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PRELIMINARY FEASIBILITY STUDY OF A WATER SPACE REACTOR WITH AN INNOVATIVE REACTIVITY CONTROL SYSTEM

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ABSTRACT

Power limitation represents a major issue within space applications aimed to human settlements on solar system planets. Among these planets, Mars is considered the most attractive because of its nearness to the Earth and the probable presence of minerals which can be used by the settlers to live off the land. In this frame, small size nuclear power plants can be an interesting solution to overcome the energy supply problem. This paper presents a preliminary feasibility study of a 100 kW_e self-pressurized water space reactor, with the aim to design a system characterized by compactness, intrinsic safety and simplicity of the main reactor control components. To this end an innovative reactivity control system, based on the control of the primary coolant mass flow rate, was adopted. The introduction of this system in the reactor design required a comprehensive core neutronics analysis in order to properly quantify the effect of the coolant on the reactor behaviour also as a function of the fuel burn-up. Here only the main results of this analysis, concerning neutron flux profiles and multiplication factors, are discussed. Moreover preliminary results on long term reactivity control are presented, showing the possibility to operate the reactor for as long as 7 years with no need of human intervention.

Key Words: Space Nuclear Reactor; Planet Settlement; Reactivity Control System; Core Neutronics

1. INTRODUCTION

The study of nuclear power plants for human settlements on solar system planets represents an attractive applied research field for nuclear engineers because of the strict requirements that an extra-terrestrial environment imposes. Several and different reactor designs have been proposed [1, 2, 3], all of them based on the following requests: (1) high reliability; (2) relatively cheap R&D programs; (3) possibility to deploy the plant within a reasonable period of time; (4) operability and control for a long time without any human intervention; (5) adaptability to some particular terrestrial applications.

Although many of the these space reactor designs were characterized by unconventional solutions, the possibility of space nuclear reactor plants based on well proven and reliable technologies was well and extensively discussed in [4]. Anyway, the design of such space systems should allow the development of different components and the simplification (or elimination) of redundant systems.

Two major problems deal with the design of the heat sink and the efficiency of the power cycle. The heat sink is a high demanding component because of the limitations on the heat transfer in a rarefied atmosphere where the thermal radiation plays the major role. This leads to extended surfaces, with a high risk to be hit and damaged by micrometeorites. Moreover heat removal by thermal radiation requires high temperatures of the condensing fluid. This means small cycle efficiency, not exceeding 12-13% in case of a conventional Rankine cycle. However this result is acceptable when compared to the efficiency of

thermoelectric devices which, in spite of their high reliability, show a conversion efficiency of few percents. Previous works [4, 5] identified the PWR reactor as the best solution to meet the requirements of a space surface application, relying on an unmatchable operating experience gained in terrestrial applications; besides, the PWR solution has been widely adopted for special applications, such as nuclear submarines and nuclear ships, whose features are even more similar to those of a space reactor. The SURE reactor proposed by Cammi et al. [4] was designed basically as a conventional PWR reactor, controlled by means of a core moving system: the best results in terms of reactivity reduction were obtained dividing the whole core in six slices rigidly movable in radial direction.
To avoid the issues concerning the mechanical design and the reliability of the core withdrawal system, an innovative reactivity control solution is proposed in this work: the reactor is now cooled and moderated by water reaching boiling conditions (as in a BWR reactor, but at an higher pressure) and no control rods neither moving systems are planned; the reactor is controlled varying the mass flow rate in the core and consequently the boiling front height. In this way the moderator density distribution, which strongly affects the core neutronics, can be adjusted to bring the reactor power at stationary levels.
Scope of the present paper is to present results of the feasibility study of a preliminary design of the space reactor in terms of core neutronics and its interaction with the coolant properties. Moreover results on long term reactor operation are presented in order to demonstrate the capabilities of the new reactivity control system.

