Forschungszentrum Karlsruhe

in der Helmholtz-Gemeinschaft

Wissenschaftliche Berichte

FZKA 6825

Review on Critical Heat Flux in Water Cooled Reactors

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2003

Impressum der Print-Ausgabe:

Als Manuskript gedruckt

Forschungszentrum Karlsruhe GmbH
Postfach 3640, 76021 Karlsruhe

Mitglied der Hermann von Helmholtz-Gemeinschaft
Deutscher Forschungszentren (HGF)

ISSN 0947-8620

ABSTRACT

REVIEW ON CRITICAL HEAT FLUX IN WATER COOLED REACTORS

Intensive investigations on the critical heat flux (CHF) have been performed in the last several decades due to its importance in nuclear engineering. This paper presents a brief overview of experimental and theoretical studies on critical heat flux with the emphasis on nuclear engineering application. The review is restricted to CHF under the normal operating condition of different types of water cooled reactors. Experimental methods using geometry modelling as well as fluid modelling are described. Representative CHF data banks obtained in different flow channel geometries (tubes, rod bundles) and in different fluids (water, Freon-12) are presented. Methods for predicting CHF are reviewed for both tubes and rod bundles, ranging from empirical correlations, look-up tables and phenomenological models.

KURZFASSUNG

WISSENSSTAND ÜBER DIE KRITISCHE HEIZFLÄCHENBELASTUNG IN WASSERGEKÜHLTEN KERNREAKTOREN

Umfangreiche Untersuchungen zur kritischen Heizflächenbelastung (KHB) wurden in den letzten Jahrzehnten durchgeführt, insbesondere für die Auslegung von wassergekühlten Kernreaktoren. Dieser Bericht gibt einen Überblick über die experimentellen und theoretischen Arbeiten zur kritischen Heizflächenbelastung und ihre Anwendung in der Kerntechnik. Experimentelle Methoden mit Modellgeometrien und Modellfluiden werden beschrieben. Wichtige Datenbanken für unterschiedliche Strömungskanäle (Kreisrohre, Stabbündel) und unterschiedliche Fluide (Wasser, Frigen-12) werden vorgestellt. Verschiedene Verfahren zur Vorhersage der KHB sowohl in Kreisrohrgeometrien als auch in Stabbündeln werden zusammengestellt und diskutiert.

Contents

1. INTRODUCTION

In a nuclear reactor system the critical heat flux (CHF) is the heat flux at which a boiling crisis occurs that causes an abrupt rise of the fuel rod surface temperature and, subsequently, a failure of the cladding material. Design of a water cooled reactor requires a sufficient safety margin with regard to the critical heat flux. The importance of CHF in nuclear engineering has led to intensive investigations worldwide over several decades. In spite of a great quantity of experimental and theoretical studies, knowledge of the precise nature of CHF is still incomplete and the mechanisms of a boiling crisis are still not well understood. This is mainly due to the very complex nature of two-phase flow with heat transfer.

It is agreed that experimental investigations on CHF have to be performed for each specific design of nuclear reactors. Due to the limitation of technical feasibility and financial expense CHF experiments have often been performed in a scaled model system. Two different modeling techniques are available, i.e. geometric modelling and fluid modeling. By the geometric modeling simplified flow channels, e.g. circular tubes, instead of prototypical rod bundles are used. By using such simple flow channels it is possible to study systematically the effect of different parameters on CHF and to gain detailed knowledge of CHF for a wide range of test parameters. By the fluid modeling a substitute fluid, e.g. Freon-12, is used instead of the original fluid (water). By a proper selection of model fluids the operating pressure, operating temperature, and the heat power required would be reduced significantly. Nowadays a huge number of experimental data are available which were obtained in different flow channel geometries and in different fluids.

In addition, a large number of empirical correlations have been developed for the CHF data base obtained from particular flow channel geometries and particular parameter ranges. The application of such correlations must be restricted to the narrow parameter ranges considered. An extrapolation of their application to parameters out of this range can obviously result in a large uncertainty and is not recommended in nuclear engineering. Efforts have been made to develop more general prediction methods, e.g. look-up tables and phenomenological models. More and more interest has been aroused in the application of such prediction methods to nuclear reactor conditions.

It has become evident that in the last several decades research work on CHF has been pushed forward due to its importance in nuclear engineering. There exists a large amount of information available relating to the critical heat flux in nuclear engineering. CHF in nuclear power plants under transient or accidential conditions can be found in the studies of Celata et al. (1992) and Theofanous (1980, 1995). Boiling crisis in fusion engineering has also been well reviewed by Boyd (1985a, 1985b) and Celata (1996). This paper provides a survey on the state-of-the-art of experimental and theoretical investigations on critical heat flux with the emphasis on its application to water cooled reactors under normal operating conditions. Future research needs are underlined.

2. OPERATING CONDITIONS OF WATER COOLED REACTORS

It is well known that CHF is dependent on geometrical conditions as well as on thermal-hydraulic conditions. The most important parameters influencing CHF in a fuel element are listed as follows:

- pressure
- mass flux
- steam quality
- fuel rod diameter
- pitch to rod diameter ratio
- fuel rods configuration
- power distribution
- spacers

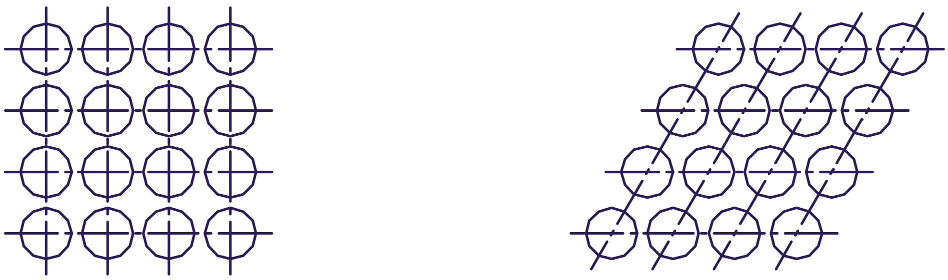
Four different groups of water cooled reactors are considered in this paper. They are pressurized water reactors (PWR), boiling water reactors (BWR), pressurized heavy water reactors (PHWR) and Russian-type pressurized water cooled reactors (VVER). Table 1 gives examples of operating conditions of each.

Table 1: examples of operating conditions of different reactors

	PWR	BWR	PHWR	VVER
Pressure *P*, [MPa]	~ 15.7	~ 7.2	~ 10.5	~ 15.7
Ave. mass flux *G*, [Mg/m²s]	~ 4.0	~ 3.0	~ 5.0	~ 4.0
Ave. outlet steam quality *X*, [-]	~ -0.15	~ 0.15	~ 0.0	~ -0.15
Fuel rod diameter *d*, [mm]	~ 9.5	~ 12.3	~ 13.1	9.1
Pitch to diameter ratio, [-]	~ 1.3	~1.3	~ 1.15	~ 1.4
Fuel rods configuration	square	square	hexagonal	hexagon al

Table 1 gives just one example from each group. It should be kept in mind that for the same reactor group, e.g. PWR, the operating conditions may differ from one design to another and vary over a wide range except for the operating pressure. For different designs of PWR's the operating pressure only ranges from 15.0 to 16.0 MPa. The mass flux changes more significantly. In a conventional PWR the mass flux is about 4 Mg/m²s, whereas in a high conversion PWR (Oldekop et al. 1982) a much higher mass flux (~ 6 Mg/m²s) is required. The same situation holds for BWR designs. In a breeding BWR with a tight lattice the mass flux is much higher (~ 6 Mg/m²s) than in a conventional BWR (~ 3 Mg/m²s). The design of PWR, PHWR and VVER reactors tries to avoid high void fraction in sub-channels. The average coolant temperature at the core outlet is somewhat below the saturation temperature. On the contrary, in a BWR high steam fractions exist at the core outlet. The average steam quality at the core outlet is about 0.15.

