EDUCATIONAL EXPERIENCES PROGRAM IN NUCLEAR ENGINEERING AREA.

SIREP 1300 NUCLEAR POWER PLANT CONCEPTUAL SIMULATOR (DFEN-ETSEIB-UPC)

I- Subject: Nuclear Reactor Physics

II- Subject: Nuclear Power Plants

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Glossary

BOL: Beginning Of Life. Reactors state at beginning of the fuel cycle life.

DDV (*DDC*): Debut De Vie (Cycle). *BOL* equivalent in French terminology.

DNBR: Departure from Nucleate Boiling Ratio. Relation that shows the boiling crisis range.

EOL: End of Life. Reactors state at the end of the fuel cycle life.

FDV (*FDC*): Fin De Vie (Cycle). *EOL* equivalent in French terminology.

GCT: Groupe de Contournement de Turbine. It allows the generators steam dump.

GV or SG: Steam Generators.

LWR: Light Water Reactor.

MAR: Antireactivity range or shutdown range.

MDV (*MDC*): Moitié De Vie (Cycle). *MOL* in French terminology.

MOL: Middle Of Life. Reactors state at the middle of the fuel cycle life.

MOX: Mixed Oxyde. Uranium-plutonium oxide mixture (UO_2/PuO_2).

PTR: Pool water cooling and clean up treatment circuit.

PWR: Pressurized Water Reactor.

RCP: Circuit Primaire. Primary Circuit.

RCV: Circuit de Contrôle Volumétrique et chimique. Chemical and volume control system.

REA: Water and Boron contribution circuit.

REC: Rapport d'Ebullition Critique. *DNBR* equivalent in French terminology.

REP: Réacteur d'Eau sous Pression. *PWR* equivalent in French terminology.

RRA: Système de Refroidissement à l'Arret. Residual heat removal system.

RRI: Système de Réfrigération Intermédiaire. Intermediate cooling system.

TEP: Effluent treatment system.

TPA: Turbo Pompe Alimentaire. Feedwater turbopump of the steam generators main feedwater system.

Introduction

Last 27 April of 2004, the *Excma. Presidenta del Consejo de Seguridad Nuclear Sra. María Teresa Estevan Bolea* and the *Universidad Politécnica de Cataluña* Rector with several academic and nuclear authorities inaugurated the *Simulador Conceptual de Central Nuclear DFEN-ETSEIB-UPC*.

Conceptual Simulator inauguration DFEN-ETSEIB-UPC by the Excma. Presidenta del Consejo de Seguridad Nuclear Sra. Maria Teresa Estevan Bolea and the UPC rector.

This facility startup has represented a pulse and educational modernization in the Nuclear specialization area, in the ETSEIB. At the present, the nuclear specialization has a 64 ECTS (*European Credit Transfer System*) length that are equivalent to 64,5 credits or 645 school hours.

The *SIREP 1300* simulator is a basic principle simulator that represents a *PWR* 4 looped power plant of *1300 MWe*.

Five independent simulators are in disposition where every student operates his own nuclear plant, following the modus operandi showed in the experiences book.

Conceptual simulator in operation.

The simulator operation with the *ETSEIB* students has proved to present an intuitive, simple and attractive user interphase. From the second experience hour, the student easy goes the simulator operation and he can focus in the concepts on nuclear physics, operational scenario analysis, determination on plant variables, results understanding, etc.

The Simulator is used in the following subjects:

- *Nuclear reactor physics* (5 ECTS = 6 credits = 60 hours). Five experiences will be done with a total length of 12 hours.

    Experience 1: Reactor kinetic variables.

    Experience 2: Subcritic approximation.

    Experience 3: Reactivity temperature effects.

    Experience 4: Isothermal coefficient and moderator coefficient.

    Experience 5: Reactor start ups and load variations.

- *Nuclear Power Plants* (5 ECTS = 6 credits = 60 hours). Five experiences will be done with a total length of 12 hours.

    Experience 6: Reactor Standard states. Transition from power operation to hot shutdown.

    Experience 7: Reactor Standard states. Transition from hot shutdown to cold shutdown.

    Experience 8: Control rod bank calibration.

    Experience 9: Reactor stabilization.

    Experience 10: Disconnection from electrical grid and house load operation.

This facility development has been possible thanks to the back by from *Consejo de Seguridad Nuclear (CSN), Asociación Nuclear Ascó–Vandellòs II (ANAV), Escola Tècnica Superior d'Enginyeria Industrial de Barcelona (ETSEIB)* and the collaboration of the *International Atomic Energy Agency (Workshop on Nuclear Power Plant Simulators for Education)* and the *Tecnatom S.A*. (Full scope simulator experiences).

1 SIREP 1300 nuclear power plant conceptual simulator presentation

1.1 Simulator model presentation

The *SIREP 1300* simulator is a basic principle simulator that represents a *PWR* 4 looped power plant of *1300 MWe*.

The *SIREP* simulator is a new update made by *CORYSS T.E.S.S.* enterprise with a programming related with objects of an already existing simulator called *SYREN,* made by the same enterprise. SIREP 1300 tries to keep the same educational objectives and basic characteristics related with the simulation. The computer science technological updates have increased the graphical interphase aspects making them much easier and attractive.

1.1.1 Model organization

The simulator model is organized in three different modules that control different simulation aspects. The modules are the following:

- *NEUTRO*: Core and Boron transport modelization.
- *MODELIX*: Thermal-hydraulic circuits (except primary circuit), electric circuits and control instrumentation modelization.
- *DRAC*: Primary circuit thermal-hydraulic modelization.

The modules detailed main characteristics are now shown.

1.1.1.1 Module *NEUTRO*

This module is in charge of the neutronics and the primary Boron concentration modelization.

The module exchanges the value of sensors, devices and interphase variables with the *MODELIX* and *DRAC* modules. It also exchanges the partial accelerations with *MODELIX*.

1.1.1.2 Module *MODELIX*

The module developed with the *MODELIX* tool includes control modelization (logic and regulations) of the electric systems and of the thermal-hydraulics sections not simulated by *DRAC*. These sections are the chemical and volume control system, the emergency cooling system, the Boron contribution system, steam, turbine, condenser and steam generators feedwater systems.

1.1.1.3 Module *DRAC*

This module developed with the *DRAC* tool is in charge of the primary equivalent loop modelization, primary pumps, pressurizer, and steam generator (secondary and primary sections).

The *DRAC* module exchanges the preceding modules and the control (primary pumps shutdown/startup, acceleration) with *NEUTRO* and *MODELIX*.

1.1.2 Main simulation contents

Some of the main simulation contents shown in the module are the following:

- Primary-secondary circuits interactions
- Level increases and decreases
- Important transitories (reactors shutdown in auto mode, house load operation, ...)
- Core physics (moderator temperature effects and Doppler effect, Xenon evolution, ...)
- Subcritic approximation
- Neutron aspects (divergence, reactivity steps, ...)
- Xenon oscillations
- Single-phase behaviour
- Cycle prolongation (stretch-out)

1.1.3 Elements and systems included in the module. Validity operation limits

The module includes:

- A neutronics, pressurizer and steam generator (thought as one single equivalent loop) detailed representations.
- A representation of the following systems and elements:
    - Primary circuit,
    - Chemical and volume control system,
    - Shutdown cooling system,
    - Water and Boron contribution system,
    - Steam secondary system,
    - Turbine,
    - Alternator,

- Condenser,
- Main feedwater system (steam generators),
- Emergency feedwater system (steam generators).

The *PWR* representation can be simplified because its objective is educational, showing the basic nuclear power plant operation principles.

The simulation is limited to the main circuits. It is only represented one single loop because they have a similar behaviour with the exception of particular unbalance that are not studied here.

The validity operation limits covers from the cold shutdown to the full power operation.

The main validity operation limits are the following:

- Primary pressure from *1 bar* up to *220 bar*,
- Every single primary circuit is single-phase with the exception of the pressurizer and the steam generator,
- Pressurizer level not null and lower than *100%* if the primary pressure is higher than *50 bars*,
- Steam generator level not null and lower than *100%*,
- Volume control tank of the chemical and volume control system not null and lower than *100%*.

An auto freeze in the simulation will be done in case of exceeding the operation limits.

The simulator module will be detailed described in chapter 2.

1.2 States and simulation management possibilities

With the aim to obtain an indispensable knowledge for a capable, versatile and effective simulator handling important aspects will be present in this part.

1.2.1 Simulation speed and partial accelerations

The simulation is load in real time but there is the opportunity to dispose a global acceleration. Also, some phenomena can be accelerated in an independent way from the rest of the model in order to allow some practical works development. These works are: fuel consumption, Xenon and Samarium poisoning, primary water heating, boration/dilution, residual power evolution, etc.

Two acceleration modes can be chosen independently: ***global acceleration*** and ***partial acceleration.*** The global acceleration is related to a simulator main acceleration or slow-motion mode. It corresponds to a simulator time step global multiplication. The partial accelerations are those ones that allow accelerating one operation or a module operation section.

The user is capable of modifying the simulation speed by selecting the desired speed in a submenu of '*vitesse*' option. The possibilities are the following:

Global accelerations:

- Real time '*temps reel*', '*reel*0,1*', '*reel*0,5*', '*reel*2*' and '*reel*3*': slow-motion or global acceleration. Every single phenomenon is modified in speed.
- Pseudo real*20 '*Pseudo reel*20*': carries out a global acceleration by a 20 factor. Only the thermal transfer inside the steam generator tubes is accelerated in a lower factor. This solution makes this acceleration mode impossible to use in the shutdown states.

Partial accelerations:

- '*Poisons*60*': a 2 value global acceleration is established with a poison appearance and disappearance speed increase (factor 60). The shown simulation time corresponds to 2 value global acceleration.

- Fuel '*Combustible*': Only the fuel consumption is accelerated in this mode. The fixed simulation time is the accelerated phenomena (50000).

- Heaters '*Chauffage*': the dissipated power by primary pumps and the pressurizer heater is multiplied by an acceleration coefficient (10). The fixed simulation time is the accelerated phenomena.

- Boron '*Bore*': Only the boration and dilution are accelerated. The shown simulation time corresponds to the studied phenomena (50).

- Residual power '*P résiduelle*': Only the residual power evolution is accelerated. The fixed simulation time is the accelerated phenomena (5000).

1.2.2 Files management

The option '*Cliché*' allows the different memorized states management. These states are classified in: *Standard*, *instructor* and *periodical*.

- The *Standard* states are related with the main states and have been made with collaboration of *CORYS T.E.S.S.* and the *INSTN*. The available list has main states in the main plant operation modes and in the different moments of the fuel cycle.

- The *instructor* states are related with the ones created and saved during users operation. It is possible to create new states by others evolutions or saved them as a simulation in operation security copy. Some of the proposed experiences have a state carried out with singularity conditions as a start point.

- The *periodical* states are the ones that are saved in auto mode every time gap. They are always related with the present simulation history and will disappear when the next simulation is started.

The instructor can get a state in every moment, that is, it can save in a determined moment the simulator state. Furthermore, the present state is saved in the instructor states list.

The options '*pannes*' and '*comandes locales*' are two simulator main characteristics.

The option '*pannes*' (damage) allows the damage management to perturb the simulation. It allows the instructor to define and to start the damages that desire to introduce during the simulation. Among them, it is possible to choose: power steps, regulation bank insertion/withdrawn, reactivity steps, damage in the instrumentation, reactor auto shutdown inhibition, primary pumps shutdown, pressurizer main relief valve opening, residual heat removal pumps shutdown, lost of *RRI,* vacuum lost in the condenser, main steam safety valve opening in the steam generators, etc.

The option '*comandes locales*' (local orders) allows the manual elements management. It allows the instructor to define and to start the elements local orders. Among them, there can be selected: Doppler effect supression, choose an initial reactivity (fuel), Boron concentration blockage and adjustment, etc.

1.3 Synoptic behaviour representation

The graphic interphase can be described as a group formed by synoptic diagrams and picture screens where the simulator presents the different systems, variables and screen phenomena. These representations are also interactive, allowing the easy elements and systems operation by the user.

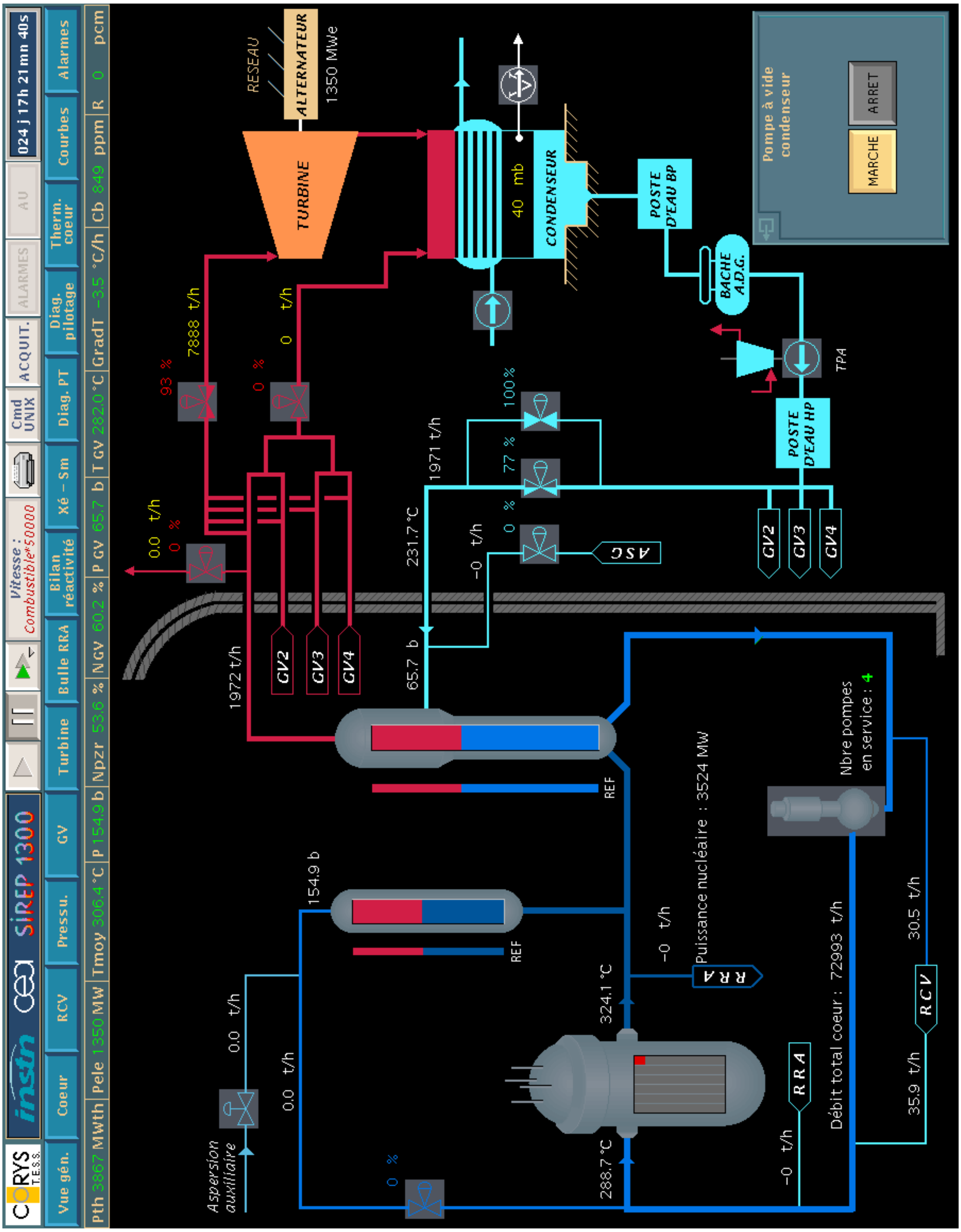
The simulator synoptic diagrams and picture screens are the following:

- Plant overview, primary and secondary circuit main views (reactor, pressurizer, steam generatos, turbine, alternator, condenser, main feedwater system)
- Reactor (vessel, core, control rods)
- Chemical and volume control system (including primary circuit scheme)
- Pressurizer (heaters and main and auxiliary sprinkling)
- Steam generator (including feedwater and steam lines)
- Turbine (including secondary circuit and electric and auxiliary systems)
- Residual heat removal system
- Reactivity balance diagram (with picture screens of every aspect in the balance contribution)
    - Fuel: Cross section and main isotopes burnup evolution graphic representation.
    - Doppler: Diminution effect and BOL and EOL graphic comparison.
    - Moderator: Moderator coefficient (α) curves related with the Boron concentration.
    - Power effects: Power effect curves in BOL, MOL and EOL.
    - Xenon: Antireactivity evolution graphic representations due to Xenon during startups and shutdowns.
    - Samarium: Shutdown and startup Samarium evolution.
    - Boron: Boron differential efficiency curves.
    - Control rods: Differential and integral efficiency graphic representations.
- Hydraulic analogous diagram of the Samarium and Xenon evolution.

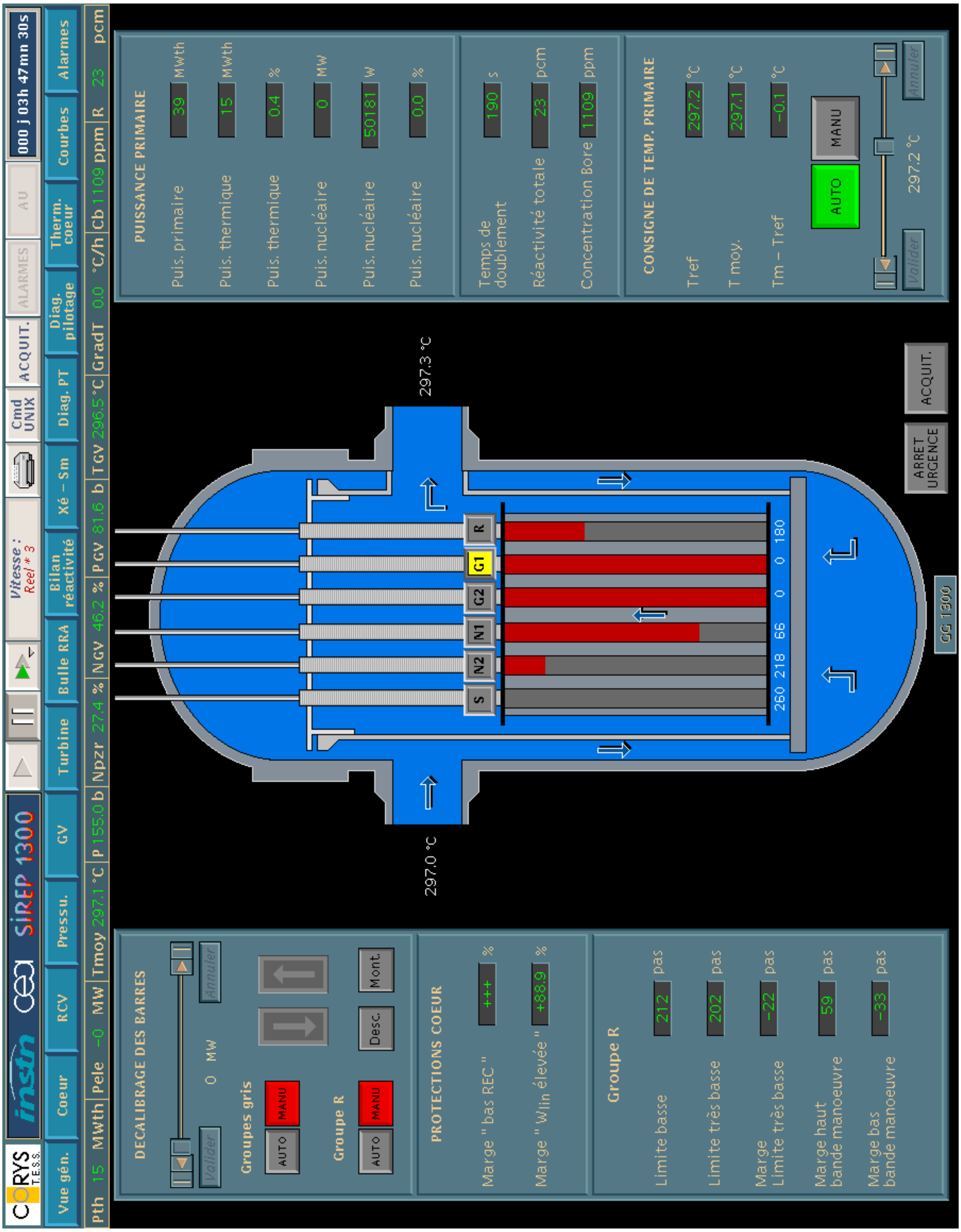
- Pressure-Temperature diagram (Operation zones description).
- Control diagram (including the axial flux, temperature and Xenon profiles).
- Thermal dynamic picture diagrams.
- Different variables evolution curves screen.
- Alarm board.

Every last described screen, that shape the simulator graphic interphase, are followed presented in figures.

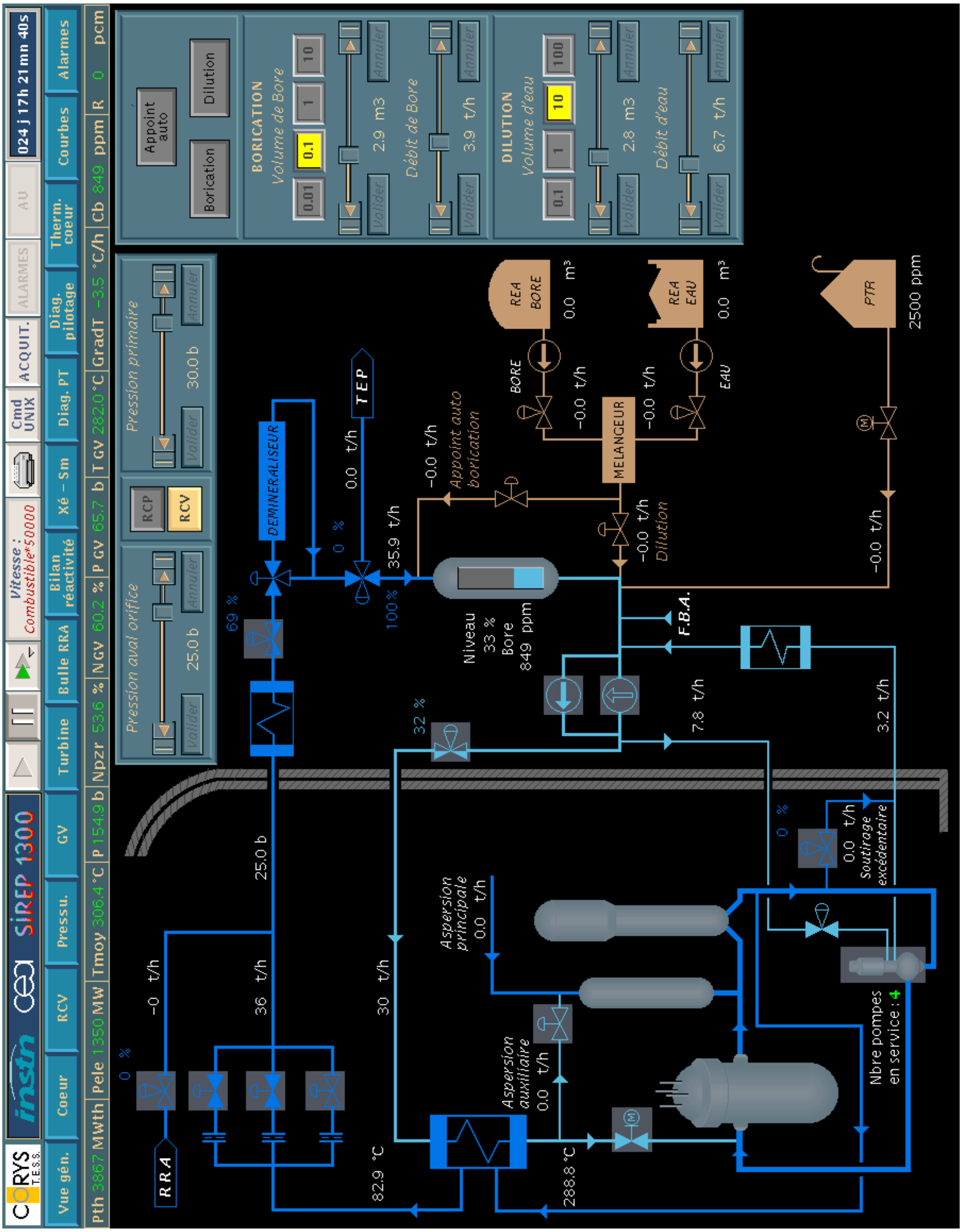
1.3.1 Plant overview representation



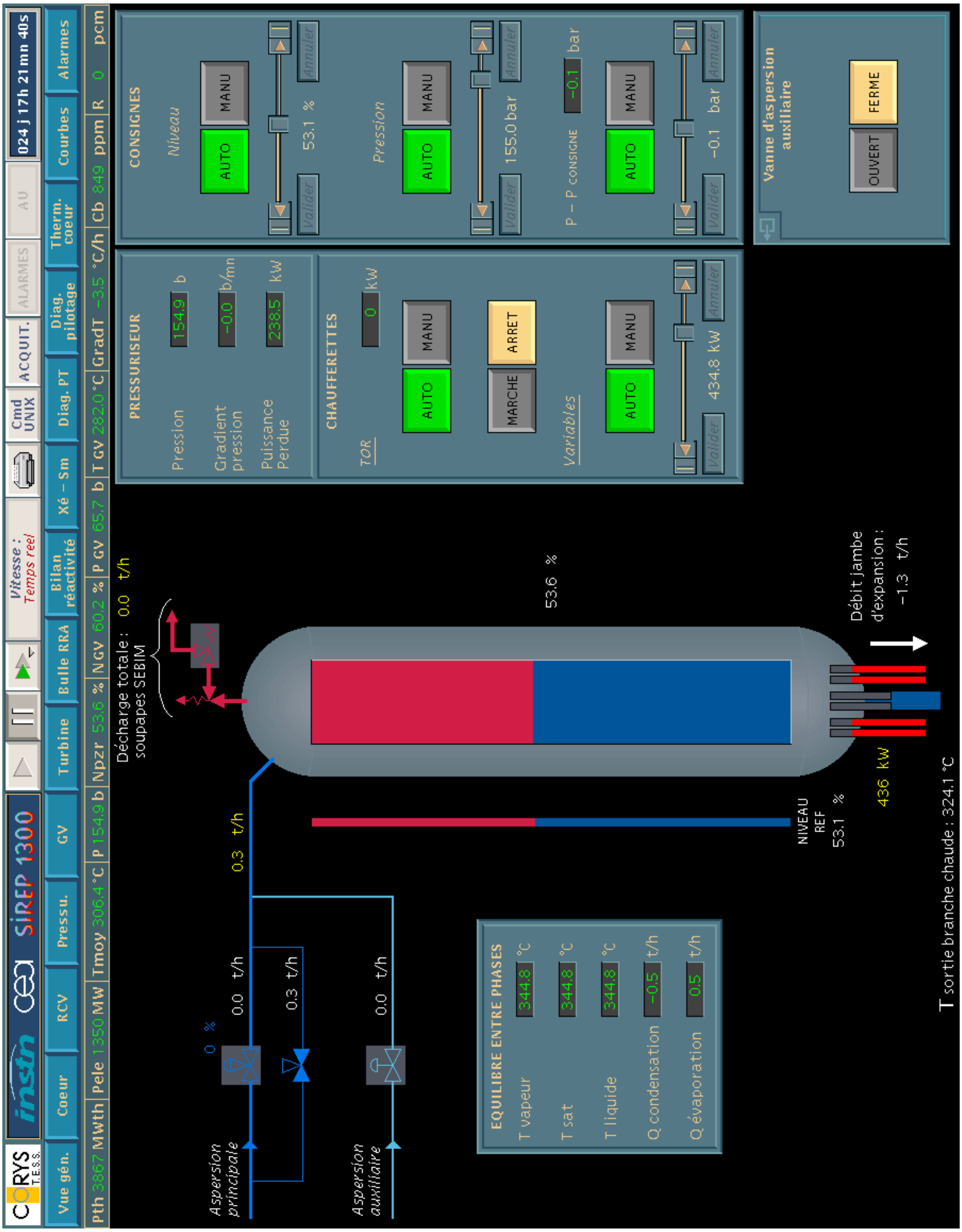
1.3.2 Reactor representation



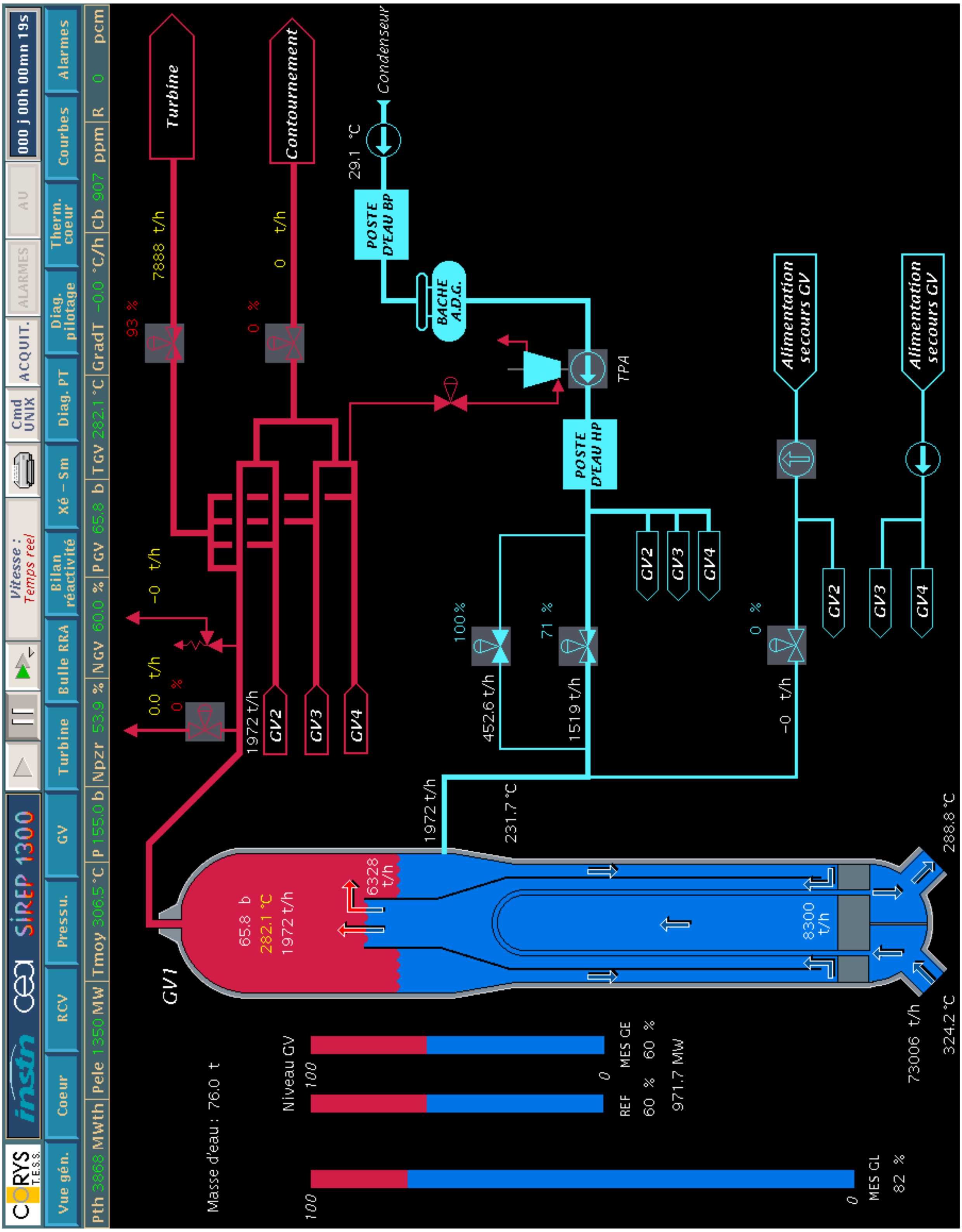
1.3.3 Chemical and volume control system representation