2. DESIGN CHOICES

The main design choices of space reactor configuration are listed in Table I. The reactor is cooled and moderated by boiling water at the pressure of 155 bar. The adoption of a direct cycle would require shielding of all the components in the water line outside the reactor vessel; moreover the turbine would work with nuclear steam in absence of any maintenance and repairs for a long period of time. For these reasons an indirect cycle is adopted and the steam is not sent directly to the turbine but flows through the steam generator where the heat is exchanged with the secondary system.
The reactor is designed with a completely innovative control system based on the amount of water hosted in the core; the water-steam mixture provides a high negative void reactivity coefficient, which is used to control the reactor simply varying the mass flow rate circulating through the core. The adopted control strategy avoids the presence in the core of the control rods, as well as of any core withdrawal systems proposed with the previous design. No movable components work within the core (except for the coolant pump), hence a marked improvement in system reliability and simplicity is achieved; the feasibility of the proposed control system is discussed in details in Section 4, analyzing the neutronic features of the reactor.
The other design choices are not significantly different from SURE reactor; the pressure is the same of a conventional PWR in favour of a better neutron moderation in the boiling region and a higher efficiency of the thermodynamic cycle, thanks to the higher saturation temperature which is about 345°C. Subcooled water enters the reactor at a temperature of 335°C. The fuel is a metal matrix composed by 45% of highly enriched uranium (93% U_{235}) and 55% of $ZrH_{1.7}$. This type of fuel has been used and fully tested on Earth in TRIGA reactors [6] for a long time and it has been also qualified for space, being utilized in the SNAP 10A reactor [7]. The presence of hydrogen mixed within the fuel in metal matrix assures that a fraction of the moderator is solid, which is the main advantage of the provided fuel. The thermal properties of uranium-zirconium hydride are listed in Table II; they are given as function of temperature T, expressed in K, and hydrogen content x in ZrH_x. As far as fuel density is concerned, changes due to thermal expansions have been neglected because such variations, over a range of temperatures of 500 K, are less than 1%. Hence, the density is evaluated at 300 K and as a function of hydrogen content only.

Table I Space reactor main parameters.

Parameter	Choice	Parameter	Result
Net power	100 kW_e	*Core*	
Design	Integral layout	Uranium mass	21.5 kg
Fuel	93% enriched uranium	Fuel mass	47.8 kg
Composition	45% U - 55% $ZrH_{1.7}$	Core height	0.241 m
Cladding	AISI 316 L	Core diameter	0.276 m
Fuel rod diameter	1.52 cm	Core power density	72.7 kW/l
Cladding OD	1.78 cm	No. of cluster	19 with 7 rods
Cladding thickness	0.112 cm		
Fuel cluster geometry	Wrapped hexagonal	*Vessel*	
Core geometry	Hexagonal	External Sphere ID	1.4 m
Inlet temperature	335°C	Internal Sphere ID	0.85 m
Outlet temperature	344.9°C	Internal Sphere thick.	0.020 m
Pressure	15.5 MPa		
Primary side mass flow	1 kg/s	*Reflector*	
Coolant area	0.016 m^2	Height	0.321 m
Coolant velocity	0.098 m/s0.016 m^2	Thickness	0.1 m
Secondary side mass flow	0.347 kg/s		
Thermodynamic cycle	Rankine	*Steam generator*	
Max temperature	325°C	Geometry	Spiral
Min temperature	165°C	OD/t	0.025/0.00225 m
Inlet turbine pressure	4.8 MPa	Total length	100 m
Net efficiency	12.5 %		
Thermal power	800 kW	Chimney height	0.22 m
		Reactor weight	5300 kg

Table II Uranium-zirconium hydride thermophysical properties.

Property	Expression
Thermal conductivity λ	$\lambda = \lambda_U \cdot V_U + \lambda_{ZrHx} \cdot V_{ZrHx}$ *
Specific heat [8] c_p [J/(mol·K)]	$c_p = 25.02 + 4.746 \cdot x + \left(3.103 \cdot 10^{-3} + 2.008 \cdot 10^{-2} \cdot x\right) \cdot T + -\frac{\left(1.943 \cdot 10^{-5} + 6.358 \cdot 10^{-5} \cdot x\right)}{T^2}$
Thermal diffusivity [9] α [cm^2/s]	$\alpha = \frac{67.9}{\left[T + 1.62 \cdot 10^3 \cdot (2.00 - x) - 1.18 \cdot 10^2\right]} - 1.16 \cdot 10^{-2}$
Density [6] ρ_{ZrHx}	$\rho_{ZrHx} = \frac{1}{0.1706 + 0.0042 \cdot x}$

* V_U, V_{ZrHx} represent the volume fractions of uranium and zirconium hydride respectively.