For water cooled reactors there are usually two different kinds of fuel rods configurations, i.e. square and hexagonal, as indicated in figure 1.



(a) square lattice *(b) hexagonal lattice*

Figure 1: configuration of fuel elements

Different definitions of sub-channels are in use. The most common approach defines sub-channel boundaries by lines between the rod centers, as shown in figure 1. This definition has been used in many sub-channel analysis codes. For a fixed configuration two geometrical parameters, e.g. rod diameter and pitch, are required to describe the size of the sub-channel. Figure 2 shows the hydraulic diameter of a sub-channel as function of the pitch to rod diameter ratio for both fuel rod configurations and for two fuel rod diameters (d=9.5 mm and d=12.3 mm).

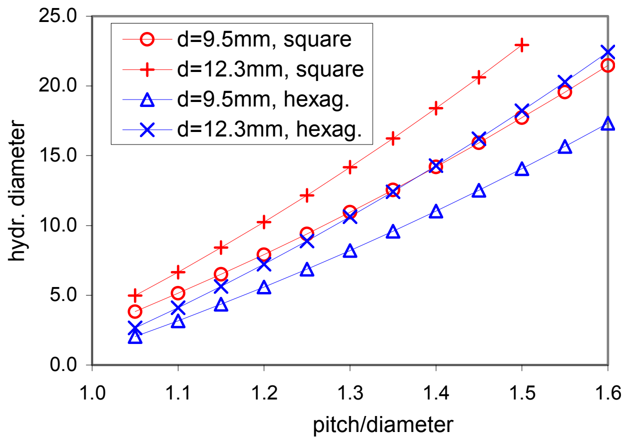


Figure 2: Hydraulic diameter of sub-channels in a rod bundle

The fuel rod configuration of a conventional PWR is of square type and the fuel rod diameter ranges from 9.0 mm to 11.0 mm. In the most recent designs of PWR (IAEA 1997) the rod diameter is about 9.5 mm. The pitch to diameter ratio ranges from 1.3 to 1.4. According to figure 2 the equivalent hydraulic diameter of the sub-channel is from 11 mm to 14 mm. For a high conversion PWR the pitch to diameter ratio is about 1.1 to 1.2, much smaller than that in a conventional PWR. In this case the fuel rods are arranged in a hexagonal lattice. The hydraulic diameter of the sub-channel is as small as 3 - 5 mm. In a BWR design the rod diameter is usually larger than in a PWR. The rod diameter is 12.3 mm in the German type BWR or in the European Simplified BWR (IAEA 1997). The pitch to diameter ratio in a BWR is similar to that in a PWR. The hydraulic diameter of the sub-channel in a BWR fuel assembly is somewhat larger (14 – 18 mm) than that in a PWR fuel assembly. Nevertheless, BWR with tight fuel lattices were also proposed, e.g. the Resource-Renewable BWR of Hitachi or the Breeding BWR of Toshiba (Okubo 1999). In both designs the rod diameter is about 10 – 11 mm. The pitch to diameter ratio is about 1.1. It is concluded from the above discussion that the hydraulic diameter of the sub-channel in a fuel assembly of water cooled reactors ranges from 3 to 18 mm.

In addition to the parameters mentioned above, two specific features which influence CHF in a nuclear reactor are heat power distribution and spacers. In nuclear reactors the axial heat flux distribution is non-uniform. It obeys more or less a cosine function. It is thus difficult to pre-determine the location of boiling crisis. This requires a complicated technique to detect the occurrence of CHF and to prevent the heating surface from burnout. In reactor cores spacers are used to fix the fuel rods. In wide lattices grid spacers are preferred, whereas in tight lattices wire wraps or ribs are used. Introducing spacers in fuel assemblies leads to (1) a redistribution of thermal-hydraulic parameters in the rod bundle and (2) a disturbance of flow conditions in the sub-channels. The first effect may be numerically quantified, whereas the second one is still not predictable. The effect of spacers on CHF can only be determined experimentally.

3. Experimental studies

The importance of CHF in nuclear engineering and the big deficiency in the modeling of the physical phenomena of a boiling crisis has led to a large number of experimental investigations and subsequently, to a large number of CHF data banks.

3.1 Experimental methods

Experimental investigations on CHF in water cooled reactors require high pressure and high heat power. Before the general design is started some details of a reactor system have to be fixed, experiments with full-scale rod bundles are not recommended. Experiments have often been performed in a scaled model system to reduce technical complexity and financial expense. Usually, geometry modeling and fluid modeling technique are in use. To gain detailed CHF knowledge over a wide range of parameters, different geometry modeling systems can be used step by step. As realized at the KRISTA test facility of the Research Center Karlsruhe (Cheng et al. 1997, 1998a, 1998b), three stages of experiments were performed to investigate CHF in a tight rod bundle of the HCPWR. At the first stage CHF tests in circular tubes of different diameters were carried out in order to study the effect of different parameters, especially the effect of the flow channel size, on CHF. At the second stage experiments in small rod bundles (7 rods) were performed with the main purpose of studying the effect of different spacers (grid spacer, wire wrap) and different heat flux distributions on CHF, and of clarifying the difference between CHF in rod bundles and those in circular tubes. At the last stage a larger rod bundle (37 rods) was used to study CHF in a prototypical fuel lattice and to assess the scaling-up of CHF results in a model system to the prototypical system.

For the fluid modeling a model fluid is used instead of the original fluid (water). In a water cooled reactor the operating pressure is high, from 7 to 16 MPa (see Table 1). Moreover, CHF experiments in water require a high heat flux of several MW/m². This leads to a high technical complexity of the test facility and a potentially very high energy consumption. By a proper selection of model fluids the operating pressure, operating temperature and the heat power required can be reduced significantly. Table 2 compares some properties of water and the commonly used Freon-12.

Table 2: properties of water and Freons

	Water	Freon-12	Freon-134a
Critical pressure P_c, MPa	22.1	4.11	4.06
Critical temperature, °C	374.0	111.5	101.1
Latent heat h_{fg} at $P = 0.5P_c$,MJ/kg	1.26	0.10	0.13

The critical pressure of Freon-12 is 4.1 MPa, about one fifth of that of water. The system pressure for CHF experiments in Freon-12 is also about one fifth of that in water. To cover the pressure range of water cooled reactors the operating pressure in Freon-12 ranges from 1.0 MPa to 2.7MPa. The saturation temperature of Freon-12 varies from 42°C to 88°C, much lower than the equivalent values in water (286° to 346°). The latent heat of Freon-12 is less than one tenth of that of water. Moreover, it has been pointed out that a reduction in heat power of about 15 times can be achieved by introducing model fluid Freon-12 into the CHF experiment. However, because of the large ozone-depletion potential (ODP) of Freon-12, a replacement fluid having a lower ODP is required. One of the most favorable replacement fluid is Freon-134a which shows no ODP. It is seen in Table 2 that the thermo-physical properties of Freon-134a are very similar to that of Freon-12.

Many test facilities have been built up worldwide to perform CHF experiments. Table 3 summarizes the design parameters of a few large-scale CHF test facilities in water and in Freon-12.