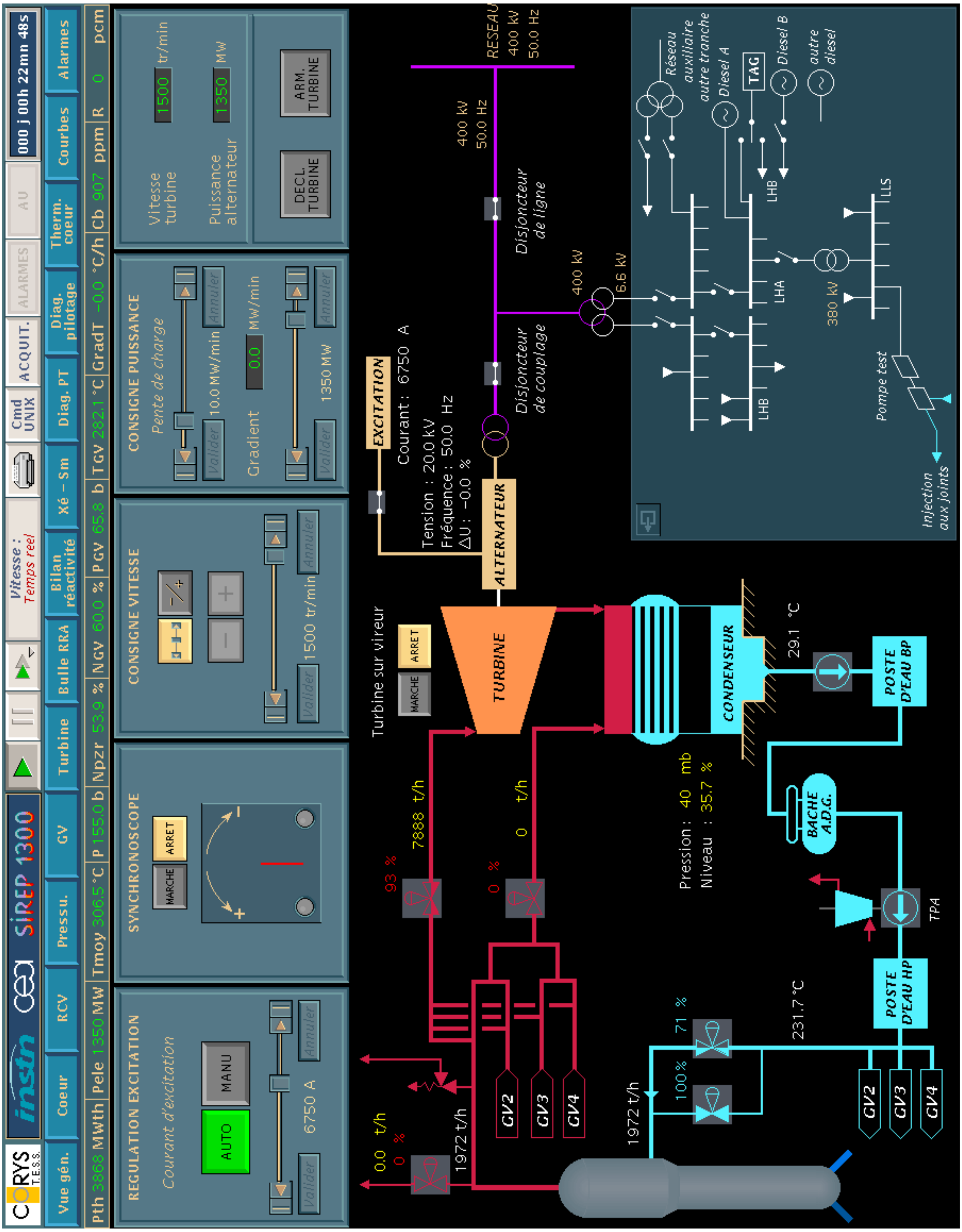
1.3.4 Pressurizer representation



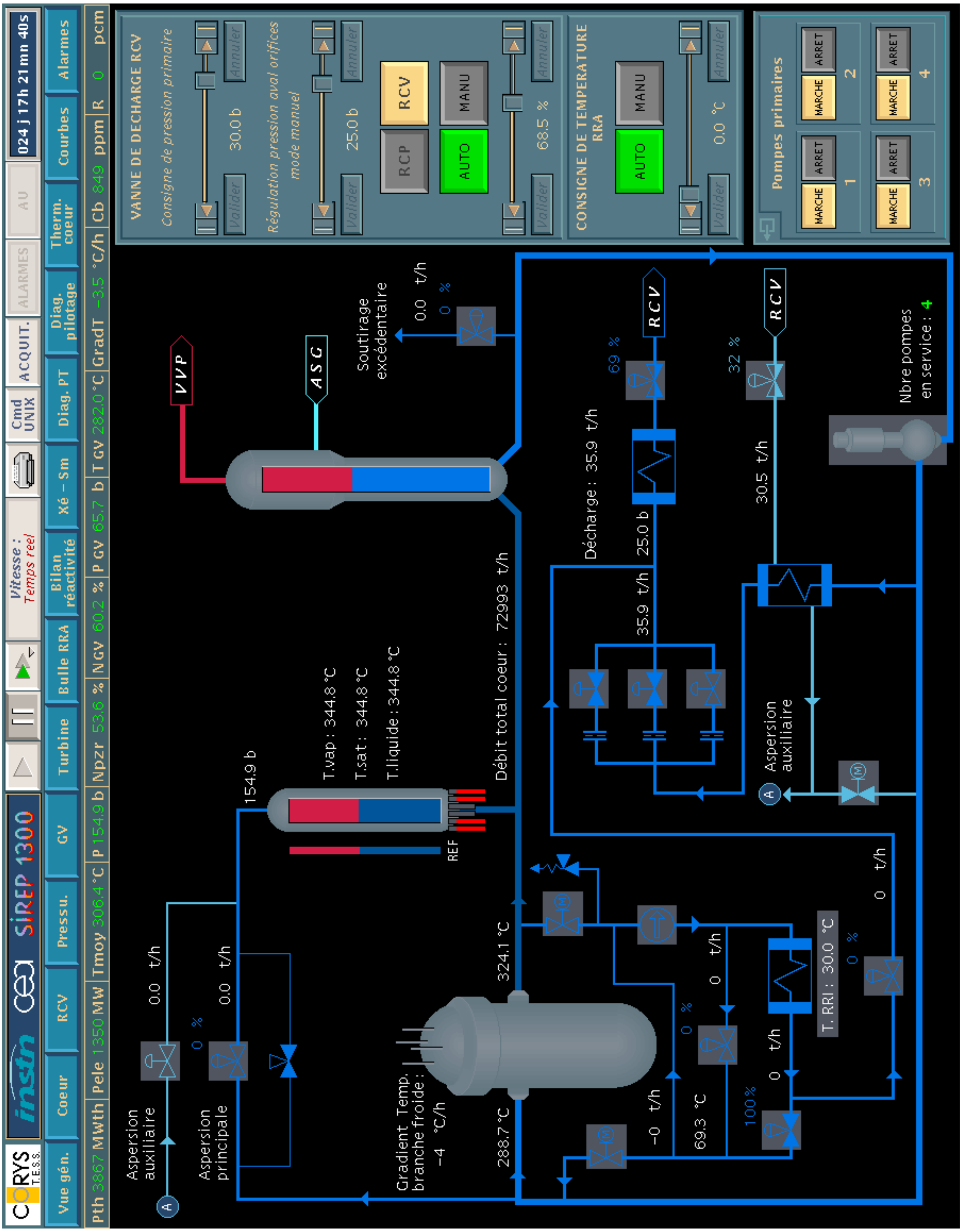
1.3.5 Steam generator representation



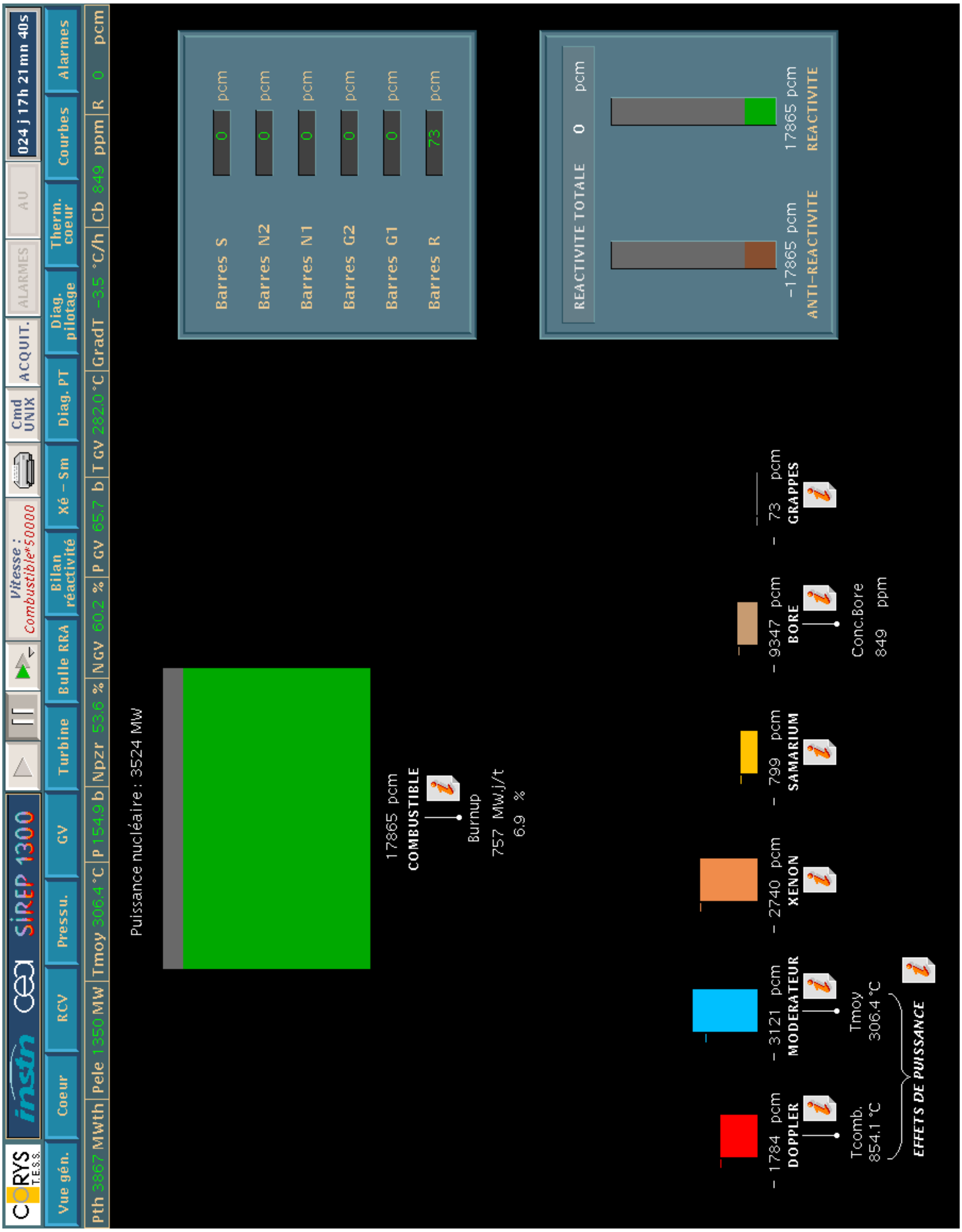
1.3.6 Turbine representation



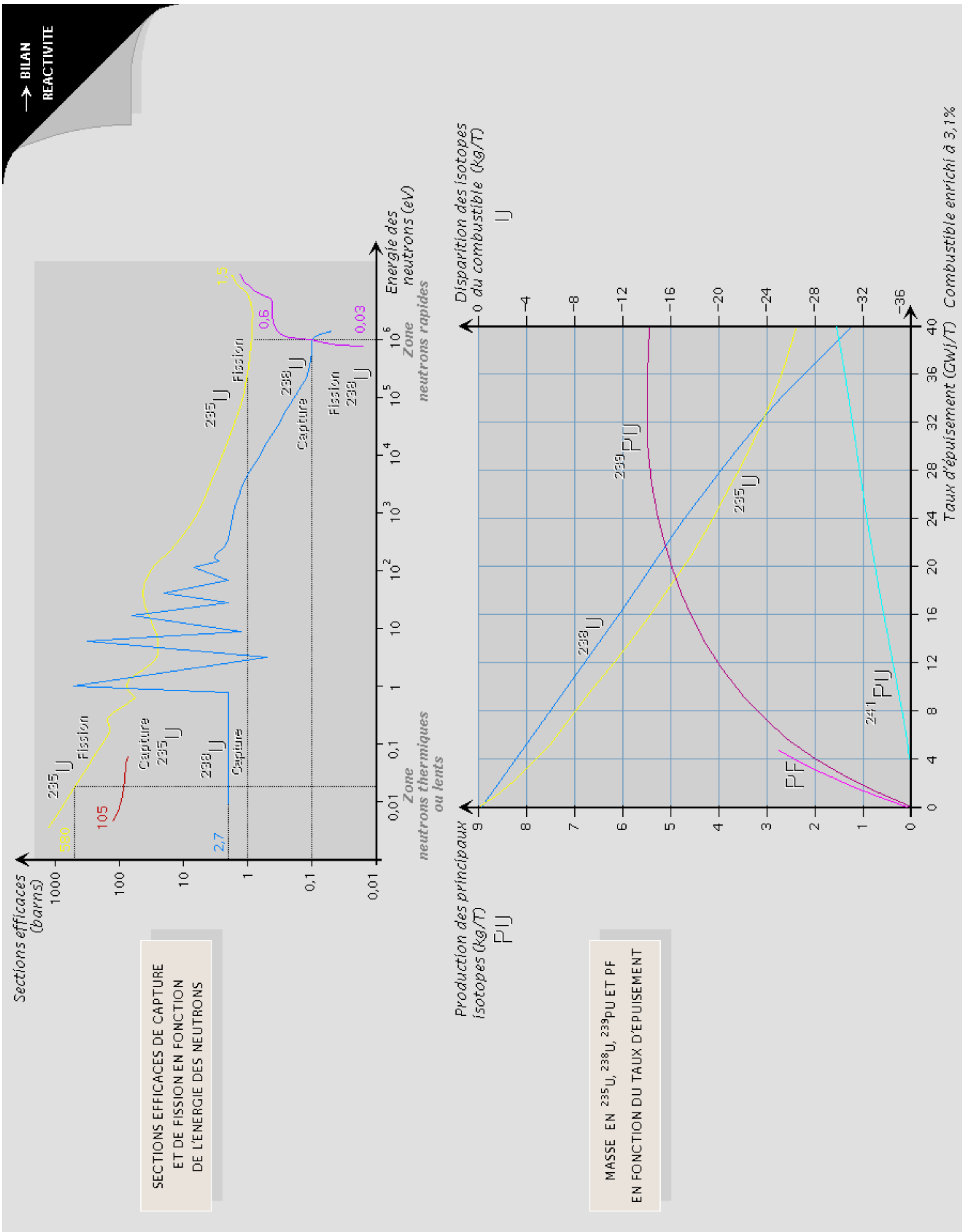
1.3.7 Residual heat removal system representation



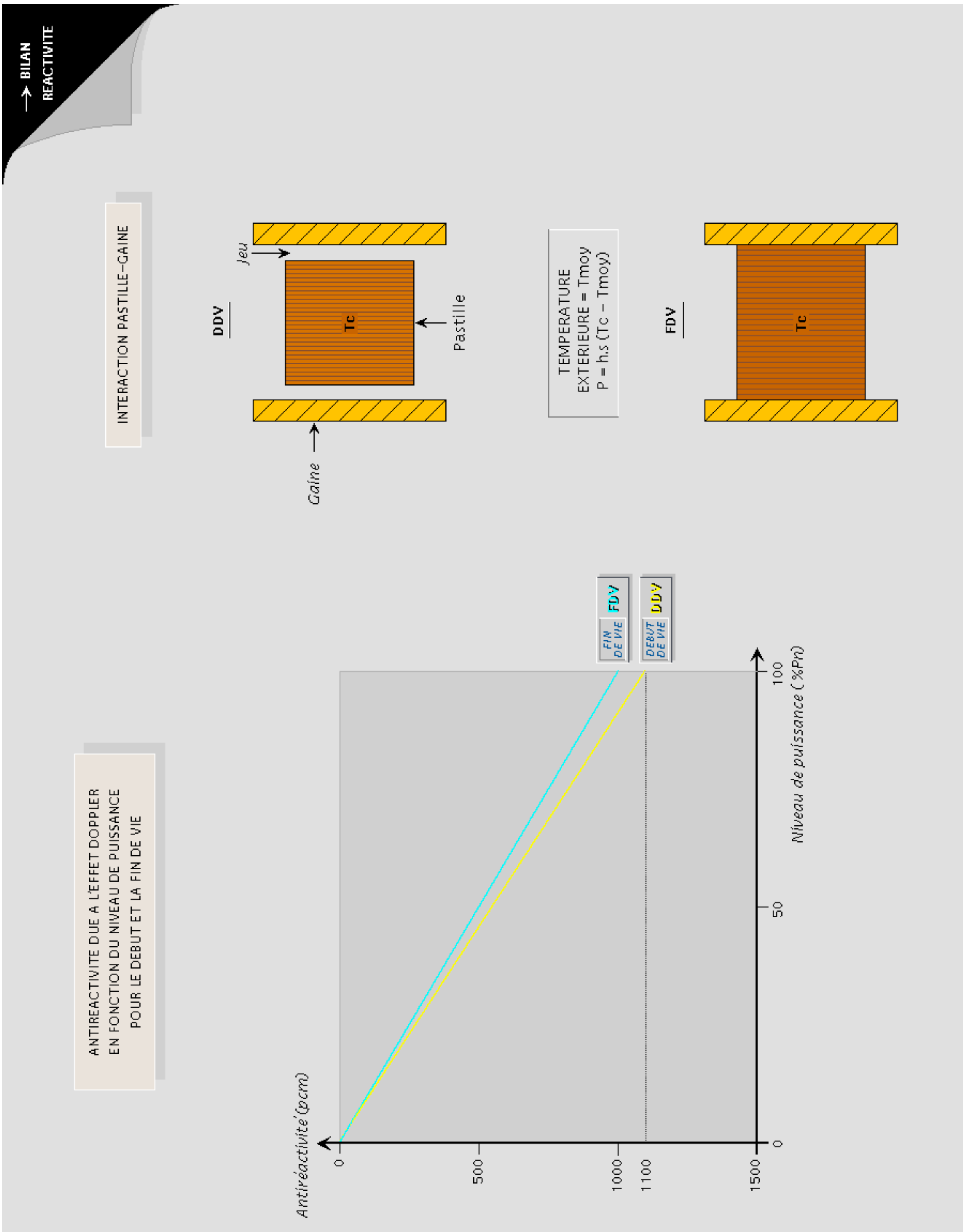
1.3.8 Reactivity balance diagram



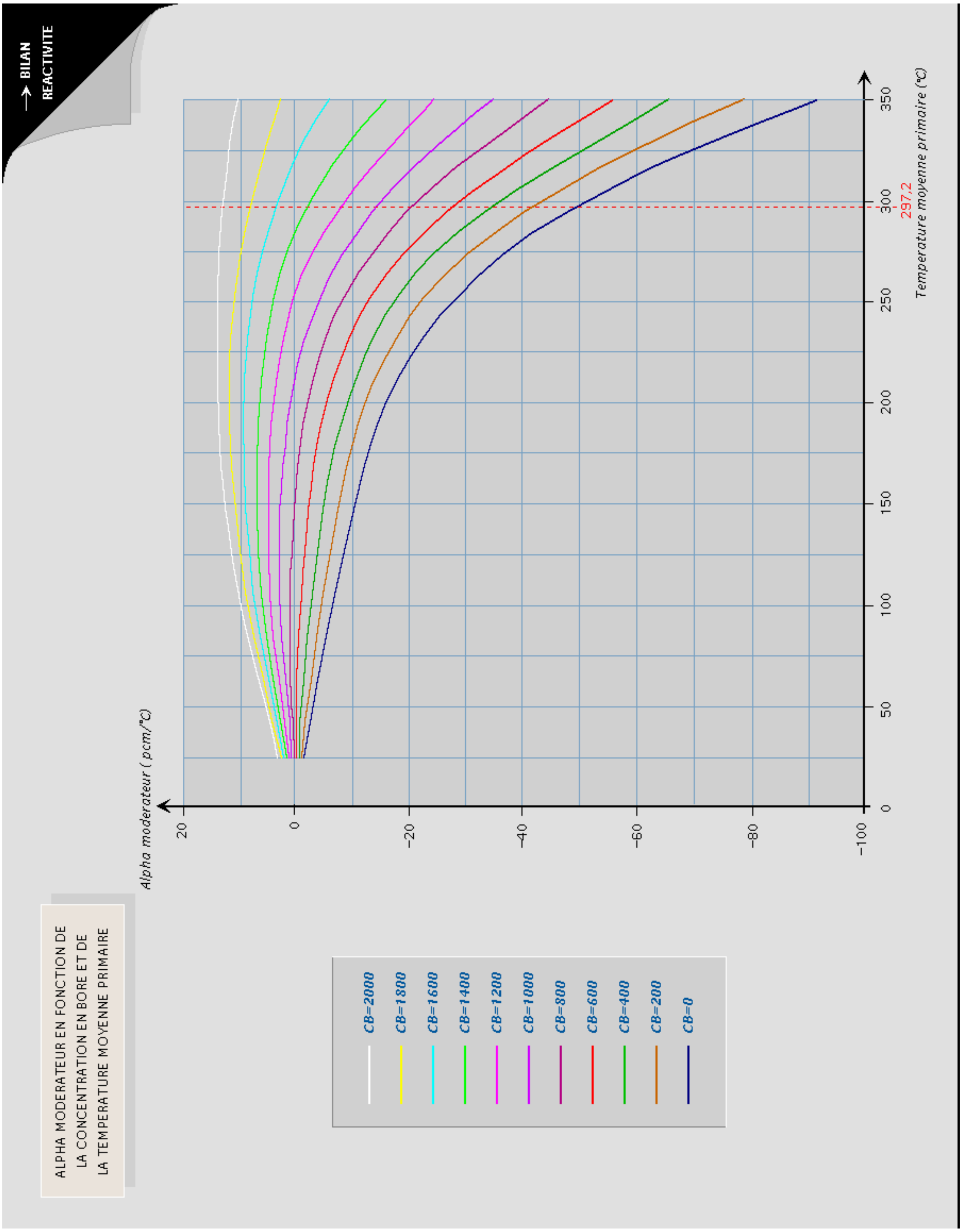
1.3.8.1 Educational images and diagrams - Fuel



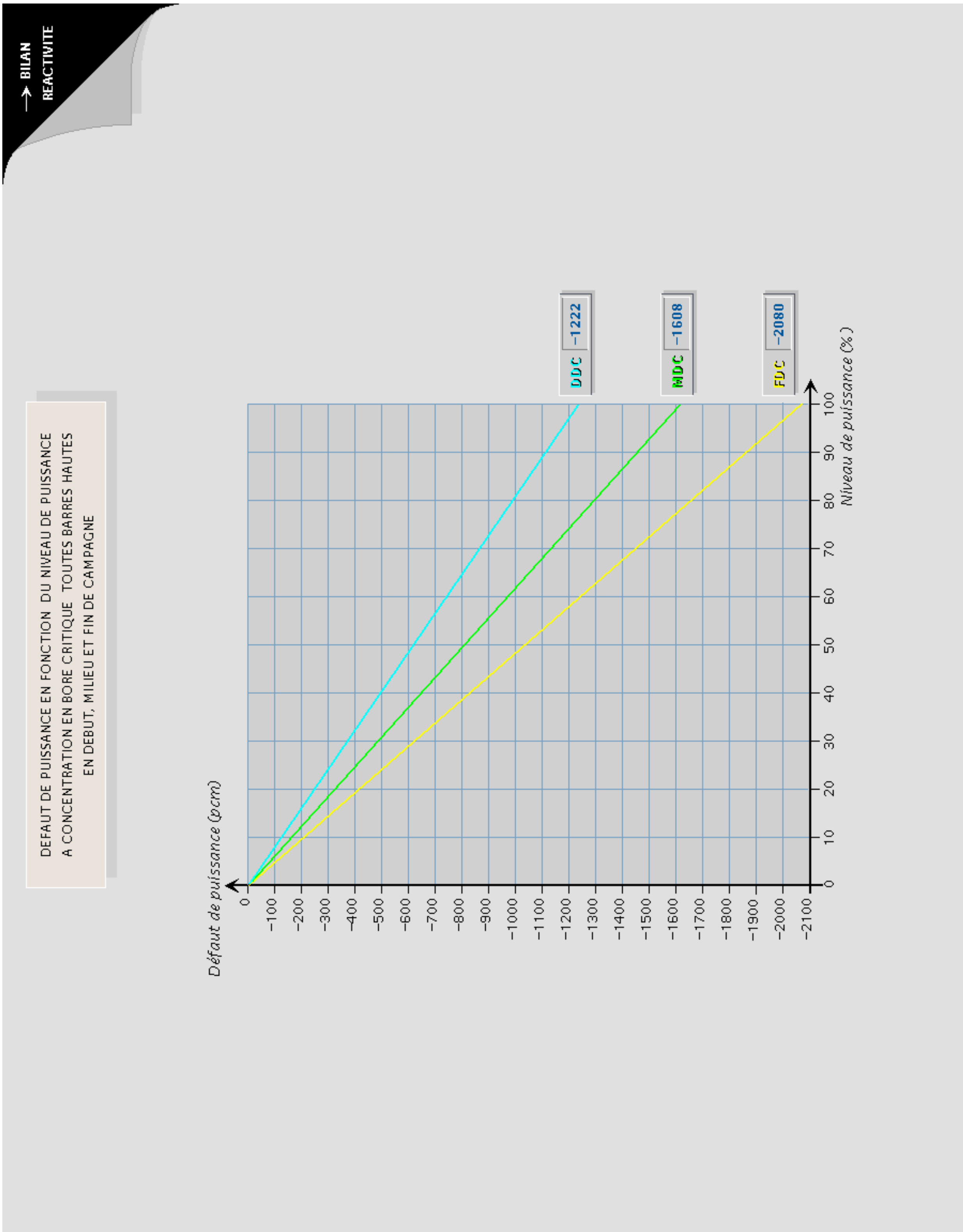
1.3.8.2 Educational images and diagrams – Doppler



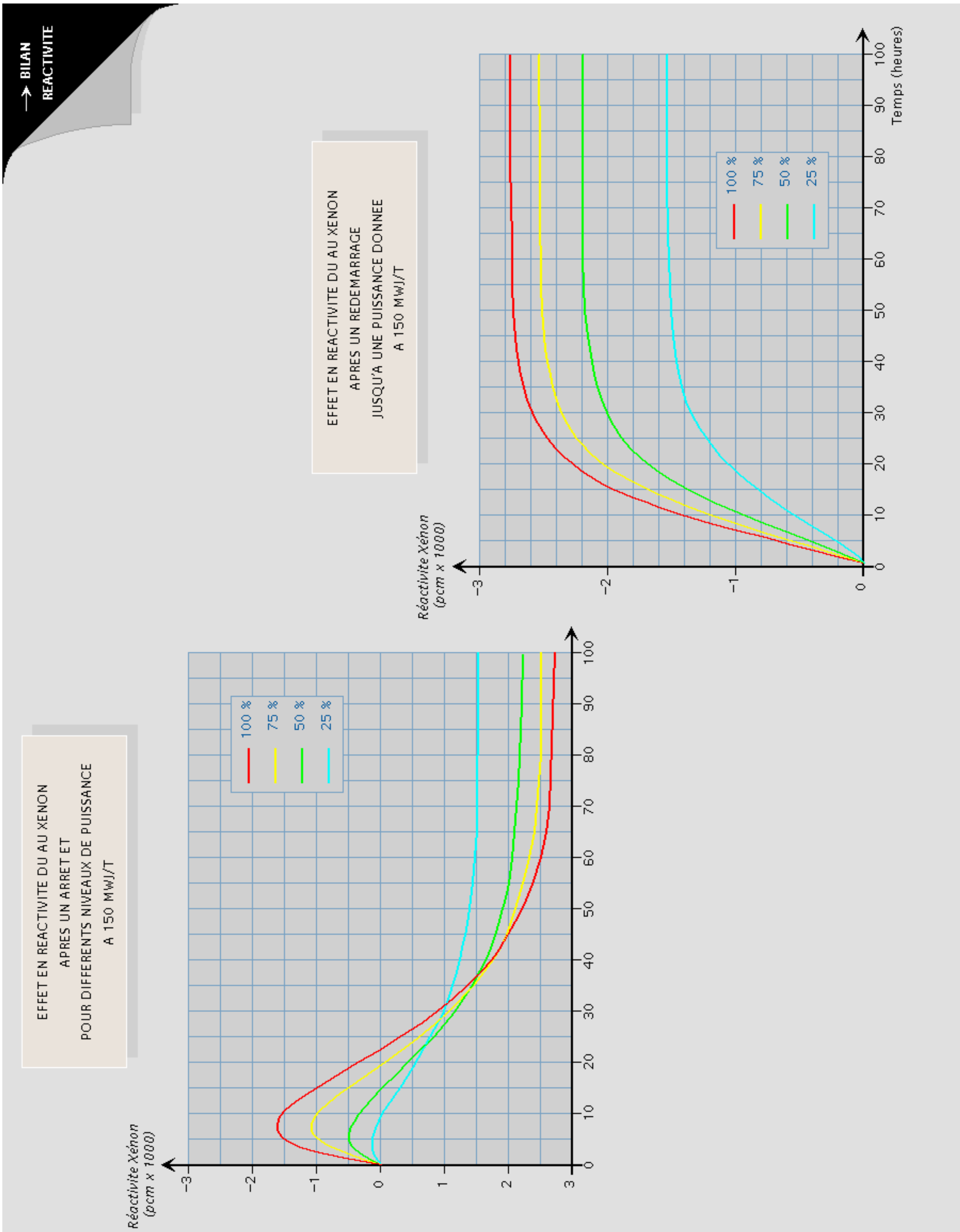
1.3.8.3 Educational images and diagrams – Moderator



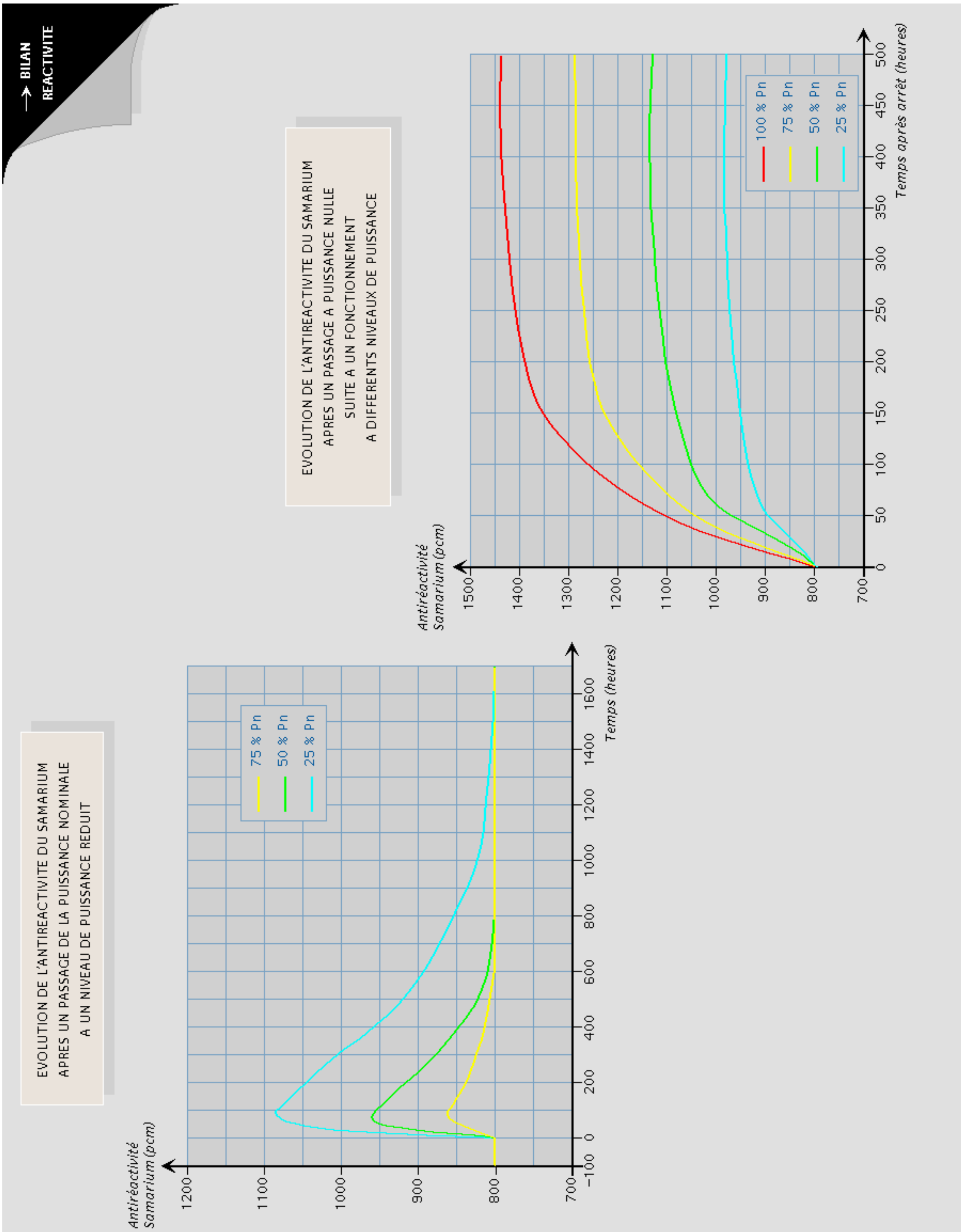
1.3.8.4 Educational images and diagrams – Power effects



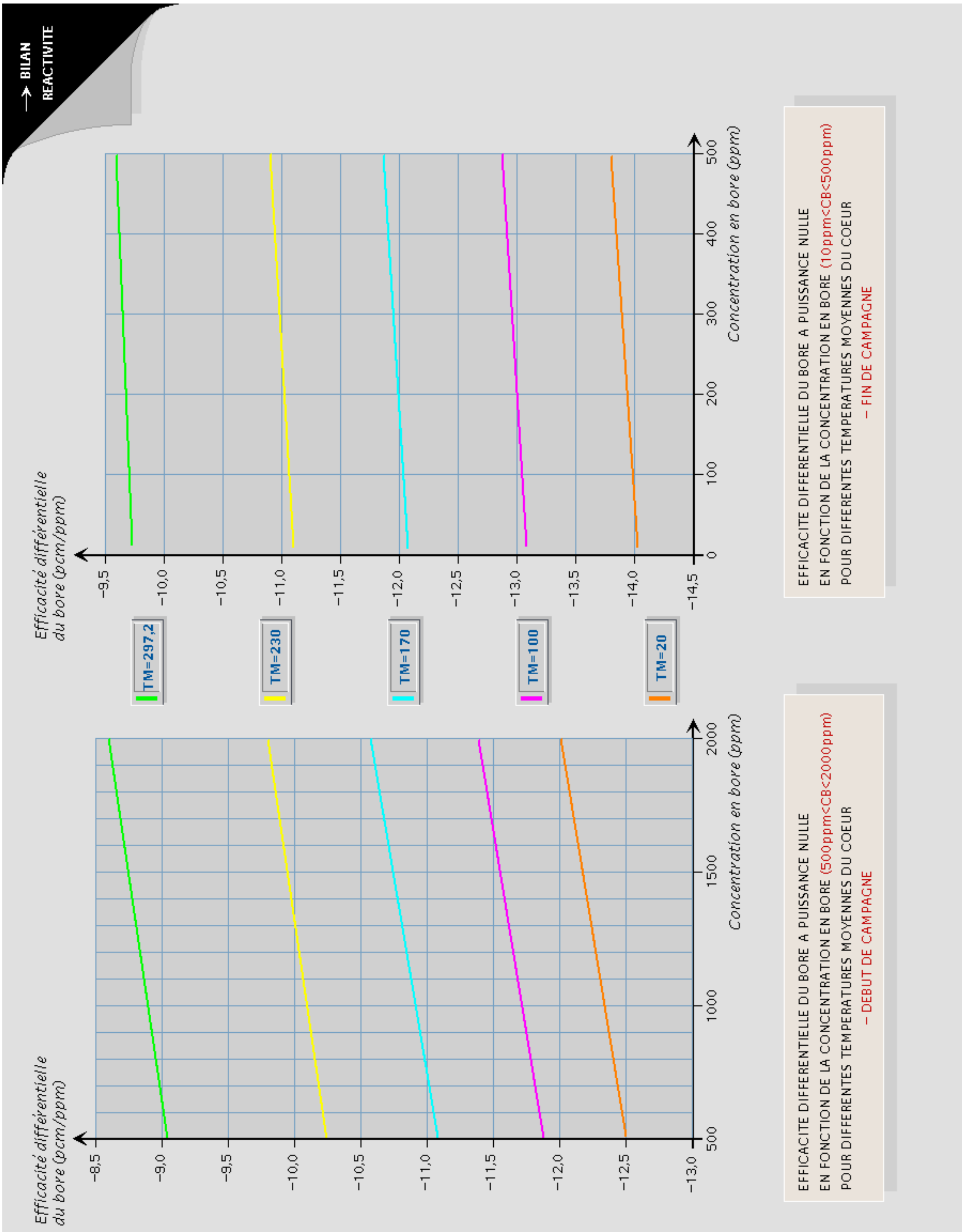
1.3.8.5 Educational images and diagrams – Xenon



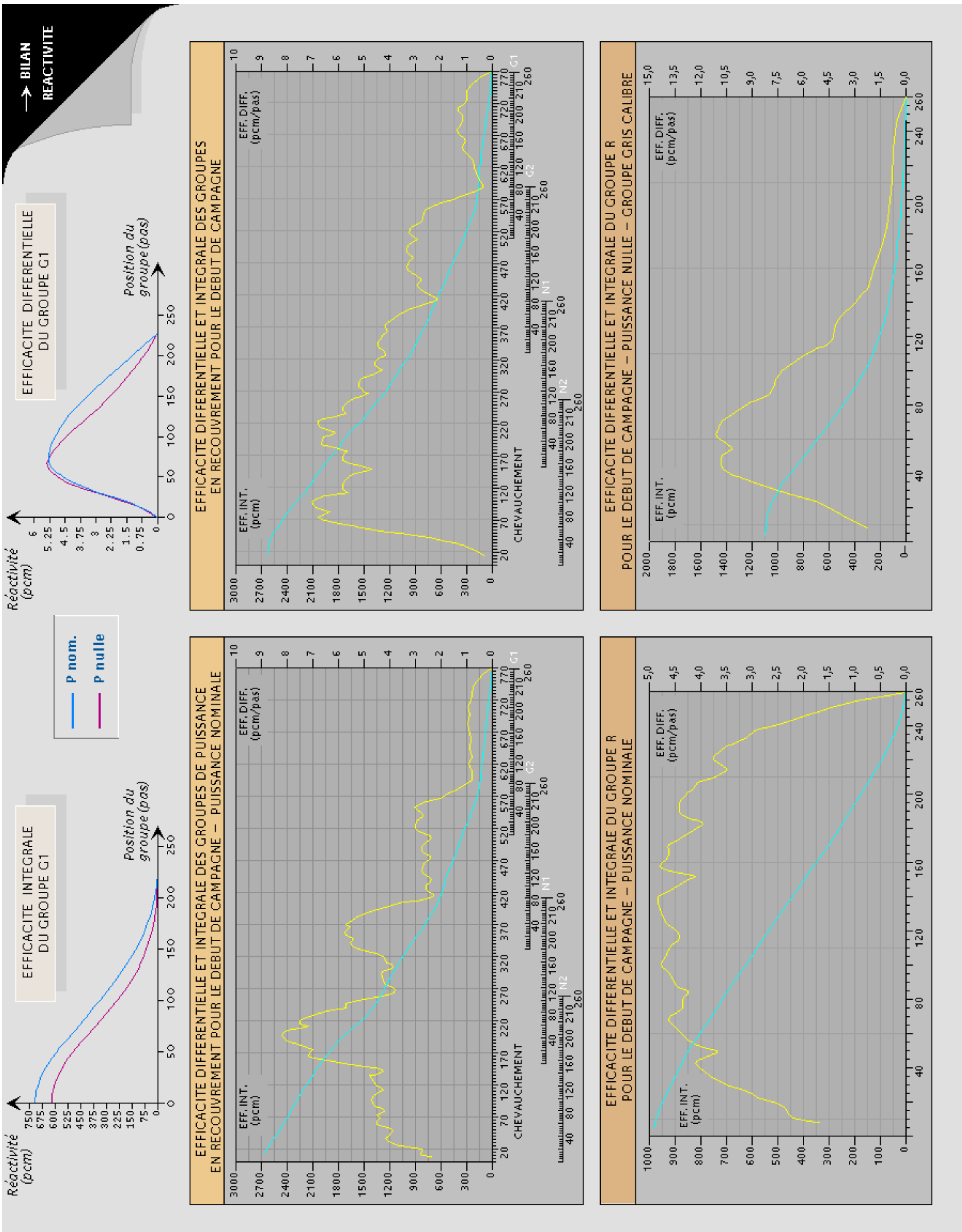
1.3.8.6 Educational images and diagrams – Samarium



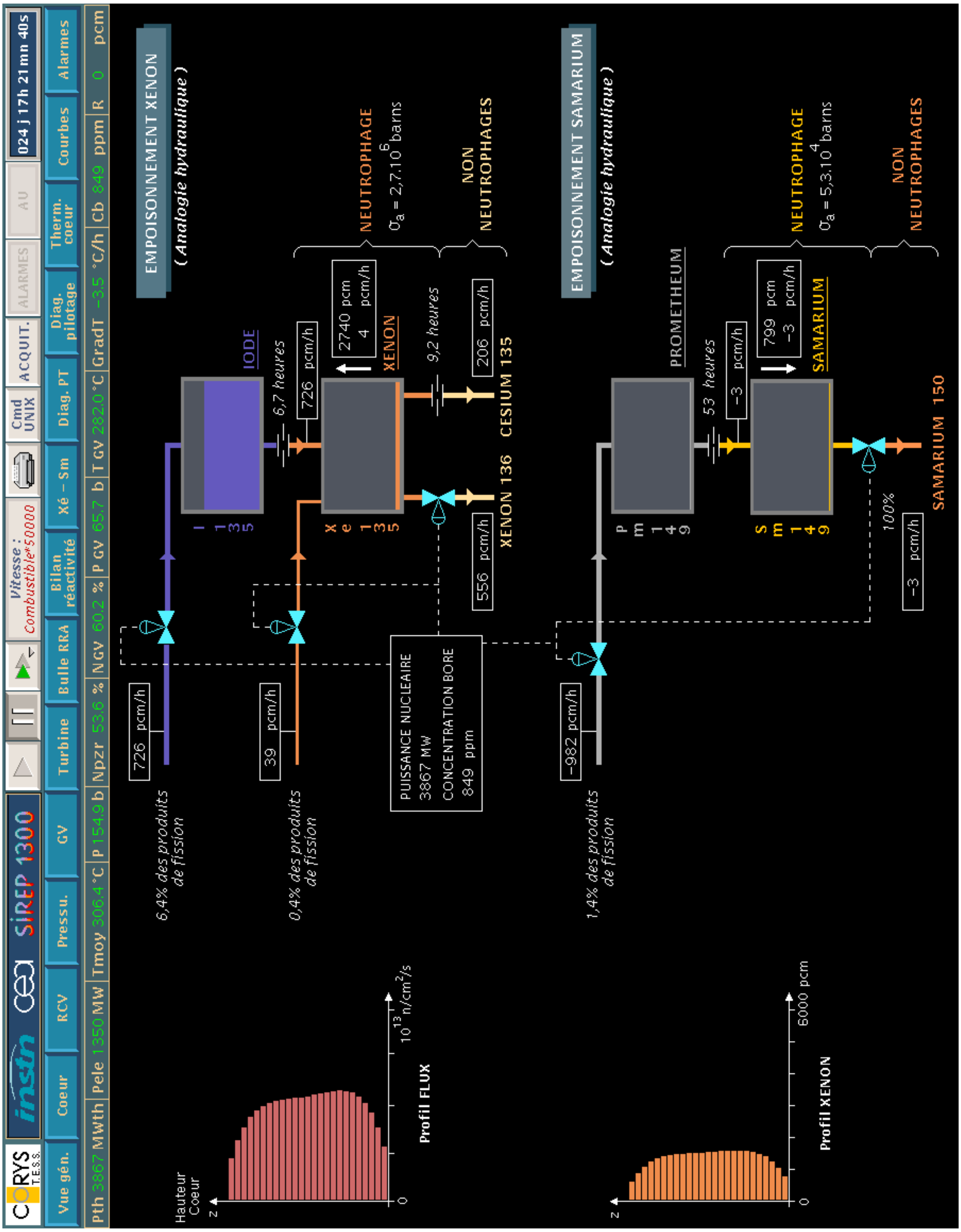
1.3.8.7 Educational images and diagrams – Boron



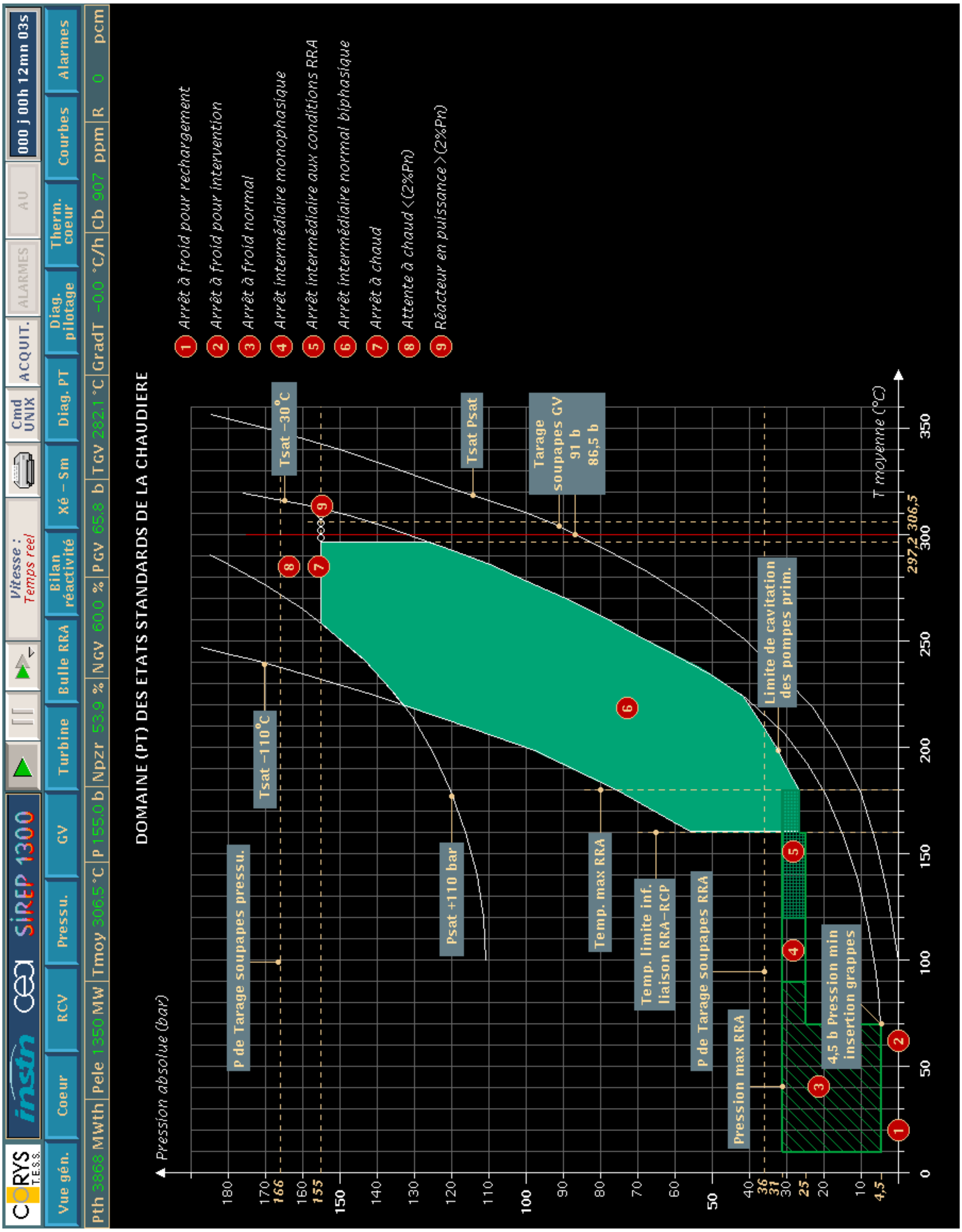
1.3.8.8 Educational images and diagrams – Control rods



