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Conceptual Design of an Inherently Safe Reactor with Modular Characteristics

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ABSTRACT

In the aim to search for solutions to increase the future competitiveness of nuclear energy a new concept of a modular reactor is presented. The neutronic-thermohydraulic and economic calculations for the steady state design of an inherently safe while economically competitive reactor (RE.M.I.SE., REactor Modular Inherentemente SEguro, in Spanish) are shown here. Primary system is subdivided in several modules with individual steam generator, fuel element and primary circuit. Enhanced safety is attained by means of passive safety systems and inherent safety features. This design presents several advantages. The total power of the core (200 Mwe) is divided in a large number of units of reduced power (2 MWth) which implies that the module can be subjected to full size tests. An identical modules design implies faster and simpler core construction and replacement. Reduction in capital costs is possible through standardized manufacturing of modules and a competitive fuel cycle cost.

Key words: Safety, nuclear, thermohydraulics, neutronics, economics, modeling

INTRODUCTION

The excellent expansion of nuclear industry in the seventies diminished greatly in the nineties and reached a very low demand state in several countries (Tashimo and Matsui, 2008). The main factors of that inflexion of the grow rate in the nucleoelectric demand were Three Mile Island (Petrangeli, 2006) and Chernobyl (Thomas *et al.*, 2011) accidents. The impact caused by those accidents was the lack of credibility in Nuclear Power Plants safety by the general public.

This tendency was partially reversed by safety improvements to actual designs from the late nineties till present time (Sha *et al.*, 2004). Nowadays, Fukushima accident (Manolopoulou *et al.*, 2011) could be a new trigger for a nucleoelectric demand decrease all around the world.

This situation imposes the search of solutions to increase the future competitiveness of nuclear energy (Malkawi, 2004). To achieve this, a new generation of nuclear reactors designs should be proposed (Magan *et al.*, 2011), with a criteria based on an enhancement of nuclear safety, without neglecting economic issues.

Nowadays, there are two main research lines for new reactor designs. One proposes improvements of conventional designs (Lestani *et al.*, 2011; Arai *et al.*, 2008), trying to progress using proved technologies. The other involves huge changes and conceptual innovations and is known as the advanced reactors design (Filho, 2011; Khan *et al.*, 2011; Terman and Khalafi, 2006).

REMISE concept presented here was developed taking into account both of this philosophy. The main features of this nuclear plant are the following:

- Medium power: focused to developing electric markets
- Passive characteristics: inherently safe systems and natural convection in the primary system
- Interchangeability of components
- No need for large components
- Convenient component manufacturing
- Feasibility of real scales tests
- Competitive cost

This reactor has being developed up to a conceptual level stage. All steady state parameters that completely define the primary module have been determined. The calculations were performed optimizing the variables from a thermohydraulic, neutronic and economical point of view. In addition, the study of operational and accidental transients is being carried out, as well as the implementation of safety systems.

This study tries to show the feasibility of a new advanced concept of medium power reactor, through a steady state analysis. This was demonstrated by neutronic, thermohydraulic and economic calculations and results.

GENERAL CHARACTERISTICS

REMISE is a power reactor with enriched uranium and light water as moderator and coolant. Its main characteristic is the partition of the primary system in modules which are pressure tubes that contain a fuel element at the bottom and a steam generator at the top.

The main advantages of the REMISE from a technical and economical point of view are listed below.

Technical advantages

Advanced modularity: REMISE concept divides the primary system in modules with the following advantages:

- Fabrication of modules in factory
- Primary system modularization

Tests at real scale and low cost could be perform for a single module, so high reliability based on experimental knowledge of the behavior for stationary and transient operation.

Possibility to verify the primary system testing each module in real scale in factory with high level of reliability and quality assurance.

- Big components are avoided that results in simple transportation

Passive safety features: In most of the nuclear plants presently in operation, reactor safety relies on active systems or actions performed by the operators. The Chernobyl (IAEA, 1986) and Three Mile Island (Kemeny, 1979) accidents showed that both, active systems and operators are not absolutely reliable. Passive safety systems constitute an attractive alternative which is based in the intelligent use of natural laws. The inherent and passive features used in REMISE are:

- The primary coolant flows by natural circulation in each pressure tube
- The residual heat removal in operative and accidental situations is performed by natural convection
- Heat transfer between the heat sink and the atmosphere is by evaporation of water
- Temperature and void reactivity coefficients are both negative
- Low power density reduces the MDNB ratio and the central pellet temperature (Tong and Weisman, 1970)
- The reactor has a water inventory able to remove the residual heat for a period from 3 to 10 days without the requirement of corrective actions

Economical advantages

Shorter construction times: An important part of the cost of conventional nuclear plants is associated with long construction times. In REMISE the modularity and simplicity of the components reduces the construction time and the complexity of assembling.