Table 3: Design parameters of some large scale CHF test facilities

Organizations/Loops	Fluid	Pressure [MPa]	Mass flow [kg/s]	Power [MW]
Columbia University (Yang, 1996)	Water	16.5	63.0	13.0
CEA/Omega (Gully, 1999)	Water	17.0	60.0	10.0
IPPE/SVD-1 (Kirillov 1997b)	Water	20.0	3.3	3.0
NUPEC (Uchida 1997, Kitamura, 1998)	Water	19.1/10.3	16.7/33.3	5.6/12.0
CEA/Graziella (Gully, 1999)	Freon-12	4.0	12.0	0.8
IPPE/STF (Kirillov 1997b)	Freon-12	5.0	11.0	1.0
NPIC (Chen, 1998)	Freon-12	4.0	14.0	0.5
FZK/KRISTA (Cheng, 1998b)	Freon-12	3.5	18.0	0.8

There exist several water loops at the Heat Transfer Research Facility of Columbia University (Yang 1996). CHF experiments in fuel bundles of PWR, BWR and PHWR are conducted. A lot of test data are produced at these test loops. At CEA two large scale loops are available, one for water (Omega loop) and one for Freon-12 (Graziella loop). CHF measurements are performed in PWR and HCPWR rod bundles (Courtaud, 1988). In IPPE/Russia there exist many test loops for different purposes (Kirillov 1997b). Two of them are indicated in Table 3. CHF experiments are conducted both in tubes and in VVER rod bundles. In NUPEC there are several loops of 10 MW size which are available for CHF experiments in PWR- and BWR-rod bundles (Uchida et al. 1997, Kitamura et al. 1998). At the Nuclear Power institute of China (NPIC) two large scale loops are being set up (Chen 1998). The construction of the Freon-12 loop has been completed. The water loop is now under construction. In both loops experiments in PWR rod bundles shall be performed and fluid-to-fluid scaling laws for rod bundle geometries will be studied. A large scale Freon-12 loop (KRISTA) was built at the Research Center Karlsruhe (FZK). A program of experimental investigations on CHF behavior in tight rod lattices has been performed using circular tubes, 7-rod bundles and 37-rod bundles (Cheng 1991, Erbacher et al. 1994). The KRISTA test facility, shown schematically in figure 3, was constructed for pressures up to 3.5 MPa, mass flow rates up to 18 kg/s and electrical powers up to 800 kW. The KRISTA test facility consists of two separate loops which are interconnected via the Freon storage tank and the purification system. A refrigeration system connected to the heat exchangers provides a wide range of inlet temperature down to –15°C.

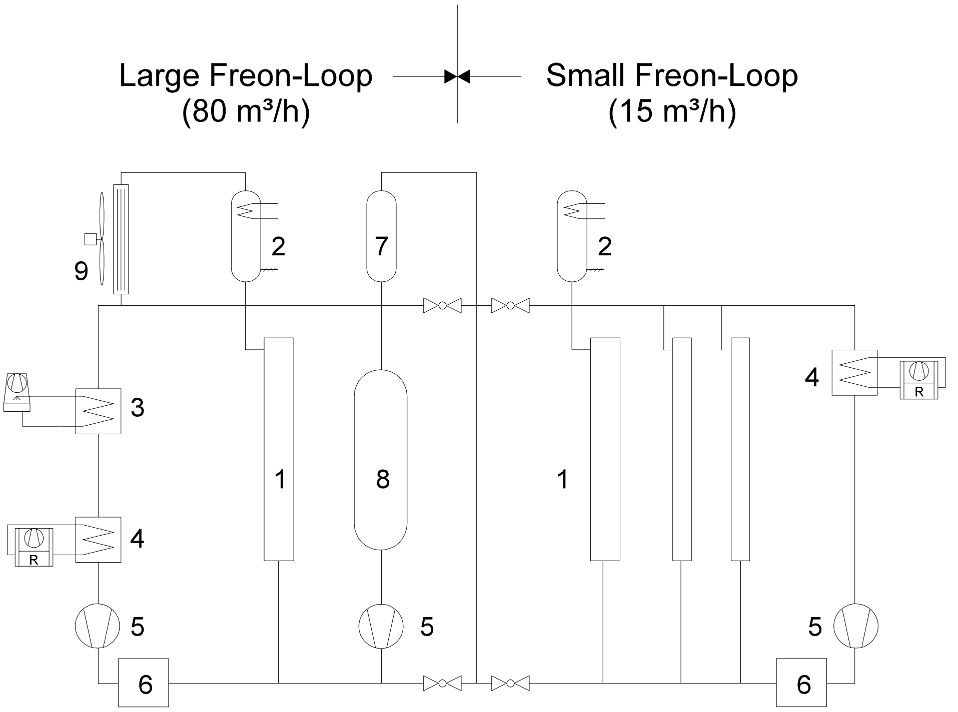


Figure 3: Schematic diagram of the KRISTA test facility at FZK
1 test section; 2 pressurizer; 3 heat exchanger (cooling tower); 4 heat exchanger (refrigerator); 5 pump; 6 preheater; 7 purification; 8 Freon tank; 9 air condenser

3.2 Data base

The experimental data available can be divided into different groups according to flow channel geometries (circular tubes, rod bundles) and to working fluids (water, Freon).

(a) circular tubes

The circular tube is the most used geometry. There exist many data banks summarizing the experimental data in circular tubes and in *water*. Two of the earlier data banks were compiled by Thomson et al. (1964) and Becker et al. (1971). Table 4 summarizes the parameter ranges of these two data banks. Thompson et al. (1964) compiled all the world CHF data available at that time for vertical, uniformly heated round tubes with sub-cooled inlet conditions. The data bank of Becker et al. (1971) contains about 3000 data points achieved in 6 different diameter tubes. More than 1500 data points were obtained in tubes of 10 mm diameter with a systematic change of hydraulic parameters.

Table 4: parameter ranges of the data banks

	pressure [MPa]	steam quality [-]	mass flux [Mg/m²s]	diameter [mm]	data points
Thomson, et al. (1964)	0.1 – 20.0	---	0.01 - 18.5	1.0 – 37.5	~ 4400
Becker, et al. (1971)	2.7 – 20.0	-0.3 – 1.0	0.12 – 7.6	3.9 – 25.0	~ 3000

Efforts have been made at many other institutions to compile CHF data banks. Some of the data banks, e.g. those complied in the former Soviet Union (Doroshchuk 1975, Kirillov 1993), are not published and, therefore, not available in the open literature. There exist also data banks for specific purposes, e.g. transient CHF (Celata 1991), CHF in fusion engineering (Boyd 1985a). In the 80‘s two big data banks were derived independently by Groeneveld et al. (1986) and by Kirillov et al. (1989). Both data banks contain more than 15000 data points and cover a wide range of parameters. More recently, a new data bank has been derived by an international working group. This data bank contains more than 30000 data points and covers the following parameter ranges (Groeneveld et al. 1999):

- pressure [MPa]: 0.1 – 21.2
- mass flux [Mg/m²s]: 0.006 – 24.27
- steam quality [-]: -1.65 – 1.57
- diameter [mm]: 0.62 – 92.4

This data bank has been deposited in the International Nuclear Safety Center Data Base at Argonne National Laboratory (IAEA 1998). However, it has to be pointed out that although each test parameter covers a wide range, the parameter matrix is not filled completely. There are still deficiencies in test data at the following parameter combinations: low pressure/low mass flux, low steam quality/low mass flux; high steam quality/high mass flux.

