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EVALUATION DU FLUX CRITIQUE EN REACTEUR

CRITICAL HEAT FLUX EVALUATION

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SYNTHÈSE :

L'évaluation du flux critique est important pour la sûreté nucléaire et représente l'un des facteurs limitatifs pour le fonctionnement des coeurs des réacteurs. Le flux critique résulte d'une forte diminution de l'échange de chaleur à niveau des crayons combustibles. La sûreté exige que ce phénomène également nommé crise d'ébullition soit évité dans des conditions nominales ou accidentelles (événéments de Classe I et II).

Une évaluation du flux de chaleur critique dans le coeur des réacteurs résulte principalement d'une démarche en deux étapes. On teste tout d'abord les assemblages de crayons dans des boucles expérimentales pour déterminer les limites du flux de chaleur critique. Puis, on effectue des calculs thermohydrauliques pour évaluer le niveau de sûreté.

Cette note précise le phénomène de crise d'ébullition et souligne la complexité ainsi que le manque de compréhension fondamentale dans le domaine. Une description des sections d'essai expérimentales nécessaires à la collecte des données est présentée. Puis on analyse les méthodes d'évaluation des marges de sûreté du flux de chaleur critique dans le coeur des réacteurs. La dernière partie de la note est consacrée aux thèmes de R&D actuellement traités sur la connaissance de la crise d'ébulition, la prédiction du flux critique ainsi que les codes de thermohydrauliques.

EXECUTIVE SUMMARY :

Critical heat flux (CHF) is of importance for nuclear safety and represents of the major limiting factors for reactor cores. Critical heat flux is caused by a sharp reduction in the heat transfer coefficient located at the outer surface of fuel rods. Safety requires that this phenomenon also called the boiling crisis should be precluded under nominal or incidental conditions (Class I & II events).

CHF evaluation in reactor cores is basically a two-step approach. Fuel assemblies are first tested in experimental loops in order to determine CHF limits under various flow conditions. Then, core thermal-hydraulic calculations are performed for safety evaluation.

The paper will go into more details about the boiling crisis in order to pinpoint complexity and lack of fundamental understanding in many areas. Experimental test sections needed to collect data over wide thermal-hydraulic and geometric ranges are described. CHF safety margin evaluation in reactors cores is discussed by presenting how uncertainties are accounted for. In the last section of the paper, areas of research and improvement are mentioned. From basic considerations to current concerns, the following topics are discussed : knowledge of the boiling crisis, CHF predictors, and advanced thermal-hydraulic codes.

CRITICAL HEAT FLUX EVALUATION

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I- Physics of the boiling crisis and CHF predictions

This section provides information on the two most important CHF mechanisms for reactor core thermal analysis (DNB and dryout). Mechanisms leading to the boiling crisis are investigated. Then, special attention is focused on technological parameters. These parameters are emphasized in order to present complexity and importance for standard safety analysis. Experimental CHF facilities are described as well as correlations needed for CHF evaluation.

I.I- Physics of the boiling crisis

Boiling crisis is a general term to describe a sudden reduction in the heat transfer coefficient between a heated surface and a cooling fluid. This variation in heat transfer properties may lead to a sharp temperature excursion of two or three orders of magnitude that can cause rod failure. Obviously, onset of the boiling crisis has to be avoided in order not to damage the first safety barrier (fuel cladding) that contains radioactive materials.

Several terms are related to the boiling crisis. First of all, the heat flux that leads to the boiling crisis is the critical heat flux. CHF experiments may be referred to as DNB (Departure from Nucleate Boiling), dryout or burn-out.

The boiling crisis is a two-phase flow phenomenon. The thermal-hydraulic parameter that influences the CHF the most is void fraction. For low void fractions, the boiling crisis is referred to as DNB, whereas for higher void fractions, the term is dryout. Burn-out refers to CHF consequences since early tests detected the crisis by the physical failure of electrically heated test elements.