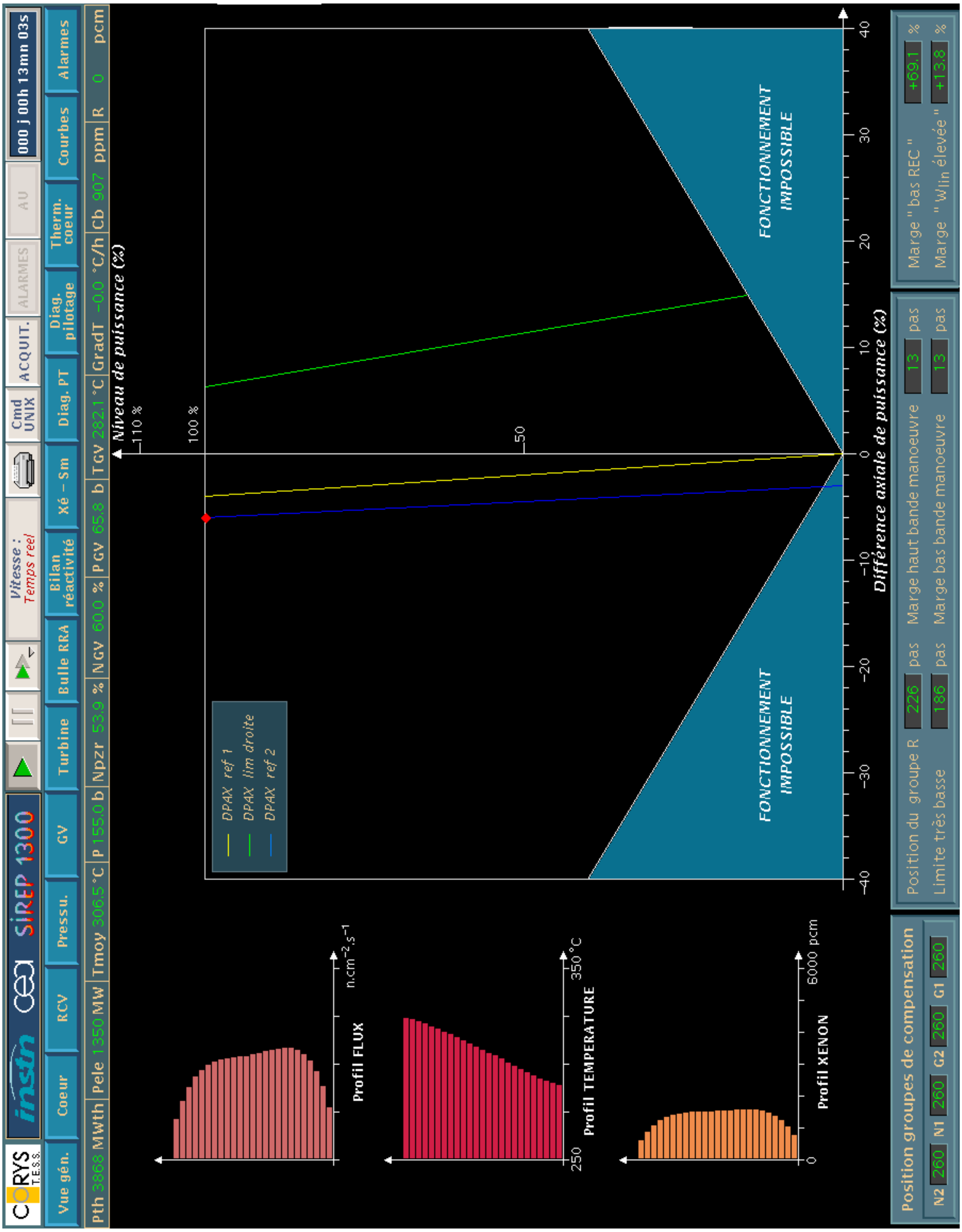
1.3.9 Hydraulic analogous Xenon and Samarium evolution representation



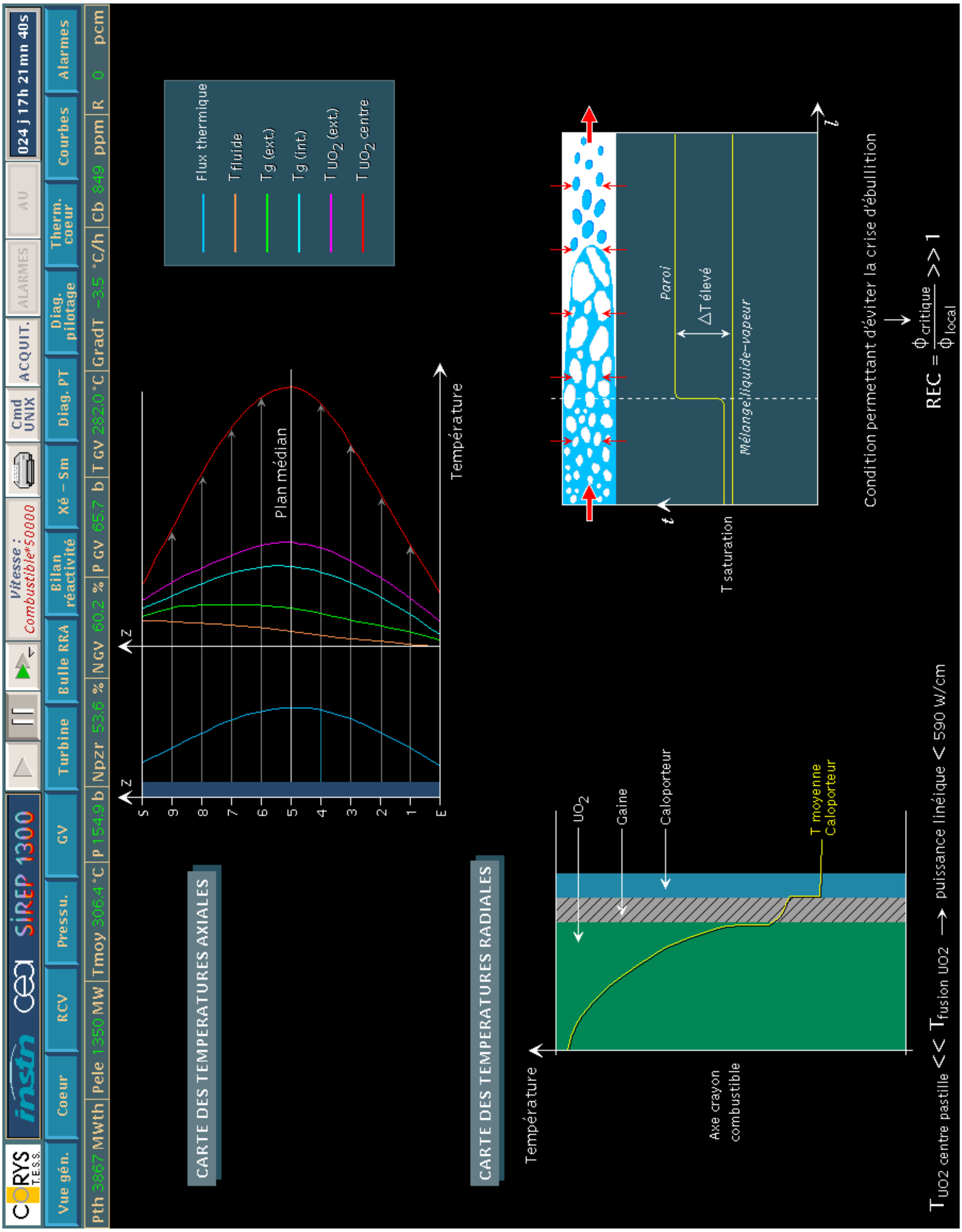
1.3.10 Pressure-Temperature representation



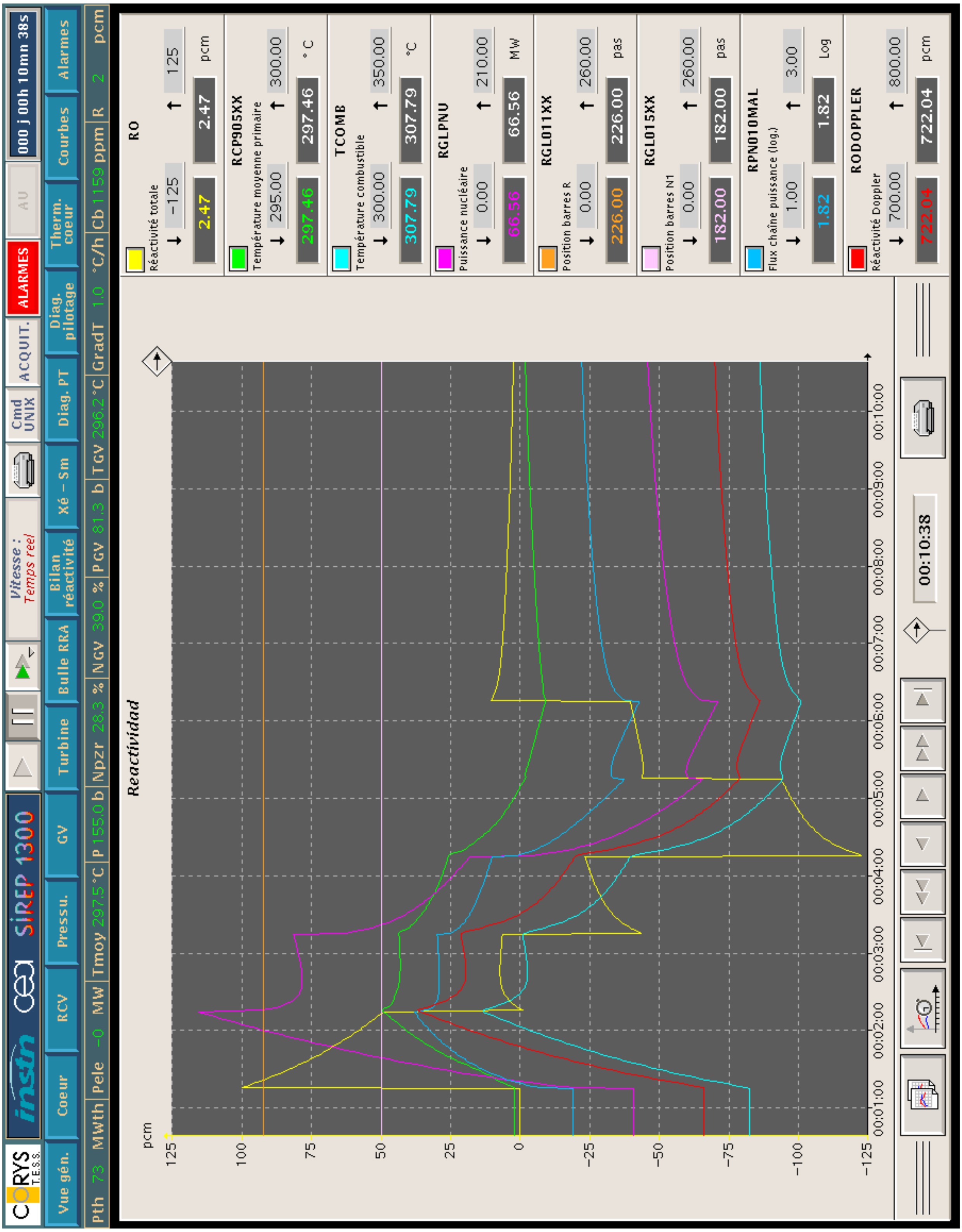
1.3.11 Control diagram representation



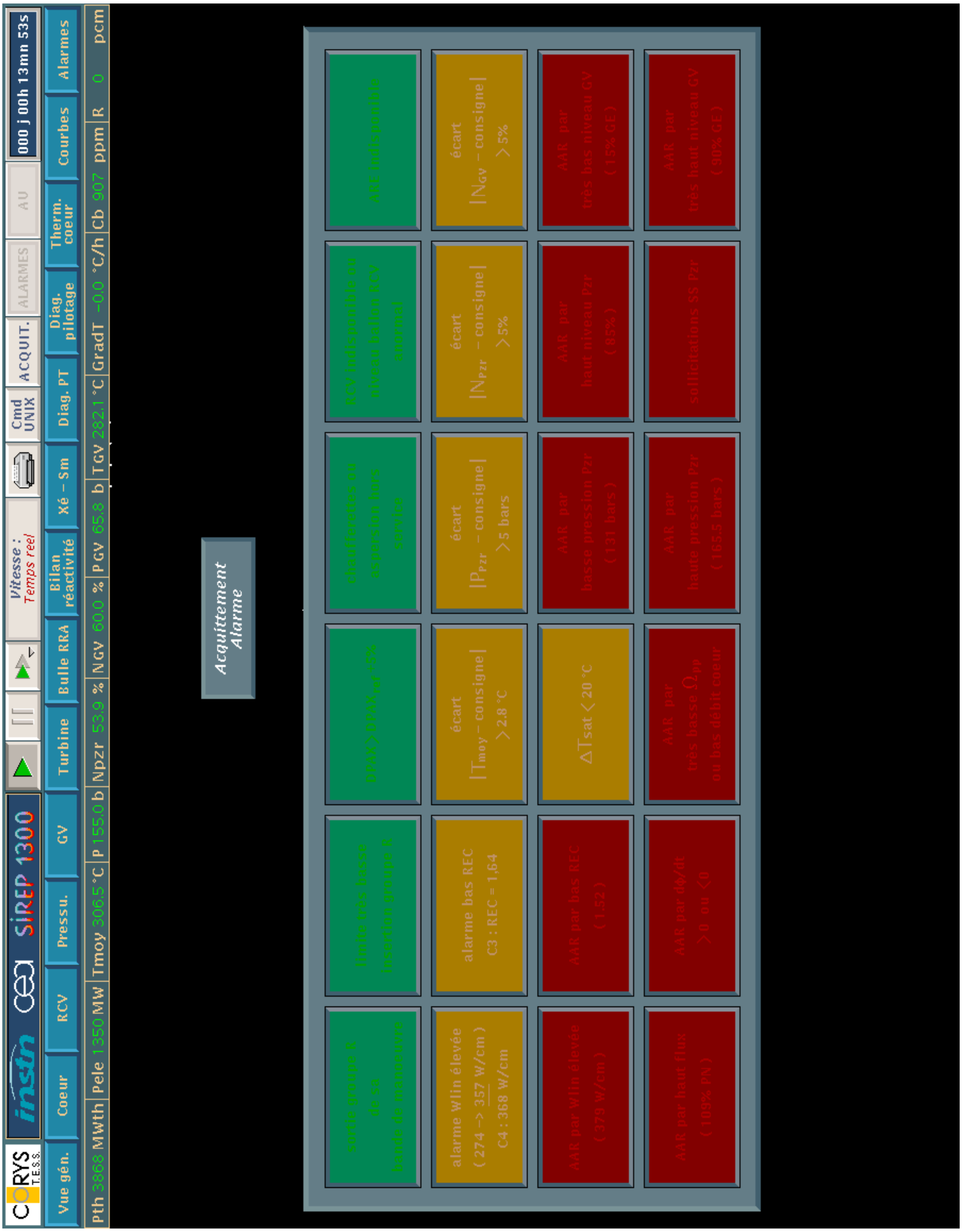
1.3.12 Thermal dynamics picture diagrams



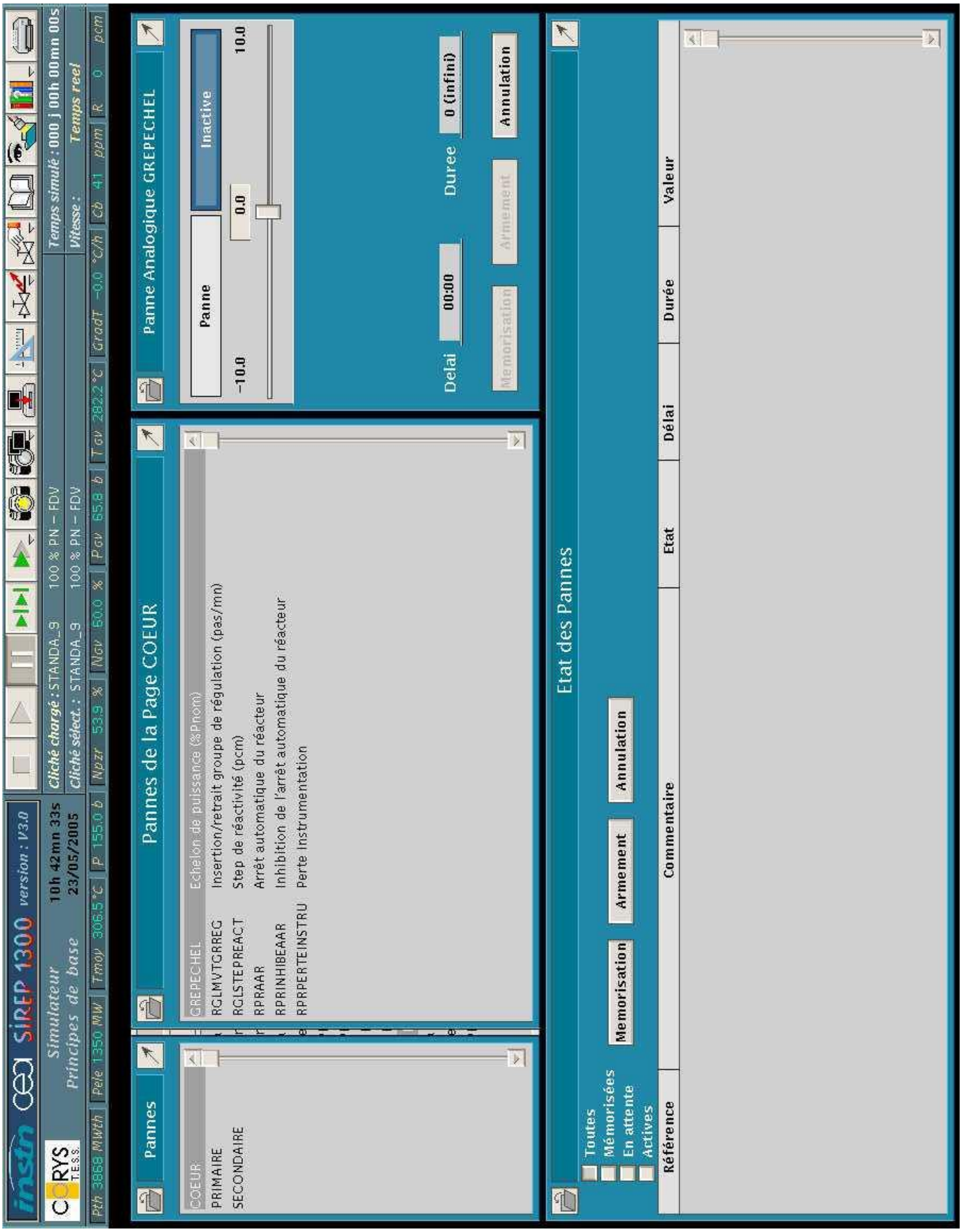
1.3.13 Curves screen



1.3.14 Alarms board



1.3.15 Instructor screen



1.4 French REP 1300 series description and main characteristics

A brief introduction in a preliminary classification has been made so that you can locate this plant inside the nuclear plants french groups. It is also made for showing the differences and peculiarities more significant of this groups.

France has a large experience in nuclear technology. Nearly 80% of its electric production has a nuclear origin. It has a total number of 58 pressurized water reactors as a result of its long-term energetic strategy. This politics pretended to achieve more oil independence with low cost and with adequate safety and radioprotection criteria. In France, there is usually a plant classification for its produced net electrical power. The reactors groups that are distinguished are: *900 MWe*, *1300 MWe* and last generation *1450 MWe*. Inside them, they are presently included the following reactors:

- 34 units *REP* of *900 MWe*, including *Fessemheim* and *Bugey*. The first of them was the first to enter in operation in 1977. All of them are controlled by EDF (*Électricité de France*).
- 20 units *REP* of *1300 MWe*, all of them are controlled by *EDF*.
- 4 reactors last generation *REP* of *1450 MWe* (The last unit is operating from 1998 in Civaux) controlled by *EDF*.
- The *FBR* (*Fast Breeder Reactor*) *Phénix* reactor that belongs to *CEA* (*Commissariat à l'Energie Atomique*) operating from 1974 with experimental purposes.

The french *REP* are globally similar. However, new technologies have been progressively introduced so the conception and operation are improved.

In a more detailed classification framework, the french nuclear installed power is classified in NPP according to standard series. A *palier* (group) is related with a similar reactor group. In the following classification, there are shown groups *CP0*, *CP1*, *CP2*, *P4*, *P'4* and *N4* characteristics. (CP: *Contrat Programme*).

The french nuclear program on February 1974 anticipated 2 launch programs of PWR (*CP1*) and of BWR (*CP2*). This last option was quickly abandoned for continuing in pressurized water technology and the 6 units, which preceded this program, were named, a posteriori, *CP0*.

CP0: Contain the 6 units of *Fessenheim 1* and *2* and *Bugey 2-5* (all approved before 1974).

CP1: ('Contrat Programme nº1'). It is related with *Tricastin 1-4*, *Gravelines 1-6*, *Dampierre 1-4* and *Blayais 1-4* units (launched in February 1974). Operation start between 1980 and 1985.

CP2: ('Contrat Programme nº2'). Comprise the *Chinon B 1-4*, *St Laurent B 1-2* and *Cruas 1-4* units (launched in February 1974). Operation start between 1981 and 1987.

The *CP1* and *CP2* groups are habitually named by ***CPY***. These plants have its main differences from their predecessors *Fessenheim* and *Bugey* in the buildings conception, the presence of an intermediate component cooling system that allows the containment building sprinkling and also allows a more flexible operation.

P4: 4 loops *REP* unit series (*1300 MWe*) referenced to *Paluel 1-4*, *Flamanville 1-2* and *St Alban 1-2* units. Operation beginning between 1984 and 1986. Some important changes were introduced respect *CPY* group. Among them, circuits and reactor protection systems can be standed out. Obviously, this power increase is translated to a 4 steam generators primary circuit, as it is needed a cooling capacity higher than *900 MWe* groups (3 cooling loops).

P'4: 4 loops *REP* unit series (*1300 MWe*). In this group we found *Cattenom 1-4*, *Belleville 1-2*, *Nogent 1-2*, *Penly 1-2* and *Golfech 1-2*. Operation beginning between 1986 and 1993. They are nearly equal to *P4* group, with the exception of some economic measurements. They have also inserted changes in the fuel building and some circuits and associated systems.

N4: 4 loops *REP* unit series (N as '*nouvelles*') (*1450 MWe*). It includes the *Chooz 1-2* and *Civaux 1-2* units. Operation beginning between 1996 and 1999. They where launched after the *900 MWe* (*CP*) and *1300 MWe (P4-P'4*) programs. This group is distinguished by the new steam generators (littler) and the primary pumps, as for some operation computerized aspects.

Particularly, *SIREP 1300* has its reference in some simulation variables phenomological correlation with **unit 1 from *Belleville NPP*** that forms part of *P'4* group.

The similarities between the *1300 MWe* plants design, more normalized than *900 MWe* reactors, have to be shown as a particularity of this reactor type.

Some units have developed from short operation fuel cycles (12 months) to large cycles (18 months), following the French main politics, as well in series *REP 1300.* The french obligatory load following together with the structural control rod damage limit the above mentioned transition to a few units. The 12 months cycles operation is kept in units in charge of fitting the global electrical production to the electric demand.

However, the base loaded nuclear power plants (without obligatory load following) are easier to operate in large cycle, in Spain. Every Spanish NPP works with the large cycle with the exception of the José Cabrera NPP.

The tables that have the units *CPY*, *P4-P'4* and *N4* main characteristics are included in sections 1.4.1, 1.4.2 and 1.4.3. Furthermore, a core graphic representation of units *P4-P'4* (*SIREP*) is included in section 1.4.4. The information shown here, not necessary shown in the simulator, can be useful and helpful for the location and understanding of the real scenario.

1.4.1 Primary circuit conceptual characteristics

	CP0		CP1-CP2	P4-P'4	N4
	Fessenheim	**Bugey**			
Nominal thermal power (MW)	2660	2785	2785	3817	4000
Maximum thermal power (MW)	2774	2905	2905	4117	4270
Configuration FA (Fuel Assembly)	17 x 17	17 x 17	17 x 17	17 x 17	17 x 17
Active high EC (m)	3,660	3,660	3,660	4,267	4,267
Number FA in the core	157	157	157	193	205
Fuel	UO_2 enrich	UO_2 enrich	UO_2 enrich	UO_2 enrich	UO_2 enrich
Nº control rods:					
Long black	48	48	41	53	65
Short black	5	5	-	-	-
Large grey	-	-	12	12	8
Number of loops	3	3	3	4	4
Primary calculation pressure (bar)	172,37	172,37	172,37	172,37	172,37
Cooling or Heating velocity:					
Normal (ºC/h)	<28	<28	<28	<28	<28
Maximum (ºC/h)	56	56	56	56	56

1.4.2 Thermal-hydraulics and neutronics characteristics

1.4.2.1 Primary thermal-hydraulics characteristics

	CP0		CP1-CP2	P4-P'4	N4
	Fessenheim	Bugey			
Primary nominal pressure (bar)	155	155	155	155	155
Nominal temperatures (ºC):					
Outlet vessel / Inlet SG	321,6	323,2	323,2	328,6	329
Outlet SG / Primary pump Inlet	284	286,9	285,8	292,8	292
Primary pump Outlet / Vessel Inlet	284,2	286	286	292,9	292,2
Core inlet	284,4	286,2	286,2	292,7	292,2
Core Outlet	323,1	326,6	326,6	332,4	329,6
Core dimension	303,8	306,4	306,4	312,6	310,9
Pressurizer	345	345	345	345	345
Core mass flow rate (t/h)	43610	45530	45530	62200	66276
Average core velocity (m/s)	4,38	4,59	4,59	5,21	5,3
Average mass core velocity (g/s.cm^2)	314	328	328	364	368
Core cross section (m^2)	3,86	3,86	3,86	4,75	5,04
Average heat flow in the core (W/cm^2)	57	60	60	57,1	60,1
Max. heat flow in the core (W/cm^2)	130	129	128	142,8	143
Average linear power (W/cm)	170,4	178,3	178,3	170,4	179,0
Max. linear power APRP (W/cm)	387	382	419	451	426
Limit APRP	2,27	2,14	2,35	2,65	2,38
Enthalpy rise factor F_H	1,55	1,55	1,55	1,55	1,55
Pellet centre temperature in nominal operation (ºC)	1830	1800	1830	1900	1900
DNBR minimum in nominal operation	1,86	1,78	1,78	1,81	-

In the simulator terminology, the nucleate boiling ratio is named REC (*Rapport d'Echauffement Critique*).

1.4.2.2 Water-Steam characteristics in SG secondary side

	CP0		CP1-CP2	P4-P'4	N4
	Fessenheim	Bugey			
Steam flow by loop at full power operation (t/h)	1718	1815	1817	1932	2164
Outlet steam pressure in the SG (bar):					
Full load	53,75	57,7	58	71,8	73,5
Null load	70,3	70,3	70,3	-	-
Steam temperature (ºC):					
Full load	268,5	268,5	273	287,5	289,1
Null load	286	286	286,1	-	-
Feedwater temperature (ºC)	216,8	219,5	219,5	229,5	229,5

1.4.2.3 Core neutron characteristics

	CP0		CP1-CP2	P4-P'4	N4
	Fessenheim	Bugey			
Enrichment (% U235)	3,25	3,25	3,25	3,10	3,40
Burn up (MWd/T):					
Off-load average	31500	31500	33000	33000	39400
Fuel rod maximum	42000	42000	45000	45000	47000
Fuel pellet maximum	45000	45000	50000	50000	50000
Boron concentration (without rods) (ppm)	1205	1205	1089	1081	1200
Long rods efficiency (pcm)	9000	9000	9000	9000	9000

1.4.3 Dimensions, components and materials characteristics

1.4.3.1 Vessel and internal elements

	CP0		CP1-CP2	P4-P'4	N4
	Fessenheim	Bugey			
Total height, with closure head (mm)	13173	13173	13173	13591	12602
Internal diameter with coating (mm)	3987,8	3987,8	3987,8	4394	4486
Mass (t):					
Vessel body	263	263	263	226,75	342,7
Vessel head	54	54	54	80	84,8
Nozzle number	2 x 3	2 x 3	2 x 3	2 x 4	2 x 4
Diameter (mm):					
Inlet nozzles	698	698	698	698	698
Outlet nozzles	736	736	736	736	736
Nº instrumentation channel	50	50	50	58	60

1.4.3.2 Fuel assembly

	CP0		CP1-CP2	P4-P'4	N4
	Fessenheim	Bugey			
Total height (mm)	4058	4058	4058	4795,4	4793,2
Side length (mm)	214	214	214	214	214
Fuel active length (mm)	3660	3660	3660	4267	4267
Total assembly weight (kg)	664	664	664	780	780
Number of fuel rods by assembly	264	264	264	264	264
Number of fuel rods in core	41448	41448	41448	50952	54120
Fuel rods:					
External diameter (mm)	9,50	9,50	9,50	9,50	9,50
Diameter gap pellet-clad (mm)	0,165	0,165	0,165	0,165	0,165
Cladding thickness (mm)	0,571	0,571	0,571	0,571	0,571
Cladding material	Zircaloy 4	Zircaloy 4	Zircaloy 4	Zircaloy 4	Zircaloy 4
Pressurizer gas	Helium	Helium	Helium	Helium	Helium
Pellet::					
Material	UO_2	UO_2	UO_2/UO_2PuO_2	UO_2	UO_2
Density, % theoretical density	95	95	95	95	95
Diameter (mm)	8,19	8,19	8,19	8,19	8,19
Length (mm)	13,5	13,5	13,5	13,5	13,5
Fuel assembly grid:					
nº fuel assembly grid	8	8	8	9	10
Material	Inconel 718	Inconel 718	Inconel 718 or zircaloy 4	Inconel 718 or zircaloy 4	Zircaloy 4
Guide tubes:					
Number by fuel assembly	24	24	24	24	24
External diameter (mm)	12,24	12,24	12,24	12,24	12,05
Thickness (mm)	0,42	0,42	0,42	0,42	0,42
Material	Zircaloy 4	Zircaloy 4	Zircaloy 4	Zircaloy 4	Zircaloy 4

1.4.3.3 Control rods

	CP0		CP1-CP2	P4-P'4	N4
	Fessenheim	**Bugey**			
Absorber rods number by cluster control rod assembly (spider):					
Black rod	24	24	24	24	24
Grey rod	-	-	8 or 12	8	8
Non absorber rods of grey rods (spider)	-	-	16 or 12	16	16
Absorbent material:					
Material	Ag-In-Cd	Ag-In-Cd	Ag-In-Cd	Ag-In-Cd	Ag-In-Cd
Composition (%)	80-15-5	80-15-5	80-15-5	80-15-5 and B_4C	80-15-5 and B_4C
Non absorber rods material of grey rods (spider)	Steel 304	Steel 304	Steel 304	Steel 304	Steel 304
Rod diameter (mm)	8,66	8,66	8,66	8,66	8,66
Absorbent length (mm)	3606,8	3606,8	3606,8	3870	3870
Drive mechanisms:					
Movement velocity (steps/min)	72	72	72	72	72
Movement velocity (mm/min)	1143	1143	1143	1143	1143
Step length (mm)	15,9	15,9	15,875	15,875	15,875
Total path (mm)	3618	3618	3618	4300	4200
Number of control rod drive mechanisms	48	48	53	65	73

1.4.3.4 Burnable poison rods and neutron sources

	CP0		CP1-CP2	P4-P'4	N4
	Fessenheim	**Bugey**			
Burnable poisons:					
Number of poison rods per control rod cluster assembly (clusters 1/2/3)	12/16/20	12/16/20	8/12/16	8/12/16	8/12/16
Burnable poison material	Boron silicate	Boron silicate	Boron silicate	Boron silicate	Boron silicate
Poison density (g/cm^3)	2,22	2,22	2,22	2,22	2,22
Absorbent length (mm)	3606,8	3606,8	3606,8	4340	4340
Neutron source:					
Primary source	Californium	Californium	Californium	Californium	Californium
Secondary source	Sb-Be	Sb-Be	Sb-Be	Sb-Be	Sb-Be

1.4.3.5 Primary pumps

	CP0		CP1-CP2	P4-P'4	N4
	Fessenheim	**Bugey**			
Rotation velocity (rpm)	1485	1485	1485	1485	1485
Nominal flow (m^3/h)	20100	22250	21250	22890	24500
Manometric height (m STP)	84,5	90,7	90,7	99	106
Nominal power absorbed by the motor (kW)	4650	5300	5300	6500	7100

1.4.3.6 Pressurizer

	CP0		CP1-CP2	P4-P'4	N4
	Fessenheim	Bugey			
Total volume (m^3)	39,6	39,865	39,865	59,3	59,5
Design pressure (bar)	172,37	172,37	172,37	172,37	172,37
Design temperature (ºC)	360	360	360	360	360
External diameter (mm)	2350	2350	2350	2800	2800
Height (mm)	12800	12800	12800	13526	13526
Heaters capacity (kW)	1400	1440	1440	2160	2160
Water volume (m^3)	37,196	37,196	37,196	55,0	55,0
Number of sprinkling valves:					
adjustable	2	2	2	2	2
continuous	2	2	2	2	2
Continuous sprinkling flow per valve (m^3/h)	0,230	0,230	0,230	0,2	0,2
Pressurizer relief:					
Type	Tandem SEBIM	Tandem SEBIM	Tandem SEBIM	Tandem SEBIM	Tandem monobloc SEBIM
Number	3	3	3	1	1
Opening pressure (bar)	162	162	162	166	166
Nominal unit capacity (t/h)	95	95	95	183	245
Safety valves:					
Type	Tandem SEBIM	Tandem SEBIM	Tandem SEBIM	Tandem SEBIM	Tandem monobloc SEBIM
Number	3	3	3	2	2
Setpoint pressure (bar)	171,5	172,5	171,5	172	172
Unit capacity (t/h)	169,15	189	172,5	183	245
Pressurizer relief tank:					
Total volume (m^3)	37	37	37	60	60

1.4.3.7 Steam Generator

	CP0		CP1-CP2	P4-P'4	N4
	Fessenheim	Bugey			
Type	51A	51A	51M or 51B	68/19	73/19E
Total height (m)	20,648	20,648	20,648	22,308	21,9
Maximum / minimum diameter (mm)	4468/3434	4468/3434	4468/3434	5040/3794	4756/3700
U-tube material	Inconel 600	Inconel 600	Inconel 600	Inconel 600	Inconel 690
Number of U-tubes	3388	3388	3361 (51M) 3330 (51 B)	5342	5600
Tubes diameter (mm)	22,22	22,22	22,22	19,05	19,05
Tubes thickness (mm)	1,27	1,27	1,27	1,09	1,09
Heat transfer surface (m^2)	4785	4785	4757 (51M) 4700 (51B)	6936	7300

1.4.4 Control and safety banks position in P4-P'4 series

The plants *P4-P'4* (case *SIREP*) diagram in the neutronic modelization shows the location of the different rod banks inside the core in figure 1.1. It is used the same simulator terminology.

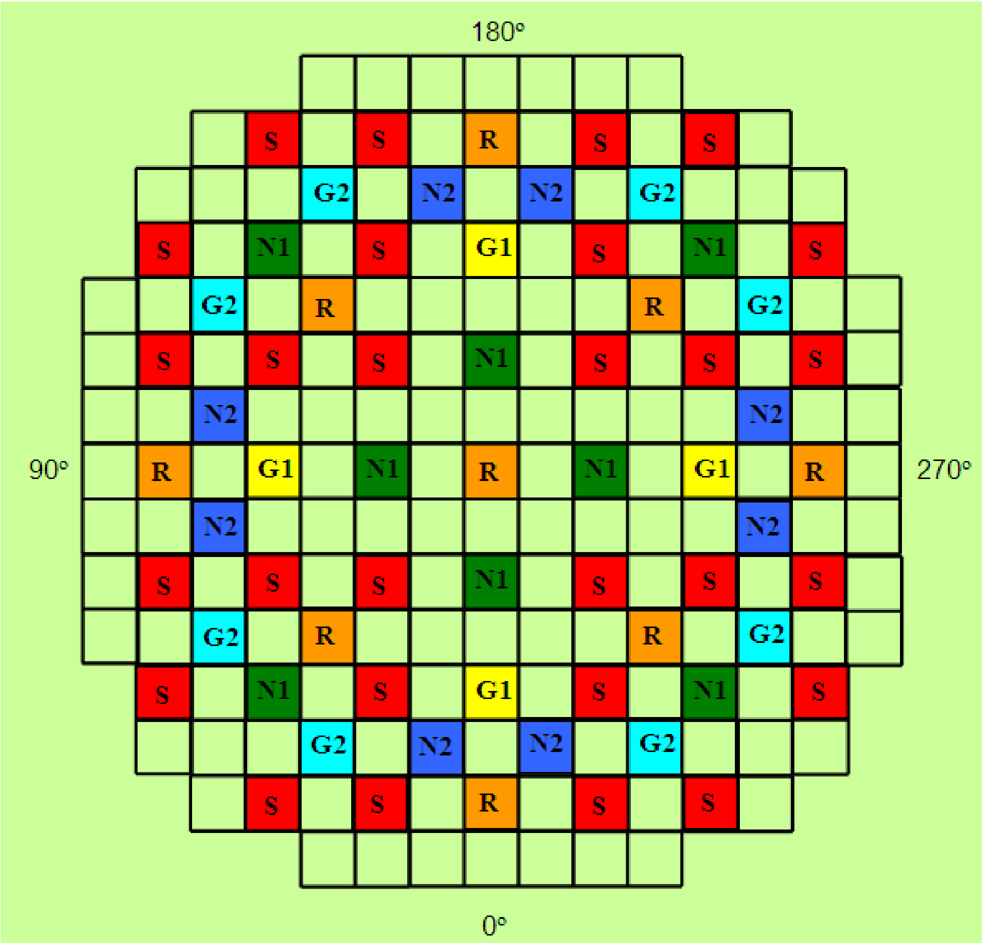


Figure 1.1.- *Safety and control rod banks location: N1 (8 black rods); N2 (8 black rods); G1 (4 grey rods); G2 (8 grey rods); R (9 black rods); S (28 shutdown rods). Case P4-P'4.*

2 Description of the simulator model

2.1 Modelization and core physics

2.1.1 Neutronic model. Simulation description and hypothesis

The model tries to simulate the nuclear power plant *REP 1300* type core and it also tries to achieve the obtaining and reproduction capacity:

- Neutron aspects (criticality, reactivity jumps, …),
- Reactivity balances.

The model uses some variables correlations (*Doppler*, efficiencies, etc…), for that reason it is named semiempiricist neutron model. The reference curves used in the model were obtained from the unit 1 of the *Belleville* plant during its 4th operation cycle.

The core modelization allows the following effects under the operation limits: neutron kinetics and dynamics, temperature effects, control rod effects, poison effects, fuel consumption effects, fuel-coolant group thermal-dynamics, main variables axial representation (neutron flux, temperatures, poison concentration), regulations, etc.

The developed core model is a point model of the neutron power, in the physics equation sense. Phenomenological correlations are used in the modelization of the axial unbalanced effects.

The specific kinetics reactor model considers the equations related to the reactivity of 6 delayed neutron groups.

The fuel evolution during a cycle in a neutronic sense is not taken in consideration.

2.1.2 Boron module. Simulation description and hypothesis

The *SIREP 1300* simulation boron module simulates the boric acid concentration balance in the primary circuit RCP, in the RCV circuit and in the REA circuit. That is to say, it allows the boron concentration calculation in the primary circuit and its attached circuits. For this reason, the different boron concentrations in the make up tanks, the primary charging system (make up system) and letdown system, the water and boron supply flows in the attached circuits (*RCV-REA*) are needed.

The boron is supposed not to become vapour. The pressurizer (continually make up by water sprinkling) is supposed to have an equal and homogeneous boric acid concentration in the whole primary circuit (cold and hot legs, core and pressurizer) with the exception of the attached circuits *RCV* and *REA*.

The instructor possible module actions are:

- Boron concentration blockage. This action allows to block the boron concentration evolution, keeping fixed the boric acid concentration in the primary circuit. This blockage can allow the primary boron concentration adjustment.

- The Boron concentration adjustment (in blockage case). This action allows to avoid the instantaneous evolution of the boric acid concentration in the core.

The total mass conservation for the boron element can be written in every node, as long as the modelization physics equations are related:

$$(MB)' = \sum Q_e B_e - \left(\sum Q_s\right) B \tag{2.1}$$

where B is the concentration, M is the mass and Q_e and Q_s are the input and output flows.

Related to the primary circuit:

- *Input flow*: Make up water (*RCV*), primary pumps joints make up and auxiliary sprinkling.

- *Output flow*: Dump flow and controlled seal leakage primary pumps.