Although a lot of data points using Freon were published, little effort was made to compile CHF data banks. Table 5 summarizes some experiments carried out in Freon-12.

Table 5: Test data obtained in Freon-12

	pressure [MPa]	mass flux [Mg/m²s]	steam quality [-]	diameter [mm]	data points
Stevens (1964)	1.05	0.5 – 4.2	≥ -0.05	11.5 – 16.0	~ 800
Merilo (1979)	1.0 – 1.5	1.6 – 8.1	0.0 – 0.6	5.3 – 12.6	~ 800
Müller-Menzel (1987)	1.0 – 2.7	1.0 – 6.0	---	2.6, 4.6	~ 900
Katto (1987)	1.0 – 3.5	0.5 – 10.4	-0.5 – 1.0	3.0 – 8.0	~ 3000
Souyri (1993)	1.0 – 3.5	0.7 – 5.4	-0.25 – 0.80	13.0	~ 500
Cheng (1997)	1.0 – 3.0	1.0 – 6.0	-0.75 – 0.6	2.0 – 16.0	~ 1700

The early experiments of Stevens (1964) and Merilo et al. (1979) are restricted to a very narrow range of test parameters with the main purpose of studying the fluid-to-fluid scaling laws. Systematic studies with wide range of parameters have been performed by Katto et al. (1987) and Cheng et al. (1997). One of the main purposes of Katto's work is to provide a large data base for developing CHF prediction methods. The test data of Cheng (1997) was obtained in tubes of 5 different diameters (2, 4, 8, 12 16 mm) covering the range of the hydraulic diameter of the fuel assembly of water cooled reactors.

Figure 4 shows an example of the test results indicating the effect of different parameters on CHF. It is seen that the critical heat flux decreases with increasing steam quality. The relationship between CHF and the exit steam quality can be approximately described by a linear function, as assumed in most empirical correlations. At low steam qualities an increasing mass flux strengthens turbulence, improves bubble transport from the heated wall and leads to a higher CHF. The pressure influence on CHF is a consequence of the fluid properties: a higher pressure leads to 1) a lower value of latent heat and a higher rate of evaporation which in turn leads to a lower CHF, 2) a higher vapor density which tends to reduce the void fraction and increase the value of CHF, and 3) a lower surface tension which in consequence leads to a lower value of CHF. Thus the influence of pressure on CHF depends on which of the factors prevail. The test results show that at low steam qualities an increasing pressure leads to a lower CHF. In this region the effects of lower latent heat and lower surface tension prevail over the effect of higher vapor density. At high steam qualities an increase in the CHF is observed by increasing pressure. In this case, the effect of increasing steam density dominates over other effects.

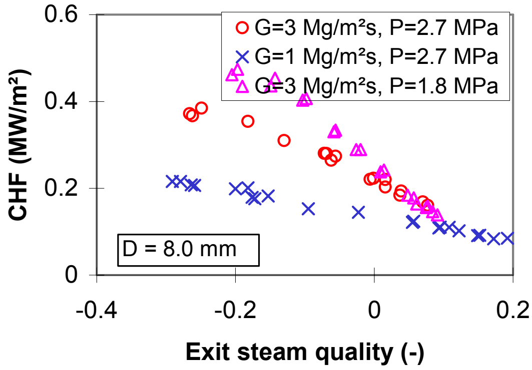


Figure 4: Example of test results obtained in Freon-12 (Cheng 1997)
P – pressure, G – mass flux, D – tube diameter

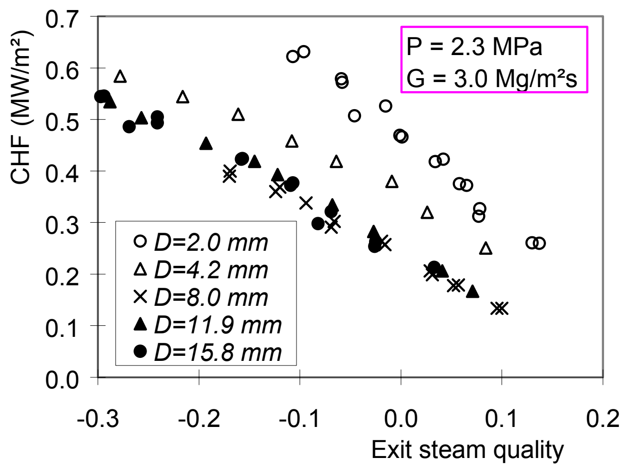


Figure 5: Effect of the tube diameter on CHF

Figure 5 shows a test example indicating the effect of the tube diameter on CHF. It can be seen that for the condition considered CHF decreases with increasing tube diameter from 2 mm up to 8 mm. By changing the tube diameter from 8 mm to

16 mm, the CHF values remain nearly unchanged. More details about the effect of the tube diameter on CHF can be found in other papers (Levitan 1980, Müller-Menzel et al. 1987, Celata et al. 1993, Cheng et al. 1997). It is concluded that the effect of the tube diameter on CHF is governed not only by thermal-hydraulic, but also by geometric conditions and can not be sufficiently described by the empirical methods available in the literature. Further research work is needed to study the influence of tube diameter.

(b) <u>rod bundles</u>

A large number of CHF experiments in rod bundles have been performed both in modeling systems and in full-scale simulation of fuel bundles of water cooled reactors. Because of their proprietory nature some of the test data are not available for public use. In early 60‘s Macbeth (1964) compiled a CHF data bank for water flow in rod bundles. This data bank contains the experimental data of (1) General Electric in 3-, 4-, 9- and 19-rod bundles, (2) Columbia University in 7-, 12- and 19 rod bundles and (3) UKAEA in 7- and 19-rod bundles. Hughes (1974) provided a compilation of bundle CHF data containing more than 4200 data points obtained in 126 bundle test sections. Some years later the Heat Transfer Research Facility of Columbia University collected over 11000 data points from 235 separate test sections (Fighetti et al. 1982). This is the largest data bank of this type in the world. These data were obtained for U.S. vendors as well as for British, Canadian, Japanese and German reactor designers. They cover a wide range of parameters of conventional PWR, BWR and PHWR fuel lattices. However, most of these bundle CHF data are proprietary information and so far have remained unpublished.

With regard to the VVER bundle geometry CHF experiments have been performed in Eastern Europe and in the former Soviet Union. In addition to some efforts made by scientists from the former Soviet Union (Bobkov 1997), a data bank for the VVER bundle geometry has been generated by Czech scientists (Cizek et al. 1997). This data bank contains more than 2600 data points from 27 hexagonal test bundles at the following conditions:

Pressure, MPa:	2.17 – 18.73,
Mass flux, Mg/m²s:	0.2 – 4.2,
Local quality, -:	-0.26 – 1.0,
Rod diameter, mm:	8.99 – 9.15,
Pitch to diameter ratio. -:	1.34 – 1.44.

In the framework of developing a high conversion PWR, experiments in semi-tight hexagonal 19- and 37 rod bundles have been performed at Siemens (Bethke 1992) and at CEA (Courtaud 1988) covering the parameter ranges summarized in Table 6.

Table 6: Parameter ranges of the CHF experiments performed at Siemens and at CEA

	Siemens	CEA
Rod diameter [mm]	9.0	8.65
Pitch to diameter ratio[-]	1.178	1.15
Pressure [MPa]	7.0 – 16.0	7.6 – 16.0
Mass flux [Mg/m²s]	2.0 – 6.0	2.0 – 8.0

As in circular tubes a lot of CHF experiments in rod bundles were carried out using Freon-12. Some of them are summarized in table 7. Herein D_h stands for the hydraulic diameter of the rod bundle.

