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Conceptual Design of the Fixed Bed Nuclear Reactor (FBNR) Concept

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1. Introduction

This is a report on the work performed at the Federal University of Rio Grande do Sul in Brazil on the conceptual design of the Fixed Bed Nuclear Reactor (FBNR) concept that constitutes a part of the IAEA Project on Small Reactors without On-Site Refuelling and is performed under the IAEA Research Contract No. 12960/ Regular Budget Fund (RBF).

The Small Reactors without On-Site Refuelling are defined by IAEA "As reactors which have a capability to operate without refuelling and reshuffling of fuel for a reasonably long period consistent with the plant economics and energy security, with no fresh and spent fuel being stored at the site outside the reactor during its service life. They also should ensure difficult unauthorized access to fuel during the whole period of its presence at the site and during transportation, and design provisions to facilitate the implementation of safeguards. In this context, the term "refuelling" is defined as the ´removal and/or replacement of either fresh or spent, single or multiple, bare or inadequately confined nuclear fuel cluster(s) or fuel element(s) contained in the core of a nuclear reactor`. This definition does not include replacement of well-contained fuel cassette(s) in a manner that prohibits clandestine diversion of nuclear fuel material."

The work plan consists of

1. Demonstration of FBNR concept applicability to small reactors without on-site refuelling.
2. Analysis of passive cooling options for FBNR.

2. Demonstration of FBNR concept applicability to small reactors without on-site refuelling.

Table 1: Summary of the requirements and their applicability to FBNR.

Requirement	Demonstration of applicability
Small in size	FBNR is small in nature. The optimum size for the FBNR is 40 MWe. The larger size up to a maximum of 60 MWe can be achieved only at the cost of a lower thermodynamic efficiency.
Modular	The reactor is modular in design; therefore, any size reactor can be built from the basic module. The modular aspect of the reactor leads to the mass production processes resulting in better economy and higher quality products.
No need for on-site refuelling	Each module is fuelled in the factory. The fuelled modules in sealed form are then transported to and from the site. The FBNR has a long fuel cycle time and, therefore, there is no need for on-site refuelling.
Proven Technology	FBNR makes an extensive use of a proven technology namely that of PWR. Its fuel is of HTGR type.
Diversity of applications	The FBNR is a land-based nuclear power plant for urban or remote localities The FBNR is designed to produce electricity alone or to operate as a cogeneration plant producing electricity and potable water or steam for industrial purposes. As an option, the FBNR may be designed for district heating.
Refuelling in the factory	No refueling on the site is needed because the fuel elements are collected in the fuel chamber and transported to the factory for refueling under surveyed condition. Refuelling is done by the replacement of fuel chamber.
Long fuel cycle time.	The length of the fuel cycle chosen depends on the economic analysis of the fuel inventory for particular situation of the reactor and its application. The HTGR fuel elements have high burn up capacity. The replacement of fuel chamber is done at any desired time interval and could be set at every 10 years or for the reactor lifetime.
No fuel reshuffling	No reshuffling of fuel is necessary because the fuel elements go from fuel chamber to the core and vice versa without the need of opening the reactor.
No fresh fuel storage on site	There is no need for fresh fuel to be stored at the reactor site since the sealed fuel chamber is transported to and from the factory where refueling process is performed.
Short period of spent fuel storage on site.	The spent fuel that are confined in the fuel chamber and kept cool by its water tank. It can be sent back to the factory at any time when the radiological requirements are met.
Inaccessibility of fuel to unauthorized individuals.	No unauthorized access to the fresh or spent fuel is possible because the fuel elements are either in the core or in the fuel chamber under sealed condition so no clandestine diversion of nuclear fuel material is possible.