- *Primary total mass*: Water mass of every primary element (pressurizer liquid phase included).

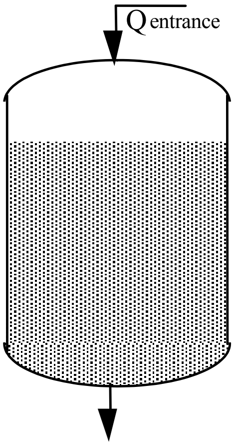


Figure 2.1- *Primary circuit simplification in a tank mode with input and output flow for the boron calculation.*

The mass balance equation resolution drives to the homogeneous boron concentration determination in the whole primary circuit.

The chemical and volume control system (*RCV*) is divided into three nodes. It is calculated:

- The boric acid concentration in the water-boron mixer that comes from the REA circuit.
- The boric acid concentration in the RCV tank.
- The boric acid concentration in the suction leg of RCV make-up pumps.

The different boron contribution reserves will have the following fixed concentration:

- Borated water storage tank (direct boration): *8000 ppm*.
- *PTR* pump suction (refuelling water storage tank): *2500 ppm*.

2.1.3 Control rods effect modelization

The control rods reactivity effect is calculated by a correlation of the insertion rod level and the power level.

Every rod bank movements (position and speed) are calculated in relation of an auto signal or an operator order. The control logic takes into account the grey mode operation, or what is the same, letting in long term the grey rods introduction.

The control rods antireactivity curves are shown in the figure of the 1.3.8.8 section as well as being the study object in experience 8.

2.1.4 Temperature fuel effect modelization

The temperature fuel effect (Doppler effect) is related, in one hand by the average fuel temperature and in the other by the core poisoning. Furthermore, the reactivity effect is represented by the temperature and the burn up correlation in the operation limits (*20-1000 ºC*).

The Doppler antireactivity is calculated by the following equation:

$$\rho_{Doppler} = k_1 + k_T T_C + \left(k_1 - k_2 \cdot burnup\right) \cdot \left(\exp\left(k_3 \cdot T_C\right) - 1\right) \tag{2.2}$$

where k_1, k_2 and k_3 constants are due to be determinated.

2.1.5 Moderators temperature effect and boric acid effect modelization

The moderators temperature effect is a physic phenomena directly dependent on the average moderator temperature and the boric acid concentration. In this sense, the correlation used in the simulator is related with the average primary water temperature and the boron concentration. The implemented correlation is valid in a wide temperature range (*50-350 ºC*) and concentrations range (*0-2000 ppm*).

2.1.6 Poisoning effects modelization

The two phenomena to be considered in a PWR reactor are joined to the Xenon and Samarium presence. These elements concentrations are obtained by the equations that represent the Iodine-Xenon and Promethium-Samarium chains. The reactivity effect is considerated proportional to the concentration.

Iodine concentration: The Iodine concentration is calculated from the high to the low core section. It is related to the neutron flux (direct appearance by fission) in the determinated zone and also related to the Iodine concentration in the disintegration preceding time as shown in the following equation:

$$I(t+\Delta t) = I(t) - \Delta t(\lambda_i I - k_{ri}\phi) \tag{2.3}$$

with:

- λ_i Iodine radioactive decay constant,
- k_{ri} Iodine dependence of flux coefficient, related to the burn up:

$$k_{ri} = k_{rx0} + burnup \cdot (k_{ri1} - k_{ri0}) \tag{2.4}$$

Xenon poisoning: The Xenon poisoning is calculated from the high to the low core cross section. It is related to the neutron flux in this zone and it is also related with the Iodine as it is shown in the following equation:

$$Xe(t+\Delta t) = Xe(t) + \Delta t \cdot (\lambda_I I + k_{rX}\phi - (\lambda_{Xe} + j_{rx}\phi) Xe(t)) \tag{2.5}$$

with:

- λ_i Iodine radioactive decay constant,
- λ_{Xe} Xenon radioactive decay constant,,
- k_{rX} Xenon dependence of flux coefficient, related to the burn up:

$$k_{rX} = k_{rx0} + burnup \cdot (k_{rX1} - k_{rX0}) \tag{2.6}$$

- j_{rX} Xenon dependence of flux coefficient, related to the burn up:

$$j_{rX} = j_{rx0} + burnup \cdot (j_{rX1} - j_{rX0}) \tag{2.7}$$

The same principle is used for the Promethium and Samarium balance establishment.

2.1.7 Fuel reactivity calculation

The fuel reactivity is only related to the burn up (*burnup*):

$$\rho_{comb} = \rho_{combinit} - k_{rho} \cdot burnup \tag{2.8}$$

The burnup variable is increased during time and related to the present power:

$$burnup(t+\Delta t) = burnup(t) + \Delta t \cdot k_{dbup} \cdot W_n / W_{nnom} \tag{2.9}$$

with:

- k_{dbup}: Fuel consumption velocity in nominal power,
- W_n: nuclear (or neutronic) power,
- W_{nnom}: Nominal nuclear power.

The neutronic calculated phenomenon connection as well as their relations are represented in the diagram below:

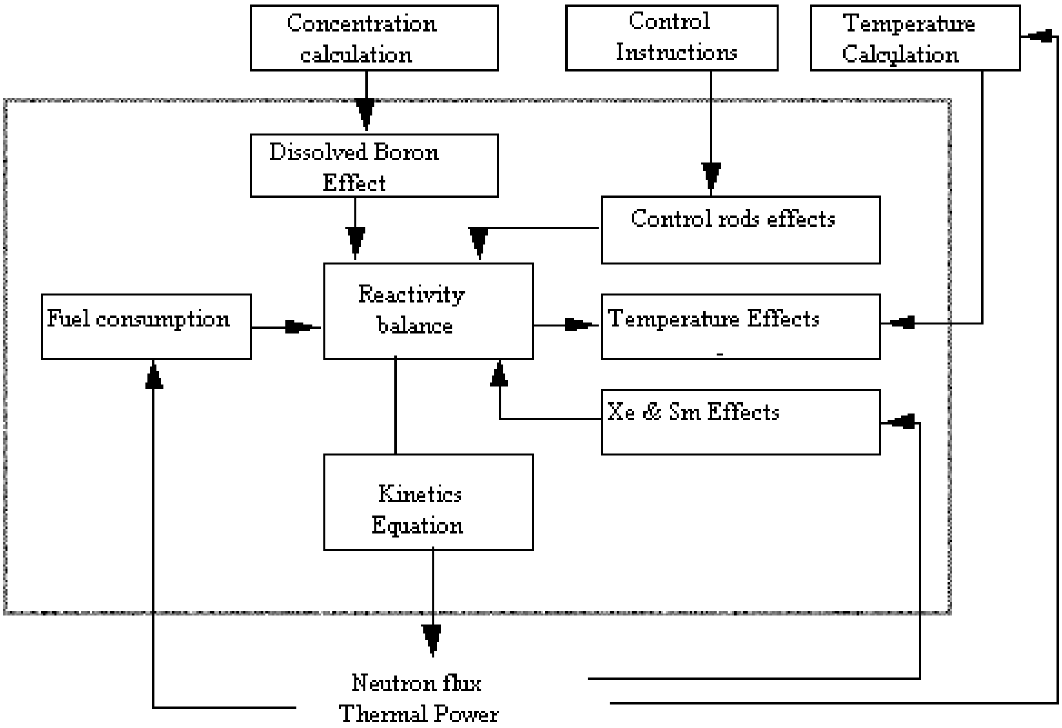


Figure 2.2- *Connection and neutronic calculated interactions diagram.*

2.1.8 Residual power calculation

The residual power (W_{res}) is calculated by three exponential equations:

$$W_{res} = W_{res1} + W_{res2} + W_{res3} \tag{2.10}$$

with:

$$W_{res1} = A_1 \exp\left(-t/\tau_1\right)$$
$$W_{res2} = A_2 \exp\left(-t/\tau_2\right)$$
$$W_{res3} = A_3 \exp\left(-t/\tau_3\right)$$

For every time instant, the residual power is calculated from the preceding time residual power value and the present one (W_n):

$$W_{res1}(t+\Delta t) = W_{res1}(t) + (A_1 \cdot W_n - W_{res1}(t)) \cdot \Delta t / \tau_1$$
$$W_{res2}(t+\Delta t) = W_{res2}(t) + (A_2 \cdot W_n - W_{res2}(t)) \cdot \Delta t / \tau_2$$
$$W_{res3}(t+\Delta t) = W_{res3}(t) + (A_3 \cdot W_n \cdot burnup - W_{res3}(t)) \cdot \Delta t / \tau_3 \qquad (2.11),(2.12),(2.13)$$

2.1.9 Core thermal calculations. Flux and temperatures axial profiles determinations

The thermal section has as objective the coolant and fuel temperatures evaluation. The calculation is made over an average channel. However the inlet and outlet water temperatures are calculated in another simulator module. The transferred power is calculated by the neutron section.

The average fuel temperature (T_C) is obtained by the law ruled by the equation of a thermal balance with a constant convection coefficient between the fuel and the moderator:

$$k_{tcomb} \frac{\Delta T_C}{\Delta t} = W_{th} - h_{conv} \left(T_C - T_{\mathrm{mod}} \right) \qquad (2.14)$$

where:

- k_{tcomb} = fuel thermal inertia (*J/ºC*),
- W_{th} = thermal power,
- h_{conv} = Fuel-moderator convection coefficient.

The neutron flux axial coefficient is determinated by a correlation between the total power and the power unbalance (*axial offset*). The developed core model is a specific integral point model of the nuclear power. Whereas the axial offset modelization is implemented using phenomenological correlations. The different variables that influence on the core axial power distribution are the following:

- Control rods steps,
- Poison distribution (fission products),
- Boron concentration,
- Power level by means of the temperature effects,
- Fuel burn up rate.

Once known the power axial profile, the coolant temperature distribution is obtained by integration, whereas the fuel temperature distribution results from the heat transfer.

Water temperature profile: The neutronic flux is proportional to the linear power. The temperature profile in water is obtained by the heat transfer on a determinated core height.

Fuel temperature profile: The fuel temperature comes from the thermal balance between coolant and fuel.

Generally, the simulators methodology summary for the flux calculation for a time step is presented in the following diagram.

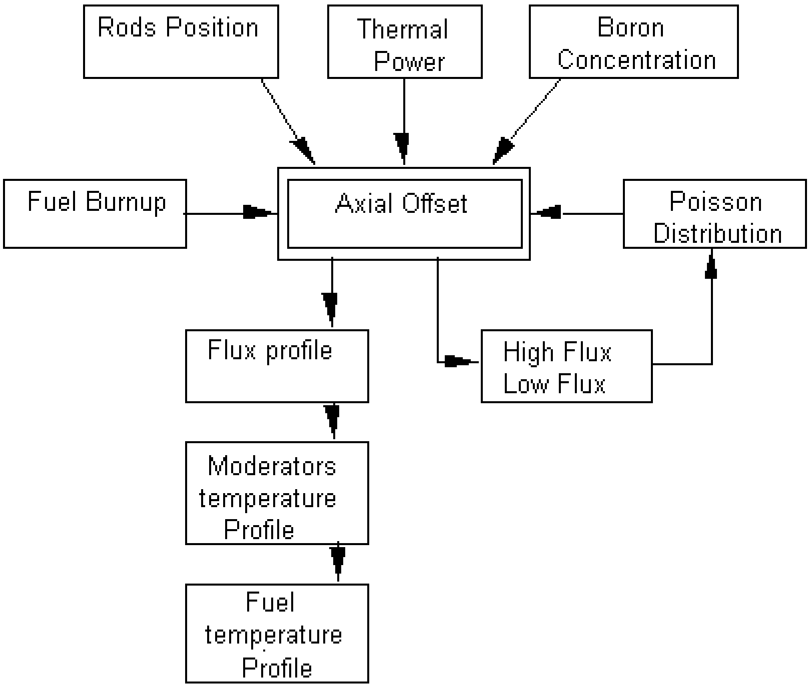


Figure 2.3- *Diagram of the unbalance determination and temperature and flux profile calculations.*

2.2 Pressurizer concept. Level and pressure primary circuit regulations

The pressurizer model is a two-phase type in thermal unbalances. The pressurizer modelization is carried out with an only volume. For this reason, the pressurizer thermal stratification is not represented. The modelization takes into account the pressurizer operation after the cold shutdown (one-phase state) until full power operation (two-phase).

The regulations that are developed in the pressurizer operation are the following ones:

- Level regulation by means of the charging flow of the chemical and volume control system.

- Primary pressure regulation by means of the letdown flow of the chemical and volume control system in one-phase operation.
- Primary pressure regulation by means of the sprinkling system and the heaters in two-phase operation.

There are also representations of the *SEBIM* valves relief opening automatisms: the first one opens when pressure exceeds *166 bar* while the second one (close safety) is opened. This last one closes when pressure diminishes of *139 bar* (only control in auto mode).

The operator has always the possibility of controlling in manual mode the relief devices. The pressurizer pressure command value can be automatically calculated in the following characteristic allowing a simulator behaviour and operation easier, especially during the coolant heating stage.

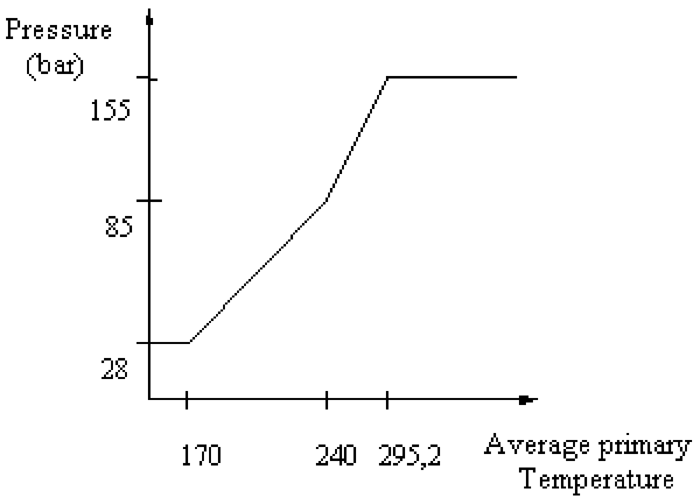


Figure 2.4- *Pressure command value evolution related with the average primary temperature.*

The pressurizer regulation junction is represented in the following diagram:

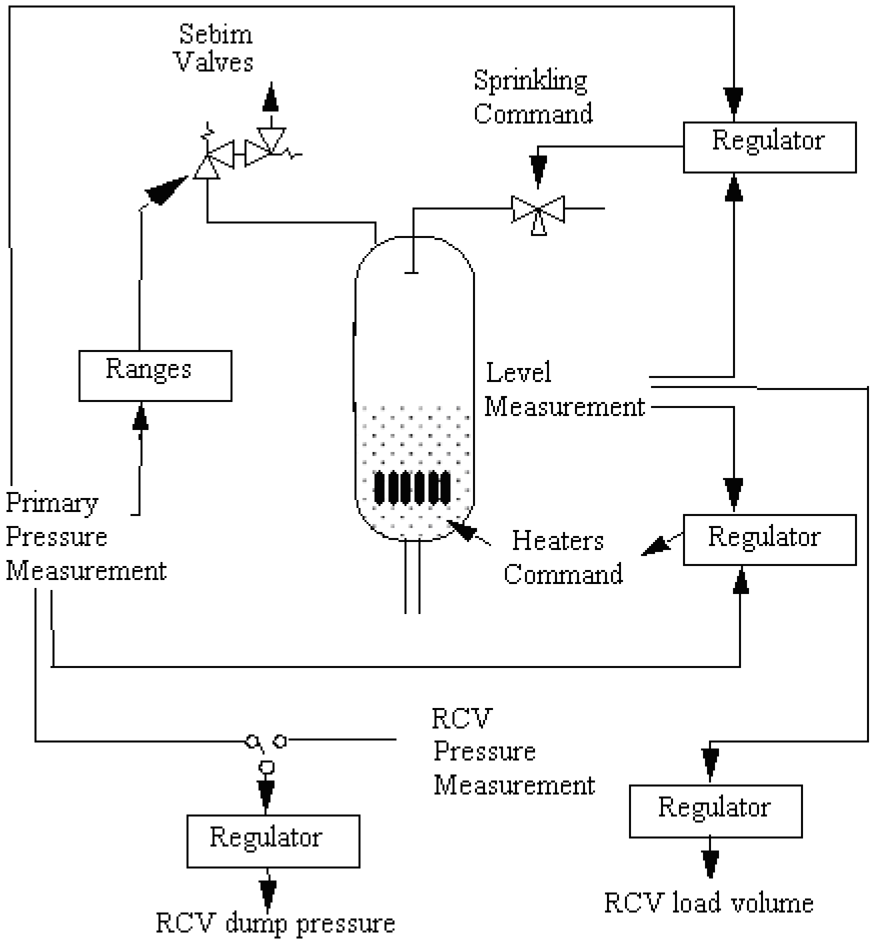


Figure 2.5- *Pressurizer regulations diagram.*

The pressurizer level command value is calculated in auto mode and in relation with the average primary temperature by the following way:

- If *T average < 295,2 ºC*, then the command level will be of *21,5 %*,
- If *T average > 307,2 ºC*, then the command level will be of *55,5 %,*
- Between them: command level linear evolution.

On the other hand, if the primary pumps are stopped, the continuous sprinkling is stopped.

2.3 Pumps and connection piping

The reactor and the pressurizer are connected together to the steam generator by means of pipes in which the primary pumps make the coolant recirculating.

The circuit total flow, imposed by the primary pumps, is calculated by the number of operating pumps and their mechanical inertia. The mechanical inertia of the start up and shutdown pumps is taken into account by the circulant mass equation solution. The dissipated power, the main start up heat source, is also modelized.

The primary thermal power takes into account the delays introduced by the canalizations transfer times and the circuits metallic mass inertia. The thermal losses are also modelized.

2.4 Secondary circuit

The secondary circuit is divided in steam generator and in classic plant sections, such as the turbine, steam dump system, alternator, condenser, make up turbopumps, etc.

2.4.1 Steam generator

The modelization allows to simulate, in a real way, the water and pressure levels variations during the different transitory times.

The steam generators command level depends on the electric power by the following way:

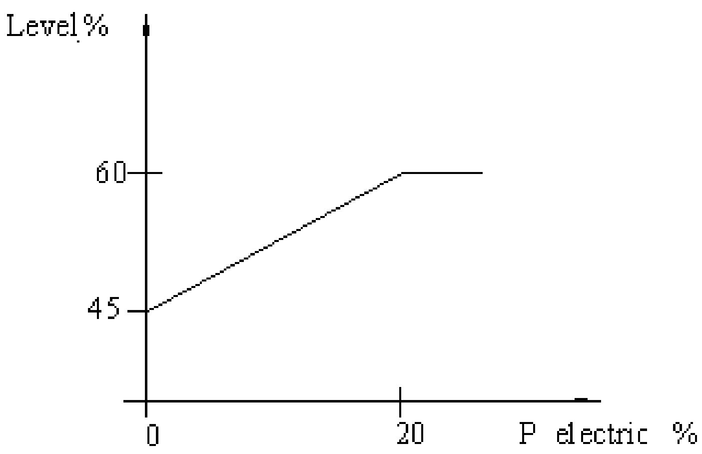


Figure 2.6- *Command level evolution curve of the steam generator related to the electric power.*

2.4.2 Steam circuit

The circuit takes into account: the atmospheric steam dump, the main feedwater, steam dump system (turbine bypass to the condenser) and the feedwater of the turbo pumps.

The turbines bypass circuits have as objective the steam dump either in the condenser (85% capacity in nominal power) or to the atmosphere (10% capacity in nominal power).

The steam dump to condenser is made of a valve group located in parallel and operated by an automatism controlled by the steam generators pressure. These valves are simulated by an only one that, with a total opening, evacuates the bypass system nominal flow. The operator has the possibility to operate the dump in pressure mode (auto) or in manual mode.

The atmospheric steam dump is controlled by a steam generators pressure regulation.

2.4.3 Turbine

The turbine is connected to the alternator. The rotation speed is calculated from the circulating mass equation. The hydraulic power and the linked performance are calculated from the steam thermal-dynamics characteristics. The increasing phases of speed, coupling, decoupling and “virador” shutdown are simulated.

2.4.4 Condenser

The condenser determinates the pressure that allows the turbine enthalpy lost calculation. The modelization allows to simulate the pressure variations in relation with the thermal load, the circulating water flow, the circulating water intake temperature and the leakage rate.

One available action in the instructor operation allows to simulate the vacuum lost in the condenser, simulating the following effects: condenser pressure increase, turbine performance diminish, etc.

The vacuum pumps and valves are not simulated.

2.4.5 Turbopumps and feedwater

In reference to the feed turbopumps, their speed is modelized by the load and the water flow necessary for their feed.

For the steam generators feedwater, there is the following distinction:

- *The main feedwater:* by means of the low flow valve (low feed flow) and the big flow valve (big feed load). The feedwater temperature is given by a correlation in relation with the secondary circuit load.

- *The emergency feedwater:* in which the enthalpy has a constant value.

The low feedwater flow valve belongs to the 20% of the modelized valve (flow opening).

The big feedwater flow valve belongs to the 80% of the modelized valve (flow opening).

The diagram below presents the relations between the modelized valve and the big and low flow valves opening.

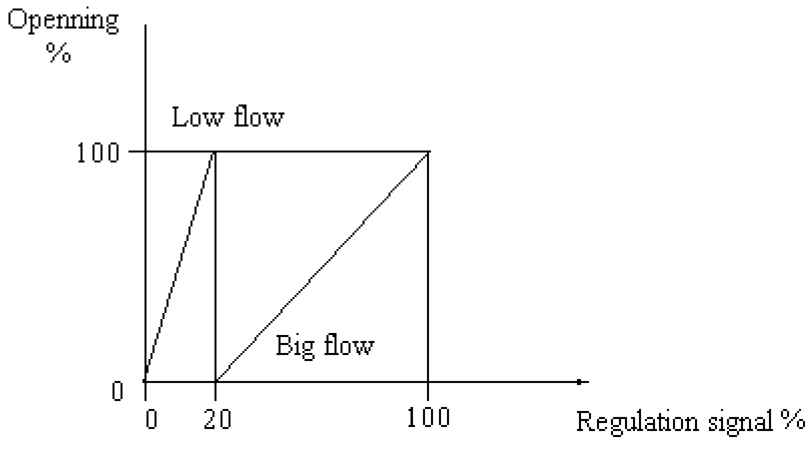


Figure 2.7- *Evolution curve that represents the low and big flow valve opening as a function of the regulation signal.*

2.4.6 Automatisms and group regulations

The different group regulations of the steam system and the feedwater system are represented in the following diagram:

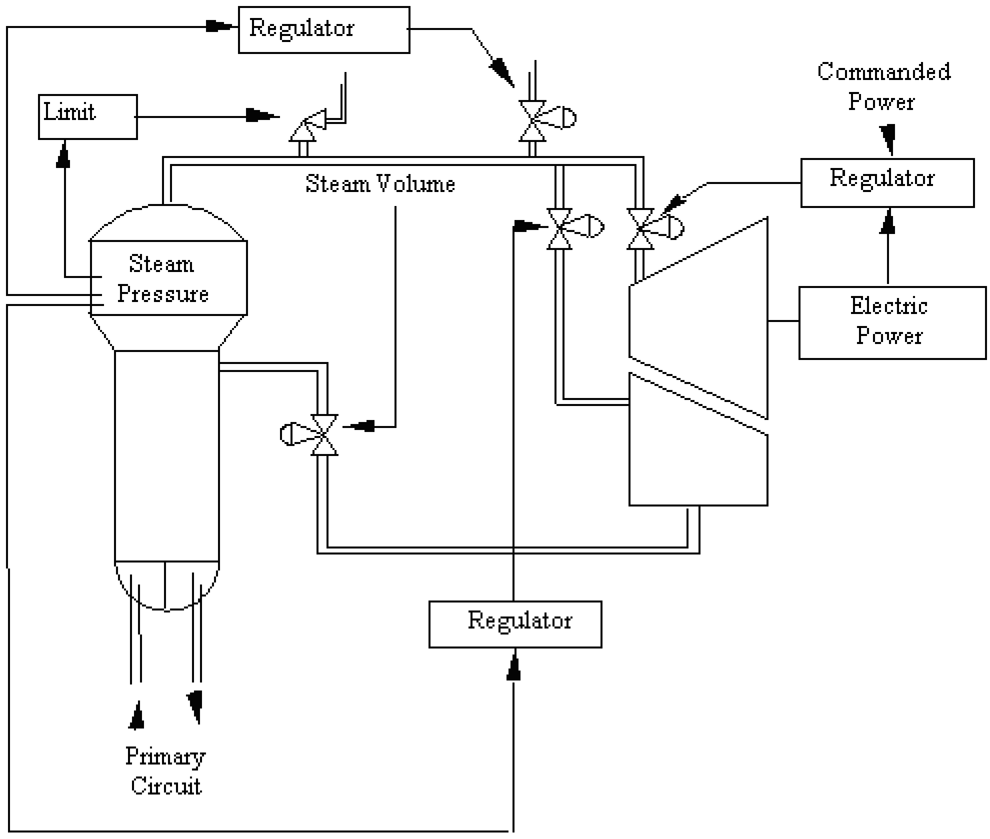


Figure 2.8- *Diagram of the main secondary group regulations.*

2.5 Chemical and volume control system

The RCV circuit functions are the primary water volume control by make up and letdown unbalance and the boric acid control. Moreover, it is the responsible of keeping some kind of mineral quality in water, corrosion products erase and primary pumps sealing.

2.5.1 Physic model

The circuit thermal phenomena are also represented: regenerated and non regenerated exchanger.

The safety injection is not simulated.

The boron concentration control is carried out by demineralized water or boron injection in the circuit. The valves are controlled by automatisms equal to the ones in the nuclear plants. These automatisms have the following possibilities:

- *Dilution*: An amount of predefined water injection in the load circuit from demineralized water tank.
- *Boration*: An amount of predefined boron injection in the load circuit from the boric acid bath tank.
- *Contribution "appoint auto borication"*: A water and boron mixture injection in auto mode in the control tank in case of a too low liquid level inside it (18% approx.)

2.5.2 Regulations

The regulations joints are shown in the following diagram of figure 2.9.

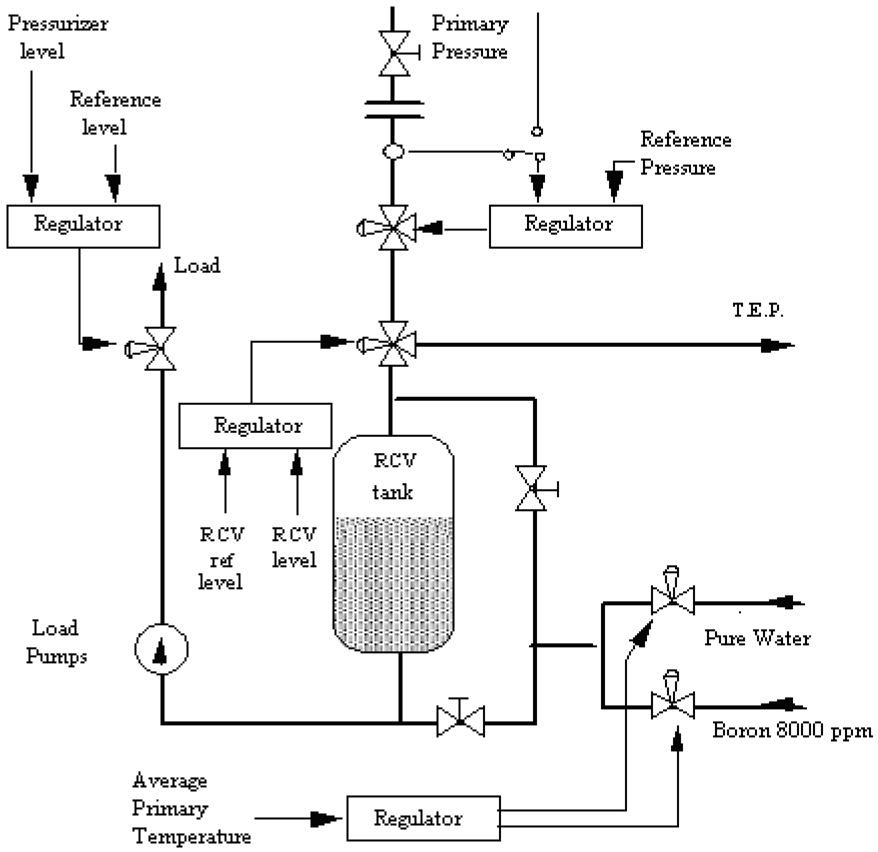


Figure 2.9- *Diagram of the chemical and volume control system regulation group.*

The three-way valve opening depends on the control tank level. If this level is lower than *49%,* the valve is open in the tank side and closed in the *TEP* (effluent treatment system) side. The opening in the *TEP* is carried out when the tank level increases the *49%*. The valve closes completely in the tank side in case the level increases the *54%,* and it will open *100%* in the *TEP* side.

The three-way valve opening diagram is represented in the figure 2.10.

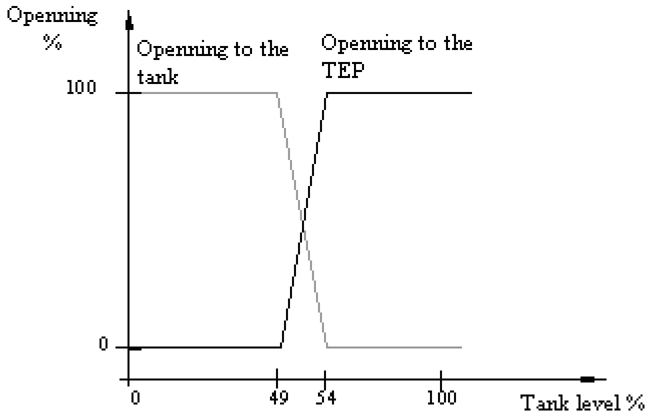


Figure 2.10- *Three-way valve opening curve in relation with the control tank level.*

The *PTR* (refuelling water storage tank) valve opening carries out when the volume control tank level is really low (*3%*).

2.6 Residual heat removal system

The residual heat removal system or the cooling system in shutdown (*RRA*) is assigned to cool the primary circuit in case of shutdown. In this moment, the thermal power must be removed since the core residual heat and the primary pumps are significant heat sources. It is mainly composed by a thermal exchanger connected to the primary circuit and cooled by the intermediate component cooling system (*RRI*).

2.6.1 Physic model

This circuit modelization is necessary to show the thermal-hydraulics phenomenon in the primary circuit during the cooling and heating phases. The *RRA* circuit is considered as a only heat exchanger and the possible pressure associated problems are not taken into account.

The total injected flow regulation in the primary circuit is adjusted by the opening of the bypass branch while the exchanger exit temperature is adjusted by the circulating flow inside it.

On one hand the model includes the flows calculation (as direct functions of valves opening) and on the other hand the temperature calculation in the different nodes.

2.6.2 Regulations

The bypass flow is regulated in relation to a command flow of the system output total flow, as in real facilities.

An additional regulation is introduced by controlling the exchanger flow in relation to a primary circuit temperature command. This fact allows the *RRA* system operation in an auto mode.

The two regulations are outlined in the following figure:

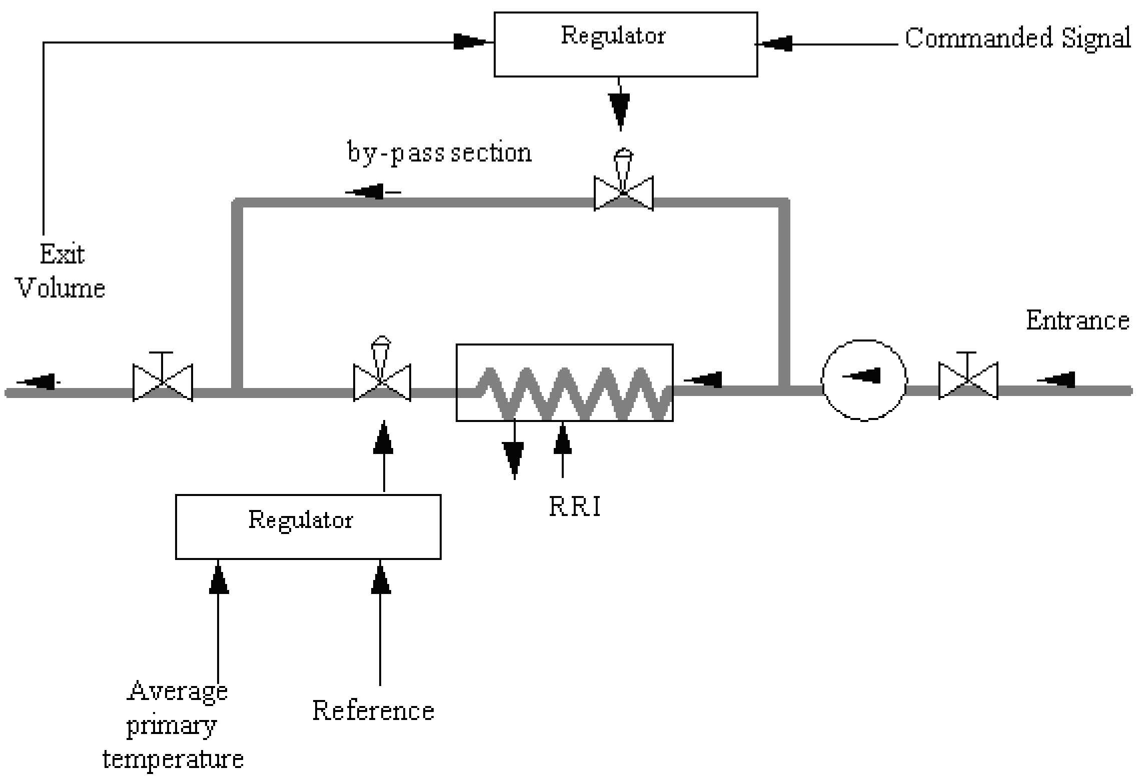


Figure 2.11- *Diagram of the exchanger and bypass flow regulations of RRA system.*

3 Proposed experiences

3.1 Reactor kinetic variables

3.1.1 Introduction

The objective of this experience is the study of the basic principles of the reactor kinetics. It is also inside the objective, the study of the parameters that influence this kinetics and the way in which the system answers to state variations leading it to subcritical or supercritical state. The study of the reactor kinetic will consider the neutron evolution without considering which could have been the causes that have modified its multiplying capacity. The reactor will be considered in a critical state and it was in a stationary regime until there has been a perturbation. This perturbation has induced changes in the reactor's reactivity, as a result of which the neutron evolution can undergo, throughout the time, diverse type of variations.

In order to carry out such intention, one will act on the system by introducing different reactivities series which will have the prompt jump shape. The student must visualize, among other variables and parameters, the reactor power evolution.

3.1.1.1 Prompt neutrons and delayed neutrons

An important aspect in the study of the reactor's kinetic is the differentiation of prompt and delayed neutrons in the model. The neutrons that are born in the fission reaction are in principle indistinguishable, to each other, except by two effects: its initial kinetic energy and the time expansion from they are born respect to the moment the fission turns to happen. The first of the differentiating characteristics is not in special importance since all neutrons are born with energy over the absorption resonances of the Uranium-238. Nevertheless, the second difference constitutes the great differentiation. Its origin is in particular disintegration chains of some fission products that end up emitting neutrons in search of a greater stability. When this happens, it appears a neutron that does not come from the fission directly, but through a disintegration chain. In general, in the disintegration chain, there is a radionucleus that has a half-life period higher than to the one of the other chain elements. This is the reason why is dominant in the time (precursor element). Actually the delayed neutrons usually are classified in 6 groups. Remembering that β_i represent the fraction of delayed neutrons of group *i* in relation to the total of emitted, the student must fill table 3.1; where there is also the decay constant and the disintegration average life-time for group i.

	U-235			Pu-239		
Group	Number of delayed neutrons for 1 fission neutron β_i	Decay Constant λ_i sec-1	Average Life t_i sec	Number of delayed neutrons for 1 fission neutron β_i	Decay Constant λ_i sec-1	Average Life t_i sec
1	0,00021	0,0124	80,4	0,00007	0,0128	78,3
2	0,00140	0,0305	32,8	0,00063	0,0301	33,2
3	0,00125	0,111	9,0	0,00044	0,124	8,1
4	0,00253	0,301	3,32	0,00069	0,325	3,07
5	0,00074	1,14	0,88	0,00018	1,12	0,89
6	0,00027	3,01	0,33	0,00009	2,69	0,37
Total	**β = 0,00640**	**λ = 0,080**	**t_r = 12,50**	**β = 0,00210**	**λ = 0,065**	**t_r = 15,4**

***Table 3.1-** Main parameters related with delayed neutrons from U-235 and Pu-239.*

3.1.1.2 Reactor kinetic equations

The equations that constitute the reactors kinetics foundation are obtained by starting of the diffusion law which dependent on time and taking it as prompt neutron source and the 6 delayed neutron groups.

$$\frac{dn}{dt} = \frac{\rho - \beta}{l^*} n + \sum_{i=1}^{6} \lambda_i C_i \qquad (3.1)$$

$$\frac{dC_i}{dt} = \frac{\beta_i}{l^*} n - \lambda_i C_i \qquad i = 1,\ldots,6 \qquad (3.2)$$

Where l^* indicates the prompt neutron generation time defined as l/k_∞ and being l the neutron life time in the moderator.

If one admits that the added reactivity in the system keeps constant at any moment, that is to say, feedback effects by temperature do not exist, the differential equation system is linear and of first order. In conclusion, the generic solution is a combination of exponential functions.

A way to obtain a simplified model is to suppose the hypothetical case of an only delayed neutron group, whose constant of disintegration λ represents one average of the six real groups.

Therefore,

$$\lambda = \frac{\beta}{\sum_{i=1}^{6} \frac{\beta_i}{\lambda_i}} \tag{3.3}$$

Introducing some simplifications in the resolution of the reactivity equation, the following temporary evolution of the neutron density is obtained:

$$n \approx n_0 \left(\frac{\beta}{\beta - \rho} \exp\left(\frac{\lambda \rho}{\beta - \rho} t \right) - \frac{\rho}{\beta - \rho} \exp\left(\frac{-(\beta - \rho)}{l^*} t \right) \right) \tag{3.4}$$

This approach can be considered suitable for lower positive reactivities down to 0.0025 and all negative reactivities. In these conditions, the first term has a positive coefficient whereas the second has negative. The second term disappears after a very short time interval. After it, the flux variation comes exclusively determined by the first term:

$$n \approx n_0 \left(\frac{\beta}{\beta - \rho} \exp\left(\frac{\lambda \rho}{\beta - \rho} t \right) \right) \tag{3.5}$$

3.1.1.3 Reactor period. Doubling Time

The *reactor period* or *reactor time constant* is defined as the temporary constant which is obtained from the exponential evolution in the expression (3.5). That is, it is the time that must pass to multiply (or to divide) the power by a factor *e*. Its expression, considering the expression (3.5), is:

$$T_P = \frac{\beta - \rho}{\lambda \rho} \tag{3.6}$$

For little reactivity insertion, $\rho << \beta$, this is:

$$T_P = \frac{\beta / \lambda}{\rho} \tag{3.7}$$

As β / λ has a defined value for each fissile species, for example *0,084* for the Uranium-235, it is a matter of fact that the reactor time constant, to low reactivities, is inversely proportional to the reactivity.

Another form to characterize the reactor evolution is through the doubling time. This parameter indicates the time to duplicate the power of the reactor. Therefore,

$$2 = \frac{P}{P_0} = \exp\left(\frac{T_d}{T_P}\right) \Rightarrow \frac{T_d}{T_P} = \ln 2 \tag{3.8}$$

being T_d the doubling time and T_P the reactor period.

3.1.2 Modus Operandi

First, it will be necessary to select the board that contains the variables that are tried to be visualize in the simulator. In the curve screen (*courbes*) click over  and select board *p1* from the list.

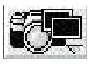
The following variables are showed in the board *p1*:

- *Flux chaîne puissance* (*W*).
- *Temps de doublement* (*s*).
- *Réactivité totale* (*pcm*).
- *Puissance nucléaire* (*MW*).

The difference between the variables *flux chaîne puissance* y *puissance nucléaire* is based in their units. The selection between either one or the other will take place in order of the power magnitude.

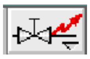
The same board *p1* will be used in all the experience.

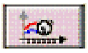
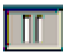
3.1.2.1 The β parameter calculation

1. Load the state corresponding to the Standard 4 of the simulator. In the instructor screen press  and choose *Standard*. Next, select in the list the state *Standard 4* and load it by pressing  in the top bar. This state is characterized by being critical in DDV at nuclear power of *50 kW* without temperature effects (hot standby).

