1 INTRODUCTION

During thermal-hydraulic design and calculus of nuclear reactors, the cooling performance of a core when the flow is driven by natural circulation forces needs to be calculated. This can be the case after a pump trip, or in a core whose normal operation mode is natural circulation.

The main characteristics of this version of the program are the following:

Reactor type: pool type

Fuel elements: MTR

Coolant channel: rectangular

Coolant: light water and air

Operating conditions: low pressure, laminar regime

Flow type: single phase

Operating conditions: steady state

Operating regime: upward flow, driven by the natural convection mechanism

Power shape: cosine with extrapolated distance or uniform

2 THERMAL-HYDRAULIC METHOD

Due to the Natural Convection characteristics of the flow, the hydraulic problem is highly coupled with the thermal problem.

The natural circulation condition states that the buoyant forces produce a flow that is balanced by sum of friction and acceleration forces.
The buoyant force on the heated channel is calculated as:

$$F_b = (\rho_c - \rho_p) \cdot A_c \cdot L_c$$

where:

ρ_c= average density in the channel = $\frac{1}{L_c}\int_0^{L_c} \rho_c(x)dx$

ρ_p= density of the coolant in the pool tank
A_c= Coolant channel cross section area
L_c = Heated length of coolant channel

The frictional forces produced by the flow velocity is expressed as:

$$F_F = \frac{(\rho V)^2_{in}}{2g} A_c \left(\frac{K_{ent}}{\rho_i} + \sum_{i=1}^{n} f \frac{\Delta z_i}{\rho i \cdot D_H} + \frac{K_{out}}{\rho_{out}} \right)$$

where:

ρ	= density of the coolant at the calculation point
V	= velocity of the coolant at inlet
g	= gravity acceleration
f	= friction factor
Δz_i	= axial node size
D_H	= hydraulic diameter
K_{ent}, K_{out}	= Inlet and Outlet form loss coefficients

Thermal distribution is calculated for the estimated velocity using heat transfer correlations and friction factors suitable for the geometry and regimes of the system. The axial distributions of the coolant temperature, wall temperature, and fuel temperature in the centerline are determined.

The resolution scheme is based on a finite difference one with variable number of nodes. Mass, energy and momentum equations are integrated within each node.

The code requires a peak factor, in order to perform calculations in the average and the hot channels.

The effect of a chimney located in the upper part of the core can be included in the calculation. The main objective is to take into account the additional buoyant force due to the hot water column located above the core and inside the chimney. Friction pressure losses due to chimney are also calculated and are included into the total pressure loss.

3 POWER DISTRIBUTION

The user gives the program the total power of a fuel element, the number and dimensions of fuel plates, as well as coolant channel dimensions.

With these data, the program calculates the power distribution in the fuel element assuming a cosine-type distribution with extrapolated distance or a uniform distribution.

Local heat flux:

* *Cosine:*

$$FI(Z) = FI0 * COS\left(\frac{\pi * Z}{2 * LP}\right)$$

Uniform:

$$FI(Z) = FI0$$

Where:

Z	= axial coordinate	(m)
FI	= local heat flux	(watt/m^2 °C)
LP	= extrapolated length + half active length	(m)
FIO	= maximum axial heat flux	(watt/m^2 °C)

$$FIO = \frac{POTBOX * PF}{(NPLATES - 1) * 2 * ACTWIDTH * ACT\_LENGTH}$$

POTBOX	= power in the fuel element (watt)	
PF	= peaking factor	
NPLATES	= Nr. of plates in a fuel element	
ACTWIDTH	= plate active width	(m)
ACT_LENGTH	= plate active length	(m)

In each node, it is assigned the integral of the power-shape distribution performed between the inlet and outlet junctions of the node.

4 VELOCITY DISTRIBUTION

The velocity profile in the channel is calculated using the mass and momentum equations, coupled with the energy equation.

Along a fuel element, the mass flow (kg/sec) is constant although the velocity and the density will vary.

