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STUDY ON NUCLEAR SPACE REACTOR DEVELOPMENT (SURE)

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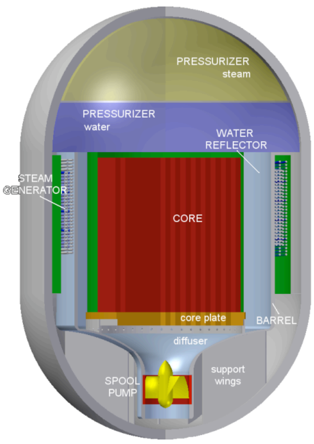

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Executive summary

Foreword.

Ambitious solar system exploration missions in the near future will require robust space power sources in the range of 10 to 200 KWe. Fission power systems are well suited to provide safe, reliable, and economic power within this range. Therefore the goal of this research program is to carry out a preliminary feasibility study of a nuclear fission reactor suited for space applications. These refer either to rocket propulsion by electricity (*NEP: Nuclear Electric Propulsion*) or to electrical power production for stationary settlements (manned or unmanned) on some planet (*Mars*), or deep space planetary surfaces, or satellites (*Moon*).

This application of nuclear energy is very demanding and it should be addressed in a gradual way, because numerous space fission power programs failed having tried to do too much too soon. Thus a good option for developing the reactor-related portion of this infrastructure and experience is to start by developing and utilizing a low-power surface fission power system: surface applications generally place less demanding requirements on the reactor and integrated system. Even if this study concerns both applications, the solutions envisaged better apply to surface applications.

The present study is a preliminary one, which in principle cannot have the ambition to give a priori a well definite answer to the problem, in the sense to reach by certain a viable proposal fit for a subsequent specific R&D program. A space nuclear reactor should respond to the following general requirements:

1. To be extremely reliable;
2. To imply an R&D program of moderate cost;
3. To be deployed within a reasonable period of time;
4. To be operated and controlled for a long time without intervention;
5. To be able to be transported into space (mass and size limit)
6. To be also used as a byproduct for some particular terrestrial application (or at least to share common technologies).

The first three items mean that the chosen reactor type must be extensively and positively tested in terrestrial applications, thus too innovative proposals are a priori excluded, at least in the medium period. Item 4) is important and again in favor of simple and reliable solutions. Item 5) is quite obviuos.

Item 6) is motivated by the usefulness to have an economic return of R&D costs from other non space applications of the same reactor concept, in fact it seems possible and probable that some technologies needed for space reactors have a terrestrial application in nuclear and non nuclear systems.

All the above considerations taken into account, it can be concluded that this reactor type should be:

- Based on the well proven technology of present terrestrial reactors, allowing obviously the development of different components and systems needed to accomplish the specific mission of a space reactor, according to known processes,
- Suitable for propulsion and stationary applications, apart from reasonable and moderate differences.

If these conclusions are accepted in this context, the first result is that the propulsion reactor has to produce electricity, in the same way as the stationary one, and its electricity will be used for propulsive scopes, by adopting suitable converting apparatus downstream the reactor.

A space reactor must satisfy a number of requirements, besides the general ones presented above. A non exhaustive list is as follows:

- produce an electrical power around 100 KW;
- last a long period of time (around 4000 days) without any intervention and fuel supply;
- minimize the overall mass and volume for rocket payload constraints;
- use high enriched uranium;
- adopt a core power density substantially lower than current reactors;
- satisfy the usual safety requirements of terrestrial reactors and besides this to assure:
    - no irradiated fuel is present at launch;
    - the core subcriticality in the case of all possible launch accidents (flooding);
    - the radiation protection without impairing weight requirements;
    - an easy decommissioning in space;
- a simple control of the reactor and the overall plant;
- a substantial reduction and simplification of maintenance and repairs;
- avoid any leakage of the contained fluids or implement systems to recuperate them;

All taken into account two reactor types are here considered: PWR and HTGR. The PWR (Pressurized Water Reactors) is the most common reactor type for terrestrial power stations and widely used for submarines propulsion: the features of space reactors are more similar, in relative sense, to those of naval reactors than those of civilian reactors, and this can be seen as a significant starting point. The HTGR (High Temperature Gas Reactors) has the peculiar feature to generate heat at much higher temperatures than PWRs, typically 800-900 °C against 300 °C. This means higher thermodynamic efficiencies and the possibility to widen the nuclear energy exploitation to other industrial applications different from electricity production. However, the experience acquired up to now, even if significant, is not comparable at all to that of PWRs; in fact important R&D programs are under way in the world.

In conclusion, this program concerns a preliminary feasibility study of a space reactor, suited either for stationary needs on a planet or for propulsion, to produce electrical energy of the order of 100 KW. It will be articulated in the following steps:

- Assume as a first choice the PWR solution as the reference system.
- Execute a rather detailed neutronic study of this reactor, which is two orders of magnitude smaller than conventional reactors (the power is three order of magnitude lower, but the power density is an order of magnitude lower).
- Define the preliminary scheme of the whole plant, under alternative solutions for electricity production: adoption of thermoelectric device or simplified conventional generators.
- Put in evidence the differences between propulsion and stationary reactor specifications and the way to fulfil them.

- Carry out analyses for a preliminary verification of its capability to satisfy the requirements listed above for a space reactor.
- Analyzing the HTGR reactors, outlining pros and cons of these reactors when compared to PWRs.
- Make a survey of Italian industry capability and willingness to participate to the development of such a reactor.
- Verify the potentialities of space reactors for particular terrestrial uses.
- Identify a research and development program including the aspects of interest for civilian (industrial) purposes in Europe;
- Draw the conclusion of the whole activity.

Introduction

Initially, a conspicuous number of analyses starting from neutronic calculations, and thus those about generator efficiency, cold well sizing, circuit definition, control, etc. were carried out. The target was to focus the main aspects of the system and to yield indications for a motivated choice of the main specifications for the final study. This was a demanding and time consuming activity; however this surveying activity will not be described in this document. The fuel enrichment is an important issue: the higher its value the lower the size and the mass of the reactor. However, the proliferation comes in, in the sense that uranium up to 20 % enrichment is not usable for a bomb, while uranium with 93 % enrichment is the “best” fuel for this military use. It is well known that the Nuclear Powers, led by USA, are against any action, which facilitates nuclear “proliferation”. This the reason why a significant fraction of our calculations referred to this 20 % enrichment. Approaching the end of the work it became clear that the proliferation political constrain was too heavy to be maintained, because of its design penalty, and by agreement with our technical interface the 20 % enriched fuel solution has been dropped.
The report is divided in three chapters and an appendix: the first chapter devoted to PWR, the second to HTGR. These chapters start from the neutronic calculations to define the reactor core, then pass to the electrical generator, the primary system, the reactivity control, the cold well. Successively a list of open issues and a preliminary indication on the potential R&D program required, the conclusions of the feasibility study and the complete list of data. The third chapter is a synthetic comparison between the two systems. The appendix details the results of the Italian industry inquiry.

The PWR

The idea is to extend as much as possible the PWR technology adopted for producing high powers in terrestrial applications to the design of a reactor suited for space conditions. However a number of modifications are needed. Let us summarize them.

Fuel composition: conventional powder of 93 % enriched uranium oxide, sintered in very small pellets. **Pellet diameter:** the chosen value is 1.8 mm, four times lower than the smallest current pellet. Fabrication process is to be defined. **Cladding material:** Stainless steel. **Cladding thickness:** 0.2 mm. **Fuel rod size:** the outer diameter is 2.2 mm, while the length is a design parameter, because it results from the core size, which is a cylinder with the diameter equal to the height. **Fuel bundle:** 19 rods are assembled in hexagonal geometry, and inserted in a hexagonal stainless steel shroud with a thickness of 0.3 mm. **Fuel burnup:** an average initial value of 60 MWd/kgU (maximum value) is assumed, which is about the same value presently adopted in PWRs.

Temperatures and pressures: the maximum operating pressure is assumed identical to PWRs, i.e. 15.5 MPa. The maximum temperature is set equal to the saturation value: 345 °C, which is about 15 °C higher than that of PWRs, while minimum temperature at the inlet is assumed equal to 335 °C, which is 45 °C higher than that of PWRs.

Cold well temperature: lower temperature means higher efficiencies, but higher cold well size: a temperature of 165 °C is a reasonable trade off between these opposite requirements.

Electrical generator: three alternate designs are considered i.e.: *thermoelectric generator, Rankine steam cycle, Rankine organic fluid cycle*. The thermoelectric generator has been discarded in this case, because the relatively small temperature difference between average core temperature and cold well temperature gives too low efficiencies, around 2-3 %. The other two cycles are characterized by an efficiency equal to 12.5 % and 18 % respectively. This leads to two values of thermal power equal to 800 KW and 555 KW.

Minimum fuel quantity: set the thermal power, the burnup, the full power duration (4000 days), we obtain for the above thermal powers the following minimum uranium masses: 53.3 and 37.0 kg respectively. This is equivalent to have a maximum fuel power density of 13.2 KW/kgUO_2, which is lower than that of conventional PWRs (38 KW/kg), while the linear power rate is much lower 0.39 against 17.8 KW/m.

Core geometry and reflector: the core geometry is based on the assumption to have a cylinder with the diameter equal to the height. The reflector is a layer of 12 cm of water all around the core.

Primary pumps: the industry has in advanced stage of development the technology of "spool pumps", which can be fully inserted in the primary circuit without any seal, because the motor can operate at high temperature inside the coolant.

Neutronic design. The program WIMS (*Winfrith Improved Multi group Scheme)* is a deterministic computation program, which uses a wide variety of calculation methods to solve the reactor physics problems. WIMS gives the reactivity in an infinite mean, thus, to obtain the reactivity of a finite reactor, it requires in input the values of axial and radial *buckling,* which is a crucial parameter in this small size reactor. As the effective multiplication factor strongly depends on the buckling values introduced in input, it seemed important to compare the results obtained by WIMS with those of a Monte Carlo program, which can be considered as an *exact* program. The comparison was made in four specific points and namely: infinite lattice and actual reflected reactor at Beginning of Life, in cold and hot conditions and by varying the moderation ratio. The Monte Carlo code here used is the well known MCNP-4C, as distributed by NEA Data Bank. The comparison turned out positive (see Figs 2.7 and 2.9), giving an indication that the WIMS should converge at the End of Life to a k_{eff} = 1.000.

The neutronic design suggested to reduce slightly the above determined minimum fuel mass for the 800 KW core, for a the moderation ratio of 6.5; for the 555 KW core the fuel mass is equal to the above mentioned value and the moderation ratio is 7 (see Figs 2.11 and 2.12). Figures 2.13 and 2.14 show the final fuel channel disposition in the 800 KW and 555 KW case, respectively. It is interesting to note that the core size is not much different for the two required powers.

The primary system. The primary system is made by the reactor vessel which contains *the reactor core, the barrel, the steam generator, the pressurizer, the circulating pump, the safety valve, the reactivity control mechanism and the instrumentation*. All these components are inside the reactor vessel, adopting the so called *integrated layout.* This allows to keep the size and the mass of the primary system to a minimum. Water flows upward through the core and then through the lower part of the upper plenum (the remaining part is filled with steam for the pressurizer), where the flow direction is reversed and the coolant is directed downward through the annular downcomer region, between the core barrel and the vessel; in this annular space the steam generator is located; the primary water flows on the outer surface of the steam generator tube, exchanging heat with the secondary fluid (water or organic compound) till the lower plenum, where the suction of the circulating pump is located; then the pumped coolant enters the reactor core to close the circuit (see Figs. 2.22-23 for the layout of the primary system).

<u>The vessel</u> shape is a cylinder with hemispheric domes (see Figs 2.18, 2.19, and 2.20), made of steel. The steel recently adopted for PWR vessels has an allowable stress of 205 MPa. By using this value the thickness are calculated, which are approximately equal to 29 mm and 14.5 mm for the cylindrical portion and the spherical domes for both powers.

<u>The barrel</u> is a simple steel cylinder not undergone to any particular load. Its thickness is determined by the requirement to have a good rigidity and to reduce fast fluence on the vessel if necessary: a value of 15 mm has been assumed.

<u>The steam generator</u> design here proposed is different from the usual one, the main difference being that all the sensible components inside the pressure vessel - i.e. tubes, headers and nozzles crossing the pressure vessel wall - are compressed instead of being stretched, because the higher primary pressure is acting on the outer surfaces: strictly speaking, primary stresses are compressive. This means that deterioration mechanisms due to high stresses, such as fatigue, should inherently be eliminated. Taking into account the limited power to be transferred in this case, it has been decided to adopt a single tube in order to eliminate any instability phenomena due to parallel channels. This would imply to choose a reasonable high value of the diameter and the length of the tube. The design results seem well within the

existing experience, especially as far as the length and the secondary pressure drops are concerned. However the thermalhydraulic behavior of helix was not well studied in past; an experimental campaign is needed for its development, also to take into account the effect of lack or reduced gravity.
For organic fluid there are differences, which probably are self compensating, so that the overall SG surface may result almost equal to the water one.
The pressurizer is a rather complex system, which can be simplified by putting the pressurizer in direct connection to the vessel (in the upper dome in our case) and bringing the outlet temperature to the saturation value, as here done. An abundant free steam volume, as 30 liters per MW, which is several times the value used in conventional PWRs, is adopted. This means in our case 24 and 17 liters for 800 and 555 KW reactors respectively. The water spray is then eliminated. These volumes are only a fraction of the upper sphere volume, which is equal to about 90 liters. Besides this free volume we have to foresee the possibility to contain the water expansion between cold and hot conditions; in fact the specific volume increases by a factor 1.64, going from ambient temperature (on the earth) to the average reactor temperature of 340 °C. This means that there are two alternatives: discharge the excess of water to an ad hoc vessel or to leave a initial *void* inside the cold vessel exactly equal to the above volume difference.
The circulating pump is of the *spool type,* which has been used in marine applications and designed for chemical plant applications requiring high flow rates and low developed head. The motor and pump consist of two concentric cylinders, where the outer ring is the stationary stator and the inner ring is the rotor, that carries high specific speed pump impellers (see Fig 2.21). The spool pump is located entirely within the reactor vessel; only small penetrations for the electrical cables are required. High temperature windings and bearing materials are under development.
The reactivity control. The PWR has inherently favorable features for control requirements, since it is characterized by a negative reactivity coefficient of temperature, which makes the reactor a *load follower.* In particular, in this reactor the temperature coefficient is still higher than in PWRs: approximately -300 against -30 pcm/°C. The overall reactivity to be controlled is about 28000 pcm against 24000 of current PWRs. The differences between PWR and this reactor are due to the high enrichment and to a reduced extent for the low power density. The control of this reactivity excursion is not an easy task. In this reactor the leakage of neutrons is so high, that a reflector poisoning may be enough to reduce the reactivity. This is to be thorough by verified. If this is the case, the control rods can be imagined not going up and down into the reflector, but made by a rotating cylinder, having on its diameter the poison plate. The rotation varies the angle of the poison plate, and then its neutron absorption capability. Here a different proposal is put forward: the principle is shown in Fig. 2.24 and 2.25: the core is divided in six moving slices each having a mass grossly of20 kg, operated by a single mechanism (to be defined). The specification is that by moving apart the slices in outside direction up to a maximum equal to the thickness of the reflector, the reactivity decreases slowly to a minimum equal to that required for the overall control. WIMS and Monte Carlo calculations show that the reactivity first rises, because the core is under moderated, then reaches a maximum and afterwards goes down rapidly: at 12 cm of distance the reactor is no longer critical.

The cold well. It is one of the most crucial component of any thermodynamic cycle for space application . Referring to the solution adopting the steam Rankine cycle with a power of 800 KW and a net efficiency of 12.5 %, the thermal power to be dissipated in the condenser is 700 KW. For the time being, only radiation has been considered. In a preliminary optimization study the conclusion was reached that the optimum condenser temperature for minimizing the overall mass is around 165 °C. By assuming a tentative view factor equal to 0.6 and a back radiation of an average temperature of 300 K, the specific surface results to 1.14 m^2/KW and then a total surface of 796 m^2. The condenser geometry is made by a bundle of 464 titanium tubes of ID/OD = 6/6.84 mm connected in parallel, having an overall weight of 1840 kg. The condenser can be imagined as a cylinder of 8 m diameter and 10 m height. In the case of organic Rankine cycle the power to be dissipated is 455 KW. Thus, adopting the same tubes, the condenser would be a cylinder of 5.3 m diameter and 10 m height.

Figs. 2.29 and 2.30 show a simplified scheme and the layout of the plant for the 800 KW reactor, respectively.

Preliminary safety considerations. A detailed safety analysis is outside the scope of this feasibility study, for its complexity and need to define the detailed requirements. In fact this reactor from one side is nor subjected to the licensing procedure of terrestrial reactors imposed by the safety authorities, from the other it must satisfy specific safety issues connected to its launch and the possibility to fall down to the earth:

- no irradiated fuel is present at launch;
- the core subcriticality in the case of possible launch accidents (flooding);
- the radiation protection without impairing mass requirements;
- an easy decommissioning in space.

The first item is inherently satisfied, because the reactor would not reach its first criticality before being outside terrestrial space. The second one seems inherently satisfied because a water reactor cannot be *flooded.* The third is a an important issue, which can be addressed only after having defined some conditions, especially for the propulsion solution. The fourth one is too indefinite at this stage of the design, that no specific consideration can be drawn.

In this study, a calculation has been done to verify whether in the case of severe accidents the fuel melting is avoided. If the fuel is no longer cooled by the water, the fuel heats up adiabatically till it reaches its melting point. However, as soon as the fuel temperature rises, a thermal radiation process takes place, the importance of which increases rapidly with the temperature. This radiation power is exchanged among the rods inside the core and from the outer rods ring toward the vessel and then from the latter toward the outside environment. Besides the radiation, there is also the convection of steam or air, which flowing inside the hot core brings its heat to the vessel walls and from them to the outside world. It is really difficult to simulate this situation by a model. A rather simplified but sufficiently realistic one has been prepared, limiting conservatively the study only to the radiation process. The results show that the maximum temperature is far from the melting point of stainless steel (1700 K) and even more from that of uranium oxide (3000 K). However, this analysis should be improved in the future to take into account the shrouds, the radial and axial flux distribution and the effect of rod pitch.

Open issues and R&D needs. This feasibility study has allowed us to find a first list of open issues to be solved for going on this route, which need a R&D program. The issues here below written in **bold letters** are those interesting for terrestrial reactors and those written in *italic* those interesting for generic terrestrial applications as well.

- Fuel
- Internals: mechanical design;
- **Increase of operating pressure: fuel implications, primary circuit materials;**
- **Saturation temperature at the reactor outlet: effect of small boiling inside the core;**
- The cold well as condenser;
- *Small steam turbines;*
- *Organic fluids: type, stability, thermal transport capabilities; Small organic fluid turbine;*
- **Fluid leakage: how much, how to cope with;**
- **Maintenance requirements of the whole system;**
- **Optimum reflector: technological aspects**
- **Pumps: development of spool pumps, reliability for long periods;**
- Fluence effects on vessel in these particular conditions;
- Shielding;
- Safety valves: reliability, how to cope their intervention;
- *Vessel material different from stainless steel*;
- **Steam Generator thermalhydraulic behavior in helical geometry** also in presence of low or no gravity;
- **Corrosion deposits inside the SG tube;**
- **The pressurizer: self pressurization, different concepts for propulsion reactor as feed and bleed, cold pressurizer,** centrifugal action**;**
- Control of the system and of the reactor and its constructive implications.

Even if this list is incomplete, no item seems to be unsolvable. An R&D program of reasonable extent may yield the needed answers, but what is important that the most demanding researches also are of interest for the new generation Light Water Reactors. Thus a cost sharing action can be proposed and duly programmed, according to the time schedule of the commercial exploitations of these terrestrial reactors.

Concluding remarks. At the end of this very preliminary feasibility study about the use of PWR system for space reactors, it can be concluded that no insoluble issues have been evidenced, which would prevent of going on along this route in order to execute a more detailed design. Then it will be possible to draw a more justified conclusion about the usefulness to follow this solution. At the beginning of the study it was supposed that the solutions for propulsion and surface application might be the same. However, this hypothesis holds only partially, because the lack of gravity and of a soil render the propulsion solution rather different and more demanding than the surface one. In particular, two aspects have been outlined for propulsion reactors: the lack of steam water separation in case of lack of gravity (pressurizer, steam moisture separation), and the need of an autonomous radiation shield, which in surface reactors can be provided by the existence of a soil. On the other side, it was anticipated in the foreword that the use of space nuclear reactors should be approached gradually starting from the easiest application, which is that for surface use: this study is a confirmation of the statement.

In the short range, future design activities should address the detailing of many aspects of the analysis presented in this report and adding new ones. Among the first ones, concerning the core, the choices to limit the fuel burnup, the use of stainless steel instead of zircaloy for cladding and shroud, the reflector material should be reconsidered: in fact these conservative choices affect the reactor size, which is an important item to define the overall mass. While for the rest of the system: cold well (in forced convection as well), reactivity and plant control. The new activities are: radiation shielding, vessel fluence, safety aspects, choice of vessel material, overall layout, containment, leakage control, ancillary circuits for start up, coolant purification, radiolisis. and other exigencies. Moreover, at the end of this further activity a preliminary R.& D. program should be detailed.

The complete list of the obtained data for the PWR reactor is detailed in the final table of par. 2.11.

The HTGR

The idea is to extend as much as possible the HTGR technology adopted for producing high powers in terrestrial applications to the design of a reactor suited for space conditions. However a number of modifications are needed. Let us summarize them.

Fuel: conventional powder of 93 % enriched uranium oxide, sintered in micro spheres of 350 μm diameter. **Cladding material and thickness:** the fuel micro spheres are protected by four carbon based layers of overall thickness equal to 400 μm and then the outer diameter turns out to be 750 μm. **Fuel "rod", said *compact*:** the micro spheres are mixed with a graphite powder and then compacted to form an hexagonal rod or *compact* having the apothem of 3.8 mm, while the length is that of the reactor height and thus it is the result of the neutronic calculations to define the core size. **Fuel-moderator-coolant channels:** the moderator is graphite under the form of hexagonal blocks having the same length of the reactor height. The blocks have an apothem which depends on the moderation ratio. The blocks are drilled by hexagonal holes: six of them are for the compacts and one for the coolant (Fig. 3.1). The blocks are then assembled together to form the reactor core. **Fuel burnup:** an average value of 100 MWd/kgU (maximum value) is assumed, which is about the same value presently adopted in HTGRs.

Temperatures and pressures: the maximum and the minimum temperatures are in both case 900/735 °C respectively. The minimum pressure is the usual value of 3 MPa, while the maximum one depends on the optimum compression ratio (1.6 in the chosen cycle, giving a maximum pressure of 4.8 MPa). **Cold well temperature:** in this case the choice depends on the generator type adopted.

Electrical generator: two alternate designs are possible i.e.: *thermoelectric generator, the Brayton gas cycle*. The thermoelectric generator is a possible and interesting solution in this case, thanks to the relatively high temperatures. In fact a net efficiency of 4.5 % has been calculated (see details below). Thus for obtaining 100 KWe net power, the reactor thermal power is to be 2219 KWth. The direct Brayton gas cycle is characterized by a much higher net efficiency equal to 24 %. This leads to a value of thermal power equal to 417 KW.

Minimum fuel quantity: set the thermal power, the burnup, the full power duration (4000 days), the following minimum UO_2 fuel masses are: 100 and 20 kg of UO_2 respectively. The possibility to adopt the above minimum masses is strictly connected to the reactor neutronic design. Present calculations show that the minimum mass for the 417 KW does not minimize the overall reactor mass and it is to be increased to 100 kg UO_2 as well; this means that the maximum fuel burnup is much lower in this case and this can be positive for a better fuel performance. **Core geometry and reflector:** the core geometry is based on the assumption to have a cylinder with the diameter equal to the height. The reflector is a layer of 5 cm of graphite all around the core.

Turbine and compressor: these are two important components of the generator, which should undergo a thorough verification for these small sizes and high reliability needed for long period of time operating at very high temperature. In particular the gas leakage raises some concern, because if present, even if to a reduced extent, would determine big impacts on the system design: containment, reinsertion in the circuit at high pressure. This concern has been coped with in this study by the decision to put all the rotating machines inside the pressure vessel.

Neutronic design. In this case the WIMS (*Winfrith Improved Multi group Scheme)* calculation program has been used, as already done for the PWR solution. As the effective multiplication factor strongly depends on the *buckling* values introduced in input, the results obtained by WIMS have been compared with those of a Monte Carlo program. This comparison has been made in four specific conditions and namely: infinite lattice and actual reflected reactors for two powers at BOL, in cold and hot conditions, by varying the moderation ratio. The comparison turned out positive (see Figs 3.4 and 3.7), giving an indication that the WIMS should converge at the End of Life to a reactivity of 1.01. Then by adopting this value and UO_2 masses of 100 kg in both cases, the value of the moderation ratio turned out to be 9.5 for the 2219 and 7.5 for the 417 KW reactor (see Figs 3.8 and 3.9). Figs 3.10 and 3.11 show the final fuel channel disposition in the 2219 KW and 417 KW reactor, respectively. The core size is not much different for the two required powers. The overall mass (core+reflector) of the 2219 KW reactor is 2588 kg, while for the 417 reactor it is 2148 kg. The difference is rather small considering that the ratio of the two powers is 5.3.

The cold well. It must dissipate in the case of Brayton cycle 317 KW. By integrating the radiation equation and considering a back radiation from the surrounding environment at 300 K, a view factor of 0.6 and an emissivity of 0.90, the average weighted value is equal to 1.16 KW/m^2 , which corresponds to a surface of 273 m^2. By assuming 450 titanium tubes of 6 mm ID and 0.5 mm thickness, the condenser can be imagined to be a cylinder of 4 m diameter and 7 m height.

Electrical power generation system by the thermoelectric device. Over the temperature range typical of HTGR (700 – 1000°C), the best thermoelectric material is SiGe (see Fig 3.14). An optimization process has been developed in order to maximize the efficiency, minimizing the area of the radiators and the thermal power to be produced. The data obtained are: T_{hot} = 1085 K, T_{cold} = 729 K, ZT = 0.6442, η = 4.73 %, A_{rad} = 160 m^2. The net efficiency, calculated in order to take into account the system absorbed power (5 KW) is 4.5%. Anyhow the absorbed power may be more significant in this case, due to the rather high pressure drops of the helium circuit. A detailed design should be carried out to this regard.

In this case, for the cold well a heat pipe solution has been adopted. The chosen heat pipe consists of a sealed aluminium container, a working fluid compatible with the container, Freon and a porous structure in aluminium. The dimensioning has been realized considering: the sun irradiation is present, each thermoelectric module produces 10 W, the view factor of each heat pipe is 0.5. As each heat pipe is mounted on 1 thermoelectric cell 10000 heat pipes are considered, the diameter of the heat pipe is 7,5 cm in order to fit with the dimension of the thermoelectric cell. The radiator obtained in order to dissipate 2119 KW (2219 -100 KW) is composed of 10000 heatpipes, of a theoretical area of 159 m^2 and of a real area of 318 m^2, 135 mm long and of a total mass of 642 kg.

The primary system. This differs substantially between the two reactors. However, both adopt a semi integrated solution, where the rotating machines are put inside the pressure vessel. Then in the 417 KW reactor the turbine, the compressor and the alternator are integrated inside the pressure vessel, while in the 2219 KW reactor only the compressor and its own motor are integrated inside the vessel (see Fig.3.18 and 3.19). In this case also the best pressure vessel shape is the cylinder surmounted by hemispheric domes: the inner dimensions are sketched in Figs.3.20 and 3.21 for the 2219 KW reactor and in Figs.3.22 and 3.23 for the 417 KW reactor.

The pressure vessel: The design pressure of the primary system is equal to the operating pressure multiplied a factor of 1.10, which takes into account the value of the safety valve setting. On this basis the pressure vessel thickness have been determined using the same steel adopted in PWR, i.e. stainless steel SA 508, Tp.3, Cl.2, with an allowable stress of 205 MPa.

The regenerator is a crucial component both for the size, transferring a power of 2.75 times that of the reactor (1153 against 417 KW), and the high temperatures and pressures involved (maximum values 750°C and 4.8 MPa). An approximate design gives an overall surface is 26 m^2.

The reactivity control. The reactivity excursions are in this case lower than those of PWR, and depending on the reactor power: 2500 and 8200 pcm for 417 and 2219 KW reactors respectively. Like in the PWR solution, the only possibilities are burnable poisons and control rods. The latter can be inserted in the reflector as already foreseen in some high power HTGRs. These *rods* can be imagined as rotating devices, as already explained for the PWR solution, or channels flowed by a *fluid* made by poisoned graphite balls inserted or extracted from the core reflector, by means of a suitable pneumatic mechanism. The problem of reactivity control seems more viable than in the PWR reactor, however the lack of a negative temperature coefficient may render the system control more delicate, implying probably a continuous operation of the control rods.

Preliminary safety consideration. A detailed safety analysis is outside the scope of this feasibility study, for its complexity and need to define the detailed requirements. In fact this reactor from one side is not subjected to the licensing procedure of terrestrial reactors imposed by the safety authorities, from the other it must satisfy specific safety issues connected to its launch and possibility to fall down to the earth. In the foreword it is mentioned that this nuclear system must satisfy the usual safety requirements of terrestrial reactors and this is what is to be defined in detail, taking into account the above consideration about the lack of a licensing procedure. Besides this the system has to assure that:

- no irradiated fuel is present at launch;
- the core sub criticality in the case of possible launch accidents (flooding);
- the radiation protection without impairing weight requirements;
- an easy decommissioning in space;

The first item is inherently satisfied, because the reactor would not reach its first criticality before being outside terrestrial space. The second one is a rather crucial one, because it requires the need to insert high absorbing materials in the core, to be extracted when the reactor will start up. Probably this is a rather demanding requirement, which deserves a specific consideration. The third one is a an important issue, which can be addressed only after having defined some conditions, especially for the propulsion solution. The fourth one is too indefinite at this stage of the design, that no specific consideration can be drawn. These reactors have the inherent feature to resist to the consequences of a LOCA, without provoking the fuel melting.

Open issues and R&D needs. This feasibility study has allowed us to find a first list of open issues to be solved for going on this route, which need a R&D program. The fuel is identical to that foreseen in terrestrial reactors, and then it can be assumed that it is or will be developed by already existing R&D programs. The issues written in **bold letters** are those interesting for terrestrial reactors and those written in *italic* those interesting for generic terrestrial applications as well.

- Reactor vessel internal layout: temperature distribution, **wall cooling, internal passages, mechanical design**;
- **Pipe design to resist to high temperature flowing fluids**;
- **Increase of operating pressure: primary circuit materials;**
- The cold well as cooler
- The cold well associated to thermoelectric device;
- *Heat pipes*
- *Gas turbine and compressor working in high temperature environment;*
- *Alternator working in high temperature and pressure environment;*
- *Thermoelectric apparatus;*
- **Fluid leakage: how much, how to cope with;**
- **Maintenance requirements of the whole system;**
- **Optimum reflector: technological aspects**
- Fluence effects on vessel in these particular conditions;
- Shielding;
- Safety valves: reliability, how to cope their intervention;

- *Vessel material different from stainless steel*;
- **The regenerator: thermal, mechanical corrosion issues;**
- **Control rods;**
- Control of the system and of the reactor and its constructive implications;
- Flooding danger avoidance.

The R&D about thermoelectricity at high temperature is of paramount importance for this reactor, because if present efficiency can be improved and assure the long term reliability by a suitable choice of materials, a thermoelectric apparatus might became the right solution for this nuclear system, instead of the much more complex Brayton cycle. Moreover, if the improvements may be obtained also at lower temperatures as those typical of PWRs, the present choice to eliminate this option for these reactors should be reconsidered. Even if this list is incomplete, no item seems to be unsolvable, and the lack of fuel development activity is greatly advantageous. On the other hand, some high temperature design issues appear demanding especially for long term operation. An R&D program of reasonable extent may yield the needed answers, but what is important that the most demanding researches also are of interest for the new generation High Temperature Gas Reactors. Then a cost sharing action can be proposed and duly programmed, according to the time schedule of the commercial exploitations of these terrestrial reactors.

Concluding remarks. At the end of this very preliminary feasibility study about the use of HTGR system for space reactors, it can be concluded that no insoluble issues have been evidenced, which would prevent of going on along this route in order to execute a more detailed design. Then it will be possible to draw a more justified conclusion about the usefulness to follow this solution.

At the beginning of the study it was supposed that the solutions for propulsion and surface application might be the same. Actually, it seems that this hypothesis holds more in this reactor than in PWR, because the lack of gravity does not determine any particular detriment to reactor operation. However, it remains the need of an autonomous radiation shield, which in surface reactors can be provided by the existence of a soil. On the other side the safety problem connected to a possible flooding seems rather demanding, also because the fuel cannot be separated from the moderator during the launch phase. If it will be confirmed in prosecution of the work that no insoluble issues are present in this proposal, it can be stated that a reasonable R&D effort and consequently a relatively limited development cost and time interval are only needed in this case.