Note: The reactor states, in the beginning, half and end of the cycle are called, in English and French: *BOL* (Begining Of Live), *MOL* (Middle Of Live), *EOL* (End Of Life) and *DDV* (Debut De Vie), *MDV* (Moitié De Vie), *FDV* (Fin De Vie), respectively. Is it also frequent to talk of ‘cycle’ instead of ‘life’, for what the previous terms can also appear as *BOC*, *MOC*, *EOC* and *DDC*, *MDC*, *FDC* in English and French respectively.

2. Activate the simulation by pressing over  which is located in upper part of all the screens.

3. Introduce a *100 pcm* step of positive reactivity in the reactor. For that reason, select the icon  (perturbations) which is placed in the upper instructors screen. In the *Choix* menu, choose *coeur* $\rightarrow$ *step de réactivité*. Then, select *100 pcm* in reactivity, active it and finally press *armement*. A progressive increase in the nuclear power will be seen in the curves screen.

4. If the time scale is reduced enough, the power jump that is produced will be able to be appreciated in the first two seconds approximately. For changing the time scale press  which is located in the lower part of the screen. Insert the new values of range of time and the step of the grid. Before closing the window it is necessary to press *Enter* to validate the changes. It can be practical to stop the simulation by pressing on  to take values of some variable. Write the values of the initial nuclear power (P_0) and the nuclear power after the jump (P_1).

5. Repeat the steps 1 to 4 for the reactivities: *50 pcm*, *-50 pcm*, *-100 pcm*, *-200 pcm*.

Note: Sometimes it is impossible to introduce exactly the suggested value. It is sufficient to select the nearest value that is allowed by simulator.

Note: As it is impossible to select reactivities with a value higher than *100 pcm*, the next method is suggested for introducing the *200 pcm* value. Select the icon  (local commands) which is placed up in the instructor screen and, inside the *Choix* menu, choose *coeur* $\rightarrow$ *réactivité initiale*. Then, reduce in *200 pcm* the predetermined reactivity activate and press *armement.*

Fill in the table with the obtained results:

ρ *(pcm)*					
P_0 *(kW)*					
P_1 *(kW)*					
β					

Table 3.2.- *Results of* β *parameter calculation.*

Related questions: 1-6

3.1.2.2 λ Parameter and period calculation with positive reactivities

1. Load the *Standard 4* state as it is explained in the section 3.1.2.1.

2. Activate the simulation by pressing  placed in the upper part of any screen.

3. Introduce *50 pcm* positive reactivity as it is explained in point 3 of section 3.1.2.1. Let the reactor evolve for about 2 or 3 minutes and check the stabilization of the doubling time.

4. Take the value of the doubling time.

5. Repeat the 1 to 4 steps with the following reactivities: *15 pcm*, *25 pcm*.

Complete the next table:

ρ *(pcm)*			
T_d *(s)*			

λ (s^{-1})			
T_p (s)			

Table 3.3.- *Results of λ parameter and period calculation results with positive reactivity.*

3.1.2.3 Reactor period calculation with negative reactivities

When negative reactivities are introduced in the simulator, it does not show information of the doubling time. For this reason, the period must be calculated by observing the evolution of the nuclear power that takes place.

1. Load the *Standard 4* state as it is explained in section 3.1.2.1.
2. Activate the simulation by pressing  placed in the upper part of any screen.
3. Introduce a *-100 pcm* negative reactivity in the reactor as it is explained in the point 3 of section 3.1.2.1. It is convenient to take the time value when introducing reactivity as this experience requires so.
4. Let the reactor evolve. Take values of the nuclear power in the 1, 2, 3, 4, 5, 6 y 7 minutes after the negative reactivity introduction.
5. Repeat the steps 1 to 4 for the following reactivities: *-50 pcm*, *-200 pcm*.

Taking two values from power at different moments it is possible to calculate the period of the reactor, since:

$$\left\{\begin{array}{l} P_1 = P_0 \exp\left(\dfrac{t_1}{T_P}\right) \\ P_2 = P_0 \exp\left(\dfrac{t_2}{T_P}\right) \end{array}\right. \quad \Rightarrow \quad T_P = \frac{t_2 - t_1}{\ln\left(\dfrac{P_2}{P_1}\right)} \tag{3.9}$$

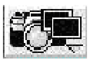
A possible way to observe the period evolution in the transitory consists of applying the previous expressions to consecutive power values. Complete table 3.4 with the collected data.

	ρ= P_0=		ρ= P_0=		ρ= P_0=	
Time after jump (min)	*P (W)*	*Period T_P (min)*	*P (W)*	*Period T_P (min)*	*P (W)*	*Period T_P (min)*

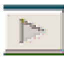
Table 3.4.- *Results in the calculation of the reactor period with negative reactivity.*

Related questions: 11-12

3.1.2.4 High reactivities insertion. Emergency Shutdown application

1. Load the state corresponding to *instructeur_1*. So, in the screen of the instructor, press on  and choose *instructeur*. Next, select in the list the state

instructeur_1 and load it by pressing on  located in the upper bar. This state is characterized by being critical to low power (*50 MW* of nuclear power) and at beginning of life (*DDV*).

2. Activate the simulation pressing on  located in the upper part of any screen.

3. Make the emergency shutdown in manual form. In order to carry out it, press on the button 

'*arret urgence*' located in the inferior part of the reactors display screen. This will be the moment of reference time.

Time after shutdown (min)	*P (W)*	*Period T_P (min)*	*Time after shutdown (min)*	*P (W)*	*Period T_P (min)*	*Time after shutdown (min)*	*P (W)*	*Period T_P (min)*

Table 3.5.- *High reactivities insertion results. Emergency Shutdown application.*

4. Let evolve the reactor and take values from the nuclear power every minute approximately until reaching about 15 min. Write down the values in table 3.5.

Applying the same method as in the section 3.1.2.3, complete the table by obtaining the reactor period for each time interval available.

Related questions: 13-15

3.1.3 Questions related with the experience

1. Why is taken as the initial reactor state at a low level of power? What would happen if it worked with an initial state of 100 % of nominal power?

2. What origin has prompt and delayed neutrons? Which is its explanation in the reactor control?

3. How evolves the β parameter (fraction of delayed neutrons) in a burned up cycle?

4. What value of the β parameter is obtained as a result in the simulations? What conclusion is given off in the fuel composition?

5. What consequences have the value of β in the reactor design and control? The cores that incorporate fuel MOX, do they see altered their kinetic?

6. What reactivity value makes critic the reactor with prompt neutrons?

7. What value of the λ parameter, stable period and doubling time are obtained as a result of the proposed simulations? Comment the results briefly.

8. What relation exists between the reactivity and the stable period or between the reactivity and the doubling time?

9. What evolution experiences the power in critical state if a step of inferior to beta reactivity is introduced? What final state is reached?

10. What evolution experiences the power in critical state if a step of reactivity superior to beta is introduced? What causes can originate the introduction of high reactivities?

11. Represent the evolution of the period for the three proposed simulations in a same graph.

12. Why the period increase in time is taken place abruptly by increasing the reactivity introduced?

13. Represent graphically the reactor period, as well as the evolution experienced by the nuclear power in a logarithmic scale.

14. Why the period is turned aside in the first minutes after the shutdown? Why the period increases progressively reaching a certain moment of time?

15. Would it be possible to induce a transitory shutdown whose period was lower to the observed one in the experience? Why? What role has the experienced period?

3.2 Subcritic Approximation

3.2.1 Introduction

3.2.1.1 Procedure description

One of the most important phases in the reactor normal operation is the one that makes reference to the start up. In this stage, special instruments that indicate the power level and allow a safe control are required. The neutron flux in each point of the reactor is a proportional parameter at the existing power level in the region. In so low power levels, devices are not adapted. So, due to these two reasons there are used the ionization chambers whose main advantage is the fast answer to local variations of power.

The interval of neutron flux in the start up stage has a huge variation (of the order of 10^{10}) causing that no individual instrument can satisfactorily measure the reactors neutron flux in all its interval from shutdown to full-power operation. For this reason it turns out to divide this interval in several regions or ranges with different associated instrumentation:

- Source range interval,
- Intermediate range interval,
- Power interval.

The way to carry out the reactors start up takes place by neutron absorbent extraction either by control rods withdrawn or by boron dilution, which have been introduced previously to assure a safe and widely subcritic state.

The most common procedure to make the start up and in which will be based the experience development, can be summarized in following the three steps:

1. *Boron dilution and determination of the critic boron concentration*. In this first stage, the reactor is far from being critical and one objective must be to eliminate, under control, part of the great amount of present boron in the primary circuit.

2. *Critical level determination of the regulation control bank for the critical boron concentration*. Once considered the boron concentration that makes critic the reactor, it is time to make the corresponding dilution. Insert previously the regulation bank for, this way, controlling the reactor with the control rods. Next, this regulation bank will be withdrawn under control until reaching criticality.

3. *Power level stabilization once reactor achieves criticality.*

3.2.1.2 Dilution water volume calculation

In order to carry out the dilutions, it will be precise to know previously the amount of water so that the absorber concentration diminishes in the wished amount.

V_0 is the primary circuit volume and c_0, c_1 are, respectively, the boron concentrations before and after the dilution. Making a matter balance for the element, *boric acid,* on the system, *primary circuit*:

Input flow = Output flow + Accumulation

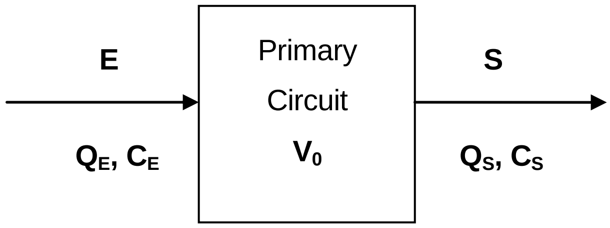


Figure 3.1.- *Simplified drawing of the primary circuit.*

Input: Make up water and pumps seal water.

Output: Letdown water and seal leak off water.

Accumulation: $\frac{dC}{dt} \neq 0$, non steady state.

$$Q_E\,C_E = Q_S\,C + V_0\,\frac{dC}{dt} \tag{3.10}$$

As $C_E = 0$ (neutron absorber free water):

$$Q_S\,C = -V_0\,\frac{dC}{dt}$$

$$Q_S\,dt = -V_0\,\frac{dC}{C}$$

Integrating by t_0 and t_1:

$$V_{H_2O} = Q_S\,(t_1 - t_0) = V_0 \ln \frac{C_0}{C_1} \qquad (3.11)$$

The expression (3.11) is important to obtain an approximate value that can be used in the dilution phase. It can be taken as a value of volume of the primary circuit 280 m^3.

3.2.2 Modus Operandi

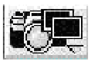
Initially, it will be necessary to choose the board which contains the most interesting variables to show its development throughout the experience. In the curves screen (*courbes*) press on  and select the list of variables *p2*.

The following variables are showed in the board *p2*:

- *Flux chaîne puissance (W)*
- *Concentration en bore du primaire (Cb)*
- *Position barres R*
- *Réactivité totale*
- *Position barres N1*
- *Position barres N2*
- *Débit dilution B*
- *Concentration en bore du ballon RCV*

The same board *p2* will be used through all the experience.

3.2.2.1 Reactor initial state characterization

1. Load the simulator state corresponding to *Standard 3*. In the instructor screen, press on  and choose *Standard*. Next, select in the list *Standard 3* state and load it by pressing on  located in the upper bar. This state corresponds to the reactor in *DDV* in hot standby.

Obtain from the upper bar the following values of generic parameters:

- Thermal Power (*Pth*):
- Electric Power (*Pele*):
- Moderator average Temperature (*Tmoy*):

- Primary Pressure (*P*):
- Boron Concentration (*Cb*):
- Total Reactivity (*R*):

Observe the rods position in the *coeur* screen:

- Safety bank or Shutdown bank (*S*):
- Power control bank (*N1*, *N2*, *G1*, *G2*):
- Regulation control bank (*R*):

Observe in the steam generators screen (GV) that the feed water valves of the GV remain closed and that the water level in the GV remains stable by making a small blowdown to the atmosphere (extracting steam by the *GCT-a* valve).

2. Activate the simulation by pressing on  placed in the upper part of any screen.

Note: In order to avoid losing information of the plant during the experience and, since the process can last enough time, it is advisable to save the present state every certain period of time (for example 15 minutes). In order to make it, it is enough with pressing on the icon  in the instructor screen. The saved state will appear in the *Instructeur* list without an associated name. If for some reason the simulation is lost and there has not kept previous states, it is possible to find the states that the program keeps automatically periodically and which they are in the list of *Periodique* states.

3.2.2.2 Previous Actions

3. In order to drive the start up operation, it will be necessary to make some modifications to prepare the secondary circuit. The following actions should be made pressing on the element whose state is desired to change.

   - Commute to automatic control the setpoint pressure (*Cons.*) of the blowdown valve to the *GCT-a* atmosphere. It will be able to eliminate the blowdown and it will allow, when the pressure of the produced steam is sufficient, the deflection of the steam towards the condenser, whose valve will also be controlled automatically.

   - Commute to automatic control the opening of the bypass of the turbine valve (*Vanne of contournement turbine*) to dump the created steam directly in the condenser.

- Start up the steam generators feedwater turbopump (*Turbo pompe alimentaire*) so that it allows to drive the condensed water into the steam generator.

- Regulate automatically the main feedwater of the steam generators with the small flow valve (*Vanne petit débit alim principal*). This valve will allow the circulation of the small water flow necessary to cool the primary circuit, during the operation.

- Cancel the small contribution coming from the auxiliary feedwater system, whose participation will not be necessary, closing completely the valve corresponding to the safety feedwater system (*Vanne d'alimentation secours*). It is necessary to validate to confirm the operation.

4. Withdraw *R* bank to level 170. Then, select in the core display screen the bank *R* and keep pressed the ascending arrow icon until reaching the indicated value. Verify the fast stabilization of the nuclear power.

5. Having selected the *N1*, *N2*, *G1* or *G2* banks indifferently, withdraw the power control banks until the commanded level *(0 % of electrical power*): *N1* bank to level 60. As the banks are overlap, the final position that will be obtained will be *N2* in the 213 step and *N1* in step 60. Verify the fast stabilization of the nuclear power.

6. The pressurizer heaters must be connected with the objective of having the pressure controlled by the pressurizer during the operation of approach to critic. In the pressurizer screen (*pressu*), locate the panel *chaufferettes* (heaters). Press '*manual*' and next '*marche*'. The heaters calorific power will increase until 1584 kW.

7. Carry out a measurement of the count rate (variable '*flux chaîne puissance*'). This value will correspond to measurement of the reference nuclear power P_0.

It is advisable the opening of a second letdown orifice with the purpose of duplicating the make up (charging) and letdown capacity, which has the primary circuit, through the chemical and volumetric control system. It is also a purpose to try to avoid problems that could appear in the pressurizer caused by variations of the primary water volume. Locate in the *RCV* screen, the make up and letdown flow from the primary circuit, the control tank, the borated water storage tank, the demineralised water storage tank and the interconnections between these elements.

8. In the Chemical and Volumetric Control System screen (*RCV*), prepare the primary circuit letdown by means of 2 lines (letdown orifices). That is, open a second orifice that allows the extraction of a greater water flow rate.

3.2.2.3 Dilution Phase

9. Reduce the primary circuit boron concentration operating the Chemical and Volumetric Control System (*RCV*). With this aim, carry out the boron dilutions by introduction of demineralised water flow in the primary circuit and extraction of the contained Bored water in the same circuit. Initially, it is proposed *200 ppm* boron dilutions, with the supposition that the primary circuit total volume is of *280 m3*. Calculate the necessary water volume for the process.

10. Once known the volume water to charge, select to the calculated volume ('*volume d'eau*') and the allowed maximum flow *30 t/h* ('*débit d'eau*') in the screen corresponding to the *RCV*, in '*dilution*'. It is necessary to validate in both cases. Finally, press '*dilution*', moment at which the valve that regulates the unloading of demineralised water storage tank is opened automatically. This valve closes itself once added the calculated volume. It can be observed that the boron concentration in control tank is always higher than the primary circuit concentration. Verify the stabilization of the flux rate and carry out the measurement of the nuclear power.

Note: Since the dilution process is quite slow, it is advisable, when the make up water flow rate has reached the *30 t/h* maximum, select the simulation speed indicated like '*Bore*50*'. For select it, press on  (upper part of the screen), which accelerates 50 times the simulation (exclusively as far as primary circuit chemical processes talks about). At this moment other interesting variations in the plant do not take place. This action will reduce the simulation time without disturbing the global result. It is also advisable to return to the simulation in real time whenever the dilution process finishes.

11. Make 3 dilutions more with the same water verifying the stabilization of the flow and taking note about the nuclear power value. Write down the results in table 3.6.

Note: It can appear the high level alarm in the control volume tank ('*RCV indisponible ou niveau ballon RCV anormal*') motivated by the letdown flow rate increase at some moment. In this point, it can be observed that the valve (that turns aside to the effluents treatment system (TEP) some letdown water) is opened automatically for the boron recovery and the water preparation.

12. It is advisable to reduce the dilution flow as the reactor state gradually comes near criticality. For example, *100 ppm* of boron may be used to obtain a slower approach to criticality. Therefore, it will be necessary to turn to calculate the water volume charging. Make the necessary dilutions until reaching a lower value of P_0/P to 0,02. Take notes from the results in tables 3,6 and 3.7.

13. Make the curve P_0/P as a function of the boron concentration (inverse count rate curve). Obtain, from the extrapolation of the curve last values, the boron concentration that makes the reactor critic.

Note: Since there must be the conditions for taking control with the rods, it is advisable to verify that the critical boron concentration (considered by the inverse count rate curve) is between *150* and *250 pcm* of equivalence below the present concentration. The calculation for a differential effectiveness *-10 pcm/ppm* must be made. If the critical concentration was lower than the actual in more than *25 ppm*, it can be made an extra dilution diminishing the boron concentration until being between the previous ranges. In that case, the critic boron concentration must be calculated again.

3.2.2.4 Rod Control Phase

14. In order to yield the control of the reactivity to the regulation control rods, place *R* bank to step 100 in a similar way to explained in point 4. It is necessary to wait for the nuclear power stabilization.

15. Once the *R* control bank has been introduced with the objective to make use of the control, dilute until the critical boron concentration (considered by the necessary water volume of the primary circuit make up). It is necessary to let stabilize the neutron flux and the nuclear power.

Note: In this point is advisable to verify that the reactors total reactivity value is not bigger than the reactivity that introduces *R* control bank. This last reactivity is located in the reactivity balance screen (*Bilan Réactivité*). In that case, the *R* bank total extraction would not reach criticality. If the reactor was in this situation it will be necessary to make a small dilution that would diminish the total reactivity below the reactivity controlled by the *R* bank.

16. Carry out a new measurement of the cameras count rate. This value corresponds to the new nuclear power reference measurement (P_0).

17. Approximate to criticality by means of the *R* bank withdrawal and write down the collect data in table 3.8. The first withdrawals can be of 10 steps, which

progressively must be reduced as an anticipation to criticality. It can be observed that the neutron flux stabilization (and, therefore, the nuclear power) is made slower with time when the reactor comes near to criticality.

Note: Initially, it is possible to work with a time scale of about 10 minutes and a power scale of *500 W* approximately. As time advances and the power is increased, it will be useful to change both, with the objective of being able to visualize better the evolutions.

Note: The neutron flux stabilization gets slower with time. By this reason, it can be practical to accelerate the simulation by 2 or 3, when the power level increases approximately *1 kW*, being selected it in the upper part of any screen.

18. Draw the curve P_0/P based on the *R* control bank position.

19. Let raise the nuclear flux and stabilize approximately, if it is possible, to a power of *0,01 %* of the nominal thermal power (*3868 MWth*).

20. Print the graphs, advisable for the accomplishment of the memory.

3.2.3 Questions related with the experience

1. What are the start up neutron sources and what are their functions?

2. What instrumentation allows to follow the critic approach from the nuclear point of view? What protections are established?

3. How evolves the neutron flux in subcritical range with neutron sources? And without neutron sources?

4. Why the reactor becomes stabilized more slowly as it comes near to the criticality?

5. Why it is observed that the control volume tank of the Chemical and Volume Control System contains a greater boron concentration than the primary circuit?

6. It becomes remarkable, especially between the last stages of the dilution process, a diminution of the nuclear power at the moments at which any dilution is carried out. What is the reason for this phenomenon?

7. What is the destination of the reactor letdown water? What amount of total boron has been extracted the system in the approach to critic process?

8. What is the reactor residual power and which is its origin? What differences are between the nuclear power and the thermal power?

9. If the cameras of source range operate between 10^{-9} to 10^{-3} % P_n, the intermediate cameras between 10^{-6} to 100 % P_n and the power ones between 0.1 to 200 % P_n, indicate what cameras provide the data during the approach to critic in the dilution stage.

3.2.4 Data diagrams

- **Boron dilution approximation**

P_0 =

V_{H2O}									
C_B (ppm)									
ρ (pcm)									
P (W)									
P_0/P									

Table 3.6- *Critic evolution for dilution with initial reference power.*

P_0 =

V_{H2O}						
C_B (ppm)						
ρ (pcm)						
P (W)						
P_0/P						

Table 3.7- *Critic evolution for dilution with the power reference change.*

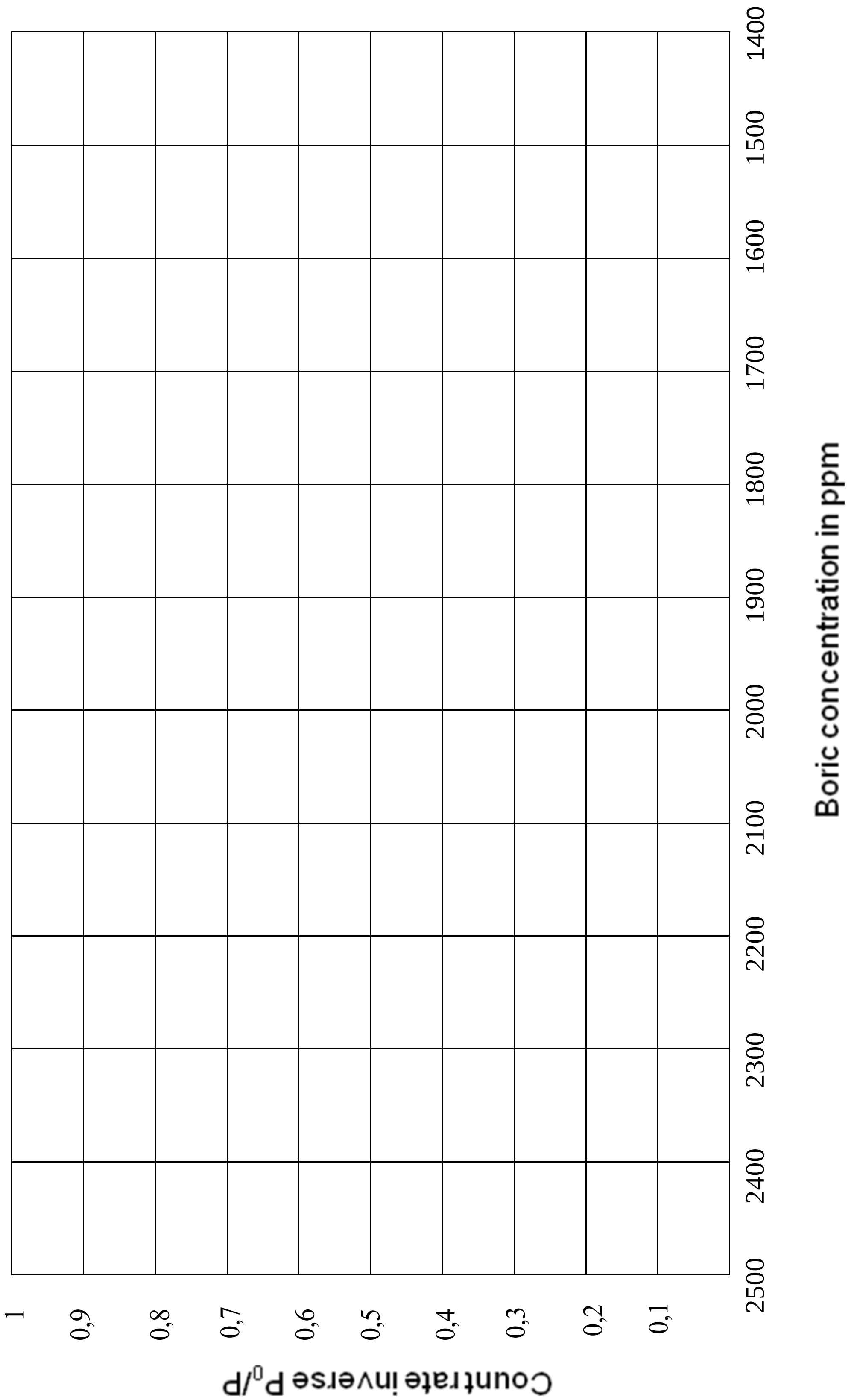
- **Bank R extraction approximation**

P_0 =

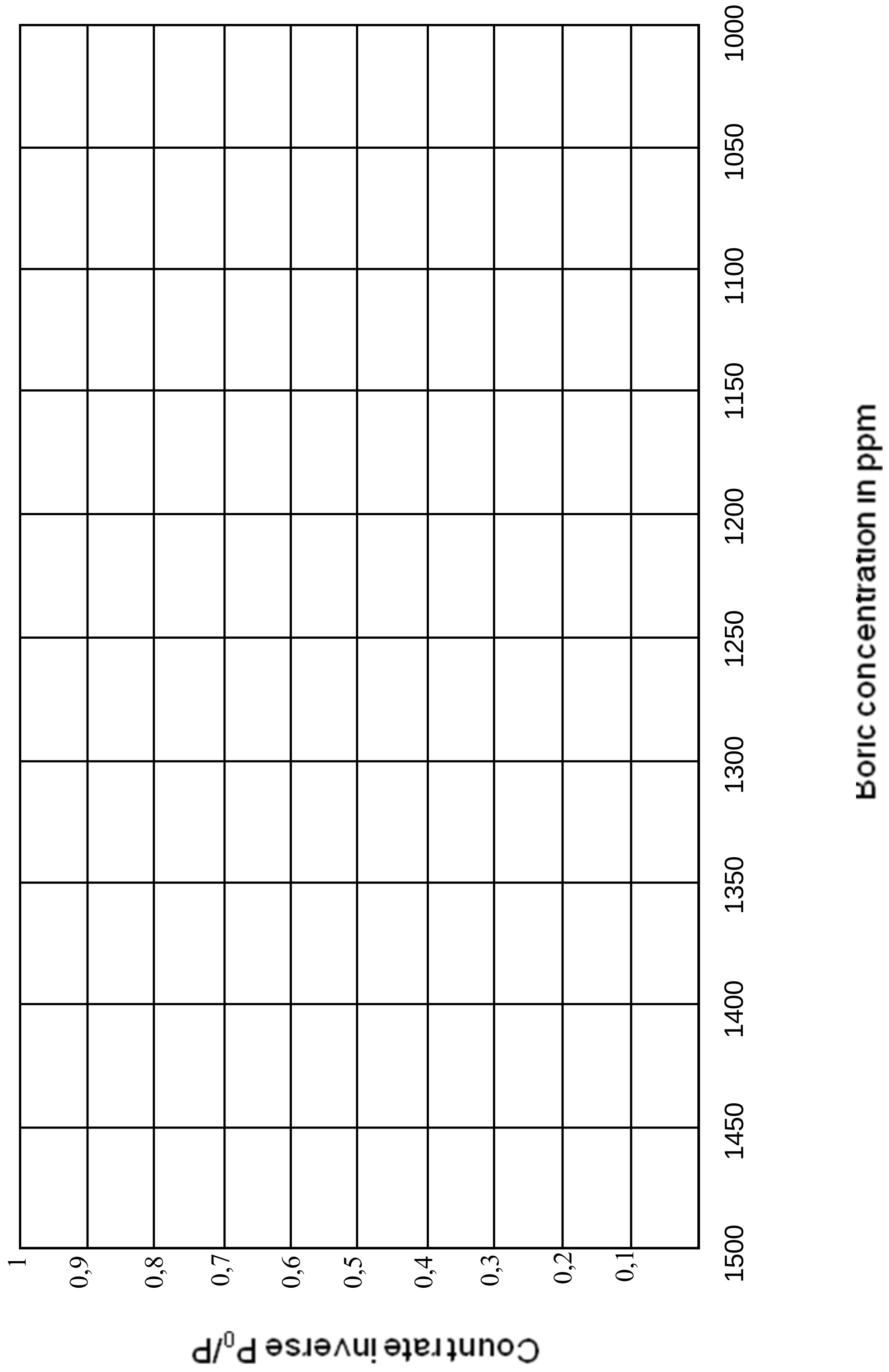
R (steps)									
ρ (pcm)									
P (W)									
P_0/P									

Table 3.8- *Critic evolution for bank R extraction.*

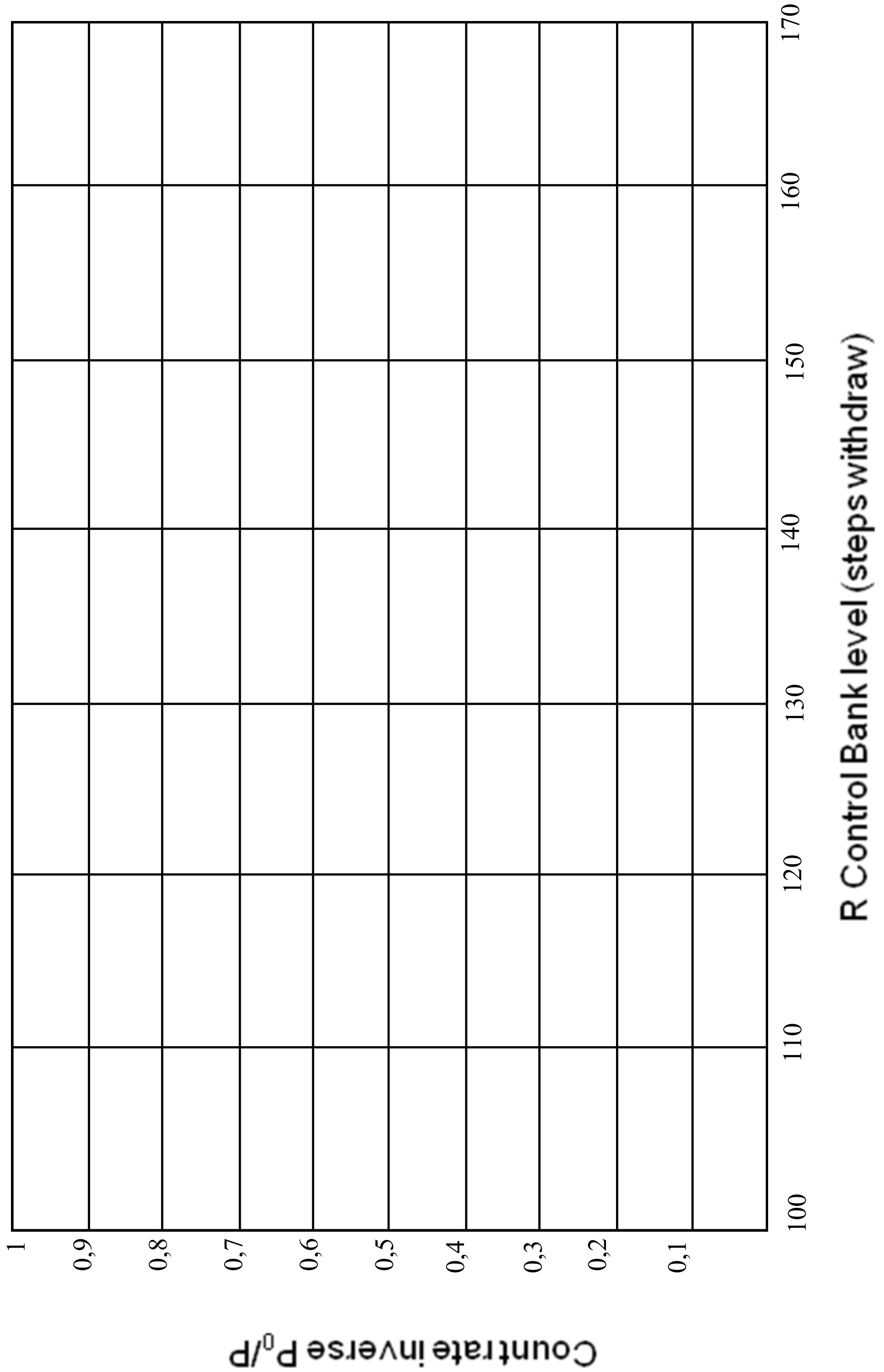
CRITIC APPROXIMATION BY DILUTION



CRITIC APPROXIMATION BY DILUTION



CRITIC APPROXIMATION BY BANK R



3.3 Reactivity temperature effects

3.3.1 Introduction

The reactors dynamics field study constitutes the main knowledge on this physical system. It is supported in the kinetics and is complemented with the incorporation of the physical mechanisms that affect the reactivity. A reducible complex phenomenology to analytical models exists according to that not only the reactor neutronics, trough the power, it affects the thermal hydraulic system. Moreover, the thermal hydraulic system affects as well to the neutron evolution by reactivity variations. It will be a complete closed feedback between the diverse variables of the reactors state, and it is extracted the more important fundamental consequence. This one consists in the possibility of designing the reactor with a configuration of a great physical stability, by which the reactor is against (as it acquires knowledge in this practice) to being separated of its nominal point of operation. This property is extremely positive, because the reactors behaviour in time and its safety do not have to depend on the control decisions or the automatic system. The own reactor, without waiting for no action of control by an external part, will modify its operation constants to be fitted at every moment to stabilized ranges. Furthermore, it will look for, after a small disturbance induced by any cause, a new definition of its critical state, and it will do it by itself with no need of a control system.

The variations of the system temperature appear with an important character between the causes that originate transitory modifications of the effective multiplication factor of a reactor in operation. Not only as a local form (for example, structural tolerances that they affect the coolant circulation in certain points) but main types affecting all the reactor as a whole (for example, variation of the coolant flow or variation of the power demand).

These temperature transitory effects have much importance in this experience. For this reason it is interesting to, not only to know if they cause increase or diminution in the multiplication factor value, but also the velocity in which these variations are pronounced.

Since the reactor is a heterogenous system, it is important to distinguish between *the fuel temperature coefficient* and *the moderator temperature coefficient* that reflect the temperature variations influence of the fuel and the moderator. These coefficients depend on different factors, so both will differ as much in magnitude as in sign. On the other hand, the time constants of the fuel and the moderator (the necessary time for a temperature variation to produce an appreciable variation of reactivity) usually are also quite different. The fuel temperature coefficient has a time constant smaller than the moderator coefficient. This fact emphasizes the fuel temperature coefficient importance in the reactors operation.

3.3.1.1 Fuel temperature coefficient

The most important temperature effect on the reactor reactivity constitutes the associated with the resonance escape probability. This phenomenon comes from the increase with the temperature of the effective resonance integral of the fuel absorption cross section diminishing the resonance escape probability.

Therefore, the resonance escape probability can suppose an important negative contribution to the temperature coefficient, in the reactors. As the effect comes determined by the fuel temperature, being independent of the moderator temperature, it is possible to hope a negative value in the fuel temperature coefficient in the reactors that use slightly enriched uranium. Therefore, in this type of reactors, the neutron capture by U-238 in the resonance region plays a very important role.

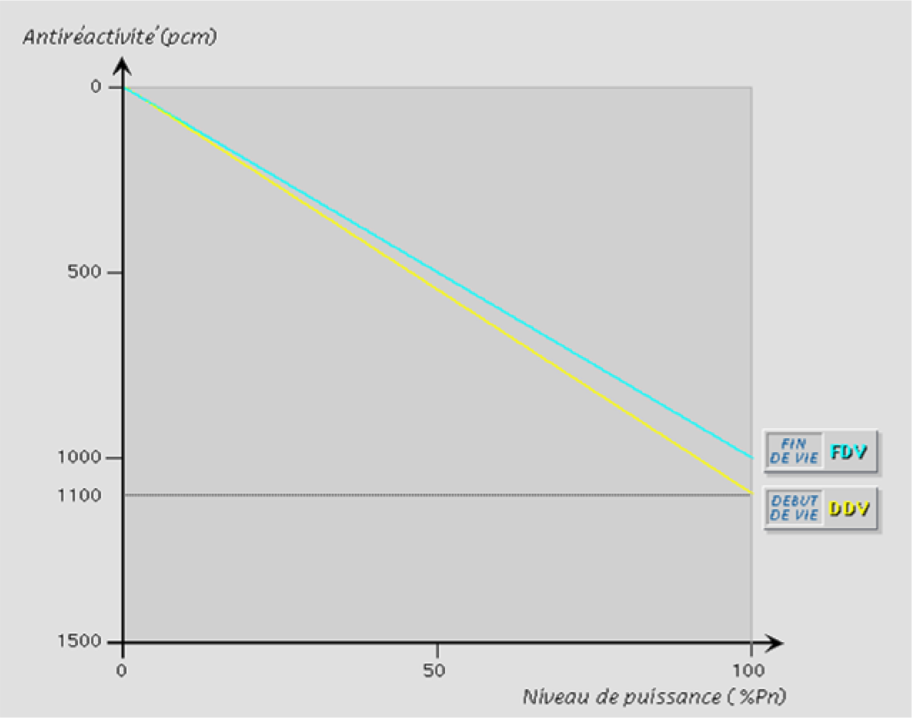


Figure 3.2.- *Doppler effects antireactivity according to the power level in BOL and in EOL.*

3.3.1.2 Moderator temperature coefficient

The moderator temperature determinates more than the fuel temperature coefficient, because this first coefficient can be positive or negative, according to the circumstances.

If the moderator (water in liquid state) increases its temperature, it will cause its expansion that will entail a diminution of the moderator concentration with respect to the fuel concentration. On the one hand, this effect will cause an increase of the thermal utilization factor. The physical meaning of this conclusion is that, when the temperature increases, more neutrons in the fuel are absorbed than in the moderator. Therefore, this phenomenon contributes with a positive value to the moderator temperature coefficient. On the other hand, the moderator density diminution also entails the resonance escape probability diminution because it is more probable that a neutron is absorbed in the fuel during their moderation process.

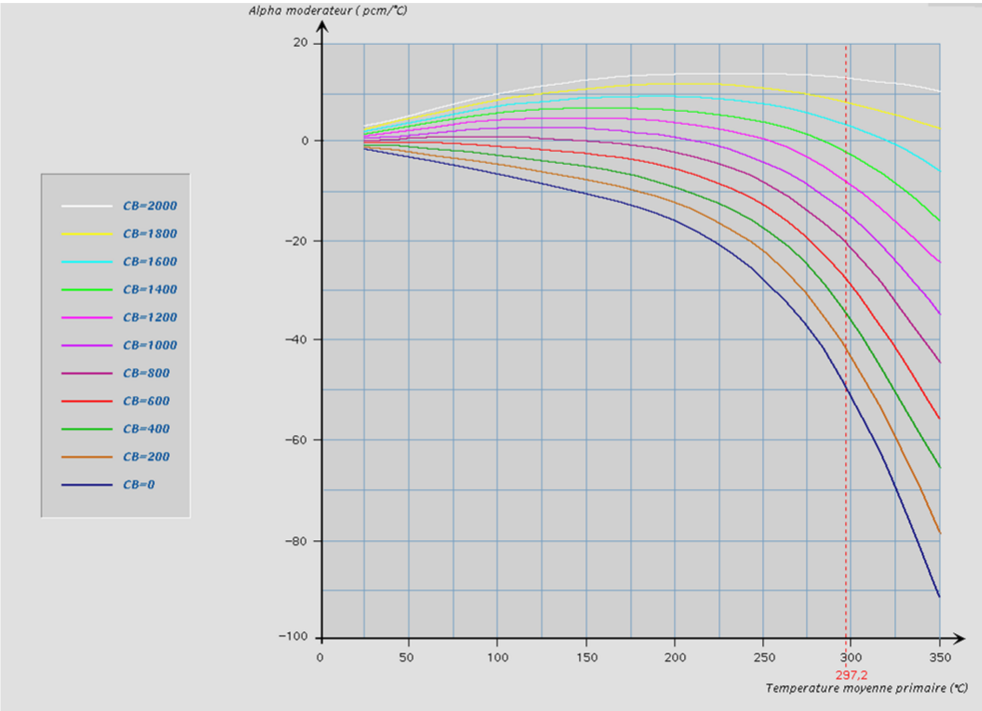


Figure 3.3.- *Moderador antireactivity according to the boron concentration and the primary circuit average temperature.*

As the resonance escape probability diminishes when the temperature increases, whereas the thermal utilization increases, habitually, the total moderator feedback coefficient value (sum of the individual coefficients) can be positive or negative. Nevertheless, a frequent case is that the resonance escape probability contribution, in absolute value, is the greatest one of the two, obtaining a total negative coefficient. In reactors *PWR*, as a design condition, the moderator coefficient is always negative during the power operation.