High fabrication quality & economy.	The FBNR is shop fabricated thus it guarantees the high quality fabrication and economic production process.
Easy transportation	The reactor is about 2 m in diameter and 6 m high, while its fuel chamber is only 2 m in diameter and 1 m high, thus the transportation to the site and return is very easy and convenient.
Easy dismantling	The reactor loop being made of relatively small components, at the end of its useful lifetime, the reactor can be dismantled and even disposed of in one piece with simplicity.
Reduced number of operators required.	The reactor can be operated with a reduced number of operators or even be remotely operated without any operator on site. This is possible due to the inherent safety characteristics of the reactor as the reactor operates when all the operating parameters are within the designed ranges. In any other situation, the electricity does not reach the pump to operate the reactor and the fuel elements will fall out of the core by the force of gravity or remain in the fuel chamber under a highly subcritical and passively cooled conditions.
Simplicity & economy	The simplicity of design and the lack of the need for large number of redundancies in control system, make the reactor highly economic.
Simple infrastructure	The infrastructure needed for the plant using FBNR is a minimum. The important processes are performed in the shop that can be in a regional centre serving many reactors.
Underground containment and environment	The inherent safety and passive cooling characteristics of the reactor eliminate the need for containment. However, an underground containment is envisaged for the reactor to mitigate any imagined adverse event, but mainly to help with the visual effects by hiding the industrial equipments underground and presenting the nuclear plant as a beautiful garden compatible with the environment acceptable to the public.
Utilization of spent fuel, nuclear waste and environment.	The spent fuel from FBNR is in a form and size (1.5 cm diameter spheres) that can directly be used as a source of radiation for irradiation purposes in agriculture and industry. Therefore, the spent fuel from FBNR may not be considered as waste, in a peaceful world of the future, as it can perform a useful function. They may also be reprocessed after their use as radiation source. Should reprocessing not be allowed, the spent fuel elements can easily be vitrified in the fuel chamber and the whole chamber be deposited directly in a waste repository. These factors result in reduced adverse environmental impact.
High conversion ratio	The moderator to fuel volume ratio of FBNR is about 0.7-0.8, compared to 1.8-2.0 for a conventional PWR. Thus, the neutron spectrum in the FBNR is harder resulting in a higher conversion ratio than the 0.55 for PWR that may be about 0.7-0.8. It may permit using MOX fuel, even in the beginning of the fuel cycle needing lower uranium enrichment, resulting in a higher conversion ratio.

Fool proof nuclear non-proliferation characteristic.	The non-proliferation characteristics of the FBNR is based on both the extrinsic concept of sealing and the intrinsic concept of isotope denaturing. Its small spherical fuel elements are confined in a fuel chamber that can be sealed by the authorities for inspection at any time. Only the fuel chamber is needed to be transported from the fuel factory to the site and back. There is no possibility of neutron irradiation to any external fertile material. Isotopic denaturing of the fuel cycle either in the U-233/Th or Pu-239/U cycle increases the proliferation resistance substantially. The use of thorium based TRISO type fuel will also contribute to this end. Therefore, both concepts of "sealing" and "isotope denaturing" contribute to the fool proof non-proliferation characteristics of FBNR.
High level of safety	Strong reliance on inherent and passive safety features and passive systems
Enhanced safeguard ability	Fuel elements are confined in the fuel chamber that could be sealed by authorities for inspection at the end of the fuel life. The reactor vessel is cladded by neutron-absorbing materials to eliminate the possibility of neutron irradiation of any external fertile material.
Technology transfer	The technology could be open to all nations of the world under the supervision and control of international authorities.
Enhanced sefety	Reactivity excursion accident cannot be provoked. The reactor core is filled with fuel only when all operational conditions are met.
Mitigation of steam generator leakage problem	The water heated in the reactor core passes through an integrated steam generator producing steam to drive the turbine.
Reduced adverse environmental impacts	Underground containment in a garden like site.
Long core lifetime	Insertion of fresh fuel into the core is performed continuously to compensate for fuel burn-up.
Resistance to unforeseen accident scenarios.	Any probable accident, through cutting off the power to the pump, causes the fuel elements fall out of the core driven by the force of gravity. The normal state of control system is "switch off". The pump is "on" only when all operating conditions are simultaneously met.
Low fuel temperature	A heat transfer analysis of the fuel elements has shown that, due to a high convective heat transfer coefficient and a large heat transfer surface-to-volume ratio, the maximum fuel temperature and power extracted from the reactor core is restricted by the mass flow of the coolant corresponding to a selected pumping power ratio, rather than by design limits of the materials.
Dual purpose plant	The FBNR can operate within a cogeneration plant producing both electricity and desalinated water. A Multi-Effect Distillation (MED) plant may be used for water desalination. An estimated 1000 m^3/day of potable water could be produced at 1 MW(e) reduction of the electric power.
Low capital investment	The simplicity of design, short construction period, and an option of incremental capacity increase through modular approach result in a much smaller capital investment.