In the short range, future design activities should address the detailing of many aspects of the analysis presented in this report and adding new ones. The new activities are: radiation shielding, vessel fluence, control, safety aspects, cold well design (in forced convection as well), choice of vessel material, vessel layout, system layout, regenerator design, containment, leakage control, ancillary circuits for start up, coolant purification and other exigencies. Moreover, at the end of this further activity a preliminary R.& D. program should be detailed.

The complete list of the obtained data for the HTGR reactor is detailed in the final table of par. 3.11.

Comparison between the two solutions.

The critical comparison between the two solutions here proposed is difficult to carry out without any external information about specific technological issues. Let's start with the PWR. The crucial issues are:

- The fuel;
- The steam turbine;
- The reactivity control;
- The pressurizer (only for propulsion solution).

The fuel is not a real technical issue, apart from the need to test it in a long irradiation program, having a so small diameter. However, other fuel alternatives are possible, as the use of a high alloyed uranium metal, similar to the one already studied for fast reactors and probably extensively used in submarine propulsion. Therefore, there are two possibilities: i) if the already existing information about fuel adopted for special reactors will become available, no specific R&D program is needed; ii) if this is not the case, a rather long and expensive R&D program is needed in order to obtain the green light to adopt this new fuel.

The steam turbine is of paramount importance for this system. These small turbines are not already developed, even if there is no particular reason to not reach such a goal, taking into account that some decrease of their efficiency is acceptable in this application. In particular, there is the leakage issue, which can impair the long term reliability of the overall system.

The reactivity control is a rather crucial mechanical issue.

The pressurizer working in absence of gravity, where steam and water cannot separate each other, is a demanding component, which for this reason no proposal has been advanced in this report.

Passing to HTGR system the crucial issues are:

- High temperature components;
- Leakage;
- Thermoelectric generator;
- Criticality during flooding:

The fuel in this case is not a problem, because the elemental micro sphere is absolutely identical to that foreseen for commercial reactors.

The high temperature is a big constrain for this reactor. In principle, there is no differences in this framework with the analogous commercial reactors. However, the high and durable reliability here required raise preoccupations about this issue. The gas leakage seems more important in this reactor than in the previous one, because helium is a mobile gas and difficult to collect, once escaped from the system. It is an aspect which requires a careful analysis. In principle, it may be supposed that this issue is more crucial for the Brayton cycle case than for that of the thermoelectric generator. The latter is the hope and the problem of this reactor. If a reasonable efficiency connected to a high reliability and durability can be demonstrated by such a device operating at high temperatures, a big push in favor of this reactor will be obtained.

The criticality danger during an accidental fall down on the sea, is not an easy task to cope with.

In the above, the cold well issue has not be mentioned in both reactors. Two are the reasons: i) it is believed that this is an optimization problem, maybe difficult and demanding one, but not unsolvable, ii) the component is not specific to these reactors and so a general and generic R&D program should be launched for this component.

In conclusion, it is clear from the above considerations how much important can be the contribution of already existing experience and knowledge to simplify substantially the R&D program needed for these reactors, but this is out of our reach.

It is probable that PWR is less suited for propulsion than for surface application, because of the lack of gravity, which makes the pressurization control a complex task. On the other hand, maybe that the a priori better reliability of such a reactor makes it more fit for surface application than the HTGR.

As for the masses, higher values are obtained for the HTGR, but the uncertainties of this estimation and the need of further ancillary components and circuits are probably higher than the differences with the PWR masses. However, an important aspect is the very low influence of the power level on the overall mass of the HTGR system, which, if confirmed, may become an advantageous item by increasing the power.

Polimi Survey on Italian Companies

The Appendix reports the results of a preliminary and surely incomplete survey on the Italian Companies operating in the nuclear industry, in many cases with experience in the space field, and interested in pursuing R&D activities related to the exploitation of the nuclear technology for the Mars exploration project. A lean format has been prepared to summarize the information: i) a brief description of the Company and its capabilities, ii) a list of possible R&D activity fields of interest for the Company, directly connected to the design of the nuclear reactor components and systems, and in some cases with technological spin-off in other industrial fields than the nuclear and space ones.

The list of the Companies that answered to the request are: ALCI, ANSALDO Nucleare, D'Appolonia S.p.A., FBM Hudson Italiana S.p.A., LABEN S.p.A., Peltech s.r.l., Silena International S.p.A.

From the compiled format it turned out a deep interest of these companies to be involved in the development of technologies needed for space reactors. The possible R&D activities with potentiality for technological spin-off are

- **(I) Thermoelectric devices**
- **(II) Components for the Heat Exchange processes**
- **(III) Mini or micro Steam turbines and Gas turbines.**

Foreword

Ambitious solar system exploration missions in the near future will require robust space power sources in the range of 10 to 200 KWe. Fission power systems are well suited to provide safe, reliable, and economic power within this range. Conventional chemical systems are near their theoretical performance limit and have very low energy density (energy released per unit mass). Solar power systems rapidly lose effectiveness as their travel farther from the sun, and are affected by orientation, radiation field, debris and eclipses. Radioisotope systems are limited to an energy density many orders of magnitude below fission, and long term supply of plutonium 238 is unknown. Each of the aforementioned power sources has a valuable place in space exploration, but only fission can truly enable ambitious exploration in the near term.

Therefore the goal of this research program is to carry out a preliminary feasibility study of a nuclear fission reactor suited for space applications. These refer either to rocket propulsion by electricity (*NEP: Nuclear Electric Propulsion*) or to electrical power production for stationary settlements (manned or unmanned) on some planet, or deep space planetary surfaces (*Mars*), or satellites (*Moon*).

This application of nuclear energy is very demanding and it should be addressed in a gradual way, because numerous space fission power programs failed having tried to do too much too soon. Then a good option for developing the reactor-related portion of this infrastructure and experience is to start by developing and utilizing a low-power surface fission power system: surface applications generally place less demanding requirements on the reactor and integrated system [6]. Even if this study concerns both applications, the solutions envisaged better apply to surface applications.

The application of nuclear energy to space needs has been considered since a long time in many demanding and expensive feasibility studies and tests under the form either of radioisotope thermoelectric generators (RTG) or real nuclear fission reactors. While the first ones have been extensively used on satellites, very few nuclear reactors were actually used in space: e.g. reactor SNAP-10A was launched in 1965 and it remains the only nuclear fission reactor launched by a Western nation.

The above goal seems to be a very complex and not a clear cut one. Therefore, the present study is a preliminary one, which in principle cannot have the ambition to give a priori a well definite answer to the problem, in the sense to reach by certain a viable proposal fit for a subsequent specific R&D program. The real goal is to see whether it is possible to develop a reliable and cheap reactor for the above mentioned space applications, possibly to be adopted in particular terrestrial civilian (industrial) applications as well, and to identify potential civilian interest in subsystem technology developments.

A space nuclear reactor should respond to the following general requirements:

1. To be extremely reliable;
2. To imply an R&D program of moderate cost;
3. To be deployed within a reasonable period of time;
4. To be operated and controlled for a long time without intervention;
5. To be able to be transported into space (mass and size limit);

6. To be also used as a byproduct for some particular terrestrial application (or at least to share common technologies).

The first item is quite obvious and does not need further explanations. This means that the chosen reactor type must be extensively and positively tested in terrestrial applications, and then too innovative proposals are a priori excluded, at least in the medium period. This conclusion is reinforced by the following considerations. Items 2) and 3) can be discussed together. The R&D of a new type of nuclear reactor requires huge investments and long development times. Typically, it can be stated, as a rule of thumb, that the costs are of the order of 2-2.5 billions € and the time span of 25-30 years, excluding the construction of the FOAK (First Of A Kind plant). These figures cannot be demonstrated by specific studies to this regard, but are based on the common opinion of experts working in this field for terrestrial reactors. It can be argued that terrestrial reactors are characterized by a much more expensive R&D program for their much higher power and much more demanding licensing procedure. However, this consideration certainly applies to the FOAK construction cost, but only to a limited extent to the R&D costs, when it is necessary to develop new materials, new processes and new systems. The above costs are also predicted for Generation IV nuclear reactors, an international initiative started in 2000 and aimed at developing new innovative reactors. If the innovative reactor foresees the development of a new fuel or a very exotic system, this will negatively affect the above figures. Moreover, if the proposed concept at the end of a relevant R&D program does not respond to the needed specifications, it must be abandoned. This was not an unusual situation in the early stages of nuclear energy development for civilian applications, when many concepts of nuclear systems and fuels were definitely abandoned after long and demanding research programs. Obviously, these objections loose their validity if these innovative reactors are already developed to a significant stage, but this does not seem the case at least for Europe.

Item 4) is important and again in favor of simple and reliable solutions. In fact previous designs foresee rather short lives, which are not coherent with the needed specifications [7]. Item 5) is quite obvious.

Item 6) is motivated by the usefulness to have an economic return of R&D costs from other non space applications of the same reactor concept. It is a reasonable requirement, which needs a careful inquiry. In fact, small nuclear reactors can be used in several specific applications, even if some difficulties may arise from the Nuclear Safety Authorities responsible to give the licensing to nuclear applications. However, this might result less problematic from a technical viewpoint than now imagined for very small simple and reliable reactors, but the energy costs are in any case so high to exclude current applications. Anyway, it seems possible and probable that some technologies needed for space reactors can have a terrestrial application in nuclear and non nuclear systems.

All the above considerations taken into account, it can be concluded that this reactor type should be:

- Based on the well proven technology of present terrestrial reactors, allowing obviously the development of different components and systems needed to accomplish the specific mission of a space reactor, according to known processes,
- Suitable for propulsion and stationary applications, apart from reasonable and moderate differences.

If these conclusions are accepted in this context, the first result is that the propulsion reactor has to produce electricity, in the same way as the stationary one, and its electricity will be used for propulsive scopes, by adopting suitable converting apparatus downstream the reactor.

The rationale of the above choices is the need to reduce the overall R&D economic burden to 0.5-0.6 billions € and the development time to 10-12 years. In fact, these figures are those predicted for the International Near Term Deployment (INTD) reactors, a parallel initiative of the above mentioned Generation IV one, which has the goal to make new nuclear reactors deployable in a reasonable period of time. These reactors do require the development of new components (excluding fuel) and systems, but always based on conventional and well proven processes, as here supposed. A further advantage derives from the mutual potential benefits of a common development of some specific component, and technology.

A space reactor must satisfy a number of requirements, besides the general ones presented above. A non exhaustive list is as follows:

- produce an electrical power around 100 KW;
- last a long period of time (around 4000 days) without any intervention and fuel supply;
- minimize the overall mass and volume for rocket payload constraints;
- use high enriched uranium;
- adopt a core power density substantially lower than that of current reactors;
- satisfy the usual safety requirements of terrestrial reactors and besides this to assure:
    - no irradiated fuel is present at launch;
    - the core subcriticality in the case of possible launch accidents (flooding);
    - the radiation protection without impairing mass requirements;
    - an easy decommissioning in space;
- a simple control of the reactor and the overall plant;
- a substantial reduction and simplification of maintenance and repairs;
- avoid any leakage of the contained fluids or implement systems to recuperate them;

Two problems of paramount importance are the design of the heat sink and the way to produce electricity. The heat sink is a high demanding component, because the lack of atmosphere or a much rarified one require extended dissipation surfaces and thus a big mass and a high risk to be hit and damaged by micrometeorites.

The electricity may be produced alternatively in two ways: a thermoelectric device or a simplified conventional generator. The first, a very proven technology also in space, does not have moving parts and thus has an high reliability. The intrinsic redundancy of the process is very high, because it is based on several hundredths thermoelectric components, and the destruction of some of them by micrometeorites is acceptable. On the other hand the efficiencies are very low as it will be shown here below. The situation is practically reversed for simplified and conventional generators: from one side, the presence of rotating or moving components implies a lower reliability and more demanding maintenance requirements, and the impact of micrometeorites requires the presence of a special protection, while from the other one a much higher efficiency is obtained. As a matter of fact, the design is strongly affected by the efficiency. With a fixed electrical output, the thermal power of the reactor may range up to a factor a factor 5.5 and this means a substantial bigger core and a much more demanding heat sink. However, there are in the world research initiatives aimed at increasing thermoelectric efficiency, and in case of success the disadvantages of thermoelectric conversion would be reduced.

Proven reactor systems here applicable are those based on thermal neutrons and the use of: i) light water as moderator and coolant and ii) graphite as moderator and gas as coolant. In principle, also fast neutron reactors may be considered, but the present solutions based on sodium coolant are now less popular for terrestrial applications for a number of technical and economic reasons and the new proposals to adopt different metallic coolants (see the above mentioned Generation IV initiative) are to be extensively tested. Gas cooled fast reactors are in a very initial stage of development, they have to solve new and demanding problems and imply anyway the development of new fuels.

The first reactors are well known as LWR (Light Water Reactors) and they are subdivided between two different types: PWR (Pressurized Water Reactors) and BWR (Boiling Water Reactors), but only PWRs can be here considered as it occurred in the past for submarines and ships propulsion. In fact, the features of space reactors are more similar, in relative sense, to those of naval reactors than those of civilian reactors, and this can be seen as a meaningful starting point.

Thermal gas reactors evolved in three different generations, here we refer to the last one, known by the general acronym HTGR (High Temperature Gas Reactors), but different solutions are under consideration, adopting different acronyms. These reactors have the peculiar feature to generate heat at much higher temperatures than LWRs, typically 800-900 °C against 300 °C. This means higher thermodynamic efficiencies and the possibility to widen the nuclear energy exploitation to other industrial applications different from electricity production. However, the experience acquired up to now, even if significant, is not comparable at all to that of PWRs; in fact important R&D programs are under way in the world.

In conclusion, this program concerns a preliminary feasibility study of a space reactor, suited either for stationary needs on a planet or for propulsion, to produce electrical energy of the order of 100 KW. It was articulated in the following steps:

- Assume as a first choice the PWR solution as the reference system.
- Execute a rather detailed neutronic study of this reactor, which is two orders of magnitude smaller than conventional reactors (the power is three order of magnitude lower, but the power density is an order of magnitude lower).
- Define the preliminary scheme of the whole plant, under alternative solutions for electricity production: adoption of thermoelectric device or simplified conventional generators.
- Put in evidence the differences between propulsion and stationary reactor specifications and the way to fulfil them.
- Carry out analyses for a preliminary verification of its capability to satisfy the requirements listed above for a space reactor.

- Analyzing the gas reactors, outlining pros and cons of these reactors when compared to PWRs.
- Make a survey of Italian industry capability and willingness to participate to the development of such a reactor.
- Verify the potentialities of space reactors for particular terrestrial uses.
- Identify a research and development program including the aspects of interest for civilian (industrial) purposes in Europe;
- Draw the conclusion of the whole activity.

1. INTRODUCTION

The feasibility study which is synthetically described in this document concerns: two fission reactors types, PWR and HTGR; two applications, propulsion and surface applications, with more attention to the latter.

Other general specifications are as follows:

- Keep the electrical power equal to 100 KWe and adjust thermal power according to the actual thermodynamic efficiency value, obtainable by different generator alternatives;
- Adopt two different fuel enrichments: the maximum obtainable one equal to 93 % of uranium 235, and the maximum “non proliferating” limit equal to 20 %;
- Assure a fuel duration without any intervention of 4000 days (about 11 years);
- Adopt a fuel burnup coherent with the already existing experience for terrestrials reactors.

This taken into account, a conspicuous number of analyses starting from neutronic calculations, and thus those about generator efficiency, cold well sizing, circuit definition, control, etc. were carried out. The target was to focus the main aspects of the system and to yield indications for a motivated choice of the main specifications for the final study. This has been a demanding and time consuming activity, also because these nuclear systems are so far from those considered for terrestrial uses, that the already existing experience of the design group was not of much help. For instance, a companion program does concern the feasibility study of a PWR, but the power is 1000 MWth, while we are here interested in powers around 1 MWth: three orders of magnitude lower, which is undoubtedly relevant for the design. Other important differences concern: enrichment, cold well temperature, reduced or absence of gravity, and control. This surveying activity will not be described in this document.

Further details are needed about the fuel enrichment issue. The higher the enrichment the lower the size and the mass of the reactor and this foreseeable result has been confirmed by neutronic calculations in both reactors. However, the proliferation comes in, in the sense that uranium up to 20 % enrichment is not usable for a bomb, while uranium with 93 % enrichment is the “best” fuel for this military use. It is well known that the Nuclear Powers, led by USA, are against any action, which facilitates nuclear “proliferation”. This is the reason why a significant fraction of our calculations referred to this 20 % enrichment. Approaching the end of the work it became clear that the proliferation political constrain was too heavy to be maintained, because of its design penalty, and by agreement with our technical interface the 20 % enriched fuel solution has been dropped. On the other hand, it is fair to mention that any other proposal in the world disregards this issue. This choice from one side will facilitate the design, on the other it will render less probable a terrestrial utilization of the reactor. As said above these 20 % enrichment calculations will not be detailed here below.

The report is divided in three chapters and an appendix: the first chapter devoted to PWR, the second to HTGR. These chapters start from the neutronic calculations to define the reactor core, successively pass to the electrical generator, the primary system, the reactivity control, the cold well. Finally a list of open issues, a preliminary indication on the potential R&D program activities, the conclusions of the feasibility study and the complete list of data are detailed . The third chapter is a synthetic comparison between the two systems. The appendix details the results of the Italian industry inquiry.

2. THE PWR

2.1 Introduction

The idea is to extend as much as possible the PWR technology adopted for producing high powers in terrestrial applications to the design of a reactor suited for space conditions. However a number of modifications are needed. Let us summarize them.

Fuel composition: conventional powder of uranium oxide, sintered in very small pellets (see below).
Fuel enrichment: 93 % in uranium 235:
Pellet diameter: this is substantially different from that of current PWRs: the high enrichment imposes a small diameter in order to avoid unacceptable flux depressions inside the pellet. The smallest current pellet is that used in fast reactors: in Superphenix the pellet diameter is equal to 7.2 mm, and with this value the flux depression in the pellet center for 93 % enrichment is clearly unacceptable as shown in Fig. 2.1.

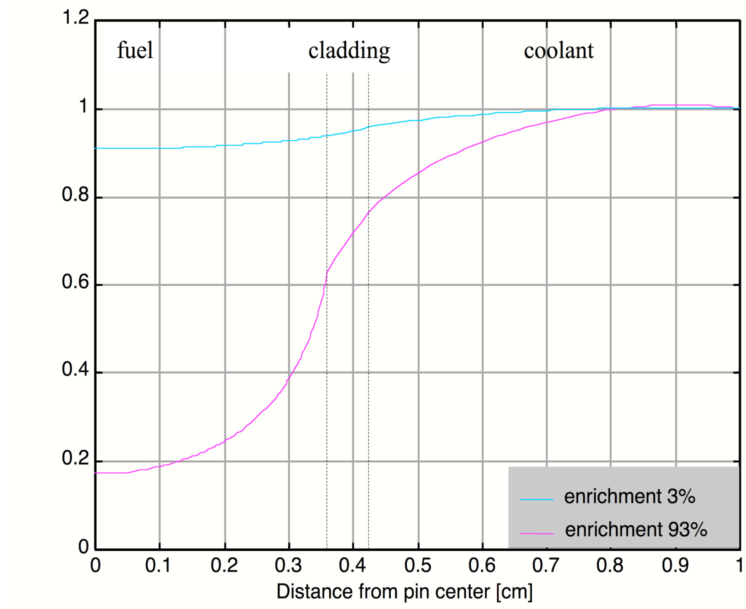

Fig. 2.1

Then a substantial diameter reduction is necessary and its value depends on fabrication limits. A check with a fabrication expert yields the conclusion that a reduction of a factor two is still possible with the current fabrication procedure, but this is still insufficient in our case. A further reduction of a factor two is reasonably obtainable, but requires anyway the adaptation of a different fabrication procedure. In conclusion, a reduction of a factor four has been adopted resulting in a pellet diameter of 1.8 mm. The corresponding flux depression is shown in Fig.2.2: the flux depression is still higher than the current one, but it was deemed acceptable. No commercial fuel industry in the world is licensed to fabricate fuel having an enrichment higher than 5 %. Therefore the supply needed for this application should be fabricated in special laboratories, under strict control in order to avoid proliferation dangers.

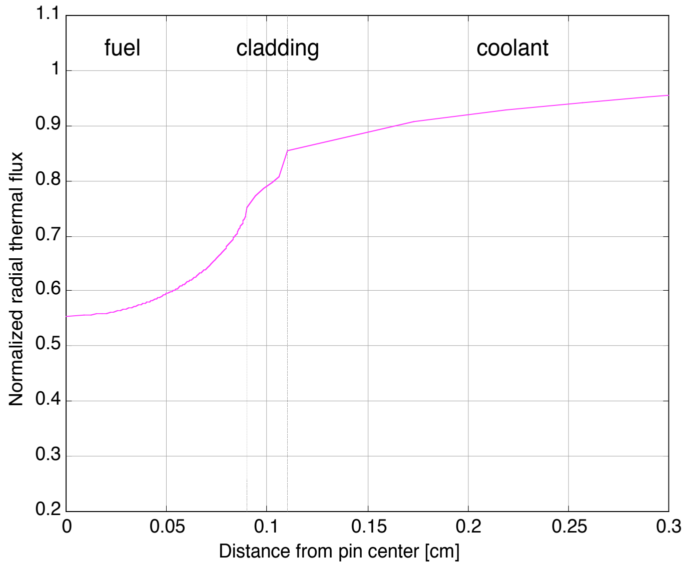

Fig. 2.2

Cladding material: Stainless steel: this choice does not derive from that of the Superphenix, but from the consideration that 4000 days of operation and a higher operating temperature (see below) might imply some concerns (corrosion, creep) in the case of Zircaloy, adopted in LWRs. On the other hand stainless steel choice, even if is a conservative solution, does not introduce a significant neutronic penalty because of the presence of high enriched fuel. This has been confirmed by a preliminary calculation.

Cladding thickness: 0.2 mm: this value is the extrapolation based on pellet diameter of the overall thickness of Superphenix rod, joining together the actual cladding and the gap thickness. In our case we decided to avoid now the design of the exact gap thickness, by adopting for this cladding a slightly reduced density equal to 6800 instead of 7800 kg/m^3 of pure stainless steel. Obviously this choice is to be duly verified by fuel performance calculations and possibly by in pile tests, but this is out of the scope of this study. However, the validity of neutronic calculations is scarcely affected by these data.

Fuel rod size: the outer diameter is 2.2 mm, while the length is a design parameter, because it results from the core size, which is a cylinder with the diameter equal to the height.

Fuel bundle: the rods are assembled in hexagonal geometry, which is better suited than the square one to obtain a regular shape of the core cross section, in the case of very small sizes. The rod number of the bundle is assumed equal to 19, a relatively low value, which is again justified by the need to have a regular core shape. The rod pitch is a design parameter, because it depends on the final value of moderation to fuel ratio, resulting from neutronic calculations.

Fuel channel: the fuel bundle is inserted in a hexagonal stainless steel shroud with a thickness of 0.3 mm, which is required for the main reason to adapt the channel flowrate to the fuel bundle power, in order to maximize the outlet temperature, set equal to the saturation value. Moreover, this choice allows an easy design of the pressurizer (see later on). While in western PWRs the fuel bundle form is square and no shroud is provided, in Russian VVERs the bundle is hexagonal, even if much bigger than this one, and a hexagonal shroud with perforated walls is adopted. A sketch of the fuel cross section in correspondence of moderation ratio 6.5 is indicated in Fig.2.3. As said for the cladding, the choice of stainless steel instead of zircaloy should be of limited effect.

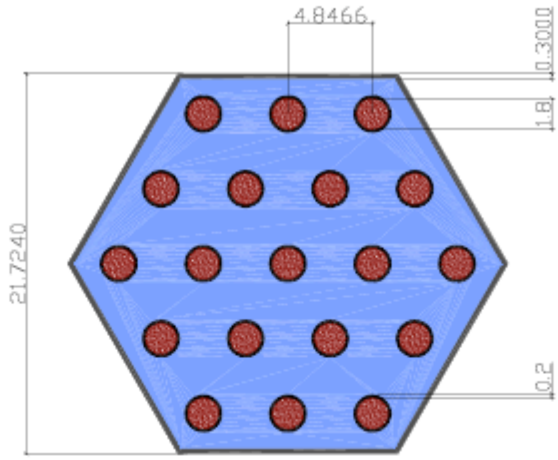

Fig. 2.3

Fuel burnup: a maximum average value of 60 MWd/kgU was assumed initially, which is equal to the value presently adopted in PWRs. In our case there are two opposite effects when compared to PWRs: i) the fuel power density is much lower and then also the corresponding maximum pellet temperatures: this is a real favorable condition, being the fuel performance much improved in these conditions; ii) the more peaked flux distribution inside the core and the lack of periodic fuel shuffling imply a higher maximum burnup value in correspondence of the same average one, then worsening the fuel damage effects in the maximum flux position. Thus the decision to maintain the same value seems completely justified. For the 800 KW core (see below), the fuel burnup has been subsequently increased to 77 MWd/kgU (see par.2.2.3).

Temperatures and pressures: the maximum operating pressure is assumed identical to PWRs, i.e. 15.5 MPa. The pressure is an important parameter to characterize the process, the fuel performance and the mass. An optimization study would be needed for an exact definition of this parameter; however, taking into account that the efficiency in our case is of paramount importance to contain the reactor thermal power and the cold well size, it can be imagined that a higher pressure would be convenient, but this would put our system outside of the present experience. In conclusion, the value of 15.5 MPa was confirmed. As for the maximum temperature, there are two requirements going in the same direction: i) to maximize it in order to improve the efficiency and ii) to have saturation temperature at the core exit, in order to use a self-pressurizer. The latter is possible only for surface application, because gravity is needed to separate liquid and steam. For propulsion applications a different type of pressurizer must be envisaged; some possible solutions are under consideration, but they will be not here detailed. Thus the maximum temperature is set equal to 345 °C, which is about 15 °C higher than that of terrestrial PWRs. The minimum temperature at the inlet is determined by fixing the temperature drop across the core. Usually this is about 40 °C, but in our case this value is too high. By considering that the fluid at the turbine inlet has a maximum temperature between the maximum and the minimum core temperatures, the efficiency is improved by increasing the minimum one. Besides, the choice to have saturation temperature at the core outlet, involves the possibility that slight incoherence between power and flowrate of a single channel might determine boiling inside it. The amount of this effect depends on the enthalpy rise across the reactor: the higher the enthalpy rise, the higher the amount of boiling. A further reason is the low channel velocity in this case, because of the very low power density, which is outside the thermalhydraulic current experience. All taken into account, a much lower temperature drop equal to 10 °C was assumed, that is an inlet core temperature equal to 335 °C, which is 45 °C higher than that of terrestrial PWRs. As above said, to reach a uniform outlet temperature, a specific flowrate for each channel is needed, according to its power. Fortunately, in the case of high enriched fuel its composition along the life is little modified, as shown in Fig.2.4. For instance, the enrichment passes from 93 % at Beginning Of Life to 82 % at End Of Life. Since the flowrate is adjusted by introducing fixed orifices at the channel inlet, their value is to be determined on the basis of an envelope of all radial power distributions along the fuel life.

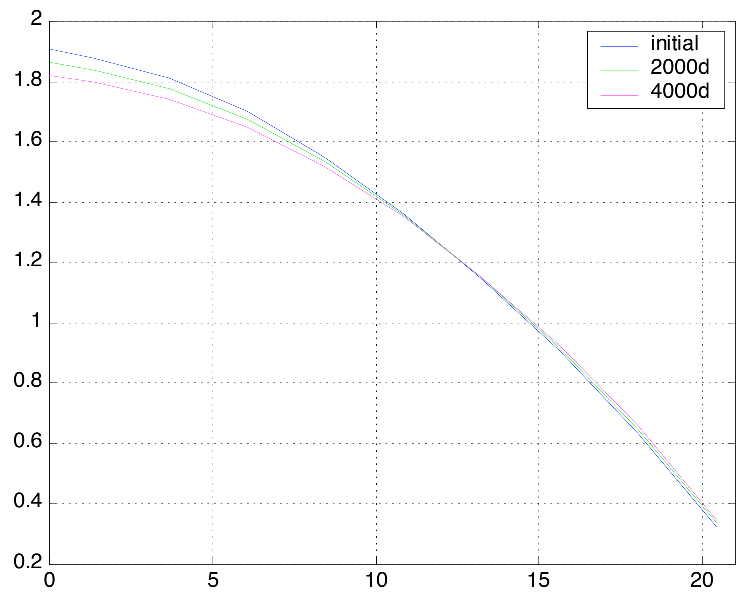

Fig. 2.4

Cold well temperature: lower temperature means higher efficiencies but higher cold well size. A preliminary optimization in order to minimize the overall mass has shown that a temperature of 165 °C is a reasonable trade off between these opposite requirements.

Electrical generator: three alternate designs are possible i.e.: *thermoelectric generator, Rankine steam cycle, Rankine organic fluid cycle*. The thermoelectric generator has been discarded in this case, because the relatively small temperature difference between average core temperature and cold well temperature gives too low efficiencies, around 2-3 %; this means a too high penalty in the overall system, even if this generator is highly reliable and experienced. The other two cycles are characterized by an efficiency equal to 12.5 % and 18 % respectively (see details in par.2.3). This leads to two values of thermal power equal to 800 KW and 555 KW.

Minimum fuel quantity: set the thermal power, the burnup, the full power duration (4000 days), we obtain for the above thermal powers the following minimum fuel masses for the initial assumption of 60 MWd/kgU: 53.3 and 37.0 kg respectively of U, or equivalent to 60.5 and 42 kg of UO_2 respectively. This is equivalent to have an average fuel power density of 13,2 KW/kgUO_2, which is lower than that of conventional PWRs (38 KW/kg UO_2), while the linear power rate is much lower 0.45 against 17.8 KW/m. The possibility to adopt the above minimum masses is strictly connected to the reactor neutronic design. In fact, the small size of these reactors, roughly two orders of magnitude lower than those of a conventional PWRs, implies big fractions of escaping neutrons from the core surface. Preliminary calculations show that these minimum masses are approximately enough for 93 % enriched fuel to satisfy the imposed fuel life. However, the requirement to obtain a reasonable moderation ratio would imply a slight variation of these masses.

Core geometry and reflector: the core geometry is based on the assumption to have a cylinder with the diameter equal to the height. Actually the neutronic optimum would be obtained for a ratio height to diameter of 0.92; however, taking into account the effect on the size and weight of the overall system it has been assumed a priori that a ratio equal to 1 would be simpler, better and not far from the usual approach. The actual size will depend on the needed mass of the fuel, which for 93 % enrichment varies only with power (also with burnup and life which are fixed in our case), and on the value of moderator to fuel ratio, which is the result of the neutronic design. The reflector is a layer of 12 cm of water all around the core. The reflector thickness is an important item because its increment serves for reducing the escaping neutrons, which are in percentage an order of magnitude higher than those in terrestrial PWRs, but on the other side it increases the mass and the size of the overall system. It is usual for these special nuclear reactors to adopt more sophisticated solutions for the moderator, which foresee the use of materials different from water (e.g. beryllium oxide) in order to optimize the above mentioned opposite requirements. However, this may be done in a prosecution of the program, but at the present stage the adopted choice seems reasonable, also because any other one would not improve substantially the overall result. Moreover it is coherent with the suggested way to control the reactivity (see later).

Primary pumps: natural circulation, initially considered, was then excluded for three reasons. On Mars we have a reduced gravity (38 % of terrestrial one), and then high elevation differences between the reactor and the steam generator (in this case located in a different pressure vessel above the core) would be necessary, moreover on a rocket the gravity is missing and to create it artificially trough a rotating platform would imply a big and probably unacceptable complication. The second reason is that forced convection gives a much higher design flexibility than the natural one. The third one is that the industry has in advanced stage of development the technology of “spool pumps”, which, opposite to canned pumps, can be fully inserted in the primary circuit without any seal, because the motor can operate at high temperature inside the coolant. In this case the only penetration in the primary circuit is that of the electrical cables to feed the pump motor.

All the reactor data, obtained on the basis of the previous specifications and the design calculation activity described in the following paragraphs, are detailed in par. 2.11

2.2 Neutronic design

2.2.1 <u>Design codes</u>

WIMS (*Winfrith Improved Multi Group Scheme)* is a deterministic computation program, which uses a wide variety of calculation methods to solve the reactor physics problems. It is suitable to study any kind of thermal reactors (1).

The program determines the spatial distribution of neutron flux by solving the transport equation. However, it requires some approximations:

- a simplified geometry: the real problem is approximated by 1D or 2D model, by imposing suitable border conditions;
- all the cross sections are averaged on discrete energy intervals (69 intervals);

Then WIMS solves a series of equations for the neutron flux for a given number of energy groups and space meshes. Moreover, it gives other important parameters as: reactivity, material compositions along the irradiation, power distribution, Xenon and other fission products transients, reaction rates.