Longer expected life of the plant: In this reactor, the modular components are designed in such a way that they can be replaced. Consequently, the total life of the plant can be extended which implies lower capital costs.

Versatility in the electric market: The conventional nuclear reactors have increased their total power to reduce the specific cost. REMISE is a medium power reactor and its total cost is lower than the cost of a high power reactor. This is convenient when potential markets are considered:

- There are several countries with low power requirements, so that low and medium power reactors are an attractive alternative to be considered
- High power implies more and more complex systems which is not compatible with the actual criteria of simple and safe reactors. REMISE presents less and simpler safety systems which is an advantage of the low and medium power reactors

Accessible technology: The most significant advantage of REMISE concept is that all the components are of simple construction.

High load factor: The low power density causes a large irradiation time which results in a high load factor due to a longer period between core reloads.

REACTOR DESCRIPTION

In this section the distinctive characteristics of REMISE reactor are described. The components and systems that are not mentioned should be considered similar to the ones in a conventional PWR. A schematic view of the plant can be seen in Fig. 1.

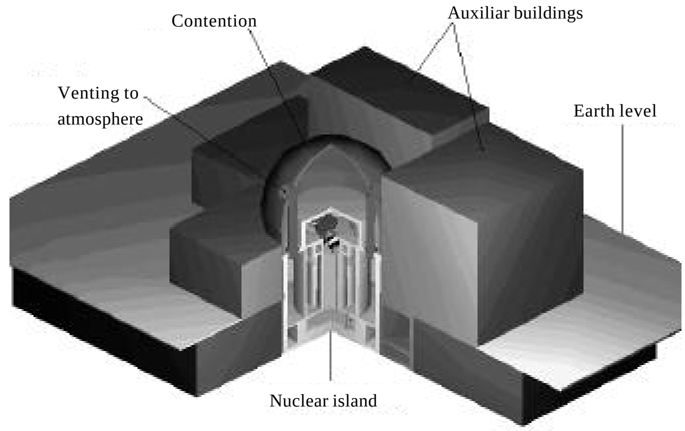

Fig. 1: Nuclear plant view

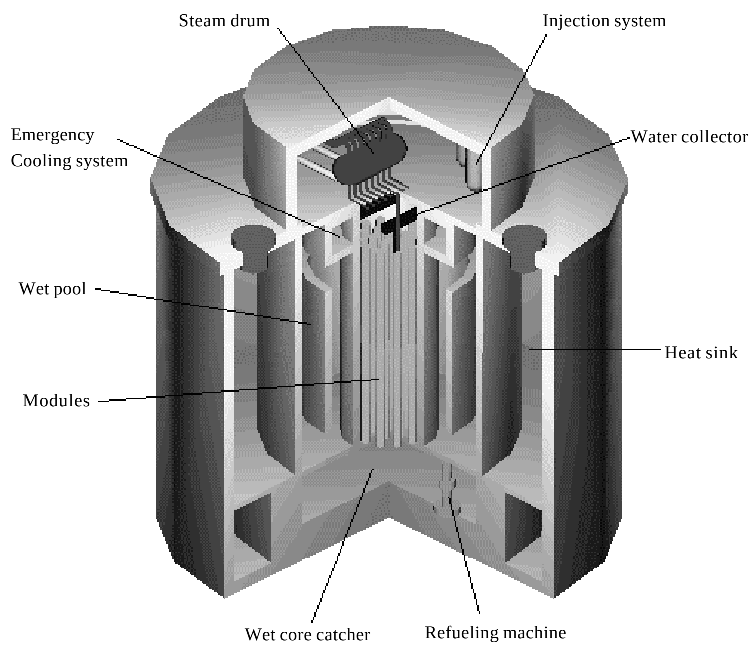

Fig. 2: Nuclear island

Primary system: Primary system is divided in modules that are located in a hexagonal array, which constitute the core, inside a dry containment. Inter module space is filled with a gas at nearly atmospheric pressure. The Volume and Chemical Control System interconnect the modules. The system involves a large number of low-diameter pipes (300 inlet pipes and 300 outlet pipes) but the concept is similar to the feeders and headers of pressure tubes in a CANDU reactor (which has 380 pressure tubes), so it is not expectable a greater complexity than in this proved design. A pressure suppression pool and an annular final heat sink are located surrounding the dry containment. The nuclear island is physically separated from the secondary and the auxiliary systems which are completely conventional (Fig. 2).

Two alternatives were considered for the pressure module concept:

- Independent modules: each unit is a close independent system
- Interconnected modules: the pressure tubes are interconnected by means of collectors

A thermohydraulic analysis was done; the results showed that each module could effectively work as a self-pressurized closed system. However, the system of interconnected modules is more convenient for the following reasons:

- Provides a better layout for purification and chemical control of the primary coolant
- Minimal number of safety valves in the primary system. A single pressurizer and a unique set of safety and relief valves are required to control the pressure of the whole primary system
- Longer discharge time in case of LOCA
- The primary coolant injection can be performed through a general collector, instead of individually to each module, which would be very cumbersome
- Since the coolant flow between modules is not considerable under operating conditions (about 5% of the circulating flow) the feasibility of real scale testing is still valid

A schematic diagram of the primary system can be seen in Fig. 3.