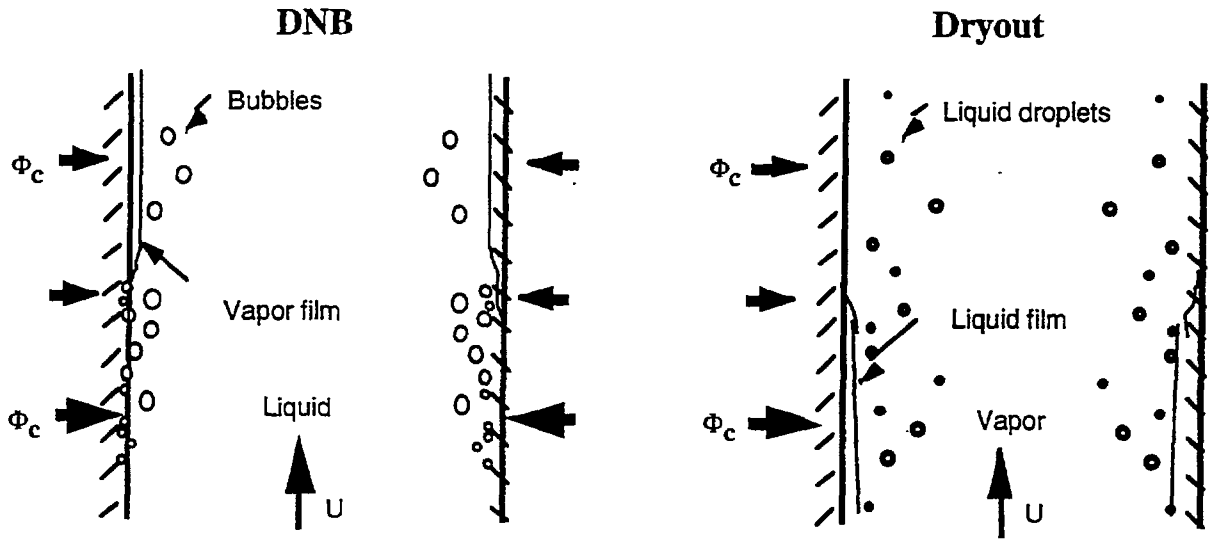


Fig 1 : The boiling crisis : DNB and dryout

Departure from nucleate boiling (DNB)

DNB has to be taken into account in pressurized water reactors (PWRs). It is reminded that very little vapor is generated within a PWR, so that under incidental (not accidental) conditions rather low quality conditions are attained.

Prior to critical conditions, heat transfer coefficients are very high because heat removal by phase change is very efficient. As the heat flux increases, more and more bubbles are being generated. This situation does not last for ever. A "vapor blanketing" effect occurs. This means that bubbles coalesce, so that a vapor layer is formed very close to the fuel rods (less than 1mm thick). The liquid phase is then insulated from the heated surface, so that a rapid temperature excursion is observed.

Dryout

For higher vapor qualities, the flow regime is annular. Most of the liquid phase is located close to the wall with some droplets being entrained by the core of the flow. Knowledge of dryout limits is of prime interest for boiling water reactors (BWR) because of bulk boiling within the core. Dryout occurs when the cooling water film breaks up, so that fuel rods are not wetted. This causes decrease in the heat transfer coefficient. Besides risks of rod failure, dryout may also cause mechanical fatigue. Dryout appears to be a smoother mechanism for the boiling crisis. As opposed to DNB, dryout is a less local phenomenon, so that history effects have to be taken into consideration.

I.2- Parameters affecting the CHF

The objective is not to provide comprehensive coverage of all parameters that may have an impact on CHF limits. Only the most significant parameters will be mentioned, so that CHF margin evaluation can be understood. Roughly speaking, these parameters fall into two categories: thermal-hydraulic and technological.

Thermal-hydraulic parameters.