However, there is a particularity added in the reactors that use boric acid like neutron poison to compensate the reactor reactivity excess just loaded. It has, in this case, two additional effects that they contribute with positive sign to the moderator temperature coefficient. In the first place, the increase of temperature causes an increase of volume in the dissolution, reason why the poison concentration in the reactor diminishes. Secondly, it makes diminish the poison absorption cross section. When there is an increase in the temperature, the effectiveness of boric acid diminishes, increasing the reactivity. As it will be seen, this fact limits the boron concentration dissolved in the coolant.

Note: The graphic representations are also available in the simulator by pressing in the reactivity balance screen (*Bilan Reactivité*).

3.3.2 Modus Operandi

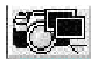
In the first time, it is required to choose the board that has the most interesting variables for showing the experience development. In the curves screen (*courbes*), press on and select the variable list *p3_1*.

In the board *p3_1*, there are showed the variables with the following viewing ranges:

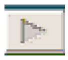
- *Température combustible* [290_320]
- *Réactivité Doppler* [650_750]
- *Température moyenne primaire* [290_320]
- *Réactivité modérateur* [2500_2600]
- *Réactivité totale* [-100_100]
- *Puissance nucleáire* [0_100]
- *Réactivité bore* [-12980_13080]
- *Réactivité grappes* [875_975]

3.3.2.1 Temperature effects appearance

In this first experience it is proposed, starting off a hot standby state, to visualize in what power levels there is the beginning of the feedback temperature effects. The reactor will become supercritical by acting on the control rods, and it will wait for the feedback temperature effects to be pronounced.

1. The simulator *Standard 4* state must be loaded. To carry out that, press on  and select *Standard* in the instructor screen. Moreover, select the *Standard 4* state and load by pressing on  located in the upper bar. This state references to the *DDV* reactor, critic, in a hot standby and with a *50 kW* of nuclear power value.

2. Take notes of the control rod banks positions and initial reactivity balance (*bilan réactivité*) in tables 3.9 and 3.10 (*Coeur* screen).

3. Activate the simulation pressing on  located in the upper section of any screen.

4. The following operations are due to prepare the secondary circuit with the purpose of being able to extract the steam generated in the primary circuit and to dump it directly to the condenser. In order to make the modifications on a certain element it is enough with selecting it. From screen GV:

    - Pass to automatic control the setpoint pressure of the blowdown valve to the atmosphere *GCT-a*. In this way, one will be able to eliminate the blowdown and to allow, when the pressure of the generated steam is sufficient, the dump of the steam towards the condenser, whose valve will also be controlled automatically.

    - Pass to automatic control the opening of the turbine by-pass valve (*Vanne of contournement turbine*) to dump the steam generated directly in the condenser.

    - Start up the feedwater turbopump of the steam generators (*Turbo pompe alimentaire*)

    - Regulate automatically the main feedwater of the steam generators with the small flow valve (*Vanne petit débit alim principal*).

    - Cancel the small contribution coming from the safety feedwater system of the steam generators by closing completely the valve corresponding to the safety feedwater (*Vanne d'alimentation secours*). It is necessary to validate to confirm the operation.

5. Locate, in the reactivity balance screen (*bilan réactivité*), the indicator of the reactivity controlled by the rods (*grappes*).

6. Eliminate *50 pcm* of rods antireactivity in the reactor by extracting the power control rods banks. To carry out this action, press on the banks *G1*, *G2*, *N1* or *N2* indifferently on the core display screen, and next, press the key that shows the ascending arrow. The withdrawal will have to be stopped when the rods reactivity indicator (*grappes*) has diminished, in absolute value in *50 pcm*. The final reached position will have approximately to be: *G2* in the step 28 and *N1* in step 201.

	safety	*control*				
	S	*N1*	*N2*	*G1*	*G2*	*R*
Initial State (St 4) *P = 50 kW*						
Final State *P = MW*						

Table 3.9.- *Control and safety rods banks position.*

	Doppler	*Moderator*	*Xenon*	*Samarium*	*Boron*	*Rods*
Initial State (St 4) *P = 50 kW*						
Final State *P = MW*						
Balance						

Table 3.10.- *Reactivity balance by the different terms from which is composed.*

Note: The reading of the reactivity value that is introduced must always be made by the way indicated, since if the total reactivity present in the core was observed, you could be observing other possible effects in addition, like a fast actuation of the feedback by temperature, although in this particular case its can be agree.

7. The variables evolution must be observed in the graphic representation. Wait until the new state stabilization arrives. It is convenient to accelerate the simulation by 3 times by selecting this option in .

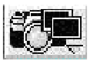
8. The graphic representation must be printed.

Complete the tables 3.9 y 3.10.

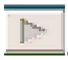
Related questions: 1-6

3.3.2.2 Reactivity feedback effects with a nominal power reactor in EOL

In the sections 3.3.2.2 and 3.3.2.3, it is advisable to work with the variables list *p3_23*, that, besides to have more suitable ranges for a correct visualization, contains the positioning variable of the *G1* bank (*Position barres G1*).

1. Load the simulator state corresponding to *Standard 9*. To carry out that, in the instructor screen you ought to press on  and choose *Standard*. Next, select in the list the *Standard 9* state and load it by pressing on  located in the upper bar. This state corresponds to the reactor in *FDV*, critic, with low boron concentration and in operation to 100 % of nominal power, equivalent to *3868* thermal MW.

2. Necessary data must be obtained for completing 3.11.

3. Simulation must be activated by pressing on .

4. With the objective to visualize the reactor natural behaviour without intervention of the automatic adjustment, it is advisable, to manually control all the control rods. Select the operation in manual mode by pressing on  as much for bank *R* as for bank *G*.

5. Introduce, by means of the *G1* control rods, a variation of *100 pcm* in a similar way as made in the point 6 of the section 3.3.2.1, but now introducing the rod.

6. Proposed rods variables evolution must be observed and plant stabilization must be checked.

7. Print the graphic representation and complete the table 3.11.

	T fuel (ºC)	*T moderator (ºC)*	*ρ_F (pcm)*	*ρ_M (pcm)*	*ρ_{rods} (pcm)*
Initial					
Final					
Balance					

Table 3.11.- *Temperature feedback effects in the bank G1 partial introduction in FDV.*

Return to carry out the experience but in this occasion with the automatic control in the *R* control rod bank. This bank participation and the variations that take place related with the previous situation must be checked. In this case, it must be noticed that the reference indicator, to consider the reactivity of *100 pcm* to be introduced, must correspond only to the associated to the bank *G1*, since the total of rods (*grappes*) will contain the variations of the *G1* and *R* banks together.

Finally, print the graphic representation and take notes of the results in table 3.12.

	T fuel (ºC)	*T moderator (ºC)*	*ρ_F (pcm)*	*ρ_M (pcm)*	*ρ_{rods} (pcm)*
Initial					
Final					
Balance					

Table 3.12.- *Temperature feedback effects in the bank G1 partial introduction with bank R in automatic mode.*

3.3.2.3 Reactivity feedback effects with a nominal power reactor in BOL

Repeat the steps as in the section 3.3.2.2, with manual regulation of *R* rods, but this time on *Standard 7* state. This other state corresponds to the critical reactor at beginning of life (*DDV*) and, therefore, with a boron concentration C_b higher to the concentration of the previous section. Nevertheless, such concentration is below the critical concentration, so the moderator temperature coefficient continues having negative sign.

	T fuel (ºC)	*T moderator (ºC)*	*ρ_F (pcm)*	*ρ_M (pcm)*	*ρ_{rods} (pcm)*
Initial					
Final					
Balance					

Table 3.13.- *Temperature feedback effects in the bank G1 partial introduction in BOL.*

Given the greater concentration of absorbent in the moderator, the system dynamics answers in a different way related to the previous situation. On the one hand, it must be observed that as the transitory takes place in a greater time, it needs more time to become stabilized. On the other hand, it is shown that the moderator final temperature is clearly inferior to the previous case, because the moderator coefficient has acted with a smaller effectiveness.

Related questions: 7-11

3.3.3 Questions related with the experience

1. What are the effects of temperature feedback?
2. What is the importance of temperature effects in the reactor stability?
3. How do the fuel and moderador feedback coefficients change as a function of temperature? Why?
4. What is the moderator coefficient design and operation condition?
5. In section 3.3.2.1 experience, in what nuclear power range can the temperature coefficients start to be detectable?
6. In section 3.3.2.1, which of the both feedback effects (fuel or moderator) have mainly compensated the reactivity introduced?
7. Discuss the evolution of the power, temperature and reactivity variables of the fuel and moderator.

8. In sections 3.3.2.2 and 3.3.2.3, what is the first temperature feedback effect to appear?

9. In this case, which of the both feedback effects (fuel or moderator) have mainly compensated the reactivity introduced?

10. Which are the differences between the reactivity feedback transitories in *BOL* and *EOL*?

11. Calculate by means of tables 3.11 and 3.13 the $\bar{\alpha}_F$ and $\bar{\alpha}_M$ average coefficients in the temperature range studied. Remember that the feedback coefficients are defined as:

$$\bar{\alpha}_F = \frac{\Delta\rho_F}{\Delta T_F} = \frac{\rho_F^f - \rho_F^i}{T_F^f - T_F^i} \qquad \bar{\alpha}_M = \frac{\Delta\rho_M}{\Delta T_M} = \frac{\rho_M^f - \rho_M^i}{T_M^f - T_M^i}$$

3.4 Isothermal coefficient and moderator coefficient

3.4.1 Introduction

The isothermal coefficient is one of the parameters that are used in the nuclear reactor characterization. Since the reactors have temperature coefficient of the global reactivity negative, the effective multiplication factor will be smaller in high temperatures than in standard temperatures. During the startup operation, the reactor temperature increases inevitably, so k_{ef} diminishes. Therefore, it is necessary to give an additional reactivity to the cold reactor, in order to increase the effective multiplication factor value. This factor value must exceed the unit in the operating temperature. Thus, for example, given a value of the average feedback reactivity coefficient in the range between the standard temperature and the operation temperature, it is necessary for the additional reactivity to have the value of the product of the average feedback reactivity coefficient by the temperature interval.

Such supposition is based on the isothermal temperature coefficient, which takes the associated hypothesis that the fuel and the moderator are in the same temperature. Referring to temperatures, this hypothesis can be considered perfectly valid because in these conditions the fuel is preheated by the moderator, making a homogenous system. Moreover, all the component presents in the reactor core have the same temperature in all the interior points. The isothermal coefficient of the reactor is defined as:

$$\alpha_{iso} = \frac{d\rho_{total}}{dT} = \frac{\Delta\rho_{total}}{\Delta T} \tag{3.12}$$

being *T* the core temperature.

The following property can be observed in these thermal uniform temperature conditions:

$$\alpha_{iso} = \frac{d\rho_F}{dT} + \frac{d\rho_M}{dT} = \alpha_F + \alpha_M \tag{3.13}$$

The objective of the present experience is based on the isothermal coefficient approximate estimation by experimentation on the plant in different situations of the reactor states from cold shutdown to hot shutdown.

On the other hand, the addition of boron to the water of a *PWR* will be limited by the appearance of positive feedback situation in reference to moderator reactivity. The operation with a greater dilution value will not be allowed than the one which produces a change of submoderation to supermoderation, in the nominal density of operation. In this sense, in the second part of the experience, the reactor will be in an operation temperature and you will work with different absorbent concentrations, always lower than the absorbent critical concentration. The objective of this experience will be to consider which is the critic boric acid concentration. That is to say, the one that turns the moderator feedback temperature coefficient to a positive value, affecting to the reactor stability in negative way.

3.4.2 Modus Operandi

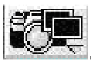
In the first place, it will be needed to load the boards that contain the variables with their ranges adapted for a correct visualization. In that case, each initial state is followed by a variables list with different ranges to improve the use and to facilitate the interpretation. The contained variables are:

- *Température combustible*
- *Réactivité Doppler*
- *Température moyenne primaire*
- *Réactivité modérateur*
- *Réactivité totale*
- *Débit RRA→Primaire*
- *Débit vapeur contournement atmosphère*
- *Pression GV (barrilet vapeur)*
- *Puissance nucléaire*
- *Débit vapeur contournement condenseur*

The names that identify the lists are: *p4_stand1*, *p4_stand2* and *p4_instru2* until *p4_instru8*.

3.4.2.1 Isothermal coefficient determination in different preheating states

It will be made the same experience on reactor states with different boric acid concentrations and different temperatures with the objective to obtain a correct parameterization of the reactor isothermal coefficient. There will be arranged, therefore, diverse states with temperatures between cold shutdown and operation temperature, with concentrations between *2600 ppm* and *2700 ppm*. The actions will be made by annulling the residual heat removal system, which will cause a moderator heating and, therefore, of the fuel, too.

1. Load the *Standard 1* state which you will find by pressing on . This state is characterized by being an intermediate moderator preheating state.

2. Take notes of the boron concentration C_b in table 3.14.

3. Activate the simulation.

4. Locate, in the residual heat removal system screen (*Bulle RRA*), the two valves that connect this system with the primary circuit and close them manually (*Vanne RRA*$\rightarrow$*primaire* and *Vanne primaire*$\rightarrow$*RRA*). The continuous coolant recirculation by the reactor will origin an increase of the moderator temperature, caused by the fuel residual power. The fuel temperature increase must also be observed slightly delayed as it has been given the reactor state of thermal homogeneity. Therefore, a contribution of negative reactivity on the fuel must take place. How evolves the moderator feedback in increasing its temperature?

5. Let evolve the reactor about 10 minutes approximately and write down the data in table 3.14. Then, determine the fuel feedback coefficient α_F, the moderator coefficient α_M and the isothermal coefficient α_{iso}.

6. Print the graphic representation.

   Repeat the same procedure on states corresponding to other intermediate preheating states that have slightly less boron concentrations and they are characterized to have higher moderator temperatures.

7. Load the “Standard 2” state. Proceed in the same way that with “standard 1” state, so, closes manually the valves that connect residual heat removal system with the primary circuit. Repeat the steeps from 2 to 6.

8. Load the "Instructuer 2" state. In this case the two valves that connect this system with the primary circuit (*Vanne RRA*$\rightarrow$*primaire* and *Vanne primaire*$\rightarrow$*RRA*) are already closed. Act on the secondary circuit. Close the blowdown valve to the atmosphere *GCT-a*. The other steeps from (2) to (6) are the same.

Intermediate Shutdown:	**Standard 1**	Nuclear Power =		C_b=	
Rods position:	N1=	N2=	G1=	G2=	R=
	ρ (pcm)	ρ_F (pcm)	ρ_M (pcm)	T_F(ºC)	T_M(ºC)
Initial					
Final					
Balance					
	α_{iso} **(pcm/ºC)**	α_F **(pcm/ºC)**	α_M **(pcm/ºC)**		
FB coeff					

Table 3.14.- *Experiences results in Standard 1 state.*

Intermediate Shutdown:	**Standard 2**	Nuclear Power =		C_b=	
Rods position:	N1=	N2=	G1=	G2=	R=
	ρ (pcm)	ρ_F (pcm)	ρ_M (pcm)	T_F(ºC)	T_M(ºC)
Initial					
Final					
Balance					
	α_{iso} **(pcm/ºC)**	α_F **(pcm/ºC)**	α_M **(pcm/ºC)**		
FB coeff					

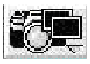
Table 3.15.- *Experiences results in Standard 2 state.*

Intermediate Shutdown:	**Instructeur 2**	Nuclear Power =		C_b=	
Rods position:	N1=	N2=	G1=	G2=	R=
	ρ *(pcm)*	ρ_F *(pcm)*	ρ_M *(pcm)*	T_F*(ºC)*	T_M*(ºC)*
Initial					
Final					
Balance					
	α_{iso} **(pcm/ºC)**	α_F **(pcm/ºC)**	α_M **(pcm/ºC)**		
FB coeff					

Table 3.16.- *Experiences results in Instructeur 2 state.*

3.4.2.2 Determination of critic boron concentration in full power operation temperature

The critical boron concentration will be determined by experimentation on diverse low power critical states with the moderator temperature around 290 ºC - 300 ºC (full power operation temperature). Such states could be representative from the different situations that can be found throughout an operation cycle of the power plant.

1. Load the *Instructeur 3* state that is pressing on . This state is characterized to arrange the reactor in hot standby, critical in low power but without temperature gradient. Therefore, the temperatures of fuel and moderator are the same.

2. Take note of the boron concentration value C_b.

3. Activate the simulation.

4. Observe the secondary circuit. The *SG* feedwater pump is in operation and the small flow feed valve remains operative in automatic mode. Thus, the system has capacity to feed the steam generator from extracted water of the condenser.

5. Locate the turbine by-pass valve (*Vanne conteurnement turbine*) in the generator screen (*GV*). In manual control, open it approximately 2 %. It is necessary to validate the action. With this operation, a dump of a steam flow to the condenser is made. It can be observed as the generator operation conditions are readjusted and, as a result of it, there is a diminution of the moderator temperature. The fuel temperature diminution must also be observed slightly delayed since the reactor is under thermal homogeneity state. Therefore, a positive total reactivity contribution must take place.

6. Close the by-pass valve once has been observed that the moderator temperature has diminished 3 ºC approximately.

7. Take a moment from time in which the reactor parameters have become stabilized and write down the data in table 3.17. Then, calculate the temperature feedback coefficients in the same way as in the section 3.4.2.1.

Hot Standby:	**Instructeur 3**	Nuclear Power =		C_b=
Rods position: N1=	N2=	G1=	G2=	R=
	ρ *(pcm)*	ρ_F *(pcm)*	ρ_M *(pcm)*	T_F(ºC) = T_M(ºC)
Initial				
Final				
Balance				
	α_{iso} **(pcm/ºC)**	α_F **(pcm/ºC)**	α_M **(pcm/ºC)**	
FB coeff				

Table 3.17.- *Experiences results in Instructeur 3 state.*

Hot Standby:	**Instructeur 4**	Nuclear Power =		C_b=
Rods Position: N1=	N2=	G1=	G2=	R=
	ρ *(pcm)*	ρ_F *(pcm)*	ρ_M *(pcm)*	T_F(ºC) = T_M(ºC)
Initial				
Final				
Balance				
	α_{iso} **(pcm/ºC)**	α_F **(pcm/ºC)**	α_M **(pcm/ºC)**	
FB coeff				

Table 3.18.- *Experiences results in Instructeur 4 state.*

Hot Standby:	**Instructeur 5**	Nuclear Power =		C_b=
Rods Position: N1=	N2=	G1=	G2=	R=
	ρ *(pcm)*	ρ_F *(pcm)*	ρ_M *(pcm)*	T_F(ºC) = T_M(ºC)
Initial				
Final				
Balance				
	α_{iso} **(pcm/ºC)**	α_F **(pcm/ºC)**	α_M **(pcm/ºC)**	
FB coeff				

Table 3.19.- *Experiences results in Instructeur 5 state.*

Hot Standby:	**Instructeur 6**	Nuclear Power =		C_b=
Rods Position: N1=	N2=	G1=	G2=	R=
	ρ *(pcm)*	ρ_F *(pcm)*	ρ_M *(pcm)*	T_F(ºC) = T_M(ºC)
Initial				
Final				
Balance				
	α_{iso} **(pcm/ºC)**	α_F **(pcm/ºC)**	α_M **(pcm/ºC)**	
FB coeff				

Table 3.20.- *Experiences results in Instructeur 6 state.*

Hot Standby:	**Instructeur 7**	Nuclear Power =		C_b=
Rods Position: N1=	N2=	G1=	G2=	R=
	ρ *(pcm)*	ρ_F *(pcm)*	ρ_M *(pcm)*	T_F(ºC) = T_M(ºC)
Initial				
Final				
Balance				
	α_{iso} **(pcm/ºC)**	α_F **(pcm/ºC)**	α_M **(pcm/ºC)**	
FB coeff				

Table 3.21.- *Experiences results in Instructeur 7 state.*

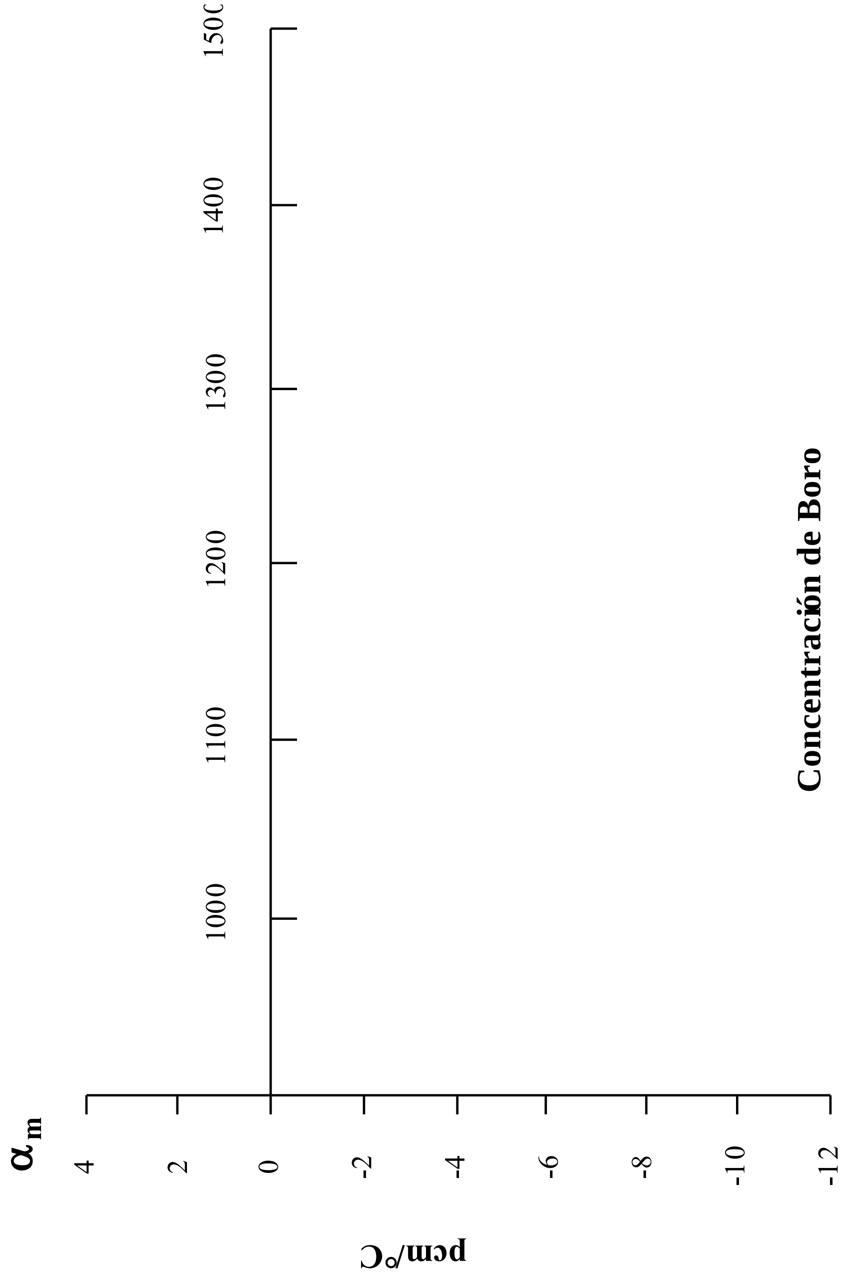
Hot Standby:	**Instructeur 8**	Nuclear Power =	C_b=	
Rods Position: N1=	N2=	G1=	G2=	R=
	ρ *(pcm)*	ρ_F *(pcm)*	ρ_M *(pcm)*	T_F(ºC) = T_M(ºC)
Initial				
Final				
Balance				
	α_{iso} **(pcm/ºC)**	α_F **(pcm/ºC)**	α_M **(pcm/ºC)**	
FB coeff				

Table 3.22.- *Experiences results in Instructeur 8 state.*

8. Print the graphic representation.

9. Repeat the same procedure on the *Instructeur 4* until *Instructeur 8* states corresponding to states with different boron concentrations.

10. Draw the graphic representation of α_M related with the boric acid concentration.



11. Determine, by extrapolation, the critical boron concentration that would provide a moderator temperature coefficient of reactivity with positive value.

3.4.3 Questions related with the experience

1. What is the reactor isothermal coefficient?

2. How is the isothermal coefficient determinated? What is its utility?

3. What sign do the isothermal feedback coefficient and the moderator coefficient have in the conditions of the section 3.4.2.1.? Why? Compare the results obtained with the graph of figure 3.3.

4. How does the moderator coefficient change with the temperature and the boron concentration?

5. For a constant boron concentration, how does the moderator coefficient change with an increase in temperature?

6. In reference to where the moderator temperature stays constant, how does the moderator coefficient change with a diminution in the boric acid concentration?

7. How does limit boron concentration evolve with the moderator temperature?

8. Explain briefly the processes that have been used during the experience to obtain to the temperature variation of the fuel-moderator system.

3.5 Reactor start ups and load variations

3.5.1 Introduction

3.5.1.1 Start up Xenon effects

The reactor start up followed by a brief standby is a process less complex than the initial start up. If there has not been an excessively great delay (some minutes), the neutron level produced by delayed neutrons can still be the sufficiently high to obtain a good instrumental data.

An important peculiarity of the nuclear reactor operation that affects the start up is constituted by the poison derived from fission products. During the reactor operation period, there are accumulated fission fragments and their numerous disintegration products. Among them, *Xenon-135* and the *Samarium-149*, play a very important role in the transitory periods of the reactor power. Consequently, when taking place a total reactivity variation, which brings with itself a neutron density variation, it is also modified the poisons concentration and this one modifies, as well, the total reactivity. Thus, although the fission products do not influence in a great measure the reactor kinetic, they carry out important effects on the reactivity. For this reason, the control systems must be projected considering this particularity, since they will have to compensate the reactivity fluctuations that come from the appearance or elimination of poisons.

Therefore, the transitory periods caused by Xenon-135 constitute an important effect that the chemical and volumetric control system must compensate in the start up and the power reduction that will be made in the experience.

3.5.1.2 Start up procedures after an emergency shutdown (A.U.)

There is a related phenomenology in neutronics poisons conditions in the reactors start up after a shutdown. In this section, it is grouped the set of operations to carry out the core to the critical state. This group is included in the procedures of the operation technical specifications.

It can be distinguished two procedures by whether there has to be done a reactivity balance or whether there has not. In case the reactivity balance is needed, three options will be distinguished. The reactor state and the procedure to be carried out in the different situations are showed in the drawing (fig 3.4).

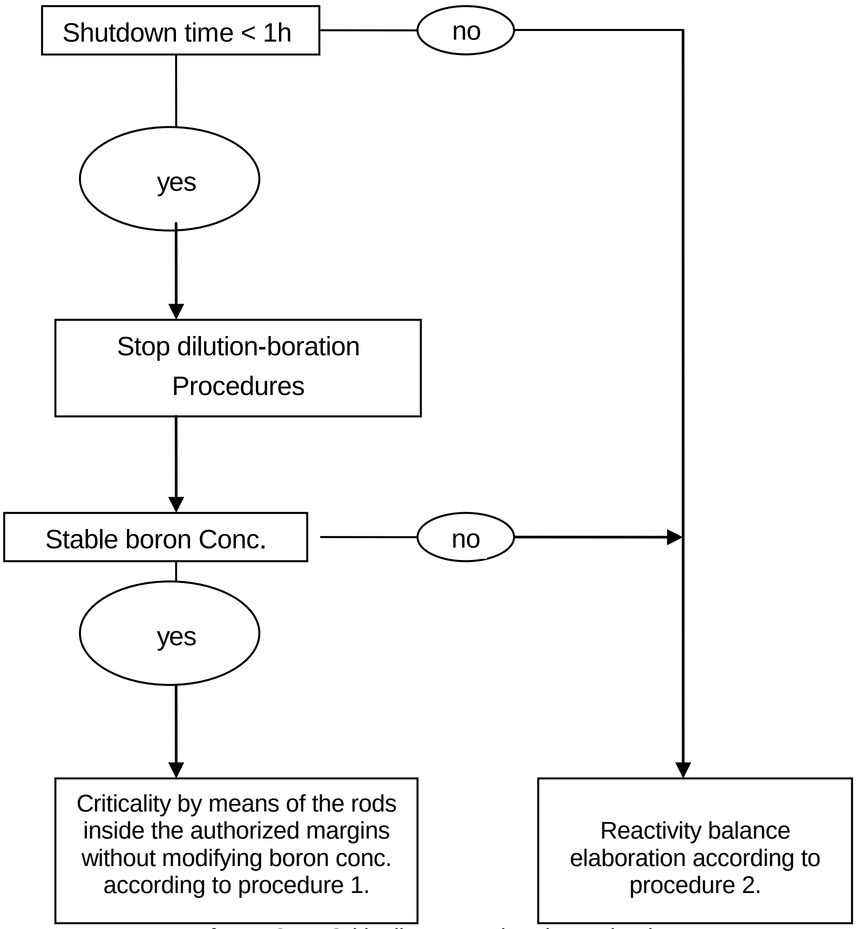


Figure 3.4.- *Criticality strategies determination.*

In the section 3.5.2.2 of this experience will be made the start up by means of making criticality by the rods, displaying the steps that are due to follow. The figure 3.5 allows to identify the most suitable case of actuations in the case of needing the reactivity balance (procedure 2). These can be summarized in the following operations:

- **Option 1**: Regulation and power rod banks withdrawal and positioning in its command step. A quick dilution in a maximum flow must be done drawing the curve $P_0 / P = f(C_b)$. The critical concentration must be estimated and it has to be diluted until criticality.

- **Option 2**: The primary circuit must be bored up to the critical concentration + *20 ppm*. It must be withdraw, regulation and power control banks. A quick dilution in a maximum flow must be done until criticality.

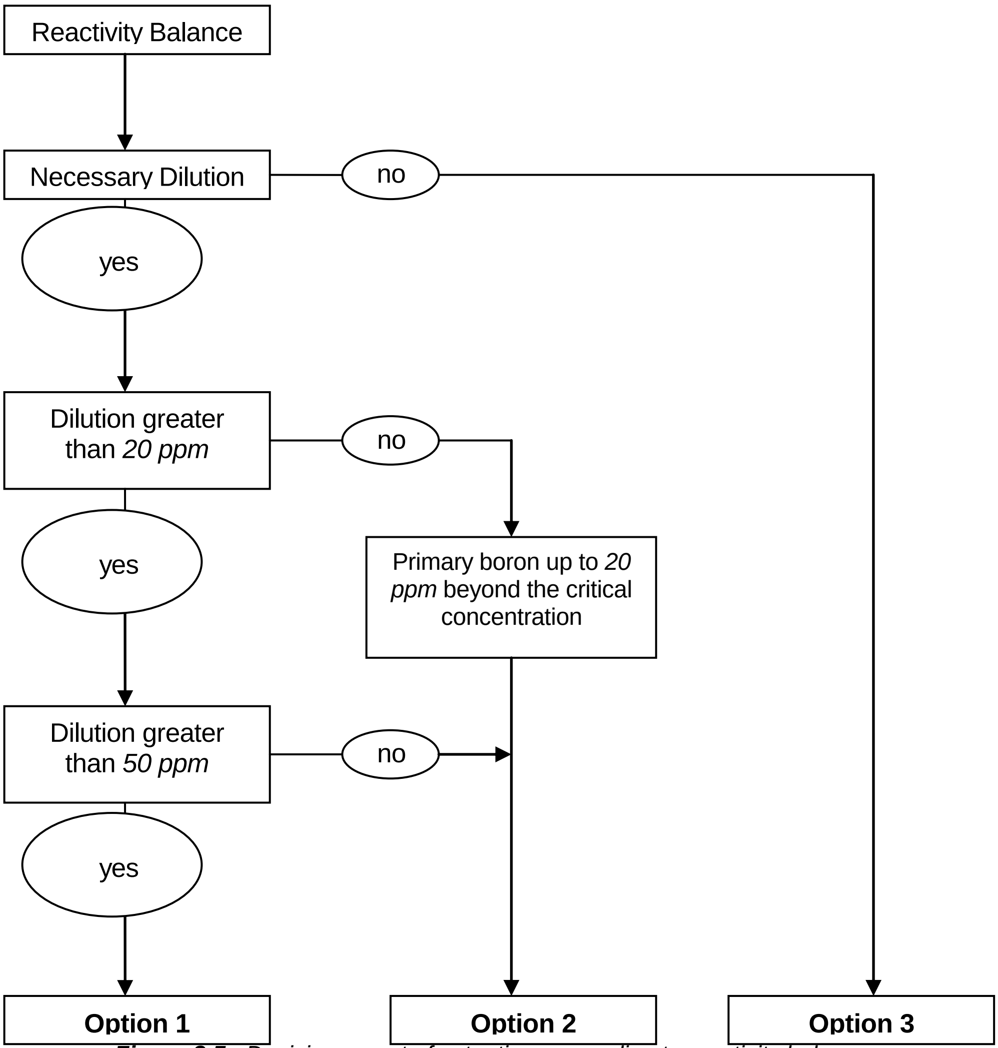


Figure 3.5.- *Decision on set of actuations according to reactivity balance.*

- **Option 3**: It must be bored, in a maximum flow, up to critical concentration + *20 ppm*. Shutdown bank withdrawal, regulation control bank withdrawal to command position and power control bank withdrawal by means of the curve P_o/P = f(level). If the curve marks criticality below the command position in a null power, the withdrawal must be stopped and then it has to be bored until critic concentration. On the other hand, if the curve shows a level which beyond the command level, place the rods in this command position and dilute until critical.

3.5.2 Modus Operandi

Given that there is a big amount of interesting phenomena produced in this experience, there are two variable lists *p5_1_1* and *p5_1_2* to carry out the 3.5.2.1 section and the *p5_2* list for the section 3.5.2.2 which contain the following variables:

- *Réactivité Doppler*
- *Réactivité modérateur*
- *Réactivité Xénon*
- *Réactivité Samarium*
- *Réactivité Bore*
- *Réactivité grappes*
- *Réactivité totale*
- *Vitesse de formation Xénon 135 totale*
- *Puissance thermique*
- *Puissance nucléaire*
- *Puissance électrique*
- *Concentration en bore du primaire (Cb)*

3.5.2.1 Reactor power reduction to half-load operation

The power plant regulation principle indicates that the reactor must follow the turbine power demand, which follows the electric net power demand. In the first experience, it has to be reduced the turbine load level so that the reactor reduces its power generated level by means of the auto rod control.

The operations which must be made for the power reduction and its power level maintenance up to the 50 % are described now.

1. Charge the *Standard 8* state by selecting . This state has critic the reactor, operating to nominal power in the half of its life.

2. Note in the 3.23 table the data referenced to the initial state of 100 % nominal power (*3868 MWth*).

3. Activate the simulation.

4. Locate the *Consigne Puissance* board in the *Turbine* screen.

5. Establish the maximum reduction rate of electric power *70 MWe/min* by means of *Pente de charge* selector. It is necessary to validate the action.

6. Indicate the desired final electric power of *675 MWe* approximately. It is necessary to validate the action. For this moment and beyond, it can be observed how the electric power is reduced in the fixed rate by means of the turbine steam admission valve progressive closing. It can be observed the control rod banks insertion until the command position in the reactors screen. The nuclear power reduction (and, therefore, the neutron flux reduction) makes the Xenon-135 appearance that will be the cause of the negative reactivity. It must be observed the rod reactivity and the feedback effects that try to compensate it.

7. Wait until the total reactivity is stabilized and write down the data in table 3.23. In this moment, the Xenon will continue to appear but it will be compensated by the *R* regulation control rod bank and by the temperature feedback. If there is no action made, it will be seen how the *R* bank withdrawal one step automatically and so on.

The Xenon effect must be compensated operating the chemical and volume control system for avoiding the *R* bank location out of range and for making the moderator temperature in the proximities of its command value.

8. There has to be diluted a *13,2 t/h* of H_2O flow rate with the objective to compensate the Xenon destabilizing effect. The neutron poison appearance (Xenon and in a lower way, Samarium) will be compensated with the soluble poison removal (boric acid). Select a big water volume which will let the continuous dilution (for example *100* m^3).

9. It is convenient to accelerate the simulation by 3 with the objective of minimizing time.

As the Xenon concentration increasing rate and the boron concentration reducing rate are values which depend on time, it is useful to watch the reactors development. If it is necessary, the make up flow of chemical system must be readapted.

10. Once an hour has past by since the reactivity stabilization, the 3.23 table must be completed. For the moment, what water volume has been introduced in the primary circuit?

11. The graphic representation must be printed.

The Xenon concentration will not arrive to its maximum until a few hours have pass from beginning of the load reduction. The boric acid removal has been compensating this effect until that moment. However, once the maximum Xenon effect passes by, its concentration will be reduced until reaching a value lower than the one reached with the reactor in nominal power. This is the reason for the chemical system to give more boron in the adequate rate so criticality is kept.

The axial flux core unbalance carried out by the partial rod insertion and the chemical system waste production justifies avoiding (as far as possible) the plant load variation.

Reactivity, in pcm	**100 % P_n**	**50 % P_n (ρ total stabilized)**	**50 % P_n (1 hour after the stabilization)**
Doppler			
Moderator			
Xenon			
Samarium			
Boron			
Rods			
Total Reactivity			

Table 3.23.- *Reactivity balance in the 50% power reduction.*

Related questions: 1-4

3.5.2.2 Reactor start up after shutdown

In this second section, a reactor shutdown will be carried out by operating until the half cycle of a 100% nominal power. The reactor will be let stabilized for an hour until reaching residual power and then, the reactor will be started up according to the following procedure.

1. Charge again the *Standard 8* state.
2. Note the reactivity balance in table 3.24.
3. The safety shutdown (*A.U.*) must be realized in manual mode from the core screen (*coeur*). It must be seen that all rod banks (safety and control) drop by gravity. In a minute, make notes about the reactivity balance in table 3.24.

4. Let the reactor stabilized for an hour approximately. The simulation ought to be accelerated by 3. The neutron flux reduction will carry out an increase in the Xenon concentration which will continue increasing all the time the reactor is shutdown.

5. You are in disposition to begin the reactor start up procedure after the waiting. Before this procedure begins, the emergency shutdown must be quit by pressing in 'acquit' (*acquittement*) in the lower part of the core screen.

6. Without carrying out any boration or dilution, withdraw the shutdown bank (*S*).

7. Withdraw the regulation control bank (*R*) until the start up position (*R=230*).

8. Withdraw the power control banks (*G*) until the command position for P_n 0 %. This command position is *N2* totally withdrawn, *N1* in 190 withdrawn steps, *G2* in 17 withdrawn steps and *G1 in 0* withdrawn steps.

9. At this point, several situations can be occurred.

   - If criticality is reached in a *G* position below command, a boration must be carried out until command value is reached.

   - If criticality is not reached in the command position, power control banks must be withdrawn a number of steps maximum equivalent to *500 pcm* value.

   If after the last operation it not appears to reach criticality, it must be procedure to dilute in maximum capacity (two letdown orifice) until criticality.

10. Take notes about the reactivities in table 3.24 for the critical state with non stabilized Xenon concentration.

Once criticality is obtained, it is important to carry out immediately the operations that drive to increase the flux level so that Xenon formed in the shutdown state is eliminated and stabilized in a lower value.

11. Withdraw slowly the control rods making supercritical the reactor and increasing its generated power.

Note: During the increase in power, the steam flow obtained in the generators output must be regulated in manual. The steam dump valve opening must be increased progressively in a way such as making the primary circuit temperature being near its command value.

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When Xenon-135 appearance velocity changes to the negative values, it will mean that the Xenon concentration is beginning to diminish. The positive reactivity contribution carried out by this phenomenon will make supercritic the reactor, helping it to increase until nominal power (*3868 MWth*). Only if every rod banks has been withdrawn and there still is a positive Xenon-135 generation velocity, there will have to be made a dilution by means of the chemical and volume control system.

12. Print the graphic representation.

Reactivity, in pcm	**100 % P_n**	**1 min after A.U.**	**1 hour after A.U.**	**Critic reactor** $\dot{\rho}_{Xe} > 0$	**Critic reactor** $\dot{\rho}_{Xe}$ compensated
Doppler					
Moderator					
Xenon					
Samarium					
Boron					
Rods					
Total react.					