Concerning the geometry the program allows to simulate different configurations:

- *Pincell*: infinite lattice of elemental identical cells; each one is made by cylindrical fuel rod axially infinite, surrounded by its cladding and its own moderator-coolant;
- *Homogeneous*: an infinite volume filled by a single material;
- *Slab*: a plane geometry made by parallel plates of fuel separated by parallel plates of moderator, axially infinite;
- *Cluster*: infinite lattice of identical clusters separated by the moderator: each cluster and the moderator is made by several concentric rings, each one having different material composition.

From this geometry description appears evident that WIMS gives the reactivity in an infinite mean, thus, to obtain the reactivity of a finite reactor, it requires as input the values of axial and radial *buckling*. These values are calculated outside the program by the usual formulas, in which the extrapolated core height

and radius are used. Here an issue rises up: the extrapolation length, usually a value which in large cores does not significantly affect the effective multiplication factor, in our small core case it turns out to be a crucial parameter. The best estimate we were able to do was adopting the formula which equals the extrapolation length to 3x0.71 times the diffusion coefficient. The latter parameter was deduced directly from the WIMS output file. The program gives the diffusion coefficient for each energy group, as well as a suitable averaged value, which is the one here adopted to calculate the extrapolation length. The values of axial and radial buckling can then determined in a straightforward manner. As the effective multiplication factor strongly depends on the buckling values introduced in input, it seemed important to compare the results obtained by WIMS with those of a Monte Carlo program. The Monte Carlo programs are highly reliable, but they have the drawback that they cannot be easily used to simulate the fuel evolution along the life. Therefore, this comparison was made in two specific points and namely: infinite lattice and actual reflected reactor at BOL, by varying the moderation ratio, in cold and hot conditions. Most k_{eff} calculations refer to 800 KW core and only a check has been done for 555 KW core.
The Monte Carlo code here used is the well known MCNP-4C, as distributed by NEA Data Bank [2]. The criticality calculation option (KCODE card) was used to determine the multiplication factor. The geometry described in the input file is fairly detailed, with a double level of lattices defined: the lower level describes the lattice of fuel rods inside a bundle, the upper level describes the lattice of bundles which forms the core.

2.2.2 WIMS and Monte Carlo comparison

The Monte Carlo and WIMS results are detailed in the following Tables and Figures. The reflector of 12 cm has been assumed, calculated starting from the equivalent diameter in both programs.

1. Tab. 2.1 gives the k_∞ values obtained by WIMS and MCNP-4C, and the k_∞ differences between the two programs, versus moderation ratio ranging from 5 to 11 (which is the range of the foreseen solution), both in cold and hot conditions; Fig. 2.5 and 2.6 show the MCNP-4C and WIMS k_∞ in graphical form in all the above conditions; while Fig. 2.7 shows the k_∞ differences between the two programs;
2. Tab. 2.2 gives the k_{eff} values obtained by both programs in cold and hot conditions versus same moderation ratio interval for the 800 KW core and surrounded only by the reflector, in correspondence of the initial fuel mass of 53.3 kg[1]. Fig. 2.8 shows k_{eff} of both programs in graphical form, while Fig. 2.9 shows the k_{eff} differences between the two programs;
3. Tab. 2.3 gives k_{eff} values obtained by MCNP-4C with and without barrel vessel and downcomer in hot conditions versus the same moderation ratio interval, and the relevant differences, for the 800 KW core, in correspondence of the initial fuel mass of 53.3 kg; Fig 2.10 shows the differences in graphical form;

[1] For these preliminary calculations this mass was assumed to be UO_2 instead of U.

4. Tab. 2.4 gives k_∞ and k_{eff} values obtained by MCNP-4C and WIMS, in hot and cold conditions and the relevant differences for the 555 KW core in correspondence of a single value of moderation ratio equal to 7 (equal to the foreseen solution). This calculation refers to a mass 42 kg of UO_2, which is coeherent with 60 MWd/kgU.

Tab. 2.1 – K_∞ : MCNP , WIMS (Beginning of life)

Geometrical data:
- Fuel diameter [mm] 1.8
- Cladding thickness [mm] 0.2
- Shroud thickness [mm] 0.3

Temperature (hot):
- Fuel [°C] 363
- Cladding [°C] 343
- Water [°C] 340
- Shroud [°C] 340

Temperature (cold):
- All components [°C] 27°C

	WIMS		MCNP		Delta K: MCNP - WIMS	
Mod. ratio	Cold	Hot	Cold	Hot	Cold	Hot
5	1.76620	1.73068	1.79095	1.76100	2475	3032
6	1.77903	1.73979	1.79914	1.76761	2011	2782
7	1.78912	1.74952	1.80517	1.77519	1605	2567
8	1.79670	1.75883	1.80879	1.78140	1209	2257
9	1.80211	1.76725	1.81201	1.78704	990	1979
10	1.80572	1.77470	1.81401	1.79287	829	1817
11	1.80782	1.78115	1.81329	1.79574	547	1459

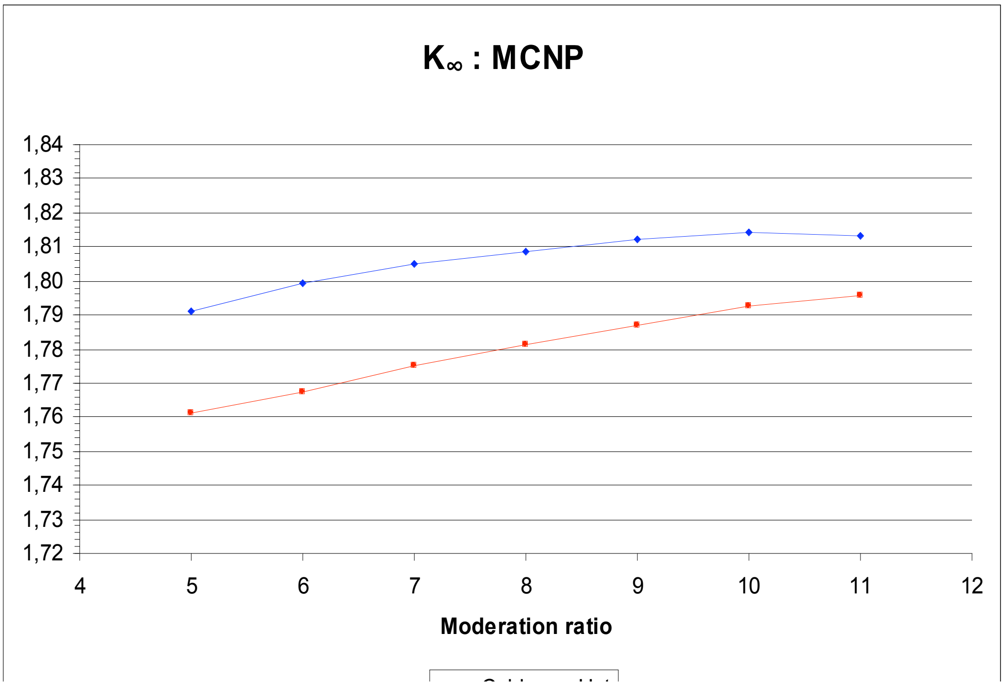

Fig. 2.5

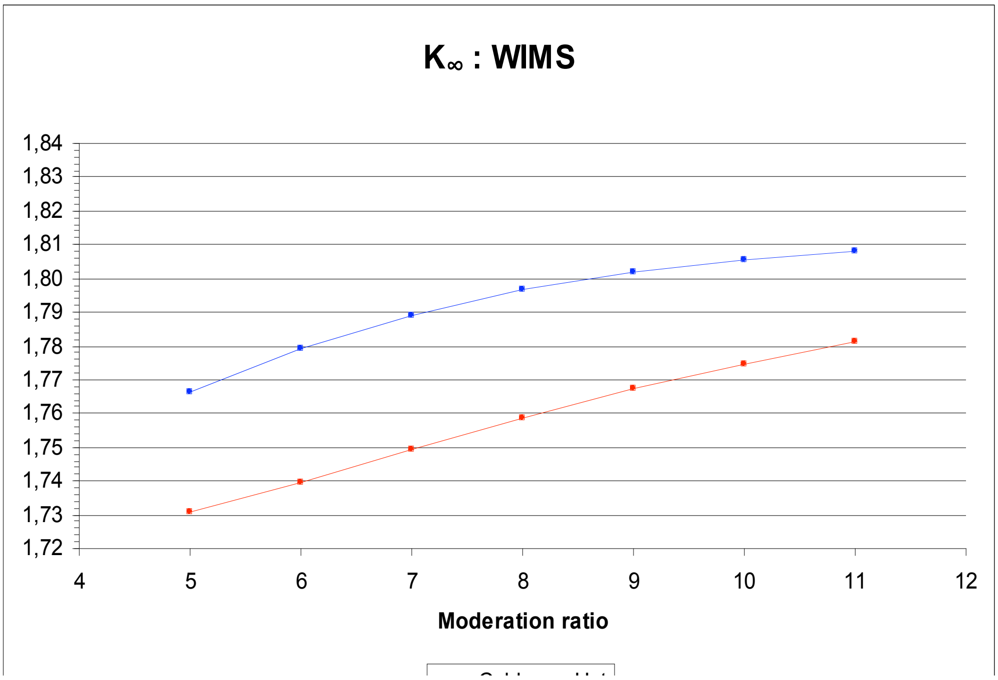
K∞ : WIMS
1,84
1,83
1,82
1,81
1,80
1,79
1,78
1,77
1,76
1,75
1,74
1,73
1,72
4
5
6
7
8
9
10
11
12
Moderation ratio

Fig. 2.6

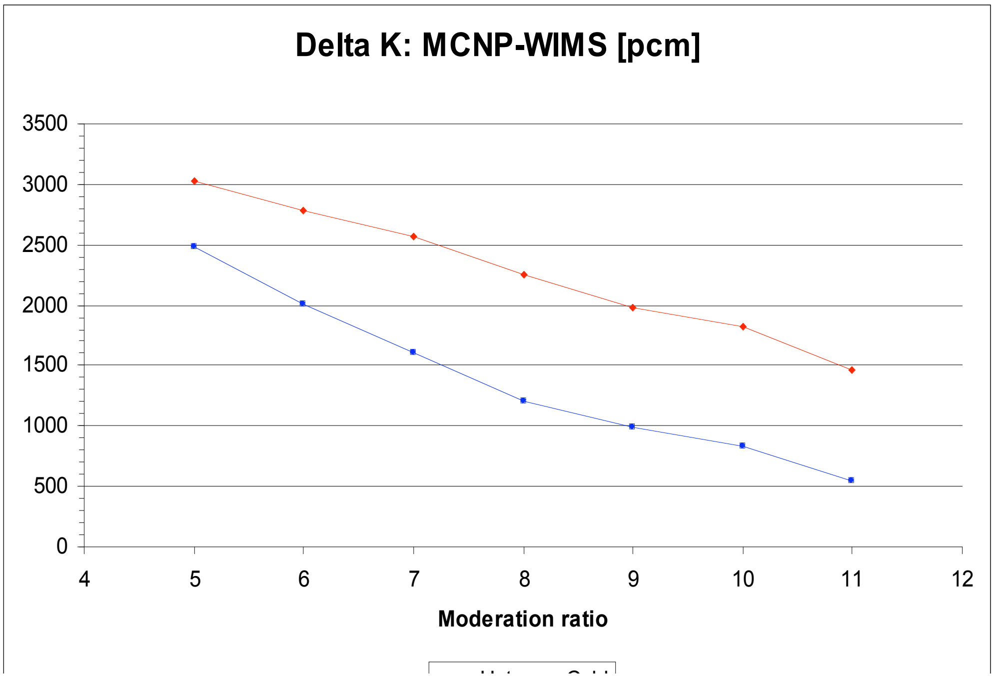
Delta K: MCNP-WIMS [pcm]
3500
3000
2500
2000
1500
1000
500
0
4
5
6
7
8
9
10
11
12
Moderation ratio

Fig. 2.7

Tab. 2.2 – K effective: MCNP-4C , WIMS (Beginning of life)

Reactor type: PWR (Rankine steam cicle)

Electric power [KW]	100
Thermal power [KW]	800

Geometrical data

Fuel diameter [mm]	1.8	Fuel mass (UO_2) [kg]	53.3
Cladding thick. [mm]	0.2	Specific power [KW/kg]	15
Reflector thick. (rad. and ax.) [mm]	120	Linear power [KW/m]	0.393

Temperature (hot):

Fuel [°C]	363
Cladding [°C]	343
Water [°C]	340
Shroud [°C]	340

Temperature (cold):

All components [°C] 27

			WIMS		MCNP		Delta K:MCNP-WIMS	
Mod. ratio	# bundle	D_{eq} [cm]	Cold	Hot	Cold	Hot	Cold	Hot
5	306	36.077	1.23273	1.05221	1.29612	1.08035	6339	2814
6	291	37.714	1.27173	1.07526	1.33020	1.10927	5847	3401
7	279	39.247	1.30411	1.09720	1.35659	1.13374	5248	3655
8	269	40.684	1.33119	1.11767	1.37707	1.15206	4588	3439
9	261	42.080	1.35397	1.13659	1.39630	1.17329	4233	3670
10	253	43.312	1.37326	1.15402	1.41379	1.19079	4053	3677
11	247	44.576	1.38962	1.17004	1.42683	1.20667	3721	3663

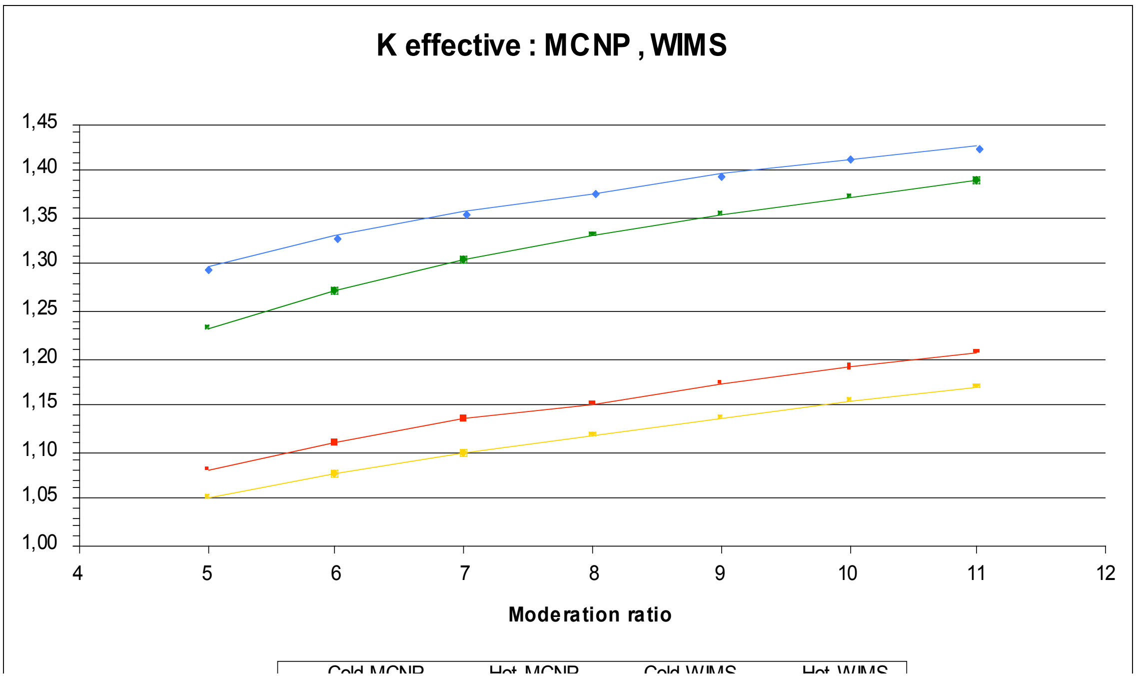

Fig. 2.8

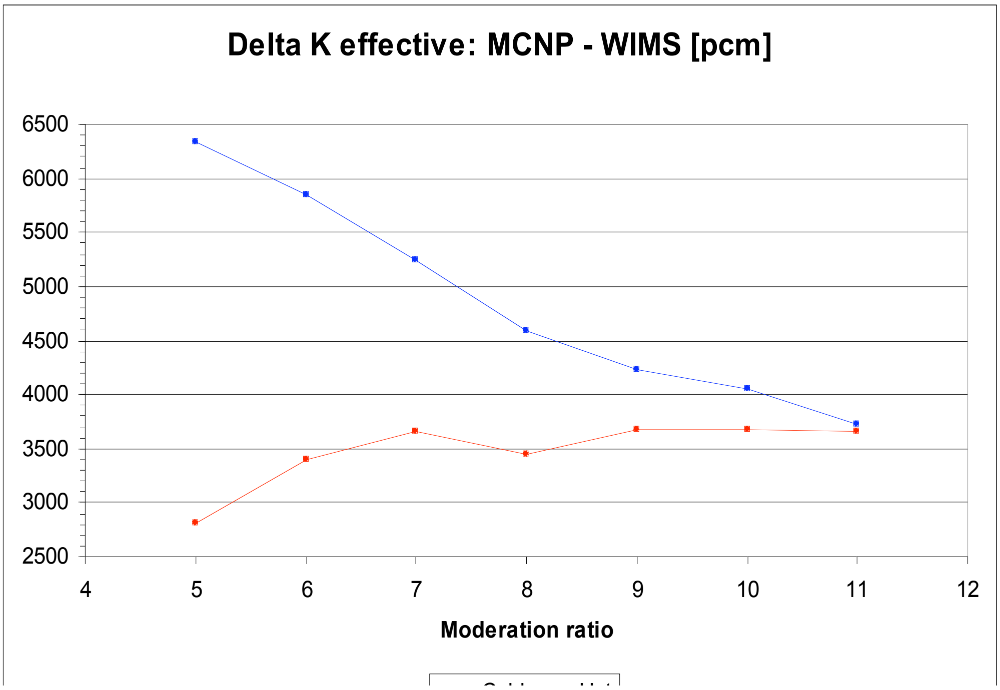

Fig. 2.9

Tab. 2.3 – K effective: Effect of vessel-barrel on reactivity in hot conditions by MCNP-4C (Beginning of life)

Reactor type: PWR (Rankine steam cicle)

Electric power [KW]	100
Thermal power [KW]	800

Geometrical data

Fuel diameter [mm]	1.8	Fuel mass (UO_2) [kg]	53.3
Cladding thick. [mm]	0.2	Specific power [KW/kg]	15
Reflector thick. (rad. and ax.) [mm]	120	Linear power [KW/m]	0.393
Downcomer thick. [mm]	33		

Temperature (hot):

Fuel [°C]	363
Cladding [°C]	343
Water [°C]	340
Shroud [°C]	340

Mod. ratio	Barrel thick. [mm]	Vessel thick. [mm]	Refl. only	With vessel	With vessel – Refl. only [pcm]
5	15	29.0	1.08035	1.09438	1403
6	15	29.7	1.10927	1.12011	1084
7	15	30.3	1.13374	1.14603	1229
8	15	30.9	1.15206	1.16349	1143
9	15	31.5	1.17329	1.18399	1070
10	15	32.0	1.19079	1.20334	1255
11	15	35.5	1.20667	1.21864	1197

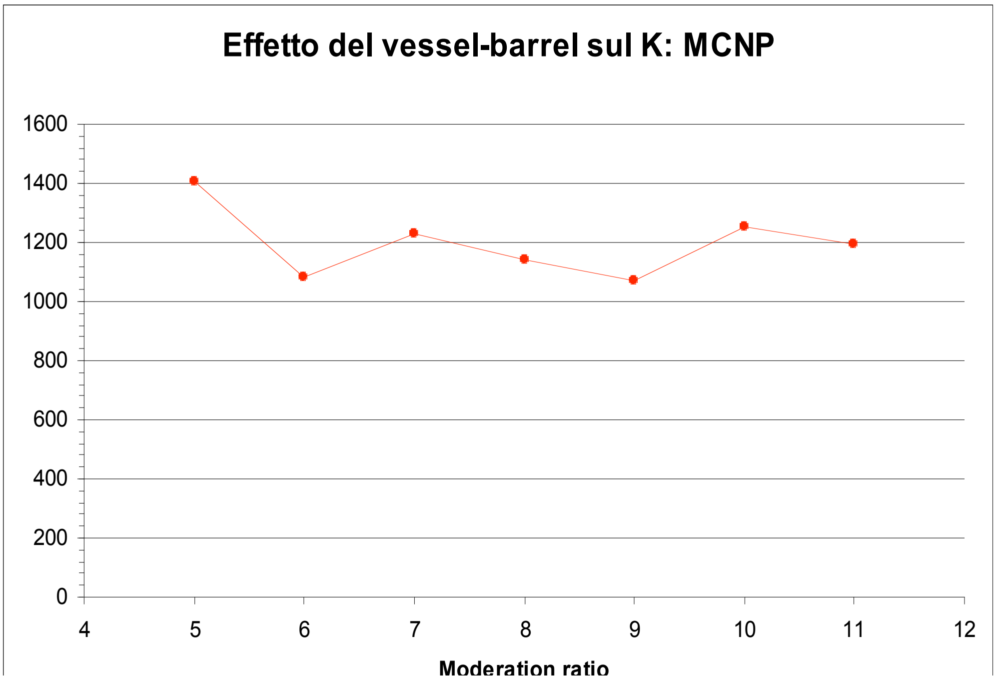

Fig. 2.10

Tab. 2.4 – K_∞ and K effective: MCNP-4C , WIMS (Beginning of life)

Reactor type: PWR (Rankine organic cicle)

Electric power [KW]	100
Thermal power [KW]	555
Moderation ratio	7

Geometrical data

Fuel diameter [mm]	1.8	Fuel mass (UO_2) [kg]	42
Cladding thickness [mm]	0.2	Specific power [KW/kg]	13
Reflector thickness [mm]	120	Linear power [KW/m]	0.393
Downcomwer thick. [mm]	33		

Temperature (hot):

Fuel [°C]	363
Cladding [°C]	343
Water [°C]	340
Shroud [°C]	340

Temperature (cold):

All components [°C] 27

	K_∞		K effective		K eff (with barrel/vessel)	Delta K:MCNP-WIMS	
	WIMS	MCNP	WIMS	MCNP	MCNP	K_∞	K effective
Cold	1.79670	1.80879	1.25903	1.31226	1.31419	1605	5323
Hot	1.75883	1.78140	1.05077	1.08311	1.09723	2567	3234

From these results the following comments can be drawn:

- k_∞ : the difference between MCNP-4C and WIMS results varies in cold condition from 2475 pcm at $V_m/V_u = 5$ to 547 pcm at $V_m/V_u = 11$, and in hot conditions from 3032 pcm at $V_m/V_u = 5$ to 1459 pcm at $V_m/V_u = 11$. The conclusion may be that WIMS shows a reasonable agreement with the *exact* Monte Carlo program, in spite of being applied outside of its range of validity both in terms of enrichment and rod size;
- k_{eff} : referring to 800 KW core, WIMS under predicts k_{eff} by roughly 3500 pcm in hot conditions, and by 6300 to 3700 pcm in cold conditions. These more pronounced differences with respect to those in k_∞ reveal that in these conditions WIMS probably overestimates the neutron leakage and thus underestimates the reactivity; referring to 555 KW core and the moderation ratio equal to 7, WIMS under predicts k_{eff} by 3200 pcm in hot conditions, and by 5300 pcm in cold conditions, which are almost the same differences obtained for the 800 KW core with the same moderation ratio.
- The MCNP-4C applied to the overall geometry, including barrel, downcomer and vessel only in radial direction, yields a reactivity increase of about 1200 pcm. This positive effect has been disregarded in this study, and it may be implemented in future calculations, when the reactor geometry will be better detailed.

In conclusion, in these particular conditions the WIMS program can be judged sufficiently reliable for the goals of this feasibility study, provided that for hot conditions the following rounded margins are assumed, both for the 800 and the 555 KW cores:

- \+ 3500 pcm: to take into account k_{eff} underestimation with respect to MCNP-4C (*exact result*);
- \- 3000 pcm: safety margin to take into account accuracy variation along the life (the above comparison has been done only at BOL), non foreseen absorbing materials, instrumentation, etc.;

Total reactivity = -500 pcm.
This means that the WIMS will be made converge at End Of Life (EOL) to $k_{eff} = 0.995$, rounded to 1.000.

2.2.3 Core design

The assumed data presented in Sec. 2.2.2 of maximum burnup (60 MWd/kgU), cycle duration at full power (4000 days) reactor thermal power (800 and 555 KW respectively), 53,3 and 37 kg U for the 800 and the 555 case respectively. The value of the moderation ratio has been determined by imposing that at EOL k_{eff} value converges to 1.00.
However, the first neutronic calculations carried out for 800 KW reactor by using this value showed that the reactivity is relatively high and thus the resulting moderation ratio turns out to be relatively small, yielding some concern about the fact that the reactor might operate with a not assured thermal spectrum. Therefore, in order to increase the moderation ratio a higher burnup value has been assumed, equal to 80 MWd/kgU, taking into account that the much lower fuel temperature would allow such an increase without much concern. The decision will be detailed in the next future.

By this hypothesis the fuel mass for this reactor is lowered from 53,3 kg to 40 of U, i.e. from 60.5 to 45.4 kg of UO_2. Moreover the 45.4 kg of UO_2 has been adjusted to 47 kg to achieve an acceptable moderation ratio of 6.5. By this value the real maximum burnup is reduced to 77 MWd/kgU instead of 80 MWd/kgU assumed a priori.

For the 555 KW reactor the maximum burnup has been kept equal to the initial value of 60 MWd/kgU, i.e. a fuel mass of 42 kg UO_2. The resulting moderation ratio turns out to be 7.

Fig. 2.11 and 2.12 show that under the above assumptions, by choosing the moderation ratio of 6.5 and 7 for the 800 and 555 KW case respectively, the specification of a EOL k_{eff} = 1.00 is satisfied. As a matter of fact the final k_{eff} is 1.008, giving a further margin to the required value of 1.000.

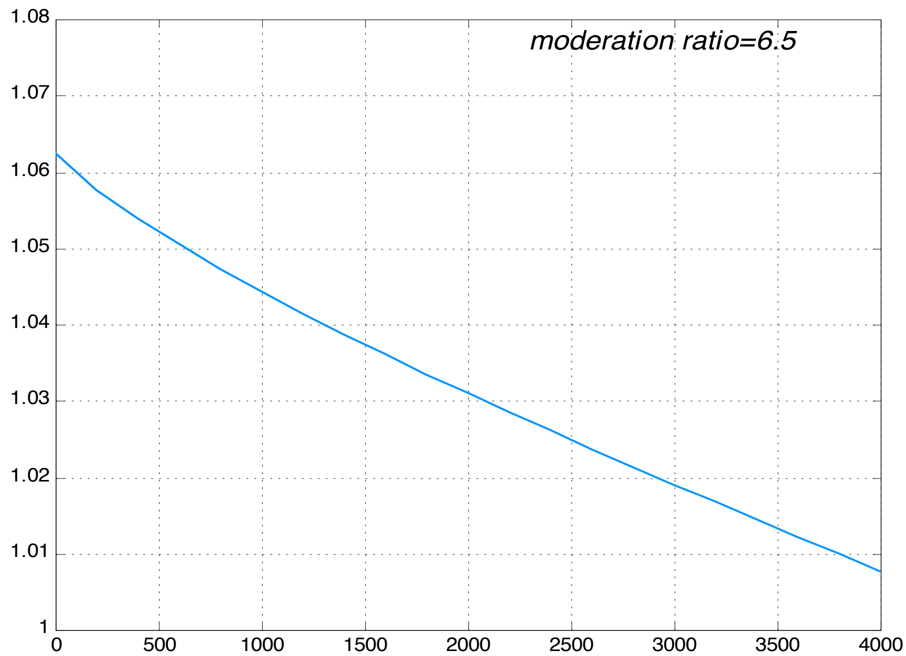

Fig. 2.11

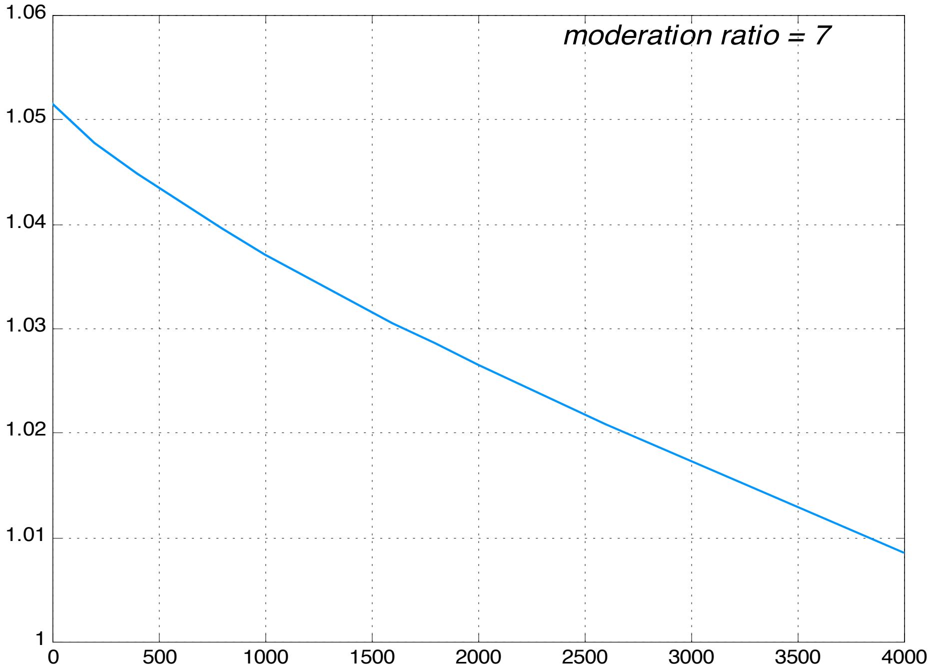

Fig. 2.12

Figures 2.13 and 2.14 show the final fuel channel disposition in the 800 KW and 555 KW case, respectively:

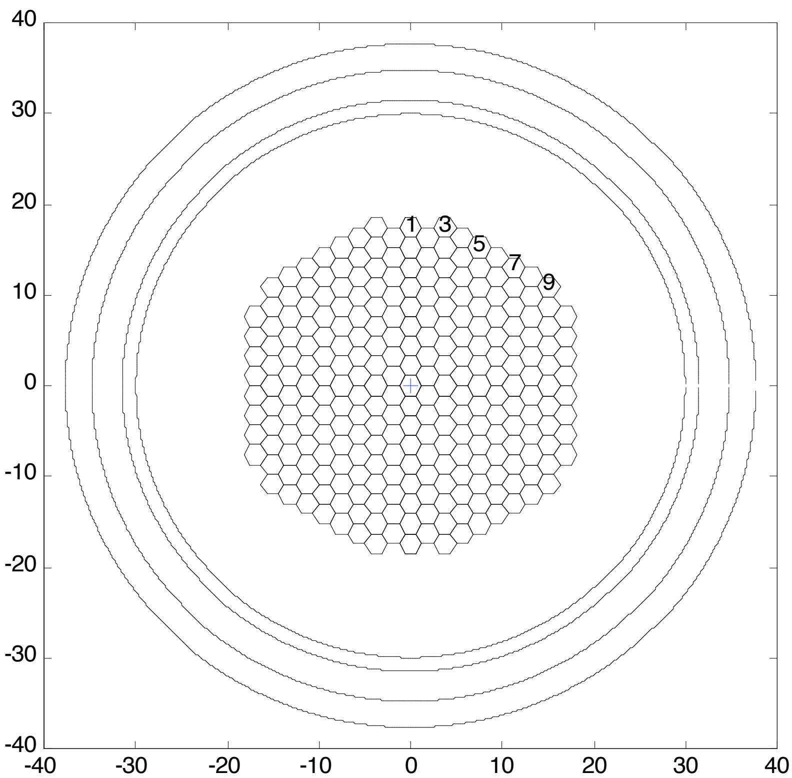

Fig 2.13

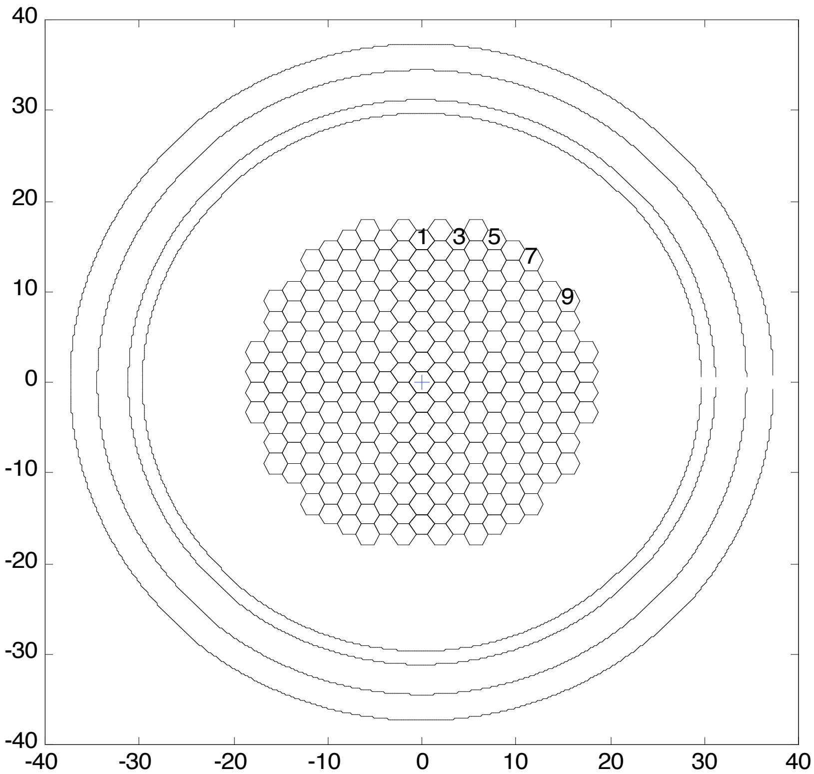

Fig. 2.14

The resulting data are given in par. 2.11. It is interesting to note that the core size is almost the same for the two required powers: the equivalent diameter is 36,8 cm for 800 KW and 36.3 cm for 555 KW core. Thus the overall mass (core+reflector) of the 555 KW case is only slightly lower: 234 against 247 kg.