One of the key parameters is quality X, that is an indication of the amount of vapor at a given elevation. It is expressed by

$$X = \frac{h - h_{l,sat}}{h_{v,sat} - h_{l,sat}}$$

where : h is the two-phase mixture enthalpy
$h_{l,sat}$ is the saturated liquid enthalpy
$h_{v,sat}$ is the saturated vapor enthalpy

It has been mentioned that the CHF is a very local phenomenon. Standard thermal-hydraulic codes do not generally compute the thermal-hydraulic field close to the rods. Quality X is usually subchannel-averaged, so that CHF is not expressed as a function of local parameters. It follows that some effects attributed to geometry can arise from this averaging process. The most significant thermal-hydraulic parameters are :

-quality X
-mass flow rate G ($kg.m^{-2}.s^{-1}$)
-pressure (Pa)
-history effects

The CHF usually decreases with quality; the more vapor, the lower the CHF. The effect of mass flow rate on the CHF turns out to be more complex. At low quality, the CHF increases with mass flow rate G because of turbulence. For dryout conditions, the CHF decreases with G because turbulence tends to break up more rapidly the liquid film close to the rod. The effect of pressure is very complex. It strongly depends on other thermal-hydraulic (T/H) parameters.

Thermal-hydraulic conditions experienced by the flow prior to the onset of CHF are referred to as history effects. The heated length may be included in correlations, although this parameter is very controversial. The axial heat flux profile has to be taken into account. The Tong factor used for non-uniform heat flux profile has been empirically expressed as a function of quality and power profile.

Technological parameters

Fuel rods in assemblies are laid out in regular bundles. Grids with mixing vanes are equally spaced, so that rods are firmly held together and turbulence is generated downstream of the grids. The following bundle and grid parameters affect the CHF :
-pitch and rod diameter. Fuel assembly correlations usually allow for these two effects.
-guide tubes are referred to as "cold walls" since no heat is generated within these rods. Guide tubes induce quality distortion in the bundle that has to be taken into consideration. These "cold wall" effects are represented by a "wetted to heated perimeter ratio" in a subchannel.
-rod bowing and rod eccentricity in a fuel assembly have a negative impact on the CHF. Safety evaluation reports have to account for these effects.
-Spacer grids are designed to enhance CHF values by generating turbulence downstream of grids. A swirl is generated by vanes, in order to obtain more turbulence for DNB conditions or to keep the water film close to the rod at dryout conditions. The CHF strongly depends on the grid design, the grid spacing (gsp) and the distance from the last upstream grid (dg). Correlations such as Westinghouse WRB-1 use dg and gsp as parameters in order to represent decay of turbulence downstream of turbulence promoters.

1.3- CHF predictions

The CHF is estimated over a wide thermal-hydraulic range for several geometric conditions. The CHF can be predicted by models or by empirical correlations.

Models are less accurate than correlations and are not used yet on a large industrial basis. It has been shown in the previous section that the boiling crisis is a very complex phenomenon, so that taking into account all technological parameters of interest is a very difficult task.
For DNB, there are mainly two models : Weisman and Katto. The former assumes that CHF occurs when the void fraction exceeds a certain level. The Katto model is based on the evolution of a dry patch. Critical conditions are assumed when the unwetted area expands at the expense of the liquid film. These two models highlight the lack of consensus on the onset of DNB.
For dryout, some models have been developed in order to predict the CHF in rod bundles with mixing grids. In this case, dryout being a smoother heat transfer transition, it is more understood. Although these models are not used extensively for safety evaluation, insight into the effects of various parameters is made possible.

The standard approach to CHF evaluation is to derive correlations. This is achieved in two steps : Fuel bundles are CHF tested for various flow conditions and the CHF location is recorded. In a second step, a thermal-hydraulic code is used, so that local conditions (quality, pressure and mass flow rate) are computed and correlated with the critical heat flux.

