</td>
<td>0.144</td>
<td>128</td>
<td>1.890</td>
<td>1.293</td>
</tr>
<tr>
<td>$x \leq 0.25$</td>
<td>1.025</td>
<td>0.158</td>
<td>156</td>
<td>1.865</td>
<td>1.316</td>
</tr>
<tr>
<td>$x \leq 0.30$</td>
<td>1.021</td>
<td>0.165</td>
<td>164</td>
<td>1.859</td>
<td>1.328</td>
</tr>
<tr>
<td>$x \leq 0.35$</td>
<td>1.029</td>
<td>0.169</td>
<td>173</td>
<td>1.853</td>
<td>1.342</td>
</tr>
</tbody>
</table>

**V. CONCLUSIONS**

This work concerned the assessment of critical heat flux (CHF) measurements taken from two 5x5 test sections...
simulating the reload fuel assembly of Angra-1 nuclear power plant. The study, although being carried out for specific geometries, reveals that the standard CHF tables for tubes appears as an interesting alternative for the determination of critical heat flux in nuclear fuel elements. In contrast with empirical correlations or semi-analytical CHF models, these tables are easy to be used and cover wide range of flow parameters.

Assuming that the local crisis hypothesis can also be applied to complex geometries, slight corrections to the CHF tables are expected to be necessary to extend them to rod bundles. However, since the prediction of CHF in subchannel geometries is considerably more difficult than the prediction of CHF in tubes, further investigations in the light of experimental results need to be conducted to confirm the above assumption. The effects of fuel assembly characteristics on CHF – such as mixing vane grid spacers, wetted and heated equivalent diameters, flow mixing, and radial and axial heat flux distributions – need to be considered.

Comparisons between calculated and measured critical heat fluxes from both the test sections indicate that the 1995 CHF table with the diameter correction factor given by Eq. (9) reproduces the CHF data in the quality range from -0.1 to 0.6 with an average error of 14.5% and a root-mean-square square error of 19.2%. About 70% of the data range from -0.1 to 0.6 with an average error of 14.5% and a root-mean-square square error reduces to 11.3% and 14.5%, respectively. The minimum departure from nucleate boiling ratio (MDNBR) with a 95x95 tolerance in the quality range from -0.1 to 0.2 was found to be 1.29.

REFERENCES