3. Analysis of passive cooling options for FBNR.

The fixed bed nuclear reactor (FBNR)[1-3] is a small reactor (40 MWe) without the need of on-site refueling. It utilizes the PWR technology but uses the HTGR type fuel elements. It has the characteristics of being simple in design, modular, inherent safety, passive cooling, proliferation resistant, and reduced environmental impact. Here, a thermal and hydraulics analysis of this reactor concept is presented, and in particular its passive cooling characteristic is demonstrated.

3.1 Reactor description

The Fixed Bed Nuclear Reactor (FBNR) is modular in design, and each module is assumed to be fuelled in the factory. The fuelled modules in sealed form are then transported to and from the site. The FBNR has a long fuel cycle time and, therefore, there is no need for on-site refuelling. The reactor makes an extensive use of PWR technology.

It is an integrated primary system design. The basic modules have in its upper part the reactor core and a steam generator and in its lower part the fuel chamber, see Fig.1. The core consists of two concentric perforated zircaloy tubes of 20 cm and 160 cm in diameters, inside which, during the reactor operation, the spherical fuel elements are held together by the coolant flow in a fixed bed configuration, forming a suspended core. The coolant flows vertically up into the inner perforated tube and then, passing horizontally through the fuel elements and the outer perforated tube, enters the outer shell where it flows up vertically to the steam generator. The reserve fuel chamber is a 40-cm diameter tube made of high neutron absorbing alloy, which is directly connected underneath the core tube. The fuel chamber consists of a helical 25 cm diameter tube flanged to the reserve fuel chamber that is sealed by the international authorities. A grid is provided at the lower part of the tube to hold the fuel elements within it. A steam generator of the shell-and-tube type is integrated in the upper part of the module. A control rod slides inside the centre of the core for fine reactivity adjustments. The reactor is provided with a pressurizer system to keep the coolant at a constant pressure. The pump circulates the coolant inside the reactor moving it up through the fuel chamber, the core, and the steam generator. Thereafter, the coolant flows back down to the pump through the concentric annular passage. At a certain pump velocity, the water coolant carries up the 15 mm diameter spherical fuel elements from the fuel chamber into the core. A fixed suspended core is formed in the module. In a shut down condition, the suspended core breaks down and the fuel elements leave the core and fall back into the fuel chamber. The fuel elements are made of TRISO type microspheres used in HTGR.

Any signal from any detector due to any initiating event is assumed to cut-off power from the pump, causing the fuel elements to leave the core and fall back into the fuel chamber, where they remain in a highly subcritical and passively cooled condition. The fuel chamber is cooled by natural convection transferring heat to the water in the tank housing the fuel chamber.

The pump circulates the water coolant in the loop and at the mass flow rate of about 141 kg/sec, corresponding to the terminal velocity of 1.64 m/sec in the reserve fuel chamber, carries the fuel elements into the core and forms a fixed bed. At the operating mass flow rate of 668 kg/sec, the fuel elements are firmly held together by a pressure of 10 bar forming a stable fixed bed. The coolant flows radially in the core and after absorbing heat from the fuel elements enters the integrated heat exchanger of tube and shell type. Thereafter, it circulates back into the pump and the fuel chamber. The long-term reactivity is supplied by fresh fuel addition and a fine control rod that moves in the center of the core controls the short-term reactivity. A piston type core limiter adjusts the core height and controls the amount of fuel elements that are permitted to enter the core from the reserve chamber. The control system is conceived to have the pump in the "not operating" condition and only operates when all the signals coming from the control detectors simultaneously indicate safe operation. Under any possible inadequate functioning of the reactor, the power does not reach the pump and the coolant flow stops causing the fuel elements to fall out of the core by the force of gravity and become stored in the passively cooled fuel chamber. The water flowing from an accumulator that is controlled by a multi redundancy valve system cools the fuel chamber as a measure of emergency core cooling system. The other components of the reactor are essentially the same as in a conventional pressurized water reactor.