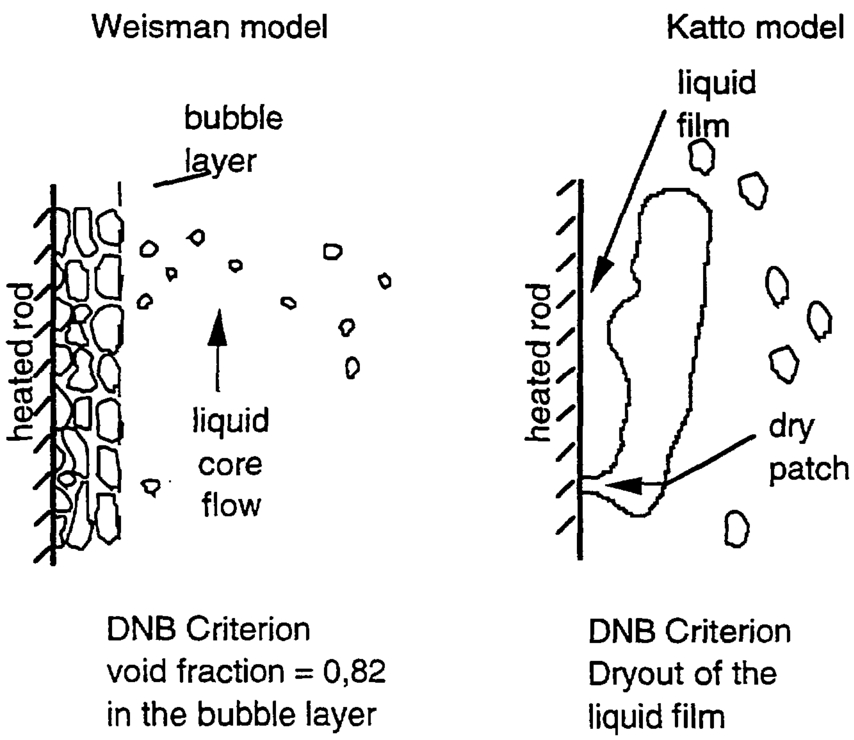


Fig 2 : postulated DNB mechanisms

CHF testing

Whole fuel assemblies cannot be CHF tested because that would require large power units. A 5x5 or 6x6 electrically heated rod bundle is tested by varying thermal-hydraulic parameters in the range of interest for safety evaluation. For PWRs:
P(100->165 bar), X(-0.1->0.3).,G(1000->5000 kg.m^{-2}.s^{-1}).

DNB conditions are attained by increasing the inlet temperature or power until a rapid temperature excursion is recorded. Electric power is then automatically shut down in order to avoid rod damage. Tests are conducted for 5x5 or 6x6 rod arrays with or without guide tubes, for various axial thermal profiles (heat flux), heated lengths and sometimes grid spacings. CHF data bases are usually fuel vendor proprietary because this information is very grid design dependent and these experiments are very expensive to run. A CHF data base generally includes about 1000 data points.

Data reduction and correlations

The objective of CHF margin calculation is to predict the CHF from local thermal-hydraulic conditions. Data collected from test sections have to be reduced in an easy way, so that predictions can be performed rapidly. Over 500 CHF correlations exist in the published literature and many others are proprietary; most of them deal with water-cooled tubes and have a narrow range of validity. This proliferation of CHF prediction methods illustrates the lack of understanding and consensus on the boiling crisis physical phenomenon. It also proves how acutely users need an accurate and reliable means of estimating boiling crisis conditions in their specific geometry and T/H range.
For PWRs, correlations come in two ways. Some include only local T/H parameters (WRB-1,W3). Others include inlet quality effects (EPRI), so that history effects are taken into account. Other T/H and geometric parameters are: local mass flow rate and pressure, grid spacing, distance from a mixing grid, rod diameter, bundle pitch. Most CHF correlations are fuel vendor proprietary. It must be pointed out that these correlations are fuel design dependent but also depend on the subchannel code used for the computation of local thermal-hydraulic conditions.

For each data point, let us call M the actual measured CHF and P the predicted CHF. Analysis of the P/M distribution yields accuracy of the correlation. Standard P/M deviations are of the order of 8%. For PWRs, uncertainties should not vary with thermal-hydraulic parameters, so that confidence in the correlation is constant for a wide T/H range. A design criterion C is calculated as follows:

$$C = \overline{\left(\frac{P}{M}\right)} + k.\sigma_{P/M}$$

with k function of the number of data points (k≈1.8). What is the meaning of C ? Under all core operating conditions, the P/M' ratio (with M' being the core heat flux) has to exceed C, so that critical conditions are not attained with a 95 percent probability at a 95 percent confidence level. For the W3 and WRB-1 correlations, C equals 1.30 and 1.17, respectively.