Each unit or module consists of a pressure tube that includes a steam generator and fuel element (Fig. 4). The pressure tubes are located in vertical position, made of stainless steel and Zry in the active region (lower part). Steam generation does not occur in a large component as in standard pressurized water reactors but is divided among the component modules. Each individual

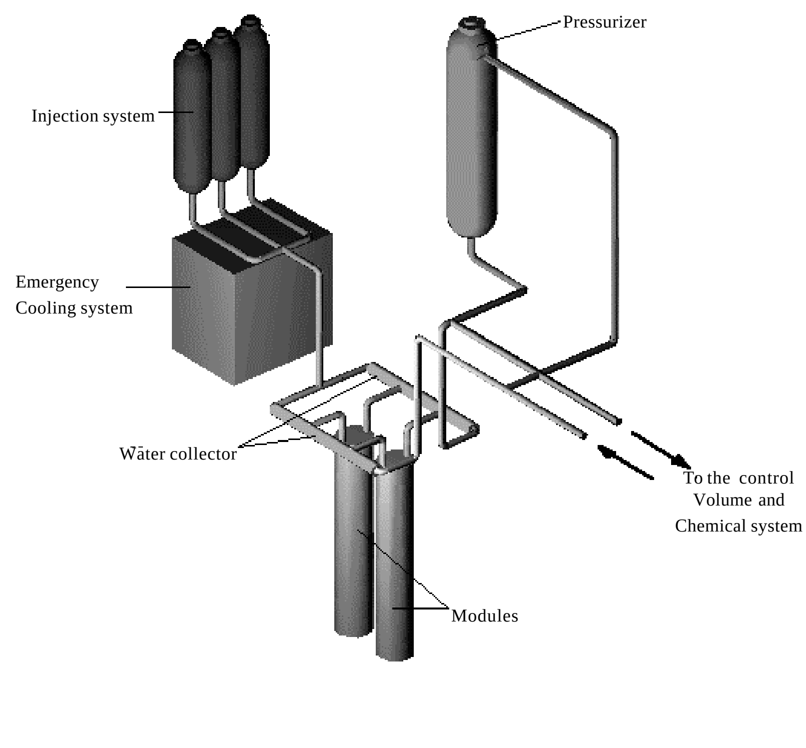

Fig. 3: Scheme of the primary system

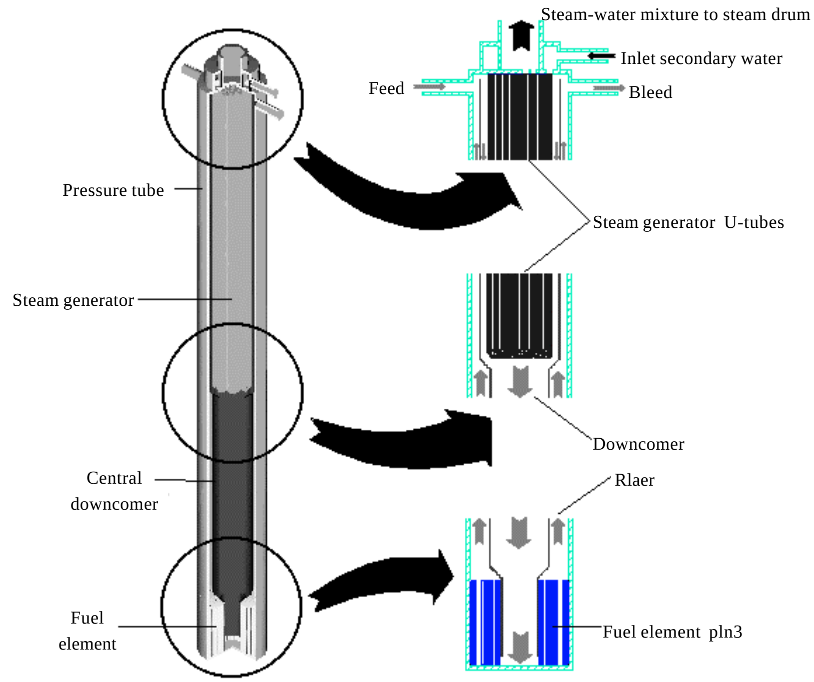

Fig. 4: Primary module cut

steam generator, at the top of the pressure tube, has 18 U tubes of Incolloy 800. The only large component that remains from standard designs is the pressurizer.

No pumps are used in the primary circuit since natural circulation is established as operating condition. The coolant is heated in the fuel element, rises in the annular region, transfers energy in the steam generator and descends through the central channel. The density difference between regions provides the buoyancy force that is balanced by the pressure drops.