Table 3.24.- *Reactivity balance during the safety shutdown and the later start up.*

Related questions: 5-10

3.5.3 Questions related with the experience

1. What is the role of the rods system and the chemical and volume control system (boron)?

2. Which control system compensates the Xenon reactivity effect?

3. Describe the reactivity variations that take place during the power reduction in section 3.5.2.1.

4. During the Xenon accumulation process in the first stabilization hour, what water volume has been introduced in the primary circuit? What is it made with the letdown water?

5. Why are there some procedures for the reactors start up from the hot shutdown?

6. What is the importance of the reactivity balances in the start up study?

7. Describe the reactivity variations and how does the criticality take place in section 3.5.2.2.

8. With the reactor at full power, how would the Xenon and Samarium evolve after a shutdown?

9. In a start up after a large in time shutdown, how much time will take to reach Xenon balance?

10. Can the reactor be started up after a one hour hot shutdown?

3.6 Reactor standard states. Transition from power operation to hot shutdown

3.6.1 Introduction

3.6.1.1 Reactor states

The reactor states correspond to the different operational situations in which the reactor and the power plant are found. In the power plant, there must exist systems that allow to make these transitions with the intention of reaching the suitable conditions that allow, according to the circumstances. Some operational situations are making tasks of maintenance or intervention, to carry out the fuel load, momentary or brief shutdown, operation to low power until operation to full power, etc.

In order to provide the reactor with the sufficient versatility to prepare, in the suitable value, its variables, the plant must have multiple systems that allow the reactor to change from one state to another.

The reactors operation with these aims is allowed to the conditions of the *Operating Manual* and of the *Performance Technical Specifications*. They constitute the operation procedure in nominal operation and accident conditions, and they define the limit values of the variables that affect the safety, as well as the actions limits of the automatic protection systems.

During the experience development, there are exposed the main restrictions of operation. It is necessary to pay attention to the vessel protection, the primary pumps, the pressurizer, the steam generators and the residual heat removal system, among others. The systems that receive special interest in the start up and shutdown are:

Primary coolant system: The primary system pressure and temperature must be such that allow to avoid the boiling point. The primary maximum temperature must be a *30 ºC* lower than the temperature of saturation at least in case of nominal operation and intermediate shutdown. The primary pumps are due to be protected against the cavitation.

Chemical and volume control system: During start up and the shutdown transitories, it is the system in charge to control and to fit the mass of water contained in the primary system, fitting the boric acid concentration of the water and the water level variations compensation of the pressurizer.

Residual heat removal system (reactor shutdown cooling): It continues evacuating the residual heat to keep the primary water in a low temperature. This system can not enter in service up to 4 hours after the shutdown of the reactor. The primary pressure must have reduced until a lower pressure down to *30 bars* (depressurized primary) and of *180ºC* temperature.

3.6.1.2 Reactivity effects calculation over core in primary system boration

The boron concentration variation, $\Delta C = C_1 - C_0$ in the core with a make up flow Q, can be calculated as the dilution flow (experience 2). Taking as reference the boron balance carried out in section 3.2.1.2:

$$Q_E\,C_E = Q_S\,C + V_0\frac{dC}{dt} \tag{3.14}$$

as $Q_E = Q_S = Q$ it is obtained:

$$Q(C - C_E) = -V_0\frac{dC}{dt}$$

Making a variable change $z = C - C_E$, it results:

$$V_{dis} = Q\cdot t = V_0 \ln\frac{C_E - C_0}{C_E - C_1} \tag{3.15}$$

being C_E the inlet concentration, that is, the concentration that there is in the tank.

It must be taken like hypothesis that during the dilution or boration processes, the injected water is mixed completely and immediately with the water of the primary system. These processes do not have their effect in an immediate moment but they take place with a delay of the effect on the core of the order of 5 minutes.

3.6.2 Modus Operandi

The variables to use in the experience development appear in several lists whose names are *p6_1_1* and *p6_1_2* for section 3.6.2.1 and *p6_2* for section 3.6.2.2. The variable ranges have been fitted so that there is a correct phenomena visualization.

The variables are:

- Nuclear power
- Electric power
- Pressure
- Average moderator temperature
- Total reactivity
- Boron concentration

- Secondary steam pressure
- Xenon reactivity
- Rods reactivity
- Flow of the residual heat removal system (RRA)
- Average temperature gradient
- Pressure gradient

The following operations are conducted taking into account the priority of the turbine over the reactor. Observe the reactors state evolution in the diagram of pressure and temperature control (*Diag. PT*). In this screen the reactors state progress can be observed along with the operational limits imposed by reasons for availability of systems, thermal hydraulics, subcooling maintenance, etc.

To avoid problems, it is convenient to save the present state every little period of time by pressing on . The saved state will appear in the *Instructeur* list.

Realize the following operations:

1. Load the *Standard 7* state corresponding to the critic reactor operating at full power (100 %) in the beginning of life (*DDV*).

2. Take notes about the initial state data in table 3.25 and start the simulation.

3.6.2.1 Hot stand by transition process

3. Load reduction of the turboalternator group until a 15% of the nominal power at a drop velocity of *50 MWe/min*. Thus, the turbine power of *1350 MWe* will be diminished to *202.5 MWe*. Carry out the reduction in a similar way as indicated in the section 3.5.2.1 of experience 5. It is advisable to accelerate by 3 the simulation in order to diminish the time. During this transitory, the rods automatic control slowly insertion of the control banks *N1*, *G2* and *G1*. Write down the state characteristics in table 3.25.

Note: Some alarms will appear:

- *Sortie groupe R de sa bande de manoeuvre* y *limite très basse insertion groupe R* because the *R* bank has been moved during the transitory.

- *RCV indisponible ou niveau ballon RCV anormal* because the 50% level of the control tank has been exceeded (caused by an increase in the primary water letdown).

4. Once it is reduced 15% the power, the turbine decoupling, in real time, must be done by pressing '*DECL. TURBINE*' located in the turbine screen. During the transitory, there can be a strong deviation in the water level in the generator activating the '*écart |N_{GV}-consigne| > 5%*' alarm. As soon as the steam generation is started again, the alarm deactivates. The electrical power production will be cancelled and the rod control will exchange to the manual operation as it is required in the low power operation. Automatically the steam that before was turbined turns aside now to the condenser, allowing this way the primary system coolant by means of the direct steam dump to the condenser. Write down the state characteristics in table 3.25.

	Initial (*St. 7*)	Step 3	Step 4 (**hot stand-by**)
characteristics	*Operation 100 % P_{elec} DDV*	*Operation 15 % P_{elec}*	*Decoupling turbine Steam dump act.*
Primary system			
Pressure (*bar*)			
Av. moderator temp. (*ºC*)			
Total reactivity (*pcm*)			
Secondary system			
SG pressure (*bar*)			
SG temperature (*ºC*)			
Steam flow (*t/h*) (GV1 x 4)			

Table 3.25.- *Variables evolution in hot stand by transition.*

The state achieved at this point corresponds to a **hot stand by** with a total negative reactivity in the order of *-60 pcm*. The antireactivity increase produced by Xenon can be observed. In this situation, the reactor could start up if it is not let exist the Xenon buildup that is produced in a several hours.

3.6.2.2 Transition process to hot shutdown

5. Introduce the power control banks *N1*, *N2*, *G1* and *G2* and the regulation *R* bank. The total reactivity in the reactor will diminish until about *-2700 pcm*. Note the data in table 3.26.

6. Keep the primary and secondary temperatures in the allowed ranges by means of controlling the pressure in the steam generator. This pressure is controlled by the opening of the valve for the steam dump to the condenser '*vanne contournement turbine*'.

Based on the time that the shutdown duration has been anticipated, the estimation of the amount of boron that is due to charge will be made. For this purpose, the Xenon kinetics constitutes the predominant factor. If it is presumed that the shutdown will be short, it will not be necessary to borate. It will appear Xenon, like neutron poison of the disintegration of its precursor Iodine-135, still present in the reactor. This Xenon concentration will contribute to diminish the reactivity. If, on the other hand, a long shutdown is anticipated (of the order of days) the total absence of neutron flow and the decay of the Iodine concentration will cause the total disappearance of Xenon. Therefore, it will be essential to compensate the antireactivity of the Xenon that exists immediately after the shutdown by Boron addition until a soluble poison concentration that allows a safe reactor state for a prolonged time.

7. Since it is not tried to return to start up the reactor, but to evolve later to cold shutdown, it is necessary to resort to the primary system boration by means of the chemical and volume control system (*RCV*). In this case, it will not only be tried to avoid the Xenon effect but also to compensate the effects of feedback by temperature until cold shutdown. Borate until obtaining a boron concentration around *1500 ppm* in primary water. In the chemical and volume control system screen (*RCV*), locate to the picture '*borication*' and select an approximately water volume of $24\ m^3$ of dissolution ($2{,}4 \times 10\ m^3$) charging a flow of *10 t/h*. In order to diminish the operation time, select '*bore x 50*'. If it is necessary to stop the boration process manually when there are reached the *1500 ppm* of concentration, turn to press on '*borication*'. Write down the state characteristics after the boration in table 3.26.

Note: Most probably, the control tank will be to half of its capacity, reason why a proportion of the letdown water (*10 t/h*, equivalent to the flow contributed from the borate water storage tank) will be turned aside to the effluent treatment system (*TEP*).

The imposed limits to keep the reactors safe operation indicate that the multiplication factor must be always lower than *0,99* or, in an equivalent way, the lower reactivity to *-1000 pcm*. Verify that this condition is fulfilled. If, on the other hand, this situation had not been satisfied it would be necessary to continue borating. The reached state corresponds to **hot shutdown**.

Visualize in the pressure-temperature diagram, the reactors small displacement as a result of diminution of the coolant average temperature caused by not having the necessity to evacuate a significant core power.

	Step 4 (**Hot stand by**)	Step 5	Step 7 (**Hot shutdown**)
characteristics	*Decoupling turbine Steam dump act.*	*Control banks inserted*	*Borated reactor*
Primary system			
Pressure (*bar*)			
Av. moderator temp. (ºC)			
Total reactivity (*pcm*)			
Secondary system			
SG pressure (*bar*)			
SG temperature (ºC)			
Steam flow (*t/h*) (GV1 x 4)			

Tabla 3.26.- *Variables evolution in the transition to hot shutdown.*

3.6.3 Questions related with the experience

1. What are the functions of the reactors states?
2. How does the reactor change one state to another?
3. How does the reactor heat remove in the several states?
4. Which are the protection systems of the reactor against coolant overpressures?
5. Indicate in which state is the reactor according the following conditions:
   - *State A*: P_n ***= 100 %***. Reactivity = ***0 pcm***. Primary pressure = ***155 bar***. Moderator Average Temperature = ***306,9 °C***
   - *State B*: P_n ***= 0 %***. Reactivity = ***-1 pcm***. Primary pressure = ***155 bar***. Moderator Average Temperature = ***297,2 °C***
6. Taking as a reference the boration carried out in point 7, calculate the boric acid concentration in the borated water storage tank by applying the equations (3.14) and (3.15).
7. What is the evolution in time that suffers the primary boron concentration during the boration/dilution processes?
8. Making the suitable hypotheses, between boration or dilution, which process obtains greater alteration in the reactivity in less time? Are there differences between making it at beginning of life or in end life?
9. Which is the Xenon concentration evolution during the whole process up to cold shutdown?

3.7 Reactor standard states. Transition from hot shutdown to cold shutdown

3.7.1 Modus Operandi

In the previous experience, the operation transition from power operation to hot shutdown was achieved, interrupted here by reasons of time. It will be begun in the initial scenario at the end of experience 6. The present experience tries to give continuity to the previous and to complement the necessary procedure until the cold shutdown.

The evolution of the reactor state must be observed in the diagram of pressure and temperature control (*Diag. PT*). In this screen, the reactors state progress can be observed along with the operational limits imposed by reasons of availability of systems, thermal hidraulics, subcooling maintenance, etc.

It is often convenient to save the present state by pressing  . The saved state will appear in the *Instructeur* list. There will be also available the periodically auto saved states. No variables have been prepared in this experience.

3.7.1.1 Transition to intermediate shutdown

Make the following operations:

1. Continue from the state of the last experience (*DDV*). It has been prepared the *instructeur 9 (parada caliente, inicio)* that corresponds to the 7 position in screen *Diagr PT*. Load this state and begin the simulation. Describe and characterize the reactor and plant initial states.

2. Take notes about the feedback effects (doppler and moderator) and the neutron poisons (Xenon and Samarium) in table 3.27.

3. The present boron concentration, of about *1500 ppm*, is sufficient to guarantee the subcriticality in case of the Xenon is completely disintegrated. However, if exceptionally the reactor has to transit up to cold shutdown to refuel or to have an intervention, the concentration must increase to *2500 ppm*. For that reason, borate up to the concentration of *2500 ± 50 ppm* by means of make up water. The borate water volume value must be *47,1* m^3 by a maximum flow of *10 t/h*. Watch the deflection from the extracted water flow towards the effluent treatment system (*TEP*). The accelerated mode '*bore*50*' can be selected. The tank stabilization will be waited because the boron unbalance in primary circuit.

It is necessary to overlap certain actions, as far as possible, to reduce the required total time in all the process. For example, a good strategy can consist in synchronizing the boration carried out with the primary circuits cooling that will take place next.

4. The Xenon profile in the *Xé-Sm* screen must be observed. Justify the axial asymmetry. Does its concentration grow or diminish in time? Justify the answer. A concentration monitoring of Xenon and Samarium must be made throughout the development of the experience.

The system is prepared to initiate its cooling. The primary circuit cooling and depressurization must be made under the limitations imposed by the components, structures and systems that take part in the intervention. These limitations can be:

- Approach to the water boiling conditions (reduction of the saturation range) in case of high temperatures for a given pressure.
- Presence of the cavitation destructive phenomenon in the primary circuit pumps. The centrifugal pumps must be operating avoiding the water local vaporization induced by a pressure hydrodynamic reduction avoiding consequences like possible erosion, noise, vibrations and loss of the hydrodynamics properties.
- Excessive pressure difference in the steam generator tubes.
- Unbalance in the pressurizer at low temperatures.
- Vessel damage induced by coolant high pressure related with its intermediate temperature and the materials fragility level.

The cooling of the primary system will be made now. The operation of preparation to cold shutdown is a process that can take hours. The main reasons that they lead to the slow execution of this process are the following ones:

- Cooling limitation up to *-28 ºC/hour*. By this way, a fast thermal shock will be avoided in the carbon steeled vessel, leaving the ductile zone and increasing its fragility and, therefore, diminishing its working life.
- The end of cooling in the primary system through the secondary circuit will involve the obtaining the necessary conditions for the placing in service of residual heat removal system (*RRA*). Since this system and its components have been projected to work in low pressure, they cannot operate until it has been obtained the conditions around *29 bars* and *170 ºC*. If these conditions had been obtained quickly, the

reactor residual power could be sufficiently great so that it could not to be evacuated by this system. In short, it must wait until the residual power generation level is the suitable one.

These are the following operations to carry out:

5. Open the three letdown orifices of the chemical and volume control system '*vanne aval orifice 1, 2, 3*' so that it allows to triple the make up and letdown flow to the primary circuit.

6. Since the water cooling entails as an associate consequence the contraction and reduction of its volume, a method of automatic water contribution to the control tank that feeds as well the primary system is due to have. In addition, this water must have a concentration of the most similar boron to the already existing one in the system, because otherwise it would be altered its concentration. This disadvantage is solved by activating the automatic contribution '*appoint auto*' in the screen *RCV*, which starts up when the tank diminishes below 18%. It feeds the tank with a mixture in the appropriate proportion on free-boron water and borate water.

7. The cooling must be made initially by steam dump to the condenser with the pressure control commanded by the pressurizer. In operation, open the deflection valve '*vanne contournement turbine*' approximately at 1% in manual mode. The pressure in the steam generator will begin to diminish, causing the vaporization of the secondary water and the cooling of the primary water. Observe the temperature gradient that takes place in the coolant. The temperature decrease rate control must be carried out by the drive of this valve. Often, the generator pressure stabilization makes diminish the cooling rate. For this reason, the valve opening is due to increase regularly. In order to diminish the simulation time, select the speed indicated like '*pseudo reel x 20*'. It is proposed to cool the primary water up to *280 ºC*.

Note: The pressure and primary temperature reduction causes the appearance, in the alarms screen, of the deflection of the operation conditions warnings. Specifically, during the cooling there will be the warning of the moderator temperature departure, since its reference is related exclusively to the operation nominal temperature.

8. In the pressurizer screen *Pressu,* commute to manual operation the command pressure and reduce it down to *145 bar*. It can be observed how the heaters are automatically disconnected and it is proceeded to spray water from the main sprinkling.

9. Continue with the cooling from the secondary system limiting it approximately to *-28 ºC/h*. If it is necessary, open additionally the bypass valve to increase the cooling rate. Make 2 or 3 pressure reductions like the indicated ones in the previous section, keeping the reactor state situation within the established operation ranges in diagram *PT*.

10. In this situation, as the core cooling requirements do not demand the water flow of nominal power operation, the operation is allowed only with one or two primary pumps. Therefore, when the average temperature is around *250 ºC,* two pumps of the primary circuit can be disconnected. For this reason, in the main screen '*Vue gén.*', select the primary pumps and to choose '*arret*' for two of them.

The cooling would have to be continued until entering in the zone limited by the maximum pressure and temperature *RRA*. In order to reduce the dedication time it is proposed to continue with the simulation from a state previously prepared.

3.7.1.2 Transition to cold shutdown

In this phase of the process, it is necessary to cancel some systems that are at the moment operating, giving the control towards others specially conceived to make this task in cold shutdown states.

11. Load the *instructeur 10 (intermediat shutdown, RRA)*, state where there are the necessary conditions to continue the cooling. Pay attention to the pressure and the primary temperature are *28,9 bar* and *175,3ºC*, which are appropriates for the start up of the residual heat removal system. The primary cooling is continued done by the steam generators, as the pressure control is commanded by the pressurizer. It must be reminded that it is convenient to start this phase in real time simulation.

Which is the main heat source in the primary system? And which if the reactor had operated for a long time in nominal power?

12. Take notes about the feedback effects (Doppler and moderator) and neutronic poisons (Xenon and Samarium) in table 3.27. Start the simulation.

13. The *S* safety bank must be totally inserted. Every single control rods banks will be totally inserted after this operation.

In this moment, the cooling and removal process of the thermal power must correspond to the residual heat removal system. Simultaneously, the primary pressure control and regulation must be yielded to the volume control system that will regulate it by means of the letdown flow. All actions that lead to make these commutations are described as follows.

14. Fit the pressurizer heaters power in *140 kW* approximately. This device has some electrical resistance to the primary pressure control that can be controlled from the board '*chaufferettes*'. The pressurizer consists of resistances with power fixed values (*1584 kW*). In order to disconnect them (if it is not already done) select the manual control and to press '*arret*'. It has also a group of variable power resistances that can reach *576 kW*. In order to establish its power down to *140 kW*, choose the manual control and select the wished value. It is necessary to validate.

15. In *RCV* screen, commute the letdown operation to *RCP* (*Circuit Primaire*) so that the letdown valve is regulated according to the commanded pressure for the primary circuit.

16. In the same screen, set a primary pressure of *29 bar* that will be tried to be approximated constant during the cooling phase remaining.

17. Since the conditions that had been reached allow the place in service of the residual heat removal system (*RRA*), connect it. Therefore, in the screen '*Bulle RRA*', open the valves that connect it to the primary circuit '*vanne RRA→primaire*' and '*vanne primaire→RRA*. Locate the valve that controls the cooled flow (*vanne échangeur RRA*) and open it, in manual mode, until 10 % (a higher opening would suppose an excessive cooling that would violate the maximum limit established in *-28 ºC/h*). Finally, start up the pump *pompe RRA* which will put into service the system and primary cooling.

The residual heat removal system allows to cool the primary circuit until reaching the cold shutdown. This circuit is connected as well to intermediate component cooling system through a heat exchanger (which is displayed in the simulator like boundary condition with a constant temperature of *30 ºC*). Finally, this system can be cooled by the last heat sink. The residual heat removal system has a deviation (bypass) to regulate the flow that goes through the exchangers. This allows to fit the primary cooling velocity, to avoid thermal fatigues in components. This deviation (by-pass) that allows to regulate the flow that goes through the exchanger can be placed in the simulator system scheme.

18. In the screen '*Bulle RRA*', find the valve that links the residual heat removal system with the volume control system letdown (*vanne de liaison RRA/RCV*). Open it until *100 %*. There must be seen the primary water letdown is carried out in the *RRA* system.

19. Open the valve '*soutirage excédenteaire*' until *100 %* in the same screen.

20. It is possible to cancel the secondary system because in this moment, there is no need to cool the primary system with the secondary one (*GV*). Fix the condenser bypass valve opening down to 0% (totally closed), in the steam generators screen. The feedwater pump must be stopped *TPA*, '*arret*' to cancel the steam generators feedwater. Without having reached the cold shutdown conditions, the secondary circuit has been isolated so that maintenance can be done in it.

In the present situation, the system is cooled by the *RRA* system, with pressure being controlled by the letdown of the volume control system.

21. The accelerated mode '*pseudo reel x 20*' is proposed for reducing simulation time during the cooling.

22. It can probably be seen the primary cooling rate has diminished strongly with time. Increase the valve opening '*vanne échangeur RRA*' for having it between the established ranges and always below *-28 ºC/h*.

23. Stop the simulation when the primary temperature is approximately *70 ºC*. Lastly, the state achieved by the reactor is the cold shutdown. Take notes about the feedback effects (Doppler and moderator) and neutron poisons (Xenon and Samarium) in table 3.27. Calculate the average temperature coefficients in the 3 states involved in the experience and compare them to what the simulators offers.

Reactivity *(pcm)*	**Hot shutdown** *P =* *T =*	**Intermediate shutdown** (RRA placed in service) *P =* *T =*	**Cold shutdown** *P =* *T =*
Doppler			
Moderator			
Xenon			
Samarium			

Table 3.27.- *Monitoring of temperature coefficients and neutron poisons (Xe and Sm).*

3.7.2 Questions related with the experience

1. How the vessel operation in ductile zone is controlled?
2. How are steel fatigues of the vessel controlled during cooling or heating? Which are the conditions that limit the vessel cooling?
3. Explain and describe briefly the hot and cold shutdown states. How the subcriticality is secured in the shutdown states?
4. Indicate in which state is the reactor placed for the following conditions:
    - *State A*: $\boldsymbol{P_n = 0\ \%}$. Reactivity = ***- 25000 pcm***. Primary pressure = ***15 bar***. Average moderator temperature = ***50,0 ^{o}C***
    - *State B*: $\boldsymbol{P_n = 0\ \%}$. Reactivity = ***- 5000 pcm***. Primary pressure = ***100 bar***. Average moderator temperature = ***250,0 ^{o}C***
5. Why is necessary for other systems to intervene in transition to cold shutdown? Which are those systems?

6. How does the core heat removal secured in cold and hot shutdowns? Which is the heat origin? And what if the shutdown is made in end of life?

7. Which is the amount of water added to the primary circuit during its cooling? Carry out the calculation considering the density variation and supposing a primary circuit volume of $280\ m^3$.

8. How is the residual heat removal system constituted?

9. Why does the heat exchanger efficiency of the residual heat removal system decrease during the cooling?

10. Calculate, by table 3.27, the average temperature coefficients for the hot, intermediate and cold states and compare them with the ones offered by the simulator.

11. Justify the Xenon and Samarium time evolution seen during the experience. What is the Xenon axial distribution? Justify the answer.

3.8 Control rod bank calibration

3.8.1 Introduction

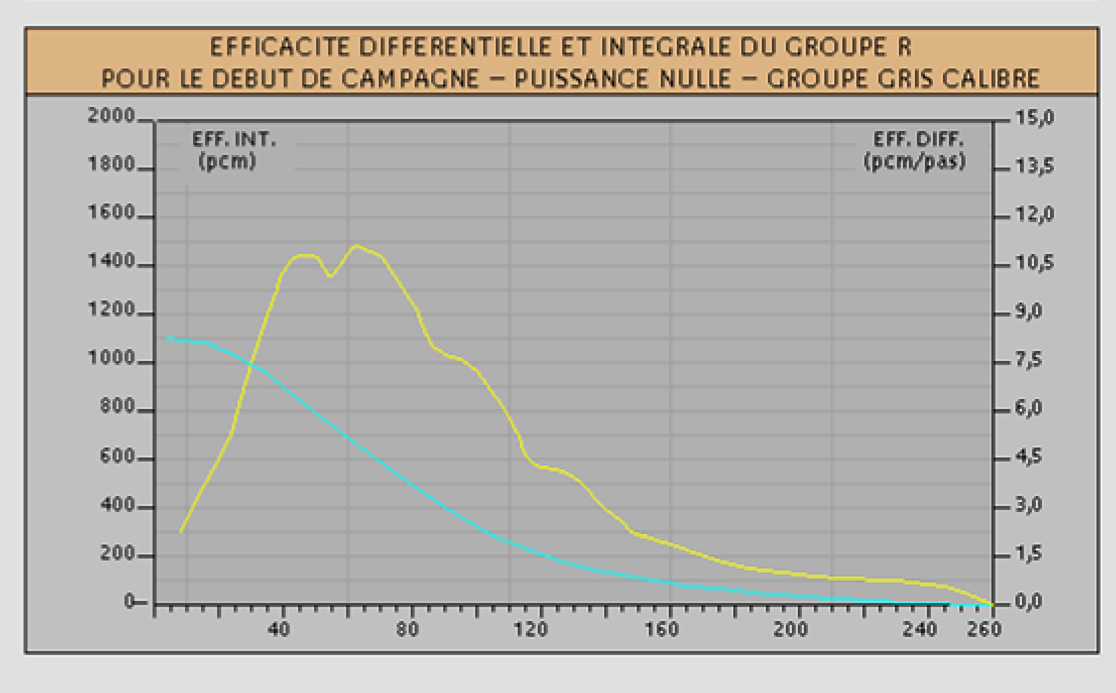
The reactors operation needs knowledge in a big amount of behaviour characteristics. There are important well-known effects to compensate during the fuel life, such as temperature effects, Xenon effects, fuel consumption, etc. They all carry out reactivity variations.

It is necessary to anticipate in the beginning of life some amount of core reactivity because the fuel burn up. This reactivity excess must be compensated with the control rods inserted and the boric acid dilution. It is necessary to know accurately the control rod banks effectiveness and the differential boric efficiency for the reactor control.

Particularly, in order to implement an effective reactor control, it is necessary to know the rods reactivity based on its insertion path in the reactor. The control rod worth is known theoretically by means of theoretical absorption and distribution flux calculations. However, a periodic and direct verification by experimental measures is essential if it is tried to know its worth with greater exactitude. Figures 3.6 and 3.7 show the effectiveness of the regulation power bank and control banks with the overlapping of the last ones.

The rods differential worth is defined as the rod value in a certain position within the core in terms of reactivity by length unit. Generally, in the core edge this differential worth is lower than in the centre. The differential worth is not constant because of the flux axial distribution. On the other hand, the integral worth corresponds with the total rod worth in its present position.

This whole values change within the fuel life. It has to be noticed that in the simulator it is not possible to calibrate a single rod, they all move together, but it must be done by single banks.



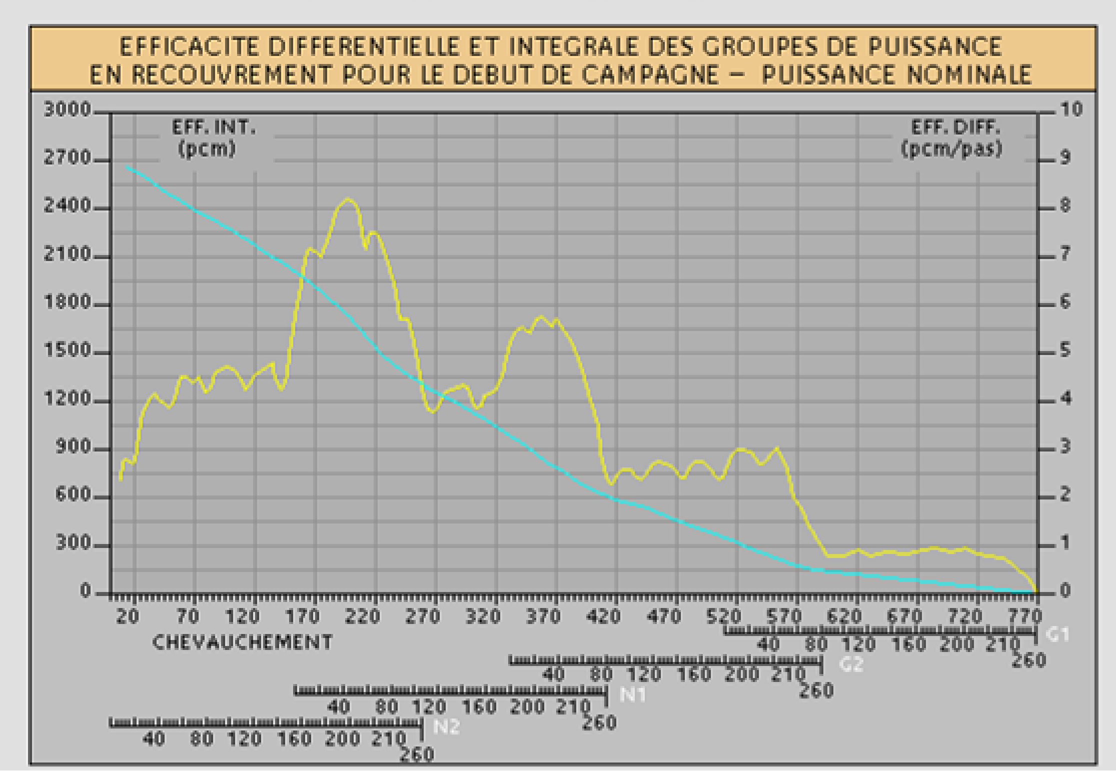


Figure 3.7.- *Differential and integral worth of power control banks with overlapping in null power*.

It is also advisable to know how the boric acid affects the reactors behaviour. The differential efficiency value that the simulator makes use of is showed in figure 3.8.

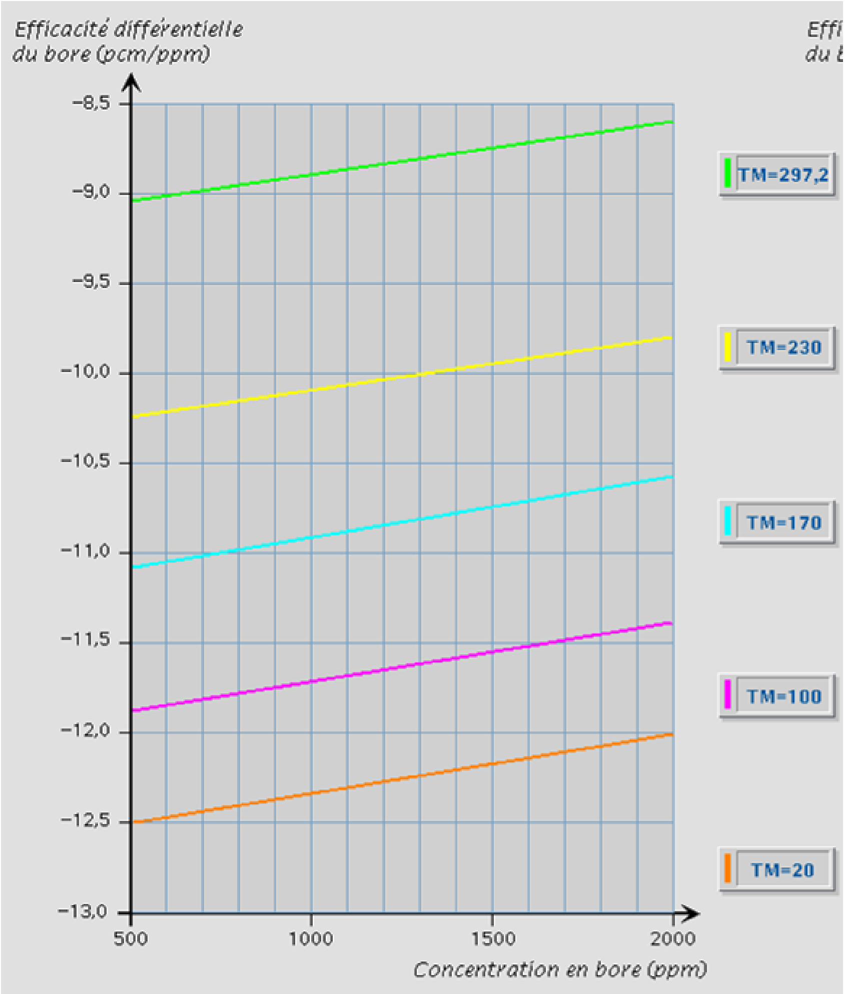


Figure 3.8.- *Boric acid differential efficiency in null power related with the boron concentration for some average core temperatures (DDV).*

3.8.2 Modus Operandi

Special interesting variables for section 3.8.2.1 are included in the variables list *p8_1*.

- *Flux chaîne puissance*
- *Temps de doublement*
- *Réactivité bore*
- *Réactivité grappes*

- *Réactivité totale*
- *Débit eau REA*
- *Concentration en bore du primaire (Cb)*
- *Position barres R*

It has not been planned the use of variables nor evolution registry in section 3.8.2.2.

3.8.2.1 *R* control bank calibration by dilution

The process consists of making a dilution starting off the reactor with a sufficient negative reactivity (very low nuclear power) compensating the dilution effect by means of antireactivity introduction of by rod insertion.

1. Load the *Standard 4* state that has a critic reactor in *BOL* with a nuclear power of *50 kW*. Nuclear test conditions. Take notes in the initial boron concentration.

2. Activate the simulation.

Preliminary actions:

3. Open an additional second letdown orifice in the chemical and volume control system screen (*RCV*).

4. Make a boration of *0,5* m^3 of boric acid with an added rate of *10 t/h* from *RCV*. This boration makes the reactor subcritical, before the calibrating bank withdrawal, as target. Furthermore, the boration tries to locate the reactor in an adecuate subcritical interval for the following procedure. The total reactivity must approximately reach *-127 pcm”*. ‘*Bore*50*’ simulation speed is advisable to be load (only in this point). Take notes about the boron concentration.

5. Withdraw totally *R* control bank (step 260) in manual operation before the calibration is started up. The power control banks will continue in their command position in null power. The total reactivity must approximately reach *-100 pcm*.

Note: There will appear the warnings in the alarms screen when *R* bank is withdrawn over their higher range.

Calibration procedure:

6. Carry out a dilution with a *20 t/h* rate. Choose sufficient water volume so that the experience can be realized, for example, *100* m^3. It is proposed to begin with real time, and when it is appropriated considered, to accelerate by 3 the simulation.

7. Next, insert the *R* bank 20 steps. Freeze the simulation when the insertion is finished. It can be supposed negligible the action and variation of the boron concentration during the time that has lasted the bank insertion. Determine the introduced reactivity by the insertion bank by means of the cursor displacement and complete the table 3.28.

Note: It is not necessary to insert the exact number of steps proposed. An approximated approach is sufficient.

8. Restart the simulation and repeat the *R* bank insertion until the total group is inserted (step 0). The total reactivity must be in the *-100 pcm* and *-200 pcm* range approximately during the whole process. This target can be solved by having less time between the consecutives rod insertions. Wait until the boration effects if it is necessary to increase the reactivity.

Note: The first insertions can be 20 steps length until reaching step 160. From this moment, it is advisable to reduce them down to 10 steps so a detailed influence study in the lower core zone can be carried out.

9. When the insertion is finished, print the graphic representation. Take notes in the final boron concentration.

Determine the boric acid differential efficiency with the injected reactivity and the boron concentration variation.

Related questions: 8-10

Position R (*steps*)	260	240							
$\Delta\rho_{salto}$ **(*pcm*)**	-								
Δz **(*steps with.*)**	-								
$\Delta\rho_{salto} / \Delta z$ **(*pcm/step*)**	-								
$\sum\Delta\rho_{salto}$ **(*pcm*)**	-								
Position R (*steps*)									
$\Delta\rho_{salto}$ **(*pcm*)**									
Δz **(*steps with.*)**									
$\Delta\rho_{salto} / \Delta z$ **(*pcm/step*)**									
$\sum\Delta\rho_{salto}$ **(*pcm*)**									
Position R (*steps*)									
$\Delta\rho_{salto}$ **(*pcm*)**									
Δz **(*steps with.*)**									
$\Delta\rho_{salto} / \Delta z$ **(*pcm/step*)**									
$\sum\Delta\rho_{salto}$ **(*pcm*)**									

Table 3.28.- *Bank R differential and integral worth data.*

3.8.2.2 Power control bank calibration by exchange

It is possible to determine a bank integral efficiency by comparison of another known bank efficiency that has been calibrated by dilution. The principle is the following one (showed in figure 3.9): consider an *X* bank whose efficiency is known and a *Y* bank whose efficiency is wanted to be known. Initially, *X* bank is totally inserted in the core but the *Y* bank is totally withdrawn making the reactor critic (position 1). Then, *Y* bank is totally inserted while *X* bank is partially withdrawn so reactivity is compensated (position 2). Later, the boron concentration must be adjusted by boration with the target of withdraw the *X* bank, always having the reactor critic (position 3). The *X* and *Y* banks change again positions so *Y* banks is turn to be totally withdrawn. The reactor must have criticality with bank *X* with N steps withdrawn (position 4). The total integral efficiency of *Y* bank will be the equivalent to *X* bank with N steps withdrawn.

If what is pretended is to obtain the differential efficiency until a certain step, the *Y* bank must be withdrawn (position 4) until the step the integral efficiency is wanted.

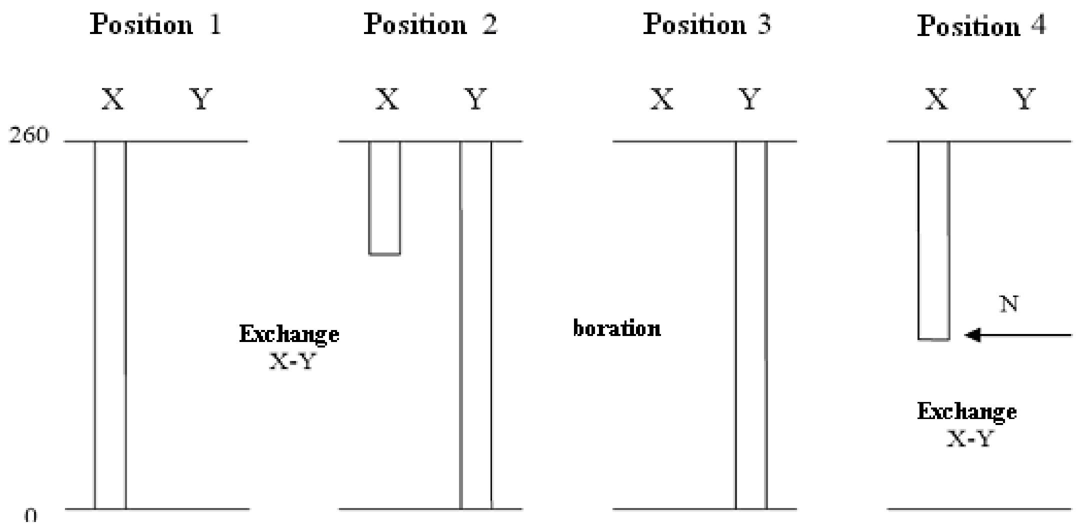


Figure 3.9.- *Bank Y integral worth (X known) schematic diagram.*

In this second part of the experience, the power control bank integral efficiency will be tried to consider by bank *R*. See the *N1*, *N2*, *G1* and *G2* bank overlapping so to try to know their worth in an independent way would be impossible.

The procedure in this experience will be slightly different. The initial situation will be equivalent to position 3. But, in the exchange moment, it will be not inserted *Y* bank but *X* calibrated bank so that constant reactivity variations will be introduced. The operations are:

1. Load the *Instructeur_11* state, being this state very similar to *Standard 4*. See the power control banks *N1*, *N2*, *G1* and *G2* are inserted and the *R* bank is totally withdrawn (position 3 of the figure 3.9).

2. Activate the simulation.

3. Insert the *R* regulation bank so *200 pcm* of antireactivity are introduce to the system. Use as a reference the indicated value in variable '*réactivité groupe R*'. Withdraw the *N2* bank until the reactor reaches criticality. In this new position, the *N2* bank integral efficiency will be *200 pcm*. Take notes on table 3.29. ***You must always only write down the compensated state.***

4. Continue inserting the equivalent to antireactivity *200 pcm* with *R* bank and withdraw the power banks until *R* bank is totally inserted.

Borating the primary circuit and returning to the regulation bank start point will be necessary for calibrating the grey rods banks.