Table 7: Rod bundle CHF experiments carried out in Freon-12

	lattice	pitch/d	D_h mm	pressure MPa	mass flux Mg/m²s	No. of rods
Stevens (1966)	Hex.	18.0/15.8	~ 6.8	1.07	0.2 – 1.4	19
McPherson (1971)	Hex.	20.7/19.7	~ 6.7	0.8 – 13.0	0.18 – 5.4	19
Fulfs (1976, 1980)	Square	14.3/10.7	~ 14.0	1.1 – 3.7	0.6 – 3.6	25 - 64
Courtaud (1988)	Hex.	9.96/8.65	~ 4.0	1.2 – 2.9	2.0 – 9.0	19
Cheng (1998a, 1998b)	Hex.	1.15, 1.18	~ 4.3	1.0 – 3.0	1.0 – 6.0	7, 37

In the experiments of McPherson (1971) and Stevens (1966) 19-rod bundles simulating CANDU fuel elements were used. Tests were carried out under low pressures with the main purpose of studying the fluid-to-fluid scaling law. The experiments at GKSS in 5x5, 6x6, 7x7 and 8x8 square rod bundles served also to assess fluid-to-fluid scaling laws and CHF correlations for PWR fuel element geometries (Fulfs et al. 1976, 1980). The corresponding tests in water were conducted at Siemens (Ulrych 1976). Tests in tight or semi-tight hexagonal rod bundles were performed by Cheng et al. (1998a, 1998b) and by Courtaud et al. (1988) for investigating CHF behavior in a high conversion PWR. Corresponding water tests in geometrically identical 19- and 37-rod bundles are also available (Bethke 1992, Courtaud et al 1988). Based on the test data of both fluids, fluid-to-fluid scaling laws for tight hexagonal rod bundles were assessed (Cheng 1998b) and developed (Courtaud 1988).

Analysis of some test data available shows that the effect of steam quality, pressure and mass flux on CHF in rod bundles is similar to that in tubes (Cheng 1997, 1998a). Intensive studies have been performed on the effect of spacing devices on CHF. In general, a significant increase in local CHF was observed just downstream of a grid spacer, as illustrated in figure 6. The increase is primarily due to the higher turbulence level of the two-phase flow, which can strongly suppress the occurrence of CHF, and the improved inter-channel mixing. The effect of wire wraps on CHF was studied by Cheng (1998a). Experiments were conducted using two different rod bundles. In the first one grid spacers were used with large spacing distance (~60 times of the hydraulic diameter). In this case the effect of the spacer on CHF is negligible small. In the second bundle wire wraps were utilized. Figure 7 compares the CHF values obtained in the bundle with wire wraps with that in the bundle with grid spacers. It is evident that the slope of the curve valid for the bundle with wire wraps is steeper than that for the bundle with grid spacers. Both curves cross over at an exit vapor quality close to zero. For low vapor qualities the CHF of the bundle with wire wraps is higher compared to that of the bundle with grid spacers. An opposite influence of wire wraps on CHF is found for high vapor qualities. These results were explained by the influence of wire wraps on local flow conditions near the wall in connection with the flow-regime transition. At low vapor qualities with bubbly flow, wire wraps tend to enhance the bubble transport from the heated wall and result in a higher CHF. At higher vapor qualities with annular flow, wire wraps tend to destroy the liquid film on the heated surface and accelerate the dryout of the heated surface.

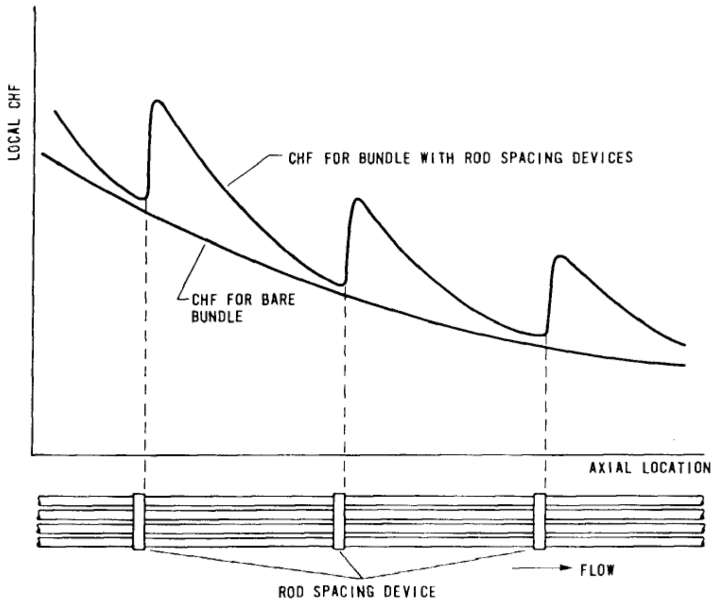


Figure 6: effect of grid spacers on CHF (Groeneveld et al. 1999)

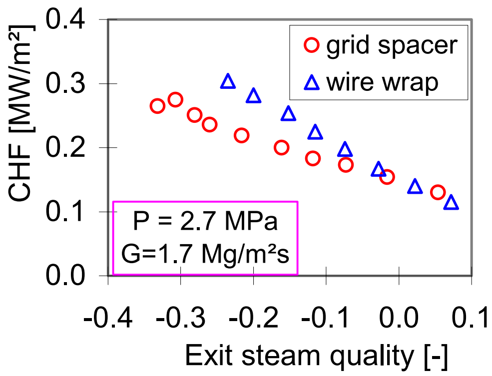


Figure 7: Effect of wire wraps on CHF

4. Prediction methods

In rod bundle geometries CHF results can be presented either by the sub-channel flow condition or by the bundle average condition. For a better understanding of the physical mechanisms of boiling crisis the use of the sub-channel condition is a more favorable method, especially in small rod bundles where strong thermal imbalance appears. Most of the CHF prediction methods available in the open literature are based on the sub-channel condition and require the use of a sub-channel analysis code, which usually contains closure equations. A sensitivity study pointed out that the empirical models describing inter-channel mixing affect strongly the calculated sub-channel conditions. Figure 8 shows the effect of the turbulent mixing coefficient on the local exit steam quality of the hot sub-channel in a 37 rod bundle. A strong effect of the turbulent mixing coefficient on the calculated sub-channel conditions is observed. In spite of intensive studies carried out worldwide there exist still deficiencies in analyzing sub-channel flow parameters under high pressure two-phase flow conditions (Wolf 1987). Further works are underway at many institutions (Okubo 1997). It has to be kept in mind that the accuracy of a CHF prediction method in the rod bundle geometry is coupled with the accuracy of determining sub-channel flow conditions.

