SUB-CHAPTER D.4 CORE THERMAL-HYDRAULIC DESIGN

0. SAFETY REQUIREMENTS

0.1. SAFETY FUNCTIONS

The safety functions carried out by thermal and hydraulic design are:

- removal of heat produced in the fuel via the coolant fluid,
- containment of radioactive substances (actinides and fission products) inside the first containment barrier.

0.2. FUNCTIONAL CRITERIA

0.2.1. Control of reactivity of the core

No impact on the thermal-hydraulic design.

0.2.2. Removal of the heat produced in the fuel

The thermal and hydraulic design must enable the removal of heat produced in the core by maintaining an efficient transfer between the fuel rods and the coolant fluid under normal and incident operating conditions.

0.2.3. Containment of radioactive products

Ensuring that there is no departure from nucleate boiling under incident (PCC-2) conditions ensures that the leak tightness of the fuel assembly is not compromised.

0.3. DESIGN REQUIREMENTS

Formally, core thermal-hydraulic design is not governed by specific codes and standards. However, the safety functions that it ensures require the application of a quality assurance program whose aim is to document and monitor all activities related to design.

0.4. TESTS

0.4.1. Pre-operational tests

The underlying features of the selected scenarios in the safety analyses must be checked during the first physical core tests. Some of these tests, such as verification of the reactor coolant flow rate or the drop time of the control rod assemblies, are carried out regularly. Other tests are only carried out in full, on commissioning of a lead plant unit.

For the later units, the only tests needed are those ensuring that the thermal hydraulic characteristics of the core are identical to those of the core in the lead plant.

0.4.2. In service monitoring

The reactor coolant flow rate and the control rod assembly drop time must be measured regularly.

0.4.3. Periodic tests

Not applicable

1. DESIGN BASIS

The overall objective of the thermal and hydraulic design of the reactor core is to provide adequate heat transfer compatible with the heat generation in the core, such that heat removal by the reactor coolant system or the safety injection system (when applicable) ensures that the performance and safety criteria requirement in Chapter D.3.0 are met.

In order to satisfy these criteria, the following design basis has been established for the thermal and hydraulic design of the reactor core.

1.1. PROTECTION AGAINST DEPARTURE FROM NUCLEATE BOILING

The requirement is that there is at least a 95% probability at a 95% confidence level that departure from nucleate boiling (DNB) will not occur on the limiting fuel rods during normal operation and operational transients and in any transient conditions arising from faults of moderate frequency (PCC1 and PCC2 events).

By preventing DNB, adequate heat transfer is ensured between the fuel clad and the reactor coolant, thereby preventing clad damage as a result of inadequate cooling. Maximum fuel rod surface temperature is not a design basis parameter as it will be within a few degrees of coolant temperature during operation in the nucleate boiling region. Limits provided by the nuclear control, limitation and protection systems are such that this design requirement will be met for transients associated with PCC2 events. There is an additional DNBR margin in rated power operation and during normal operating transients.

The use of simplified RFTC [DNBR] on-line calculations in the protection system and in the surveillance system enables the design criterion to be met by defining a low RFTC [DNBR] for reactor trip (DNB_{RT}) and a Limiting Condition of Operation (DNB_{LCO}) with regard to DNB, directly based on the reconstructed variable representative of the phenomenon to avoid.

The on-line calculated values are provided by systems which use measurements to reconstruct the local conditions using an algorithm, and apply the chosen Critical Heat Flux predictor to calculate the DNBR.

The reconstruction uncertainties and the measurement accuracy are considered when establishing the setpoints for the on-line DNBR calculated value. The setpoints are set such that there is a 95% probability at a 95% confidence level that DNB will not occur when the DNBR on-line calculated value is equal to the DNBR threshold.

1.2. FUEL TEMPERATURE

During modes of operation associated with PCC1 and PCC2 events, there is at least a 95% probability that the peak W/cm fuel rods will not exceed the fuel melting temperature at the 95% confidence level.

The melting temperature for UO_2 is 2800°C for an unirradiated element; the melting temperature for MOX is 2737°C for an unirradiated element. These values decrease with burn-up.

By precluding fuel melting, the fuel geometry is preserved and possible adverse effects of molten fuel on the cladding are eliminated.

1.3. CORE FLOW DESIGN BASIS

A minimum of 94.5% of the thermal flow rate passes through the fuel rod region of the core and is effective for fuel rod cooling. Coolant flow through the thimble tubes, as well as the leakage from the core barrel-baffle region into the core is not considered effective for heat removal.

Core cooling evaluations are based on the thermal flow rate (minimum flow) entering the reactor vessel. Under hot dome conditions, a maximum of 5.5% of this flow may be allocated as bypass flow. This includes rod cluster control guide thimble cooling flow, head cooling flow, baffle leakage, and leakage to the vessel outlet nozzle.

1.4. HYDRO-DYNAMIC STABILITY OF THE CORE

The operating modes associated with PCC1 and PCC2 events must not lead to hydro-dynamic instability of the core.

2. DESCRIPTION OF THE LIMITING PHYSICAL PHENOMENA – DESIGN CRITERIA

2.1. SUMMARY

Values of pertinent parameters along with critical heat flux ratios, fuel temperatures, and linear heat generation rates are presented in D.4 TAB 1 for all coolant loops in service. The reactor is designed to achieve a defined minimum RFTC [DNBR] and to avoid fuel centreline melting in normal operation, operational transients, and faults of moderate frequency.

2.2. CRITICAL HEAT FLUX RATIO OR DEPARTURE FROM NUCLEATE BOILING RATIO AND MIXING TECHNOLOGY

The minimum RFTC [DNBR] in the limiting flow channel will occur downstream of the peak heat flux location (hot spot) due to the increased downstream enthalpy rise.

The RFTC [DNBR] is calculated using the correlation and the definitions described in 2.2.1 and 2.2.2 within this Sub-chapter. The FLICA calculation code is used to determine the distribution of flow in the core and the local conditions in the hot channel for use in the RFTC [DNBR] correlation.

2.2.1. Departure from nucleate boiling technology

1. Critical heat flux - Correlations

Early experimental studies of DNB were conducted with fluid flowing inside single heated tubes or channels and with single annulus configurations with one or both walls heated. The results of the experiments were analyzed using many different physical models for describing the DNB phenomenon, but all resultant predictors were highly empirical in nature.

As testing methods progressed to the use of rod bundles, instead of single channels, it became apparent that the bundle average flow conditions could not be used in DNB prediction. Test results showed that predictions based on average conditions were not accurate for DNB heat flux. This indicated that knowledge of the local sub-channel conditions within the bundle is necessary.

In order to determine the local sub-channel conditions, the FLICA code was developed. In this code a rod bundle is considered to be an array of sub-channels each of which includes the flow area formed by four adjacent rods. The sub-channels are also divided into axial steps such that each may be treated as a control volume. The local fluid conditions in each control volume are calculated by solving simultaneously the mass, energy, and momentum equations. The predicting critical heat flux is performed by using the sub-channel local fluid conditions calculated by the design code and the FC correlation.

2. Use of test data for the FTC [CHF] correlation

The experimental basis of the FTC [CHF] correlation is mainly provided by fuel assembly tests carried out by AREVA NP.

The tests were performed for the following conditions:

- uniform axial flux distribution
- non-uniform axial flux distribution
- typical sub-channels
- sub-channels adjacent to a guide thimble

The tests were performed over the following parameter ranges:

- Pressure 20.7 < p < 170.6 bar.
- Mass velocity 930 < G < 4790 Kg/m²/s.

- Quality $-0.22 < X < 0.44$.

These ranges are representative of operating conditions of the EPR reactor. In nominal operating conditions, the local values of parameters with minimum RFTC [DNBR] are approximately as follows:

- Pressure = 155 bars.
- Mass velocity = 3500 kg/m²/s.
- Quality ~ - 0.10.

3. Form of the correlation for a uniform axial thermal flux

The FTC [CHF] correlation is given in analytical form in terms of the following parameters:

- thermo-hydraulic variables: pressure p, mass velocity G, and quality X
- fuel geometry i.e. distances between the grids
- type of sub-channel: i.e., differences between the two types, typical sub-channel and guide thimble sub-channel

The main term of the uniform flux correlation does not depend on the fuel geometry. It is only function of the thermal hydraulic variables. This term is assumed to depend linearly on the X variable via the following relation:

$$\Phi_{HCF} = A(p,G) - B(p,G)*X$$

The other terms associated with the fuel depend on the following geometric effects:

- spacing between grids
- distance between the expected location of the critical heat flux and the location of the upstream grid

The FTC [CHF] correlation has the following form: FCRIT $(P, G, X, d_g, g_{sp}, r_{tg})$

with

$$FCRIT= a(P,G,X,d_g) + c(P,G,X,g_{sp}) + d(P,X,g_{sp},r_{tg})$$

4. Form of the correlation for a non-uniform axial thermal flux

The FTC [CHF] values measured in rod bundles with non-uniform axial flux distributions are lower than those obtained with uniform distributions for the same local conditions. Application of FTC [CHF] correlations thus gave a predicted flux higher than the measured flux. The predicted value therefore required correction. This was achieved by applying the TONG non-uniform flux factor. The corrected flux is expressed as:

$$\Phi = \Phi_u / F_{NU}$$

where

Φ is the corrected flux value

Φu is the predicted flux value assuming a uniform axial heat flux distribution

F_{NU} is a non-uniform flux factor

2.2.2. Definition of the critical heat flux ratio

The RFTC [DNBR] as applied for both typical and thimble cold wall cells is:

$$DNBR = \frac{q''_{DNB,N}}{q''_{loc}}$$

where:

q''_{loc} is the actual local heat flux,

$$q''_{DNB,N} = \frac{q''_{DNB,EU}}{F}.$$

and $q''_{DNB,EU}$ is the uniform DNB heat flux as predicted by the FC- FTC [CHF] correlation.