Therefore, using an initial guess value for the mass flow, the program computes the friction pressure drop of each node and zone of the fuel element, the buoyancy driving force and the acceleration pressure drop.

After performing the calculation for all nodes in a fuel element, the program compares the total buoyancy term with the total acceleration and friction terms. If the buoyancy is greater than the others, the program computes a positive correction to the mass flow, otherwise, the correction is negative. With this new value the momentum equation is once more calculated until the difference between the buoyancy term and the friction and acceleration terms is smaller than a convergence criteria.

The buoyancy force is calculated using a reference coolant density at pool temperature.

4.1 Hot channel vs. mean channel calculations

Two sequenced calculations are performed by CONVEC: first a global thermal-hydraulic parameters calculation using the average power of the core, and then the calculation of thermal-hydraulic parameters of the hot channel.
For example, if the case is a core with 25 fuel elements operating at 200 kWatt, then the code requests on line for the average power per F.E., that is:

$$Average\,fuel\,element\,power = \frac{200\,Kwatt}{25\,Fuel\,elements} = \frac{8\,Kwat}{F.E.}$$

Using the total power peaking factor of the core and a given cosine profile (defined by the extrapolated distance) or a uniform profile, the axial peaking factor is estimated. Typically, for light water as moderator the extrapolated value is approximately 8 cm, giving a typical value for axial peaking factor of around 1.3.

Then, the program calculates the hottest fuel power as follows:

$$\text{Hottest fuelelement power} = \text{Average fuelelement power} \cdot \text{Radial peaking factor}$$

$$\frac{8Kwatt}{F.E.} \cdot \frac{TotalPeak\,Factor}{AxialPeak\,Factor} = 18.5\,Kwatt(\,hottest\,F.E.\,)$$

When an average channel calculation is performed, the total pressure loss in the core is calculated with the average power per fuel element in the core.
The program includes the capability to model the pipe that exists in between the grid plate of the FE and the flap valve.
When the hot channel calculation is performed, the pressure loss in this pipe is set using the value obtained previously from the core average calculation.

4.2 Friction correlations

The program has a set of appropriate friction correlations used for calculating the friction pressure drop in inlet nozzle and outlet plenum, as well as in the coolant channel.

All correlations are valid for the laminar regime. The program verifies that the regime is really laminar (Reynolds less than 2000). If this were not the case, a warning message will be printed in the output file.

4.2.1 Distributed friction factor

* *Parallel plates friction correlation* (Ref 1 and 1):

$$f = \frac{16.}{\left(\frac{2}{3} + \frac{11}{24}\alpha.(1-\alpha)\right)} \qquad \alpha = \frac{gap\,channlel}{Span\,channel}$$

$$DP = \frac{f \cdot \Delta Z \cdot V \cdot \mu_f}{2 \cdot D_{hy}^{\,2}} \cdot CORREC$$

$$CORREC = \left[\frac{\mu_w}{\mu_b}\right]^{0.28} \qquad \text{(water)}$$

$$CORREC = \left[\frac{\mu_w}{\mu_b}\right]^{1.0} \qquad \text{(air)}$$

Where:

α	=	Aspect ratio of the rectangular channel
f	=	Laminar Fanning friction factor
DP	=	Pressure loss (Pa)
V	=	Velocity in between the plates (m/s)
D_{hy}	=	Hydraulic Diameter (m)
ΔZ	=	Node length (m)
μ	=	Coolant Viscosity (b: bulk; w: wall) (kg/m^3)