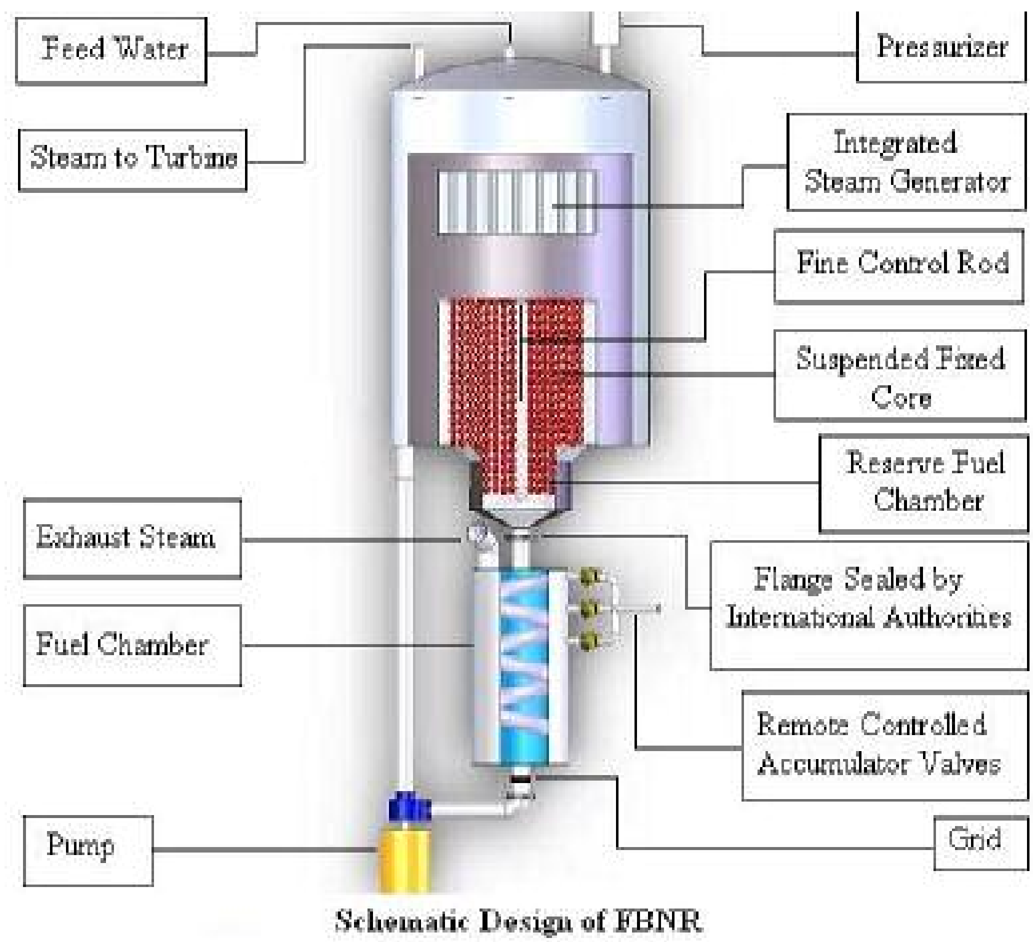

Fig.1: Schematic Design of the Fixed Bed Nuclear Reactor (FBNR).

3.2 Characteristics of the reactor

Table 1 shows a summary of the requirements of a small reactor without on-site refuelling and their applicability to FBNR. The detailed technical information are presented in table 2.

Table 2. Summary of the parameters of the Fixed Bed Nuclear Reactor module

Parameter	**Value**
Power:	
Net power generation (MWe)	40
Power generation (MWt)	134
Core power density (KWt/lit)	33.7
Pump power (MWe)	3.4
Hydraulics:	
Coolant volume (m^3)	12
Coolant mass flow (kg/sec)	668
Coolant pressure (bar)	160
Pressure loss in the loop (bar)	100
Pressure loss in the bed (bar)	9.5
Terminal velocity (m/sec)	1.64
Thermal:	
Coolant inlet temperature (ºC)	290
Coolant outlet temperature (°C)	326
Coolant inlet enthalpy (kJ/kg)	1284
Coolant inlet density (kg/m^3)	747
Enthalpy rise in the core (kJ/kg)	1490
Film boiling convective heat transfer coefficient at 300 ºC (W/m^2ºC)	454
Fuel element average thermal conductivity (W/m.ºC)	30.58
Fuel element average specific heat (J/kg.ºC)	802.5
Fuel element average density (gr/cm^3)	4.041
Maximum fuel temperature after a LOCA (ºC)	< 357
Coolant temperature rise after a LOFA after 10 days (ºC)	< 1
Water needed to cool during 10 days after LOCA (m^3)	0.45
Module dimensions:	
Core height (cm)	200
Core inner diameter (cm)	20
Core outer diameter (cm)	160
Core volume (m^3)	3.96
Fuel in the core (Ton)	9.6
UO2 in the core (Ton)	4.8
Fuel element	
Fuel element diameter (cm)	1.5
SiC clad thickness (cm)	0.1
Number of microspheres in a fuel element.	165
Number of fuel elements in the core.	1.34×10^6
UO2 in each fuel element (% vol)	19.3
Dense graphite in each fuel element (% vol)	27.8
Porous graphite in each fuel element (% vol)	7.4
SiC in each fuel element (% vol)	45.5
UO2 density (gr/cm^3)	10.5
PYC porous density (gr/cm^3)	1.0
PYC dense density (gr/cm^3)	1.8
SiC density (gr/cm^3)	3.17