F is the TONG flux shape factor which accounts for the non-uniform axial heat flux distribution.

2.2.3. Mixing effect between sub-channels

In a rod bundle, the sub-channels formed by four adjacent fuel rods are open to one another through the gaps between two neighbouring fuel rods. There is a cross-flow between channels because of the pressure differential between the channels.

The mixing effect reduces the enthalpy rise in the hot channel.

In the energy balance equation of the design code, a term is included to model the turbulent enthalpy exchange between adjacent channels. This term is proportional to the enthalpy difference between adjacent channels. In the proportionality factor, a coefficient appears which is named the Turbulent Mixing Coefficient.

Commonly the value of this coefficient is determined after performing a series of special tests on the type of grid assembly considered.

To decouple the EPR design from the fuel assembly design as far as possible and to be conservative, a lower bound value established from tests is used.

2.2.4. Manufacturing parameters uncertainties

Theses uncertainties take into account the manufacturing variation in fuel rod and fuel assembly materials and geometry.

There are two typesof manufacturing uncertainties:

- the effect of the pellet eccentricity and clad ovalization on the critical flux,
- the impact of the grid manufacturing tolerance on the critical flux.

a) Effect of the pellet eccentricity and clad ovalization on the critical flux

Some pellets may be eccentric with respect to the clad at the start of life. The clad may ovalize with time. In these two cases, there is an azimuthal variation of the flux over a small axial distance.

In the case of an eccentric pellet, the local flux peak in a given angle extends axially over a distance of at most a few pellet lengths because of random contact between the pellets and the clad at the start of life. This random character of the angle of contact is caused by variations in the shape of the ends of the pellets and by variations in the diameter of the pellets.

In the case of an ovalized clad, the local flux peak in a given angle extends axially over a distance of at most a few pellet lengths, taking into account the random azimuthal distribution of the fragments of the cracked pellet.

The hot channel uncertainties take into account the fact that the geometry and the materials of the rod and the assembly are not perfect.

The following uncertainties are identified regarding the hot channel.

- Hot spot engineering factor (F_Q^E)

This uncertainty is used to evaluate the maximum local power peak (the hot spot) and is determined by statistically combining the tolerances in the fuel pellet diameter, density and enrichment.

However, RFTC [DNBR] tests with local heat flux peaks showed that it is not necessary to take account of a specific uncertainty in the local heat flux.

- Nuclear enthalpy rise factor in the hot channel ($F_{\Delta H}^E$)

This is determined by statistically combining the effects on the enthalpy rise of manufacturing tolerances for fuel density, fuel enrichment, and rod position.

b) Effect of grid manufacturing tolerances on the critical heat flux(F_{LC}^E)

This uncertainty directly represents the grid manufacturing tolerance impact on RFTC [DNBR] values, or more precisely, the impact of the grid pressure loss coefficient tolerances on the core flow redistribution.

For fuel assemblies of the same design, the impact of the variability in RFTC [DNBR] values is negligible.

2.2.5. Effects of rod bowing in the reactor on the critical heat flux

The RFTC [DNBR] can be influenced by the rod bow phenomenon which has been detected from examinations of irradiated assemblies. This phenomenon consists of a rod being displaced from its nominal position within a channel. It is strongly fuel dependent and is modelled using a methodology derived from the French experience, described below.

The induced change in flow geometry results in a reduction in the FTC [CHF] at which DNB occurs.

The FTC [CHF] rod bow penalty is quantified by a convolution of two models:

- an envelope law defining the order of magnitude of the rod bowing, i.e. channel closure as a function of fuel burnup based upon measurement of rod bowing in irradiated fuel assemblies.

- a law defining the RFTC [DNBR] penalty based on the closure of the channel. The penalty law used is that approved by the NRC in 1979. It requires differentiation between full flow rate operation and reduced flow rate operation.

The resulting model gives the RFTC [DNBR] penalty as a function of fuel burnup.

Based on AREVA NP fuel experience, at burnup values below around 16000 MWd/t there is no rod bow penalty. Above this burnup limit, the penalty increases linearly but may it be limited at high burnups. Previous analyses have indicated that beyond burnups of around 35000 MWd/t, rods do not present the maximum value of nuclear $F\Delta H$.

Based on analyses of the $F\Delta H$ decrease with burn-up in EPR fuel management schemes, the burn-up limit that is considered for assessment of the rod bow penalty is 40000 MWd/t.

This describes the way rod bow is presently addressed in EPR design studies. The final method will depend on the actual fuel characteristics.

2.3. HEAT FLUX AND LINEAR POWER DEFINITIONS

The methods used to determine core average and maximum heat fluxes and linear power are given in Chapter D.3.3.

2.4. HEAT TRANSFER AND VOID FRACTION CORRELATIONS - RADIAL POWER DISTRIBUTION

The flow model is based on a two phase flow model (D.4 FIG 1) taking into account a thermal disequilibrium for the liquid phase and unequal velocities for liquid and vapour phases. This model is deduced from mass, momentum and energy balance equations for a turbulent two-phase flow. The liquid phase enthalpy balance equation allows sub cooled boiling calculations. To close this equations field, one needs physical modelling to describe phase interactions, turbulent mixing and fluid-wall interactions. These closure relationships are:

- a wall friction model

- a heat transfer model

- a velocity slip ratio model to take into account different velocities for liquid and steam phase,

- turbulent viscosity and diffusivity coefficients which are calculated from an algebraic model, permitting mixing effects to be described

To perform PWR core thermal hydraulic design calculations, and more precisely to compute local fluid properties needed to predict the critical heat flux margins, the reference sub-channel-type code FLICA with their own heat transfer and void fraction models is used.

2.5. HYDRODYNAMIC INSTABILITY

Boiling flows may be susceptible to thermo-hydrodynamic instabilities. These instabilities are undesirable in reactors since they may cause a change in thermal hydraulic conditions that may lead to a reduction in the DNB heat flux relative to that observed under steady flow conditions or to undesired forced vibrations of core components. Therefore, a thermal hydraulic design criterion was developed which states that modes of operation under PCC1 and 2 events shall not lead to thermo hydrodynamic instabilities.

Two specific types of flow instabilities are considered for PWR operation. These are the Ledinegg or flow excursion type of static instability and the density wave type of dynamic instability.

Ledinegg instability involves a sudden change in flow rate from one steady-state to another. This instability occurs when the slope of the reactor coolant system pressure drop-flow rate curve ($\partial P/\partial G$ internal) becomes algebraically smaller than the loop supply (pump head) pressure drop-flow rate curve ($\partial P/\partial G$ external). The criterion for stability is thus $\partial P/\partial G$ internal $\geq$ $\partial P/\partial G$ external.

The mechanism of density wave oscillations in a heated channel can be described as follows. Briefly, an inlet flow fluctuation produces an enthalpy perturbation. This perturbs the length and the pressure drop of the single phase region and causes quality or void perturbations in the two-phase region which travel up the channel with the flow. The quality and length perturbations in the two-phase region create two-phase pressure drop perturbations.

However, since the total pressure drop across the core is maintained by the characteristics of the fluid system external to the core, then the two-phase pressure drop perturbation feeds back to the single phase region. These resulting perturbations can be either attenuated or self-sustained.

The conclusion of the assessment is given in 4.6 within this Sub-chapter.

2.6. REACTOR VESSEL AND CORE HYDRAULICS

2.6.1. Bypass flow rate

The coolant flow enters into the reactor vessel via the inlet nozzles. It then flows down through the downcomer annulus formed by the reactor vessel and the core barrel and up into the core and the coolant outlet plenum. It leaves the reactor vessel through the outlet nozzles.

There are, however, several bypass paths:

a) Flow through the spray nozzles into upper head for head cooling purposes.

This bypass path derives water from the downcorner annulus.

The fluid is then directed from the upper dome to the upper plenum.

In "hot dome" configuration, which is the design option used, this flow rate is directed downwards in certain guide tubes in normal operation. In other guide tubes, there is an up-flow.

b) Flow entering into the control rod assembly guide thimbles which cools the control rods, the poison rods (if used), and the sources instrumentation.

c) Leakage flow from the vessel inlet nozzle to the vessel outlet nozzle directly through the gap between the vessel and the barrel.

d) Flow which is introduced between the heavy reflector and the core barrel, and inside the heavy reflector, for the purpose of cooling these components and which is not considered available for core cooling.

e) Flow in the gaps between the fuel assemblies on the core periphery and adjacent to the heavy reflector

The various contributions above are represented in D.4 TAB 2. The value of the maximum core bypass flow calculation is equal to 5.5% of the total vessel flow rate.

Out of the total, 2% is associated with the core and the rest with the internal equipment. Calculations have been made using tolerances in the worst direction, and taking account of pressure loss uncertainties.

2.6.2. Core inlet flow misdistribution

The inlet flow distribution is generally non uniform. Investigations with FLICA involving decreasing the flow rate over a limited core inlet area indicate that there is a rapid readjustment over one third of the core height and that consequently the inlet flow misdistribution has in practice a negligible impact on the hot channel RFTC [DNBR]. This flow redistribution is due to the readjustment of fluid velocities. Therefore, the flow misdistribution at the core inlet induces no penalty on the RFTC [DNBR] value or location. For RFTC [DNBR] calculations no penalty is therefore applied.