2- CHF safety margin evaluation

The objective of safety thermal analysis is to determine margins with respect to critical conditions. Nuclear plant safety analysis must demonstrate the plant operation within safety limits during normal operation and transient conditions. Any change to the reactor after original issue of an operating license must be made in accordance with regulatory requirements. Core operating conditions of interest for CHF margin evaluation are listed in the primary system/fuel compatibility report. Typical accidents are : reduction in the primary flow, pressure drop, control rod withdrawal at power. This section presents how local coolant conditions are evaluated and uncertainties (CHF correlation, operating conditions, geometry....) taken into consideration in order to determine safety margins.

2.1- Thermal-hydraulic calculations

The same accuracy in coolant conditions is not required in all parts of the core when it comes to CHF predictions. It has been shown that CHF predictions are carried out on a subchannel basis. A 900 MW reactor contains approximately 50 000 fuel rods. If the CHF were to be evaluated in all subchannels, local coolant T/H conditions evaluation would require very long computational times. Therefore, most T/H codes compute the T/H field in areas that are most at risk with CHF. Neutronic design calculations provide power radial distribution in the core, so that areas of high thermal power are easily identified. Every T/H code is tested against experimental data in order to receive agreement from safety authorities.

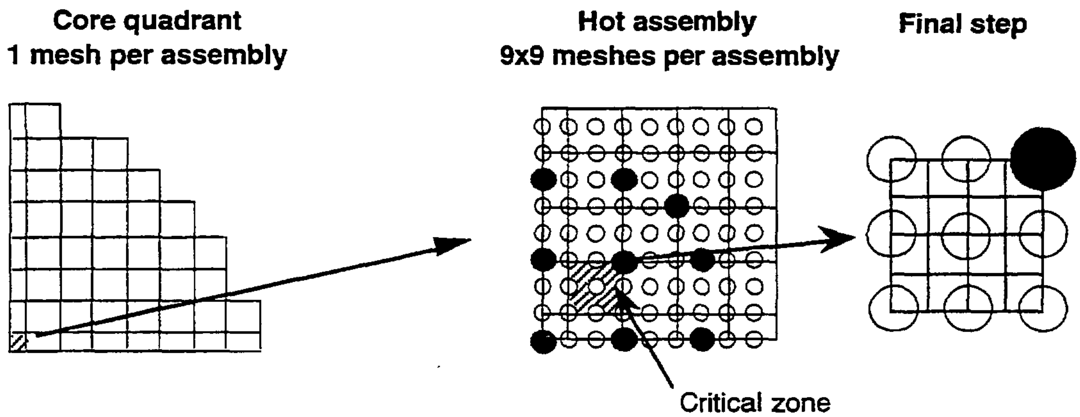


Fig 3 : THYC core calculation methodology

There are two approaches to T/H calculations. Subchannel codes such as THINC, CEA's FLICA-3M use a coarse mesh for assemblies where CHF predictions are not performed. The hot subchannel and neighboring subchannels are represented as well as the rest of the assembly. An alternate way is to carry out successive calculations by progressively restricting the computational domain (EDF's THYC, ABB's TORC). The whole core is represented by a coarse meshing. In a second step, the critical assembly is divided into a certain number of zones and first-step calculations are used as boundary conditions. The final step analyzes the most critical zone. This approach is shown in figure 3. Using a rapid core computation methodology should not affect calculation accuracy. Therefore, it must be verified that accuracy has not been lost by this procedure.

2.2- Uncertainties and margin evaluations

A core calculation methodology and a CHF correlation being available, safety margins have to be computed for various operating conditions. This section is devoted to standard approaches to uncertainties. Uncertainties may arise from :

-CHF correlation (this point has been mentioned in §1.3)
-geometry (e.g. rod bowing)
-operating conditions (system pressure, inlet mass flow rate, thermal power,......)
-code calculations

This paper will not explain in great details how all these uncertainties are allowed for. For the sake of clarity, only the first two types will be considered. More explicit literature on this subject may be found in [Tong]. Uncertainties can be combined in different ways. Safety evaluation techniques come in two ways : deterministic and statistical.