* *Friction correlation in a general section* (Ref. 1)

$$DP = \frac{32 \cdot L \cdot V \cdot \mu_f}{D_{hy}^{2}}$$

where L is the section Length

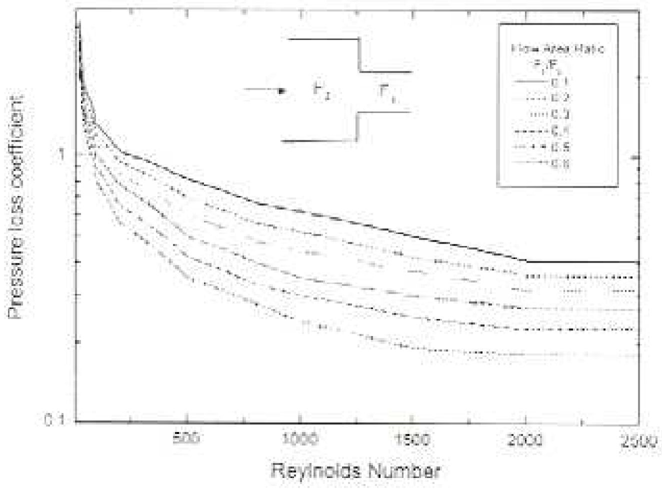
4.2.2 Form losses coefficient

* *Area contraction irreversible pressure drop:*

From Ref 1 the following values as a function of area ratio (F) and Reynolds number (Re) in the contracted area were adopted to calculate the form loss in a sudden contraction.

$$DP = \frac{1}{2} K(\mathrm{Re}, F) \rho \cdot V^2 \qquad \mathrm{Re} = \frac{V_1 \cdot D_{h1} \cdot \rho}{\mu}$$

F	*Reynolds Number*										
	20.	*30.*	*40.*	*50.*	*100.*	*200.*	*500.*	*800.*	*1000.*	*1500.*	*2000.*
0.1	3.20	2.40	2.00	1.80	1.30	1.04	0.82	0.67	0.62	0.50	0.410
0.2	3.10	2.30	1.84	1.62	1.20	0.95	0.70	0.56	0.52	0.42	0.360
0.3	2.95	2.15	1.70	1.50	1.10	0.85	0.60	0.48	0.44	0.37	0.315
0.4	2.80	2.00	1.60	1.40	1.00	0.78	0.50	0.40	0.35	0.30	0.270
0.5	2.70	1.80	1.46	1.30	0.90	0.65	0.42	0.33	0.30	0.25	0.225
0.6	2.60	1.70	1.35	1.20	0.80	0.56	0.35	0.28	0.24	0.19	0.180

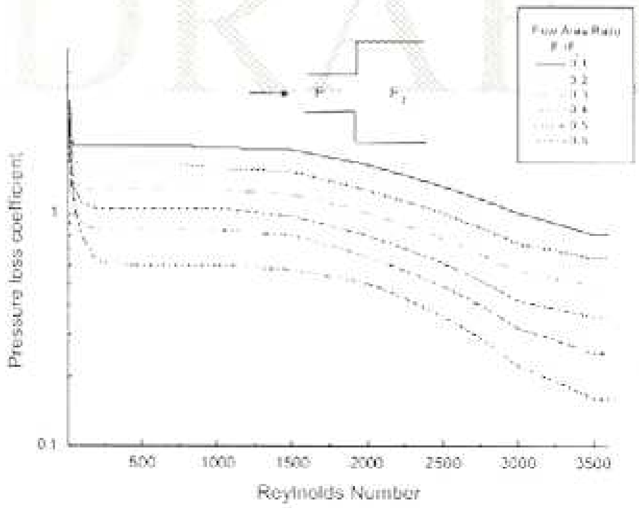


* *Area expansion irreversible pressure drop:*

From Ref **Error! Reference source not found.** the following values as a function of area ratio (F) and Reynolds number (Re) in the contracted area (Re) were adopted to calculated the form loss in a sudden expansion:

$$DP = \frac{1}{2} K(\mathrm{Re},F)\,\rho\,V^2 \qquad \mathrm{Re} = \frac{V_1 \cdot D_{h1} \cdot \rho}{\mu}$$