2.6.3. Core vessel pressure losses

Pressure losses are due to wall friction and changes in the geometry of the walls guiding the fluid. It is assumed that the flow is single phase and turbulent and the fluid is incompressible for pressure loss calculations in the core and the vessel performed to determine the flow rate in the primary system. Two-phase effects are neglected in the evaluation of pressure losses in the vessel since the core average void level is negligible.

The two-phase nature of the flow is taken into account in thermal-hydraulic analyses of the core sub-channels.

Because of the complex core and vessel flow geometry, precise analytical values for the form and friction loss coefficients are not available. Therefore, experimental values for these coefficients are obtained from geometrically similar models.

The pressure loss coefficients in the core are determined from hydraulic tests on the 17 x 17 advanced fuel assemblies. These tests are carried out in a test loop at a wide range of Reynolds numbers, including those encountered in a PWR core.

The pressure losses in the vessel are obtained by combining the core pressure loss with data from hydraulic tests on reduced-scale mock-ups performed on different vessels, and also using correlations and pressure loss models.

Primary flow measurements will be taken (see 5.1 within this Sub-chapter) during plant start-up tests to check that the design flow rates which are determined using to the method described above are conservative.

2.6.4. Hydraulic loads

The highest hydraulic loads on vessel equipments are reached for the maximum flow rate conditions.

For nominal operating conditions, the hydraulic loads are calculated using the mechanical flow rate, which uses a minimum core bypass flow rate value.

For cold shutdown conditions, the hydraulic loads are calculated with the same flow rates (vessel and core bypass), but with a different coolant density. This provides an enveloping case for normal operation.

Pump over-speed transient conditions, which could produce flow rates 20% greater than that assumed in the mechanical design, are used to define the hydraulic load envelope for transient conditions.

Hydraulic tests have been used to check the value of hydraulic loads during pump over-speed transients at mechanical design flow and under both hot and cold conditions.

2.6.5. Internal equipment hydraulic design

The EPR reactor pressure vessel and internal equipment have different characteristics compared to previous models, and these characteristics are important for hydraulic design. The developments are the result of new design options, new requirements or the convergence between existing French and German designs.

The most important developments are as follows:

Lower internals:

- the Flow Distribution Device, located below the core support plate, which ensures a relatively flat flow rate distribution at core inlet and prevents vortices form forming inside the lower plenum. This new feature results from the EPR option of deletion of reactor pressure vessel bottom penetrations, which leaves the lower plenum empty.

- the Heavy Reflector, which is a massive component that replaces the baffle plates and former around the core cavity.

Upper internals:

- The control rod guide tube and normal support columns design has been adapted to fit both control rod drive mechanism design (inspired from the KONVOI design) and EPR core and fuel assembly characteristics (inspired from French standards).

In addition, other evolutionary changes have taken place in the reactor pressure vessel design: i.e. core size increase, increase in the number of radial keys (eight instead of six), increase in radial gap in downcomer annulus.

Because of these changes, it must be confirmed that the global reactor pressurized vessel hydraulics meet the hydraulic performance objectives. This means the flow distribution at the core inlet and outlet remains acceptable, that sufficient mixing takes place between the loop flows upstream of the core inlet, that the core internals receive adequate cooling and their mechanical design enables safe operation.

The internal equipment hydraulic design is performed in three steps:

- First step: preliminary design

  based on experience feedback from other plants (N4 and P4 in France and KONVOI in Germany) or on experimental results

- Second step: design checking and optimization

  achieved by performing local CFD calculations, with a computer code validated against representative mockup tests (using the STAR-CD code).

- Third step: validation of the design

  Achieved by validation tests on the final design:

  a. final validation tests before manufacture of the reactor vessel

  b. commissioning tests before start-up of the plant.

2.7. THERMAL EFFECTS DURING NORMAL TRANSIENTS

DNB core safety limits are generated as a function of coolant temperature, pressure, core power, and axial and radial power distributions. Operation within these DNB safety limits ensures that the DNB design basis is met.

Section 4.1 within this Sub-chapter gives a description of the low RFTC [DNBR] protection and the low RFTC [DNBR] trip threshold setting. This function provides adequate protection for both steady state operation and for anticipated operational transients that are slow with respect to fluid transport delays in the reactor coolant system. Additional specific protection functions are provided for fast transients and transients from hot zero power.

2.8. UNCERTAINTIES

2.8.1. Critical heat flux

a) Treatment of uncertainties in RFTC [DNBR] calculations

A statistical approach is used to combine the uncertainties affecting the RFTC [DNBR].

Uncertainties which are random and follow a precisely known probability law are statistically treated: others are treated deterministically.

This approach is used to ensure that the RFTC [DNBR] criterion is met for all transients except for uncontrolled control rod withdrawal from a sub critical or low power startup transient and steam line break transients for which uncertainties are combined deterministically.

b) Assessment of the overall uncertainty with respect to the RFTC [DNBR].

b-1) Statistical approach

In order to relate the uncertainties affecting the RFTC [DNBR] to the RFTC [DNBR] variation, a variable, defined by the following equation, is used:

$$Y = \frac{DNBR_r}{DNBR_c}$$

$DNBR_r$ is the real RFTC [DNBR] value and $RFTC_c$ [$DNBR_c$] is the calculated value determined by considering the values of all the parameters related to the RFTC [DNBR] calculation at their best estimate values.

$DNBR_c$ is the RFTC [DNBR] value calculated on-line by the algorithm implemented in the I&C equipment.

Prob ($DNBR_r$ > T) = 95% with a 95% confidence level is equivalent to: Prob ($DNBR_c$ x Y > T) = 95% with a 95% confidence level.

If m and σ are the mean value and the standard deviation of the probability distribution for the random variable Y, the Prob (DNBRc x Y > T) = 95% with a 95% confidence level is achieved, keeping $DNBR_c > T/ m_Y (1-1.645\ V_Y)$.

$DNBR_r$ is a random variable that can be expressed as the product of the following random variables:

$$DNBR_r = \frac{\Phi_{rc}}{\Phi_{cp}} x \frac{\Phi_{cp}}{\Phi_{IDC}} x \frac{\Phi_{IDC}}{\Phi_{rl}} xP$$

Where:

Φ_{rc} = real critical heat flux,

Φ_{cp} = critical heat flux predicted by the FTC [CHF] predictor,

Φ_{IDC} = local heat flux calculated by the Design Code,

Φ_{rl} = real local heat flux in the same TH conditions,

P = Penalty factor,

$\frac{\Phi_{cp}}{\Phi_{IDC}}$ = $DNBR_{DC}$, i.e. the RFTC [DNBR] value calculated by the design code.

$DNBR_{DC}$ is a random variable which is a function of the system random variables (T°, P, Local Power, etc).

$DNBR_{DC}$ can be also decomposed as follows:

$$DNBR_{DC} = \frac{DNBR_{DC}}{DNBR_{DC0}} x \frac{DNBR_{DC0}}{DNBR_{ao}} x DNBR_{ao}$$

Where:

$DNBR_{DC0}$ = DNBR calculated by the design code at the best estimate values,

$DNBR_{ao}$ = DNBR calculated on-line by the I&C algorithm at the same best estimate values.

Consequently:

$$Y = \frac{DNBR_{r}}{DNBR_{ao}} = \frac{\Phi_{rc}}{\Phi_{cp}} x \frac{DNBR_{DC}}{DNBR_{DC0}} x \frac{DNBR_{DC0}}{DNBR_{ao}} x \frac{\Phi_{IDC}}{\Phi_{rl}} xP$$

Y is the product of the factor P and the following random variables:

1. $\frac{\Phi_{rc}}{\Phi_{cp}}$: The probability distribution of this variable is provided by the implementation of the FC-CHF correlation. It is a normal distribution characterized by a mean value m_c and a standard deviation σ_c.

   This is discussed in section 4.4.2.2.

2. $\frac{DNBR_{DC}}{DNBR_{DC0}}$: This random variable is a function of the following independent random variables:

- Plant operating parameters (which are measured online):

  a. inlet temperature,

  b. reactor pressure,

  c. local power,

  d. measured relative primary flow rate.

- Parameters which are not measured but which influence the RFTC [DNBR]:

  a. pellet uncertainties related to enrichment/diameter/dishing $\left(F^E_{\Delta H}, F^E_Q\right)$,

The probability distribution of this variable, representative of overall system uncertainty, is characterized by a mean value m_s and a standard deviation σ_s.

3. $\frac{DNBR_{DC0}}{DNBR_{ao}}$: This random variable accounts for the I&C RFTC [DNBR] calculation algorithm uncertainty.

The probability distribution of this variable is characterized by the two parameters m_a and σ_a.

4. $\frac{\Phi_{IDC}}{\Phi_{rl}}$: This random variable accounts for the design code uncertainty. Its probability distribution is characterized by the two parameters: m_{DC}, σ_{DC}.

An additional uncertainty must be taken into account: the transient versus steady - state uncertainty.

This uncertainty accounts for any discrepancy introduced by using local fluid properties from transient accident analysis to determine the RFTC [DNBR] under steady - state conditions. It is independent of the uncertainties noted above.

The parameters characterizing the probability distribution are m_{tss}, σ_{tss}.

The factor P corresponds to all the uncertainties which are treated deterministically:

- absolute total flow rate,
- core bypass flow,
- rod bow effect: RBP,
- neutronic data,
- actuation of the trip limit,
- etc.