The deterministic evaluation

For a CHF correlation, a design criterion is computed from the P/M distribution calculated from experimental CHF data points. Safety requires that DNB conditions are not attained for a number of setpoints. Let us take for instance a setpoint of a 900 MW reactor core. T/H conditions are defined by an inlet temperature Ti, mass flow rate G, thermal power W, enthalpy rise in the hot subchannel,...
The deterministic approach to margin evaluation consists of taking these T/H values at their most penalizing value. Thus, a new setpoint (NSP) is defined by :P-ΔP, G-ΔG, Ti+ΔTi, W+ΔW. For given core calculation conditions, the minimum of the P/M ratio, also called the MDNBR, is computed. It is compared with the design criterion and a safety margin is defined by :

$$\mathrm{Margin} = \left(\frac{P}{M}\right)_{NSP} - C$$

The statistical approach

This is a more realistic combination of uncertainties. It is assumed that parameters vary in a statistical way around the average setpoint. For instance, pressure is defined by a distribution (e.g gaussian) characterized by a mean value, and a standard deviation.

Different approaches to statistical uncertainties have been used. For instance, the ITDP method (Improved Thermal Design Procedure) consists in calculating sensitivity of MDNBR to the main operating parameters p_i considered as independent variables :

$$Si = \frac{\partial MDNBR}{\partial pi}$$

Sensitivity parameters and uncertainties being known, an overall sensitivity parameter V_{REC} is computed. These operating uncertainties being statistically added, the design criterion C defined in §1.3 is used to compute a new design criterion C_N :

$$C_N = \frac{C}{(1 - V_{REC})}$$

This new criterion allows for correlation uncertainty and operating parameters uncertainty. The safety margin is calculated as follows:

$$\mathrm{Margin} = \left(\frac{P}{M}\right)_{SP} - C_N$$

Both uncertainties being taken into account by the design criterion, P/M is computed at nominal setpoint (SP) conditions. It has been shown that this more realistic combination provides enhanced safety margins.