5. Open a second letdown orifice of the chemical and volume control system.

6. Borate the system to compensate the forward *R* bank withdrawal. The antireactivity controlled by *R* bank appears in the reactivity balance screen '*Bilan réactivité*'. From expression (3.15) and preceding data, obtain an approximation of boron volume to be introduced in primary system. Select the maximum flow allowed of *10 t/h*. Modify also the simulation speed by choosing '*Bore*50*'.

7. Continue with the *R* bank withdrawal. During this withdrawal there can be two possible situations:

    - The *R* bank can be totally withdrawn without reaching criticality. In this case, withdraw totally the *R* bank and compensate the negative reactivity with the power banks withdrawal until reaching criticality. The compensated reactivity will count in the power banks integral efficiency.

    - If the criticality is happened to be before the *R* bank total withdrawal, this withdrawal must be stopped when criticality is reached. The change to be realized will be to change the reference value of the reactivity controlled by *R* bank to the present value.

8. Return to section 3 until *R* bank is totally inserted.

9. The process will be finished when the four banks are totally inserted.

Control banks position (steps)					Boron Conc. (*ppm*)	React. R (*pcm*)	Integral Efficiency (*pcm*)
N2	N1	G2	G1	R			
0	0	0	0	260		0	0

Table 3.29.- *Power banks integral efficiency calibration data.*

Related questions: 11-12

3.8.3 Questions related with the experience

1. Why rods are calibrated?
2. How rods must be calibrated? How rod bank can be calibrated by one already calibrated?
3. What are the differences in composition between grey and black rods? What are their roles?
4. What are the parameters that are related to the banks efficiency?
5. How does the boron efficiency (*pcm/ppm*) change in relation with the moderator boron concentration C_B?
6. How does the boron efficiency (*pcm/ppm*) change related with temperature for a constant boron concentration?
7. How does the flux evolve if in nominal power a step from a totally withdrawn rod is introduced? And what if the rod is in the core centre?
8. Draw the integral and differential graphic representation of the *R* bank calibration by data collected from table 3.28.
9. What is the value of the estimated boron differential efficiency? Compare the obtained value with the one given by the simulator. Does this value depend on instant during the fuel life?
10. Can the rods be calibrated by boration? How would it be made?
11. What is the reason for the regulation rods overlapping movement?
12. Draw the integral calibration curve of power control banks by table 3.29 data.

3.9 Reactor stabilization

3.9.1 Introduction

The reactor stabilizing characteristics vs power variations when it operates at full power have been studied in the past experiences. The autostabilization effectiveness will depend on the temperature feedback coefficients and its relative value. These coefficients will change their relation during the fuel life and are directly related with the amount of boron in the primary circuit.

New stabilizations experiences with the reactor operating in half of its nominal power will be carried out during this practice. The differences between the perturbations introduced or by the primary circuit or by the second one will be emphasized.

Two perturbations types can happen during the reactor operation:

- Primary perturbations corresponding to a reactivity insertion or extraction.
- Secondary perturbations, for example, load fast change in ± *10 %*.

In this experience, it will be differentiate between both situations. Furthermore, there will be several experiences in different fuel life states. In sections 3.9.2.1 to 3.9.2.4, there will be a study on the different secondary and primary perturbations consequences over the plant and the differences between beginning of life (*BOL/DDV*) and middle of life (*MOL/MDV*).

A negative moderator feedback coefficient α_M must be always kept during the reactor operation. This is the reason why is important the knowledge of the boron limit concentration and never increase it. Moreover, the fuel effect is always negative. For this reason, two other situations with hypothetical operation conditions will be studied.

From safety and control point of view the most important role is played by the fuel coefficient α_F that is always negative. Its relevancy is made by the quick effect in external perturbations. In section 3.9.2.5, an experience where this effect is eliminated $(\alpha_F = 0)$ and there are only the moderator effect in reactor autostabilization, will take place.

In section 3.9.2.6, it will carry out an experience in a state with a boron concentration higher than critic finding a positive feedback moderator coefficient α_M. The reactor will not have autostabilization characteristic; it will not be able to achieve a balanced power and thermal state, depending on the new moderator coefficient value (if this coefficient is higher than the Doppler coefficient).

3.9.2 Modus Operandi

The values in lists *p9_1*, *p9_2*, *p9_3*, *p9_4*, *p9_5* y *p9_6* will be used for the sections 3.9.2.1 to 3.9.2.6 in this experience. These variables are: fuel temperature, Doppler reactivity, moderator average temperature, moderator reactivity, total reactivity, nuclear power, electric power, steam generators pressure, rods reactivity and steam dump flow to the condenser.

3.9.2.1 Primary perturbation in the beginning of life (BOL)

Follow the following operations:

1. Load the *Standard 12* state. In this state, the reactor is in 50% nominal power, steady-state (critic) and in the beginning of life.

2. Activate the simulation and complete the tables 3.30 and 3.31 in the initial state.

3. Convert the rod control system to manual mode for observe the reactor natural behaviour. Select ‘*manu*’ for the *G* and *R* banks in the ‘*coeur*’ screen.

4. Withdraw in manual mode 33 steps the *G2* bank. This withdrawn will approximately end in a *100 pcm* reactivity insertion because each step is roughly equivalent to *3 pcm*.

5. Once the reactor reaches stabilization, complete the tables 3.30 and 3.31.

6. Print the graphic representation.

***State**, P_n*	T_M (°C)	T_F (°C)	ρ_M (pcm)	ρ_F (pcm)
Initial, ***50 %***				
Final				
Balance				

Table 3.30.- *Temperature feedback effects in DDV caused by primary perturbation.*

Reactivity Control	**Initial**	**Final**
C_b (ppm, pcm)		
R (steps, pcm)		
G1 (steps, pcm) ***G2 (steps, pcm)***		
N1 (steps, pcm) ***N2 (steps, pcm)***		
S (steps, pcm)		
TOTAL RODS (pcm)		

Table 3.31.- *Boron and rods reactivity control systems (banks classification).*

3.9.2.2 Primary perturbation in the middle of life (MOL)

Repeat the same procedure as in section 3.9.2.1 loading the *Standard 5* state that has the reactor operating in *50 %* of nominal power, steady-state (critic) and in the middle of life. Complete tables 3.32 and 3.33. Print the graphic representation.

***State**, P_n*	T_M (°C)	T_F (°C)	ρ_M (pcm)	ρ_F (pcm)
Initial, ***50 %***				
Final				
Balance				

Table 3.32.- *Temperature feedback effects in MDV by primary perturbation.*

Reactivity Control	Initial	Final
C_b (ppm, pcm)		
R (steps, pcm)		
G1 (steps, pcm) ***G2 (steps, pcm)***		
N1 (steps, pcm) ***N2 (steps, pcm)***		
S (steps, pcm)		
TOTAL RODS (pcm)		

Table 3.33.- *Boron and rods reactivity control systems (banks classification).*

Related questions: 1-7

3.9.2.3 Secondary perturbation in beginning of life (BOL)

Follow these operations:

1. Turn to load the *Standard 12* state.

2. Activate the simulation. Complete the table 3.34 in reference to the initial state.

Introduce a load *+10 %* variation of electric power (increase) in the secondary side (turbine). This perturbation will be made with the rod control system in manual mode with the objective of study the reactors natural behaviour. For example, *677 MWe* to *745 MWe* increase in electric power with a *40 MWe/min* increasing rate is proposed. Follow these operations:

3. Commute to manual mode the *G* and *R* rods control system.

4. Locate the '*consigne puissance*' board in the '*turbine*' screen. Select the variation rate in *40 MWe/min* and the final power in *745 MWe*. It is necessary to validate in both operations.

5. Once the reactor reaches stabilization, complete table 3.34.

6. Print the graphic representation.

***State**, P_n*	T_M (°C)	T_F (°C)	ρ_M (pcm)	ρ_F (pcm)
Initial, ***50 %***				
Final				
Balance				

Table 3.34.- *Temperature feedback effects in DDV by secondary perturbation.*

3.9.2.4 Secondary perturbation in middle of life (MOL)

Follow the same operations as in section 3.9.2.3 loading the *Standard 5* state which has the reactor operating in *50 %* of nominal power, steady-state (critic) and in the middle of life. Complete the table 3.35 and print the graphic representation.

***State**, P_n*	T_M (°C)	T_F (°C)	ρ_M (pcm)	ρ_F (pcm)
Initial, ***50 %***				
Final				
Balance				

Table 3.35.- *Temperature feedback effects in MDV by secondary perturbation.*

Related questions: 8-13

3.9.2.5 Doppler effect significance. Hypothetical operation without Doppler effect

The reactors operation with a null fuel coefficient is not possible because the fuel coefficient of *LWR* reactors (composition, enrichment, geometry) is always negative (Doppler effect). Nevertheless, a new state will be created with the reactor operating in *50 %* P_n critic and steady-state. The Doppler effect action is inhibited (null α_F coefficient) while moderators coefficient α_M is kept negative.

1. Load the *Standard 5* state. In this state, the reactor is in 50% nominal power, steady-state (critic) and in the middle of life.

2. Activate the simulation. Complete table 3.36 in reference to the initial state.

3. Select the  icon (local commands), that is located in the upper section of the instructor screen. Then, inside the menu *Choix*, choose *coeur* $\rightarrow$ *Suppression de*

l'effet doppler. After, activate it by pressing '*non*' $\rightarrow$ '*oui'* and finally, load it by pressing *armement.*

4. Commute the rods control system in manual mode so that it is possible to watch the reactors natural behaviour. Select '*manu*' for the banks *G* and *R* in the '*coeur*' screen.

5. Withdraw in manual mode 33 steps of bank *G2*. This withdrawal will end in a *100 pcm* reactivity insertion because each step is roughly equivalent to *3 pcm* approximately.

6. Once the reactor reaches stabilization, complete table 3.36.

7. Print the graphic representation.

***State**, P_n*	T_M (°C)	T_F (°C)	ρ_M (pcm)	ρ_F (pcm)
Initial, ***50 %***				
Final				
Balance				

Table 3.36.- *Temperature feedback effects in MDV by primary perturbation (without Doppler).*

Repeat the same procedure operating on the turbine in the 10% power increase instead of modifying the rods positions. Remember keeping the control banks in manual mode. Complete table 3.37 and print the obtained graphic.

***State**, P_n*	T_M (°C)	T_F (°C)	ρ_M (pcm)	ρ_F (pcm)
Initial, ***50 %***				
Final				
Balance				

Table 3.37.- *Temperature feedback effects in MDV by secondary perturbation (without Doppler).*

Related questions: 14-16

3.9.2.6 Moderator coefficient effect significance. Hypothetical operation with high boron concentration

The reactor operation with a positive moderator coefficient is prohibited, as it is considered in the plant design and as it is established in the performance technical specifications. So, the operation will always be below a limit boron concentration.

However, a new reactor state with low power, critic and steady-state will be created. Its high boron concentration will allow a α_M positive coefficient. In the other hand, the α_F coefficient will be negative.

Follow these operations:

1. Load the *instructeur 12* state which corresponds to a reactors operation of *3 MW* nuclear power and *37 MW* thermal power, beginning of life with the highest fuel reactivity allowed and high boron concentration for its compensation (*1435 ppm*). Observe that rods control is operating in manual mode.

2. Activate the simulation. Complete tables 3.38 and 3.39 referenced to the initial state.

3. Modify the average moderator temperature from the secondary circuit. It is proposed to do it by a steam dump to the condenser. Select the deflection valve '*vanne de contournement turbine*' in the steam generator screen. Commute its operation to manual mode and open it until 40 %. It is necessary to validate.

4. Stop the simulation in a minute since the valve opening and take notes of the reactivity balance in tables 3.38 and 3.39.

5. Restart the simulation. Keep the valve open during two more minutes and finally close it and return to auto mode.

6. Stop the simulation and take notes of the results in tables 3.38 and 3.39 in the bypass valve closing moment.

7. Print the graphic representation.

Note: The preceding transitory carries out the alarms appearance, because the important variables excursions in pressurizer and steam generators as well as the moderator temperature decrease.

State	T_M (°C)	T_F (°C)	ρ_M (pcm)	ρ_F (pcm)
Initial (1)				
1 min later (2)				
Balance (2) - (1)				
3 min later (3)				
Balance (3) – (2)				

Table 3.38.- *Temperature feedback effects in DDV by secondary perturbation (α_M positive).*

Reactivity Control	Initial	1 min later	3 min later
Fuel (pcm)			
Doppler (pcm)			
Moderator (pcm)			
Xenon (pcm)			
Samarium (pcm)			
Boron (ppm, pcm)			
Rods (pcm)			

Table 3.39.- *Reactivity control systems (fuel, Doppler and moderator, poisons, boron and rods).*

Related questions: 17-20

3.9.3 Questions related with the experience

1. What is the autostabilization?
2. If reactivity is changed from the primary system, how would the reactors power evolve?
3. How do the fuel and moderator temperatures evolve?
4. Which is the first feedback effect to appear? Why?
5. How do the feedback effects appear during fuel life?
6. In the middle of life, does the reactor take more or less in stabilizating? Why?

7. Determine the moderator and fuel average feedback coefficients in the studied range.

8. If the secondary load is change abruptly, for example in *± 10%*, how does the reactor power evolve?

9. How do the fuel and moderator temperatures evolve?

10. Which is the first temperature feedback effect to appear? Why?

11. Determine the moderator and fuel average feedback coefficients in the studied range.

12. Which are the reactor limits for the transitories with changing load?

13. What occurs if the limits are exceeded? Which is the condenser steam dump function and capacity? And the steam generators safety valves?

14. How do the moderator temperature and the power evolve in both situations?

15. Is the compensation immediately after the perturbation?

16. Which are the differences and the shape if the experience was in the beginning of life?

17. How do the moderator temperature and the moderator antireactivity evolve?

18. Does the reactor power increase or decrease during the moderators cooling? Which is the more important coefficient (moderator or fuel) in the experimental conditions?

19. Calculate the fuel and moderator average feedback coefficients in both studied intervals. Compare them to the ones in figure 3.3.

20. Why does the power diminish after a several time, making itself subcritic?

3.10 Disconnection from electrical grid and house load operation

3.10.1 Introduction

Special attention will be paid to the operation diagram, axial power unbalance control and turbine regulation during this experience. This attention will be during the house load operation transitory. On the other hand, reactor shutdown margin will be tried to be considered.

It will carry out the power transitory named *house load operation* in first part of experience. The house load operation (‘*ilotage*’) consist in reduce the plant generated power until a level slightly higher than necessary to power supply the auxiliary systems and reactor regulations automatic controls. The house load operation is the most severe operation transitory for the plant, taking part in the electrical grid failures. This operation main advantage is to allow the power fast recovery, once solved the possible electric problem without being influenced negatively by the poison Xenon-135 in the reactor. An event which leads to the hot shutdown would cause the increase of neutron poisons that could disable the start up until their disintegration.

Later, after the house load operation, it will come to the turboalternator connection to the electrical grid and to the power increase until 100 %. Special interest to the operation diagram, power coefficient of reactivity and shutdown margin will be needed.

Next, some basic aspects for the experience development will be defined.

3.10.1.1 Power coefficient of reactivity

The temperature effects (moderator and fuel) are important for the light water reactors, as seen before. They are translated to power effects so that a balance on these effects is possible. The power coefficient of reactivity is defined as the coefficient of the temperature reactivity variation divided by the power variation:

$$\alpha_P = \frac{\Delta\rho\ (doppler + moderador)}{\Delta P} \qquad \left(pcm / \%P_n\right) \tag{3.16}$$

During the fuel life, the moderator coefficient evolves importantly, as seen in lasts experiences. Therefore, the power coefficient changes according the power level and the reactors state (for a same power state, in the end of life, the power coefficient is more important than in beginning of life).

3.10.1.2 Operation diagram limits

Neither the hot spot factor, nor the lineal power are directly available in a reactor. However, it is necessary to search for a conversion between these variables and the power and axial offset. This conversion is needed for operation ranges representation over a diagram called *operation diagram*.

The available variables to that purpose are the ionization chambers currents I_H and I_L that are located in the core upper side and core lower side (figure 3.10). The axial offset is written as:

$$AO = \frac{I_H - I_L}{I_H + I_L} = \frac{P_H - P_L}{P_H + P_L} \tag{3.17}$$

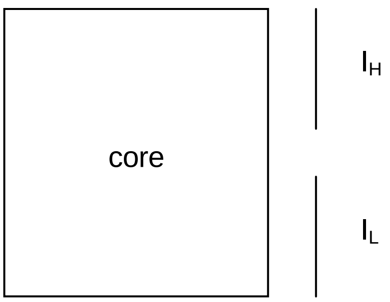


Figure 3.10.- *Power drawing in the core upper and lower side.*

Experimentally, a correlation between the maximum power peak F_Q^T and the axial offset has been established in several operation situations (different rods steps, different boron concentrations, several Xenon distributions). The different obtained dots are represented in a diagram called '*flyspeck*' included in figure 3.11.

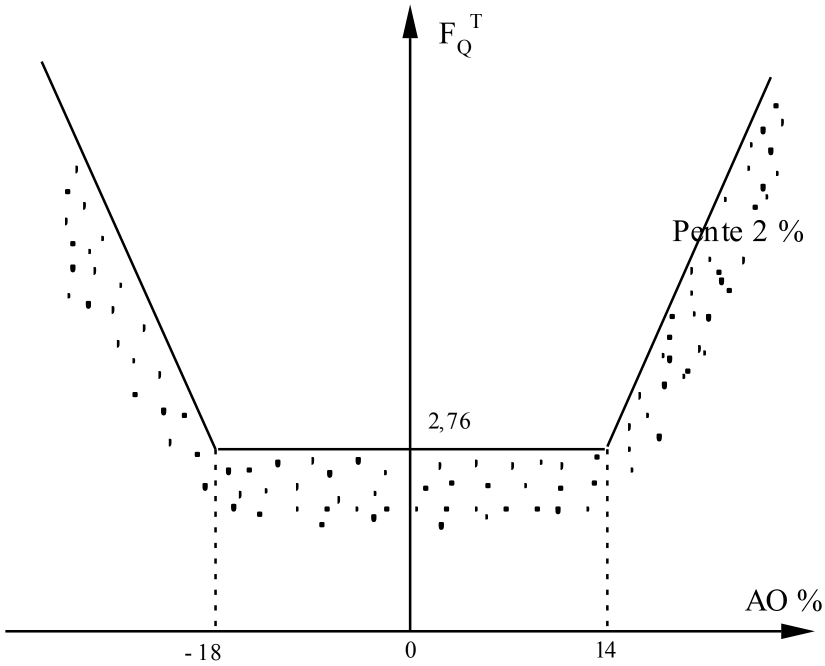


Figure 3.11.- *Typical 'flyspeck' diagram by Westinghouse (CPY 900 MW example).*

A curve over these dots can be drawn, composed by a horizontal part $F_Q^T = 2,76$ (between the axial offset values -18 % and 14 %) and in two straight lines with an equal slope of 2 % in increase of F_Q^T by % of axial offset increase.

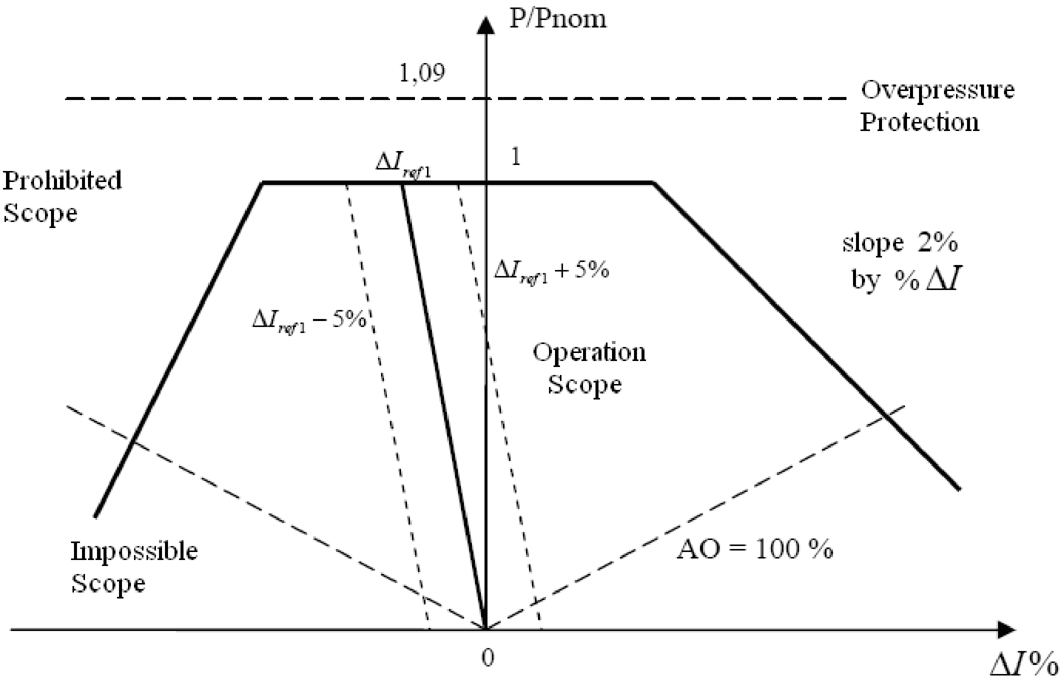


Figura 3.12.- *Operating conditions obtained by the 'flyspeck'* diagram.

However, the *normalized axial offset* is normally used as a reference variable. It is defined as in equation (3.18) in which the reactors nominal power is considered.

$$AON \equiv \Delta I = \frac{P_H - P_L}{\left(P_H + P_B\right)_{P_{nom}}} = \frac{P_H - P_L}{P_{nom}} \tag{3.18}$$

The shape, seen in figure 3.12., can be easily demonstrated considering the nominal power and the power during operation.

The power redistribution effect obligates to operate with a slight controlled offset in the core lower side. Taking into account the relation:

$$\Delta I = \frac{P}{P_{nom}} AO \tag{3.19}$$

it means that operation with $AO = cte$, the vertical straight line will become an oblique straight line that will pass through the origin (if $P = P_{nom} \rightarrow \Delta I = AO$, if $P = 0 \rightarrow \Delta I = 0$). On the other hand, the axial offset will change between *+100%* and *-100%*. In the extreme situation of $AO = 100\%$ it is obtained $\Delta I = P / P_{nom}$ that matches with the straight line of *45º* with x axis (figure 3.12). Therefore, after the transformation, the lower space is physically inaccessible.

The Xenon transitories are used to have big periods (hours) so operating with a slight deviation in the axial power difference is allowed. Furthermore, the operating conditions can be divided in different sections with time limits so that important Xenon transitions are avoided.

Particularly, for *REP 1300* reactor, a concrete and strict ΔI limit does not exist so figure 3.12 trapeze is suppressed. However, the operation is limited in the right side so Xenon oscillations (figure 3.13.) are avoided. On the other hand, it will be necessary to operate near $\Delta I_{ref\,2}$ straight line, defined below:

$$\begin{aligned} \Delta I_{ref\,2}(100\%\,P) &= \Delta I_{ref\,1}(100\%\,P) - 2\% \\ \Delta I_{ref\,2}(0\%\,P) &= \frac{\Delta I_{ref\,2}(100\%\,P)}{2} \end{aligned} \tag{3.20}$$

It will operate over the straight line $\Delta I_{ref\,2}$ slightly more negative than the original reference, modifying with the *R* bank insertion. The conditions are showed in figure 3.13 and they are represented by the parallelepiped *ABCD*.

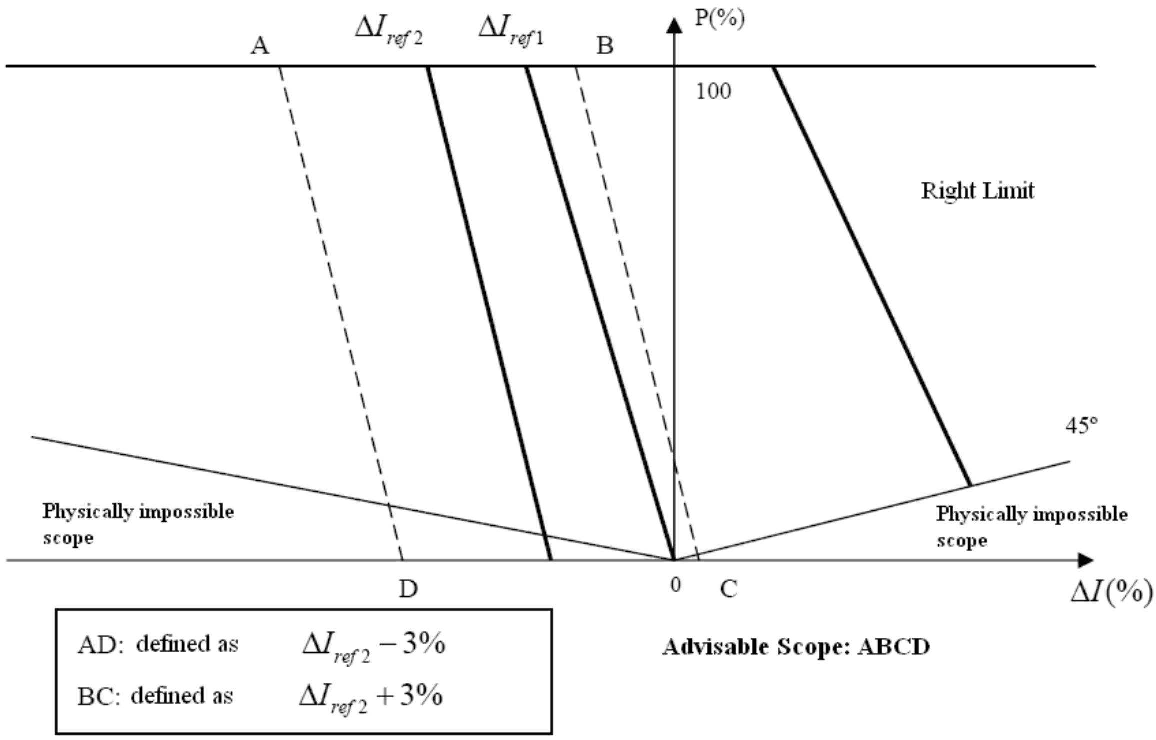


Figure 3.13.- *Operating diagram for REP 1300 reactor.*

3.10.1.3 Shutdown margin characterization

It is named antireactivity margin or shutdown margin for a fixed instant to the reactivity balance obtained in an emergency shutdown that drives the reactor to null power from a *P* power state. This shutdown must be with every single rod drop except for one that is supposed to be stucked in the core high position, which it controls more reactivity.

$$MP = \begin{bmatrix} Antirreactividad \\ introducida\ \ por\, la \\ caída\ de\ (N-1)\ barras \end{bmatrix} - \begin{bmatrix} Reactividad\ liberada \\ por\ el\ descenso\ de \\ potencia\ de\ P\ a\ 0\ MWt \end{bmatrix} \tag{3.21}$$

The power range is deduced with the help of:

- The (*N-1*) rods efficiency measured in the null power tests (*0% P_n*).
- The power effect between *100% P_n* and *0% P_n* integral.

This shutdown margin is necessary for covering the steam pipes breakage of the secondary system as well as to the sudden opening of the safety valves. For this reason, a negative reactivity core after the shutdown must be had. Therefore, the safety antireactivity is variable related with the boron concentration (influenced by the moderator temperature coefficients variation) and would not have to be lower than approximately *1000 pcm.*

The criteria imposed to the control rods can be the following ones:

- To be able to reduce, in every moment, the reactors nominal power and reach the hot shutdown with only the rods.

- To have a core safety antireactivity that allows a reactivity in case of a breakage in steam pipe, sudden opening in the steam dump to the condenser valve or generators safety valves. For example, a hypothetical primary temperature drop of -*36 ºC*, about *180 pcm* in beginning of life (*5 pcm/ºC*) or *1800 pcm* in the end of life would be necessary to compensate. In the shutdown margin balance calculation, there will be a value higher than *1000 pcm*.

- To suppose the rod with more antireactivity out of the core, applying the simple failure criterion (stuck rod criterion).

A 10% error range must be applied to the calculations.

The perturbation theory must be taken into account for describing this experience. This theory shows that the reactivity controlled by one rod is related with the squared flux:

$$\frac{\rho}{\bar{\rho}} = \left(\frac{\phi}{\bar{\phi}}\right)^2 \tag{3.22}$$

where $\bar{\rho}$ and $\bar{\phi}$ are the reactivity and flux average values, respectively.

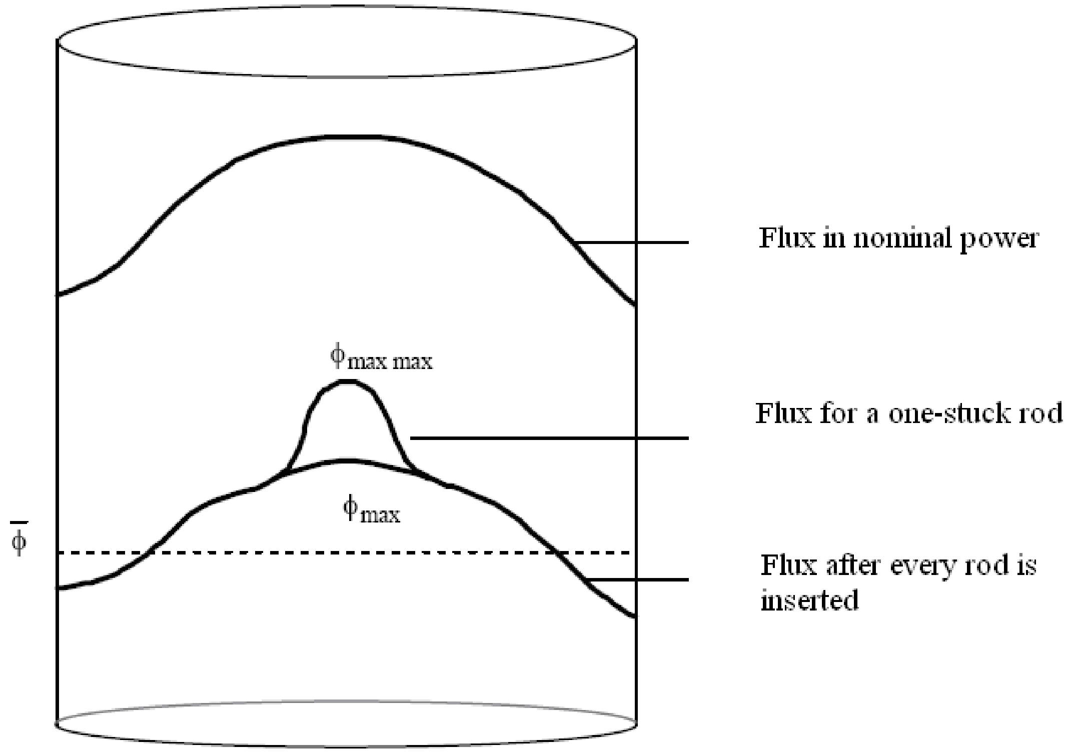


Figure 3.14.- *Neutronic flux perturbation in stuck-rod.*

3.10.1.4 Power redistribution effects

The cooling water enters and leaves at the same temperature when the reactor is in hot stand by operation. If rods are withdrawn, the neutron flux over the vertical axis has a symmetric shape. However, the leaving water has a higher temperature than the entering one when the reactor operates in nominal power. This variation consequence is that the power is redistributed in the reactors lower side by the moderators temperature effect. The symmetric shape has been lost. The power unbalance that has place is measured by the neutron chambers (high and low) and its mathematical expressions in equation (3.17). In nominal power and in beginning of life, the axial offset is about *-7%*. Obviously, during the power reduction, the neutron flux axial profile will try to recover its symmetric shape, locating its power peak in the middle of the reactor.

The tendency explained previously on the *AO* increase during the temperature drop comes amplified by the fact that in the core upper side, less Xenon exists than in the lower side. So, as a consequence of the poisoning, after the shutdown, the axial offset will become positive.

The power redistribution reactivity effects consist of a negative reactivity appearance when power increases (contributing to global stability). The reactivity effect will be different (*-350 pcm* in beginning of life and *-800 pcm* in the end of life). This effect must be taken into account in the establishment of the reactivity balances.

3.10.1.5 Void effects

During the reactors power operation and in the core middle part, a phenomenon called nucleate boiling takes place. It helps to increase the heat transferred to water. However, this phenomenon reduces the water density, being related then with the moderator coefficient (void effect).

The void coefficient is defined as the multiplication factor *k* relative variation for a unitary percentage variation of the moderator voids fraction. This effect is for about *50 pcm* between nominal power and null power in hot stand by, in a pressure water reactor.

A reactor reactivity balance can be carried out during the end of life where temperature effects are more important, with the lastly exposed phenomenon and variables.

3.10.2 Modus Operandi

3.10.2.1 Electrical grid disconnection and house load operation

In habitual operation, the circuit breaker is always closed. The electric power must be higher than *60 MW*, imposed by the auxiliary systems consumption, for carrying out the house load operation.

The studied variables are showed in the lists *p10_1_1* and *p10_1_2*. These variables are: primary thermal power, electric power, average temperature gradient, load slope, Doppler reactivity, moderator reactivity, Xenon reactivity, rods reactivity, lineal power margin, *REC* margin, primary pressure, pressurizer level, primary average temperature, steam temperature in *SG*, steam flow in *SG* output.

1. Load the *Standard 9* state in the simulator. This state corresponds to the reactor operating at 100 % nominal power, critic and at the end of life (*FDV*).

2. Take notes about the present state characteristics related with the reactor control, primary circuit and secondary circuit in tables 3.40, 3.41, 3.42 and 3.43.

3. Activate the simulation.

4. Disconnect the line breaker circuit (‘*disjoncteur de ligne*’) in the turbine screen. During the transitory period, pay attention the important variables evolution in the graphic representations screen. Observe the power axial offset in the operation diagram ‘*diag. Pilotage*’, too.

5. **Stop the simulation** when the stationary state in lower power is reached (approximately 20 mins). Take notes of significant data in tables 3.40, 3.41, 3.42 and 3.43.

6. Print the adequate graphic representation.

7. Calculate the power coefficient α_P in $pcm/\%$, by means of table 3.44.

Reactor control Reactivity (*pcm*) and related variables	***Initial state*** P_n = P_t = P_e =	***house load operation*** P_n = P_t = P_e =	***Recovery 100% P_n*** P_n = P_t = P_e =
Doppler (*pcm*) T average fuel (^oC)			
Moderator (*pcm*) T average moderator (^oC)			
Xenon (*pcm*)			
Samarium (*pcm*)			
Boron (*pcm*) C_b (*ppm*)			
Rods (*pcm*) Position R (steps) Position G1 (steps) Position G2 (steps) Position N1 (steps) Position N2 (steps) Position S (steps)			
Fuel (*pcm*)			

Table 3.40.- *Variables related with the reactors reactivity control system.*

Primary system	***Initial state*** P_n = P_t = P_e =	***house load operation*** P_n = P_t = P_e =	***Recovery 100% P_n*** P_n = P_t = P_e =
Reactor T inlet (*ºC*) T outlet (*ºC*) Pressure (*bar*)			
Pressurizer Level (*%*) Heaters (*kW*) Sprinkling (*t/h*)			
Pumps in operation Flow (*t/h*)			
Chemical and volume control Make up flow (*t/h*) Letdown flow (*t/h*) Tank level (*%*) C_b tank (*ppm*)			

Table 3.41.- *Variables related with the primary system.*

Steam generator	***Initial state*** P_n = P_t = P_e =	***house load operation*** P_n = P_t = P_e =	***Recovery 100% P_n*** P_n = P_t = P_e =
Primary side Water flow (*t/h*) T inlet (^oC) T outlet (^oC)			
Secondary side Feedwater flow (*t/h*) Steam flow (*t/h*) Steam pressure (*bar*)			
Heat transfer zone Water level in GE (*%*) Water total flow (*t/h*) Recirculation flow (*t/h*) Steam flow (*t/h*)			

Table 3.42.- *Variables related with the steam generators.*

Secondary circuit	***Initial state*** P_n = P_t = P_e =	***house load operation*** P_n = P_t = P_e =	***Recovery 100% P_n*** P_n = P_t = P_e =
Turbine Admission valve (*%*) Pressure (*bar*) Steam flow (*t/h*)			
Steam dump Valves (*%*) Pressure (*bar*) Steam flow (*t/h*)			
Condenser Pressure (*mb*) Level (*%*) Residual power (*MW*)			

Table 3.43.- *Variables related with the secondary system.*

Effect	**Initial situation** 100 % P_n	**house load operation** P_n =	$\Delta\rho$
Doppler (pcm)			
Moderator (pcm)			
		TOTAL	

Table 3.44.- *Power coefficient calculation in the transitory period of the house load operation.*

3.10.2.2 Electrical grid connection and nominal power recovery

The energy necessary to rotate the axis (rotor) must be given so that the generator can start up and connect to the grid. The next operation must be follow for achieving grid synchronism.

- Accelerate the turbine until its frequency is similar to the one in the grid (synchronous speed). The alternators synchronous speed will be *1500 rpm* because it is a two-pole device.

- Change the rotors excitation current to obtain voltages similar between the grid and the stator void voltage.

- Achieve the synchronism between the alternator phases and grid during the *synchronism* procedure. Change slightly the rotation speed until synchronizing the phases and then connect it to the grid.

The connection can be made if the turbine rotation frequency is lower slightly to *1500 rpm* (a difference lower than a *1 Hz*) and the relative voltage difference is up to 5 %. In addition, the synchronoscope must be connected so that the phases synchronism can be achieved.

Carry out the following operations in the '*turbine*' screen over the last state.

1. Diminish slightly the turbines admission valve so that the rotation speed can be changed up to *1500 rpm*. See that the alternators excitation current changes in auto mode for keeping constant the alternator terminal voltage.

2. In the '*synchronoscope*' board, activate the synchronoscope by pressing '*marche*'. The needle rotation indicates the difference between the grid frequency and the alternator rotation speed.

3. Connect to the grid by closing the line circuit breaker '*fermé*'.

The procedure that carries out from house load operation until nominal power sets by normal procedure.

4. Open slowly and in manual mode the turbine admission valve. The *21,3 %* value is proposed for this opening. See how the flow that was dump to the condenser now is turbined meanwhile the rods are withdrawn, in auto mode, from the core. The *5 %* value is proposed for every opening until 100% and let the power get stabilized.

5. See that the Xenon concentration is not stabilized. Insert the regulation control bank in manual mode so that the power unbalance is compensated.

6. Complete the tables 3.40, 3.41, 3.42 and 3.43 with the reactor in *100 %* P_n.

3.10.2.3 Shutdown margin calculation and power coefficient of reactivity

The shutdown margin or antireactivity margin (*MAR*) is a variable that has to be always controlled. Particularly, the shutdown margin has a special interest in the end of life, moment where the power coefficient is higher and the moderators feedback is more important. Take into account the following considerations and simplifications in the shutdown margin estimation.

- Get all rods global efficiency (control and safety). Calculate the average reactivity controlled by every single rod $\bar{\rho}$.

- Suppose a stuck antireactivity related with the stuck rod. Take the (3.22) equation with the following perturbations:
    - Flux radial variation $\frac{\phi_{\max}}{\bar{\phi}} = 1,5$.
    - Flux local perturbation by stuck rod $\frac{\phi_{\max\max}}{\phi_{\max}} = 1,7$.

- Consider a *1500 pcm* safety antireactivity so that it is necessary to avoid critic situations in primary quick cooling.

- Suppose that in one operation moment, the *R* regulation bank could be totally inserted. Furthermore, the reactivity of this bank must be erased from the balance.

RECOVERED REACTIVITY (*pcm*)	**RODS AVAILABLE ANTIREACTIVITY (*pcm*)**
Doppler effect:	*Rods efficiency:*
Moderator effect:	*Stuck rod:*
Power redistribution:	*TOTAL:*
Void effect:	*Uncertainty10 %:*
Safety antireactivity:	*Introduced R bank:*
BALANCE:	**BALANCE:**

Table 3.45.- *Reactivity balance and shutdown margin calculation*

Calculate the antireactivity margin.

Note: In the simulator, the power redistribution appears to be with the antireactivity by the Doppler effect.

3.10.3 Questions related with the experience

1. What are the restrictions contained in the operation diagram?
2. What is the house load operation? What is it supposed to do and with what target?
3. Which are the reactor power regulation systems?
4. Which are the turbine regulation systems?
5. How is the electrical system constituted?
6. What is the reactors shutdown margin?
7. Why is the *R* regulation bank limited in its insertion?
8. How is the power coefficient determinated?
9. Calculate the power coefficient.
10. In which burn up phase (beginning or end of life) has the power redistribution effect more importance?