Except for the rod bow, all these uncertainties are directly taken into account by parameters used in the transient analysis.

D.4 TAB 3 summarises all the variables used.

As all the above variables (different measurements, I&C algorithm, FTC [CHF] predictor, design code, manufacturing uncertainties) can be considered as independent and the perturbations from the mean value are small, the variation coefficient accounting for the uncertainty distribution related to RFTC [DNBR] is calculated as:

$$V_Y^2 = \left(\frac{\sigma_Y}{m_Y}\right)^2 = \left(\frac{\sigma_c}{m_c}\right)^2 + \left(\frac{\sigma_s}{m_s}\right)^2 + \left(\frac{\sigma_a}{m_a}\right)^2 + \left(\frac{\sigma_{tss}}{m_{tss}}\right)^2 + \left(\frac{\sigma_{DC}}{m_{DC}}\right)^2 \qquad (1)$$

Moreover, the probability distribution function of Y approaches a normal distribution with:

mean value: $m_Y = m_c \times m_s \times m_a \times m_{tss} \times m_{DC} \times P$,

standard deviation $\sigma_Y = m_Y \times V_Y$.

Consequently, the probability that the RFTC [DNBR] will be greater than a threshold T is 95% with a 95% confidence level if RFTC [DNBR] is greater than the threshold DNB_{th} defined by:

$$DNB_{th} = \frac{T}{m_Y\left(1-1.645V_Y\right)}.$$

All the terms of the equation (1) are calculated separately except for σ_S/m_S which is assessed with a Monte-Carlo method.

b-2) Deterministic approach

All the uncertainties mentioned above are handled in a deterministic manner.

Moreover, as the simplified on-line RFTC [DNBR] calculation is not used to protect the core against DNB in transients, neither by using the protection system actuation nor by using the DNB LCO surveillance, each parameter which impacts RFTC [DNBR] has to be compared to a specific LCO and its uncertainty has thus to be accounted for.

These parameters are, for example, as follows:

- the average primary temperature
- reactor pressure
- local power
- analysis of the transients is carried out using the worst power distribution.

<u>Uncertainty due to calculation code and the mixing coefficient</u>

The results of a sensitivity study with the design code show that the minimum RFTC [DNBR] in the hot channel is relatively insensitive to variations in the core-wide radial power distribution (for the same value of $F^N_{\Delta H}$.

Studies have been performed to determine the sensitivity of the minimum RFTC [DNBR] in the hot channel to the radial and axial discretisation, the inlet velocity, the pressure drop coefficient, the power distribution, the mixing coefficients, and the void model.

The results of these studies show that the minimum RFTC [DNBR] in the hot channel is relatively sensitive to variations of three of these parameters: the mixing coefficients, the two-phase model and the radial distribution of the grid pressure drop coefficients.

For the fuel grid, numerous mixing tests have been performed using the same experimental configurations as the one used for FTC [CHF] tests.

These mixing data have resulted in an average value of the mixing coefficient which is higher than the calculation value used in the design calculation codes (see 2.2.3 within this Sub-chapter).

2.8.2. Justification for statistical combination of uncertainties

In accordance with the explanations above, a statistical approach is used to combine the following uncertainties affecting the RFTC [DNBR]:

- critical heat flux correlation uncertainty (m_C, σ_C)
- overall system uncertainty (m_S, σ_S)
- algorithm uncertainty (m_a, σ_a)

- design code uncertainty (m_{DC}, σ_{DC})
- transient versus steady state calculation uncertainty (m_{tss}, σ_{tss})

Independent parameters for which the uncertainty presents a random feature and the probability law is precisely known are statistically treated.

<u>CHF correlation uncertainty</u>

The assessment of the performance of the FC- FTC [CHF] correlation, based on a comparison with FTC [CHF] test results has led to a definition of the probability distribution for the ratio of measured to predicted CHF. The ratio is found to be normally distributed (see Chapter D.2.2.2).

<u>Overall system uncertainty</u>

Two main types of uncertainties are defined, each of which can be split up into several uncertainties.

Uncertainties in physical parameters measured during operation:

The following plant operating parameters are used for calculating the RFTC [DNBR]: the inlet temperature, the pressurizer pressure, the relative measured primary flow rate and the local power. The inlet temperature is derived from cold leg temperature sensors, the pressurizer pressure is derived from primary pressure sensors, the relative measured primary flow rate is derived from the Reactor Coolant Pump (GMPP) [RCP] speed sensors and the power density distribution of the hot channel is directly derived from the nuclear in-core instrumentation (Self Powered Neutron Detectors (SPND)). Each measuring process is independent of the others (Temperature sensor in the cold leg, pressure sensor on the top of pressurizer, speed sensor on the GMPP [RCP]and in-core for the SPND). A given uncertainty for a temperature sensor due to a calibration error has, for example, no relation to that of the pressuriser sensor or that of the GMPP [RCP] speed sensor.

Each of the devices between the sensor and the use of the signal in the protection system (e.g. for temperature: ohm-ampere converter, ampere-volt converter, isolation module if required and analog/numerical converter) has a random and independent uncertainty that statistically treated.

The power density distribution in the hot channel involves the following uncertainties: the Aeroball Measuring System accuracy (taking into account the activation rate accuracy, the relative power density reconstruction, the burnup discretisation and the number of instrumented assemblies) and the SPND signal accuracy (drift, allowance for burnable absorbers…).

The global uncertainty can be split up in several independent probability distributions (sensor, transmitter calibration, etc, uncertainties. The resultant probability distribution for such a large number of random variables is a normal distribution, as generally observed for measurement uncertainty.

Uncertainties due to pellet manufacturing tolerance:

The $F^{E}_{\Delta H}$ factor accounts for variations in those fabrication variables which affect the heat generation rate in the flow channel. These variables are pellet diameter, density and U 235 enrichment. Uncertainties in these variables are determined from sampling of manufacturing data. The resulting uncertainty is independent of the uncertainties noted above. The distribution is found to be normal.

<u>Algorithm uncertainty</u>

This uncertainty accounts for the difference between the design code calculation and the on-line RFTC [DNBR] algorithm calculation for the same thermal and hydraulic conditions. The algorithm is adjusted using design code calculations.

A statistical analysis allows the determination of the probability distribution of differences between algorithm and design code. The distribution is found to be normal.

Design code uncertainty

The design code uncertainty includes all the effects of analyzing the full core with a numeric code. As the analysis of the heat flux tests to define the correlation performance is done with the design code, the range of the predicted heat flux for each experimental data point includes the design code uncertainty and consequently can be used to define the parameters m and σ for the predictor.

Transient versus steady state calculation uncertainty

This uncertainty accounts for any discrepancy introduced by the calculation of RFTC [DNBR] under steady-state conditions using local fluid properties from transient analyses. It is independent of the uncertainties noted above. It is assumed to have a normal distribution.

2.8.3. Fuel and Cladding temperatures

The fuel temperature is a function of the oxide, clad, gap, and pellet conductance. Uncertainties in the fuel temperature calculation are essentially of two types:

- fabrication uncertainties such as variations in the pellet and clad dimensions,
- pellet density, and model uncertainties such as variations in the pellet conductivity and gap conductance.

These uncertainties have been quantified by comparison of the thermal model with in-pile thermocouple measurements by out-of-pile measurements of the fuel and clad properties and by measurements of the fuel and clad dimensions during fabrication. The resulting uncertainties are then used in all evaluations involving the fuel temperature.

In addition to the temperature uncertainty described above, the uncertainty in local power due to measurement uncertainty and density and enrichment variations are considered in establishing the heat flux hot channel factor.

Uncertainty in in the cladding temperature results from uncertainties in the oxide thickness. Because of the excellent heat transfer between the surface of the rod and the coolant, the film temperature drop does not appreciably contribute to the uncertainty.

2.8.4. Hydraulics

a) Uncertainties related to core and vessel pressure drops

Core and vessel pressure drops based on the best estimate flow. They are based on the uncertainties in both the test results and the analytical extension of these values to the reactor application.

A major use of the core and vessel pressure drops is to determine the reactor coolant system flow rates. In addition, tests on the RCP [RCS] prior to initial criticality will be made to verify that a conservative RCP [RCS] coolant flow rate has been used in the design and analyses of the plant.

b) Uncertainties due to inlet flow misdistribution

The effects of uncertainties in the inlet flow misdistribution criteria used in the core thermal analyses are discussed in section 2.6.2 within this Sub-chapter.

c) Uncertainties concerning flow rates

The thermal design flow is defined for use in core thermal performance evaluations which accounts for both prediction and measurement uncertainties.

In addition, a maximum of 5.5% of the thermal design flow is assumed to be ineffective for core heat removal capability because it bypasses through the various available vessel flow paths described in section 2.6.1 within this Sub-chapter.

d) Uncertainties concerning hydraulic loads

As explained in 2.6.4 within this Sub-chapter, the enveloping hydraulic loads on the fuel assembly are evaluated in normal operation for cold shutdown conditions and in transient conditions for a pump over speed transient which creates flow rates 20% greater than the mechanical design flow. The mechanical design flow is 8% greater than the best estimate or most likely flow rate value for the actual plant operating condition.

e) Uncertainties concerning internals hydraulic design

Uncertainties are taken into account either by considering conservative hypothesis for the boundary conditions of the calculations or through conservatism inherent to the code or to the numerical schemes used to perform the calculation.

3. THERMAL-HYDRAULIC OPERATING PARAMETERS

3.1. TEMPERATURE / POWER OPERATING MAP

The relationship between the temperature of the primary system and the power is shown is D.4 FIG 2 for the thermal-hydraulic flow rate, the most likely flow rate and the mechanical flow rate.

