5.8 THERMAL AND HYDRAULIC DESIGN

5.8.1 Introduction

This section describes the main features of the thermal-hydraulic design for the reactor and provides information that demonstrates that core integrity is preserved and fuel cladding is protected by adequate cooling during Reactor operational states and anticipated operational occurrences.

5.8.2 Codes and Standards

The thermal-hydraulic design bases conform to the following standards and documents:

IAEA Safety Series No. 35, Safety Requirements for Research Reactors (draft, February 1999).

Criteria for the Design of Nuclear Installations, Regulatory Guideline RG-5, draft, December 1998.

5.8.3 Design Bases

Table 5.8/1 summarises the thermal and hydraulic design bases given below.

5.8.3.1 General Design Bases

a) Avoidance of thermally or hydraulically induced fuel damage during normal steady-state operation and during anticipated operational occurrences is the main thermal-hydraulic design basis. Specific aspects of fuel integrity are discussed in 5.3.2.

b) In order to satisfy the design basis for steady state operation and anticipated operational transients, design limits are established with an adequate margin to safety limits. Violations of these design limits will not necessarily result in fuel damage.

c) The Reactor Protection Systems (RPS) provide for automatic reactor trip or other corrective action before these design and safety limits are violated.

5.8.3.2 Power Peaking Factor Related Design Basis

The total Power Peaking Factor (PPF) is related to the non-homogeneous spatial distribution of the heat flux over the core. This distribution determines that in certain spots of the core the local heat flux is higher than the average value. For thermal-hydraulic calculations a cosine profile with a maximum PPF value of 3 is conservatively adopted.

For every core operational configuration, it must be fulfilled that PPF ≤ 3

The average and the maximum heat fluxes are calculated in the following way:

a) The average heat flux (q''_{ave}) is calculated as the power removed by the Primary Cooling System divided by the total heat transfer area.

b) The maximum operational heat flux (q''_{max}) is obtained from the product between the average heat flux and the PPF.

5.8.3.3 Safety Relevant Design Bases

Research reactors that are cooled and moderated by water are limited, from the thermal point of view, by critical phenomena leading to boiling crisis and damage in the fuel assemblies.

Power is established to a maximum within reasonable safety margins compatible with the imposed operative conditions.

To establish those margins it is necessary to know the limits above which critical phenomena appear leading to a rapid increase in cladding temperature and fuel plate damage.

Different phenomena can lead to the boiling crisis and they are considered of importance to define the design criteria. These phenomena relevant to the reactor safety are:

a) Departure from Nucleate Boiling – DNB
b) Flow Instability Phenomenon – RD
c) Low Flow Burn-out Phenomenon - BO
d) Fluid-Structure Interactions - v_{crit}

5.8.3.3.1 Critical Heat Flux Related Design Bases

A design basis related to the critical heat flux is set by adopting and ensuring an appropriate margin for these variables involved in the heat transfer regime, taking into account phase changes and boiling crisis (DNB ratio).

The ratios between the critical phenomena and the maximum heat flux (q''_{max}) or the integrated power (P_{max}) in the hot channel, define the design criteria that must be fulfilled.

5.8.3.4 Non-Safety Relevant Design Bases

In research reactors using plate-type fuel assemblies, specific thermal-hydraulic conditions are commonly used as design constraints or warnings in order to anticipate critical phenomena. The phenomena appearing before the critical phenomena are referred as non-safety relevant conditions.

The only non-safety relevant condition for this design is the Onset of Nucleate Boiling (ONB). In this sense a design basis is adopted for the cooling system requiring that it be designed in such a way that it provides enough cooling to the reactor core in any operational situation, and local boiling should not be reached in any point during steady state normal operation”

5.8.3.5 Irradiation Facility Related Design Bases

The following design bases have been defined for the Thermal and Hydraulic Design of the Irradiation Facilities in order to comply with the requirement that, during the irradiation process, the heat generated in the material must be removed to avoid any thermally induced damage.

a) The criterion of Onset of Nucleate Boiling heat flux is adopted in the Bulk Irradiation Facilities.
b) For the Bulk Irradiation Facilities it was conservatively adopted that the maximum wall temperature shall not exceed the saturation temperature at the channel exit.
c) The cooling in the other rigs must remain sufficient when any bulk production irradiation rigs are removed from their positions during normal operation.

5.8.4 Description

The power released in the reactor core resulting from the fission process is removed by different cooling modes depending on the Reactor State.

The thermal-hydraulic design is performed by integrating: geometric data; results of assessments on power distributions and data on cooling system characteristics and conditions. Calculations are performed by codes using computational models and experimental correlations. Details on the validation and verification of the programmes and codes are given in Section 5.10.

One of the distinctive features of the Reactor Facility is that the core cooling flow circulates in the upward direction, with an inlet plenum below the core to allow flow homogenisation. The upward circulation enables a dynamic pressurisation of the core, increasing the saturation temperature and hence also the thermal margins with respect to a downward circulation scheme. It also avoids flow inversion following a pump coast down.

5.8.4.1 Safety Relevant Phenomena

5.8.4.1.1 Flow Instability

5.8.4.1.1.1 General

Flow instabilities must be avoided in heated channels as flow oscillations affect the local heat transfer characteristics and may induce a premature burnout. In low-pressure subcooled boiling systems, flow excursions leading to burnout have been observed (Maulbetsch, J.S. and Griffith, P., 1965; Whittle, R.H. and R. Forgan, R., 1967). The burnout heat flux occurring under unstable flow conditions was well below the burnout heat flux for the same channel under stable flow conditions. For practical purposes in plate-type fuel design, the critical heat flux that leads to the onset of flow instability is more limiting than the heat flux for stable burnout.

The most common flow instabilities encountered in heated channels with forced convection are the flow excursion and density wave oscillation types.

The flow excursion or Ledinegg instability is initiated (but, in some transient conditions, may not be completed) when the slope of the channel demand pressure drop-flow rate curve becomes algebraically smaller than or equal to the slope of the loop supply pressure drop-flow rate curve.

In research reactors using plate-type fuel, the core can be thought of as an array of many parallel channels. The supply characteristic with respect to flow perturbations in a channel, mainly in the hot channel, is essentially horizontal, and independent of the pump characteristics. Thus, the criterion of zero slope of the channel demand pressure drop-flow curve is a good approximation for assessing the onset of the excursive flow instability for quasi-static onset of the excursion, i.e.,

$$\frac{\partial\ (\Delta P)_{channel}}{\partial G} = 0$$

where ΔP is the pressure loss and G is the mass flux and given by the density of water times water velocity.

Functionally, the channel pressure drop-flow curve depends on the channel geometry, inlet and exit resistances, flow direction, sub-cooled steam void fraction and heat flux, along the channel.

Density wave oscillations are low-frequency oscillations in which the period is approximately the time required for a density wave to travel through the channel. Inlet flow perturbations in a heated channel result in a delayed mixture-density affecting the local mixture velocity and the pressure drop in a channel. Under certain conditions, the inlet flow perturbations satisfy a self-exciting relationship such that sustained oscillations with considerable amplitudes appear in the system.

There is a critical value for the inlet sub-cooling below which flow instability would be caused by density wave oscillations only and no flow excursion would occur.

For a high-pressure system, this critical sub-cooling value corresponds to a very low inlet temperature. However, for a low-pressure system, this critical sub-cooling value corresponds to a high inlet temperature very close to the saturation temperature.

For most research reactors, the steady state operating system pressure is low and the inlet coolant temperature is much lower than the saturation temperature. It can, therefore, be concluded that:

a) flow excursion would occur for a given flow rate only at a high enough heat flux
b) density wave oscillations will not occur under normal operating conditions

As a general rule, the operation of research reactors in the range of 1MW to 50MW, cooled and moderated by water at low pressures, is limited, from the thermal point of view, by the flow redistribution phenomenon. This phenomenon could lead to cladding failure and it may occur as a result of:

a) A temporary power increase.
b) A flow rate decrease such as, for example, a loss of flow in the primary coolant circuit.