3. REACTOR LAYOUT

A preliminary layout of the space reactor is depicted in Fig. 1. All the primary system components, i.e. the reactor core, the barrel, the steam generator, the main coolant pump and the instrumentations are hosted inside the reactor vessel, resulting in the so named "integral layout" commonly adopted in several Generation III+ reactors designs. The chosen configuration minimizes the size and the mass of the primary system.
The core is composed of 19 wrapped hexagonal fuel assemblies. Every assembly contains 7 fuel pins for a total number of 133 fuel rods. The wrapped arrangement permits to tune the inlet mass flow rate in each fuel assembly by means of orifices in order to have a radial outlet temperature distribution as flat as possible. The water flows upward through the core and then through a chimney which promotes natural circulation; once the flow reaches the top of the chimney, the coolant is directed downward through the annular downcomer region, where the steam generator tubes are located. The steam-water mixture flows on the outer surface of the steam generator tubes, exchanging heat with the secondary fluid (water) down to the lower plenum, where the coolant is collected and then rises again through the core by means of the circulating pump.

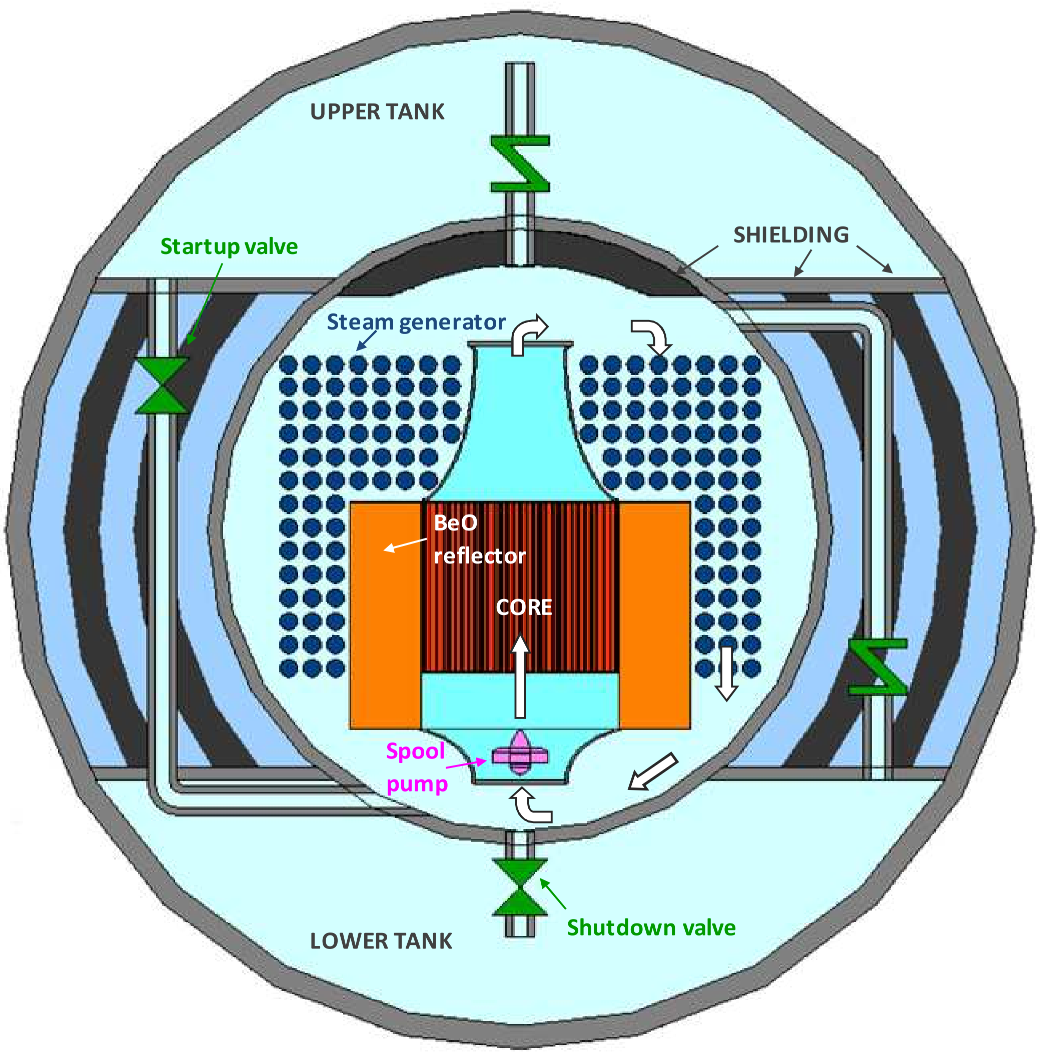

Figure 1 Layout of the space reactor pressure vessel.

The reactor vessel, characterized by a spherical shape, is made of stainless steel AISI 316 L. The primary system is provided with an upper and lower tank, aimed at controlling the mass of water hosted inside the vessel and designed to start up the system and shut down quickly the reactor in case of an accidental event. The two tanks are connected by an external pipe with a charging pump, not shown in Fig 1. The steam generator is located in the annular space between the reactor barrel and the vessel; a spiral tubes configuration has been adopted with six tubes located around the chimney and eight more tubes in the core region. The higher primary system pressure acts on tube outer surface, hence primary stresses are of compressive type; this means that most deterioration mechanism leading to tube failure, such as intergranular stress corrosion cracking, should be inherently eliminated. The coolant pump is of spool type; it assures the criticality of the reactor varying the mass flow rate in the primary system and it 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 for terrestrial reactors, as in the IRIS case [10].