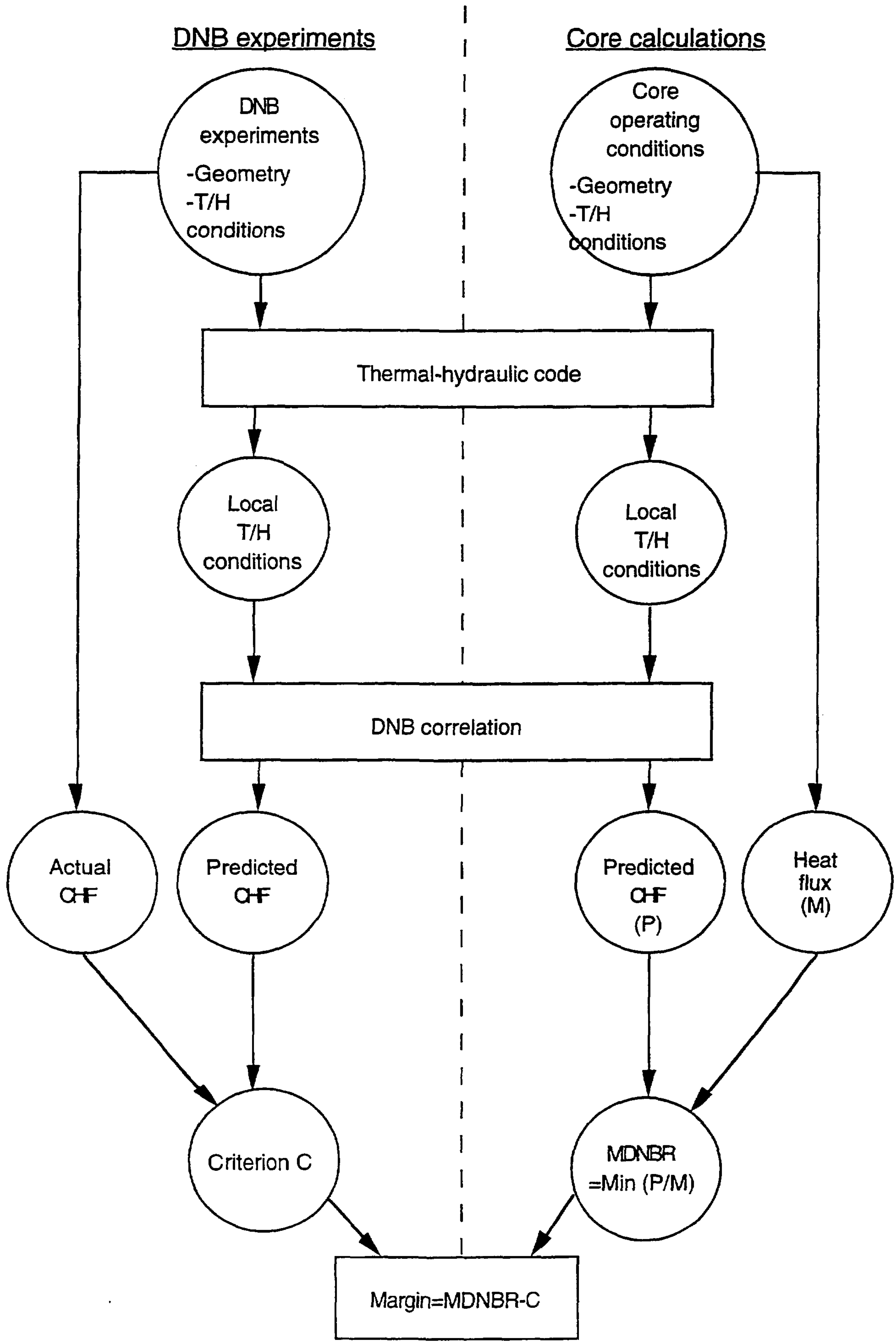


Figure 3 : Deterministic margin calculation

A more advanced statistical method (MSG) does not assume that the effect of T/H parameters (inlet temperature, pressure, ...) can be added as independent variables. The real P/M distribution is computed by a random walk approach (i.e. all parameters vary simultaneously). This requires longer computational times because the real (P/M) distribution has to be evaluated. This distribution is then statistically combined with correlation uncertainties. This approach is now licensed by safety authorities. Power upratings have been reached by this method.

So far, BWR safety evaluation has not been mentioned. The approach to safety is basically the same except that BWR safety evaluation reports do not use the MDNBR but the CPR (Critical Power Ratio) instead. The CPR can be defined as the ratio of assembly power at which CHF occurs to the actual power. It indicates how much the power can be increased before the CHF phenomenon occurs. For any given assembly power, the CPR is lower than the DNB ratio. The CPR ratio is obtained by an iterative procedure, that allows for variations in T/H parameters as critical conditions are being met.

3- Current concerns and R&D programs

Fuel evolution represents one of the major changes. The fuel design may be modified in order to improve CHF performances. Consideration of mechanical properties or corrosion may also lead to changes in the grid design to allow new fuel management. Fuel assemblies can also be purchased from different suppliers with various grid designs.

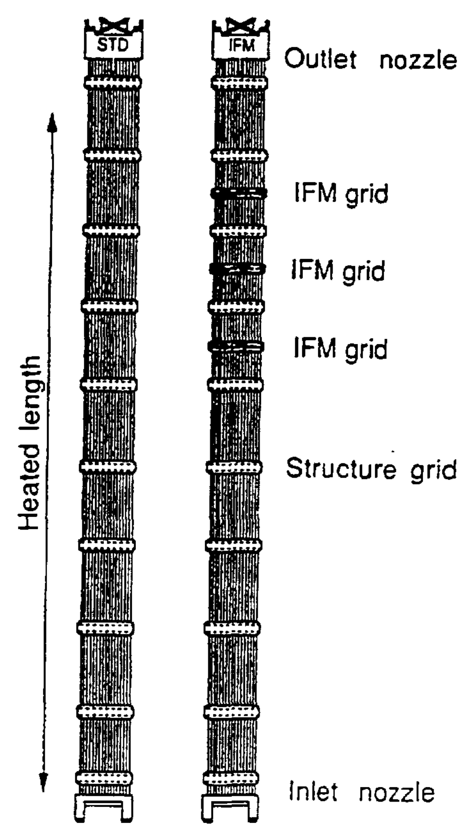
When a core is reloaded, only a third or a fourth of the core is renewed. When new fuel assemblies are introduced into the core, thermal-hydraulic compatibility between these assemblies has to be verified. Different grid designs induce different pressure drop coefficients, so that local coolant conditions are affected by these changes. A standard fuel assembly is depicted as well as a new fuel design assembly with Intermediate Flow Mixers (IFMs). IFMs are laid out in the upper part of the core, a location most at risk with CHF because of higher quality. Additional turbulence promoters increase DNB limits of reactor cores but can cause crossflows between assemblies. One concern is to address these T/H problems.