3.2. THERMAL-HYDRAULIC CHARACTERISTICS

The thermal-hydraulic characteristics are given in D.4 TAB 1.

4. ANALYSIS METHODS AND DESIGN DATA

4.1. METHODOLOGY USED FOR TRANSIENT ANALYSIS

Low DNBR Reactor Trip and Limiting Conditions of Operation (LCO) settings.

The transients to be analyzed are PCC2 events and also certain PCC3 and PCC4 events (e.g. steam line break).

The methodology depends on the type of reactor protection channel actuated during the transient: i.e. the low RFTC [DNBR] channel or otherwise.

In case the low RFTC [DNBR] channel is used, the requirement is to define the thresholds for the low RFTC [DNBR] channel. In the second case, the purpose is to define a RFTC [DNBR] criterion which must be met for each transient considered.

Transients are divided into three categories:

- Type I transients: transients from power for which the low RFTC [DNBR] protection is effective,

- Type II transients: transients from power for which the low RFTC [DNBR] protection is not effective,

- Type III transients: transients from hot zero power (uncontrolled control rod bank withdrawal in a sub-critical or low power start up, and steam line break).

4.1.1. Transients in power operation (low RFTC [DNBR] reactor trip and LCO setting)

As the DNB protection channel and the DNB surveillance channels are based directly on the DNBR variable calculated by the DNBR algorithm, the low RFTC [DNBR] trip threshold and the DNB LCO threshold take into account calculational uncertainties and measurement errors.

DNB is avoided by keeping the on-line RFTC [DNBR] calculated values above the corresponding thresholds.

The uncertainties may be different for the two systems, as they are related to system accuracy and system operational conditions.

To set the two thresholds, the transients are divided into two classes:

Transients for which low DNBR reactor trip is effective (type I transients)

These are characterized by the following conditions:

- parameters influencing RFTC [DNBR] in the transients are input parameters for the low RFTC [DNBR] protection channel,

- evolution of the parameters is not too fast to be detected by the low RFTC [DNBR] protection channel.

The low RFTC [DNBR] I&C channel trip is actuated whenever the RFTC [DNBR] reaches the safety criterion.

The variable $Y = \frac{DNBR_r}{DNBR_{PS}}$ is characterized by a mean value (m_Y^{PS}) and a standard deviation (σ_Y^{PS}).

$DNBR_{PS}$ is the DNBR value calculated by the protection system algorithm.

The low RFTC [DNBR] I&C channel trip is actuated when the DNBR value calculated by the protection system is lower than the threshold value $DNBR_{PS}$ calculated from

$$DNBR_{PS} = \frac{SC}{m_Y^{PS}\ (1\text{-}1.645\ V_Y^{PS})}\ .$$

Which implies that the probability to avoid boiling crisis is above 95% at the 95% confidence level.

Transients for which low DNBR protection is not effective (type II transients):

In this case reactor trip is achieved using only specific parameters (in many cases only one, e.g. low pump speed). Therefore, it is necessary to assume initial values for other parameters input to the AAR [RT] channel to define the LCO surveillance system.

During normal operation, the on-line calculated DNBR value must be kept above a low DNBR threshold (LCO) such that DNB is avoided the anticipated transients.

A typical type II transient is complete loss of forced reactor coolant flow.

In order to design the DNB LCO I&C channel to cope with type II transients, the maximum RFTC [DNBR] variation is calculated for all transients of this category.

The transient leading to the maximum RFTC [DNBR] variation is used to determine the low RFTC [DNBR] LCO threshold:

$$DNBR_{LCO} = \frac{\left[SC + (\Delta DNBR)\max\right]}{m_Y^{SS}\ (1\text{-}1.645\ V_Y^{SS})}$$

Where m_Y^{SS} and V_Y^{SS} are the mean value and the variation coefficient of the variable $Y = \frac{DNBR_r}{DNBR_{ss}}$ and $DNBR_{SS}$ is the DNBR value calculated by the surveillance system.

During normal operation, the DNBR must always remain above the low DNBR threshold: thus the DNB criterion will be met if a type II transient occurs.

4.1.2. Transients from hot zero power (DNBR design limit)

These transients, referred to as type III transients, are:

- uncontrolled control rod withdrawal from a sub-critical or low power start-up,

- main and minor steam system line failure (steam line break (RTV) [SLB]).

Specific analyses are performed for these transients. Specific protection is necessary because low RFTC [DNBR] I&C channel is not effective in these cases.

A minimum RFTC [DNBR] is calculated for the transient. This must always be greater than the RFTC [DNBR] design limit assessed in a deterministic way.

This limit includes only:

- the correlation/design code uncertainty,
- the rod bow uncertainty except for RTV [SLB] as there is no rod bow effect exists at low pressure

4.2. INFLUENCE OF POWER DISTRIBUTION

The core power distribution is largely established at beginning-of-life by fuel enrichment, loading pattern, and core power level but is also a function of variables such as control rod worth and position, and fuel depletion throughout life. The core radial enthalpy rise distribution, as determined by the integral of power in each channel, is more important for DNB analyses. This power distribution is characterized by $F_{\Delta H}^{N}$ and axial heat flux profiles.

Given the local power density q'(W/cm) at a point x, y, z in a core with N fuel rods and height H, the nuclear enthalpy rise hot channel factor is given by:

$$F_{\Delta H}^{N} = \frac{\text{hot rod power}}{\text{average rod power}} = \frac{\max \int_{O}^{H} q'(x_o, y_o, z)\,dz}{\frac{1}{N_{\text{allrods}}} \Sigma \int_{O}^{H} q'(x, y, z)\,dz}$$

Where:

x_O, y_O are the position coordinates of the hot rod.

The way in which $F_{\Delta H}^{N}$ is used in the DNB calculations is important.

The location of minimum RFTC [DNBR] depends on the axial power profile, and the value of RFTC [DNBR] depends on the enthalpy rise to that point. Basically, the maximum value of the rod power integral is used to identify the most likely rod for minimum RFTC [DNBR]. An axial power profile is obtained which, when normalized to the value of $F_{\Delta H}^{N}$, recreates the axial heat flux along the limiting rod. The surrounding rods are assumed to have the same axial profile with rod average power, which are typical distributions found in hot assemblies. In this manner, worst case axial profiles can be combined with worst case radial distributions for reference DNB calculations.

It should be noted again that $F^N_{\Delta H}$ is an integral and is used as such in DNB calculations. Local heat fluxes are obtained by using hot rod and adjacent rod explicit power shapes which take into account variations in horizontal power shapes throughout the core.

The enthalpy rise hot channel factor FΔH corresponds to:

$$F\Delta H = \max_{xy} \left(P_{\Delta H}(x, y)\right)$$

Where $P_{\Delta H}$ (x, y) is the radial enthalpy rise distribution of the channel (x, y):

$$P_{\Delta H}(x, y) = \frac{\int_O^H P(x, y, z)\, dz}{\int_O^H dz}$$

And P(x, y, z) is the relative power in the channel (x, y) at the height z.

For each channel, the average relative power distribution is assessed from the rod relative power distribution weighting the relative power of each rod surrounding the channel by the heated perimeter of the part of the rod within in the considered channel.

The design studies are performed with power distributions calculated with neutronic design codes.

For type I and type II transients (see 4.4.4.1) the core related I&C protection functions prevent the core from developing adverse power distributions.

The power density distribution in the hot channel is directly derived from nuclear in-core instrumentation by Self Powered Neutron Detectors (SPND).

Consequently the two uncertainties are associated with the power density distribution, as follows:

- reconstruction uncertainty,
- measurement accuracy.

For type III transients (see chapter 4.4.4.1) the uncertainty in the power distribution is the calculation uncertainty that applies to the DNBR value predicted by the design code.

For loss of coolant analyses APRP [LOCA] a decoupling design $F^N_{\Delta H}$ value equal to 1.80 is used, which takes into account the uncertainties in the power distribution calculations, the xenon penalty for azimuthal and radial oscillations, and a provision for fuel loading.

4.3. CORE ANALYSIS TOOLS

The objective of the reactor core thermal design is to determine the maximum heat removal capability in all flow sub-channels and show that core safety limits are not exceeded taking account of engineering and nuclear effects. The thermal design considers local variations in dimensions, power generation, flow redistribution, and mixing.

The thermal design uses the FLICA code which used realistic matrix models that account for hydraulic and nuclear effects on the enthalpy rise in the core. The behaviour of the hot assembly is determined by superimposing the power distribution among the assemblies on the inlet flow distribution while allowing for flow mixing and flow distribution between assemblies. The average flow and enthalpy in the hottest assembly is obtained from a core-wide, assembly by assembly analysis. The local variation of power, fuel rod and pellet fabrication, and mixing within the hottest assembly are then superimposed on the average conditions of the hottest assembly in order to determine the conditions in the hot channel.

The transient and steady state FLICA code has been used to determine coolant density, mass velocity, enthalpy, vapour void, static pressure, and RFTC [DNBR] distributions in the parallel flow channels in the reactor core under all expected operating conditions.

4.4. CORE THERMAL RESPONSE

The general thermo-hydraulic characteristics of the core in steady-state operating conditions are given in ***4.4. TAB 1.***

4.5. HYDRAULIC DATA

The bypass flow rates are given in D.4 TAB 2.