The power generated inside the core is not uniformly distributed. In some channels (hot channels) more power must be removed and boiling could be reached much earlier than in the rest of the parallel channels.

Reactor operating conditions are defined by power, pressure, coolant inlet temperature, coolant flow rate and pressure drop through the core, variables that are common to all parallel channels.

The operating point of each channel is fixed by the intersection of two curves:

a) pressure drop through the channel as a function of flow rate
b) core pressure drop, which is a constant.

This channel characteristic according to the imposed flow conditions, coolant inlet temperature and pressure, has two inflection points. This leads to non-unique operating points for channels for some conditions, allowing the potential to move from one to the other.

If conditions are such that the flow rate decreases dramatically, with consequent significant increase in vapour bubble formation, the effect is known as a Ledinegg excursion or flow redistribution.

5.8.4.1.1.2 Prediction of Flow Redistribution Condition

The correlation used to predict flow redistribution is the Whittle and Forgan (1967) correlation, which estimates the redistribution power (P_{RD}) along the height of the core.

This power is divided by the total power generated in the channel, to obtain the minimum ratio to redistribution flow.

5.8.4.1.1.3 Assessment of the Predictive Model

This correlation, as implemented within the TERMIC programme of the MTR_PC package, was compared against experimental data resulting in good agreement as shown in Figure.5.8/1 (Doval, A, 1998).

5.8.4.1.2 Departure from Nucleate Boiling

5.8.4.1.2.1 General

For reactor design purposes, an acceptable prediction method for burnout heat flux is needed since Departure from Nucleate Boiling (DNB) is potentially a limiting design constraint.

Critical Heat Flux (CHF) correlations for water flowing in channels with the FA geometry under various conditions are commonly used as a prediction method.

At a high flow rate, burnout occurs due to vapour blanketing on the heated wall, and the resulting CHF depends only on the local conditions considering that the effect of flow orientation (up-flow or down-flow) is small. This burnout mechanism is a DNB type. The heat flux leading to this situation is named CHF, Burn-Out flux or DNB flux, q''_{DNB}.

At these conditions fuel plate wall temperatures are likely to reach the melting point.

5.8.4.1.2.2 Prediction of Departure from Nucleate Boiling Condition

A literature survey of Departure from Nucleate Boiling (DNB) correlations applicable to low-pressure research reactors points to the Mirshak correlation as one providing conservatively low DNB heat flux predictions for narrow rectangular channel flow. This correlation is recommended in the evaluation carried out in the US sponsored RERTR programme (IAEA, 1980), and is adopted for assessing the reactor DNBR in steady state conditions.

5.8.4.1.2.3 Assessment of Departure from Nucleate Boiling Correlation

The existing correlations and schemes for predicting the DNB in rectangular channels have been investigated (Sudo, Y. and Kaminaga, M., 1993; Sudo, Y., 1996; Kaminaga, M. et al., 1997). The effect of dominant factors on CHF was investigated for the existing CHF experimental data for flat-plate-type fuel.

To evaluate the Mirshak DNB correlation, predictions from it were compared with those from the Sudo and Kaminaga (1993) prediction method for conditions near those proposed for the reactor.

Both correlations show the same trend, with the Mirshak correlation prediction being more conservative.

An additional assessment was made using the experimental data from Yucel and Kakac (1978).

To consider the error of the predictive method, the standard statistical treatment of a finite sample size converges to $k\,\sigma = 1.64\,\sigma$, where k incorporates the uncertainties of the sample and σ is the standard deviation of the correlation prediction. This value provides a 95% probability that the CHF will not occur, with 95% confidence.

5.8.4.1.3 Low Flow Burn-out

5.8.4.1.3.1 General

A periodic instability known as "pulsed boiling" can appear for low power, low coolant velocities. The sequence of events characterising this phenomenon is as follows:

(i) Convection and Onset of Nucleate Boiling: the coolant inside the channel, where the maximum axial heat flux takes place, begins to heat up, bubbles appear and pressure increases.

(ii) Coolant expulsion: vapour flashes at both ends of the coolant channels.

(iii) Re-introduction: sub-saturated coolant re-enters the channel and the sequence starts again.

Pulsed boiling is a non-destructive phenomenon but its appearance means that the core will support an oscillatory cooling regime.

When the heat flux is approximately 2 to 4 times greater than the pulsed boiling heat flux, burnout appears. There is a large and permanent increase in the wall temperature due to a vapour film formed at the hot spot, leading to fuel damage.

5.8.4.1.3.2 Prediction of Low Flow Burn-out Condition

The burn-out heat flux (q_{BO}) for low flow can be calculated using the prediction given by the following correlations:

a) Fabrega (1971):

b) Sudo (1996):

The minimum value predicted from both correlations is adopted as a safety limit.

In the region where the mass flux is very low or the flow condition is a counter-current flow, the burnout heat flux is predicted by Mishima's equation

$$q_{BO}{}^{*} = 0.7 \frac{A}{A_H} \cdot \frac{\sqrt{W/\lambda}}{\left\{1 + (\rho_g / \rho_l)^{1/4}\right\}^2}$$

5.8.4.1.3.3 Assessment of Low Flow Burn-out Correlations

Extended experimental investigations in this region have been reported in several studies (Sudo, Y. and Kaminaga, M., 1993; Mishima, K. and Nishihara, H., 1985; Mishima, K. and Nishihara, 1987) for rectangular channels. A verification of previous correlations was made using the experimental data from Sudo and Kaminaga (1993) and from Mishima and Nishihara (1985). Figures 5.8/4 and 5.8/5 show the results of the measured value of burn-out against the mass flux and the correlation predictions. The test section was in both cases a rectangular channel heated from both sides. As can be seen, Sudo and Kaminaga's (1993) correlation fits the experimental points. Fabrega's (1971) correlation predicts a constant value, as it is independent of mass flux. Even so, the experimental data from Sudo and Kaminaga (1993) and from Mishima and Nishihara (1985) are for steady state, presumably, forced convection conditions. Fabrega´s (1971) and similar correlations (Gambill W. and Bundy R., 1964) give a minimum value of burn-out heat flux in pure natural circulation at intermediate-low flow, while Sudo (1996) gives a better prediction in low-flow high quality condition (X ~ 1). Under single phase natural circulation predictions from Fabrega's (1971) burnout correlation are sufficiently conservative. The Mishima prediction (extended by Sudo, Y., 1996) is the minimum value under flooding condition, that is, very low mass flux or flow reversal.

5.8.4.1.4 Fluid-Structure Interactions

5.8.4.1.4.1 General

When a high-speed flow passes through a gradually narrowing passage, the pressure head of the stream is converted into a velocity head, which creates a suction force on the wall. If the wall is movable or flexible, the flow area could be reduced to zero and the flow stopped. As soon as the flow becomes stagnant, the stream pressure increases, pushing the wall back, thus providing a wide flow area again. These suction and pushing forces can act periodically and thus make the structure vibrate.

This mechanism appears to govern the vibration of parallel fuel plates. These hydraulic vibrations can result in a large deflection of the plates, causing local overheating and possibly a complete blockage of the coolant channels.