3.3 Pressure loss in the reactor loop

The pressure loss in the reactor loop is calculated in order to determine the pump work necessary to operate the reactor. The minimum coolant velocity in the reserve fuel chamber should be equal to the terminal velocity for the fuel elements in order to carry the fuel elements out of the reserve fuel chamber into the core. The higher the coolant velocity, the higher is the core heat removal and consequently higher thermal power is produced. However, the higher coolant velocity requires higher pumping power. Therefore, the choice of operating coolant mass velocity is based on the power fraction for the pump one is prepared to pay for power generation.

3.3.1 Terminal velocity

The terminal velocity of 1.64 m/sec in the reserve fuel chamber is calculated by the method of NASA [4]. This is the minimum velocity for the coolant should be maintained in order to guarantee that the fuel elements are in the core maintaining a fixed bed. The mass flow in the loop is 22 times that correspond to terminal velocity. Therefore, at higher velocities the pressure holding the fuel elements together in the bed is increased.

NASA proposes the following expression for determination of the terminal velocity Ut .

$Ut = \sqrt{\dfrac{2.W}{C_d.r.A}}$; where, W = Weight of fuel element $= 7.145 \times 10^{-3} kg * 9.81 m/s^2 = 0.07N$

C_d = drag coefficient = 0.38 (by FOX), r = coolant density $= 774 kg/m^3$, A = fuel element cross section area $= 1.77 \times 10^{-4} m^2$, results in Ut = 1.64 $^{\text{m}}\!/_{\text{sec}}$.

3.3.2 Pressure loss in fixed bed

The pressure loss in the fixed bed is calculated using Idelchick Handbook of Hydraulic Resistance[5]. The coefficient for pressure loss (k) due to flow in the fixed bed of spherical fuel elements are calculated by:

$$k = \frac{\lambda l_0}{d_{e1}} + \Delta\zeta_t \quad ; \quad \lambda = \frac{A_1}{\text{Re}_1} + B_1 \quad ; \quad \text{Re}_1 = \frac{\rho v_1 d_{e1}}{\mu} \quad ; \quad de_1 = \varphi_1 d_{gr} \quad ;$$

$$\Delta\zeta_t = 2\frac{T_{ex} - T_{in}}{T_m} \quad ; \quad T_m = \frac{T_{in} + T_{ex}}{2} \quad ; \quad A_1 = 360\frac{(1-\varepsilon')^2}{\varepsilon'^3} \quad ; \quad B_1 = B'\frac{(1-\varepsilon')}{\varepsilon'^3} \quad ;$$

$$\varepsilon' = 1 - \frac{\pi}{6(1-\cos\theta)\sqrt{1+2\cos\theta}}$$

Where, B'=1.8 (for bodies with smooth surface), φ_1=1(coefficient of the body shape), and d_{gr} =0.015 m (mean size diameter of the body).

The above formula is recommended for Reynolds of up to 10^3 while in the present system it is in the order of 10^4. The variation of pressure loss as the function of mass flow is shown in Fig. 2. In the operational condition of mass flow of 668.53 kg/sec, the pressure loss in the fixed bed is 9.52 bar.