2.3 Electrical power generation system

To transform the reactor thermal power to electrical power a thermodynamic generation system is needed. In PWR case, we discarded the possibility to use a thermoelectric device, because the relatively small temperature difference between hot and cold wells, would yield a too low efficiency, of the order of 2-3 % (see for more details par. 3.4).

Therefore, two conventional thermodynamic cycles have been considered:

- A Rankine cycle with steam;
- A Rankine cycle utilizing an organic fluid.

Before starting the efficiency calculations, a preliminary optimization study has been carried out concerning the cold well temperature. This optimization is based on the minimum mass of the overall system including the reactor, the cold well, and a contingency of 20 %. In both cycles, the result has been that the optimum is in the 160-180 °C range, with a rather flat behavior. Then a value of 165 °C has been chosen at this stage of the work. The maximum temperature can be obtained by a detailed design and optimization of the steam generator (see later on). A preliminary study to this procedure has yielded a value for this temperature approximately equal to the minimum primary temperature of 335 °C.

2.3.1 <u>The Rankine steam cycle</u>

This well known and widely adopted cycle has been calculated by imposing specific requirements tending to simplify as much as possible the system and by assuming reasonably low turbine efficiency to take into account its small size, which is outside the present technology. In particular, in spite of the fact that with the relatively low steam temperature a saturated steam cycle would be convenient from the efficiency stand point, this solution has been discarded to reduce as much as possible the moisture along the cycle and at the condenser inlet and to avoid in any case the need of a moisture separator. A high moisture value is in general viewed as a danger for its effects on turbine blade erosion, which would become unacceptable for 4000 days of operation without maintenance. Then we adopted the following hypotheses and data:

- Inlet steam temperature: 335 °C;
- Condensing temperature 165 °C;
- No regeneration along the turbine expansion;
- No re-superheating in the middle of turbine expansion;
- No moisture separator along the expansion;
- Maximum discharge moisture 3 %;
- Turbine efficiency: 73 %;
- Feed pump efficiency: 75 %;

By adopting these data an efficiency equal to 13.1 % has been obtained, in correspondence of a boiling pressure of 5.7 MPa and an inlet superheating of 63 °C. This is a gross efficiency, which takes into account only the power absorbed by the feed pump. Assuming that the system will grossly need a power of 5 KW for the primary pump, the cold well cooling system (to be detailed), and the ancillary circuits, the net efficiency is obtained by multiplying the gross efficiency by the ratio 100/105, and then the resulting value is equal to 12,5 % .

In conclusion, in order to produce 100 KWe of net power, the reactor thermal power is set equal to 800 KWth.

2.3.2 <u>Organic fluid Rankine cycle</u>

The cycle efficiency is of paramount importance in order to minimize sizes and masses of the overall system. By using a conventional steam Rankine generator system the above efficiency results rather low. It is well known that in relatively low temperatures and low temperature differences between hot and cold wells, an organic fluid (C_xH_y) would give higher values. This advantage is counterbalanced by some drawbacks as the presence in the system of a different fluid, which requires an ad hoc supply system and ad hoc reservoir to cope with possible leakage in the long run, and the possibility to have an insufficient thermal stability of the compound for the 4000 days period. These issues will be addressed in a prosecution of the program, here we limit our analysis to a rough estimate of the efficiency to focus the possible advantages of such an alternate generator.

The efficiency has not been calculated directly, because the variety of fluids and their complex thermodynamic properties behavior would render such an approach too challenging in this stage of the work. The approach has been to extrapolate a published datum to our conditions.

By referring for instance to the scheme represented in Fig. 2.15**,** the authors of the paper in rif. 8 calculated the efficiency of a cycle using the organic compound *tetrametilbenzene* ($C_{10}H_{14}$)*,* working between 300 and 60 °C and with a turbine efficiency of 75 %. The resulting efficiency is equal to 28 %. The rather usual way to extrapolate the Rankine efficiencies to other conditions is that to adopt the same ratio with the Carnot efficiency. In this case the Carnot efficiency would be 41.9 %, then the ratio actual efficiency over Carnot efficiency is 67 %.

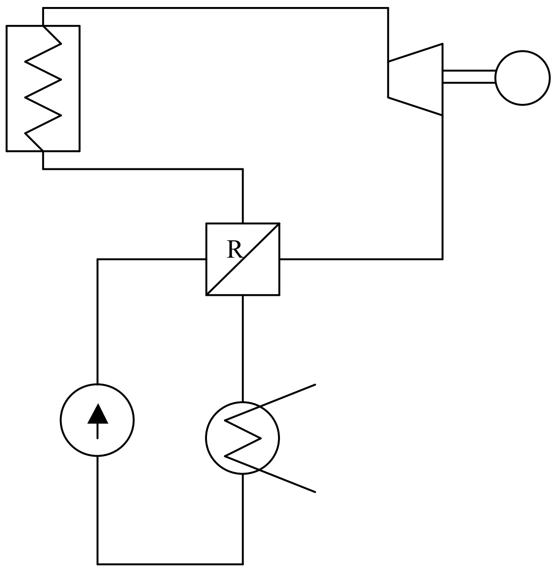

Fig. 2.15

In our condition the organic inlet temperature is 335 °C and the discharge temperature 165 °C[2]. The corresponding Carnot efficiency is 28 %, which multiplied by 67 % gives the cycle efficiency of 18,76 %. To complete the preliminary study of an organic fluid application, this value has been assumed for defining the whole system.

This is the gross efficiency, which, taking into account a power absorption of 5 KW, as above done for steam cycle, is reduced to 17,9 %, finally rounded to 18 %.

By adopting this value the reactor thermal power is set equal to 555 KW.

[2] This extrapolation procedure might be a little conservative, being the temperature interval lower than the reference one.

2.4 The primary system

The primary system is made by the reactor vessel which contains *the reactor core, the barrel, the steam generator, the pressurizer, the circulating pump, the safety valve, the reactivity control mechanism and the instrumentation.* All these components are inside the reactor vessel, thus adopting the so called *integrated* solution. This allows to keep the size and the mass of the primary system to a minimum, together with the elimination of the possibility to have any break of the pipes, which is the main cause of having Loss Of Coolant Accidents (LOCAs); moreover the fast neutrons fluence on the vessel is reduced. Therefore, the only pipes connected to the vessel are the ones of the steam and the liquid of the secondary generator system; note that these pipes are connected to a closed circuit inside the vessel, and thus no connection with the primary system is possible.

Water flows upward through the core and then through the lower part of the upper plenum (the remaining part is filled with steam for the pressurizer), where the flow direction is reversed and the coolant is directed downward through the annular downcomer region, between the core barrel and the vessel; in this annular space the steam generator is located; the primary water flows on the outer surface of the steam generator tube, exchanging heat with the secondary fluid (water or organic compound) till the lower plenum, where the suction of the circulating pump is located; finally the pumped coolant enters the reactor core to close the circuit (see Fig. 2.22-23 for the layout of the primary system).

The design pressure of the primary system is equal to the operating pressure multiplied a factor of 1.10, which takes into account the value of the safety valve setting. This component is absolutely needed for each pressurized system, but in our case implies an important issue concerning the consequences of its operation in the case of pressure transients above 1.10 operating pressure, which means a discharge of a substantial fraction of the primary water. For the time being we adopted a higher value than the usual one in order to reduce the possibility of its intervention (1.10 against 1.07). The design pressure has been then set to 15.5x1.10 = 17.05 MPa.

2.4.1 The reactor vessel

Three possible geometry were considered for this component: sphere, cylinder with hemispheric domes, cylinder with semi ellipsoidal domes. A parametric study of the masses and volumes of the three alternatives, which are to satisfy the space requirements of a cylindrical core, arrived at the conclusion that all were almost equivalent, with a slight preference for the cylinder with hemispheric domes. The latter is also preferable for accommodating the steam generator in the annular space between the barrel and the vessel. Moreover, the spherical domes are better suited than ellipsoidal ones for accommodating the pressurizer in the upper dome and the pumps in the lower one (see below). Then the final choice is the cylinder with hemispheric domes.

The vessel size has been determined by adopting the scheme of Fig 2.14 and 2.15. The final choice has been to use the following data: a = 120 mm (equal to the reflector thickness), b = 33 mm (to accommodate the steam generator), c = 120 mm (to better protect the vessel from the escaping radiation in the corner points). The value of b is important for controlling the neutron fluence and the radiation shielding capability inside the vessel: both issues are to be addressed in a prosecution of the work. Here it can be stressed that any increment of this value would be rather penalizing for the overall mass.

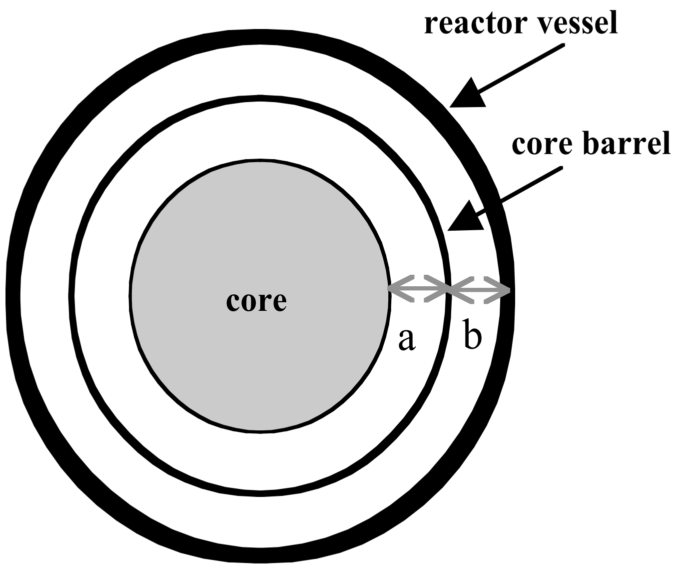

Fig. 2.16

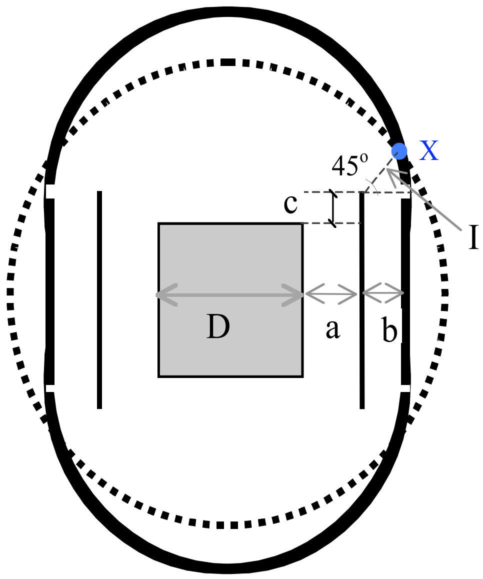

Fig. 2.17

The vessel material is steel; this is a conservative choice, anyway motivated by the fact that the only possible alternate material is Titanium. This material has a density of 4500 kg/m^3, which is rather lower than that of steel (7800 kg/m^3), however its features both in terms of strength and corrosion resistance are to be established, taking into account that a variety of alloys exists, having different behavior (the best experience is in Russia). Moreover, Titanium is not much resistant to creep, as shown by the maximum operating temperature established by ASME code equal to 315 °C (600 °F), which is 30 °C lower the outlet core temperature; fabrication processes are more difficult, especially the welding. The first guesses show that only moderate advantages are possible by using this material, thus a more detailed analysis and the corresponding final decision have been delayed to a next activity.

Steel recently adopted for PWR vessels has an allowable stress of 205 MPa. As a matter of fact the conventional vessels are made by carbon steel plated on the inside surface by stainless steel. The above allowable stress refers to carbon steel. In our case, it may be probable that the whole vessel will be made only by stainless steel, taking into account the small value of the thickness, when compared to that of conventional vessels. In this case the allowable stress must be referred to the particular stainless steel adopted. For the time being the above value has been adopted yielding a thickness for the two reactors equal to 29.3/29.1 mm and 14.65/14.55 mm respectively for the cylindrical portion and the spherical domes.

The barrel has the function to separate the reactor core zone from the annular peripheral one where the out of core coolant is circulated at the reactor inlet and at the same time cooled down by exchanging heat to the steam generator, here located. The barrel is a simple steel cylinder not undergone to any particular load. Thus its thickness is determined by the requirement to have a good rigidity and to reduce fast fluence on the vessel if necessary: a value of 15 mm has been assumed.

The vessel plus barrel mass is equal to 641 kg and 624 kg for 800 and 555 KW respectively.

The overall reactor masses are 1112 kg for the 800 KW and 1078 kg for the 555 KW reactor, assuming conservatively cold water completely filling the pressure vessel. All mass and volume data for the primary integrated circuit are detailed in par. 2.11.

The sketches of the horizontal cross section are shown in Fig. 2.16 and 2.17 referring to the 800 and the 555 KW cores, respectively. The sketch of the vertical cross section is shown in Fig. 2.20 referring only to the 800 KW core since, as said before in sec. 2.2.3, the core sizes are almost the same for both powers and then also the pressure vessel.

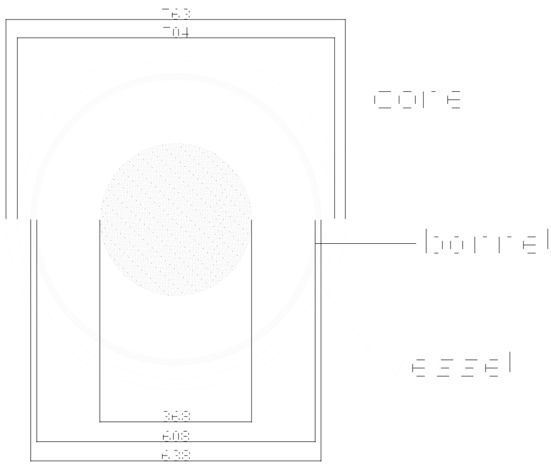

Fig. 2.18

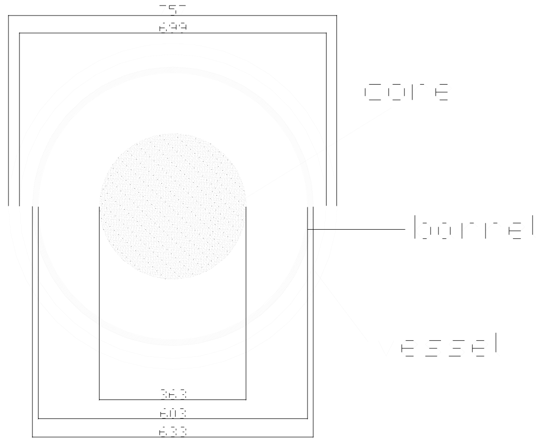

Fig. 2.19

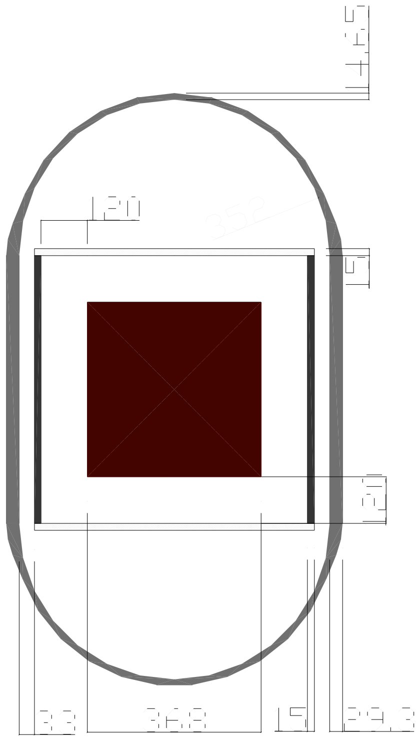

Fig. 2.20

2.4.2 <u>The steam generator</u>

Several configurations are possible for the steam generator (SG): straight tube, U-tube, C-tube, bayonet tube, helical tube. Based on a thorough analysis carried out for an integrated PWR reactor (IRIS), a helical coil tube bundle has been selected [9]. This is a proven design that has operated in various reactors, including the French Superphenix. There is also the positive ten years operating experience of the German ship Otto Hahn. This design is capable of accommodating thermal expansion without excessive mechanical stresses, has high resistance to flow induced vibrations, and is designed to have thermal performance second only to a straight tube design, which is to be discarded for the high loads created by temperature transients.

As far as the reliability of such a component is concerned, it is well recognized that current PWR steam generators face various degradation mechanisms, which imply extended inspections, maintenance and repairs. The latest annual report on SG performance published by EPRI [10], in the frame of the Steam Generator Management Project (SGMP), details the major issues and degradation mechanisms contained within SG degradation database: for instance, in 1999 for the 238 operating plants a total of 8546 tubes were repaired owing to different degradation mechanisms, mainly due to stress corrosion cracking (SCC). In general, SGs show a shorter life than that of the overall plant, which makes their troublesome periodic replacement a common practice.

Therefore, it would seem that SGs inside the pressure vessel be the cause of impairing substantially the reliability of the plant as in the current ones. However, the steam generator design here proposed is so different that the present degradation experience of current SGs is not directly applicable.

The main design difference is that all the sensible components inside the pressure vessel - i.e. tubes, headers and nozzles crossing the pressure vessel wall - are compressed instead of being stretched, because the higher primary pressure is acting on the outer surfaces: strictly speaking, primary stresses are compressive. In the case of tubes, stability requirements to avoid their collapse imply the adoption of a tube thickness that is about twice as much the value needed to resist to the pressure compressive stress, based on full (primary) outer pressure. Considering that the secondary side pressure is 5.7 MPa, in normal operation the thickness is about three times the value needed to resist to the compression stress. This means that deterioration mechanisms due to high stresses, such as fatigue, should inherently be eliminated; those ones connected to SCC on both surfaces are not possible from a mechanical point of view, i.e. considering only the compressive primary stresses.

Magnetite and copper impurities, or other such as lead, on the inner surface in contact with the secondary fluid can be an issue, which should be thorough addressed when choosing the materials of the turbine system.

In this design, several orifices or ferrules are to be inserted in the inlet section of the tubes, in order to create an additional pressure drop in the water zone both to stabilize the flow from the parallel channel instability phenomenon and to promote an even flow distribution through the tubes in the tube bundle. This additional pressure drop is not yet defined, but should be of the order of the overall pressure drop in the tube. This aspect deserves a special attention both for constructive and functioning reasons; the latter one means that some erosion-corrosion phenomenon may here originate.

Taking into account the limited power to be transferred in this case, it has been decided to adopt a single tube in order to eliminate the above mentioned instability phenomena. This would imply to choose a reasonable high value of the diameter and the length of the tube.

A rather detailed mechanical design of a tube subjected to external pressure of 17.05 MPa and zero pressure inside yields a thickness value of 11 % of internal diameter, by using the material inconel TT690.

In this case the chosen inner diameter is 20 mm, then the thickness is 2.2 mm and the outer diameter 24.4 mm. The well known RELAP5 [5] calculation program suitable for thermalhydraulic system analysis, both in steady state and transient conditions has been used for sizing the SG. Some adaptations were introduced in order to take into account the particular geometry of this SG. However, no benefit for the helix geometry effect on the inside heat transfer coefficient and for the outside spacers on the outside heat transfer coefficient are taken into account; that means that the overall size here detailed may be over estimated. The results are shown in the Table 2.5. They seem well within the existing experience especially as far as the length and the secondary pressure drops are concerned. However the thermalhydraulic behavior of helix was not well studied in past, then a experimental campaign is needed for its development, also to take into account the effect of lack or reduced gravity.

Tab. 2.5 : Steam Generator data

Power [KW]	800
Secondary fluid	Water
Geometry	Helical single tube around the barrel
Annular gap width [mm]	33
ID/OD/t [mm]	20/24.4/2.2
Tube length [m]	50
Coil diameter [m]	~ 0.68
Coils number	23
SG height [m]	0.80
SG weight [kg]	60
Primary fluid temperatures inlet-outlet [°C]	345-335
Secondary fluid temperatures inlet-outlet [°C]	165-335
Primary/secondary pressures [MPa]	15.5/5.7
Primary flowrate [kg/s]	10.1
Max. primary mass flux [kg/m^{2}s]	549
Secondary flowrate [kg/s]	0.347
Secondary mass flux [kg/m^{2}s]	1103
Primary SG pressure drops [KPa]	5
Secondary SG pressure drops [KPa]	500

For organic fluid there are three differences: i) the power to be transferred is 555 instead of 800 KW, ii) the transport properties of complex organic fluids, as here assumed, are worse than the water ones, iii) the organic cycle uses only saturated fluid, then inside the SG there is no superheated zone. Probably these differences are self compensating, so that the overall SG surface may result almost equal to the water one. If this is the case, also the overall layout will be similar, being the core size practically equal. These issues will be clarified when and if the organic fluid will be chosen for this application. This requires a definite choice of the fluid and consequently of its physical properties.

2.4.3 The pressurizer

The pressurizer is a big issue in PWRs. It is made by a pressure vessel where saturated water and saturated steam are in equilibrium exactly at the operating pressure. The water zone is connected by a relatively small pipe to the primary circuit. Since the maximum water temperature at the reactor outlet is well below the saturation pressure (about 15 °C), the pressurizer water is to be heated by special electrical heaters. The pressure is controlled by acting on the electrical power of the heaters and by spraying a small flow rate of colder water into the steam zone. It is a rather complex system, which can be simplified by putting the pressurizer in direct connection to the vessel (in the upper dome in our case) and bringing the outlet temperature to the saturation value, as here done. The sprayed water may be eliminated if a big steam volume is chosen and a certain pressure oscillation of moderate entity is permitted. On the contrary in a big PWR the pressure must be well within given limits to reduce the mechanical stresses, which are particularly compelling in this case.

An abundant free steam volume, as 30 liters per MW, which is several times the value used in conventional PWRs, is here adopted. That means in our case 24 and 17 liters for 800 and 555 KW reactors respectively. The spray water is then eliminated. These volumes are only a fraction of the upper sphere volume, which is equal to about 90 liters.

Besides this free volume we have to foresee the possibility to contain the water expansion between cold and hot conditions; in fact the specific volume increases by a factor 1.64, going from ambient temperature (on the earth) to the average reactor temperature of 340 °C. This means that there are two alternatives: discharge the excess of water to an ad hoc vessel or to leave a *void* inside the cold vessel exactly equal to the above volume difference.

The first solution seems to penalize the system in terms of mass and volume, because this excess of water is to be discharged during the start up operation in an external reservoir; however, a given amount of water is probably needed to cope with probable leakages during the long period of operation.

The second one imply some issues. Assuming for simplicity a vessel made by a cylinder with diameter D equal to the height and spherical domes (see Fig 2.22), its volume is equal to $5/12(\pi D^3)$. Disregarding any solid component inside the vessel, this approximately represents the total volume of water; reducing this volume by 1.64 times an initial height of cold water in the cylinder equal to 0.67 D is obtained. This means that the initial level does not cover entirely the whole core. The acceptability of this condition depends on the start up procedure, which in turns depends on some specific requirements, not well defined at this stage of design. Maybe that an intermediate solution between the two ones is advisable.

Obviously the above pressurizer, integrated inside the vessel, is possible only in presence of a given value of the gravity to separate steam and water. This is the case for surface reactor, but not for propulsion one. Some pressurization alternatives can be imagined, but each of them is to be carefully studied and experienced. We leave it for a future activity.

2.4.4 <u>The circulating pump</u>

In the introduction it has already been said that the natural circulation of the primary water has been excluded, thus a circulating pump is needed. The chosen pump is the *spool type,* [3] which has been used in marine applications and designed for chemical plant applications requiring high flow rates and low developed head. The motor and pump consist of two concentric cylinders, where the outer ring is the stationary stator and the inner ring is the rotor, that carries high specific speed pump impellers.
A sketch of the spool pump is shown in Fig. 2.21.

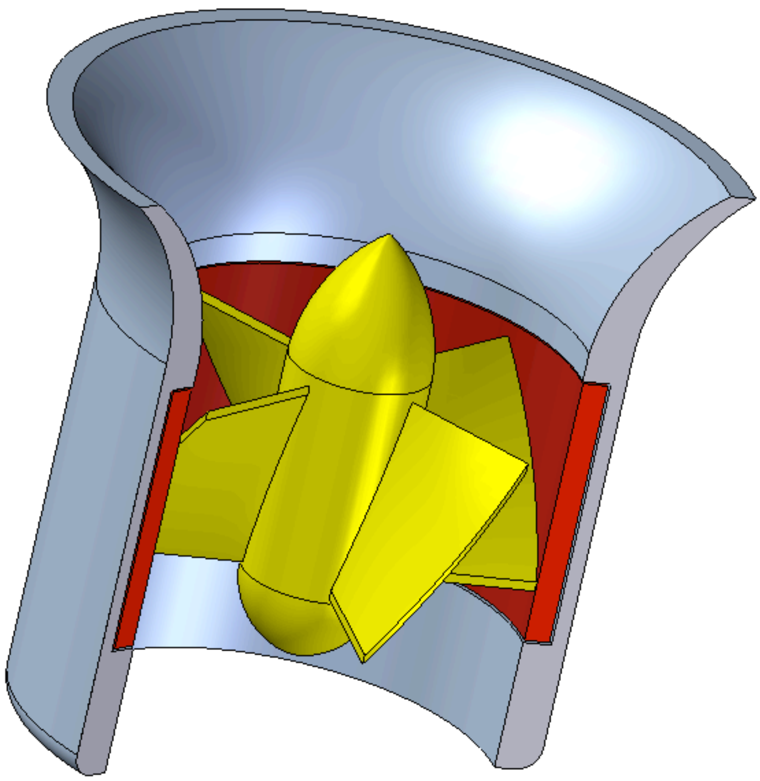

Fig. 2.21

The spool pump is located entirely within the reactor vessel; only small penetrations for the electrical cables are required. High temperature windings and bearing materials are being developed in a US industry to eliminate the need of cooling water and the associated piping penetrations trough the reactor vessel. Moreover, the demonstration of long lasting operating capability, without any maintenance and inspection activity, is part of this development program. The achievement of these goals is prejudicial for this application, because simplicity and reliability is a must for all components.

In addition to the advantages derived from its integral location, the spool pump geometric configuration provides high inertia/coast-down and high run-out flow capability, that contributes to mitigating the consequences of loss-of-flow accidents (LOFAs).

Because of their low developed head, spool pumps have not previously been considered for nuclear applications. This integral core configuration and low coolant path pressure drop, however are an ideal match for these pumps and can take full advantage of their unique characteristics. For this application, the pump is much smaller than those under development and in principle this seems to be an advantage, however a verification should be done.

The main preliminary characteristics of the spool pumps are detailed in the Table 2.6. By using these data, the pump features amply remain in the range of axial pump zone.

Tab. 2.6 : Spool pumps characteristics

Reactor thermal power (KW)	555	800
Fluid	Water	Water
Operating pressure [MPa]	15.5	15.5
Assumed head [KPa]	20	20
Operating temperature [°C]	335	335
Mass flow [kg/s]	7.0	10.1
Volumetric flow [liters/s]	11.1	16.1
Efficiency	0.6	0.6
Power [W]	370	540

The remaining components are the safety valve, the control mechanism and the instrumentation. The safety valve does not require particular consideration apart from the issue mentioned above about the effects of its operation. The control mechanism will be addressed in the next paragraph. The instrumentation can be considered rather conventional, but how to treat the relative information is an open issue, which should be considered in the following.

The overall resulting structure of the primary system is depicted in Figs. 2.22 and 2.23.