Fuel assembly evaluation depends on a T/H code and a CHF correlation. These two tools vary from one supplier to another. Therefore, power utilities have to verify CHF performance with one single tool. Moreover, one must keep in mind that the reactor operating system is controlled and protected by one correlation. Therefore, this correlation has to predict CHF in the core in a conservative (safe) way for all types of fuel assemblies.

R&D programs

Examples of current status of topics discussed previously are presented: boiling crisis understanding, CHF predictions, core calculations and approaches to safety.

Efficient models have been found for dryout conditions. Predictions of critical power ratio CPR by General Electric based on a two-fluid, multi-field model compare favorably well against experimental data with a relative percentage error of the order of 5%. The predicted trends in critical power versus



some important physical parameters are also found to be in close agreement with experiments.
DNB still remains a very challenging problem. Understanding of the onset of DNB requires very acute knowledge of flow conditions close to the rods, so that advanced techniques are necessary in order to capture the flow structure. Very modern two-phase flow probing (LDV, optical fibers) has been developed to obtain insight into the two-phase layer within a tube. EDF has launched an extensive program in that respect called Aphrodite in a freon test tube where DNB data points are collected in order to derive local information. This experiment is now extended to a 3-rod bundle (Poseidon). The objective of these experiments is to develop very local models to predict the boiling crisis. Local information such as the void fraction distribution is then used to be compared with CFD codes such as the ASTRID code. Basic experiments may be used in order to investigate analytical effects (e.g. heated length; turbulence promoters, surface conditions...).such as the DEBORA freon CHF test section (CEA).

Apart from basic considerations, it has been shown that CHF evaluation basically depends on a CHF correlation and a T/H code.

CHF are usually established by standard regression techniques. A new statistical technique has been developed by the CEA in order to improve the accuracy of CHF predictors. Spline functions are used so that the complex CHF phenomenon can be described more precisely. In addition, optimal smoothing is performed by using the generalized cross validation technique. This method has turned out to yield lower design criteria. Moreover, it is a very efficient way to analyze experimental data in order to study special effects (fuel performance comparison, technological and thermal-hydraulic effects).

In the area of thermal-hydraulic analysis, codes used for licensing purposes are based on a one dimensional approach. This means that the flow is assumed to be predominantly axial. This causes some limitations when three-dimensional flows have to be considered. This situation can be encountered in mixed cores or in a steam line break accident.
In recent years, EDF has developed a three dimensional code THYC. Its scope of application is single or two-phase flow in rod bundles. It is especially devoted to heat and mass transfer in the following nuclear components: reactor cores, steam generators and condensers. The code differs from subchannel analysis which assumes a prevailing axial component of velocity and uniform pressure at each elevation. A fully three-dimensional representation of the flow is proposed in conjunction with a porous-body approach. THYC has been validated against test data, so that accuracy of correlations and models is established.

CONCLUSION

The boiling crisis is a very complex phenomenon, not fully understood from a theoretical point of view, so that empirical relationships are still necessary to predict the critical heat flux. Moreover, in fuel assemblies, many parameters have a strong impact on the CHF. Therefore, extensive experimental programs are still necessary to obtain a wide application range. For reactor cores, it has been shown that core design requires a methodology that is both thermal-hydraulic and statistical, so that fast and accurate calculations can be performed for operating conditions relevant to safety. Fuel design evolution induces new studies in the field of CHF margins. CHF performances have to be checked as well as compatibility with other fuel designs.
In order to improve mixing grid design and to obtain larger validity ranges, mechanistic models are being developed. These programs are based on sophisticated experimental studies as well as advanced two-phase flow CFD codes. In the area of CHF evaluation, more accurate correlations are being developed, so that CHF can be predicted to a better extent. The general trend in the field of core thermal-hydraulic calculation is to use three-dimensional codes, so that more realistic situations can be dealt with. These code development programs are carried out in conjunction with tests against advanced validation data.

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