F	Reynolds Number												
	20	30	40	50	100	200	400	500	800	1000	1500	2000	3000
0.1	3.00	2.40	2.10	1.95	1.94	1.93	1.92	1.92	1.92	1.90	1.85	1.60	1.00
0.2	2.80	2.20	1.85	1.65	1.60	1.60	1.60	1.60	1.60	1.55	1.50	1.25	0.74
0.3	2.60	2.00	1.60	1.40	1.27	1.25	1.25	1.25	1.25	1.25	1.20	1.00	0.56
0.4	2.40	1.80	1.50	1.30	1.10	1.05	1.05	1.05	1.05	1.05	0.97	0.80	0.42
0.5	2.30	1.65	1.35	1.15	0.90	0.85	0.85	0.85	0.85	0.85	0.80	0.65	0.32
0.6	2.15	1.55	1.25	1.05	0.80	0.62	0.60	0.60	0.60	0.60	0.57	0.50	0.22



5 TEMPERATURE DISTRIBUTION

5.1 Fluid temperatures

Fluid temperatures are calculated in the following way:

Using the current iteration channel flow rate, the program integrates the energy equation beginning from the lower end towards the top of the channel. In this way, for one generic node the temperature is calculated with the following equation:

$$Tout\_fluid(j) = \frac{Power(j)}{Flowrate \cdot cp(j)} + Tinlet\_Fluid(j)$$

With,

$$Tinlet\_fluid(j) = \begin{cases} Tpool & j = 0 \\ Toutlet\_fluid(j-1) & j > 1 \end{cases}$$

where the nodes "j" are numbered from bottom to top .

The fluid specific heat (cp) is calculated at an average node temperature. Due to this reason, an iteration is performed within each node.

5.2 Wall temperatures

Wall temperatures are calculated using suitable heat transfer correlations for flow regimes and geometries of the MTR coolant channels.

In this version the program has two heat transfer correlations. One of them is for forced laminar flow, and the other for free natural convection.

Following the criteria indicated in Ref 2, the parameter for using one or another is:

If the quotient Grashof Nr/(Reynolds Nr)^2 is greater than one, the natural convection effects on the velocity profile are significant and the natural convection correlation is used. On the other hand, if this quotient is smaller than one, the velocity profile is not strongly dependent on buoyancy effects and the forced laminar correlation is used.

$$Twall(j) = \frac{Heat\_flux(j)}{h\ (j)} + Tfluid(j+1/2)$$

$$Tfluid(j+1/2) = \frac{Tfluid(j) + Tfluid(j+1)}{2}$$

where h(j) is the heat transfer coefficient at the node "j"

The temperature in the centerline of the fuel element is calculated with the following expression:

$$Tcl(j) = Twall(j) + q'' \cdot \left(\frac{e_{meat}}{2.K_{meat}} + \frac{e_{cladding}}{K_{Aluminum}} + \frac{e_{oxide}}{K_{oxide}} \right)$$

where K_i is the thermal conductivity of the fuel meat, Aluminum or Oxide, and e_i the material thickness.

5.3 Heat transfer correlations

5.3.1 Laminar forced convection regime:

The condition that verifies this regime is:

$$\frac{Grashof\ Nr}{(Reynolds\ Nr\ )^2} < 1.0$$

The Nusselt number value ($Nu_{x,H}$) is obtained from the following equations valid for two parallel plates (Ref. 1) with heated walls:

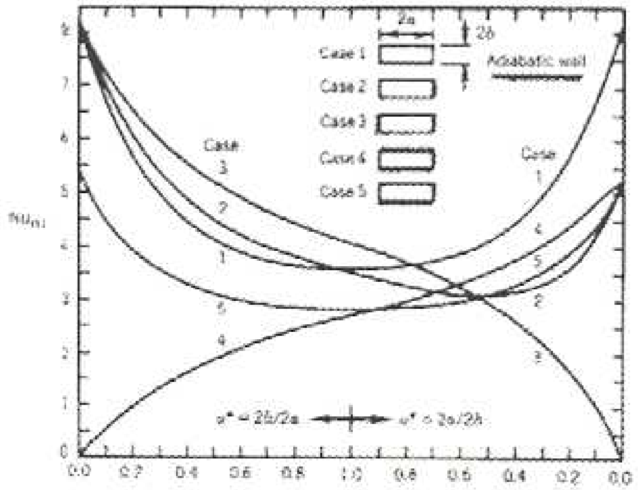
$$Nu_{x,H} = \begin{cases} 1.490 \cdot x^{*-1/3} & for\ x^{*} \le 0.0002 \\ 1.490 \cdot x^{*-1/3} - 0.4 & for\ 0.0002 < x^{*} \le 0.001 \\ 8.235 + 8.68(10^{3} \cdot x^{*})^{-0.506} \cdot e^{-164x^{*}} & for\ x^{*} > 0.001 \end{cases}$$