5.8.4.1.4.2 Prediction of Critical Velocity

Parallel fuel plate vibrations have been analysed by Miller, among others, (see Nuclear Regulatory Authority (Argentina), 1998a) who derived a formula for critical velocity based on the interaction between the changes in cross-sectional areas, coolant velocities and pressures in two adjacent channels. According to Miller, the critical velocity above which significant vibration will occur for fuel plate assemblies with long edges attached to side plates is given by:

$$V_{crit} = \left[\frac{15 \times 10^5 \ E \ t_p^{\ 3} \ t_w}{\rho W^4 (1 - \upsilon^2)} \right]^{1/2}$$

where:

- V_{crit} : critical flow velocity (ms^{-1})
- E : modulus of elasticity (bar)
- t_p : fuel plate thickness (cm)
- t_w : water channel thickness (cm)
- ρ : density of water (kgm^{-3})
- υ : Poisson’s ratio (dimensionless)
- W : water channel width (cm).

5.8.4.2 Non-Safety Relevant Phenomena

In research reactors using plate-type FAs, specific thermal-hydraulic conditions are commonly used as design constraints in order to anticipate critical phenomena or to limit non-safety issues.

5.8.4.2.1 Onset of Nucleate Boiling

The Onset of Nucleate Boiling (ONB) is considered as the first indication of the potential for critical phenomena, and the heat flux that initiates ONB, q''_{ONB}, is frequently used as a thermal design constraint. The ONB is taken as a warning in steady state conditions as it does not actually correspond to any critical event.

The heat flux leading to ONB is calculated by means of the Bergles-Rohsenow correlation:

5.8.4.2.1.1 Wall Temperature Calculation

To calculate the wall temperature of the fuel plate along the hot channel it is necessary to calculate the heat transfer coefficient. Depending on the cooling regime different correlations are used.

5.8.4.2.1.1.1 Single-Phase Laminar Regime

With the aim of calculating the heat transfer coefficient for forced laminar regime, the following set of empirical equations (Shah and London, 1978) are used, depending on geometrical parameters and on Reynolds and Prandtl numbers:

5.8.4.2.1.1.2 Single-Phase Transition Regime

To evaluate the heat transfer coefficient h_c in the transition regime, the correlation proposed by Kreith was used

$$hc \cdot \Pr^{2/3} \cdot \frac{(\mu_{bulk} / \mu_{wall})^{-0.14}}{c_p \cdot V \cdot \rho} = \mathrm{A}$$

5.8.4.2.1.1.3 Single-Phase Turbulent Regime

The Nusselt number is calculated with Dittus Boelter's correlation:

Reynolds > 10000.

$$Nu = \frac{h_c \cdot D_h}{\lambda} = 0.023 \cdot \mathrm{Re}^{0.8} \cdot \Pr^{0.4}$$

5.8.4.2.1.1.4 Sub-cooled Boiling Laminar Regime

No boiling is expected under reactor nominal steady-state operating conditions. Therefore, correlations for nucleate boiling heat transfer and two-phase pressure drop, as well as for void fraction, are not included here

5.8.4.2.1.1.5 Sub-cooled Boiling Turbulent Regime

No boiling is expected under reactor nominal steady-state operating conditions. Therefore, correlations for nucleate boiling heat transfer and two-phase pressure drop, as well as for void fraction, are not included here

5.8.4.2.2 Saturated Boiling

For low flow condition, an additional warning accounting for maximum power is the “boiling power ratio” (BPR). It represents the ratio between the power, for the condition in which the exit temperature reaches the saturation temperature and the equilibrium quality is zero, and the integrated power in the hot channel. A simple energy balance using the present mass flux of the system is performed to calculate it:

$$BPR = \frac{m \cdot c_p \cdot (T_{sat} - T_{inlet})}{P_{\max}}$$

where:

m : Mass Flow (kg s^{-1})

c_p : Specific heat of the coolant (kJ kg^{-1} $°C^{-1}$)

T_{sat} : Saturation Temperature (°C)

T_{inlet} : Core's inlet temperature (°C)
P_{max} : Integrated power in the hot channel (kW)

5.8.5 Uncertainties in the Thermal Margin Calculations

Uncertainties in estimates are applied as a safety factor in thermal-hydraulic calculations. These uncertainties can be grouped into four categories:

a) Uncertainties due to reactor control and power.

b) Uncertainties in reactor geometry: geometric and material tolerances attained during fuel fabrication stage (tolerances, chemical composition, etc.) and during the reactor life (deformation, erosion, physical and chemical changes in materials and, specially, in the fuel).

c) Uncertainties due to thermodynamic local conditions: thermodynamic parameters depend on variables such as pressure, density, velocity, enthalpy and heat flux.

d) Uncertainties in correlations: empirical correlations have a defined uncertainty band.

A statistical method with direct error propagation is used to derive the equations to evaluate the uncertainty of the safety-related variables or functions. For this purpose, these variables *X* are considered arbitrary functions of independent parameters x_i, and it is also assumed that the errors for $x_1,\ldots,x_M$ are of statistical nature and those for $x_{M+1},\ldots,x_N$ are systematic .

$$E(X) = \sum_{M+1}^{N} \left| E(X, x_i) \right| + \sqrt{\sum_{i=1}^{M} [E(X, x_i)]^2}$$

5.8.5.1 Basic Selected Parameters for Uncertainty Analysis

The independent parameters selected, where uncertainties are postulated and the treatment used to combine them to calculate the global error of the redistribution power, are shown below.

Parameters
Fuel fabrication
Amount of U_{235} per fuel plate
Active surface of plates
Coolant channel gap
Homogeneity of Uranium in the meat
Uranium meat Thickness
Core inlet temperature fluctuation
Core inlet temperature fluctuation
Reactor Power control
coolant channel velocity
atmospheric pressure
water level
Additional margins
ONB correlation
Redistribution power correlation
Heat transfer correlation
Inlet temperature shift

Core power shift

The values adopted are consistent with fuel manufacturing tolerances and fluctuations in the operating conditions.

The design method based on the hot channel factor model (PPF) provides an additional safety margin since the probability is low that all adverse effects which are combined to give the conditions for the hot channel occur simultaneously at any one point in the core.

5.8.6 Thermal and Hydraulic Analysis of the Core for the Power State

The nominal reactor fission power in the Power State is 20 MW, and it is removed from the core by forced convection through the action of the PCS pumps. A minimum flow rate of 1900 m^3h^{-1} passes through the core. Refer to Chapter 6, Section 6.2, for a detailed PCS description.

5.8.6.1 Geometric Description of the Fuel Assembly and the Core

In order to comply with the core performance requirements a standard FA as described in Section 5.3 is considered for the analysis. The core configuration is described in Section 5.7.

5.8.6.2 Coolant Velocity

The coolant velocity through the FA must be established with a sufficient margin to the critical velocity.

As a design criterion, the coolant velocity is recommended to be $\leq$ 2/3 of the critical velocity.

The Miller correlation described in Section 5.8.4.1.4.2 is used to estimate the critical velocity

The Fuel Assembly design includes a structural reinforcement at the inlet end of the fuel plates that results in the critical velocity being much higher than that predicted by the Miller correlation. This component was not considered in the application of Miller correlation, so the predicted critical velocity for inner channels is a conservative estimate.

5.8.6.3 Modelling

Calculations were split in two parts: the first one dealing with hydraulic calculations and the second one with thermal-hydraulic calculations. Calculations were performed using the codes in the MTR_PC package, specifically, CAUDVAP for the first step and TERMIC for the second.

5.8.6.3.1 Hydraulic Calculations

In this step, the coolant velocity distribution inside the coolant channels and the core pressure drop for a given flow, coolant pressure and temperature are calculated.

To predict these parameters two different models are implemented:

a) A full core or lumped model to calculate the core pressure drop and an average velocity.

b) A second model considering only the fuel plates to calculate coolant velocities in the different core channels.

5.8.6.3.1.1 Model A: Full Core Model

According to the core configuration and geometry, and due to CAUDVAP input requirements, both FA and Control Rod Plates (CRPs) are modelled as single units, or channel types. Depending on CRP positions, they are considered either partially inserted (PI) or completely withdrawn (TO). Although the final results are largely unaffected by the CRP position it is relevant for the pressure drop distribution inside the Control Rod Guide Box (CRGB).