4.6. HYDRODYNAMIC INSTABILITY

The EPR pump head curve has a negative slope ($\partial P/\partial G$ external < 0), whereas the reactor coolant system pressure drop-flow curve has a positive slope ($\partial P/\partial G$ internal > 0) over the PCC1 and PCC2 operational ranges. The criterion for stability is met and thus Ledinegg instability will not occur.

The application of the Ishii method to reactors indicates that a large margin exists to density wave instability, e.g., increases on the order of 200% of rated reactor power would be required for the predicted inception of this type of instability.

The method of Ishii applied to the EPR plant is conservative due to the parallel open channel feature of PWR cores. For such cores, there is little resistance to lateral flow leaving flow channels of high power density. There is also energy transfer from channels of high power density to lower power density channels. This coupling with cooler channels has led to the view that an open channel configuration is more stable than an equivalent closed channel configuration under the same boundary conditions.

Additional evidence that flow instabilities do not adversely affect thermal margin is provided by the data from the rod bundle DNB tests. Many rod bundles have been tested over wide ranges of operating conditions with no evidence of premature DNB or of inconsistent data which might be indicative of flow instabilities in the rod bundles.

In summary, it is concluded that thermo hydrodynamic instabilities will not occur under PCC 1 and 2 modes of reactor operation. A large power margin, greater than doubling rated power, exists to predicted interception of such instabilities.

5. TESTS AND VERIFICATION

5.1. TESTS PRIOR TO INITIAL CRITICALITY

Reactor coolant flow tests are performed following fuel loading and at several power levels after the plant startup. The results of the successive enthalpies balances performed allow determining the coolant flow rates at reactor operating conditions. These tests verify that proper coolant flow rates have been used in the core thermal and hydraulic analysis.

5.2. INITIAL POWER AND PLANT OPERATION

Core power distribution measurements are made at several core power levels. These tests are used to ensure that the values of the parameters used in the evaluation of the power distribution and the enthalpy rise factor are correct.

5.3. COMPONENTS AND FUEL INSPECTIONS

Inspections are performed on the manufactured fuel. Fabrication measurements critical to thermal and hydraulic analyses are obtained for verifying that the engineering hot channel factors in the design analyses are met.

6. INSTRUMENTATION REQUIREMENTS

6.1. LOW RFTC [DNBR] FUNCTIONS

There are two low RFTC [DNBR] I&C functions:

- low RFTC [DNBR] protection functions, which actuate reactor trip (AAR) [RT],
- low RFTC [DNBR] surveillance functions which limit the conditions of operation (LCO).

The use of RFTC [DNBR] on-line calculations in the protection and surveillance system(s) allows the RFTC [DNBR] criterion to be met by defining a low RFTC [DNBR] reactor trip channel and a low RFTC [DNBR] LCO channel based directly on derived variables representative of the phenomenon to avoid.

The low RFTC [DNBR] protection function actuates a reactor trip which protects the fuel against DNB, during accidental transients, for all postulated initiating events that could lead to an uncontrolled decrease of the RFTC [DNBR] value. The low RFTC [DNBR] surveillance function (LCO) ensures a sufficient margin to the RFTC [DNBR] criterion during normal operation in order to accommodate postulated initiating events leading to a significant decrease in RFTC [DNBR]. During PCC1 events, the RFTC [DNBR] value must be kept above the $RFTC_{LCO}$ [$DNBR_{LCO}$] threshold so that if an event occurs for which the low RFTC [DNBR] protection is not efficient, DNB is avoided (see 4.1 within this Sub-chapter).

Exceeding this LCO initiates the following countermeasures:

- on reaching the first threshold there is an alarm, blocking of control rod withdrawal and blocking of load increases,

- on reaching the second threshold, a reactor power reduction is effected by insertion of rod banks and if it is necessary a suitable turbine load reduction.

Both I&C RFTC [DNBR] algorithm for protection and surveillance are based on the same principles.

The calculation of the minimum RFTC [DNBR] uses the following parameters:

- Power density distribution of the hot channel, derived from the neutronic in-core instrumentation (SPND). The signals of the in-core detectors calibrated in power density units provide both, local power and integrated power along the hot channel using a polynomial development.

- Inlet temperature: derived from the cold leg temperature sensors.

- Pressure: derived from the primary pressure sensors

- Core flow rate: derived from the speed sensors.

The critical heat flux (FTC) [CHF] is calculated by the FC-FTC [CHF] correlation using local thermo hydraulic parameters, i.e. pressure, quality, mass flow.

These parameters are calculated by a simplified single channel model representing the hot channel without considering exchanges with neighbouring channels.

Therefore it is fitted on thermal hydraulic design code to take into account mass and energy exchanges between channels.

6.2. HIGH LINEAR POWER DENSITY (HLPD) FUNCTIONS

There are two high linear power density functions:

- the high linear power density protection function,

- the high linear power density surveillance function (LCO).

The meeting of the safety criteria on the melting at the centre of the fuel pellet is ensured by meeting the decoupling criteria on the linear power density at the hot spot which must remain lower than a certain limit.

Thus, the protection and surveillance systems allow the safety criteria concerning the melting at the centre of the fuel pellet to be respected by defining an HLPD reactor trip channel and a HLPD LCO channel directly based on the reconstruction of the linear power density at the hot spot.

The HLPD protection function actuates a reactor trip which protects the fuel against the melting at the centre of the fuel pellet, during accidental transients, whatever the event leading to an uncontrolled increase of the linear power density is.

The HLPD surveillance function (LCO) mainly ensures compliance with the core integrity criterion in case of loss of coolant accident APRP [LOCA] or event like "GMPP [RCP] shaft break" and/or "rod ejection".

The exceeding of this LCO initiates the following countermeasures:

- on the first threshold, an alarm, the blocking of RCCA withdrawal or RCCA insertion depending on the axial power shape and the blocking of load increase,

- on the second threshold, a reactor power reduction by the insertion of rod banks and if it is necessary the suitable turbine load reduction.

A distortion of the power shape can be a cause, among others, for this limit value to be reached. The limit value depends on the core axial location (lower for the upper core half than for the lower core half) that means this surveillance function limits also the axial power shape.

The calculation of the maximal linear power density value (W/cm) is directly issued from the neutronic in-core instrumentation by SPNDs.

6.3. FIXED IN-CORE NEUTRONIC INSTRUMENTATION

The fixed in-core nuclear instrumentation system consists of SPNDs which are used to measure the local power density generated by the fuel elements.

The detector fingers are located where they can give the maximum information about power density variations and their impact on the key core parameters (F_Q, F∆H) especially under perturbed conditions.

The SPNDs are distributed homogeneously radially in the core such that their signals are representative of the key core parameters for different perturbation modes and fuel managements.

12 fuel assemblies are instrumented. Therefore the core is divided into 12 radial zones and each zone is surveyed by one SPND finger.

The SPND fingers are distributed such that practically the whole area of the core is covered.

An example of the position of instrumented assemblies and the various zones are presented in D.4 FIG 3.

The 12 SPND fingers contain each one 6 detectors.

In each finger, three SPNDs are located in the upper core half and the other three in the lower core half to detect peak power density occurring in upper and lower halves and thus covering all possible power distributions (normal and accidental) Optimal axial positions will be determined after analyzing axial power shapes in transient conditions.

Axial locations are always between two grids to avoid flux depression in the vicinity of the grids.

Each finger is assigned to a part of the core volume called the "surveillance zone". The fingers are calibrated by the aeroball measuring system to reproduce the peak power density and the power distribution of the channel containing the maximum power density (aeroball measuring system distribution) in its radial surveyed zone since the SPND signals are used for minimum RFTC [DNBR] calculation.

6.4. AEROBALL SYSTEM

The basic function of this movable in-core neutronic instrumentation system is to measure the neutron flux at different spatial points of the core. The preliminary aeroball probes location is shown in D.4 FIG 4. The flux map is used to get the axial neutronic power distribution of the hot channel of each fuel assembly (3-D image of the distribution).

From axial distributions, the following core parameters are deduced:

- axial neutronic power of the hot channel of each fuel assembly, and maximum core value FQ parameter,
- average axial neutronic core power distribution,
- integrated power along each assembly and hot channel allowing to deduce rise in enthalpy level of each assembly, and maximum core value F∆H parameter then, the min DNB ratio is calculated by mixing with operating thermal hydraulic conditions,
- quadrant power tilt ratio.

Core parameters can be used to perform the following tasks:

- verify the core conformity at first startup and during reloading,
- calibrate fixed in-core and ex-core neutronic instrumentations,
- justify measurement uncertainties that are taken into account in monitoring systems,
- follow up fuel assembly burn up,
- make investigations and diagnostics when operating in particular or abnormal conditions.

The aeroball movable system enables:

- all the different regions of the core (fuel variety, local effects of control rods, radial surface) to be equitably taken into account in the flux maps. Therefore, when the measured assemblies are reflected into a single eighth of the core, nearly all of assemblies (beside those located at RCCA positions) are instrumented,
- measurements of symmetric assemblies (with regard to the quarter core) to be performed in different regions of the core,
- measurements distributed uniformly over active height of the fuel assembly to be performed assembly

6.5. EX-CORE NEUTRONIC INSTRUMENTATION

The output of the three ranges (source, intermediate, and power) of detectors, with the electronics of the nuclear instruments, is used to limit the maximum power output of the reactor within their respective ranges.

The neutron flux detectors are installed around the reactor in the primary shield:

- the 3 chambers for the source range detectors are installed at radial positions at the nearest points to reactor core.

- the 4 chambers for the intermediate range detectors are installed at radial positions that are the nearest points to reactor core and positioned at an elevation corresponding to 1/2 of the core height.