$$x^{*} = \frac{x / D_h}{\mathrm{Re\,Pr}} \qquad\qquad Nu_{x,H} = \frac{H \cdot D_h}{k}$$

Where:

- x = axial distance form the entrance (m)
- D_h = Hydraulic diameter (m)
- H = Heat transfer coefficient (w/m2/C)
- k = Bulk water conductivity (w/m/C)
- Re= Reynolds Number
- Pr = Prandlt Number

The following correction factors have been adopted to consider a heated rectangular duct instead a flat duct. From the Ref. 1, according to the heated surfaces, different correction factors as a function of the aspect ratio of the channel are proposed for a fully develop condition:



The following expression fits correctly the case 3 of the previous figure:

$$Nu = N_{x,H}.(1.-1.4122\ \alpha+2.3473\alpha^{2}-2.8983\ \alpha^{3}+2.0629\alpha^{4}-0.6077\alpha^{5})$$

The following expression has been fitted, according to data of Ref 1, to correct the influence of the entrance distance in developing flow condition:

$$x^{*} = \frac{x^{*}}{6.0 - 5.0 \cdot \exp^{\frac{0.75 \alpha}{0.5257}}}$$

5.3.2 Natural free convection regime:

The condition that verifies this regime is:

$$\frac{Grashof\ Nr}{(Reynolds\ Nr)^2} > 1.0$$

From Ref 2 the Elenbaas correlation, appropriate for infinite vertical parallel plates, was selected:

$$N_u = \frac{H \cdot R_h}{K} = \frac{1}{24} \cdot \frac{R_h}{L} \cdot (Grashof \cdot Prandtl) \cdot (1 - \exp^{(-CTE)})$$

With

$$CTE = \left[\frac{0.5 / L}{R_h \cdot Grashof \cdot Prandtl} \right]^{0.75}$$

H	=	heat transfer coefficient
K	=	fluid thermal conductivity
D_h	=	hydraulic diameter
C_p	=	fluid specific heat
μ	=	fluid viscosity
β	=	fluid thermal expansion coefficient
R_h	=	hydraulic radius = $D_h/2$
g	=	gravity acceleration
ρ	=	fluid density
V	=	fluid velocity
L_h	=	heated length

$$Grashof = \rho^2 \cdot g \cdot \beta * (Twall - Tfluid) * \frac{L_h^3}{\mu^2}$$

$$Prandtl = \frac{Cp \cdot \mu}{K}$$

$$Reynolds(D) = \rho * V * \frac{D_h}{\mu}$$

6 THERMAL-HYDRAULIC SAFETY MARGINS

Thermal-hydraulic safety margins are evaluated for the reactor core using this code with the information of the fuel design parameters. The input parameters required for analysis of the limiting (hottest) channel in the core include:

1) nuclear power peaking factors for the limiting fuel element
2) engineering hot channel factors that account for fuel fabrication tolerances, uncertainties in calculated parameters, and uncertainties in the ability to measure certain variables such as the reactor power.

The nuclear power peaking factors may be introduced as an axial power distribution and a radial peaking factor.