Three different types of channels were defined:

- FA
- CRP-PI
- CRP-TO

Each channel type is axially divided in regions, representing the geometric changes, and for each region the following parameters must be specified:

- Length
- Flow area
- Hydraulic diameter

For the FA channels, the regions are:

- Lower part of the Fuel Clamp (FC)
- Entrance box
- Lower top of the FC
- Upper top of the FC
- FC
- Nozzle region
- Window region
- Comb (Structural Reinforce) region
- Comb SR + Fuel Plate (FP)
- FP region
- Upper end of the FA

Figure 5.8/2 shows the different regions of the FA considered in the analysis.

A lumped flow area is defined for the FP region (x) of Figure 5.8/2, corresponding to the twenty internal channels and two external channels representing those channels between the external FP and the chimney, the CRGB walls or the adjacent FA sidewall, depending on the FA position. Figure 5.8/3 shows this lumped channel.

As all the channels are connected in a parallel array to a common inlet plenum and the chimney (outlet plenum), the same inlet and outlet pressures are assumed for the different channels, providing a good approximation for the average velocity calculations.

As FA window openings result in pressure equalisation, before entering the fuel plate region, the plate region flow distribution depends only on channel geometry. Experimental measurements performed in a dummy FA show that the velocity distribution in the internal channels is quite uniform.

For CRP-PI channels, the regions are:

CRGB entrance box

Grid region

Empty CRGB + CRP stem

Region of the guide box with the CRP+ CRP stem

Region of the guide box with the CRP

Empty CRGB

Restrictors

For CRP-TO channels the seven regions are the same as for CRP-PI channels, the only difference being the region lengths.

The core pressure drop is estimated from the fastener inlet up to the FA upper plenum.

5.8.6.3.1.2 Model B: Isolated Fuel Plate Model

In order to estimate the different velocities in the FA channels with different channel gaps a partial nodalisation is analysed.

In the analysis, all the coolant channels of a group of four FAs are considered. There are four different types of channels:

a) Internal: corresponding to the twenty internal channels.

b) External: it represents those channels between the external FP and the chimney/GB walls

c) Between two FAs: it corresponds to the channel formed by the external plates of a FA and the adjacent FA sidewall.

d) Lateral channels: formed by the sidewall of the FA and the chimney/CRGB walls

CRPs are not considered in this model. Figure 5.8/4 shows a scheme of these channels.

The total flow rate for cooling the fuel plates and the number of each type of channel is supplied as input data. With this data and the specified geometry, CAUDVAP calculates the velocity distribution for every type of channel.

5.8.6.3.2 Thermal-hydraulic Calculations

For this second step of calculations the TERMIC code is used to predict the critical heat fluxes for the coolant velocities determined from the hydraulic calculations.

The code calculates the heat fluxes for ONB, $q"_{ONB}$; the critical power for redistribution, P_{RD}, and the critical heat flux for DNB, $q"_{DNB}$

These predicted values allow for their dependence on coolant velocity, FA geometry, inlet temperature, flow direction, and water column above the top of the core.

5.8.6.4 Calculated Thermal-hydraulics Parameters for the Power State

Table 5.8/2 shows the velocities and core pressure drop from the fastener up to the FA outlet as calculated with CAUDVAP for the Power State.

The flow distribution is as follows:

a) 98.6% through the FAs

b) 1.4% through the CRGB

Figure 5.8/5 shows TERMIC best-estimate predictions for the hot channel temperature distribution along the wall, as well as saturation and ONB temperatures (including uncertainties). These predictions apply for the velocity of 8.2 ms^{-1} and for the nominal core power of 18.8 MW removed by the PCS.

Table 5.8/3 summarises calculated temperatures (best estimate temperatures) for the Power State.

5.8.6.5 Design Evaluation

From previous results the design bases parameters are calculated. Table 5.8/6 summarises those parameters.

Predictions are given in Table 5.8/4 for an equivalent nominal power of 20 MW. They are also given for the provisional core power safety system setting (corresponding to a reactor power of 23 MW). Under the same initial conditions, i.e. a core flow of 1900 m^3h^{-1} and coolant inlet temperature of 37°C, the maximum core power that can be removed by the PCS without exceeding the required safety limit corresponds to a reactor power of 31.9 MW

The analysis described above demonstrates that the minimum core flow of 1900 m^3h^{-1} fulfils the thermal-hydraulic design bases.

5.8.7 Thermal and Hydraulic Analysis of the Core for Shutdown State

An analysis has been carried out to determine the time after reactor shutdown when the PCS pumps can be stopped such that the ensuing core cooling by natural convection is adequate. The analysis verifies that the thermal-hydraulic parameters fulfil the thermal-hydraulic design bases.

After the reactor is shut down, the decay power in the core rapidly decreases to values lower than 10% of the nominal reactor power. This low power level can be readily removed by just one PCS pump. A minimum time interval must be defined before this PCS pump stops, flap valves open and natural circulation is established. Refer to Chapter 6, Section 6.2, for further information on PCS operation modes.

This time interval is set by the decay power that can be removed by the coolant by natural circulation without violating the design criterion.

The adopted power is around 400 kW and corresponds to the decay power reached 30 min after reactor shutdown. This time is estimated from the curve of decay power calculated according to ANSI/ANS-5.1-1979 standard, for the core operating for indefinite time and assuming an adequate safety factor. Figure 5.8/6 presents the decay power curve normalised to the nominal core power.

One of the two pumps can be manually disconnected once the reactor is shutdown while the remaining pump would normally be kept running for some 30 minutes.

Thermal-hydraulic calculations are carried out to verify that the core is properly cooled under these conditions. These calculations are performed using the MTR_PC package.

5.8.7.1 Modelling

Steady state calculations are performed with CONVEC (in MTR_PC package) in order to verify that the core is properly cooled in the Shutdown State.

This programme balances the forces acting in the single-phase momentum equation calculating the resultant flow established by the buoyancy along the coolant channel.

For a given geometry of the loop, boundary conditions (such as the pool water temperature and the pressure at the top of core), and the power per FA, the natural circulation flow and the wall and coolant temperature distributions in the average and hot channels are calculated. Additional data such as the maximum heat flux and the burn-out heat flux are also given.

The initial conditions for these calculations for the shutdown state are shown in Table 5.8/5.

The core of the reactor is cooled by the coolant flowing in the upward direction. Once the PCS pumps stop and the flap valves open, the ensuing natural circulation is in the same direction, so that no flow reversal takes place.

If the pump stops before 30 minutes after the reactor shuts down, the pump flywheel, as well as the flap valves, provide adequate core cooling during the transition to the natural convection regime.

5.8.7.2 Engineering Factors

Uncertainties are included in the calculations described above. Engineering factors were included in the calculation and treated deterministically. These factors consider uncertainties in the heat flux and heat transfer correlation.

5.8.7.3 Calculated Thermal-hydraulic Parameters for the Shutdown State

From the initial conditions of Table 5.8/5, and adopting a conservative scheme for carrying out design calculations (i.e. by affecting the characteristics of the system to a less favourable situation), the results presented in Table 5.8/6 are obtained. Figure 5.8/7 shows the temperature distribution in the hot channel for the coolant and on the wall.

As shown in Table 5.8/6, the BOR is equal to 5.3. Table 5.8/7 shows the BOR calculated for different times after shutdown.

It is worth noticing that the calculations show that to avoid the onset of nucleate boiling, a minimum time of only 15 minutes after shutdown is required before the second PCS can also be stopped. .

If a PCS pump remains running more than 30 minutes after reactor shutdown, the BOR becomes larger as the decay power decreases.

5.8.7.4 Design Evaluation

A power around 400kW can be removed in the natural circulation regime. This power is reached 30 minutes after reactor shutdown.