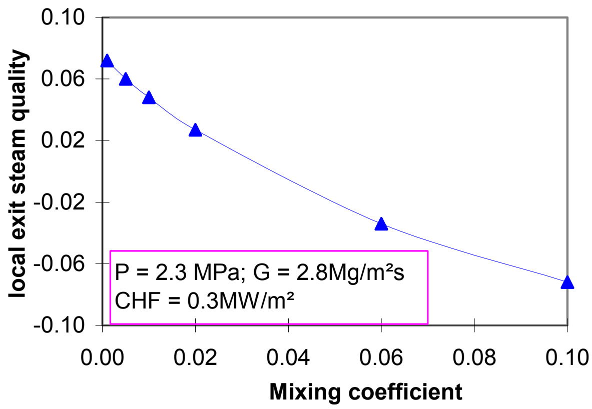


Figure 8: effect of the mixing coefficient on the exit steam quality of the hot sub-channel

4.1 Fluid-to-fluid scaling

Fluid modeling is widely used in thermal-hydraulic design of nuclear reactors. One of the main tasks concerning the fluid modeling is the transfer of the test data obtained in the model fluid to the prototypical fluid (water). An analysis shows that by neglecting geometric scaling and the effect of heat transfer surface properties on CHF, there exist still more than 10 independent dimensionless numbers, which describe the boiling crisis under forced flow conditions (Cheng 1991). It is obviously impossible to achieve a complete similarity using different fluids, because of the limiting degree of freedom available for experimentation. During the CHF experiments four thermal-hydraulic parameters can normally be adjusted, i.e. pressure, mass flux, steam quality and heat flux. It is commonly agreed that the scaling of pressure, steam quality and heat flux should obey the following equations:

$$\left(\frac{\rho_f}{\rho_g}\right)_M \Bigg/ \left(\frac{\rho_f}{\rho_g}\right)_P = 1$$

$$X_M / X_P = 1$$

$$\left(\frac{CHF}{G \cdot h_{fg}}\right)_M \Bigg/ \left(\frac{CHF}{G \cdot h_{fg}}\right)_P = 1$$

Where ρ_f and ρ_g are the density of the liquid phase and the vapor phase. The subscripts *'M'* and *'P'* stand for model and prototype, respectively. The difference between the fluid-to-fluid scaling laws is the derivation of the scaling factor of mass flux, which is defined as the ratio of the mass flux of the original fluid (water) to that of the model fluid (Freon-12). Based on CHF test data obtained in seven different fluids Groeneveld et al. (1992) pointed out that the Weber number is a reliable parameter for determining the scaling factor of mass flux, i.e.

$$\left(\frac{G^2 \cdot D}{\sigma \cdot \rho_f}\right)_M \Big/ \left(\frac{G^2 \cdot D}{\sigma \cdot \rho_f}\right)_P = 1 \ ,$$

where σ is the surface tension. A thorough assessment of the existing scaling laws was also made by Katsaounis (1986) and Cheng (1991). It was concluded that the fluid-to-fluid scaling law of Ahmad (1973) is well suited for fluid-to-fluid modeling in circular tubes and for square type wide rod bundles. According to the Ahmad's scaling law the scaling factor of mass flux is determined by the following equation:

$$\left\{\left(\frac{G^2 \cdot D}{\sigma \cdot \rho_f}\right)^{2/3}\left(\frac{G \cdot D}{\mu_f}\right)^{-2/15}\left(\frac{G \cdot D}{\mu_g}\right)^{-1/5}\right\}_M \Big/ \left\{\left(\frac{G^2 \cdot D}{\sigma \cdot \rho_f}\right)^{2/3}\left(\frac{G \cdot D}{\mu_f}\right)^{-2/15}\left(\frac{G \cdot D}{\mu_g}\right)^{-1/5}\right\}_P = 1$$

Where μ_f and μ_g are the dynamic viscosity of the liquid phase and the vapor phase, respectively. Figure 9 compares the CHF data obtained in a 8 mm tube and in Freon-12 with the look-up table of Groeneveld et al. (1996). The water table data are transferred to Freon-12 equivalent conditions by using the Ahmad's scaling law. The good agreement shows that the test results obtained in Freon-12 can be well transferred to water conditions by using the Ahmad's scaling law.

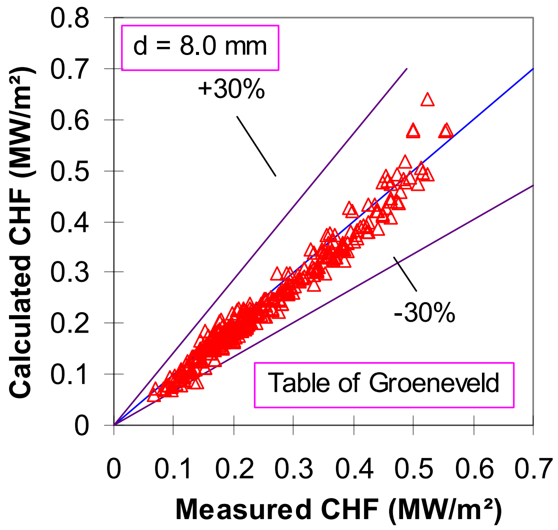


Figure 9: comparison of the Freon-12 data with the look-up table

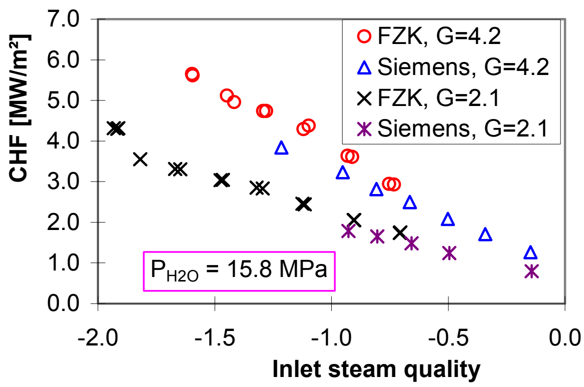


Figure 10: comparison of the bundle data in Freon-12 with that in water

The application of some scaling laws to tight hexagonal lattices was assessed by Cheng et al. (1998b). Figure 10 compares the test data in two geometrically identical 37-rod bundles with different fluids. The experiments in water were conducted by Siemens (Bethke 1992) and the test data in Freon-12 were obtained at FZK (Cheng 1998b). For comparison the test data in Freon-12 are transferred to water conditions by using the scaling law of Ahmad. It is found that the CHF values of water are higher than that of Freon-12 transferred to water equivalent conditions.

4.2 Empirical correlations

Nowadays, there exist a large number of empirical correlations that have been developed by correlating available CHF data base obtained from particular flow channel geometries and parameter ranges. For a uniformly heated circular tube the number of empirical correlations exceeds 400, all of which can be divided into two different types; i.e. fluid specific and fluid independent correlations. In the earlier years correlations were derived mostly for water conditions. Two of them are:

- W-3 correlation (Tong 1967)

$$
\begin{aligned}
CHF = 3.1546 \cdot & \left[(2.022 - 0.0624 \cdot P) + (0.172 - 0.0143 \cdot P) \cdot e^{(18.177 - 0.599 p) X}\right] \\
& \left[(0.109 - 1.173 \cdot X + 0.127 \cdot X \cdot |X|) \cdot G + 1.037\right] \cdot \left[1.157 - 0.869 \cdot X\right] \\
& \left[0.266 + 0.836 \cdot e^{-0.124 D}\right] \cdot \left[0.826 + 0.341 \cdot 10^{-6} \cdot (h_{sat} - h_{in})\right]
\end{aligned}
$$

- correlation of Levitan (Levitan 1975)

$$
CHF = \left[10.3 - 17.5 \cdot \frac{P}{P_C} + 8.0 \cdot \left(\frac{P}{P_C}\right)^2\right] \cdot G^{\left(0.68 \cdot \frac{P}{P_C} - 1.2 \cdot X - 0.3\right)} \cdot e^{-1.5 \cdot X} \cdot \left(\frac{D}{8.0}\right)^{-0.5}
$$

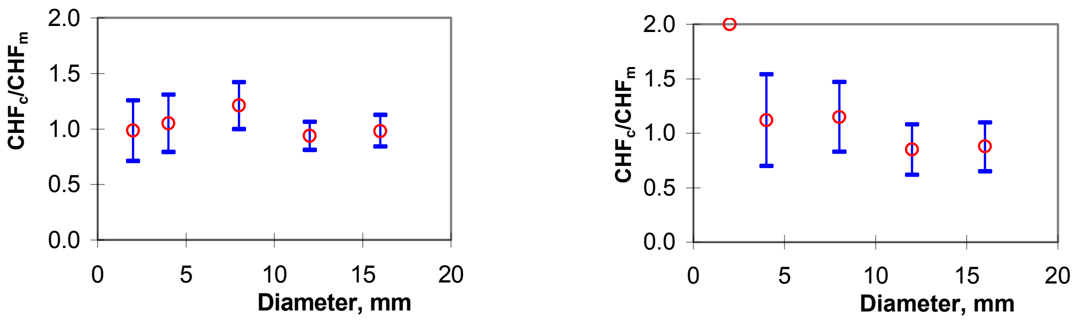
The CHF is given in MW/m². The dimensions and the valid ranges of the parameters are summarized in table 8.