The power conversion system is the well known Rankine cycle, duly simplified. Reasonably low turbine efficiency is assumed, to account for its small size, which stands outside present technology possibilities. A superheated steam cycle is considered, instead of a potentially more efficient saturated steam cycle, operating between the same temperatures, which have been discarded to avoid the need of moisture separators. Moisture presence is reduced, limiting turbine blade erosion problems in absence of long-term maintenance. The cold well finally assures the heat transfer to the environment only by means of radiation processes. According to a preliminary optimisation study, the overall mass is minimized with a condenser temperature of about 165°C, resulting in an overall cycle efficiency of 12.5%, a thermal power to be removed of 700 kW and a total condenser surface of 1100 m^2.

4. NEUTRONIC DESIGN

A detailed neutronic analysis has been performed to investigate the feasibility of the control system based on the amount of water mass contained in the reactor core and to evaluate the relationship between water level and system reactivity. The analysis has been carried out using the Monte Carlo code MCNP [11]. Parametric studies on the neutronic properties of the reactor permitted to build simple correlations which have been introduced in a more comprehensive dynamic model, in order to account for the thermal-hydraulics and neutronics coupling. The MCNP model includes the core where the fuel rods are arranged according to hexagonal geometry, the radial reflector of BeO and water. The description of the dynamic model and the discussion of some preliminary transient results are out of the scope of this paper, where just a stationary analysis is presented. The nuclear data used in the neutronic calculations are based on JEFF3.1 library [12] where neutron cross sections of more than 300 isotopes are available and are evaluated over a pretty wide range of temperatures, allowing to determine temperature coefficients in different operating conditions. IAEA cross section tables [13] were used for the $S(\alpha,\beta)$ treatment [11] of low energy neutron scattering with the H_xZr, H_2O and BeO molecules.
By means of MCNP it was possible to characterize the system in terms of radial and vertical neutron flux profiles as shown in Fig. 2. The radial profile was calculated by averaging the neutron flux in each fuel rod positioned along one diameter of the core. The radial peak factor, defined as the ratio between the peak value and the average value, results equal to F_r=1.11. For the axial profile it can be noticed how the thermal flux becomes more important near the reflector region, as expected. The axial peak factor is higher than the radial value, more precisely it is equal to F_z=1.24. Thus the total peak factor is given by $F_{tot}=F_rF_z$=1.38. The maximum total flux value is about 10^{14} $n\cdot cm^{-2}\cdot s^{-1}$.