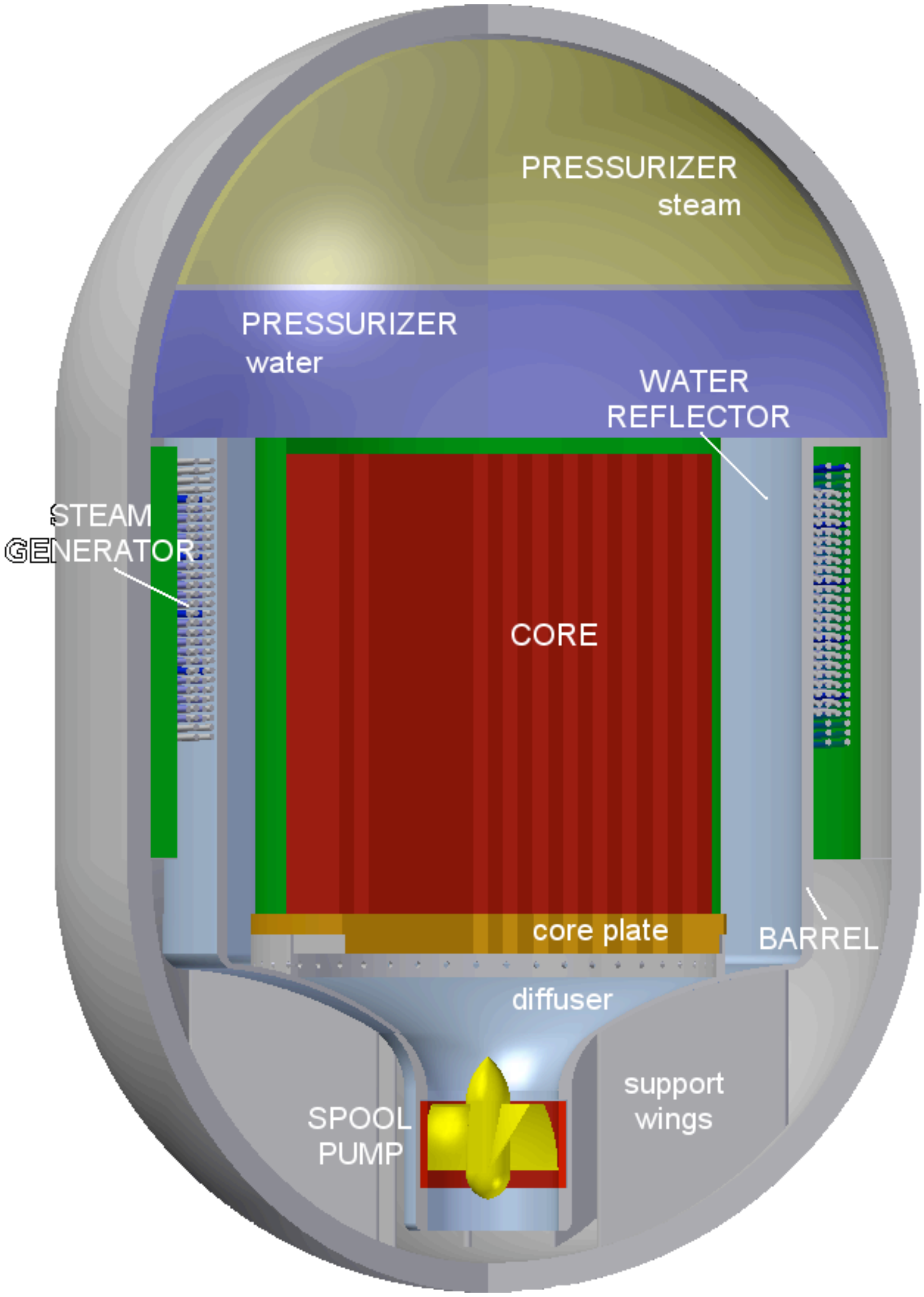
PRESSURIZER
steam
PRESSURIZER
water
WATER
REFLECTOR
STEAM
GENERATOR
CORE
core plate
BARREL
diffuser
support
wings
SPOOL
PUMP

Fig. 2.22

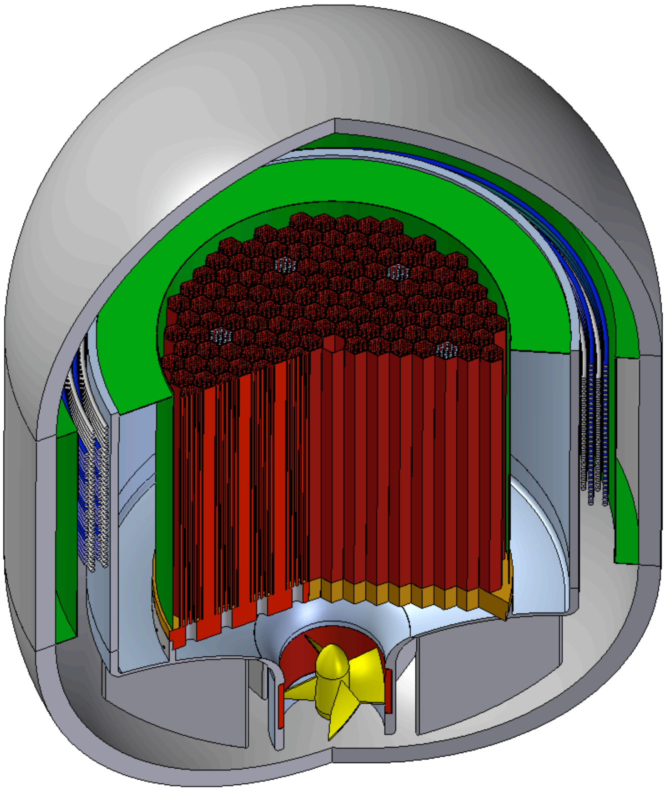

Fig. 2.23

2.5 The reactivity control

The PWR has inherently favorable features for control requirements, since it is characterized by a negative reactivity coefficient of temperature. Any increase of water temperature implies a reduction of its density, and because the core is designed to be *undermoderated* the reactivity reduces too. A further effect would be due a resonance widening in the absorption cross section of the fuel for a temperature increase (the Doppler effect), but this effect is not practically present in this reactor for the very low fraction of uranium-238, and the very low temperature increase inside the fuel rod, being the linear power value much smaller than in current PWRs. On the other hand, any decrease in the inlet temperature implies an increase of water density and an increase of the reactivity: this yields an increase of reactor power, which reestablishes the previous average core temperature. In this case it is to be verified that the transient is not too fast to produce a power peak before the feedback action can enter into effect. This must be addressed in the safety analysis. Apart from this concern, the negative temperature coefficient makes the reactor a *load follower:* any request of a higher power from the secondary system produces the opening of the inlet steam throttle valve, which in turns reduces the steam pressure and thus its temperature and that of the primary system through the steam generator: the reactivity increases and the reactor power goes up to the required value. The self regulating capability of these reactors holds within a given power range variation, beyond which the reactivity control system has to operate. In this reactor the temperature coefficient, around the operating temperature, is much higher than in PWRs: approximately

equal to -300 against -30 pcm/°C, and this would improve appreciably the self regulating capability of the reactor. This is important for safety aspects, however, this feature should not be exploited currently for the control, because the reactor should operate at constant power.
On the other hand the reactivity excursions in these reactors are rather important. In a typical PWR the total reactivity difference between a cold condition with the fuel at the Beginning Of Life (BOL) and the hot reactor at full power with the fuel at the End Of Life (EOL) is approximately equal to 24000 pcm (10^{-5}). A logical subdivision of this reactivity variation is as follows (in brackets the typical values for PWRs in pcm, 10^{-5}):

1. Reactivity decrease going from cold condition to operating temperature (6000);
2. Reactivity decrease from zero power to full power conditions (this term includes the absorptions by Xe and Sm, which reach their equilibrium value in tenths hours) (4000 = 1200 for Doppler + 2800 for Xe and Sm);
3. Reactivity decrease along the fuel life to cope with reduction of fissile material and accumulation of poisoning fission products (14000);
4. Reactivity margin to control the reactor power (500).

The control of this big amount of reactivity is done by the combined use of several systems:

a. A neutron poison (Boric acid) diluted into the coolant, the concentration of which is varied according to the needs. However, its maximum concentration and then the maximum reactivity absorption is limited by the condition that the temperature reactivity coefficient remain negative; in fact the water expansion due to a temperature increase makes decrease the poison concentration as well and above a given concentration this effect in terms of reactivity is such to counterbalance the negative temperature effect of pure moderator. The system which controls the poison concentration during the reactor operation is rather complex and its action is obviously rather slow.
b. The use of a neutron absorbers spread on the fuel surface (ZrB_2) or dispersed inside the fuel, gadolinium or erbium, which capturing a neutron transmute themselves in low absorbers elements. Thus their effect on reactivity is maximum at BOL and reduces along the fuel life. The requirement is that their effect be negligible at EOL, condition not satisfied by all alternatives, and to not increase the initial flux peaks along the fuel life. However, in general the reactivity reduction does not compensate exactly the one of fuel consumption and accumulation of fission products and thus the other systems have to operate.
c. Control rods: mechanical devices made by high neutron absorbers which are moved inside or outside the core, to vary the poisons content of the core. This system is very fast and is used also to shut down rapidly the power (SCRAM). Each rod is moved singly, with small movements. This system is mechanically complex, expensive and above all requires important penetrations in the vessel, because the mechanisms controlling the movement are located outside the vessel on its spherical dome.

These systems are well developed and positively experienced in hundredths of PWRs, however they cannot be straightforward used in this reactor. The present requirements are: simplicity and no or few vessel penetrations. This can be partially facilitated by the fact that this reactor is to be brought at full power and then remain there for the whole period of operation.

The overall reactivity excursions and the value of each single term is detailed in Table 2.7 (in brackets typical PWR data):

Tab. 2.7 : Reactivity terms in pcm.

Power (KW)	555	800
Term 1 (6000)	23000	23000
Term 2 (4000)	500	500
Term 3 (14000)	4300	5500
Term 4 (500)	$\sim 0^+$	$\sim 0^+$
Total (24500)	27800	29000

The differences between PWR and this reactor are due to the high enrichment and to a reduced extent for the low power density. In particular, Term 2 is relatively small because of the low temperature increase inside the rod and the low thermal flux: both the Doppler effect and the absorptions by Xe and Sm are of modest impact.

The total reactivity is higher than that of PWR, because the much higher value of the term 1 is only partially counterbalanced by the lower values of terms 2,3, and 4.

The first system, using variable boric acid concentration in the coolant requires a complex control circuit outside the reactor, which is not advisable to implement in this reactor.

The second one might be used. However, in high enriched cores the poison concentration must be very high, but the very low thermal fluxes typical of these cores prevent to burn substantially the poisons at EOL, and this has been verified for ZrB_2 , which cannot be used in these cores. Probably the best poisons are those characterized by high absorption cross sections as in the case of gadolinium. However the design of burnable poison solution is complex in any reactor, and then this activity has been postponed.

For the control rods there is not an easy answer: being high the reactivity to be controlled, several of them are needed, necessarily moved in parallel by a single bar crossing the vessel and operated by an external motor as used for each single rod in a PWR. In this solution the space into the vessel above the core should be enough to accommodate the entire length of the rods when they are out of the core. For this reason and other ones, many designs foresee the use of rods inside the reflector instead of in the moderator. In this reactor the leakage of neutrons is so high, that a reflector poisoning may be enough to reduce the reactivity. This has to be thorough verified. If this is the case, the control rods can be imagined not going up and down into the reflector, but made by a rotating cylinder, having on its diameter the poison plate. The rotation varies the angle of the poison plate, and then its neutron absorption capability.

Here a different proposal is put forward, based on the fact that the core is very small and its portions can be probably moved apart rather easily. By this proposal we do not intend to discard the previous one, which will be considered in detail during the prosecution of the work.

The principle is shown in Fig. 2.24 and 2.25.

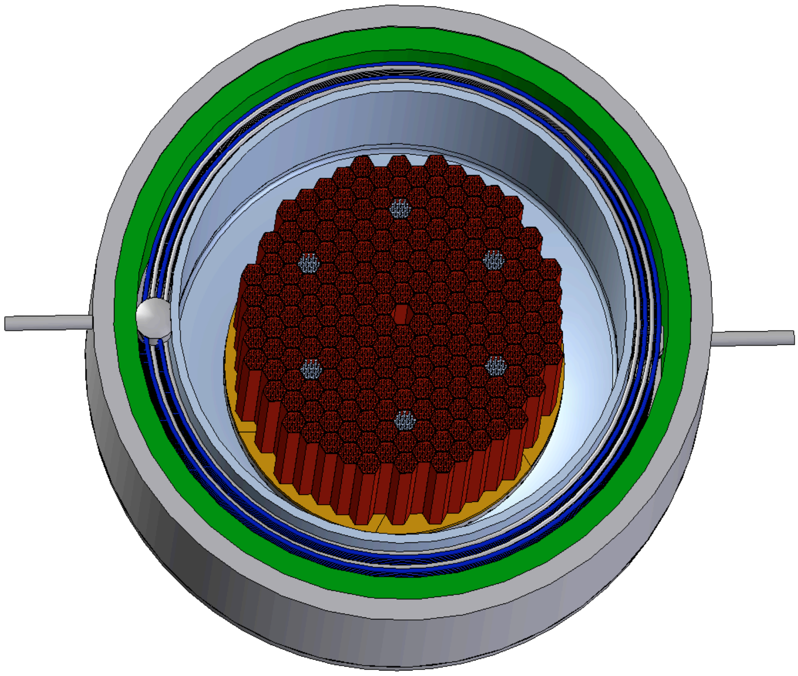

Fig. 2.24

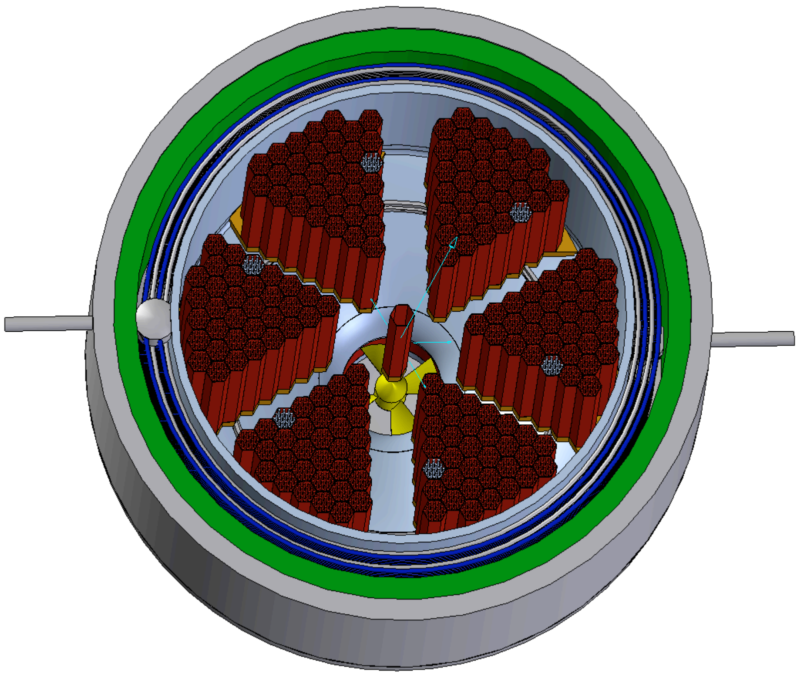

Fig. 2.25

The core is divided in six moving slices each one having grossly a mass of 20 kg, operated by a single mechanism (to be defined), The specification is that by moving apart the slices in outside direction up to a maximum equal to the thickness of the reflector, the reactivity decreases slowly to a minimum equal to that required for the overall control. The issues are the reactivity behavior versus distance and the overall controlled value. For the latter, in the case it were not enough, the number of slices can be augmented and/or the barrel against which the slices are pushed at the end of their movement can be poisoned[3]. The first one is more delicate, because in the first part of the slices movement the reactivity tends to increase slightly, because the reactor is undermoderated and by this movement the moderator is increased.
The neutronic calculation in such an articulated structure is rather complex[4]. Then it has been approached gradually. A first approximation has been obtained by using the geometry qualitatively depicted in Fig. 2.26, which represents two homogeneous axially infinite slabs, each one 4.5 cm thick and surrounded by 10 cm reflector which can be split on both sides.

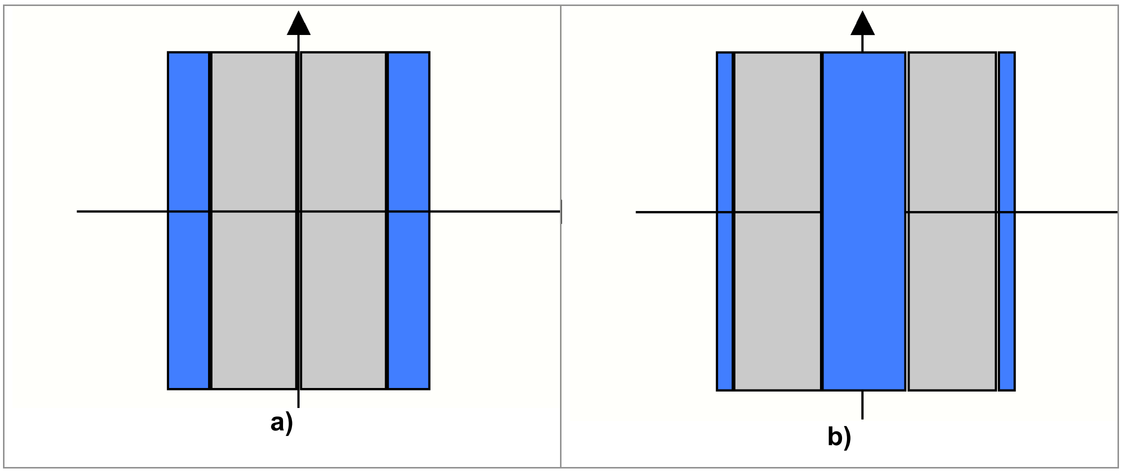

Fig. 2.26

First the slabs are joined together as shown in Fig. 2.26 a. The slabs are formed only by uranium oxide 93 % enriched and hot water (340 °C) and their ratio has been varied till obtaining approximately the same initial k_{eff} of the 800 KW core equal to about 1.08[5] . The calculations have been done by WIMS program. When the slabs are progressively taken away, as shown Fig. 2.26 b, the k_{eff} follows the curve detailed in Fig. 2.27.

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[3] The neutronic calculations done without simulating the barrel implicitly assume that all the escaping neutrons outside the reflector are lost and thus this is equivalent to have a poisoned barrel.
[4] These calculations were done in parallel with those concerning the final choice of the reactor configuration. Therefore, the input data here adopted are not completely coherent with those detailed in par.2.2.3. In our opinion the results here given should not be not appreciably affected by this discrepancy.
[5] Uranium-235 = 5.1 10^{-4}, Uranium-238 = 3.79 10^{-5}, Oxygen = 2.67 10^{-2}, Hydrogen = 5.35 10^{-2}, Boron = 7.5 10^{-6} in atoms/barn cm.

The reactivity first rises, because the initial slab is under moderated, then reaches a maximum and afterwards goes down rapidly: at 10 cm of slab distance k_{eff} is close to 1.01, not far from the final reactivity of the same 800 KW core.

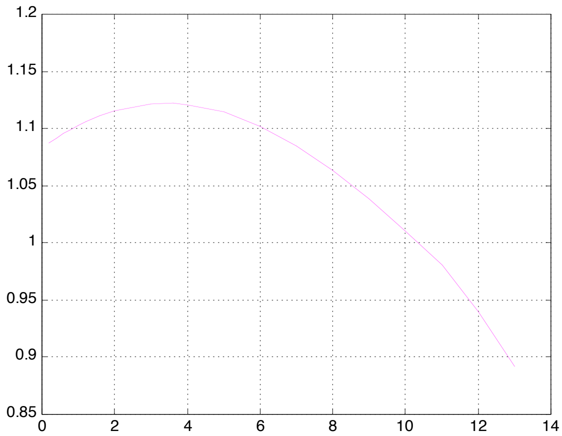

Fig. 2.27

These first results seem encouraging, then more detailed Monte Carlo calculations have been performed (see input data in Tab.2.4, but doing an interpolation for a moderation ratio = 6.5). The core has been divided in six slices as shown in the previous Fig. 2.24 and 2.25 and then starting from the compact configuration, they are taken away by three discrete steps 2.17, 8.70 and 10.86 cm respectively. The first two steps have been calculated with an equivalent reflector thickness of 11.2 cm instead of 12 cm (the present value for the reactor), The results are shown in the following Table:

Tab. 2.8 : k_{eff} for different radial displacement values

Radial slice displac. (cm)	k_{eff} – hot	k_{eff} – cold
0	1.09964	1.33186
2.17	1.12742	1.34292
8.70	0.95844	1.06874
10.86	0.85388	0.98500

These results confirm the above conclusions and in particular:

- The first k_{eff} rise and then the sharp decrease;
- A bigger decrease in cold conditions than in hot ones (24600 against 34700 pcm);

In conclusion one can imagine a sequence as follows:

- The reactor starts in cold conditions at the maximum outside displacement, where the reactivity reaches the minimum below criticality; the slices are gently approached one other until K_{eff} =1 is obtained; the reactor starts producing power;
- As soon as the temperature rises, the slices are further approached one another in order to keep constant at k_{eff} = 1;

- As soon as the temperature reaches the operating one, the full power is produced and the fuel starts burning;
- To compensate the reactivity reduction due to fuel burning, the slices are progressively approached one another till the point of maximum reactivity, which may not be the above one, because this is to be calculated with poisoned fuel;

In any condition the temperature reactivity coefficient is negative, because the cold k_{eff} curve is always higher than the hot one. Moreover, it has been verified that in correspondence of the maximum reactivity in hot conditions (1.12742) the temperature coefficient is equal to -266 pcm/°C.

If this is confirmed by further analyses, it can be said that this control procedure can in principle be adopted. The demanding issues is the design of the slice command mechanism and how to avoid that the water trapped between the slices does not mix with the outlet coolant, to avoid the lowering of its temperature below the saturation. If this is the chosen solution, the reflector should be necessarily water, as here already chosen.

2.6 The cold well

The cold well is one of the most crucial component of any thermodynamic cycle for space application . In a terrestrial power station adopting the Rankine cycle a typical specific surface of a condenser cooled by water coming from a river or a sea is about 0.02 m^2/KWth. For instance, a nuclear power station of 1000 MWe dissipates in the condenser about 2000 MWth and thus the overall condenser surface is 40`000 m^2.
Referring to the solution adopting the steam Rankine cycle with a power of 800 KW and a net efficiency of 12.5 %, the thermal power to be dissipated in the condenser is 700 KW[6]. By applying the above parameter, a surface of 14 m^2 would be obtained. However, this value is impossible to realize, because of the lack of any cooling agent and the only way to dissipate heat is radiation. The Mars atmosphere is practically made only by carbon dioxide at a pressure of 500 Pa, 200 times less than the atmospheric terrestrial pressure. Thus, this gas has very low heat transport capability, even if it cannot be discarded for this cooling action. In this case the gas is to be circulated by a blower. Some studies in the literature show that this solution is possible and perhaps more convenient than the pure radiation mechanism.
However, in the present study this possibility has not been considered, deeming that the system would be more complex and its reliability reduced, because of the blower, working with a not well defined fluid.
In a preliminary optimization study the conclusion was reached that the optimum condenser temperature for minimizing the overall mass is around 165 °C. We adopt the following relationship:

$$Q = \alpha \cdot \varepsilon \cdot S \cdot \sigma \cdot \left(T_s^4 - T_0^4\right)$$

[6] Actually the cycle gross efficiency (13.1 %) is to be taken into account for the condenser power, which is equal to 695 KW. However, the 5 KW absorbed by the various auxiliary systems are in part transformed in heat, which in some way is to be dissipated in a similar manner as for the main steam. In conclusion, the above number of 700 KW seems a good approximation of the practical situation.

where ε is the surface emissivity (0.9 as a reasonable value), S [m^2] the ideal radiating surface (without view factor reduction), $\sigma = 5.67 \cdot 10^{-8}$ [W/m^{2}K^4] Stefan-Boltzman constant, T_s [K] the tube outer temperature, T_0 [K] is assumed equal to 300 K (this value intends to simulate the back radiation coming from the Mars surface and it is equivalent to 0.25 KW/m^2), α is the view factor assumed equal to 0.6. In fact, if the tubes are disposed to form a cylinder, inside which the power system is located, the view factor would be lower than 1 and 0.6 is a tentative value: the view factor can be defined when the condenser layout is established.

With the above data, a value of 1.14 m^2/KW is obtained (57 times more than the above typical value).

A parametric study shows that the condenser geometry is made by a bundle of several tubes connected in parallel, having the following features:

ID = paramter to be optimized (inner diameter);

t = 7 % of ID (tube thickness)

L = 80 m (length of each tube)

N = tube number: depends on ID

The internal pressure of 0.7 MPa is relatively low, and the above thickness of 7 % ID is well beyond the value needed to satisfy the stress limit; then it has been fixed on the basis to assure sufficient tube rigidity, to cope with a given corrosion and to resist to micrometeorites impact. The latter requirement probably would impose a constant thickness instead of a constant percentage of ID; however, taking into account the scarcity of knowledge on this issue, it seemed more reasonable the choice of a constant percentage of ID, as usually done in terrestrial applications.

The material, working in this case at relatively low temperature, may be titanium, the density of which is 4500 against 7800 kg/m^3 of steel.

In fig. 2.28 the resulting values of overall mass and pressure drops of the condenser are shown. The overall mass takes into account a 30 % increase of the overall tube mass for spacers and headers. The graph shows that a linear decrease of the overall mass can be obtained by reducing the inner diameter. However, the condenser pressure drops are inversely proportional to the third power of ID: this means a reduction of the cycle efficiency, because its backpressure increases accordingly. A simplified evaluation indicates that in correspondence of 20 KPa of backpressure increase, the efficiency reduces by about 1 %. This efficiency reduction is reasonably acceptable and it is obtained with ID = 6 mm. Probably an optimization study would suggest that even a further reduction in efficiency may be convenient, taking into account the corresponding mass saving. However a smaller ID would imply a proportional decrease in tube thickness, and probably a not acceptable reduction in tube strength against meteorites.

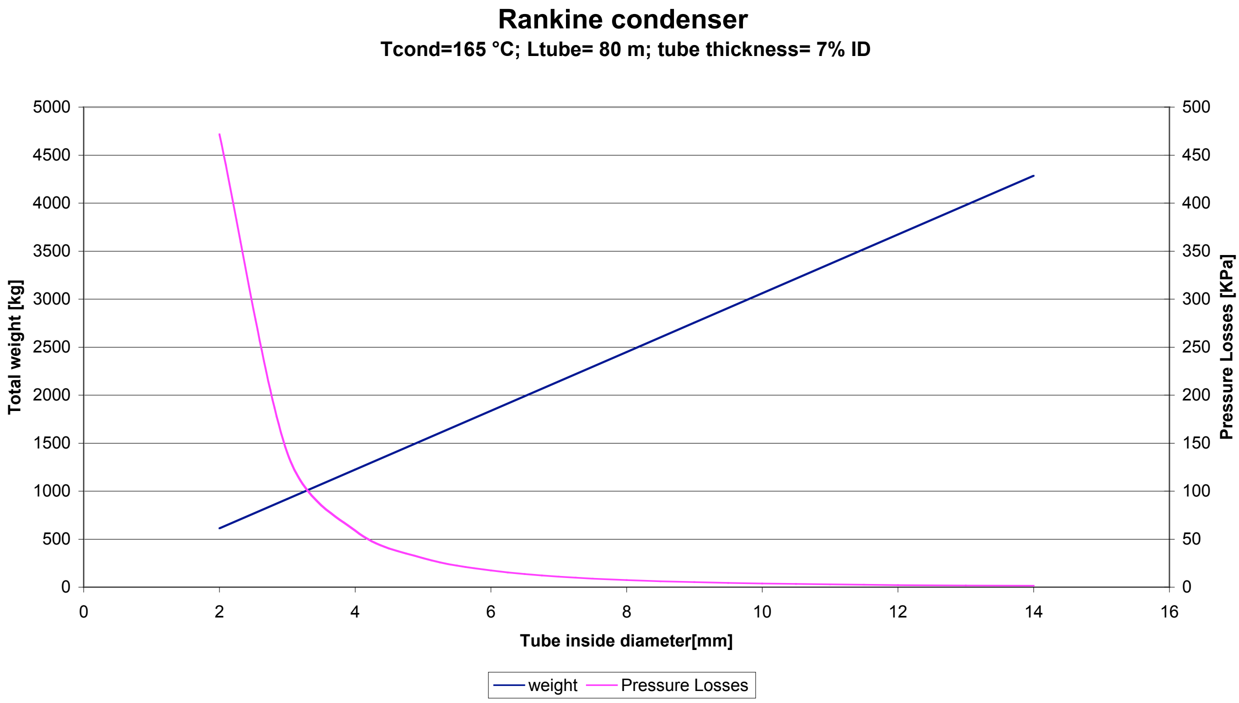

Fig.2.28

Then the overall mass results 1840, the overall surface 796 m^2 and 464 tubes (2.6 kg/KW or 2.3 kg/m^2).
It has been verified that the temperature drop inside the tube between the wall and the fluid is few tenths of degree, then negligible: the radiation temperature of 165 °C is practically identical to the condenser temperature. The pressure drops inside the tubes are about 20 Kpa, with respect to the absolute pressure of 700 KPa.
The 80 m long tubes are supposed to form 4 contiguous U, 10 m high, for a total of 3708 legs. Assuming that the tube legs are welded together, an overall linear dimension of 3708x0.00684 = 25.4 m is obtained. In conclusion, the condenser can be imagined to be a cylinder of 8 m diameter and 10 m height. The final design would take into account the real pay load size of the launching rocket. In particular, if necessary the condenser can be divided in several identical pieces to be assembled on the site.
In the case of organic Rankine cycle the power to be dissipated is 555-100 = 455 KW. Even if in this case we do not know the thermodynamic properties of the organic fluid, we can imagine, on the basis of the above results, that the radiation mechanism is the controlling one, so that the overall dimensions are about proportional to the power to be dissipated. Thus, adopting the same tubes their number is decreased to 464x(455/700) = 302. By the same hypotheses, the condenser would be a cylinder of 5.3 m diameter and 10 m height, while the total mass is 1840x(455/700) = 1196 kg.
All the above data are listed in par. 2.11. Figs. 2.29 and 2.30 show a simplified scheme and the layout of the plant for the 800 KW reactor, respectively.

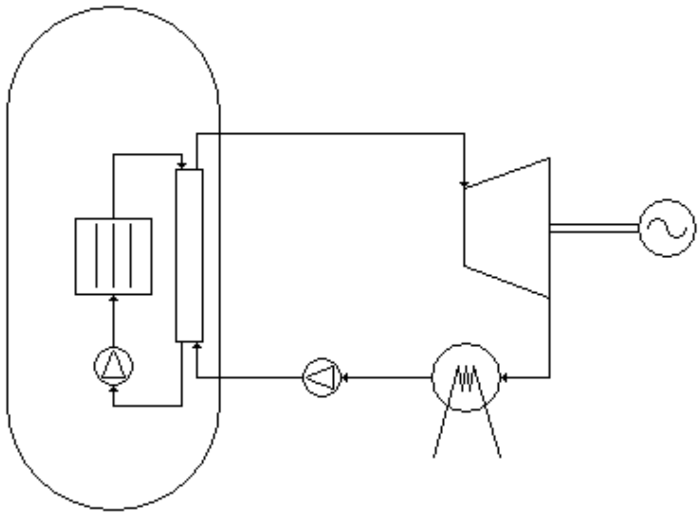

Fig. 2.29

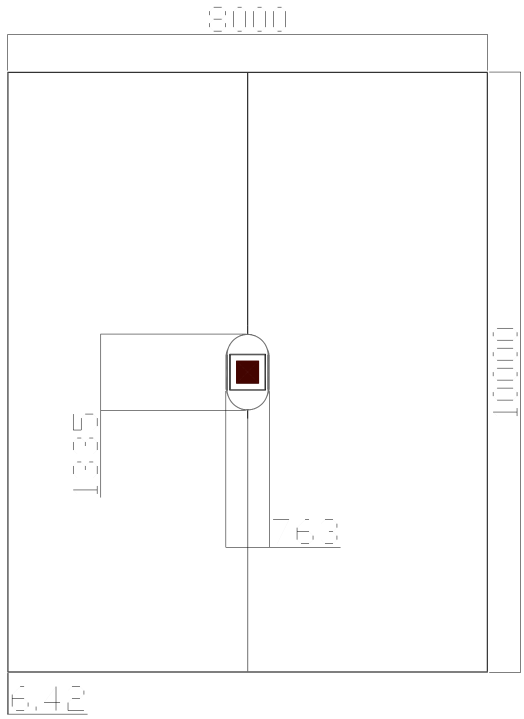

Fig. 2.30

2.7 Masses

In the previous sections the masses of the core, primary system and cold well has been detailed for both the 800 and the 555 KW cores. We have not considered some further components of the complete system yet, as for example the steam turbine, the tubes, valves, auxiliary circuits, internals and so on. Concerning the small steam turbine, it is worth noting that this issue certainly requires an R. & D. program in order to build steam turbine of about 100 KW with very compact size and high efficiency at the same time. We estimate for the moment a value of 2 kg/KWe for the mass per unit electrical power, resulting in a mass of 200 kg for this component. The mass of the other not considered parts is globally assumed to be about 200 kg for both reactors. Actually, in 555 KW the pressure of the secondary circuit is lower but there is as an extra component, the regenerator.

At the end the mass of any single component has been increased of a margin of 5 % to take into acount any kind uncertainty or disregarded parts. The final sum has then been increased by a further 10 % as a contingency. The detailed values are shown in Tab 2.9.

The resulting overall masses are 3941 and 3158 kg for 800 and 55 KW respectively. This is equivalent to a specific mass, which ranges between 40 and 32 kg/KWe, which seems a reasonable interval, being in agreement with other proposals in the literature.

Tab. 2.9 : Masses in kg of PWR

	800 KW				555 KW			
Fuel (UO_2 only)	47 +				42 +			
Cladding	15 +				14 +			
Shroud	16 +				15 +			
Moderator (cold)	30 +				29 +			
Reflector (cold)	139				134			
Core+reflector		= 247 +				= 234 +		
Barrel		138 +				135 +		
Vessel		502 +				489 +		
Downcomer water (cold)		42 +				41 +		
Dome water (cold)		183				179		
Overall reactor			= 1112	+ 5 % 1168			= 1078	+ 5 % 1132
Cold well			1840	+ 5 % 1932			1196	+ 5 % 1256
Steam Generator			60	+ 5 % 63			60	+ 5 % 63
Turbine			200	+ 5 % 210			200	+ 5 % 210
Other components			200	+ 5 % 210			200	+ 5 % 210
Overall system				3583				2871
Contingency = 10 %				358				287
Total Mass of the System				**3941**				**3158**

2.8 Preliminary safety considerations

A detailed safety analysis is outside the scope of this feasibility study, for its complexity and need to define the detailed requirements. In fact this reactor from one side is nor subjected to the licensing procedure of terrestrial reactors imposed by the safety authorities, from the other it must satisfy specific safety issues connected to its launch and the possibility to fall down to the earth.

In the foreword it is mentioned that this nuclear system must satisfy the usual safety requirements of terrestrial reactors and this is to be defined in detail, taking into account the above consideration about the lack of a licensing procedure. Besides this the system has to assure that:

- no irradiated fuel is present at launch;
- the core subcriticality in the case of possible launch accidents (flooding);
- the radiation protection without impairing mass requirements;
- an easy decommissioning in space;

The first item is inherently satisfied, because the reactor would not reach its first criticality before being outside terrestrial space. The second one seems inherently satisfied because a water reactor cannot be *flooded.*

The third is a an important issue, which can be addressed only after having defined some conditions, especially for the propulsion solution. In fact, for surface reactor, the shield cannot be transported from the earth and it is to be provided by a suitable system layout on the Mars surface (regoliths around, underground siting, big distances). For propulsion reactor an intermediate solution is to be found by balancing the addition of a small shield around the core with the reduction of the radiation danger by locating the reactor far away from the sensible zone (a separate capsule for the reactor?) or by locate an extra shield only on the reactor portion *viewed* by the sensible zone.

The fourth one is too indefinite at this stage of the design, that no specific consideration can be drawn.

In this study, a calculation has been done to verify whether in the case of severe accidents the fuel melting is avoided. It is well known that fuel melting represents the most feared situation for terrestrial reactor, and the nowadays attitude for new reactor designs is to avoid this in any foreseeable circumstance. It is not an easy task, but in the present case, the very small size of the reactor may be of big help. If this is the case, it is justified the choice to eliminate any protection system to cool down the fuel in the case of a complete loss of coolant (LOCA).