Table 8: Dimensions and valid ranges of different parameters

	W-3 correlation	Levitan’s correlation
Pressure P, [MPa]	6.9 – 15.9	2.94 – 19.6
Mass flux G, [Mg/m²s]	1.36 – 6.78	0.75 – 5.0
Local steam quality X, [-]	-0.15 – 0.15	-75°C to boundary steam quality
Diameter D, [mm]	5.0 – 17.8	---

The W-3 correlation was widely used in western countries, whereas the correlation of Levitan has found a wide application in the former Soviet Union.

With the use of model fluids more and more efforts have been made to develop fluid independent correlations, of which the correlation of Katto (1984) and Shah (1987) are widely



(a) Correlation of Katto et al. (1984) *(b) Correlation of Shah (1987)*

Figure 11: Comparison of correlations with CHF data in Freon-12 and in circular tubes

Both correlations utilize dimensionless numbers and can be applied to different fluids and to extremely wide parameter ranges. Figure 11 compares both the correlations with test data obtained in Freon-12 of different diameter tubes (Cheng 1997). The circle indicates the average value and the lines stand for the standard deviation of the ratio of the calculated to the measured CHF values. The subscripts *'c'* and *'m'* stand for calculation and measurement, respectively. The comparison bases on the direct substitution method. A good agreement between the Shah's correlation and the test data is obtained. Except for the data in the very small diameter tube (D = 2 mm) the CHF test results are reasonably reproduced by the Katto's correlation.

As mentioned above, experimental investigations on CHF have to be performed for each specific design of fuel elements and operating conditions. Validated prediction methods for a specific design condition must be derived. Obviously, many correlations of rod bundle CHF are in existence. However, most of them are of proprietory nature and not available in the open literature. Table 9 gives the valid parameter ranges of three CHF correlations selected. All three correlations are based on sub-channel flow conditions. The EPRI-1 correlation (Fighetti et al. 1983) has been developed by using more than 3600 test points from 65 test sections simulating square PWR & BWR fuel assemblies. The effect of grid spacers on CHF was quantified in terms of the grid loss coefficient. The correlation was also verified to non-uniform axial heat flux distribution. To obtain the sub-channel flow parameters the sub-channel Code COBRA-III was used. The Winfrith Sub-channel Correlation WSC-2 (Bowring 1973) was developed based on more than 1000 bundle data from 54 clusters simulating fuel assemblies of PWR, BWR and PHWR. The sub-channel flow conditions were determined by using the HAMBO code. It consists of three equations, namely for (1) the inner rodded square sub-channels, (2) the inner rodded triangular sub-channels and (3) the outer sub-channels containing unheated walls. The test data from a hexagonal lattice were obtained at a pressure lower than 9 MPa and a steam quality higher than +6%. The KfK-3 correlation was mainly developed for tight hexagonal lattices. It consists of two different sets of equations valid for gridded bundles and for wire wrapped bundles, respectively.

Table 9: Bundle CHF correlations and their valid parameter ranges

	D_h, mm	P, MPa	G, Mg/m²s	X
EPRI-1 (Fighetti 1983)	8.9 – 14.0	0.4 - 17.0	0.3 - 5.5	-0.25 - 0.75
WSC-2 (Bowring 1976)	5.1 – 30.0	3.4 – 15.9	0.27 – 5.0	-0.2 – 0.86
KfK-3 (Dalle-Donne 1991)	2.3 – 6.6	2.7 - 14.0	≤ 5.4	-0.44 - 0.96

Assessment of bundle correlations was made by many researchers, e.g. Cuta (1983), Yoder (1985), Cheng (1998a).

4.3 Look-up tables

One of the attempts to standardize empirical correlations is the CHF look-up table method. Compared to other available prediction methods, the table approach has a higher accuracy, wider range of application and can be easily upgraded. The representative work was done by the scientists of the former Soviet Union (Doroshchuk 1975). The table approach has been continued at the Institute of

Physics and Power Engineering in Russia (Kirillov et al. 1989, 1995), at the Chalk River Nuclear Laboratory and at the University of Ottawa in Canada (Groeneveld 1986). Most recently, a new CHF look-up table has been proposed based on an international CHF data bank containing more than 30000 data points. The table is valid for pressure from 0.1 – 20 MPa, mass flux from 0 to 8 Mg/m²s and local steam quality from –0.5 to 1.0. The CHF look-up table was derived by statistically averaging CHF data points within each interval of parameters. At conditions where experimental data were not available, CHF points were obtained by extrapolations, using known trends. The look-up tables have been successfully assessed by many researchers, e.g. Becker et al. (1992), Baek et al. (1996).

It has been strongly recommended by Groeneveld et al. (1999) to apply the tube look-up table to rod bundle geometry on the basis of sub-channel conditions. Several correction factors are introduced to modify the tube CHF to bundle conditions to account for bundle-specific and sub-channel specific effects:

$$CHF_{bundle} = CHF_{table} \times K_1 \times K_2 \times K_3 \times K_4 \times K_5 \times K_6 \times K_7 \times K_8$$

K_1 sub-channel or tube-diameter cross-section geometry factor,
K_2 bundle-geometry factor,
K_3 spacer factor,
K_4 heated-length factor,
K_5 axial flux distribution factor,
K_6 radial or circumferential flux distribution factor,
K_7 flow-orientation factor,
K_8 vertical low-flow factor.

The eight correction factors are described by Groeneveld et al. (1999). Clearly, the accuracy of the look-up table for the bundle condition depends on the accuracy of the correction factors, which have to be determined empirically. The application of the tube look-up tables to rod bundles has been assessed by Ulrych (1985, 1993), Bethke et al. (1993), Herer (1993) and Cevolani (1997).

For the VVER hexagonal rod bundle a CHF look-up table was recently derived at IPPE (Bobkov 1997b). This look-up table is based on more than 4000 CHF data in 22 VVER hexagonal bundles and valid for the following conditions:

Rod diameter:	9.36 mm,
Pitch to diameter ratio:	1.40,
Pressure:	0.1 – 20.0 MPa,
Mass flux:	0.05 – 5.0 Mg/m²s,
Local steam quality:	-0.5 – 0.9.