The engineering hot channel factors are applied as three separate components corresponding to:

1) Uncertainties that affect the heat flux, F_{QF}
2) Uncertainties in the flow or enthalpy change in the channel, F_{ER}
3) Uncertainties in the heat transfer to the coolant, F_{HT}

These factors are introduced as:

$$q''_{hc} = F_{QF} \cdot q''_{nc}$$
$$h_{hc} = h_{nc}/F_{HT}$$
$$MFR_{hc} = MFR_{nc}/F_{ER}$$

where the notation *"hc"* refers to the hot channel and *"nc"* refers to the nominal channel values for the heat flux (q"), the heat transfer coefficient (h), and the mass flow rate (MFR). The hot channel factor for flow cannot be applied directly because the flowrate is determined iteratively from the balance between buoyancy and friction. Therefore, the square of this factor is applied to increase the frictional component of the balance. In this manner, the flowrate is reduced approximately by this factor.

6.1 Nuclear Power Peaking Factors

The total nuclear power peaking factor used in evaluating the thermal-hydraulic safety margins is always fixed in the maximum design value. The neutronic calculations include an error factor to assure the design value.

The total power peaking factor is defined as the product of two components:

1) a radial factor defined as the average power density in each fuel element divided by the average power density in the core
2) an axial factor defined as the peak power density in each fuel element when the control plates are fully-withdrawn divided by the average power density in that fuel element.

This total peaking factor is required as input data in CONVEC when the option of cosine profile is used. The value of the extrapolate distance is used to determine the axial peaking factor

6.2 Engineering Hot Channel Factors

The hot channel factors selected for analysis of the thermal-hydraulic safety margins are based in part on the fuel plate specifications for the reactor core that is calculated, and in part on steady-state computations with natural convection flow.
The hot channel factors are applied as three separate components corresponding to the heat flux, F_{QF}, the flow or enthalpy change in the channel, F_{ER}, and the heat transfer to the coolant, F_{HT}. These hot channel factors are typically broken down into subfactors based on uncertainties in the manufacturing process, measurements and tolerances in the specifications.
The subfactors may be combined by simply multiplying them together, by treating them statistically, or a combination of the two. The first method is conservative, but somewhat unrealistic. The second two methods recognize that all of these conditions do not occur at the same time and location. The subfactors are combined statistically as:

$$F = 1 + \sqrt{\sum_i (1 - f_i)^2}$$

where the fi are the subfactors of F.

The specifications for the typical MTR fuel plates give the following fabrication tolerances:

Fuel Plate Thickness	< ± 4.0 %
Meat thickness	< ± 5.0 %
Meat width	< ± 1.0 %
235U Loading Variation	≈ ± 2% per plate
Homogeneity of ^{235}U in Fuel Meat (U_3Si_2)	≈ ± 12 %
Coolant Channel Thickness	< 5.0 %

Typically, the fuel plate tolerance is assigned directly to the fuel meat. The local tolerance in the meat density is assigned as the local homogeneity variation.
It is worthwhile to mention that the global input values like the average FE power and the pool temperature must be considered with conservative values (i.e. including deterministic error factors)

We can consider two cases to analyze the influence of the tolerances:

a) The reactor works with nuclear power under natural circulation
b) The reactor is cooled by natural circulation after shutdown

6.2.1 Operation under natural circulation

These tolerances translate into the following fabrication subfactors, increasing the heat flux:

Uncertainty In	Hot Channel Subfactor
Fuel Meat Thickness (Local)	1.05
^{235}U Loading per Plate	1.02
^{235}U Homogeneity (Local)	1.12

Concerning to the channel thickness, as it is shown in section 4.2.1, the friction pressure loss has dependence with the inverse of hydraulic diameter. However, the flow in a natural circulation is practically self-adjusting because the buoyant forces increase with heating and is offset by the friction forces, so the net effect in the velocity is negligible.
On the other hand, as it is shown in section 5.3.1, the heat transfer correlation for fully developed flow has dependence with the inverse of hydraulic diameter, and a weakly dependence with the inverse of hydraulic diameter for developing flow. So, considering the case of maximum gap and averaging its influence, we assume:

Uncertainty In	Hot Channel Subfactor
Flow Channel Thickness (5%):	
Increase in bulk temperature	-
Increase in clad surface temperature	1.02

Additional uncertainties that were included are:

Uncertainty In	Hot Channel Subfactor
Power Distribution (5 %)	1.05
Heat Transfer Coefficient (15%)	1.15
Coolant Flow Rate (10%)	1.10

The heat transfer coefficient Subfactor is treated as a multiplicative factor in F_{HT}.
Again, since the flow in a natural convection reactor is somewhat self-adjusting (the buoyant forces increase with heating and are offset by the friction forces), uncertainties in the coolant flow rate due to friction, orificing, and plenums are difficult to assess. An uncertainty of 10% in the coolant flow rate has been assumed.

The hot channel factors and the hot channel components typically used in the calculations for the MTR reactor are summarized in the following table:

Uncertainty	F_{qr}	F_{ER}	F_{HT}
Fuel Meat Thickness	1.05	-	-
235U Loading	1.02	1.02	-
235U Homogeneity	1.12	-	-
Coolant Channel Thickness	-	-	1.02
Power Measurement	1.05	1.05	-
Coolant Flow Rate	-	1.10	-
Heat Transfer Coefficient	-	-	1.15
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Statistical Combination	1.14	1.11	1.17*

6.2.2 Core Cooling by natural circulation after shutdown

The principal uncertainty after the core shutdown is the determination of the decay power. Concerning this case, it is recommended to calculate the thermal-hydraulic parameters using an overestimation of the heat decay in the core.
The heat decay should be calculated according to the ANSI/ANS (Ref. 10) recommendation, considering the power history of the core working at 105 % and using a global safety factor not less than 1.15 as uncertainty in the power. Using this approach, the values of F_{QF} and F_{ER} should be equal to 1.

6.3 Limiting phenomena correlations

The following phenomena are commonly used to evaluate the safety margins: Onset of Nucleate Boiling, Boiling of the coolant, Pulsed Boiling and Burnout.
The Onset of Nucleate Boiling is considered as the first warning and the heat flux that initiates it is frequently used as a thermal design constraint. Actually, the ONB is taken as a limit in steady state conditions although it does not correspond to any critical event.
The code doesn't perform calculation under two phase flow conditions so, the distance to limiting phenomena are based on extrapolation using correlations developed ad hoc.

6.3.1 Onset of Nucleate Boiling:

The wall temperature at onset of nucleate boiling is calculated using the subcooled boiling correlation of Bergles & Rohsenow correlation (Ref. 6) with the local heat flux:

$$Tw_{ONB} = Tsat + \left(\frac{q''_{Loc}}{15.6\ P^{1.156}} \right)^{1/n}$$

$$n = 2.30/P^{0.0234}$$

q''_{Loc} = Local Heat Flux in (Btu/hr-ft^2)
P = Pressure (Psia)
Tsat = Saturation temperature in (F)
Tw_{ONB} = Wall Temperature for Onset of Nucleate Boiling (F)

Two conditions are considered:

a) If Twall < Tw_{ONB}: It means that the present heat flux is smaller than the one needed to establish Nucleate Boiling. The heat flux to ONB is calculated using the Bergles & Rohsenow correlation and the single-phase heat transfer equation:

$$H_{1phase} \cdot (Tw_{ONB} - T_{liq}) = q_{ONB}$$

$$q_{ONB} = 15.6\ P^{1.156} (Tw_{ONB} - T_{sat})^n$$

$$n = 2.30/P^{0.0234}$$

where
q_{ONB} = ONB Heat Flux in Btu/hr-ft^2.
P = Pressure Psia,
Tsat = Saturation temperature in °F
H_{1phase} = Single-phase heat transfer coefficient (see section 5.3)

Both equations are solved with the Tw_{ONB} as independent variable, and with this value the Heat flux to ONB is calculated.

b) If Twall > Tw_{ONB}: There is Nucleate Boiling and the calculation stops.