3.3.3 Pressure loss in the circuit

The pressure loss in the circuit, that is the pressure loss in the loop excluding the reactor core, was calculated using the following expression resulting from the balance of energy.

$$\left(p_1 + 10000.\frac{\rho}{g}.\alpha_1.\frac{V_1^2}{2} + 10000.\rho.z_1\right) - \left(p_2 + 10000.\frac{\rho}{g}.\alpha_2.\frac{V_2^2}{2} + 10000.\rho.z_2\right) = h_{lT}$$

where, p being pressure at the point of interest in bar, ρ is coolant density in kg/cm³, α, the correction factor for flow profile, V the average coolant velocity m/sec, g acceleration of gravity m/sec², and z the height at the point of interest in m. lt is the total pressure loss in bar being the sum of pressure loss due to friction and localized loss due to the change of geometry.

The flow in the loop is found to be very turbulent having Reynolds numbers of the order of 10 millions. Therefore, the friction coefficients were practically constants varying between 0.013 and 0.038. A fully developed flow was assumed, thus α =1 was used in the calculations.

Therefore, the pressure losses due to friction were calculated using the relation. $hl = fe.\left(\frac{L}{D}\right).\frac{V^2}{2}$ Where L and D are tube length and diameter respectively. V is the average coolant velocity in the tube. The localized pressure losses are calculated by the formula. .

$$h_l = K\left(\frac{V^2}{2}\right)$$

Where, the values of K were obtained from the Idelchik, Handbook of Hydraulic Resistance. The values of K vary between 0.11 in gradual expansion to 7.9 in the case of change of flow direction in 180°.

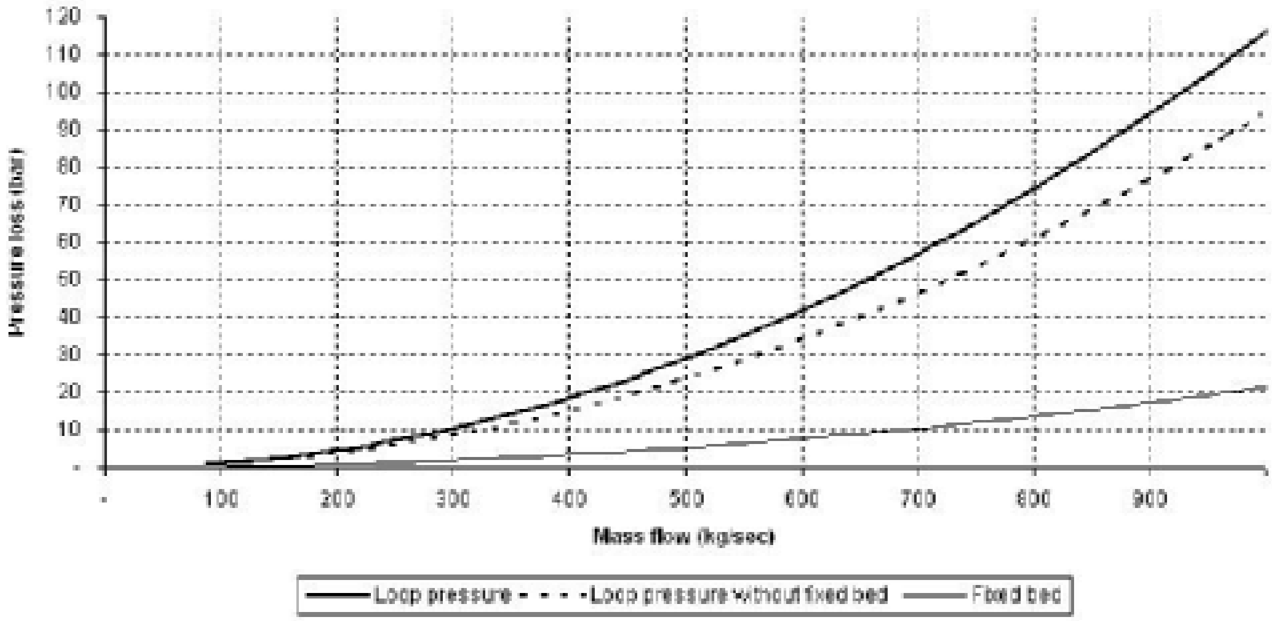

Fig 2. Pressure losses in the loop and the bed as a function of coolant mass flow rate.