The fuel pins are identical to the ones of a PWR reactor in materials, densities and radii. Although the core is relatively short (1.1 m) the burnup is reasonable (31000 MWd/Ton for $\varepsilon = 3.5\%$ and a three zone refueling strategy). The axial neutronic leakage through the inter-module space is not significant and consequently there is no need of axial reflectors.

Secondary system: The steam-water mixture produced in each module is transported to steam drums (Fig. 5) with a mean void fraction of 20%. The dry steam is separated and then conducted to the turbine, with a typical thermal cycle of a pressure water reactor. Feed water is pumped through distributors to the steam generator of each module. Also the secondary system is designed to work under natural circulation in accidental conditions with SCRAM, as an alternative to the Residual Heat Removal System.

Control mechanisms: The reactivity control is achieved by neutron absorbing rods that operate vertically from the top and are located in the dry containment. They do not work under pressure and coolant flow. Therefore, these are very simple mechanisms, driven by gravity and step by step engines.

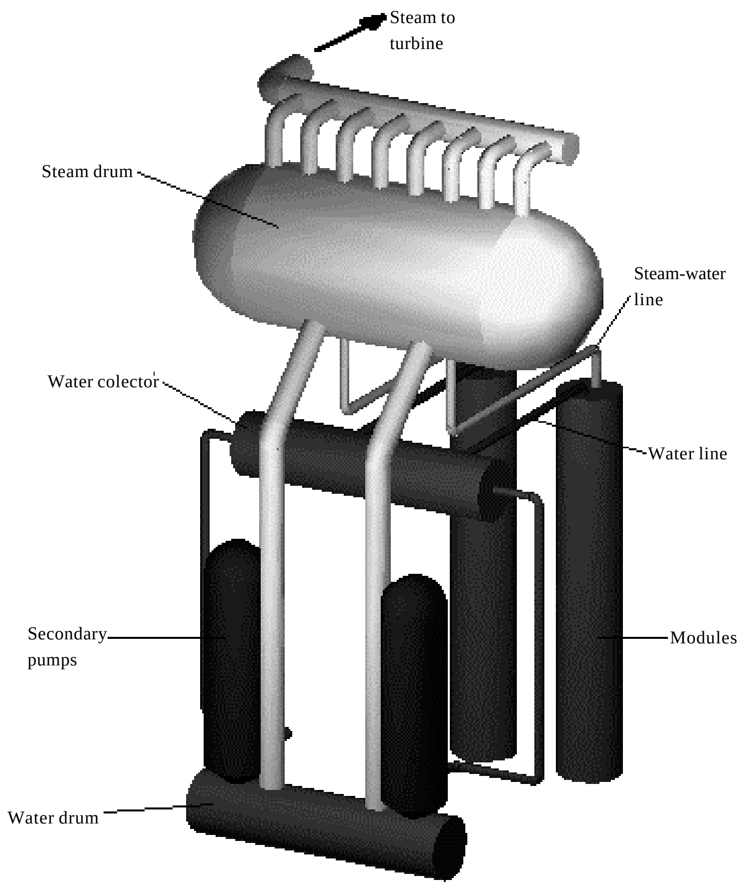

Fig. 5: Main components of the secondary system

The dry containment is expectable to be flooded under shutdown conditions or accidental situation. Therefore, the first shutdown system is expected to be inserted in the inter-module space before the flooding. In case of First Shutdown System failure, the flooding could include a borated solution to avoid a criticality accident. These points will be analyzed in detail when transient calculations are completed.

Refueling: Batch refueling is performed during the reactor shut down, under depressurized conditions. One third of the core is replaced in every refueling. Due to the low power density, the irradiation time is longer than typical PWRs and refueling is required approximately every eighteen months. The pressure tube has a plug at the bottom for the refueling. Underneath the fuel elements there is a water pool at low pressure that is used for the operation of refueling. Also it acts as wet core catcher in case of severe accident or plug failure. Otherwise, this wet core catcher could lead to a steam explosion. The probability of this sequence should be analyzed once accident calculations are completed and Probabilistic Safety Analysis is performed.

Safety systems: All calculations performed at this stage were done for steady state operation. Safety systems are only proposed at this stage. Their design is left for the next stage of development where operational transients and incidental situations should be analyzed. The safety systems are:

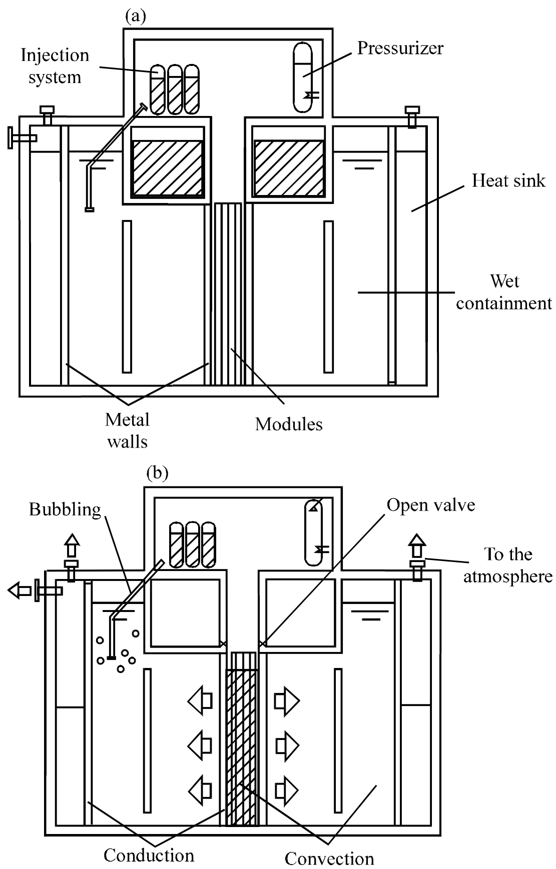

Fig. 6: Residual heat removal system

- Shut down system located in the dry containment
- Passive core residual heat removal system (PRHRS). This system injects water in the dry containment in case of normal cooling failure. By this way it is possible to cool the modules externally. Natural convection is settled inside the modules, the heat is conducted through the pressure tube wall to the water injected in the dry containment, and then finally evacuated by natural convection to a surrounding wet containment (Fig. 6)
- High and medium pressure injection system to the collectors of the primary system is under consideration to prevent the fuel element uncovering
- Wet containment to provide an important amount of water to remove the heat during 3 to 10 days. In such case, heat is transferred to the last heat sink by conduction through metal walls. The sink transfers heat by evaporating water (not contaminated) to the atmosphere. The only active corrective action to be performed is the reflooding of water to the sink. Eventually, it can be cooled by natural convection of air. This alternative is left for a next stage of development

REACTOR CALCULATION

In this item, neutronic, thermohydraulic and economical aspects are discussed. Then the global optimization is described and the final values of design parameters are shown in Table 1.

Table 1: Value of the main variables of the reactor

Variable	Description	Value
Np	Number of fuel element pins	145
ε	Enrichment	UO2,3.5% ^{235}U
R_p	Pin radius	0.5 cm
L_N	Core length	109.8 cm
$q_{N''}$	Medium surface power density	40 W/cm^{-2}
R_i	Internal fuel element tube radius	3.62 cm
R_{tp}	Pressure tube radius	9.5 cm
e_{tp}	Pressure tube thickness	7.1 mm
N_{MOD}	Number of modules	300
R_N	Core radius	2.54 m
K_{eff}	Multiplication effective factor	1.1452
Q	Medium core burnup	30960 MWd/TonU
t_{irr}	Irradiation time	574 d
N	Number of refuelling zones	3
α_v	Void feedback coefficient	-165 pcm/% r
F_p	Power peaking factor	1.18
R_{TGV}	Steam generator tube radius	0.75 cm
e_{TVG}	Steam generator tube thickness	1.1 mm
R_{GV}	External radius of the SG	7.85 cm
R_{DC}	Down comer radius	5.95 cm
P_p	Primary system pressure	12.0 MPa
P_s	Secondary system pressure	5.5 MPa
T_i	Inlet core temperature	285°C
T_o	Outlet core temperature	325°C
T_f	Steam temperature	270°C
η	Plant efficiency	0.327
f_c	Load factor	0.9
L_{TOT}	Module total length	12.90 m
MDNBR	Minimum CHF ratio	2.3
x_s	Outlet fuel element void fraction	1%
C_{CYC}	Fuel cycle cost	10 mills KWh^{-1}
C_{CAP}	Plant investment cost	25 mills KWh^{-1}
$C_{O\ and\ M}$	Operation and maintenance cost	7 mills KWh^{-1}
C_{TOT}	Total cost	42 mills KWh^{-1}

Neutron physics: As REMISE is a new reactor concept, the fuel element geometry is not fully determined in the first steps of the design. Therefore, the array of fuel pins inside the pressure tube has to be determined. This is an important difference with reactors such as PWRs, where the geometry is well defined and only second order changes are performed in core design.

Neutronic calculations of the fuel element were performed by the standard cell code WIMS-D4. Heat removal demands the existence of two regions in the pressure tube: one where the coolant flows upward, heated by the fuel pins and the down comer. The results showed that an annular geometry for the fuel pins is more convenient, because the neutron current coming from the central channel balances the one incoming from the neighbor modules. This results in a more uniform neutron flux than when the fuel element is in the central region. A view of this array is shown in Fig. 7.