The fuel melting is connected to the fact that in the case of LOCA the reactor is inherently shut down, because the loss of moderator makes the reactivity go down to very low values, but the fuel goes on producing some power, the so called *decay heat.* If the fuel is no longer cooled by the water, the fuel heats up adiabatically till it reaches its melting point. However, as soon as the fuel temperature rises, the thermal radiation process takes place, the importance of which increases rapidly with the temperature. This radiation power is exchanged among the rods inside the core and from the outer rods ring toward the vessel and then from the latter toward the outside environment. Besides the radiation, there is also the convection of steam or air, which flowing inside the hot core brings its heat to the vessel walls and from them to the outside world.

It is really difficult to simulate this situation by a model. A rather simplified but sufficiently realistic one has been prepared, limiting conservatively the study only to the radiation process.

The temperature distribution inside the core has been evaluated using a computer program, based on a model in which the whole core is divided in unit cells, having the geometry indicated in Fig. 2.31 [4]. Each cell is made by four rod segments, each one equal to one quarter of circumference and by four imaginary surfaces. The cells are connected one another by means of common imaginary surfaces, which transform the impinging heat flux in a isotropic flux for the next unit.

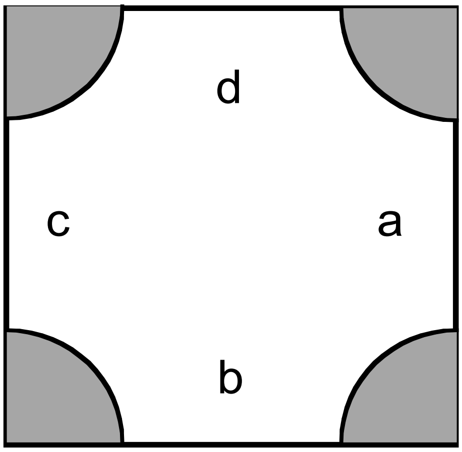

Fig. 2.31

The hypothesis are:

- The fuel rod are in a square open lattice and are axially infinite;
- The rod power is axially uniform;
- The surface rod temperature is uniform;
- The emissivity of all surfaces is the same;
- Emission and reflection are isotropic;
- The rod surface is gray, while the connecting surfaces between two cells are black;
- The calculation is done in steady state, i.e. the decay power is assumed constant with time.

Two differences between the core geometry assumed in the first hypothesis and the present one: the lattice is now hexagonal, and the rods are assembled in 19 rods and inserted in hexagonal channels. Both differences should not modify appreciably the result: for the first point what is important is the moderator to fuel ratio more than the lattice shape; for the second difference the effect can be more pronounced, but anyway these shroud surfaces should introduce a further thermal resistance among the rods, but being its temperature relatively high their effect should be modest. However both these differences will be analyzed in further detail in the future.

Once fixed the rod pitch, the distance of the outer rod row from the cold well, the rod diameter, the program calculates the shape factors, defined as the fraction of the radiation leaving a surface A_i in whichever direction and intercepted by a surface A_j. These coefficients are then used to calculate the shape factors modified ψ_{ij}, which represent the absorbed energy per unit time (the intercepted energy minus the reflected one) by surface A_j when a given heat flux is emitted from surface A_i. Thus, determined the modified shape factors and the connection method among the cells, the system of the balance equations is solved. Assigned the power of each rod and the temperature of the cold well, the temperature profile along the core radius can be calculated. The equations number is equal to the rods number along the core radius and the unknowns are the net energies per unit time which leave each rod. The form of each equation is:

$$q_i = \varepsilon_i \cdot A_i \cdot \sigma \cdot T_i^4 - \sum_j \left(A_j \cdot \sigma \cdot T_j^4 \cdot \psi_{ji} \right)$$

where ε_i is the emissivity of surface A_i, σ is the Stefan-Boltzmann constant, equal to $5.67 \cdot 10^{-8}$ W/m^2 K^4 and T_i is the absolute temperature of surface A_i. The first term in the second member represents the emitted power from surface A_i, while the second term is the absorbed power by A_i coming from all surfaces A_J.

The power per unit rod surface has been calculated as

$$P_s = \frac{P_d}{rodnumber \cdot \pi \cdot D_{rod} \cdot H_{core}}$$

where P_d is the decay power, supposed 1 % of the nominal thermal core power. The last one is the thermal power of the reactor and this means that a flat radial and axial profile has been assumed.

The vessel wall temperature is set at 750 K, but its value is not critical. In fact in one typical case going from 500 to 1000 K the central rod temperature changes by 80 K. The rod emissivity has been set at 0.75. The rod pitch is 6.9 (V_m/V_u = 17.2) and 5.6 mm (V_m/V_u = 10.8) for 555 and 833 KW respectively. The 833 KW power was used when these calculations were done, and then it was reduced a little bit coherently with an increased value of the cycle efficiency.

Once fixed the above data the only controlling parameter is the reactor power. The results are detailed in Fig. 2.32 for the two powers of 555 and 833 KW. It can be seen that the maximum temperature is far from the melting point of stainless steel (1700 K) and even more from that of uranium oxide (3000 K). If instead of the average power of each rod the maximum radial one equal to 1.5 times is used the resulting temperature is increased of about 100 K. However, this analysis should be improved in the future to take into account the shrouds, the radial and axial flux distribution and the effect of rod pitch.

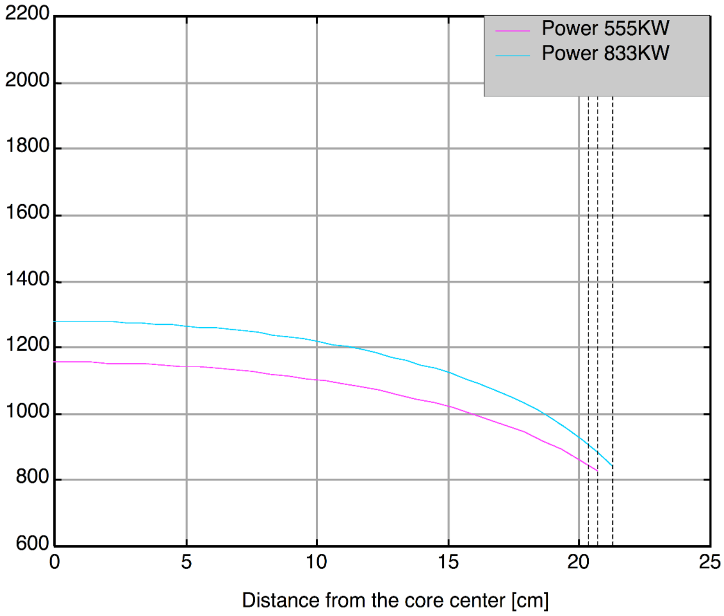

Fig. 2.32

2.9 Open issues and R&D needs

This feasibility study has allowed us to find the open issues to be solved for going on this route, which needs a R&D program. Here below a list of them, which is inevitably incomplete, is detailed; the order is not based on their importance. The issues written in **bold letters** are those interesting for terrestrial reactors and those written in *italic* those interesting for generic terrestrial applications as well.

- Fuel fabrication process of small diameter pellets;
- Fuel irradiation behavior due to the new geometry;
- Fuel gap definition;
- Cladding material: stainless steel or zircalloy;
- Hexagonal shrouds: material, dimension stability, fabrication;
- Internals: mechanical design;
- **Increase of operating pressure: fuel implications, primary circuit materials;**
- **Saturation temperature at the reactor outlet: effect of small boiling inside the core;**
- The cold well as condenser;
- *Small steam turbines;*
- *Organic fluids: composition stability, thermal transport capabilities;*
- *Small organic fluid turbine;*
- **Fluid leakage: how much, how to cope with;**
- **Maintenance requirements of the whole system;**
- **Optimum reflector: technological aspects**
- **Pumps: development of spool pumps, reliability for long periods;**

- Fluence effects on vessel in these particular conditions;
- Shielding;
- Safety valves: reliability, how to cope their intervention;
- *Vessel material different from stainless steel*;
- **Steam Generator thermalhydraulic behavior in helical geometry** also in presence of low or no gravity;
- **Corrosion deposits inside the SG tube;**
- **The pressurizer: self pressurization, different concepts for propulsion reactor as feed and bleed, cold pressurizer,** centrifugal action**;**
- Control of the system and of the reactor and its constructive implications.

Even if this list is incomplete, no item seems to be unsolvable. An R&D program of reasonable extent may yield the needed answers, but what is important that the most demanding researches also are of interest for the new generation Light Water Reactors. Thus a cost sharing action can be proposed and duly programmed, according to the time schedule of the commercial exploitations of these terrestrial reactors.

2.10 Concluding remarks

At the end of this very preliminary feasibility study about the use of PWR system for space reactors, it can be concluded that no insoluble issues have been evidenced, which would prevent of going on along this route in order to execute a more detailed design. Then it will be possible to draw a more justified conclusion about the usefulness to follow this solution.

At the beginning of the study it was supposed that the solutions for propulsion and surface application might be the same. However, this hypothesis holds only partially, because the lack of gravity and of a soil render the propulsion solution rather different and more demanding than the surface one. In particular, two aspects have been outlined for propulsion reactors: the lack of steam water separation in case of lack of gravity (pressurizer, steam moisture separation), and the need of an autonomous radiation shield, which in surface reactors can be provided by the existence of a soil. On the other side, it was anticipated in the foreword that the use of space nuclear reactors should be approached gradually starting from the easiest application, which is that for surface use: this study is a confirmation of the statement.

If it will be confirmed in prosecution of the work that no insoluble issues are present in this proposal, it can be stated that a reasonable R&D effort and consequently a relatively limited development cost and time interval are only needed in this case.

In the short range, future design activities should address the detailing of many aspects of the analysis presented in this report and adding new ones. Among the first ones, concerning the core, the choices to limit the fuel burnup, the use of stainless steel instead of zircaloy for cladding and shroud, the reflector material should be reconsidered: in fact these conservative choices affect the reactor size, which is an important item to define the overall mass. While for the rest of the system: cold well in forced convection, reactivity and plant control. The new activities are: radiation shielding, vessel fluence, safety aspects, choice of vessel/barrel material, overall layout, containment, leakage control, ancillary circuits for start up, coolant purification, radiolisis. and other exigencies. Moreover, at the end of this further activity a preliminary R.& D. program should be detailed.

2.11 PWR reactor list of data

	PWR - 800	PWR - 555
Power		
Thermal [KW]	800	555
Electric [KW]	100	
Primary circuit		
Fuel		
Composition	UO_2	
Enrichment BOL	93%	
Enrichment EOL	~ 82%	
Diameter [mm]	1,8	
Cladding		
Material	Stainless steel	
Thickness [mm]	0,2	
Equivalent density [kg/m^3]	6800	
Fuel channel		
Geometry	Hexagonal	
# of rod	19	
Fuel rod diameter [mm]	2,2	
Height [mm]	361,5	352,7
Moderation ratio	6,5	7
Fuel pitch [mm]	4,846	4,996
Shroud material	Stainless steel	
Shroud thickness [mm]	0,3	
Shroud density [kg/m^3]	7800	
# of fuel bundle	261	239
total # of fuel rod	4959	4541
Burnup [MWd/kg] (assumed)	77	60
Pressure [MPa] (assumed)	15,5	
Temperatures		
T max [°C] (assumed)	345	
T min [°C] (assumed)	335	
Secondary circuit		
Cold well temperature [°C]	165	
Thermodinamic cycle	Rankine steam	Rankine organic fluid
Efficiency [%]	12,5	18

Miscellaneous data		
Fuel quantity (UO_2) [kg]	47	42
Specific power [KW/kg]	17	13,2
Linear power [KW/m]	0,39	
Reflector		
Material	water	
Thickness [mm]	120	
Core designs		
keff EOL (assumed)	1,000	
Mass (core+reflector) [kg]	247	234
Electrical power generation		
Working fluid	water	$C_{10}H_{14}$
Inlet steam temperature [°C]	335	
Condensing temperature [°C]	165	
Max discharge moisture (assumed)	3%	N.A.
Turbine efficiency (assumed)	73%	N.A.
Feed pump efficiency (assumed)	75%	N.A.
Absorbed el. power [KW] (assumed)	5	
Reactor vessel		
Design pressure [MPa]	17,05	
Geometry [mm]		
a	120	
b	33	
c	120	
Material	SA 508, TP.3, Cl.2	
Thickness [mm]		
Cylindrical portion	29,3	29,1
Spherical portion	14,7	14,5
Barrel		
Geometry	Cylinder	
Material	SA 508, TP.3, Cl.2	
Thikness [mm]	15	
Vessel mass [kg]	502	489
Barrel mass [kg]	138	135
Core+reflector mass [kg]	247	234
Water (cold)	225	220
Overall vessel mass filled with cold water [kg]	1112	1078
Steam generator		
Geometry	Helical single tube around the barrel	
Annular gap width [mm]	33	
ID/OD/T [mm]	20 / 24,4 / 2,2	
Tube length [m]	50	
Tube material	Inconel TT 690	
Coil diameter [m]	0,68	
Coils number	23	
SG height [m]	0,80	

SG mass [kg]	60	
Primary fluid temp. [°C]		
inlet	345	
outlet	335	
Secondary fluid temp. [°C]		
inlet	165	
outlet	335	
Primary/Secondary pressures [MPa]	15,5 / 5,7	15.5 / N.A.
Primary flow rate [kg/s]	10,1	N.A.
Max primary mass flux [kg/m^{2}s]	549	N.A.
Secondary flow rate [kg/s]	0,347	N.A.
Secondary mass flux [kg/m^{2}s]	1103	N.A.
Primary SG pressure drops [KPa]	5	N.A.
Secondary SG pressure drops [KPa]	500	N.A.
Pressurizer		
Type	Integrated	
Steam volume [liters]	24	17
Circulating pump		
Type	Spool pump	
Operating pressure [MPa]	15,5	
Head [KPa] (assumed)	20	
Operating temperature [°C]	335	
Mass flow [kg/s]	10,1	7,0
Volumetric flow [liters/s]	16,1	11,1
Efficiency	0,6	
Power [W]	540	370
Reactivity control		
Term 1 [pcm]	23000	
Term 2 [pcm]	500	
Term 3 [pcm]	5500	4300
Term 4 [pcm]	~ 0^+	
Total [pcm]	27800	29000
# of moving slices	6	
slice mass [kg]	~ 20	
Cold well		
Type	Bundle of tubes	
Tube geometry		
Inner diameter [mm]	6	
Outer diameter [mm]	6,84	
Thickness [mm]	0,42	
Length [m]	80	
Tube number	464	302
Internal pressure [MPa]	0,7	N.A.
Temperature [°C]	165	
Specific surface [m^2/KW]	1,1	
View factor	0,6	
Surface area [m^2]	796	302
Mass [kg]	1840	1196
Dimension (cylinder)		
height [m]	10	
diameter[m]	8	5,3

Masses [kg] (+5%)		
Overall vessel weight filled with cold water	1168	1132
Cold well	1932	1256
Steam generator	63	
Turbine	210	
Other components	210	
Overall system	**3583**	**2871**
Contingency [10%]	358	287
Total	**3941**	**3158**

3. THE HTGR

3.1 Introduction

The idea is to extend as much as possible the HTGR technology adopted for producing high powers in terrestrial applications to the design of a reactor suited for space conditions. However a number of modifications are needed. Let us summarize them.

Fuel composition: conventional powder of uranium oxide, sintered in micro spheres (see below).

Fuel enrichment: 93 % in uranium 235:

Fuel micro sphere diameter: the HTGR technology foresees different values of this diameter all around 300-600 μm. In this case the choice has been 350 μm.

Cladding material and thickness: the fuel micro spheres are protected by four carbon based layers: a low density pyrolitic carbon buffer, high density inner pyrolitic carbon layer, silicon carbide layer, high density, outer pyrolitic carbon layer. The overall thickness is assumed equal to 400 μm. For the neutronic calculations the layers are assumed to be equal to pure graphite (density 1800 kg/m^3).

Overall micro sphere diameter: the outer diameter turns out to be 750 μm.

Fuel "rod", said *compact*: the micro spheres are mixed with a graphite powder and then compacted to form an hexagonal rod or *compact* having the apothem of 3.8 mm, while the length is that of the reactor height and thus it is the result of the neutronic calculations to define the core size. Taking into account the graphite layers of the fuel micro spheres and the graphite powder, the overall uranium oxide volume content in the compact is 9.4 %.

Fuel-moderator-coolant channels: the moderator is graphite under the form of hexagonal blocks having the same length of the reactor height. The blocks have an apothem which depends on the moderation ratio: for a moderation ratio of 7.5 and 9.5, its value is 29.4 and 32.7 mm respectively. The blocks are drilled by two types of hexagonal holes of the same size: six of them are for the compacts and one for the coolant. The six are symmetrically located in the hexagon, while the coolant one is located in the center as shown in Fig. 3.1. The blocks are then assembled together to form the reactor core. The moderator is then made by these blocks and by the graphite already in the compacts.

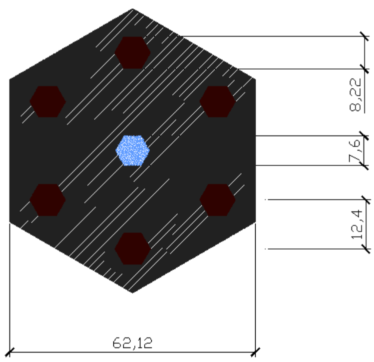

Fig. 3.1

Fuel burnup: an average value of 100 MWd/kgU (maximum value) is assumed, which is about the same value presently adopted in HTGRs. In our case there are two opposite effects when compared to HTGRs: i) the fuel power density is much lower and then also the corresponding maximum compact temperature: this is a real favorable condition, being the fuel performance much improved in these conditions; ii) the more peaked flux distribution inside the core and the lack of periodic fuel shuffling imply a higher maximum burnup value in correspondence of the same average one, then worsening the fuel damage effects in the maximum flux position. In spite of the fact that the first item seems more than to counterbalance the second one, the decision was taken to adopt the same HTGR value.

Temperatures and pressures: the maximum reactor temperature at the outlet is a parameter of paramount importance both for the thermodynamic efficiency and for the technological constrains of the system. A reasonable and well experienced value is 800 °C, however several designs for terrestrial reactors now under development foresee higher values up to 900 °C and beyond. In this study the value of 900 °C has then been adopted. The minimum temperature value at the reactor inlet is important for the thermoelectric generator, because its efficiency depends on the average temperature between the maximum and minimum one, while for a gas cycle it is the result of the optimization of the coupling between the reactor and the cycle (see below). Because for gas cycle a value of 725 °C turned out to be the optimum value, the same relatively high value has been adopted for the thermoelectric device as well (see par.3.3 for some considerations about this value). As far as the pressure is concerned, its value in practice does not impact on the efficiency, but only on the transport properties of helium. The final value will be better chosen when a detailed design of the core, the regenerator and the cold well will be defined. For the time being the minimum pressure has been assumed a rather usual value equal to 3 MPa, while the maximum one depends on the optimum compression ratio (1.6 in the chosen cycle, giving a maximum pressure of 4.8 MPa).

Cold well temperature: in this case the choice depends on the generator type adopted. See pars.3.3 and 3.4.

Electrical generator: two alternate designs are possible i.e.: *thermoelectric generator, the Brayton gas cycle*. The thermoelectric generator is a possible and interesting solution in this case, thanks to the relatively high temperatures. In fact a net efficiency of 4.5 % has been calculated (see details in par.3.4), which is a rather low value, if compared with efficiencies obtainable with gas cycles, but sufficient to justify the use of this option, considering that this generator is simple, reliable and experienced. Then for obtaining 100 KWe net power, the reactor thermal power is to be 2219 KWth. The direct Brayton gas cycle is characterized by a much higher net efficiency equal to 24 % (see details in par.3.3). This leads to a value of thermal power equal to 417 KW.

Minimum fuel quantity: set the thermal power, the burnup, the full power duration (4000 days), we obtain for the above thermal powers the following minimum UO_2 fuel masses: 100 and 19 kg of UO_2 respectively. The possibility to adopt the above minimum masses is strictly connected to the reactor neutronic design. In fact, the small size of these reactors, roughly two orders of magnitude lower than those of a conventional HTGRs, implies big fractions of escaping neutrons from the core surface. Present calculations show that these minimum masses are not enough for 417 core to satisfy the imposed fuel life, unless a rather thick reflector is adopted (see below). This means that the maximum fuel burnup is lower and this can be positive for a better fuel performance.

Core geometry and reflector: the core geometry is based on the assumption to have a cylinder with the diameter equal to the height. Actually the neutronic optimum would be obtained for a ratio height to diameter of 0.92; however, taking into account the effect on the size and mass of the overall system it has been assumed a priori that a ratio equal to 1 would be simpler, better and not far from the usual one. The actual size will depend on the needed mass of the fuel and the value of moderator to fuel ratio, which are given by the neutronic design. The reflector is a layer of 5 cm of graphite all around the core. The reflector thickness is an important item because its increment serves for reducing the escaping neutrons, which are in percentage an order of magnitude higher than those in terrestrial HTGRs, but on the other side it substantially increases the mass and the size of the overall system. It is usual for these special nuclear reactors to adopt more sophisticated solutions for the moderator, which foresee the use of materials different from graphite (e.g. beryllium oxide) in order to optimize the above mentioned opposite requirements. However, this may be done in a prosecution of the program, but at the present stage the adopted choice seems reasonable, also because any other one would not improve substantially the overall result.

Turbine and compressor: these are two important components of the generator, which should undergo a thorough verification for these small sizes and high reliability needed for long period of time operating at very high temperature. In particular the gas leakage raises some concern, because if present, even if to a reduced extent, would determine big impacts on the system design: containment, reinsertion in the circuit at high pressure. This concern has been coped with in this study by the decision to put all the rotating machines inside the pressure vessel (see par. 3.5).

3.2 Neutronic design

3.2.1 <u>Design codes</u>

In this case the WIMS (*Winfrith Improved Multi group Scheme)* [1] calculation program has been used, as already done for the PWR solution. WIMS is a deterministic computation program, which uses a wide variety of calculation methods to solve the reactor physics problems. It is suitable to study any kind of thermal reactors. A synthetic description is given in sec.2.2.1, to which it is referred.

As the effective multiplication factor strongly depends on the *buckling* values introduced in input (see sec.2.2.1), it seemed important to compare the results obtained by WIMS with those of a Monte Carlo program.

The Monte Carlo programs are highly reliable, but they have the drawback that they cannot be easily used to simulate the fuel evolution along the life. Therefore, this comparison was made in four specific

conditions and namely: infinite lattice and actual reflected reactors for two powers at BOL, in cold and hot conditions, by varying the moderation ratio.
The Monte Carlo code here used is the well known MCNP-4C, as distributed by NEA Data Bank [2].

3.2.2 WIMS and Monte Carlo comparison

The Monte Carlo results are detailed in the following Tables and Figures.

1. Tab. 3.1 gives the k_∞ values obtained by WIMS and MCNP-4C and the k_∞ differences between the two programs, versus moderation ratio from 1 to 20, both in cold and hot conditions; Figs.3.2 and 3.3 show the MCNP-4C and WIMS k_∞ in graphical form in all the above conditions; while Fig.3.4 shows the differences in k_∞ between the two programs;
2. Tab. 3.2 gives the k_{eff} values obtained by both programs in cold and hot versus moderation ratio from 7 to 11 (which is the range of the foreseen solution). k_{eff} is calculated for both powers with 100 kg of UO_2 fuel (see also sec.3.2.3). Preliminary neutronic calculations show that this mass allows to minimize the overall mass of fuel+moderator+reflector+vessel. In fact, lower fuel masses require higher moderation ratio and higher reflector thickness in order to obtain the needed reactivity. Keeping constant the fuel mass in both reactors the k_{eff} at BOL does not depend on power, but only on the moderation ratio; for this reason the data presented refer to both the 2219 and the 417 KW cores; Figs.3.5 and 3.6 show k_{eff} of each program in graphical form, while Fig.3.7 shows the k_{eff} differences between the two programs;

Tab 3.1 – K_∞ : Comparison WIMS-MCNP

	WIMS		MCNP		delta K : MCNP - WIMS	
MR	Cold	Hot	Cold	Hot	Cold	Hot
1	1.671124	1.66680	1.71442	1.71471	4330	4791
2	1.627721	1.62199	1.66840	1.66823	4068	4624
3	1.621616	1.61531	1.65460	1.65443	3298	3912
4	1.631869	1.62534	1.65706	1.65769	2519	3235
5	1.648817	1.64216	1.66835	1.66786	1953	2570
6	1.668138	1.66136	1.68300	1.68262	1486	2126
7	1.687726	1.68092	1.69943	1.69835	1170	1743
8	1.706729	1.69979	1.71435	1.71277	762	1298
9	1.724672	1.71751	1.72972	1.72834	505	1083
10	1.741408	1.73397	1.74394	1.74248	253	851
11	1.756904	1.74915	1.75850	1.75691	160	776
12	1.771200	1.76312	1.76992	1.76779	-128	467
13	1.784402	1.77596	1.78119	1.77902	-321	306
14	1.796551	1.78775	1.79179	1.78930	-476	155
15	1.807742	1.79859	1.80144	1.79842	-630	-17
16	1.818060	1.80857	1.81114	1.80760	-692	-97
17	1.827608	1.81778	1.81915	1.81567	-846	-211
18	1.836440	1.82629	1.82693	1.82340	-951	-289
19	1.844600	1.83433	1.83386	1.82969	-1074	-464
20	1.852158	1.84142	1.84044	1.83637	-1172	-505

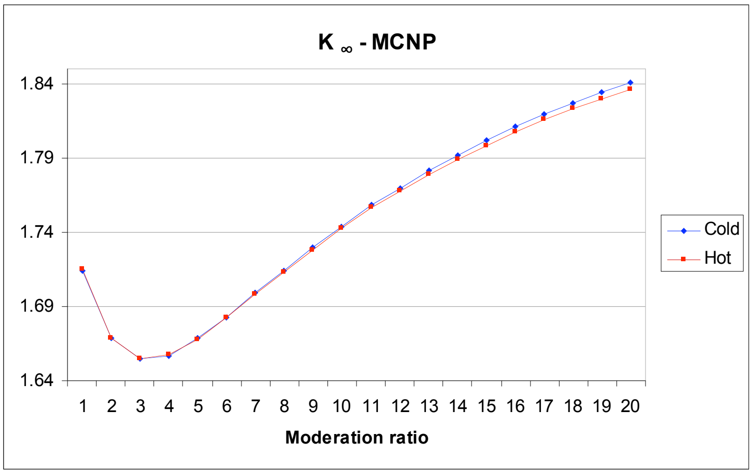

Fig. 3.2

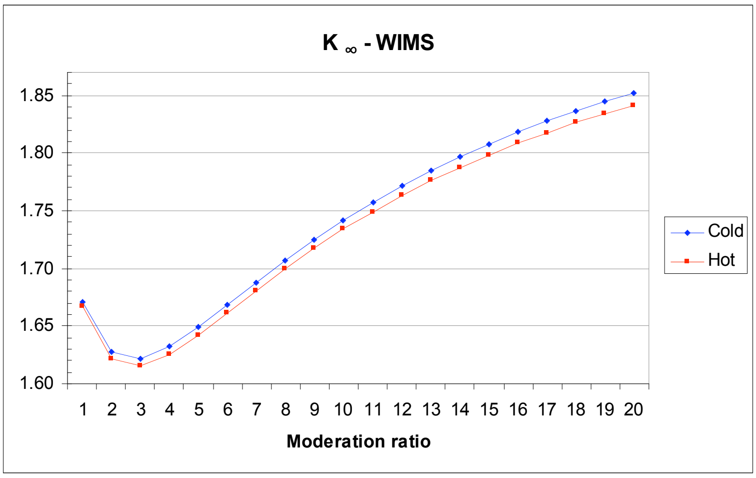

Fig. 3.3

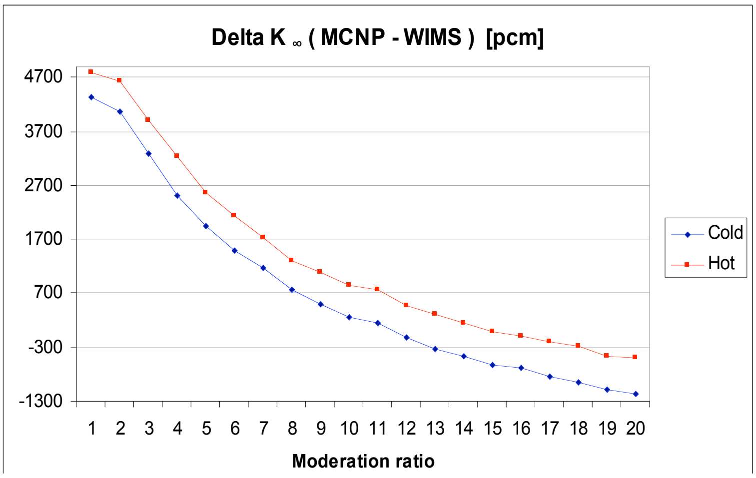

Fig. 3.4

Tab. 3.2 - K effective: MCNP- 4C , WIMS (Beginning of life)

Reactor type			**HTGR**						
	Electric power [KW]		100						
Geometrical data									
	fuel diameter [mm]		4,32		UO$_2$ fuel mass [kg]				100
	reflector thickness [mm]		50		Specific power [KW/kg]				13,77
Temperature (hot):					Temperature (cold):				
	fuel [°C]		775		All components [°C]			27	
	moderator [°C]		775						
	coolant [°C]		775						

			MCNP		WIMS		delta K MCNP - WIMS	
MR	# bundle	D [cm]	Cold	Hot	Cold	Hot	Cold	Hot
7	288	101,42	1,02301	1,02547	1,01425	1,00997	876	1550
8	277	105,68	1,05685	1,05842	1,04606	1,04131	1079	1711
9	268	111,35	1,09299	1,09274	1,07568	1,07040	1731	2234
10	259	114,43	1,11734	1,11949	1,10324	1,09737	1410	2212
11	252	116,62	1,14266	1,14380	1,12886	1,12235	1380	2145

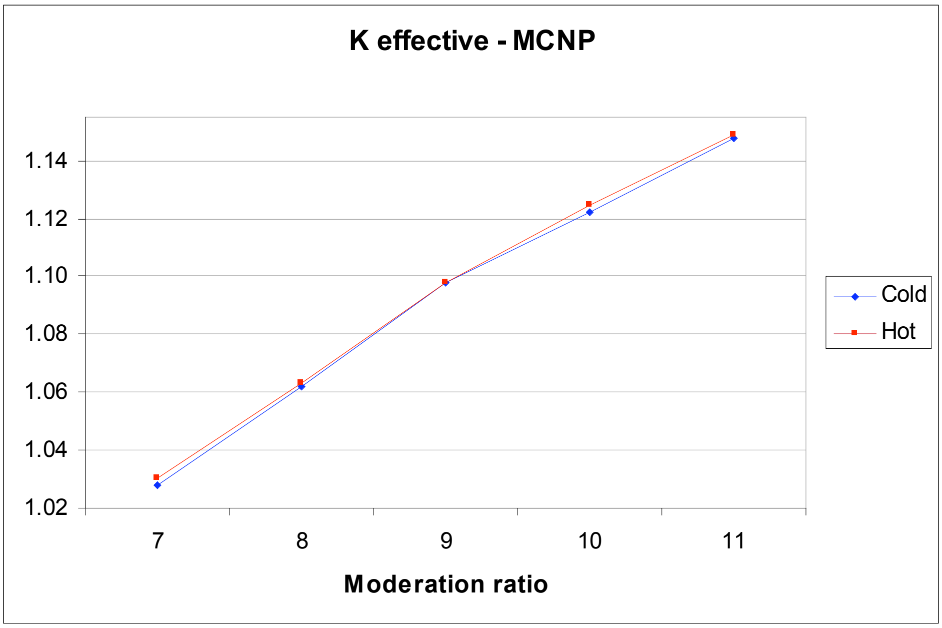

Fig. 3.5

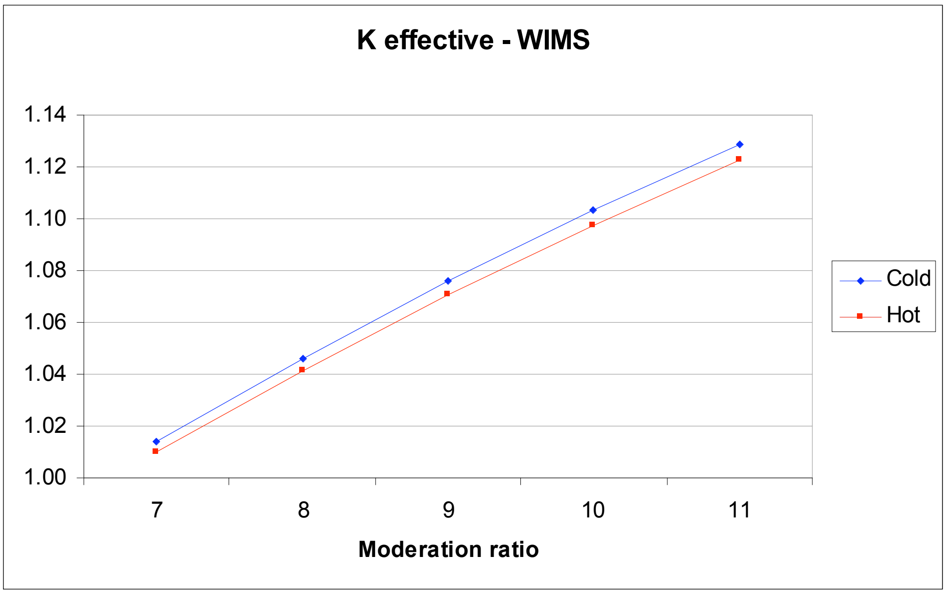
K effective - WIMS
1.14
1.12
1.10
1.08
1.06
1.04
1.02
1.00
7
8
9
10
11
Moderation ratio
Cold
Hot

Fig. 3.6

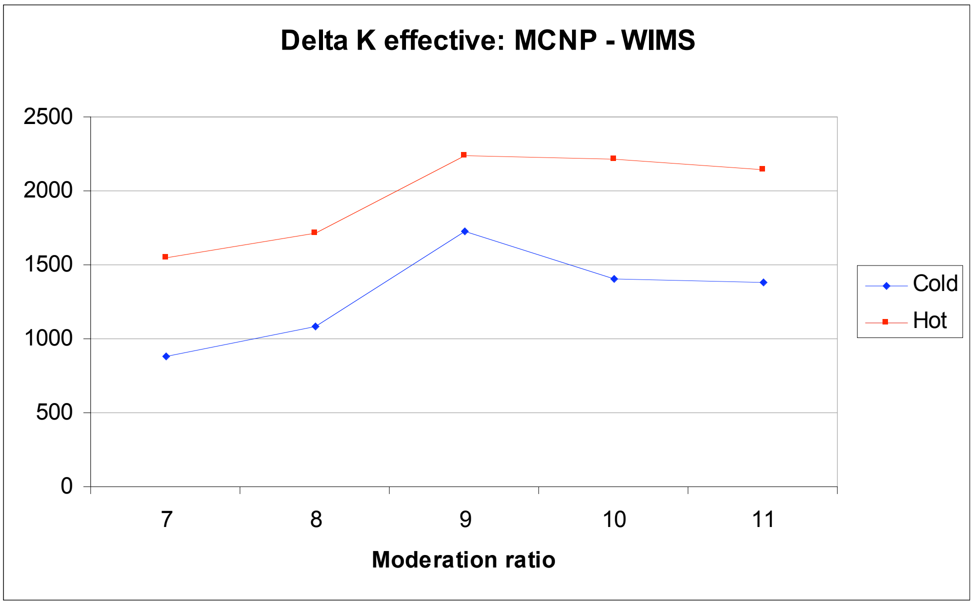
Delta K effective: MCNP - WIMS
2500
2000
1500
1000
500
0
7
8
9
10
11
Moderation ratio
Cold
Hot

Fig. 3.7

From these results the following comments can be drawn:

- k_∞ : the difference between WIMS and MCNP-4C results varies in cold condition from - 4330 pcm at V_m/V_u = 1 to +1172 pcm at V_m/V_u = 20, and in hot conditions from –4791 pcm at V_m/V_u = 1 to +505 pcm at V_m/V_u = 20. Limiting the comparison to the 7÷10 moderation ratio interval, which is the most interesting for this application, the k_∞ difference remains within –1743 and –253 pcm for both cold and hot conditions. The conclusion may be that WIMS shows a reasonable agreement with the *exact* Monte Carlo program, in spite of being applied outside of its range of validity both in terms of enrichment and compact size;
- k_{eff} : as already said, it is calculated for both powers with 100 kg of UO_2 fuel and then the core k_{eff} at BOL does not depend on power but only on the moderation ratio. In the interval 7÷11, WIMS under predicts k_{eff} by 1500÷2200 pcm in hot conditions, and by 900 to 1700 pcm in cold conditions. These are practically the same as those in k_∞.
- k_{eff} and k_∞ variations with temperature are very limited, because the density variation of graphite is much lower than that of water.