For a normal reactor shutdown transition, the reactor operator is required to manually stop one of the PCS pumps once the reactor is shutdown, and then wait for a minimum time of 30 min after reactor shutdown before manually stopping the second pump. After that time, the flap valves open and natural circulation establishes, assuring core cooling. In the long term, the normal procedure is to remove the decay heat from the Reactor Pool by means of the Long Term Pool Cooling Mode of the RSPCS. Alternatively, the decay power can be transferred to the large water reservoir present at the Reactor Pool.

A minimum time of 15 minutes after shutdown is required to avoid the occurrence of ONB under natural circulation conditions.

In the Shutdown State, the BOR exceeds the design criterion by a very large margin. This ensures adequate cooling of the core.

If the pump fails before 30 minutes after reactor shutdown, core cooling is assured by the engineered safety features considered in the design, such as the pump flywheel and flap valves.

On the other hand, if a pump is to remain running for more than 30 minutes after shutdown, the BOR becomes larger, but this does not imply better cooling conditions.

5.8.8 Thermal and Hydraulic Analysis of the Core for the Refuelling State

The Refuelling State is a nominal zero power state. Any heat removal requirement due to the long-term decay heat is removed by natural convection. Calculations made for the Shutdown State cover this case.

5.8.9 Thermal-hydraulic Analysis of the Core for the Physics Test State

A core power of 400 kW can be removed by natural circulation. This is similar to the decay power to be removed by natural circulation 30 minutes after shutdown. The analysis of 5.8.7 described above indicates that the core power released in Physics Test State can be removed by natural circulation.

As with the Shutdown State, the BOR in the Physics Test State exceeds the design criterion. This ensures adequate cooling of the core.

5.8.10 Thermal and Hydraulic Analysis of the Control Rod Plates and Control Rod Guide Box

The cooling of CRPs must be analysed to define the appropriate flow, which must be neither too high nor too low.

A high flow can represent a significant core bypass or exert an unacceptable drag force on the plate from a safety point of view. On the other hand, a low flow value can result in wall temperatures exceeding the Onset of Nucleate Boiling temperature design criterion.

To comply with the design basis and the design criterion the analysis was split into two parts, hydraulic and thermal-hydraulic.

5.8.10.1 Design Analysis

The maximum allowable coolant velocity to avoid the dragging of the CRP and the pressure drop distribution inside the CRGB are determined.

5.8.10.1.1 Allowable Velocity

Figures 5.4/2 and 5.4/3 show the CRPs inside the CRGB.

The flow area between the CRP and the CRGB, the CRP weight and the area normal to the coolant flow direction, besides the total core pressure drop, are required to determine the velocity at which the upward drag on the CRP becomes unacceptable.

To comply with a drag of 75% of the CRP weight, the maximum allowable pressure drop along the plate is approximately 42 kPa.

The maximum coolant velocity giving a 42 kPa pressure drop in the CRP region is predicted by allowing for friction on the walls of the plate and pressure changes due to the area changes at the entrance and exit regions of the plate. This maximum allowable velocity between the CRPs and the CRGB is 4.0 ms^{-1}, (for the Safety and Compensating plates) and 3.8 m s^{-1} (for the Safety and Regulating plate).. A maximum allowable velocity of 3.8 m s^{-1} was adopted.

5.8.10.1.2 Estimation of the Pressure Drop Distribution

Since the FA and the CRPs are placed inside the same plenum, they have the same pressure drop between the coolant inlet and outlet.

Calculations performed for the most demanding conditions, give a total core pressure drop of 270 kPa and a coolant velocity inside the CRGB of 2.7 m s^{-1}, in the CRP region.

The elastic deformation of the CRGB, caused by the pressure difference, was conservatively considered by adopting a design with a positive higher pressure inside the CRGB. For this reason restrictions to limit the flow are located at the outlet of the CRGB.

To verify that the pressure distribution along the control plates and inside the guide box complies with the safety requirements, a calculation was performed using CAUDVAP from the MTR_PC System. A full core model was considered in which the sixteen FAs were represented as in Section 5.8.6.

The FA regions are as presented in Figure 5.8/2.

The CRPs and the CRGB were divided in five different regions listed below and shown in Figure 5.8/8.

The same five regions apply for the four Safety and Compensating plates and for the Safety and Regulating plate.

- (i) Entrance region
- (ii) Region with the follower
- (iii) CRP region
- (iv) Empty CRGB region
- (v) Region with restrictions

Due to the condition on the pressure drop stated at the beginning of this section, the same pressure was assumed for all channels in the CRGB. This is a good approximation for calculations of the average velocity.

The pressure distribution along the FAs and inside the CRGB is given in Figure 5.8/17. This figure shows that the pressure inside the CRGB is always higher than the pressure in the FAs. The x-axis indicates absolute pressure, i.e., dynamic, frictional and hydraulic pressure components. The same pressure distribution applies for the cold state.

5.8.10.2 Calculated Thermal-hydraulic Parameters for the Control Rod Plate and Control Rod Guide Box

The maximum heat flux that can be removed in the control plate and guide box is determined here for the velocity calculated above.

For the lower velocity between the Safety and Regulating CRP and the Safety and Compensating CRP, the heat flux leading to ONB, (**q''_{ONB}**), is equal to 84 Wcm^{-2}.

A total heating in the Safety and Regulating CRP was calculated giving a maximum heat flux of 34.5 W cm^{-2}. The ratio between q''_{ONB} and the maximum heat flux is equal to 2.4, which largely fulfils the design criterion.

The CRP wall temperature result was around 80°C.

5.8.10.3 Design Evaluation

To establish cooling conditions fulfilling the design bases and criteria during the steady state for the CRPs and for the CRGB, a thermal-hydraulic analysis has been carried out:

a) The maximum allowable cooling velocity inside the CRGB, over the CRPs (region III), is 3.8 ms^{-1}.

b) Calibrated restrictions at the outlet of the CRGB provide a positive pressure difference between the inside and the outside faces of the CRGB.

c) The ratio between the q''_{ONB} and q''_{max} for nominal conditions is 2.4.

5.8.11 Thermal and Hydraulic Analysis of the Irradiation Facilities for the Power State

A thermal-hydraulic analysis of irradiation rig cooling for the reactor Power State has been performed to ensure adequate cooling of the bulk production and the large irradiation facilities, and to estimate the pressure drop along these rigs. A description of the analysis is presented in this Section.

One of the main functions of the Reactor Facility is the production of radioisotopes. For this function there are several target materials at different Irradiation Facilities located at different positions in the Reflector Vessel (A description of the Reflector Vessel is given in section 5.2.4.6). Irradiation Facilities (IF) can be grouped into:

a) Bulk Production IFs

b) Long Residence Time General Purpose IFs

c) Short Residence Time IFs

d) Large Volume IFs

Materials in these facilities produce heat that is removed by the RSPCS. The RSPCS capability allows the utilisation of a wide variety of target designs, being remarkable the flexibility of the system to adjust the cooling flow on each individual position by the utilisation of flow restrictors.

For each target and rig geometry to be used in the irradiation positions, thermal-hydraulic calculations should be performed to demonstrate that an adequate cooling condition can be achieved with the RSPCS capability.

As an example, this section includes the thermal-hydraulic analysis performed on specific geometries for the bulk productions.

Bulk Production and Large Volume IFs have been analysed at this stage as they represent the facilities having the highest heat flux. The heat fluxes from Long Residence Time General Purpose and Short Residence Time IFs are estimated to be low and the cooling of these facilities is accommodated within the design margins given for the RSPCS.

5.8.11.1 Description of the Irradiation Facilities

5.8.11.1.1 Bulk Production Irradiation Facilities

Seventeen irradiation tubes are placed vertically inside the Reflector Vessel. They contain Uranium-235, Tellurium dioxide and Iridium metal to produce Molybdenum-99, Iodine-131 and Iridium-192.