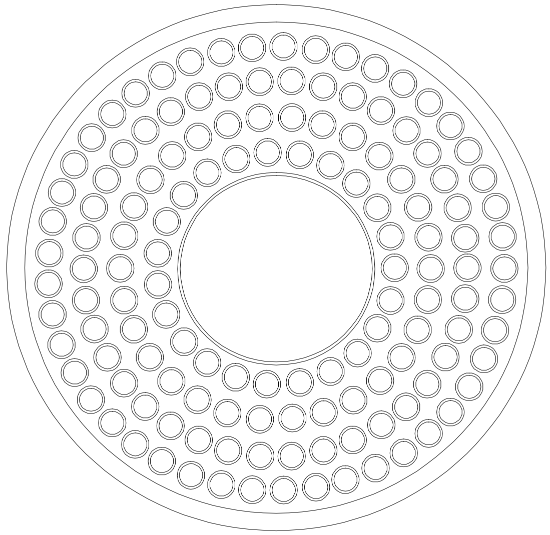

Fig. 7: Cluster array of fuel pins

It is important to notice that, although in this configuration the neutron flux is more uniform, it is necessary to reduce the power peaking factor of the central ring. It is caused due to over moderation from the central water channel. To attain this, ^{10}B can be used as burnable poison uniformly dispersed in the fuel.

One of the constraints imposed to the reactor is the negative reactivity feedback coefficient. Void reactivity coefficient should be sufficiently negative and it is a determinant factor in the geometry of the fuel element. It is very dependent of the moderator to fuel ratio:

$$V_{M/F} = \frac{A_{H_2O}}{A_{Fuel}} = \frac{R_{tp}^2 - N_p r_p^2}{N_p r_p^2}$$

It is desirable to have a moderator-fuel ratio similar to a PWR, where α_v is negative, typically:

$$V_{M/F} = 1.4$$

For a given pressure tube radius, module power and pin diameter (also similar to a PWR), the fuel element geometry is completely defined by these constraints. α_v was calculated changing the coolant density.

Fuel element burnup was determined by the cell code WIMS-D4, for a final reactivity of 1500 pcm. Considering a lineal dependence of Keff vs. burnup (which was effectively observed), core burnup was estimated by:

$$Q_N = \frac{2N}{N+1} Q_{FE}$$

where, N is the number of refueling zones in the core.

Thermohydraulic calculus: The hypothesis and general considerations done for the primary circuit and steam generator are explained below.

Primary circuit: One of the requirements imposed to the design is natural circulation in the primary circuit. To determine the dimensions that allow natural circulation, the system is divided into three sections (fuel element, riser and steam generator) with two regions (up and down), with average densities and temperatures for each. Design goal was to minimize the pressure drops in the whole circuit for a given radius of the pressure tube, to obtain a minimum raiser length.

Thermohydraulic calculus was done for steady state and involves the following hypothesis:

- Uniform pressure of 12 MPa in the primary and 5.5 MPa in the secondary circuits. Pressure losses are much lower than total pressure
- No heat transfer between riser (annular) and down comer (central) and between pressure tube and dry containment
- The system is treated as one-dimensional
- Heat transfer coefficients are estimated from Dittus-Boelter correlation for single phase and Tong correlation for two-phase flow
- Pressure drop is calculated using the Moody diagram for the friction factor, with the homogeneous two-phase flow multiplier for the secondary side
- Tube thickness, for the steam generator and pressure tubes are calculated from thin wall cylinder approximation

Steam generator: Two alternatives were considered for the steam generator: primary or secondary inside the tubes. They are shown in Fig. 8.

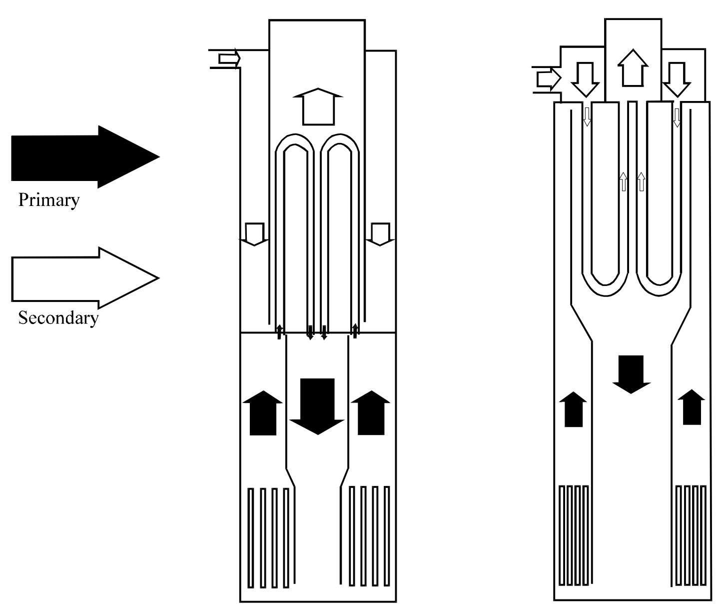

Fig. 8: Steam generator alternatives

In alternative (a) pressure drop inside the tubes is so high, that natural convection at full power in the primary side is not established for reasonable module heights. Since the module must be extracted from the top, after removing the fuel element from the bottom, the total length should not exceed 12 m (which is about half of the total height of the pressure vessel of a BWR) (Roberts, 1981). For these reasons, this alternative was discarded.