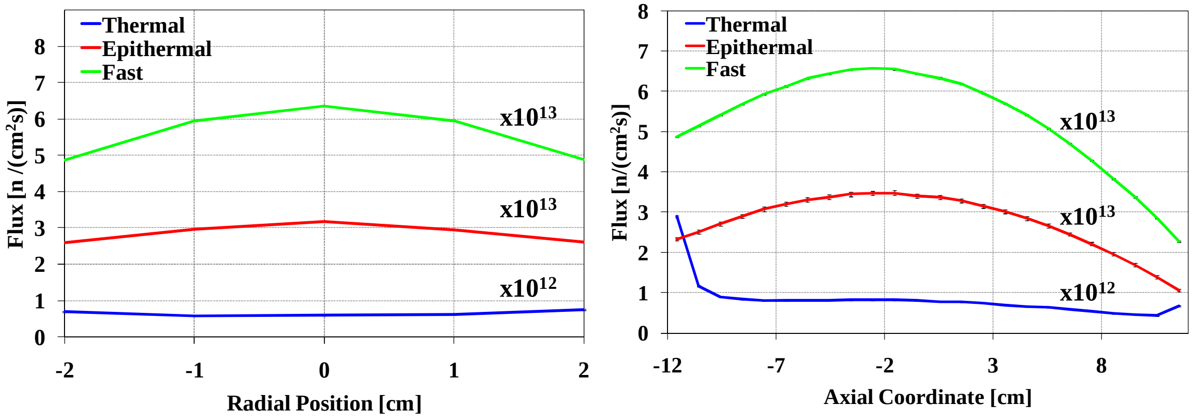

Figure 2 Radial and axial neutron flux profiles.

In order to fully characterize the reactor core neutronics at different operating conditions, the change of the effective multiplication factor for different values of coolant level was calculated assuming a temperature for the cold system of 300 K. In this case, the coolant in the core is essentially water in equilibrium with saturated steam. The same calculation has been repeated at zero power at the equilibrium temperature of 600 K and system pressure of 15.5 MPa. The results of the calculation show that at 300 K the system reactivity variation from core top to bottom of the coolant level is around 20000 pcm, whereas at 600 K the variation is 8000 pcm smaller. The coolant, in addition to its moderating function, works as a reflector.

For the cold system the criticality is reached when the coolant level is about 7.55 cm from the core bottom, whereas for the system at 600 K the criticality occurs when the level is 9.75 cm. This is mainly due to the smaller density of the coolant at this temperature (and pressure) and to the fuel temperature effect. A third calculation has been performed considering the coolant as a homogeneous mixture in nominal operating conditions, i.e. at the power of 800 kW. At this power the fuel temperature is assumed equal to 900 K and the rest of the system at the temperature of 600 K. The assumption of homogeneous mixture can be considered reasonable given the boiling process occurring along the channel core. In this situation a very important role is played by the top reflector represented by the coolant itself at the exit of the core. In Fig. 3 the behaviour of the MCNP calculated effective multiplication factor as a function of the homogenous mixture density is presented. The system results critical when the homogenous mixture density is equal to 282 kg/m^3. The moderator density reactivity coefficient is 30 $pcm/(kg \cdot m^{-3})$, which means a change of reactivity of about -60 pcm for a unitary change of inlet coolant temperature. The curves of Fig. 4 are really important for the understanding of the reactor start-up; they show the behaviour of the effective multiplication factor as a function of the coolant mass at the different operating conditions (cold system, hot system at zero power and nominal operating conditions). A tentative scheme for the reactor start-up can be summarised as follows:

A. at the beginning water is added to the core at room temperature (300 K), keeping the reactor subcritical (blue curve);
B. when still subcritical, the water is heated up to 600 K, and the new reactor state is to be found on the red curve;
C. other water is added in order to increase the system reactivity till the system becomes critical;
D. at this point the reactor starts up and the fuel temperature starts increasing, reducing reactivity because of its negative feedbacks;
E. in order to counterbalance the negative fuel temperature coefficient, water has to be added in the core till the reactor reaches the operating point.