In conclusion, in these particular conditions the WIMS program can be judged sufficiently reliable for the goals of this feasibility study, provided that the following rounded margins are assumed, both for the 417 and the 2219 KW cores:

- + 2000 pcm: to take into account k_{eff} underestimation;
- - 3000 pcm: safety margin to take into account burnup effects, non foreseen absorbing materials, instrumentation, etc.;

Total margin: - 1000 pcm.
This means that the WIMS will be made converge at EOL to a k_{eff} of about 1.01 instead of 1.00.

3.2.3 Core design

The considerations presented in sec. 3.2.2 confirmed the need to use masses higher than the minimum ones for the 417 KW core. Thus by adopting 100 kg of UO_2 in both cases the fuel burnup turns out to be 100 MWd/kgU for 2219 KW core and 20 MWd/kgU for 417 KW core.
As already mentioned in par.3.1, for the time being a 5 cm graphite reflector has been chosen.
The value of the moderation ratio has been determined by imposing that at EOL k_{eff} value converges around the above number of 1.01. Figs.3.8 and 3.9 show that, by choosing the moderation ratio of 9.5 for the 2219 and 7.5 for the 417 KW reactor, this specification is satisfied.

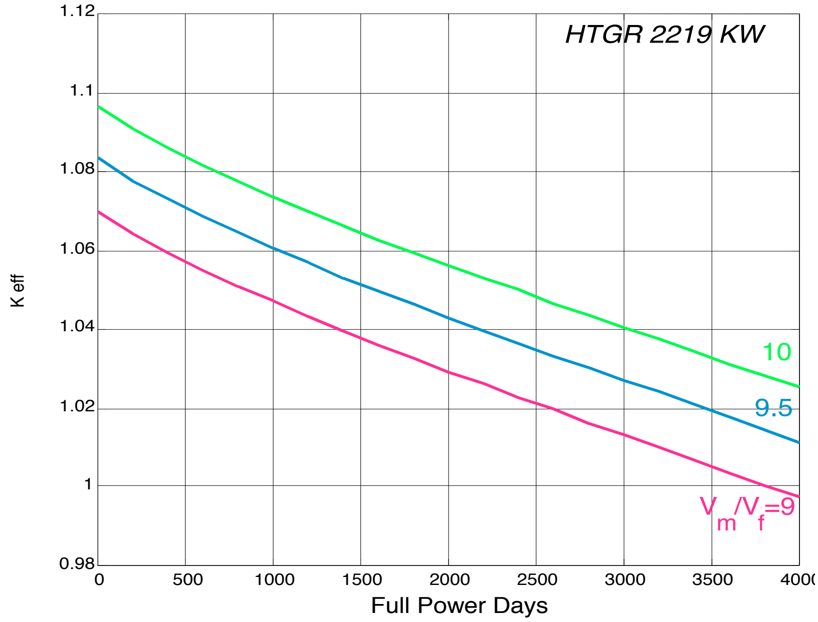
HTGR 2219 KW
K eff
Full Power Days
10
9.5
V_m/V_f=9

Fig. 3.8

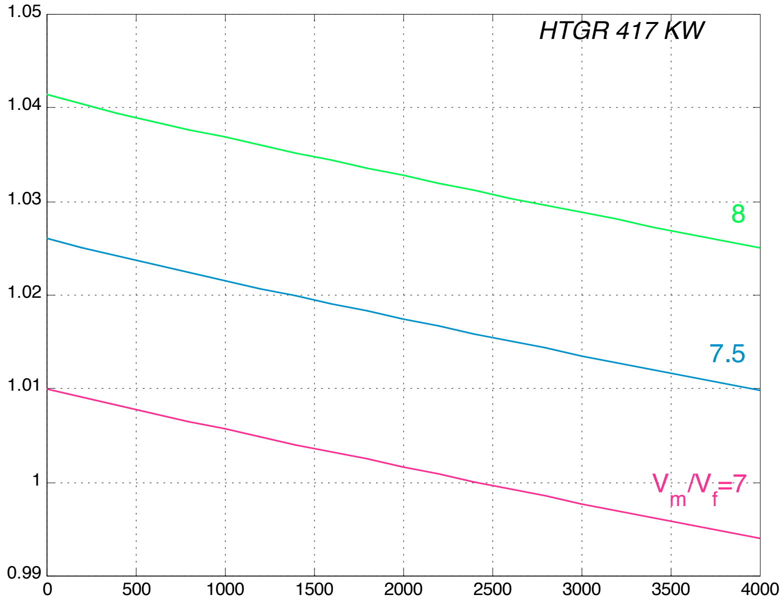
HTGR 417 KW
8
7.5
V_m/V_f=7

Fig. 3.9

Figs.3.10 and 3.11 show the final fuel channel disposition in the 2219 KW and 417 KW reactor, respectively.

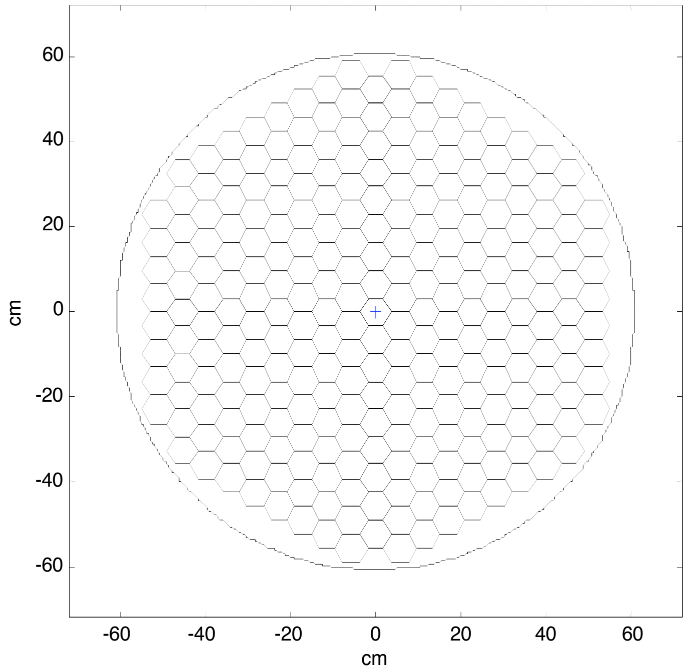

Fig. 3.10

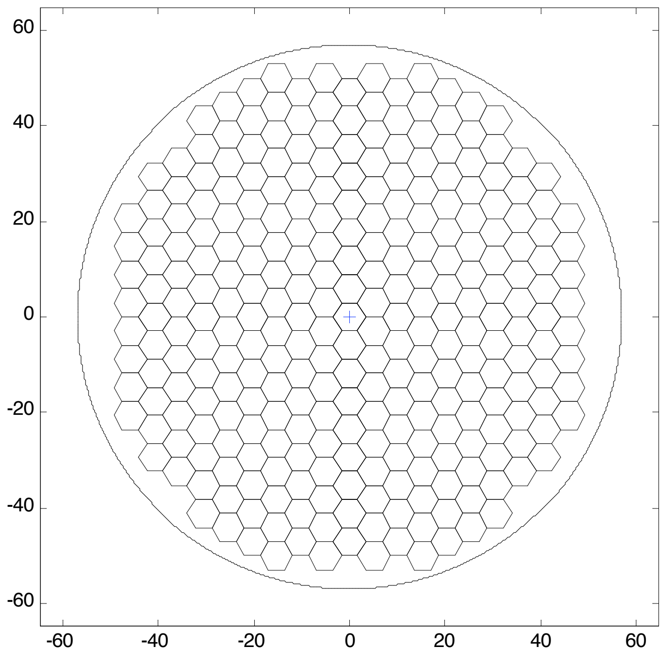

Fig. 3.11

The resulting core data are detailed in par. 3.11. It is interesting to note that the core size is not much different for the two required powers: the fuel content is the same and the difference is due to a higher moderator content in the 2219 KW core. The overall mass (core+reflector) of the 2219 KW reactor is 2588 kg, while for the 417 KW reactor is 2148 kg. The difference is rather small considering that the ratio of the two powers is 5.3.

3.3 Electrical power generation by the Brayton gas cycle

In this reactor cooled by helium and working at high temperature is straightforward the choice of the thermoelectric cycle or the Brayton gas cycle. The first one will be described in the par. 3.4 concerning the thermoelectric device, while the Brayton cycle will be described here below.

The maximum cycle temperature at the reactor outlet, as said in the introduction, is 900 °C. The first alternative is the one to use or not a regenerator between the turbine outlet and the reactor inlet. Without regenerator the efficiency results very low, around 11 %, but on the other side the cold well inlet temperature is very high equal to about 615 °C, and even if its outlet temperature is brought down to 135 °C, the cold well specific surface (m^2/KW) results very small indeed. With regenerator, the efficiency goes up substantially, around 25 %, while the cold well specific surface increases substantially because of the much lower inlet temperature of 266°C. Moreover the regenerator is a demanding component, which exchanges a power about 2.75 times that of the reactor. However, the advantage of obtaining a much higher efficiency (more than a factor 2), also in terms of power to be dissipated in the cold well, and the relatively low cold well surface with respect to PWR, for the higher efficiency, push in the direction of the solution with regenerator.

Thus a simplified but sufficiently reliable calculation program for the Brayton cycle has been implemented and applied to the following input data:

outlet reactor temperature: 900 °C;
minimum cold well temperature: 135 °C;
turbine efficiency: 0.80;
compressor efficiency: 0.80;
regenerator efficiency: 0.95;
minimum pressure: 3 MPa;
pressure drops over fluid pressure ($\Delta p/p$) in reactor, cold well, regenerator (two sides): 0.01

The cycle data are detailed in Fig.3.12 e 3.13. The gross efficiency is 25.36 %, which, for an auxiliary systems absorption of about 5 KW, gives a net value of 24.1 % rounded to 24 %. Thus the thermal power of the reactor is 417 KW.

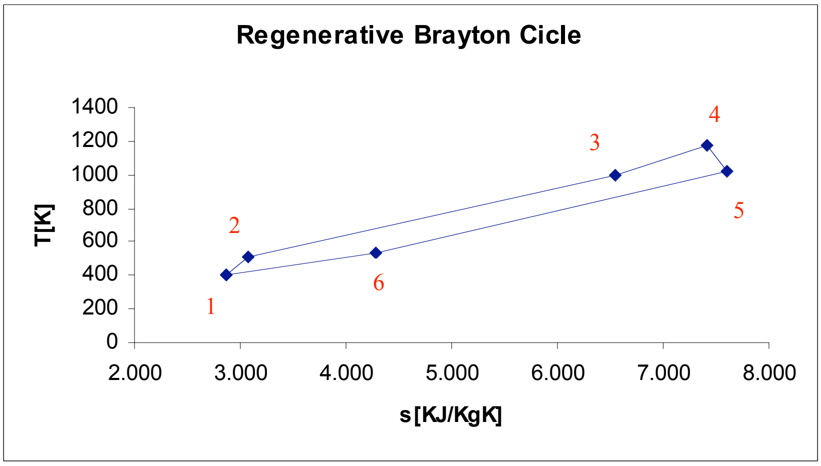

Fig.3.12

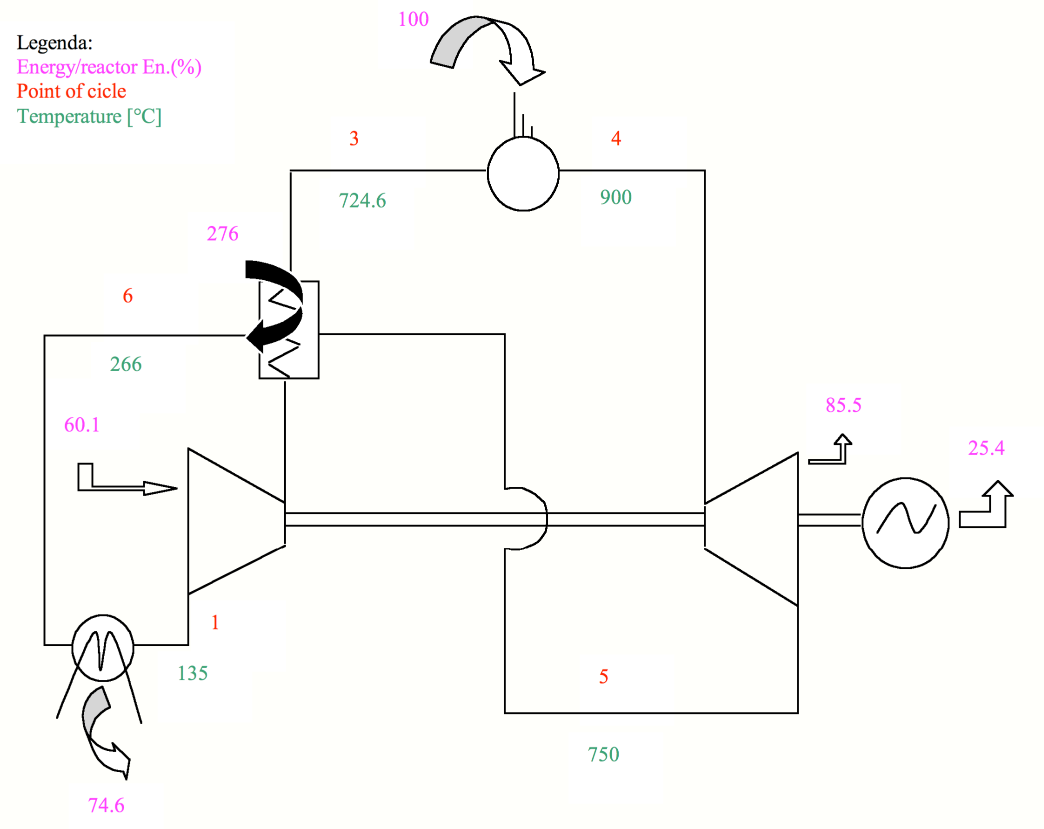

Fig. 3.13

Two data deserve some consideration: the reactor inlet temperature of 725 °C, and the cold well temperature interval of 135-266 °C. The first is a relatively high value, which might involve some technological issues, because in these reactors, characterized by very high outlet temperatures, the structural reactor components are put in contact with the minimum temperatures and not with the maximum ones. Thus the minimum temperature should be reasonably low, and probably the value of 725 °C does not satisfy this requirement and therefore some specific design choice and protection of the involved walls are to be implemented.

The 135-266 °C interval for the cold well temperatures is not an optimized value as in the case of PWR, but it has been assumed as a sound judgment on the basis of the consideration that, when the efficiency is higher also the cold well temperature should go in the same direction. When the radiation is the controlling mechanism, the average temperature cannot be the arithmetic one, but a suitably weighted average, which gives the same specific surface. So doing the average value turns out to be close to 190 °C, value which is reasonably higher that of 165 °C already assumed in PWR.

The cold well with the above efficiency must dissipate (417- 100) = 317 KW. By integrating the radiation equation already mentioned in par.2.6, by considering a back radiation from the surrounding environment at 300 K, a view factor of 0.6 and an emissivity of 0.90, the specific powers varies from 2.33 and 0.6 KW/m^2 at the inlet to the outlet conditions respectively. The average weighted value is equal to 1.16 KW/m^2 , which corresponds to a surface of 273 m^2.

By assuming 450 tubes of 6 mm ID and 0.5 mm thickness (t/ID is slightly higher than in PWR because the pressure is 3 against 0.7 MPa and the maximum temperature is 266 against 165 °C) a tube length of 28 m and a pressure drop of 30 KPa, corresponding to a Δp/p of 0.01 are obtained (note that this agrees with the *a priori* assumed value of 0.01 used to calculate the cycle efficiency). Adopting also in this case the titanium material, the overall tube weight is 571 kg, which, increased by 30 % for auxiliary components, gives an overall cold well weight of 742 kg (2.34 kg/KW or 2.72 kg/m^2).

The 28 m long tubes are supposed to form 4 contiguous U, 7 m high, for a total of 1800 legs. Assuming that the tube legs are welded together , an overall linear dimension of 1800x0.007 = 12.6 m is obtained. In conclusion, the condenser can be imagined to be a cylinder of 4 m diameter and 7 m height. The final design would take into account the real pay load size of the launching rocket. In particular, if necessary the cold well may be sectioned in several identical parts and assembled on the site. All the data are detailed in the par. 3.11.

3.4 Electrical power generation by the thermoelectric device

Thermoelectric (TE) materials are solid-state devices with no moving parts, their heating ability combined with their highly reliable, silent, and vibration-free operational mode, lack of compressed gases, chemicals, or other consumables; and complete scalability makes them attractive for electrical energy generation. However, because of their relatively high cost and low efficiency, TE devices up to now have been restricted to applications for which high reliability, portability, or small size are important such as automotive seat coolers and generators in satellites and space probes.

Thermoelectric converters have considerable space flight experience. The radioisotope thermoelectric generators (RTGs) which powered the Voyager spacecraft comprised thermoelectric converters in conjunction with a radioisotope heat source, as the Cassini Mission to Saturn and many other robotic missions. Radioactive isotopes serve as heat source to enable the use of radioisotope TE generators, for extended periods. In fact to send a satellite away from the sun, to the outer planets it is almost impossible to use photovoltaic cells – there is not enough solar energy at those distances to generate useful power. The Cassini probe and the quarter century old Voyagers contains radioisotope thermoelectric generators in which a mass of plutonium-238 (serving as heat source) is coupled to a thermoelectric material to produce electricity without relying on solar panels and is still in operation.

The SNAP-10A 0.5kWe space reactor and the Russian ROMASHKA type reactor system both used thermoelectric converters. They have proved reliable for continuous operation up to 10 years. For the SP-100 reference design a system efficiency of 4% was conservatively assumed .

The dimensionless thermoelectric figure of merit, *ZT*, determines the dependence of device efficiency upon material properties, and is defined as follows,

$$ZT = TS^2 \sigma / \kappa$$

where *T* is the absolute temperature, *S* the Seebeck coefficient, σ and κ the electrical and thermal conductivity, respectively, and the latter is the sum of the electronic κ_e and lattice κ_L components at the temperature *T*. A good thermoelectric material must have a large *S*, to produce the required voltage, high σ, to reduce the thermal noise (joule heating, I^2R), and a low κ, to decrease thermal losses from the thermocouple junctions. A low thermal conductivity in a good thermoelectric material means a low value of its lattice component κ_L as a major contributor, since its electronic component κ_e ,as a minor contributor, is proportional to the electrical conductivity σ.

These physical properties are better satisfied by semiconductors. However, the electrical properties of these materials can change dramatically with temperature. As a result, semiconductors can only function as thermoelectric materials over certain temperature ranges, which will vary for each semiconductor. Fig.3.14 shows the effectiveness of most commonly used semiconductor materials for thermoelectric devices, as measured by the figure of merit (*ZT*).

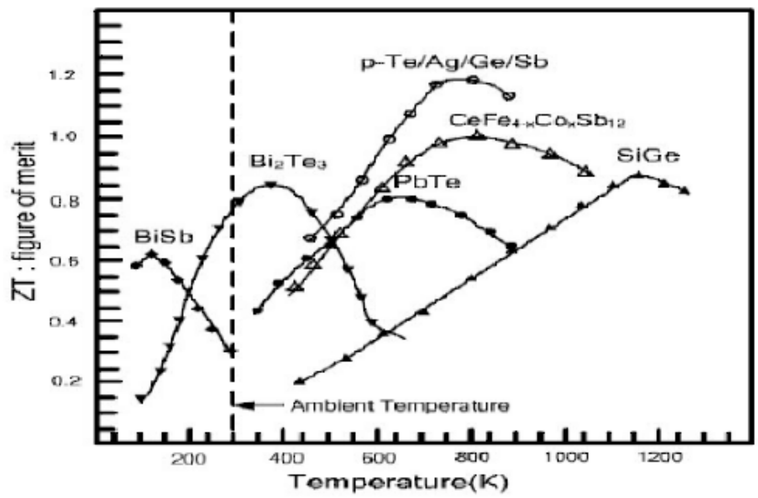

Fig. 3.14

Higher *ZT* yield better thermoelectric performance. Known thermoelectric materials fall into three categories depending upon their temperature range of operation. Bismuth Telluride (Bi_2Te_3) and its alloys have the highest *ZT*, and are extensively employed in *terrestrial cooling* applications. The most commonly used semiconductor material for cooling applications, Bismuth Telluride system, has a maximum performance at approximately 80 ^oC with an effective operating range (EOR) of -100 ^oC to +200 °C. Bi-Sb alloys are useful only at low temperatures. Lead Telluride (PbTe), the next most commonly used material, is typically used for power generation, but is not as efficient as Bi_2Te_3 in cooling applications. PbTe reaches a peak *ZT* at 350 ^oC and has an EOR of 200 to 500 ^oC. PbTe is typically used for power generation because its higher operating temperatures yields more efficient power generation when the heat is rejected at ambient temperatures. TAGS are the alloys $(TeGe)_{1-x}(AgSbTe)_x$, where x ~0.2, and has an EOR of 400-600 ^oC. Silicon Germanium, SiGe, has an EOR of 800-1000 °C and have been widely used in *thermoelectric generators* for space applications together with TAGS. The Skutterudite, $CeFe_3CoSb_{12}$, has an EOR of 400 to 700 ^oC but is not used in practice since TAGS are superior in the same temperature range.

Over the temperature range 100 – 1000°C, two different couples are the best for the space reactor application :

- PbTe is the best (maximum factor of merit) for medium temperature below 700°C
- SiGe is adapted to higher temperature (700 – 1000°C).

The efficiency of the entire system is defined:

$$\eta = \eta_{carnot}\,\eta_{thermo}$$

where η_{carnot} is the Carnot (ideal thermodynamic) cycle efficiency and η_{thermo} is the thermoelectric junction efficiency, given by:

$$\eta_{thermo} = \frac{\sqrt{(1+ZT)}-1}{\sqrt{(1+ZT)}+\dfrac{T_{cold}}{T_{hot}}}.$$

This gives the total thermoelectric power conversion system efficiency as:

$$\eta = \frac{\Delta T}{T_{hot}} \times \frac{\sqrt{(1+ZT)}-1}{\sqrt{(1+ZT)}+\dfrac{T_{cold}}{T_{hot}}}$$

Where: ΔT = temperature difference between hot and cold junctions

T_{hot} = temperature of hot junction (K)

T_{cold} = temperature of cold junction (K)

The hot well temperature is fixed from the reactor, the average of the inlet and outlet reactor temperatures is T_{hot} = 1085 K.
Fig.3.15 shows the efficiency vs. the cold well temperature. As clear in figure the optimum is reached as the cold well temperature is as low as possible.

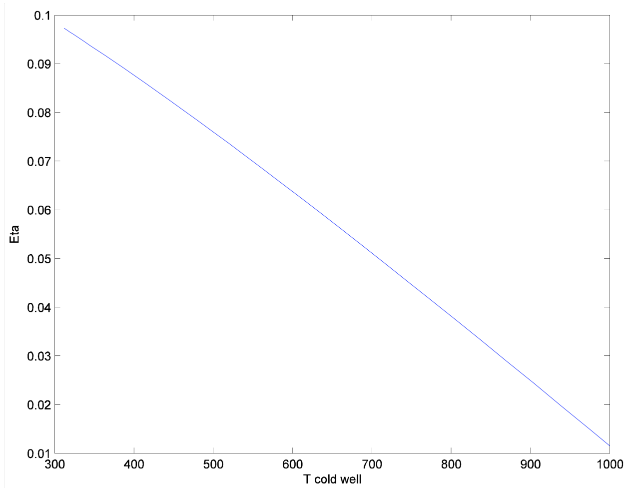

Fig. 3.15

As the temperature of the cold well decreases, the efficiency increases and the thermal power of the space reactor decreases.
At the same time as the cold well temperature decreases, the area of the radiators increases, in fact this area is proportional to the ratio of the power to be wasted to the fourth power of the cold well temperature (Fig.3.16):

$$P = A(\varepsilon\sigma T_{cold}^4 - P_{sun})$$

where P_{sun} is the specific power irradiated by the Sun on Mars, approximately equal to 0.25 KW/m^2.

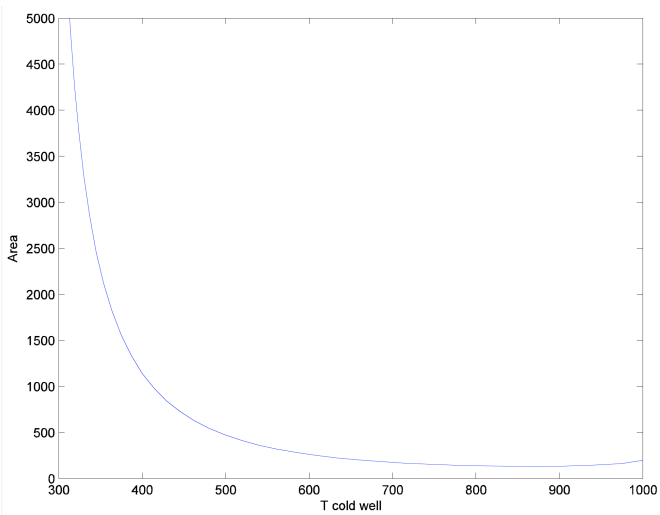

Fig. 3.16

As the efficiency is very high, the area of the radiator is large and its mass is considerable; on the other side if the efficiency is low the required thermal power is high and so the mass of the core increases. All these elements have to be considered doing a feasibility study. An optimization process has been developed in order to maximize the efficiency, minimizing the area of the radiator and the thermal power. This optimization has been developed finding a minimum of a constrained nonlinear multivariable function F = f(x) chosen as the ratio between the area of the radiator A_{rad} and the overall efficiency η; this function depends on the cold well and the hot well temperatures, T_{cold} and T_{hot}, respectively.
At the end only boundary constraints have been set up:

- limitation of the highest and lowest temperature for the hot well
- limitation of the highest and lowest temperature for the cold well.

The function F is shown in Fig. 3.17.

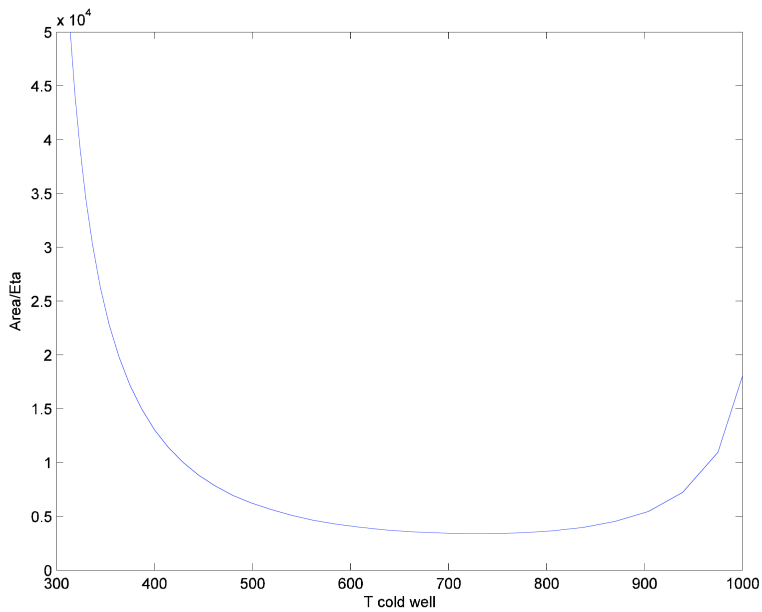

Fig. 3.17

The results of this optimization procedure are:

T_{hot} = 1085 K

T_{cold} = 729 K

ZT = 0.6442

η = 0.0473

A_{rad} = 160 m^2

The net efficiency, calculated in order to take into account the system power requirements with the same hypotheses adopted for the PWR is :

η_{net} = 0.0473·100/105 = 0.045.

However, the system power requirement might be significant, because of the relatively high pressure drops in the circuit, which must be carefully optimized.

The thermoelectric conversion unit obtained should have a mass of no more than 100 kg.

In par. 2.1, it has been said that the thermoelectric device are not convenient for the PWR. Actually, in this case the hot temperature would be about 340°C. This value, combined with the fact that the cold well temperature cannot be too low in order to irradiate the thermal power, yields always very low thermoelectric conversion efficiency. As the same optimization is done for the PWR reactor (using PbTe, the best available material in the temperature range considered) the results obtained are:

T_{hot} = 587 K
T_{cold} = 438 K
ZT = 0.4213
η = 0.0251
A_{rad} = 2725 m^2

As clear with such a low gross efficiency and such a huge area, the thermoelectric device, by now, is not suitable to the PWR.

