CHAPTER D. REACTOR AND CORE

SUB-CHAPTER D.1. SUMMARY DESCRIPTION

Chapter D describes the nuclear, hydraulic and thermal characteristics of the reactor, the proposals made at the present stage of the EPR design with regard to the mechanical characteristics of the fuel assemblies, and the objectives of the nuclear and of the thermal-hydraulic designs.

1. SUMMARY DESCRIPTION OF THE CORE AND FUEL ASSEMBLIES

The reactor core contains fuel material where the fission reaction, which produces energy occurs.

All the core equipment is used either to physically support the fissile material, to control the fission reaction or to channel the coolant.

The reactor core consists of a specified number of fuel rods which are held in bundles by spacer grids and top and bottom fittings. The fuel rods consist of uranium or MOX (uranium plus plutonium) pellets stacked in a cladding tube plugged and seal welded to encapsulate the fuel. The bundles, known as fuel assemblies, are arranged in a pattern which approximates to a circular cylinder.

Each fuel assembly is formed by a 17 x 17 array, made up of 265 fuel rods and 24 guide thimbles.

The 24 non-fuel rod positions in the array are equipped with guide thimbles joined to the grids and the top and bottom nozzles. The guide thimbles are used as core locations for rod cluster control assemblies, for neutron source rods and internal instrumentation or they are fitted with plugging devices to limit the bypass flow.

The grid assemblies consist of an "egg-crate" arrangement of interlocked straps. The straps contain spring fingers and dimples for fuel rod support as well as coolant mixing fins.

At the present stage of the EPR design, the exemplary initial core consists of 241 assemblies organised into 3 regions with different enrichments.

For future reloads, the number and the characteristics of the fresh assemblies depend on the type of the fuel management foreseen, i.e., cycle length, type of loading, MOX core, etc. Fuel cycle lengths of 22 and 18 months, IN/OUT fuel management types uranium and MOX fuel are possible and may be used with the core described herein.

The core is cooled and moderated with light water at a pressure of 155 bar in the primary coolant system.

2. SUMMARY DESCRIPTION OF THE REACTIVITY CONTROL METHOD

The moderator coolant contains soluble boron as a neutron poison. The boron concentration in the coolant is varied as required to control relatively slow reactivity changes including the effects of fuel burnup. Additional neutron poison (Gadolinium), in the form of commixed burnable poisoned rods, is used to establish the required initial reactivity and power distribution.

The core reactivity and the core power distribution are also controlled by movable control rods consisting of absorber rods that allow the performance of rapid reactivity variations.

Each control rod assembly consists of a group of individual absorber rods fastened at the top end to a common hub or spider assembly.

The control rod assemblies are arranged in several groups.

The control rod drive mechanisms RGL [CRDM] are used to insert, hold or withdraw the control rods.

They consist of electromechanical devices fixed to the reactor vessel cap.

They are used to change the control rod assembly position and to ensure reactor trip by gravity drop. The control rod gravity drop is obtained by cutting off electrical supplies to the RGL [CRDM]

3. OBJECTIVE OF NUCLEAR AND THERMO-HYDRAULIC DESIGN ANALYSES

The nuclear design analyses and calculations establish physical locations for control rods, burnable poison rods, and physical parameters such as fuel enrichments and boron concentration in the coolant. The nuclear design evaluation confirms that the reactor core has inherent characteristics which, together with corrective actions of the reactor control and protective systems, provide adequate reactivity control even if the highest reactivity worth control rod assembly is stuck in the fully withdrawn position.

The design also provides for inherent stability against diametrical or radial and axial power oscillations and for control of induced axial power oscillation through the use of control rods.

The thermal-hydraulic design analyses and calculations establish coolant flow parameters which ensure that adequate heat transfer is provided between the fuel cladding and the reactor coolant. The thermal design takes into account local variations in dimensions, power generation, flow distribution, and mixing. The mixing fins incorporated in the fuel assembly spacer grid design induce additional flow mixing between the various flow channels within a fuel assembly as well as between adjacent assemblies. Instrumentation is provided in and out of the core to monitor the nuclear, thermal-hydraulic, and mechanical performance of the reactor and to provide inputs to automatic control functions.

4. COMPILATION OF MAIN DATA

The main reactor nuclear, thermalhydraulic and mechanical design parameters are presented in 4.1 D.1 TAB 1.

5. DESIGN TOOLS AND METHODS

The analytical techniques used in the core design are presented in table D.1 TAB 2.

TAB 1 : REACTOR DESIGN PARAMETERS[1]

A - Thermal and hydraulic design parameters:	
1 - Reactor core heat output (100 %) (MWth)	4500
2 - Number of loops	4
3 - Heat generated in fuel (%)	97.4
4 - Nominal system pressure (bar)	155
5 - DNB predictor	FC-CHF[2] Correlation
6 - Minimum DNBR under nominal operating conditions ($F\Delta H$ = 1.61 - cos 1.45)	2.6
7 - Minimum initial DNBR for the Basic Design transient analyses	see Chapter P
B - Coolant flow:	
8 - Thermal design flowrate/loop (m^3/h)	27195
9 - Core bypass flowrate (%)	5.50
10 - Core flow area for heat transfer (m^2)	5.9
11 - Average velocity along fuel rods (m/s)	4.8
12 - Core average mass velocity ($g/cm^2.s$)	356