Besides a strong moderator density coefficient, the core is characterized by a relative high fuel temperature coefficient. This coefficient is essentially due to two phenomena: the resonance broadening with the temperature (Doppler effect) and the interaction of neutrons with the hydrogen atoms in the zirconium hydride molecules. The influence of the fuel temperature on the multiplication factor was investigated by means of MCNP, carrying out criticality calculations for different values of fuel temperature. At the nominal power, the value of the fuel coefficient is estimated to be around -3 pcm/ºC, which is a small value with respect to the Highly Enriched Uranium (HEU) TRIGA fuel [14]. This can be explained by the fact that the Heavy Metal (HM) inventory constitutes the 45% in weight of the fuel composition. This leads to a reduction of the effect due to the interactions with hydrogen atoms, differently from the HEU fuel where the HM inventory is the 10%.

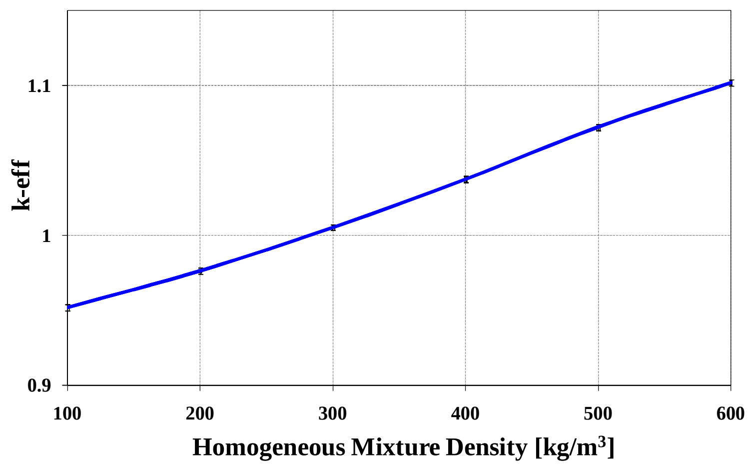

Figure 3 Reactor effective multiplication factor as a function of the homogeneous mixture density.

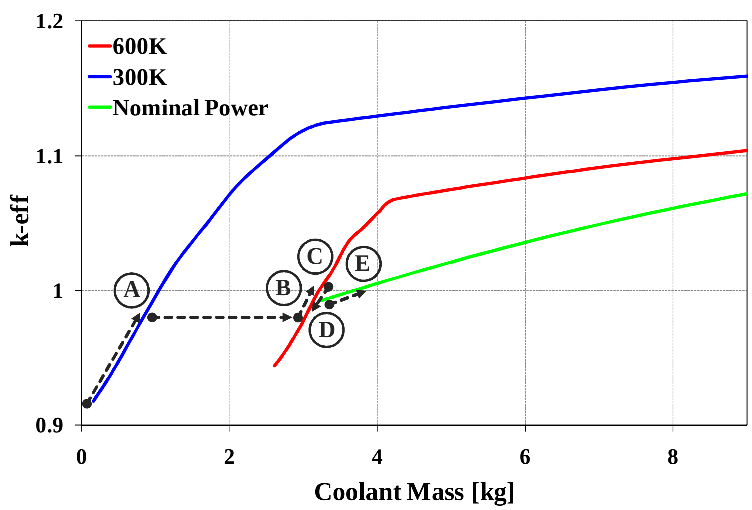

Figure 4 Reactor effective multiplication factor in different operating conditions as a function of the coolant mass, with explanation of a tentative start-up procedure.

5. STATIONARY ANALYSIS

A stationary analysis has been performed by calculating thermal-hydraulic channel parameters, i.e. exit vapour quality and mass flow rate, as a function of power (Fig. 5) and core inlet coolant temperature (Fig.

6) at nominal power. For each operating condition the value of the mass flow rate corresponds to a critical core. As it can be noticed from Fig. 5, as the power increases the exit vapour quality decreases. The increase of the power causes a negative reactivity insertion into the system because of the feedback of fuel temperature. In order to counteract this effect the mass flow rate must be increased. As for the inlet temperature dependency, a decrease of inlet temperature has the effect to introduce positive reactivity into the system. The only way to bring the reactor back to a critical configuration is to decrease the mass flow rate. This decrease of the mass flow rate is such that the exit steam quality is anyway lower.