As for the PWR already described, the cold well is one of the most crucial component of any energy conversion system for space application.

In this preliminary study of the thermoelectric option, a heat pipe solution has been adopted because heat pipes have an effective thermal conductivity many thousands of times that of copper. A heat pipe is a simple device that can quickly transfer heat from one point to another. Up to now they have been adopted for several space applications. A heat pipe is a hermetically sealed evacuated tube normally containing a mesh or sintered powder wick and a working fluid in both the liquid and vapour phase. When one end of the tube is heated, the liquid turns to vapour absorbing the latent heat of vaporization. The hot vapour flows to the colder end of the tube, where it condenses and gives out the latent heat. The condensed liquid then flows back through the wick to the hot end of the tube. The chosen heat pipe consists of a sealed aluminium container, a working fluid compatible with the container, Freon and a porous structure in aluminium. A feature of the heat pipe is that it can be designed to transport heat between the heat source and the heat sink with very small temperature drops and the second one is that relatively large amounts of heat can be transported by small lightweight structures.

The dimensioning has been realized considering some hypothesis: the sun irradiation is present, each thermoelectric module produces 10 W, the view factor of each heat pipe is 0.5.

The operating temperature is 729 K, resulting from the optimization previously realized for the entire system. As each heat pipe is mounted on 1 thermoelectric cell 10000 heat pipes are considered, the diameter of the heat pipe is 7,5 cm in order to fit with the dimension of the thermoelectric cell.

The power to dissipate is proportional to the fourth power of the cold well temperature:

$$P = A(\varepsilon\sigma T_{cold}^4 - P_{sun})\ ;$$

the real area to consider is obtained dividing the theoretical area for the view factor:

$A_{real} = 2 \cdot A_{theorical}$

The ratio between the total area of the radiators and the total area of the thermoelectric device should be maintained on each heat pipe so the length, L of each heat pipe is easily obtained:

$\xi = A_{real} / A_{thermoelectric\ cell}$

$A_{heat\ pipe} = \pi \cdot D \cdot L = \xi \cdot A_{thermoelectric\ cell}$

The thickness of the container has been assumed t=0.75 mm, because of structural reasons. The radiator obtained in order to dissipate 2119 KW (2219 -100 KW) is composed of 10000 heatpipes, of a theoretical area of 159 m^2 and of a real area of 318 m^2, 135 mm long and of a total mass of 642 kg.

3.5 The primary system

The primary system differs substantially between the two reactors. However, both adopt a semi integrated solution, where the rotating machines are put inside the pressure vessel. Then in the 417 KW reactor the turbine, the compressor and the alternator are integrated inside the pressure vessel, while in the 2219 KW reactor the compressor and its own motor (see Fig.3.18 and 3.19).

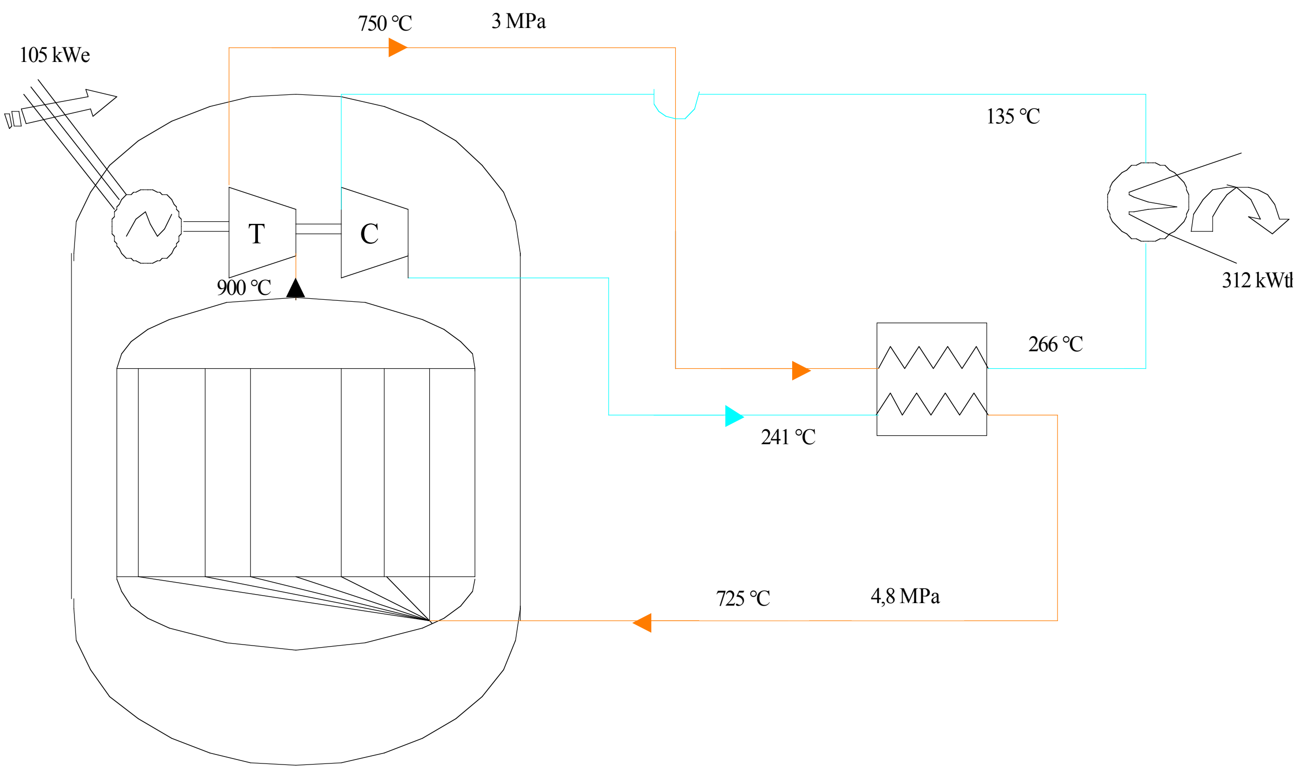

Fig. 3.18

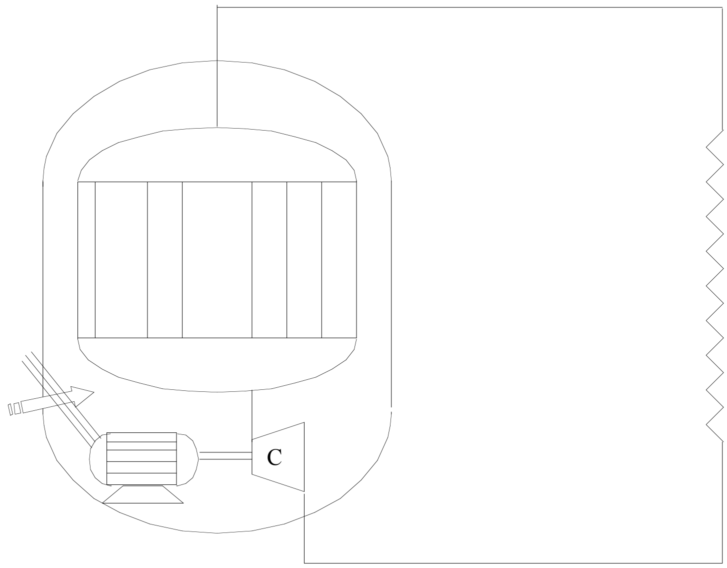

Fig. 3.19

This choice is based on the following considerations:

- to recuperate the small gas leakage from the turbomachines without the need of an ad hoc containment and recharging compressor;
- to cool the pressure vessel walls by a relatively cold gas, otherwise the mechanical constrain would result excessive, unless an ad hoc insulation is implemented to cover all over the inner walls; this cooling can be done rather simply in the 417 KW reactor by making licking up the cold gas on the inner pressure vessel wall; while in the 2219 KW one, an auxiliary circuit is necessary to produce this small flow rate of rather cold gas following the same inner path.
- The 900 °C pipe is eliminated outside the vessel in the 417 KW reactor but not in the 2219 KW one.
- The vessel mass is practically the same

The issues to be coped with by this choice are:

- the inlet and outlet nozzles are passed through by hot gas, (725 - 750 °C) and (725 - 900 °C) for the 417 KW and 2219 KW reactor respectively and thus a reliable insulation solution for the wall must be implemented;
- the same for the connecting pipes to the regenerator (417 KW) and the thermoelectric device (2219 KW);
- the rotating machines work in high temperature environment and cannot be inspected and maintained (in any case very difficult in space reactors);
- the above requirement seems more worrying for the alternator (417 KW) and the motor (2219 KW), which usually operate at room temperature and copper wires can be sensitive to temperature.

In this case also the best pressure vessel shape is the cylinder surmounted by hemispheric domes: the inner dimensions are sketched in Figs.3.20 and 3.21 for the 2219 KW reactor and in Figs.3.22 and 3.23 for the 417 KW reactor.

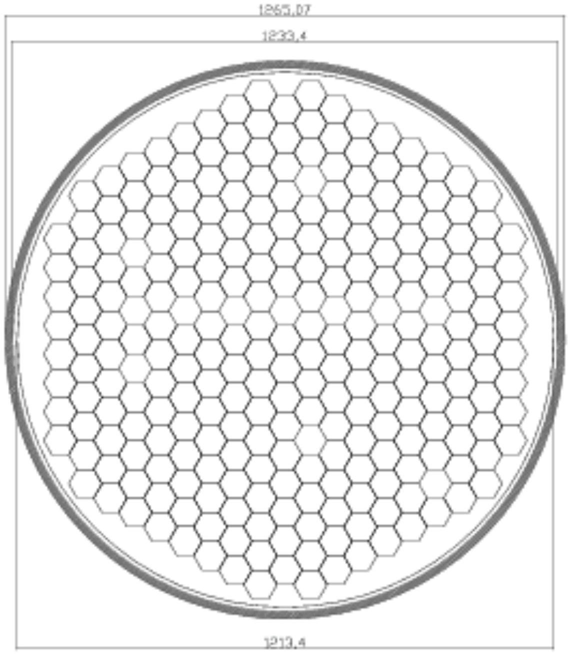

Fig. 3.20

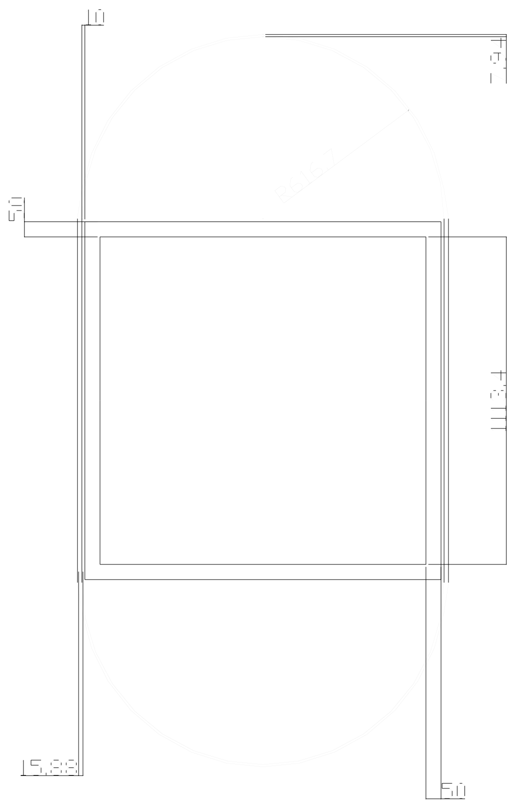

Fig. 3.21

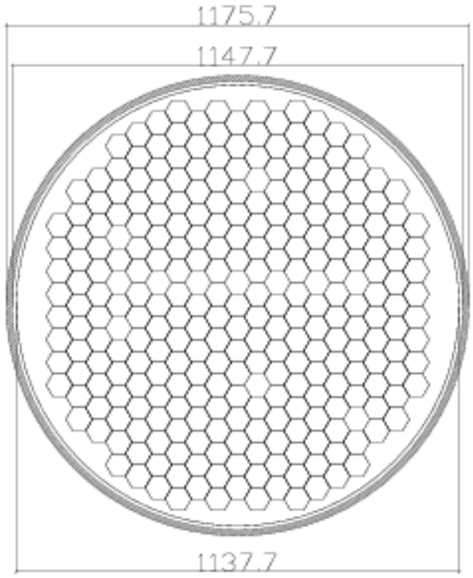

Fig. 3.22

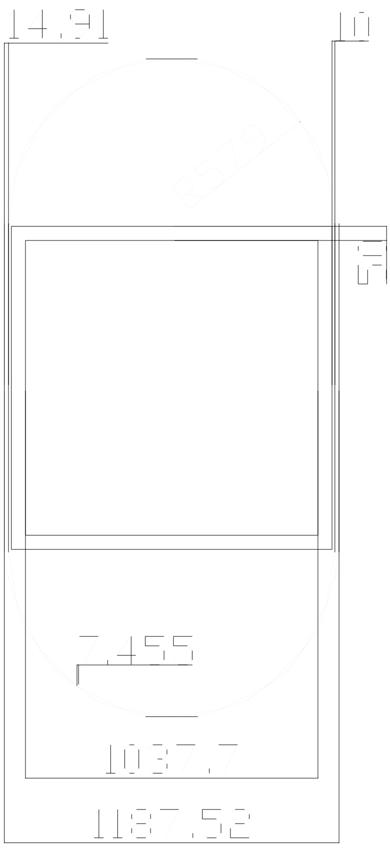

Fig. 3.23

3.5.1 The pressure vessel

The design pressure of the primary system is equal to the operating pressure multiplied a factor of 1.10, which takes into account the value of the safety valve setting. This component is absolutely needed for each pressurized system, but in our case implies an important issue concerning the consequences of its operation in the case of pressure transients above 1.10 operating pressure, which means a discharge of a substantial fraction of helium. For the time being we adopted a higher value than the usual one in order to reduce the possibility of its intervention (1.10 against 1.07). The design pressure has been then set to 4.8x1.10 = 5.28 MPa.
On this basis the pressure vessel thickness have been determined using the same steel adopted in PWR, i.e. stainless steel SA 508, Tp.3, Cl.2, with an allowable stress of 205 MPa[7]. The thickness are detailed in par. 3.11 and the vessel mass turns out to be 890 kg and 736 kg for the 2219 and 417 KW reactors respectively.

3.5.2 The regenerator

The regenerator is a crucial component both for the size, having a power of 2.76 times that of the reactor (1151 against 417 KW), and the high temperatures and pressures involved (maximum values 750°C and 4.8 MPa). A detailed design of this component has not been carried out yet. From the relevant literature we found that a plate fin exchanger with 6 fins/cm provides about 1300 m^2 per cubic meter of volume. Operated at a frontal velocity of 3 m/s, the heat transfer coefficient based on prime surface area would be around 1.8 KW/m^2 K. The average temperature drop is 25 °C. Therefore the overall surface is 26 m^2. Recalling the above surface over volume ratio, the overall volume results 0.02 m^3. In other words that means a cube of 27 cm side.
The pressure drops are to be verified once defined the exact geometry of the heat exchanger, however, the assumed value of 0.01 of 4.8 and 3 MPa respectively of the two sides seems to be sufficiently high.
The remaining components are the safety valve, the control mechanism and the instrumentation. The safety valve does not require particular consideration apart from the issue mentioned above about the effects of its operation. The control mechanism will be addressed in the next paragraph. The instrumentation can be considered rather conventional, but how to treat the relative information is an open issue which should be considered in the following.

[7] It might be a little higher because the temperature is lower than in PWR, but this is to be verified in detail.

3.6 The reactivity control

For the reactivity control, the general considerations developed for PWR systems are still valid, apart the solution to use boric acid dilution in the moderator, which obviously is not possible in this reactor, but anyway it has been also excluded for other reasons in the PWR.

The reactivity excursions in these reactors are due to total reactivity difference between a cold condition with the fuel at the Beginning Of Life (BOL) and the hot reactor at full power with the fuel at the End Of Life (EOL). A logical subdivision of this reactivity variation is as follows:

1. Reactivity decrease going from cold condition to operating temperature;
2. Reactivity decrease from zero power to full power conditions (this term includes the absorptions by Xe and Sm, which reach their equilibrium value in tenths hours);
3. Reactivity decrease along the fuel life to cope with reduction of fissile material and accumulation of poisoning fission products;
4. Reactivity margin to control the reactor power.

The typical values for a commercial HTGRs are not well known, because their design is still in progress. The values found in the present design are as follows:

Tab. 3.3 : Reactivity terms in pcm.

Power (KW)	417	2219
Term 1	~ 500	~ 500
Term 2	~ 300	~ 500
Term 3	1700	7200
Term 4	$\sim 0^+$	$\sim 0^+$
Total	2500	8200

Term 1 is small because the density variation in solid graphite between cold and operating temperature is much lower than in water; Term 2 is relatively small because of the low temperature increase inside the rod and the low thermal flux both the Doppler effect and the absorptions by Xe and Sm are of modest impact. Term 3 is that shown in Figs. 3.8 and 3.9.

Like in the PWR solution, the only possibilities are burnable poisons and control rods. The first ones require as already said in par.2.5 a detailed study, which is now postponed and its goal is to reduce the overall reactivity to be controlled by the control rods. The latter can be inserted in the reflector as already foreseen in some high power HTGRs. These *rods* can be imagined as rotating devices, as already explained for the PWR solution, or channels flowed by a *fluid* made by poisoned graphite balls inserted or extracted from the core reflector, by means of a suitable pneumatic mechanism.

The problem of reactivity control seems more viable than in the PWR reactor, however the lack of a negative temperature coefficient may render the system control more delicate, implying in probably a continuous operation of the control rods.

3.7 Masses

In the previous sections the masses of the core, pressure vessel and cold well has been detailed for both the 2219 and the 417 KW cores. We have not considered some further components of the complete system yet, as for example the tubes, valves, auxiliary circuits, internals and so on. The mass of the other not considered parts is globally assumed to be about 200 kg for both reactors.
At the end the mass of any single component has been increased of a margin of 5 % to take into account any kind uncertainty or disregarded parts. The final sum has then been increased by a further 10 % as a contingency. The detailed values are shown in Tab3.4.

Tab. 3.4 : Masses in kg of HTGR

	2219 KW				417 KW			
Fuel	268 +				268 +			
Moderator	1746 +				1378 +			
Reflector	574				502			
Core+reflector	= 2588 +				= 2148 +			
Vessel	890 +				736 +			
Overall reactor		= 3478	+ 5 %	3652		= 2884	+ 5 %	3028
Cold well		642	+ 5 %	674		742	+ 5 %	779
Compressor		100	+ 5 %	105		100	+ 5 %	105
Turbine / Thermoelectric devices		100	+ 5 %	105		100	+ 5 %	105
Other components		200	+ 5 %	210		200	+ 5 %	210
Overall system				4746				4227
Contingency = 10 %				475				423
Total Mass of the System				**5221**				**4650**

3.8 Preliminary safety consideration

A detailed safety analysis is outside the scope of this feasibility study, for its complexity and need to define the detailed requirements. In fact this reactor from one side is nor subjected to the licensing procedure of terrestrial reactors imposed by the safety authorities, from the other it must satisfy specific safety issues connected to its launch and the possibility to fall down to the earth.

In the foreword it is mentioned that this nuclear system must satisfy the usual safety requirements of terrestrial reactors and this is what is to be defined in detail, taking into account the above consideration about the lack of a licensing procedure. Besides this the system has to assure that:
- no irradiated fuel is present at launch;
- the core sub criticality in the case of possible launch accidents (flooding);
- the radiation protection without impairing mass requirements;
- an easy decommissioning in space;

The first item is inherently satisfied, because the reactor would not reach its first criticality before being outside terrestrial space. The second one is a rather crucial one, because it requires the need to insert high absorbing materials in the core, to be extracted when the reactor will start up. Probably this is a rather demanding requirement, which deserves a specific consideration.

The third is a an important issue, which can be addressed only after having defined some conditions, especially for the propulsion solution. In fact, for surface reactor, the shield cannot be transported from the earth and it is to be provided by a suitable system layout on the Mars surface (regoliths around, underground siting, big distances). For propulsion reactor an intermediate solution is to be found by balancing the addition of a small shield around the core with the reduction of the radiation danger by locating the reactor far away from the sensible zone (a separate capsule for the reactor?) or by locate an extra shield only on the reactor portion *viewed* by the sensible zone.

The fourth one is too indefinite at this stage of the design, that no specific consideration can be drawn.

These reactors have the inherent feature to resist to the consequences of a LOCA, without provoking the fuel melting. It is well known that fuel melting represents the most feared situation for terrestrial reactor, and the nowadays attitude for new reactor designs is to avoid this in any foreseeable circumstance, as it happens in these reactors by adopting a relatively high value of the ratio lateral surface over volume of the core. In fact, this choice limits the power of these reactors to 200÷300 MWth range. In a complete loss of coolant, the fuel and the core heat up and the reactivity goes down till the complete shut down of the neutronic chain, but the fuel goes on producing some power, the so called *decay heat*. Actually this transient is to be verified in this reactor because of its very low temperature coefficient. If this is the case, the fuel temperature rise is curbed by the thermal radiation of the core toward the vessel and from the latter towards the nearby environment. Since the fuel resists to very high temperature without undergoing damages or ruptures, there is no need of any protection system intervention.

However, in the case of a LOCA in a terrestrial reactor it is necessary to avoid the air entrance into the core, otherwise the graphite reaching high temperature reacts with the oxygen of the air and burns. However, in the literature it is mentioned the fact that high purity nuclear grade graphite reacts very slowly with oxygen and it will be classified as non combustible by conventional standard. This potential danger in terrestrial reactors is coped with different protection actions and in particular a containment, the goal of which is to avoid the air entrance into the core. In the present reactor, the danger is not well defined, because in propulsion application there is no atmosphere and in surface application the exact composition of the atmosphere is not well known. Anyway this issue deserves attention.

3.9 Open issues and R&D needs

This feasibility study has allowed us to find the open issues to be solved for going on this route, which needs a R&D program. Here below a list of them, which is inevitably incomplete, is detailed; the order is not based on their importance. The fuel is identical to that foreseen in terrestrial reactors, and then it can be assumed that it is or will be developed by already existing R&D programs.
The issues written in **bold letters** are those interesting for terrestrial reactors and those written in *italic* those interesting for generic terrestrial applications as well.

- Reactor vessel internal layout: temperature distribution, **wall cooling, internal passages, mechanical design**;
- **Pipe design to resist to high temperature flowing fluids**;
- **Increase of operating pressure: primary circuit materials;**
- The cold well as cooler
- The cold well associated to thermoelectric device;
- *Heat pipes*
- *Gas turbine and compressor working in high temperature environment;*
- *Alternator working in high temperature and pressure environment;*
- *Thermoelectric apparatus;*
- **Fluid leakage: how much, how to cope with;**
- **Maintenance requirements of the whole system;**
- **Optimum reflector: technological aspects**
- Fluence effects on vessel in these particular conditions;
- Shielding;
- Safety valves: reliability, how to cope their intervention;
- *Vessel material different from stainless steel*;
- **The regenerator: thermal, mechanical corrosion issues;**
- **Control rods;**
- Control of the system and of the reactor and its constructive implications;
- Flooding danger avoidance.

The R&D about thermoelectricity at high temperature is of paramount importance for this reactor, because if present net efficiency (gas pressure drops may require high compressor power absorption) can be improved and assure the long term reliability by a suitable choice of materials, a thermoelectric apparatus might became the right solution for this nuclear system, instead of the much more complex Brayton cycle. Moreover, if the improvements may be obtained also at lower temperatures as those typical of PWRs, the present choice to eliminate this option for these reactors should be reconsidered.
Even if this list is incomplete, no item seems to be unsolvable, and the lack of fuel development activity is greatly advantageous. On the other hand, some high temperature design issues appear demanding especially for long term operation. An R&D program of reasonable extent may yield the needed answers,

but what is important that the most demanding researches also are of interest for the new generation High Temperature Gas Reactors. Then a cost sharing action can be proposed and duly programmed, according to the time schedule of the commercial exploitations of these terrestrial reactors.

3.10 Concluding remarks

At the end of this very preliminary feasibility study about the use of HTGR system for space reactors, it can be concluded that no insoluble issues have been evidenced, which would prevent of going on along this route in order to execute a more detailed design. Then it will be possible to draw a more justified conclusion about the usefulness to follow this solution.

At the beginning of the study it was supposed that the solutions for propulsion and surface application might be the same. Actually, it seems that this hypothesis holds more in this reactor than in PWR, because the lack of gravity does not determine any particular detriment to reactor operation. However, it remains the need of an autonomous radiation shield, which in surface reactors can be provided by the existence of a soil. On the other side the safety problem connected to a possible flooding seems rather demanding, also because the fuel cannot be separated from the moderator during the launch phase.

If it will be confirmed in prosecution of the work that no insoluble issues are present in this proposal, it can be stated that a reasonable R&D effort and consequently a relatively limited development cost and time interval are only needed in this case.

In the short range, future design activities should address the detailing of many aspects of the analysis presented in this report and adding new ones. The new activities are: radiation shielding, vessel fluence, control, safety aspects, cold well (in forced convection as well), choice of vessel material, vessel layout, system layout, regenerator design, containment, leakage control, ancillary circuits for start up, coolant purification and other exigencies. Moreover, at the end of this further activity a preliminary R.&D. program should be detailed.

3.10 . HTGR reactor list of data

	HTGR - 2219	HTGR - 417
Power		
Thermal [KW]	2219	417
Electric [KW]	100	
Primary circuit		
Fuel		
Composition	UO_2 sintered in micro spheres	
Enrichment BOL	93%	
Enrichment EOL	~83%	~91%
Diameter [ì m]	350	
Rod geometry	Hexagonal	
Apothem [mm]	4,1	
Cladding		
Material	4 carbon based layers	
Thickness [ì m]	400	
Density [kg/m^3]	1800	
Fuel-moderator-coolant channel		
Geometry	Hexagonal	
Apothem [mm]	32,7	29,4
# of fuel rod / coolant holes	6 / 1	
Holes apothem	3,8	
Height [mm]	1113	1037
Moderation ratio	9,5	7,5
total # of fuel rod	1590	1686
Burnup [MWD/kg]	100	20
Pressure		
P max [MPa]	4,8	
P min [MPa] (assumed)	3	
Compression ratio	1,6	
Temperatures		
T max [°C] (assumed)	900	
T min [°C] (assumed)	725	
Secondary circuit		
Cold well temperature [°C]	456	135 - 266
Thermodinamic cycle	Thermoelectric	Brayton
Net efficiency [%]	4,5	24

Miscellaneous data		
UO_2 Fuel quantity [kg]	100	
Specific power [KW/kg]	22,19	4,17
Reflector		
Material	Graphite	
Thickness [mm]	50	
Core designs		
keff (EOL) (assumed)	1,01	
Mass (core+reflector) [kg]	2588	2148
Electrical power generation		
Working fluid	///	He
Inlet reactor temperature [°C]	725	
Outlet reactor temperature [°C]	900	
Turbine efficiency	///	0,80
Compressor efficiency	0,80	
Regenerator efficiency	///	0,95
Minimum pressure [MPa]	///	3
Pressure drops (deltap/p)	///	0,01
Cold well		
Type	Heat pipes	Bundle of tubes
Tube geometry		
Inner diameter [mm]	73,6	6
Outer diameter [mm]	75	7
Length [m]	0,13	30
Tube number	10000	450
Pressure drop [KPa]	///	30
Internal pressure [MPa]	///	0,7
Pressure drops (deltap/p)	///	0,01
Temperature [°C]	456	135 - 266
Specific power [KW/m^2]	///	1,16
View factor	0,5	0,6
Surface area [m^2]	317	273
Material	Aluminium	Titanium
Mass [kg]	642	742
Mass per unit radiating surface [kg/m^2]	2,03	2,72
Cold well dimension (cylinder)		
height [m]	///	7
diameter[m]	///	4
Reactor vessel		
Design pressure [MPa]	5,28	
Material	SA 508, TP.3, Cl.2	
Thickness [mm]		
Cylindrical portion [mm]	15,9	14,9
Spherical portion [mm]	7,9	7,5

Vessel mass [kg]	890	736
Core+reflector mass [kg]	2588	2148
Overall vessel mass [kg]	3478	2884
Regenerator		
Surface [m^2]	///	26
Volume [m^3]	///	0,02
Reactivity control		
Term 1 [pcm]	500	
Term 2 [pcm]	500	300
Term 3 [pcm]	7200	1700
Term 4 [pcm]	~ 0^+	
Total [pcm]	8200	2500
Masses [kg] (+5%)		
Overall reactor mass	3652	3028
Cold well	674	779
Compressor	105	
Turbine+alternator/Thermoelectric devices	105	105
Other components	210	
Overall system	4746	4227
Contingency (10%)	475	423
Total	**5221**	**4650**

4. COMPARISON BETWEEN THE TWO PROPOSALS

The critical comparison between the two solutions here proposed is difficult to carry out without any external information about specific technological issues.

Let us start with the PWR. The crucial issues are:

- The fuel;
- The steam turbine;
- The reactivity control;
- The pressurizer (only for propulsion solution).

The fuel is not a real technical issue, apart from the need to test it in a long irradiation program. As said in the introduction, the rod diameter is so small with respect of usual UO_2 rods, that a verification program is necessary, even if no particular difficulties can be envisaged. However, other fuel alternatives are possible, thanks to the high enrichment. For instance, instead of uranium oxide it is possible to imagine the use of a high alloyed uranium metal, as the one already studied for fast reactors and probably extensively used in submarine propulsion. In this case the geometry can be different from a rod as a curved plate, many of them suitably assembled is square boxes (Material Testing Reactor type). Therefore, there are two possibilities: i) if the already existing information about fuel adopted for special reactors will become available, no specific R&D program is needed; ii) if this is not the case, a rather long and expensive R&D program is needed in order to obtain the green light to adopt this new fuel.

The steam turbine is of paramount importance for this system. These small turbines are not yet developed, even if there is no particular reason to not reach such a goal, taking into account that some decrease of their efficiency is acceptable in this application. In particular, there is the leakage issue, which can impair the long term reliability of the overall system: it is difficult to build perfectly sealed components, when they are connected to outside by rotating shafts, as that which connects the turbine to the alternator. Anyway the leakage issue applies to the primary system as well, because losses of water are possible across the pressure vessel penetrations, necessary for the instrumentation electrical cables, and for the control mechanism (this may require a mechanical rotating shaft crossing the pressure barrier or an electrical cable in the case of using electrical motors inside the pressure vessel). Finally, the alternator may require some cooling need, not easily solvable in the space.

The reactivity control has been already discussed in par.2.5, however let us here recall the crucial mechanical aspects of this system.

The pressurizer working in absence of gravity, where steam and water cannot separate each other, is a demanding component and then for this reason no proposal has been studied in this report.

Passing to HTGR system the crucial issues are:

- High temperature components;
- Leakage;
- Thermoelectric generator;
- Criticality during flooding:

The fuel in this case is not a problem, because the elemental micro sphere is absolutely identical to that foreseen for commercial reactors. Maybe some more verifications and tests are necessary, but this will be done by the R&D programs associated to these commercial reactors. Moreover the limited constrains of this reactor fuel is a further assurance that the required specification will be met.

The high temperature is a big constrain for this reactor. In principle, there is no differences in this framework with the analogous commercial reactors. However, the high and durable reliability here required, together with a certain perplexity that in the commercial reactors they will be able to reach completely such a high temperature value, raise preoccupations about this issue.

The gas leakage seems more important in this reactor than in the previous one, because the helium is a mobile gas and difficult to collect, once escaped from the system. It is an aspect which requires a careful analysis. In principle, it may be supposed that this issue is more crucial for the Brayton cycle case than for that of the thermoelectric generator.

The latter is the hope and the problem of this reactor. If a reasonable *net* efficiency connected to a high reliability and durability can be demonstrated by such a device operating at high temperatures, a big push in favor of this reactor will be obtained.

The criticality danger during an accidental fall down into the sea, is not an easy task to cope with. This issue has not be analyzed in detail in this study and then a further analysis is needed before drawing founded conclusions.

In the above, the cold well issue has not been mentioned for both reactors. Two are the reasons: i) it is believed that this is an optimization problem, maybe difficult and demanding one, but not unsolvable, ii) the component is not specific to these reactors and so a general and generic R&D program should be launched for this component.

In conclusion, it is clear from the above considerations how much determinant can be the contribution of already existing experience and knowledge to simplify substantially the R&D program needed for these reactors, but this is out of our reach.

It is probable that PWR is less suited for propulsion than for surface application, because of the lack of gravity, which makes the pressurization control a complex task. On the other hand, maybe the a priori better reliability of such a reactor makes it more fit for surface application than the HTGR.

As for the masses, higher values are obtained for the HTGR, but the uncertainties of this estimation and the need of further ancillary components and circuits are probably higher than the differences with the PWR masses. However, an important aspect is the very low influence of the power level on the overall core mass of the HTGR system, which, if confirmed, may become an advantageous item by increasing the power.

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