These tubes provide a physical independence between the irradiation rigs and the Reflector Vessel, described in section 5.2.4.6. They allow light water from the pool to remove the heat generated in the target by flowing downwards between the rigs and the tubes.

Each of the seventeen-irradiation tubes contains a rig with up to five irradiation targets. Twelve of these facilities are designed for Molybdenum-99 production, three for Iodine-131 production and two for Iridium-192 production.

A maximum heat flux equal to 110 W cm^{-2} and a total power per rig of 125 kW are the design conditions for the Mo production facility.

The power limit is used to calculate the temperature rise of the coolant along the rig. The maximum heat flux limit is considered to calculate the wall temperature of the targets and verify that the estimated coolant velocity is enough to fulfil the design criterion.

The total rig power and thermal loads on those rigs containing Tellurium dioxide and Iridium metal were estimated according to the maximum neutron flux in the irradiation position in the reflector vessel and were conservatively increased by an engineering factor.

5.8.11.1.2 Large Volume Irradiation Facilities

Large Volume Irradiation Facilities (LVFs) are provided for neutron transmutation doping of single-crystal silicon ingots.

A total of six irradiation rigs are placed in the reflector tank tubes, in a similar way to the bulk production IFs. Two are of large diameter, one of medium diameter and three of medium large diameter.

As with the bulk production IFs, a geometry illustrative of these facilities is adopted for calculation purposes.

The most demanding facility from the thermal-hydraulic point of view was adopted as the envelope case. A total power of 8 kW was assumed and it corresponds to one of the medium-large diameter facilities. The same minimum velocity was defined for all the facilities.

5.8.11.2 Modelling

5.8.11.2.1 Thermal Analysis

A thermal analysis provides predictions of the maximum wall temperature reached in the envelope rig. The rig power is removed by the RSPCS. The main features of the RSPCS provide a basis for predicting the coolant temperature increase and the reference temperature, which are required for physical property evaluation.

The total thermal load removed by the RSPCS consists of the thermal loads listed in Tables 5.8/8, together with the thermal load due to the energy deposited in the Long and Short Residence Time General Purpose facilities. A preliminary conservative estimate of the total thermal load is approximately 2100 kW. This value considers, also, the heat load in the Service Pool.

The input conditions imposed by the RSPCS are given in Table 5.8/8.

5.8.11.2.1.1 Velocities and Wall Temperature Calculation

Targets inside the rigs must be cooled without exceeding the design criteria. Coolant velocities are related to the wall-coolant temperature difference via a heat transfer coefficient calculated from the Dittus–Boelter correlation:

$$Nu = 0.023\, Re^{0.8}\, Pr^{0.4}$$

(standard notation), which is valid for Reynolds numbers greater than 10000. Reynolds numbers for the Bulk production IFs are around 40000.

For laminar flow, (Re < 2100), the Sieder Tate correlation (see Krieth, F., 1973) is adopted

$$Nu = 1.86\,(Re\,Pr\,\mathrm{D_h/L})^{0.33}\,(\mu_f / \mu_w)^{0.14}$$

where $\mathrm{D_h}$ is the equivalent hydraulic diameter, L is the heated length, μ_f is the coolant viscosity at the fluid temperature, and μ_w is the coolant viscosity at the wall temperature.

5.8.11.2.1.1.1 Safety margin and Engineering Factors

As stated in 5.8.3.5, a margin between the heat flux at which nucleate boiling is initiated and the maximum heat flux in each rig is adopted.

Heat transfer coefficients are weighted with an engineering factor for uncertainty purposes.

5.8.11.2.1.1.2 Sequence of the Analysis

Depending on the thermal load of each rig and using the same temperature rise, a partial flow rate is calculated.

With the assumed geometry and the partial flow, the coolant velocity is estimated and the temperature difference between the coolant and the wall, the wall temperature and the pressure drop are also calculated.

5.8.11.2.1.2 Onset of Nucleate Boiling Temperature

The Onset of Nucleate Boiling (ONB) is considered as an indication that the critical phenomena may be approached. Hence the heat flux that initiates ONB, q''_{ONB}, is frequently used as a thermal design constraint for steady state condition. Although ONB is taken as a limit in steady state conditions, it does not actually correspond to any safety related phenomenon.

The heat flux leading to ONB is calculated by means of the Bergles-Rohsenow correlation (see 5.8.4.2.1)

5.8.11.2.1.3 Minimum Velocities

The minimum velocities in the Bulk Production IFs given as example fulfilling the design criterion are calculated before analysing the flow distribution among the different rigs.

5.8.11.2.2 Hydraulic Model

An integral model of all the rigs, based on the geometries previously presented, was constructed to predict the total pressure drop of the system and the flow distribution among the different rigs.

Following the CAUDVAP structure, several parallel resistances representing the IFs were connected to an inlet plenum, that is the pool, and to an outlet plenum provided by the suction box.

In order to comply with the operational design requirement, a variable number of empty channels were also defined.

For completeness, additional resistances representing the cooling flow for the Long and Short Residence Time General Purpose Facilities and the Surveillance Program Probe positions were included in the model to properly simulate the flow distribution. A scheme is given in Figure 5.8/9.

5.8.11.2.2.1 Restrictions

Restrictions have been introduced so that operational removal of rigs does not result in inadequate cooling of the remaining rigs.

A set of restrictions was estimated to comply not only with the minimum required velocities for the Mo-99 IF, but with the operative design requirement dealing with the unperturbed cooling condition, as well.

These hydraulic restrictions have been designed allowing that up to two Mo and one Iodine positions are empty. If this number is exceeded, additional restrictions or plugs must be placed in the empty positions.

5.8.11.2.3 Calculated Thermal-hydraulic Parameters for the Irradiation Facilities

From the operation conditions and the assumed preliminary rig geometries, velocities and pressure drops were estimated for the different rigs and a given number of empty positions. From maximum heat fluxes, q”, and conditions stated in Table 5.8/8, the minimum coolant velocities were estimated to fulfil the design criterion regarding the margin to ONB heat flux.

For the calculations targets in the fast and short residence time facilities were loaded.

5.8.11.3 Design Evaluation

Both the Bulk production and the Large Volume IFs were analysed in order to verify that they were properly cooled for the nominal conditions of the reactor Power State.

Illustrative geometries were assumed for cooling purposes estimations.

The design allows for a certain number of irradiation tubes to be without irradiation rigs without jeopardising the cooling of the targets being irradiated.

Velocities through the IFs were estimated so as to fulfil the design criterion that, as well as other design requirements being satisfied, the maximum wall temperature and the corresponding heat flux will remain below q''_{ONB} for a given number of empty rig positions.

An integral model was defined to estimate the pressure drop and the flow distribution for the different rigs.

5.8.12 Thermal and Hydraulic Analysis of the Irradiation Facilities for Shutdown State

When the reactor is shut down, the RSPCS pump is expected to remain operating for some minutes. This time interval is recommended for operational purposes to remove the decay power on the rigs by the RSPCS. After this time, the RSPCS pump is shut down, so the downward flow through the irradiation tubes is retained by the coast down of the pump until the flap valves on the RSPCS line open. As the forced flow in the irradiation tubes is downwards, there is an inversion of the flow through the irradiation tubes during the transition to upward natural circulation flow. Details on the operation of the RSPCS for this transition are given in Chapter 6, Section 6.3.

The demonstration that the rigs are properly cooled is covered by the case of loss of flow in the RSPCS given in Chapter 16, Section 16.15. This case assumes that the RSPCS pump is stopped and the reactor is shut down due to low RSPCS flow.

5.8.13 Thermal and Hydraulic Analysis of the Irradiation Facilities for Refuelling State

Apart from long term decay, no power is deployed or generated in the Irradiation Facilities during the Refuelling State, as this is a zero power State.