In alternative (b), when the water of the secondary system flows inside the tubes, the pressure drop is higher than in (a), because of the two-phase flow. But in the secondary system pumps establish the flow. The decay heat can still be removed by natural circulation in secondary circuit at low flow. Steam drums are used to collect and separate the vapor. In order to fulfill the natural convection requirement at decay heat, the position of the steam drum must be determined to provide the buoyancy force necessary to balance the pressure drop. The aspect ratio of steam generators tubes is not different from the one corresponding to large steam generators of large power reactors, besides tubes amount is sensible small. Therefore no construction difficulties are expected.

Economics: At this stage only fuel cycle cost can be calculated using physical parameters of the reactor. Plant amortization and maintenance cost is estimated from cost breakdown and scaling to channel-type reactor (CANDU) (IAEA, 1983).

Fuel cycle cost: For a given module power, pin diameter and moderator-fuel ratio, the total core length L_N and surface heat flux density q''_N are related to total power:

$$P = 2\pi R_p N_p L_N q''_{GV}$$

These two variables are determined minimizing the fuel cycle cost which is equal to the first core amortization cost plus the refueling cost. Surface heat flux density affects the capital cost of the first load. For larger values of q''_N, the core becomes shorter. Therefore, lower uranium inventory is required for the same power with a reduction in the first core amortization cost. On the other hand, for a smaller core, neutronic axial leakage increases decreasing fuel burnup and so increasing the refueling cost.

A minimum fuel cycle cost, for a module power of 2 MWth, of approximately 10 mills/KWh was found for q''_N = 40 W/cm^2 and L_N = 1.1 m. In the range of 40-45 W cm^{-2} the fuel cycle cost is nearly flat as a function of q''_N (Fig. 9). Due to the flatness of this region it is convenient to choose lower values of q''_N, so the Minimum Departure from Nucleate Boiling Ratio (MDNBR) is improved.

Plant amortization cost: Nowadays, the introduction of modularity concepts in nuclear industry has caused a reduction in overnight and leveled costs (Lapp and Golay, 1997; Williams, 1997). REMISE concept presents advanced modularity criteria, which starts in the nuclear island. This concept would result in further reduction of the NPP cost. On the other hand, low reactor power is associated with a higher specific cost per unit of power.

Since the primary system is divided in channels, the refueling machine is inside the containment, and control rods work at low-pressure conditions, the reactor is closer to a CANDU than to a typical PWR or BWR. Therefore a CANDU cost breakdown was adopted to perform the analysis of REMISE plant amortization cost.

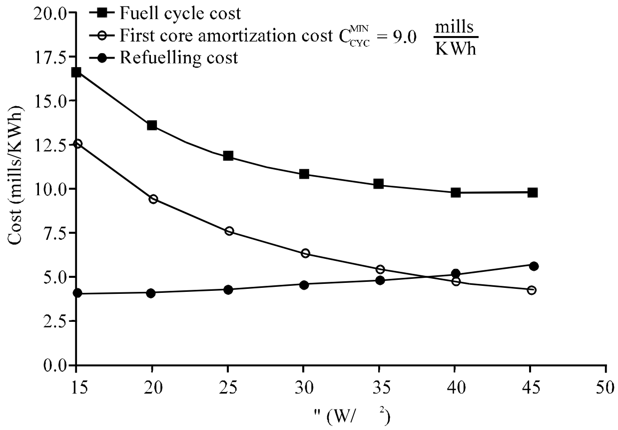

Fig. 9: Optimization of the fuel cycle cost

Two CANDU reactors of 658 Mwe in South Korea have a 1424 U$S/KWe installation overnight cost (IEA and NEA, 1993). With a 0.93 factor of cost reduction for the construction of two NPP instead of only one (IAEA, 1991). A single NPP would cost 1007 MU$S. A classical model of cost scaling (IAEA, 1984) can be applied to estimate the total cost of a 196 Mwe NPP. With a 0.77 scaling coefficient and not including the heavy water cost (~63 MU$S), a total investment cost of 361 MU$S is obtained.

This gives an overnight cost 30% higher than a competitive CANDU NPP. On the other hand, if a CANDU cost breakdown is used and conservative values for cost reduction are adopted for:

- The advanced modularity in the primary system
- Classical strategies of modular construction
- New systems of C and I

the plant investment cost could be reduced. For example, assuming a 20% cost reduction in the reactor, 20% in electrical equipment, 50% in indirect cost construction and 20% in engineering indirect costs, the plant investment cost would be 298 MU$S.