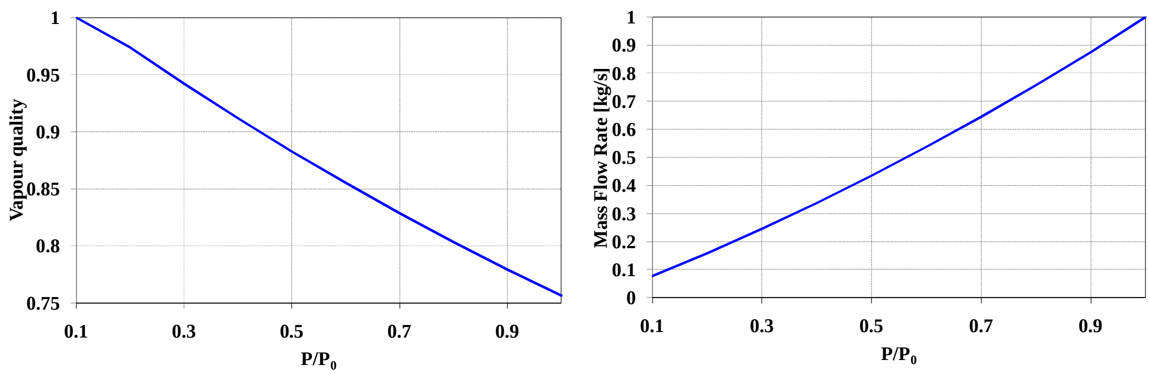

Figure 5 Primary system exit vapour quality and mass flow rate needed to keep the reactor critical as a function of reactor thermal power.

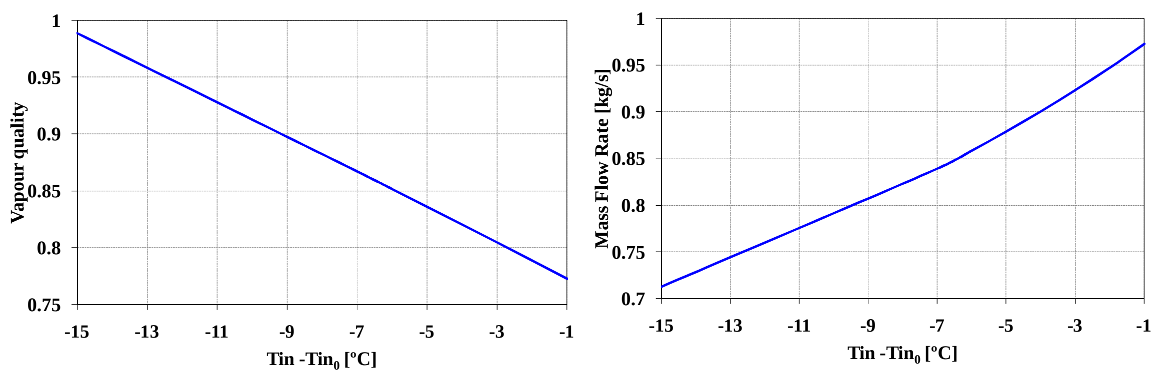

Figure 6 Primary system exit vapour quality and mass flow rate needed to keep the reactor critical as a function of inlet temperature reduction.

The reactivity control system of typical nuclear reactor is usually realized with rods made of absorbing material. The large amount of reactivity which can be managed by increasing or decreasing the coolant mass in the space reactor core suggested a control system based on a pump capable of finely changing the coolant mass flow rate in the core channel. In this hypothesis, a study was carried out in order to estimate the reactor behaviour as a function of the burn-up effects. In the developed model the control of the mass flow rate has been realized by means of a PI controller with a reference law given by P = 800 kW (k_{eff}=1). The input is given by the reactivity variation due to the burn-up of the fuel calculated with BGCore [15], which is a fuel depletion code coupled with MCNP. Fig.7 shows the behaviour of the exit vapour quality as a function of time at full power. As shown in the figure, after a period of time of about 7.2 years, the mass flow rate control is not anymore capable of counteracting the effect of burn-up. At this point the vapour quality is zero. The mass flow rate reaches a maximum value of about 11 kg/s whereas the initial mass flow rate is about 1 kg/s. It is worth to notice that as the burn-up increases the mass flow rate has to be increased more rapidly in order to introduce the amount of reactivity requested by the fuel consumption.