Any long-term decay heat is removed by natural convection. The arguments given for the Shutdown State (Section 5.8.12) cover this case.

5.8.14 Thermal and Hydraulic Analysis of the Irradiation Facilities for Physics Test State

The demonstration that, for this power level, the rigs are properly cooled is covered by the case of loss of flow in the RSPCS due to a loss of power supply event given in Chapter 16, Section 16.15.

5.8.15 Thermal-hydraulic Effects of Anticipated Operational Occurrences

The evaluation of the core capability to withstand the thermal effects resulting from anticipated operational occurrences, and from postulated accidents, is covered in Chapter 16.

5.8.16 Calculation of Thermal-hydraulic Limiting Conditions

This section presents results of calculations performed for assessing the safety limits, safety system settings, and safety margins for the thermal-hydraulic parameters of the reactor core during the Power State.

Calculations have been performed using the tools described in Section 5.10, with the uncertainty factors discussed in Section 5.8.5 and for steady state conditions.

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End of Section

Table 5.8/1 Summary of Thermal and Hydraulic Design Bases

Parameter	Value
Power peaking factor (PPF)	≤ 3.0
Safety Relevant Design Bases	
Critical velocity ratio	≤ 2/3
Burn-Out ratio (BOR)	≥ 2.0
Flow Redistribution ratio (RDR)	≥ 2.0
Departure from Nucleate Boiling ratio (DNBR)	≥ 2.0

Table 5.8/2 Calculated Hydraulic Parameters for the Power State

Parameter	Value
Core Flow Rate (m^3h^{-1})	1900
Coolant Average Velocity Through the Core (ms^{-1})	8.2
Core pressure drop (kPa)	205

Table 5.8/3 Calculated Temperatures for the Power State

Parameter	Value
Minimum saturation temperature (ºC)	118
Coolant Exit Temperature (ºC)	
Average channel	45.6
Hot Channel	60.9
Maximum wall temperature (ºC)	
Average channel	60.4
Hot Channel	98.4

Table 5.8/4 Design Basis Fulfilment for minimum core flow (*)

Phenomena	Design Basis	Design Basis
	Predictions for an equivalent core power of 20MW	Predictions for core nominal safety system setting
PRD/RDR	2.28 (&)	1.97
DNBR	2.58 (&)	1.96
Critical Velocity Ratio	0.55	0.55

(*) Acronyms are defined in Table 5.8/1

(&) with uncertainties.

Table 5.8/5 Initial Conditions for the Shutdown State Natural Convection Thermal-hydraulic Calculations

Parameter	Value
Decay power	420 kW
Time after reactor shutdown	30 min
Power Peaking Factor	3
Pool water temperature	40°C

Table 5.8/6 Calculated Thermal-hydraulic Parameters for the Shutdown State

Parameter	Value
Average coolant temperature rise	10
Core pressure drop	159 Pa
Maximum wall temperature in the hot channel	93 °C
Maximum heat flux	4.9 Wcm^{-2}
Burn-out heat flux	26 Wcm^{-2}
Boiling power ratio	4.4
Burn-out ratio (BOR)	5.3

Table 5.8/7 Shutdown State – Burnout Ratios at Different Times after Reactor Shutdown

Time After Shutdown (s)	Decay Power (kW)	BOR
100	958	2.3
300	637	3.5
600	545	4.1
1200	467	4.8
1800	416	5.4

Table 5.8/8 Input Conditions Imposed by the Reactor & Service Pool Cooling System for the Power State Thermal-hydraulic Analysis of the Irradiation Facilities

Parameter	Value
Coolant	Light water
Flow direction	Downwards
Inlet Temperature (ºC)	37
Inlet Pressure (kPa)	200

Table 5.8/9 Minimum Allowed Velocities in the Bulk Production Irradiation Facilities in the Power State

Bulk Production Irradiation Facility	ONBR
Molybdenum	1.33
Iodine	1.36
Iridium	1.33

Table 5.8/10: Velocities, Mass Flow and Temperature Differences for the Example Targets in Physics Tests State

	Mo	Te	Ir	Si(Very)	Si(Large)	Si(Med)
m (kg s-1)	$7.14\ 10^{-1}$	$6.9\ 10^{-3}$	$1.32\ 10^{-2}$	$1.4\ 10^{-2}$ - $5.\ 10^{-2}$		
ΔT (°C)	12	2.7	0.43	3		

Table 5.8/11 Parameter Values at Safety Limits

Point	Equivalent Core power [MW]	Core flow [m^3h^{-1}]	Core inlet temperature [C]
Single parameter variations from nominal conditions of point 4			
1	20	1900	70
2	20	1045	37
3	31.9	1900	37
4	**20**	**1900**	**37**
Parameter variations from provisional safety system settings of point 5			
5	**23**	**1710**	**43**
6	23	1710	56
7	23	1355	43
8	27.6	1710	43
9	23	1900	61
10	29.6	1710	37
11	20.7	1710	62
12	23	1241	37

Table 5.8/12 Safety Limits

Parameter	Limits	Nominal value
Total fission Core power (MW)	31.9	20
PCS flow rate (m^3h^{-1})	1045	1900
Core inlet temperature (ºC)	70	37

End of Tables

Figure 5.8/1 Comparison between Redistribution Heat Flux Experimental Data and the Correlation Implemented in the TERMIC Code

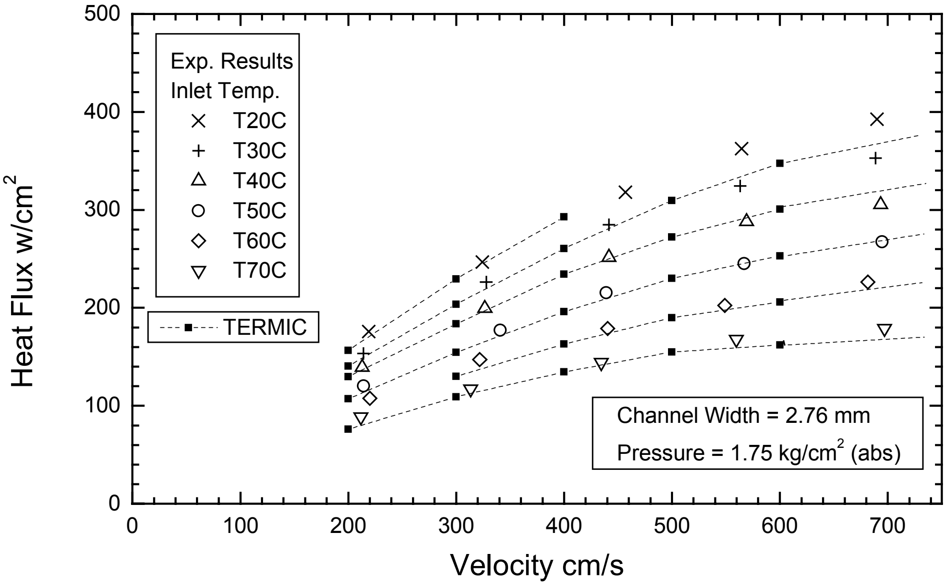


Figure 5.8/7 **Shutdown State – Temperature and heat flux Distributions in the Hot Channel**