The application of the table can be extended to a wide range of parameters by introducing correction factors:

$$CHF_{bundle} = CHF_{table} \times K_1 \times K_2 \times K_3 \times K_4 \times K_5 \times K_6$$

These correction factors take different effects into account and are summarized in table 10:

Table 10: Correction factors and their valid ranges

Factor	Effect	Valid range
K_1	hydraulic diameter	2.8 – 21 mm
K_2	pitch to diameter ratio	1.02 – 1.52
K_3	heated length to diameter ratio	40 – 300
K_4	grid spacer	
K_5	axial heat flux distribution	
K_6	radial heat flux distribution	

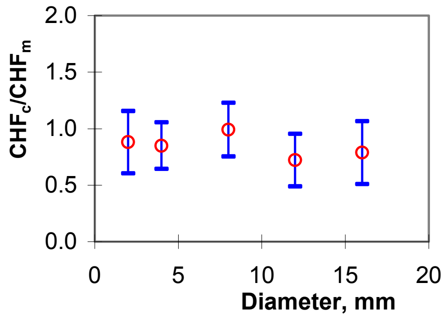
Based on an analysis of some CHF data in the fuel lattice of PWR and VVER reactors, Kirillov (1997a) pointed out that there is no significant difference between the CHF values in square rod bundles and in hexagonal lattices. The look-up table developed for hexagonal lattices could also be applied to square rod bundles.

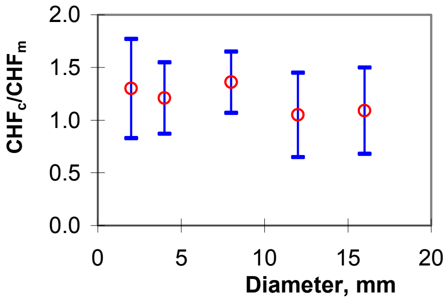
4.4 Phenomenological models

Extensive research efforts have been made on modeling the relevant physical processes leading to critical heat flux. However, due to the difficulty in performing detailed flow visualization of the near wall region at high heat fluxes approaching CHF, many of the published CHF models have been based on postulated mechanisms and not verified through direct observations. For CHF in forced convective channel flow, two different types of boiling crisis are considered. In the subcooled or low steam quality region a boiling crisis occurs by the transition from nucleate boiling to film boiling, or departure from nucleate boiling (DNB). In the higher steam quality region, mostly in annular flow, the boiling crisis originates from a depletion of the liquid film (dryout). In the fuel assembly of PWR, PHWR and VVER reactors the first kind of boiling crisis (DNB) is mostly expected because of low steam quality in sub-channels. In the fuel assembly of a BWR attention is paid to the boiling crisis of the second kind (dryout).

The situation is confusing concerning the boiling crisis mechanism in the low steam quality region (DNB). Different mechanisms have been postulated. The 'boundary layer separation model' proposed by Kutateladze et.al (1966) postulates that CHF is caused by the flow stagnation due to injection of vapor from the heated wall. Weisman & Pei (1983) proposed the so-called 'near wall bubble crowding model'. Turbulent interchange between the bubble layer and the core region is the limiting mechanism for the onset of CHF for this case. They postulated that CHF occurs when the void fraction in the bubble layer exceeds the critical value at which an array of ellipsoidal bubbles can be maintained without significant contact between the bubbles. Lee and Mudawar (1988) proposed the 'liquid sublayer dryout model'. According to this model there exists a thin liquid sublayer beneath a vapor blanket flowing over the heated surface. Dryout of the thin liquid sublayer occurs when the rate of sublayer mass loss by evaporation exceeds that of liquid entering the sublayer from the core region. Based on the liquid sublayer dryout concept different prediction models have been proposed by Katto et al. (1990), Celata et al. (1994, 1998), Lin et al. (1996), Liu et al. (1999). The major difference between these models is the calculation of the microlayer thickness and the vapor blanket size. Brief reviews of CHF models have been provided by Tong et al. (1972), Weisman (1992), Katto

(1994) and Bricard et al. (1995). Some phenomenological models have been assessed by Celata et al. (1994), Bricade et al. (1995) and Cheng et al. (1997).





(a) Model of Weisman & Pei (1983) *(b) Model of Lee & Mudawar (1987)*

Figure 12: Comparison of test data with CHF models

Figure 12 compares the test data obtained in circular tubes and in Freon-12 with the 'near wall bubble crowding model' of Weisman (1983) and the 'liquid sublayer dryout model of Lee (1988). The comparison is based on direct substitution method. A satisfying agreement between the test data and the CHF models is observed. On the average, the Weisman's model underpredicts the experimental data, whereas the model of Lee overpredicts the experimental results.

All of the models mentioned above were originally developed for circular tube geometries. Efforts were also made to extend them to rod bundle geometries (Weisman 1985, Lin 1989). Generally, it is assumed that the sub-channels in rod bundles are similar to a circular tube so that similar thermal-hydraulic behaviour may be expected. To apply a circular tube approach to a rod bundle, it is necessary to determine the local conditions through the use of sub-channel analysis codes. Weisman et al. (1986) and Lim et al. (1990) extended their models to account for the effect of mixing vanes of various types and the effect of unheated walls. A satisfying agreement between the models and the bundle CHF test data was achieved.

In comparison to DNB the processes of dryout are well understood. The flow rate of the liquid film is determined by evaporation, droplet entrainment and droplet deposition. CHF corresponds to the condition in which the film flow rate falls to zero. The dryout process was first described by Whalley et al. (1974) by means of the so-called ‘annular flow model‘. This model was extended by Hewitt et al. (1990). A good agreement between the model and the test data from uniformly heated tubes was obtained. The annular flow model was also successfully assessed by many other authors, e.g. Azzopardi (1995).

Whalley (1977) extended the dryout modeling to rod bundles. Additional terms of mass balance have been introduced to take into account inter-subchannel exchange of mass by means of turbulent two-phase mixing and cross flow. The effort, to apply dryout models to rod bundles, was also made by many other authors (Saito 1978, Mitsutake 1990). All the recent models have produced reasonable prediction of bundle test results. However, it has to be kept in mind that difficulties still remain with regard to an accurate determination of the conditions for the onset of annular flow and in reliably modeling the entrainment rate and the deposition rate.

5. Conclusions

Design of a water cooled reactor requires a sufficient safety margin to the critical heat flux. Due to the incomplete knowledge of boiling crisis mechanisms, especially burnout, experimental investigations on CHF have to be performed for each specific design of nuclear reactors. Validated prediction methods for the design condition must be derived. This leads to a large amount of CHF data banks and CHF prediction methods available in the literature. This paper gives a brief overview of experimental and theoretical investigations on CHF with the emphasis on its application to water cooled reactors under normal operating conditions. The following main conclusions can be drawn:

- Geometry and fluid modeling approaches are inevitable in order to reduce technical complexity and financial expense of the CHF experiments.
- A large amount of test data in circular tubes is available. Scaling up of these data to rod bundles requires an accurate description of the effect of the geometric parameters on CHF. This is still not reliably predictable.
- Fluid-to-fluid scaling laws have been successfully applied and assessed to CHF in circular tubes and in wide square rod bundles. Nevertheless, further works are necessary to assess the fluid-to-fluid scaling in tight rod bundles and to take into

account the effect of spacers and heat flux distributions. The applicability of the fluid-to-fluid scaling to the new replacement fluid, e.g. Freon-134a, should be verified for different conditions.

- Empirical correlations are the mostly used CHF prediction methods in rod bundles. A successful application of the look-up table approach to rod bundles requires (1) an accurate determination of sub-channel flow conditions and (2) a reliable description of the bundle specific effects on CHF, e.g. spacer effect. Research works in this field are underway at many institutions.
- One of the key tasks in future CHF investigations is an improved understanding of the physical mechanism of the boiling crisis. Visualisation experiments are highly desirable with suitable measurement techniques to observe and to determine the boiling crisis processes at different ranges of parameters.

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