[1] The sizes are given in cold conditions (20°C)
[2] Critical Heat Flux

REACTOR DESIGN PARAMETERS

C - Coolant temperature:	
13 - Nominal inlet (°C)	295.7
14 - Average rise in vessel (°C)	34.2
15 - Average rise in core (°C)	36.0
16 - Average in core (°C)	313.7
17 - Average in vessel (°C)	312.8
D - Heat transfer:	
18 - Heat transfer surface area (m²)	8005
19 - Average core heat flux (W/cm²)	54.7
20 - Maximum core heat flux (nominal operation) (W/cm²)	157.3
21 - Average linear power density (based on cold dimensions) (W/cm)	163.4
22 - Peak linear power for normal operating conditions (W/cm)	470
23 - Peak linear power protection setpoint (W/cm)	590
24 - Peak linear power for prevention of centreline melt (W/cm)	> 590
25 - Power density in hot conditions (KW/core litre)	94.6

REACTOR DESIGN PARAMETERS

E - Vessel and core pressure losses:	
26 - Reactor vessel (bar):	1.66
27 - Core (bar):	2.55
F - Fuel assemblies (The sizes are given in cold conditions (20°C)):	
28 - Rod array	17 x 17
29 - Number of fuel assemblies	241
30 - Rods per assembly	265
31 - Fuel assembly pitch (cm)	21.504
32 - Fuel assembly length without hold-down spring (cm)	480
33 - Lattice rod pitch (cm)	1.26
34 - Overall transverse dimensions (cm)	21.4 x 21.4
35 - Fuel weight per assembly (kg)	598 UO_2, 527.5 U
36 - Number of grids per assembly	10
37 - Composition of grids	Zircaloy & Inconel
38 -Number of guide thimbles per assembly	24
39 -Number of instrumentation thimbles per assembly	0

REACTOR DESIGN PARAMETERS

G - Fuel rods (The sizes are given in cold conditions (20°C))	
40 - Number	63865
41 - Outside diameter (mm)	9.50
42 - Diametral gap (mm)	0.17
43 - Clad thickness (mm)	0.57
44 - Clad material	M5
H - Fuel pellet (The sizes are given in cold conditions (20°C)):	
45 - Material	UO_2 or MOX
46 - UO_2 density (% of the theoretical density)	95
47 - UO_2 + PuO_2 density (% of the theoretical density)	94.5
48 - Diameter (mm)	8.19

REACTOR DESIGN PARAMETERS

I - Rod cluster control assemblies (The sizes are given in cold conditions (20°C)):	
49 - Absorber:	
1) AIC part (lower part)	
Composition (% wt):	
Ag	80
In	15
Cd	5
Density (g/cm³)	10.17
Absorber outer diameter (mm)	7.65
Length (mm)	1500
2) B4C part (upper part)	
Composition: natural boron (19.9 atoms of percent B10)	
Density (g/cm³)	1.79
Absorber diameter (mm)	7.47
Length (mm)	2610
50 - Cladding:	
Outer diameter (mm)	9.68
Inner diameter (mm)	7.72
Thickness (mm)	0.98
Material	Stainless steel
51 - Number of clusters, full length	89
52 - Number of absorber rods per cluster	24

REACTOR DESIGN PARAMETERS

J - Active core (The sizes are given in cold conditions (20°C)) :	
53 - Equivalent diameter (mm)	3767
54 - Core average active fuel height (mm)	4200
55 - Height-to-diameter ratio	1.115
56 - Total cross-section area (cm^2)	111 440
K - Heavy, radial reflector (The sizes are given in cold conditions (20°C)):	
57 - Thickness (mm)	Between 77 and 297 (average 194)
58 - Composition (% volume)	Roughly 95 % steel, 4 % water

REACTOR DESIGN PARAMETERS

L - Fuel enrichment:	
For UO2 fuel assemblies (% wt):	
59 - Region 1 of cycle 1	2.1 %
60 - Region 2 of cycle 1	3.2 %
61 - Region 3 of cycle 1	4.2%
62 - Fresh assemblies for UO_2 - INOUT - 18 months	5.0 %
63 - Fresh assemblies for UO_2 - INOUT - 22 months	5.0 %
For MOX fuel assemblies (% wt)[3]:	
64 - Maximum fissile Pu enrichment for the zone 1	7.44 %[1)]
65 - Middle fissile Pu enrichment for the zone 2	6.44 %[1)]
66 - Minimum fissile Pu enrichment for the zone 3	3.44 %[1)]
67 - Average fissile Pu enrichment	7.0 %[1)]
68 - Assumption for UO_2 enrichment in MOX fuel (% U235 wt)	0.2 %[1)]
M - Pu vector for MOX fuel assemblies:	
Issued of a UO_2 fuel burned up to 60 GWd/t (% wt):	

[3] The fissile Pu enrichments is defined as $e = \frac{Pu239 + Pu241}{(U + Pu + Am)}$

69 - Pu 238	4.0
70 - Pu 239	50.0
71 - Pu 240	23.0
72 - Pu 241	12.0
73 - Pu 242	9.5
74 - Am 241	1.5

TAB 2 : ANALYTICAL TECHNIQUES IN CORE DESIGN

ANALYSIS	TECHNIQUE	COMPUTER CODE
1 - Nuclear design		
- Cross sections and group constants	Macroscopic data	APOLLO 2
- Power distributions, fuel burnup critical boron concentration, xenon distributions, reactivity coefficients, rod worths.	3-D, 2-group, diffusion evolution theory	SMART
2 - Thermal - Hydraulic design	Subchannel analysis of local fluid conditions (treatment performed for the core, the assembly and the hot channel)	FLICA