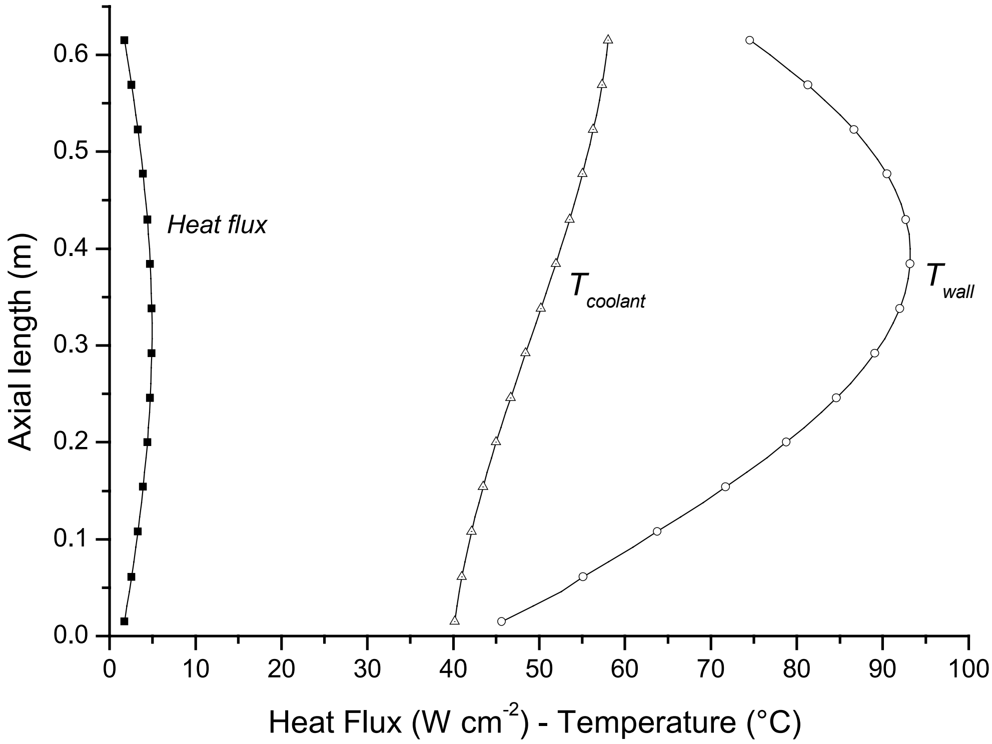


Figure 5.8/8 Control Rod Plate and Control Rod Guide Box Thermal-hydraulic Model

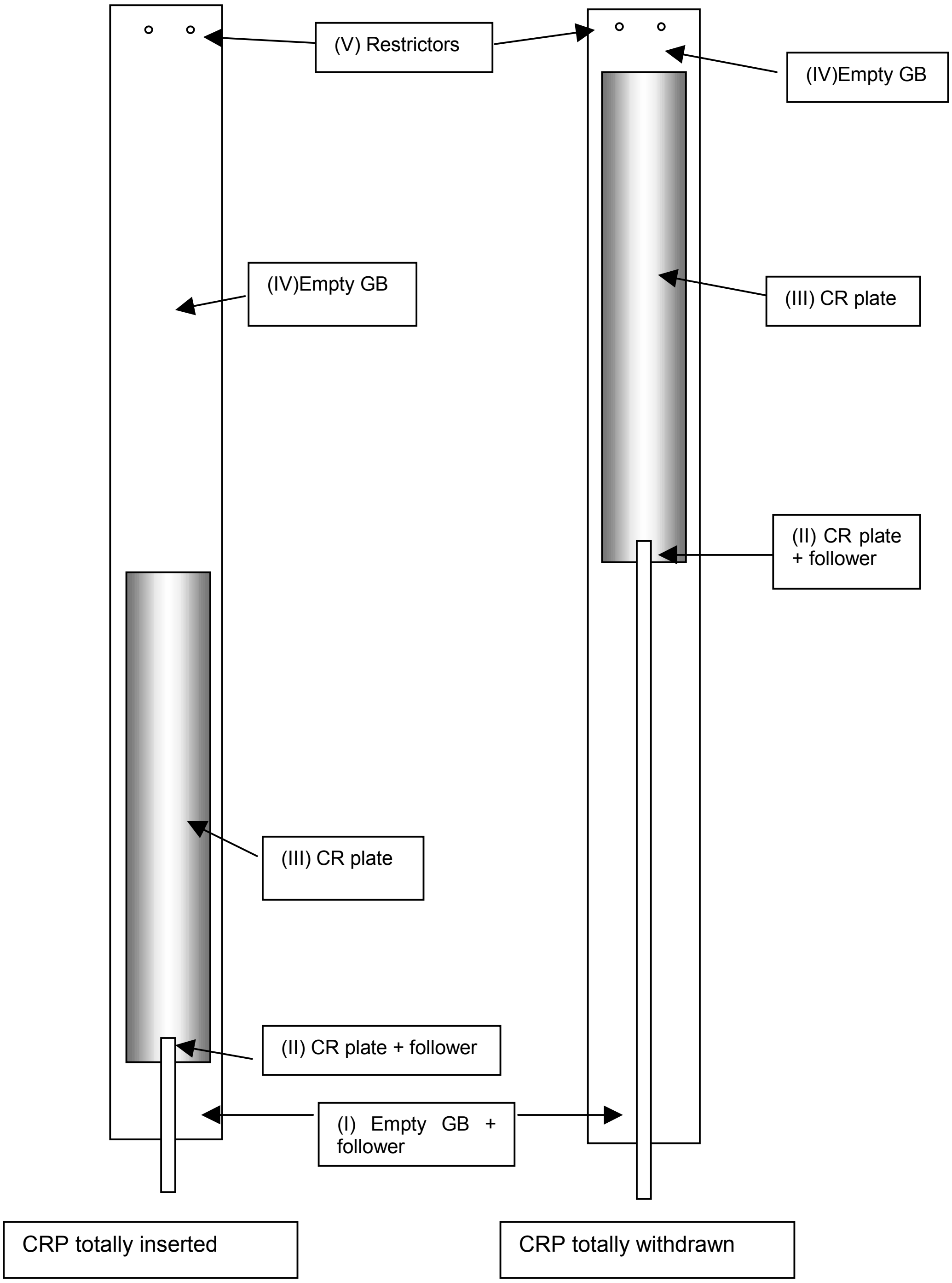
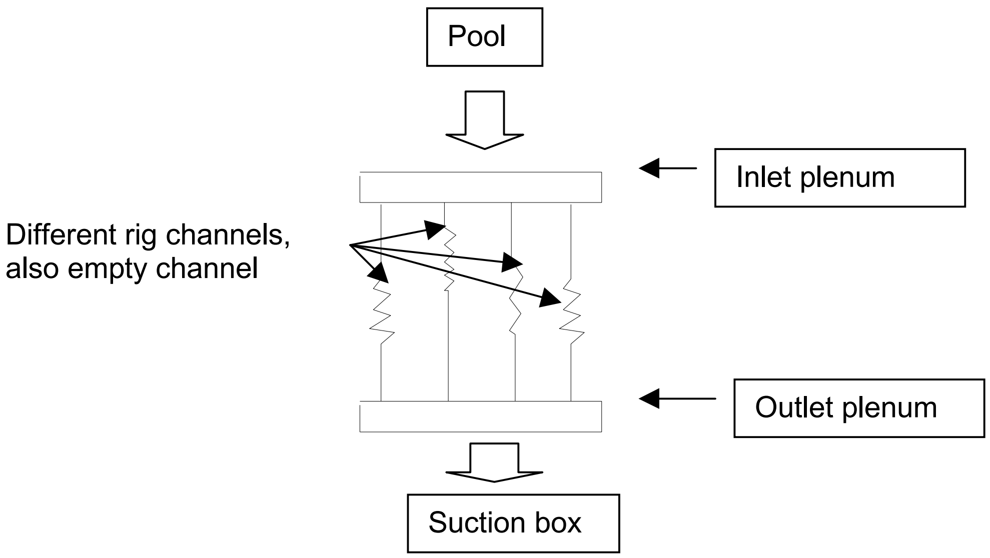


Figure 5.8/9 **Integral model for the Hydraulic Analysis of IF**



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