- the 4 bisection detectors for the power range are placed as near as possible to the reactor vessel. Axially, the detectors are distributed over the whole active core height.

The three ranges of detectors are used as inputs to monitor neutron flux from a completely shutdown condition to 120% of fuel power, with the capability of recording higher overpower excursions.

The output of the power range channels may be used for:

- Protecting the core against uncontrolled positive reactivity insertion. It is used mainly to cope with the safety analyses of PIEs of PCC3 (e.g. single control rod assembly withdrawal at power) or PCC4 (e.g. control rod ejection).

- Alerting the operator to an excessive power unbalance between the quadrants.

- Banks blocking

D.4 TAB 1: THERMO-HYDRAULIC DESIGN PARAMETERS

Nominal core power (MWth)	4500
Number of loops	4
Nominal primary pressure (absolute) (MPa)	15.5
Coolant flow:	
Core cross sectional area (m^2)	5.9
Average velocity in the core (m/s)	5
Average mass flux in the core ($g/cm^2/s$)	356
Total mass flow rate (kg/s)	22225
Mass flow rate in the core (t/h)	75609
Thermo-hydraulic design flow rate/loop (m^3/hr)	27180
Best Estimate flow rate/loop (m^3/hr)	28315
Mechanical design flow rate/loop (m^3/hr)	30580
Coolant temperature (°C):	
Nominal inlet	295.7
Average increase in vessel	34.2
Average increase in core	36.0
Average in core	313.7
Average in vessel	312.8
Heat exchange:	
Heat transfer area (m^2)	8005
Average core thermal flux (W/cm^2)	54.7
Maximum thermal flux in the core (normal operation) (W/cm^2)	157.3
Average linear power density (with a cold geometry) (W/cm)	163.4
Peak linear power for normal operating conditions (W/cm)	470
Peak linear power protection setpoint (W/cm)	590

D.4 TAB 1: THERMO-HYDRAULIC DESIGN PARAMETERS (CONTINUED)

RFTC [DNBR] (for illustration): Minimum RFTC [DNBR] in nominal operating conditions with: $F\Delta H$ = 1.61 - cos 1.45	 2.6
Fuel assembly: Number of fuel assemblies Pitch of fuel assemblies (cm) Active height of the fuel (cm) Pitch of fuel rods (cm) Number of fuel rods per assembly Number of guide thimbles per assembly Outside diameter of the fuel rod (cm) Guide thimble diameter (cm)	 241 21.504 420 1.26 265 24 0.95 1.245
Core power characteristics: Power density (hot conditions) (MW/m^3)	 94.6

D.4 TAB 2: CORE BYPASS FLOW RATE

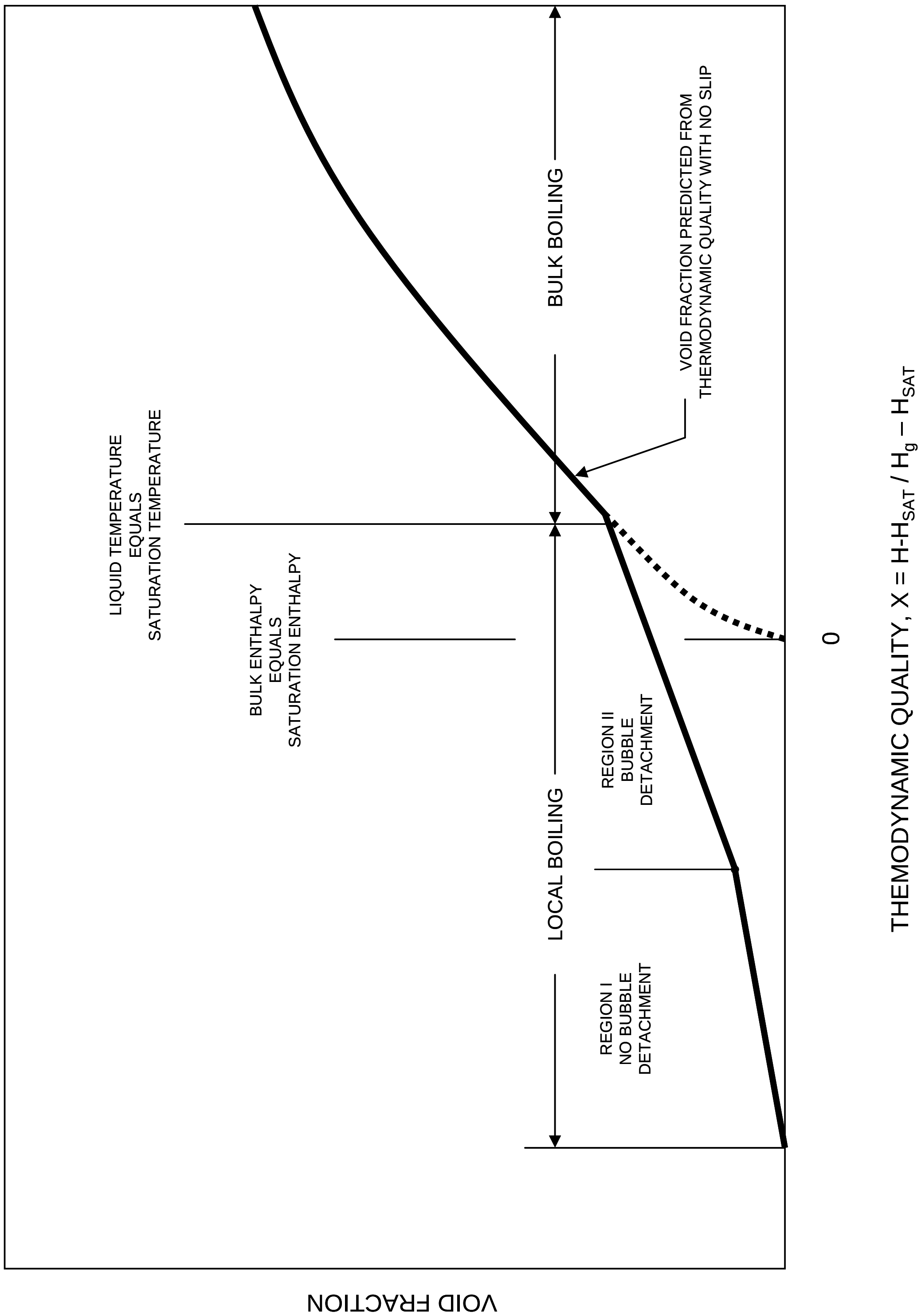
I : Core bypass flow rate	CALCULATION VALUE (in % of the vessel's total flow rate)
Head cooling flow rate	0.5
Cooling flow in the guide thimbles	2.0
Leakage to outlet nozzles	1.0
Flow through the heavy reflector	1.5
Flow between the heavy reflector and the core	0.5
Total core bypass flow rate	**5.5**

D.4 TAB 3: UNCERTAINTIES IN RFTC [DNBR]

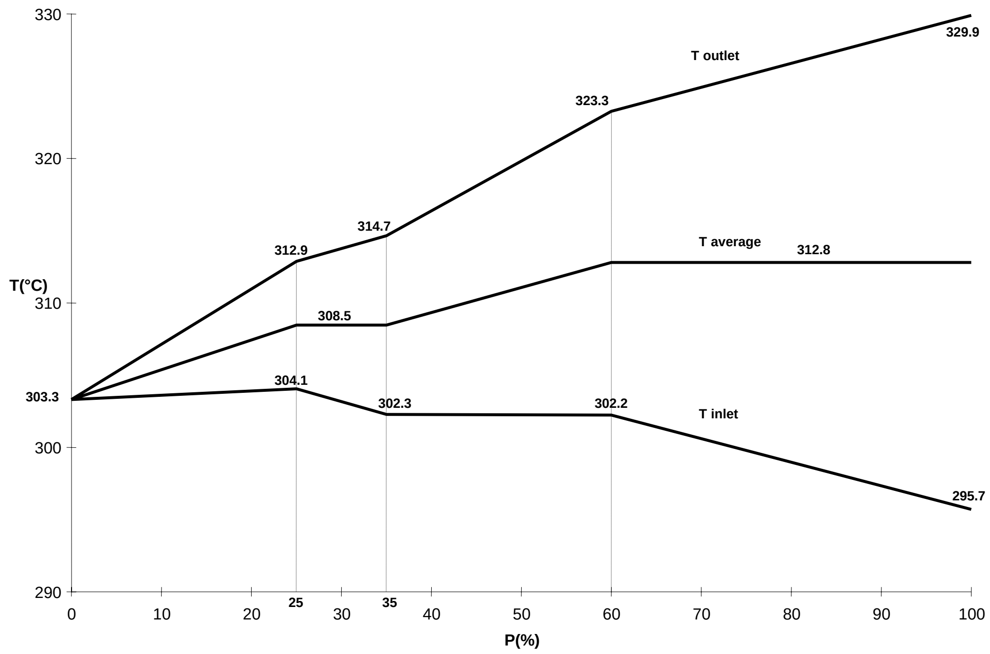
VARIABLE		Distribution	Typical variance	Average value
Critical heat flux correlation	Φ_{rc}/Φ_{cp}	Normal	σ_c	m_c
System parameters[1]:	$\frac{DNBR_{DC}}{DNBR_{DC0}}$	Normal	σ_S	m_S
- Inlet T°			σ_T	m_T
- Reactor pressure			σ_p	m_p
- Local power			σ_{lp}	m_{lp}
- Measured relative primary flow rate			σ_Q	m_Q
- $F^E_{\Delta H}$			$\sigma\ F^E_{\Delta H}$	$m\ F^E_{\Delta H}$
- F^E_Q			$\sigma\ F^E_Q$	$m\ F^E_Q$
I&C algorithm	$\frac{DNBR_{DCD}}{DNBR_{a0}}$	Normal	σ_a	m_a
Transient versus steady-state	$\Phi lDC / \Phi rl$	Normal	σ_{tss}	m_{tss}
Design code		Normal	σ_{DC}	M_{DC}
Penalties:	P	Constant	0	P
- Rod bow				RBP
- total absolute flow				Thermal-hydraulic
- core bypass flow				Thermal-hydraulic

[1] For these parameters, in the absence of the part evaluated in a deterministic manner, the average value is equal to 1.