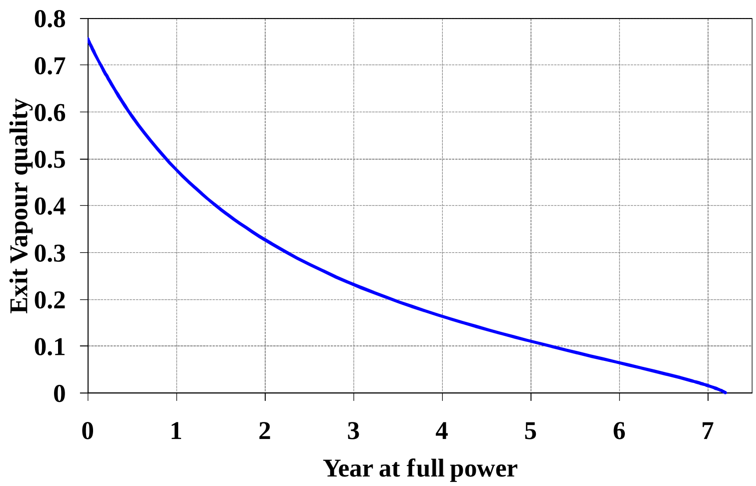

Figure 7 Exit vapour quality as a function of time at full power.

6. OPEN ISSUES

The performed neutronic analysis has demonstrated the feasibility of a water reactor for space applications, controlled by varying the mass flow rate circulating through the core. An R&D program of large extent will be of course necessary, dealing with the many aspects discussed in this paper, as well as with not considered drawbacks.

First of all, the use of uranium-zirconium hydride as nuclear fuel requires further investigations, considering also the possibility of its replacement with a more conventional uranium-oxide fuel. The reactor control, based on time-changing core mass flow rate, must be in-depth analyzed, in particular when dealing with critical situations such as start-up and shut-down transients. As for emergency reactor scram obtained by draining the water of the core into the lower tank, it is of interest to consider the possible establishment of natural circulation driven by the decay power. The dynamic behaviour of the reactor is currently under analysis by means of three different simulation codes: a dynamic point-wise core model implemented with SIMULINK [16], RELAP [17] code and MODELICA [18]. In particular, the latter will give the chance to approach the reactor design within an object oriented environment, where sub-models developed for different systems can be easily reused, allowing the implementation of multi-physics systems in the same application.

The power cycle needs a further optimisation process; it must satisfy the requirements assigned on cycle efficiency maintaining system parameters consistent with existing technological limits. In particular, two issues deserve a careful investigation: the turbine, whose size is outside present technology possibilities, and the opportunity to use an organic fluid as working fluid, which could assure higher efficiency and lower technological limits on the size of the turbine. A detailed check on the value of critical heat flux in the primary system is furthermore needed to prevent any risk of thermal crisis, given the boiling characteristics of primary coolant. In addition to all the activities mentioned above, other subjects will have to be included in future R&D programs. A non-exhaustive list is: radiation shielding, vessel fluence, safety features, overall layout, containment, leakage control, coolant purification and radiolysis.

7. CONCLUSIONS

A preliminary feasibility study on a water reactor for space applications has been presented. The reactor proposed is cooled by a boiling mixture of water and steam. An innovative control strategy has been adopted and investigated; the reactivity control system is based on the regulation of water mass flow rate in the primary system by means of a pump. The peculiarity of the adopted design is the absence of any movable components in the core (except for the coolant pump). Neither control rods nor a complex core moving system are required, differently from the previous configuration of SURE reactor [4]. Reactor reliability and simplicity are therefore deeply enhanced. A detailed neutronic model was implemented in MCNP in order to perform a comprehensive analysis aimed to the evaluation of the system reactivity variation as a function of the coolant level, which acts as control variable. The behaviour of the mass flow rate at different operating conditions was studied, the results representing a first control strategy of the reactor. As concerns fuel burn-up effects, a progressive increase of the mass flow rate can counteract the reactivity loss for 7.2 years of operation at full power at most. A R&D program of large extent concerning different issues will be necessary; a schedule of short range activities has been identified and several studies are currently under development. Anyway no unsolvable issues arise from the performed preliminary design study.

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