In this conditions, the overnight cost would be 1521 U$S/KWe, which gives a leveled amortization cost of approximately 22.5 mills/KWh for an interest rate of 10%. Therefore, a 30% increase in the specific installation costs due to the low power of the REMISE NPP would be balanced by the reduction due to modular construction, in a conservative way.

Global optimization: The calculations discussed above can be performed for several values of module power, reactor power, moderator-fuel ratio, etc. The value of all reactor variables can be optimized for a minimum fuel cycle cost of the NPP, taking into account the constraints in the design parameters such as the void coefficient, the total module length, minimum departure from critical heat flux, etc. To fulfill this objective, a neutronic-thermohydraulic computer code was developed. A schematic flowchart of the calculations is shown in Fig. 10, where the feedback between different variables is detailed. Final values of the design parameters are shown in Table 1.

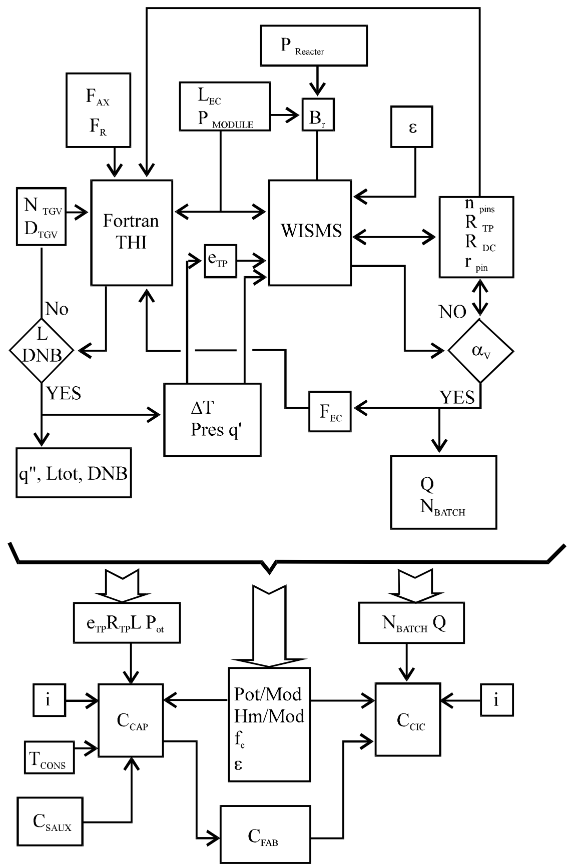

Fig. 10: Neutronic-thermohydraulic-economic optimization

List of variables in Fig. 10:

$P_{REACTOR}$	:	Reactor thermal power
P_{MODULE}	:	Module thermal power
F_{AX}	:	Axial peaking factor
F_R	:	Radial peaking factor
F_{FE}	:	Fuel element peaking factor
L_{TOT}	:	Total module length
L_{FE}	:	Fuel element length
B_R	:	Radial buckling
ε_{U5}	:	Uranium enrichment
N_{TGV}	:	Number of steam generator tubes
D_{TGV}	:	Diameter of steam generator tubes
e_{TP}	:	Pressure tube thickness
R_{TP}	:	Pressure tube radius
R_{DC}	:	Down comer radius
r_{PIN}	:	Fuel pin radius
n_{PINS}	:	Number of fuel pins
α_V	:	Void feedback coefficient

MDNBR	:	Minimum departure of nucleate boiling ratio
ΔT	:	Core temperature drop
Pres.	:	Primary system pressure
q'	:	Linear power density in the fuel element
q"	:	Surface power density in the fuel element
Q	:	Medium core burnup
N_{BATCH}	:	Number of refueling zones
f_C	:	Load factor
η	:	Plant efficiency
C_{CAP}	:	Plant amortization cost
C_{CYC}	:	Fuel cycle cost
C_{SAUX}	:	Auxiliary systems cost
C_{FAB}	:	Fabrication cost
I	:	Interest rate
T_{CONS}	:	Construction time

CONCLUSIONS

Results showed that the concept is technically and economically feasible in a preliminary steady-state analysis. The total feasibility of the concept will be proved once transient analysis is completed.

Transient analysis now are being carried out, in particular the system behavior under a LOCA situation, related to a rupture in one module inlet or outlet pipe. This analysis, together with another transients or initiating events involves a lot of calculations which are expected to be presented in future study.

A minimum cost for a module power of 2 MWth was achieved for a surface heat flux density of 40 W $cm^{-2}$and a total core length of 1.1 m. This is a very important result, as it differs from forced convection reactors, where the power density is maximized. In this reactor with natural convection, the optimal power density is determined from economical fuel cycle criteria, due to the geometric constraints and the neutronic penalties.

Based in the conceptual results achieved at this stage, the feasibility of the REMISE concept was determined.

This reactor shows excellent features for intermediate power markets, due to its low total cost and enhanced safety which are the principal characteristics required to recover the growth in nuclear energy generation.

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