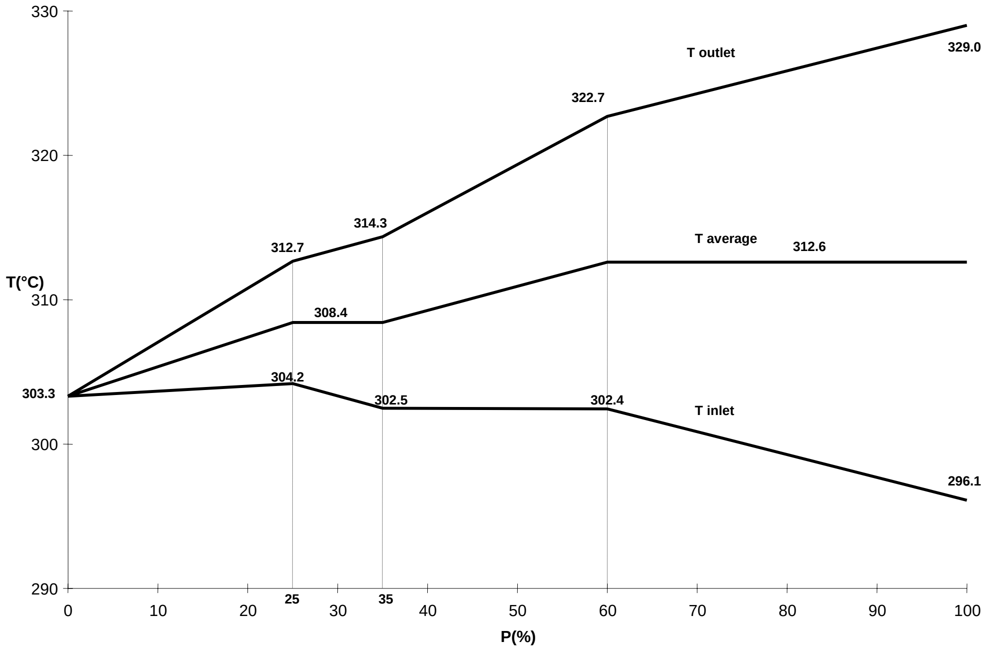
D.4 FIG 1: VOID FRACTION VERSUS THERMAL-DYNAMIC QUALITY



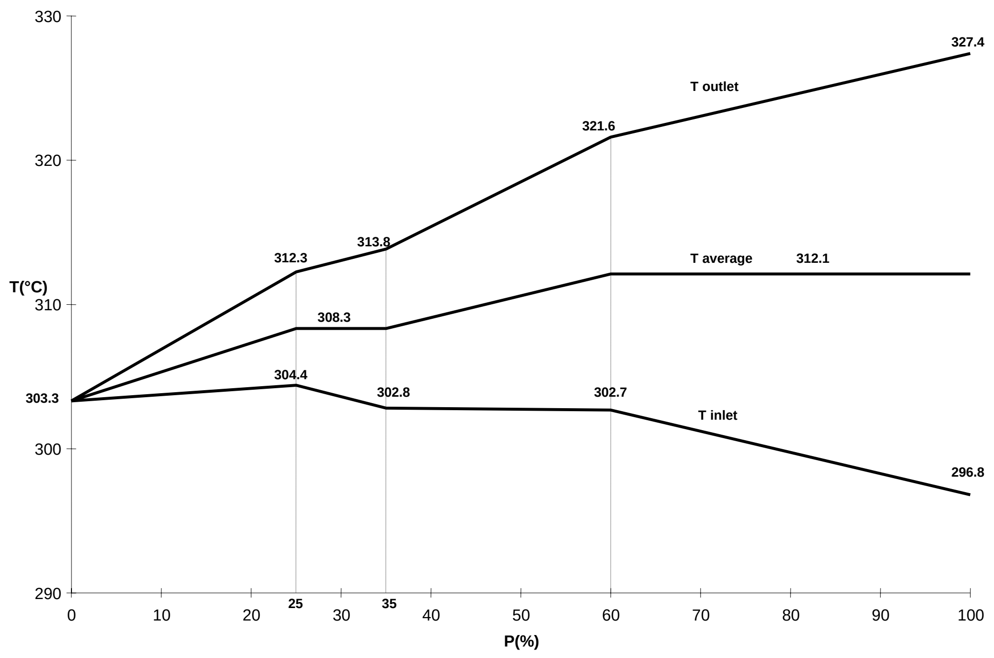
D.4 FIG 2: PART LOAD DIAGRAM (AT THERMO-HYDRAULIC FLOW RATE)



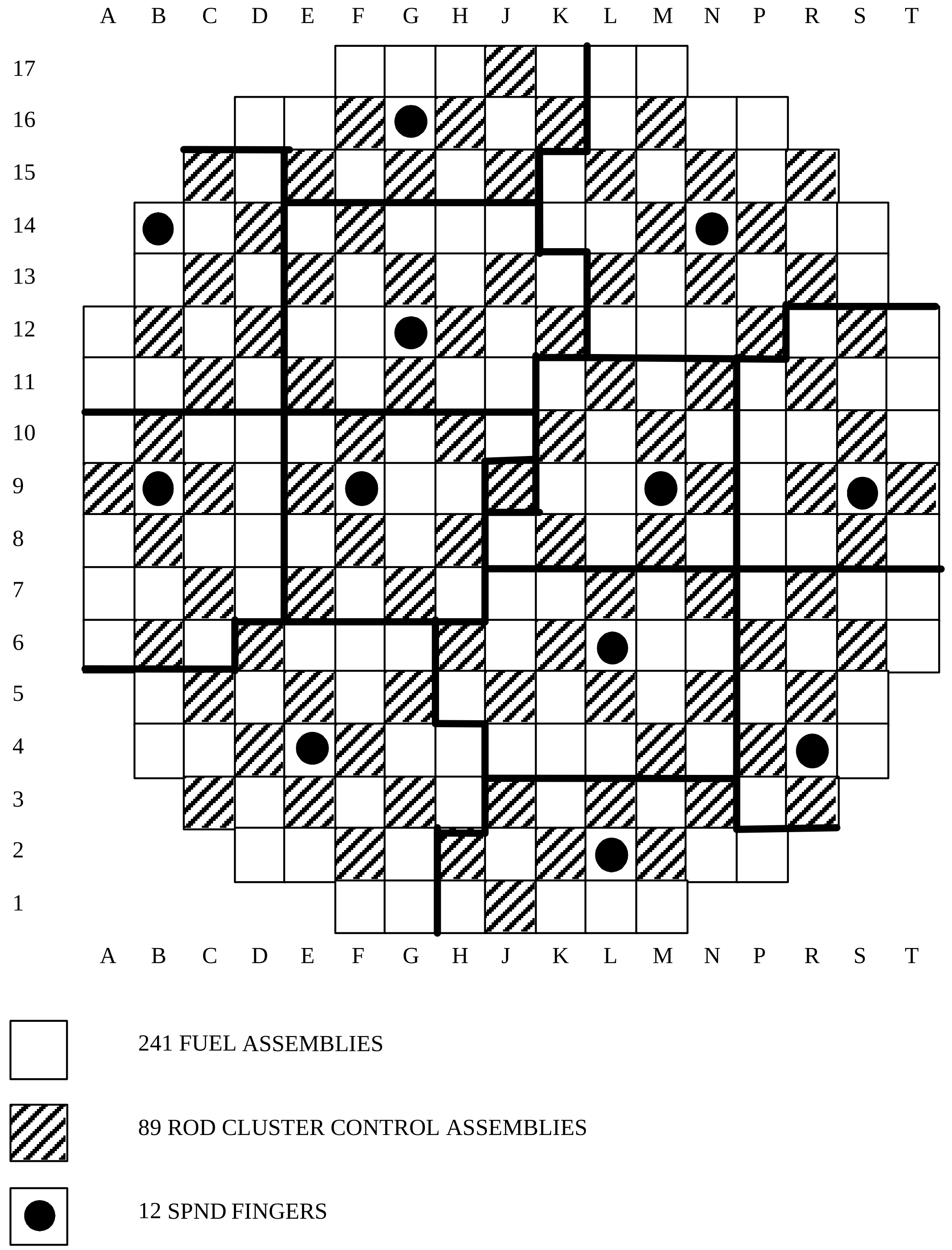
D.4 FIG 2: PART LOAD DIAGRAM (AT BEST ESTIMATE FLOW RATE)



D.4 FIG 2: PART LOAD DIAGRAM (AT MECHANICAL FLOW RATE)



D.4 FIG 3: RADIAL POSITIONS OF THE SELF-POWERED NEUTRON DETECTOR FINGERS



D.4 FIG 4: RADIAL AEROBALL PROBES LOCATIONS