4. Pump work

A centrifugal pump is envisaged for the coolant circulation in the reactor. However, when the mass flow is 930 kg/sec, the pressure drop in the loop is about 100 bars that according to Henn [6] it represents a pressure limit for making the use of a centrifugal pump. This will cause a limit on the size of a FBNR module to be 186 MWt. According to Fig. 5. The maximum power produced is when the mass flow is 1407 kg/s corresponding to 281 MWt (93 MWe). At higher flow rates, the net power production will decrease. The maximum possible net power produced by this reactor module is 61 Mwe.

The pump work is calculated by the expression: $P_{pump} = \dot{m}\ g\ \Delta p$, where,

P_{pump} is power pump (W), $\dot{m}$ mass flow (kg/sec), g acceleration of gravity (m/sec^2), and Δp the total pressure loss (bar).

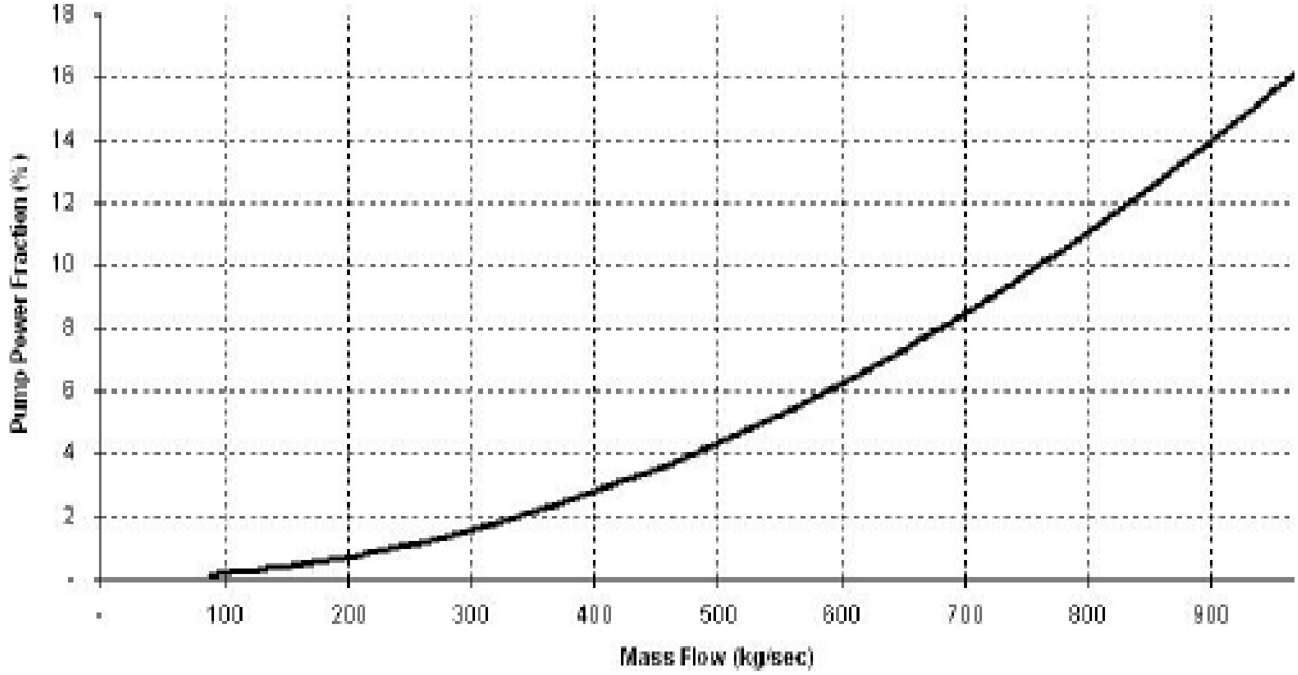

Fig.3: Pump power fraction as a function of coolant mass flow

5. Reactor power generation

The power generated is calculated by the enthalpy rise of the coolant passing through the reactor core. The inlet and outlet temperatures are assumed to be the same as for a conventional PWR, namely 290 °C and 326 °C with their respective enthalpies of 1284 kJ/kg and 1490 kJ/kg under 160 bar pressure. For electric power calculation, a thermodynamic efficiency of 33% is assumed. The power generation as a function of mass flow rate is shown in Fig. 4.

Should the reactor be cooled by supercritical steam under 250 bar pressure with the inlet/out temperatures of 290/416°C, having corresponding to enthalpies of 1280/2743 kJ/kg respectively, the reactor power can be increased to 977 MWt corresponding to 325 MWe electrical power.

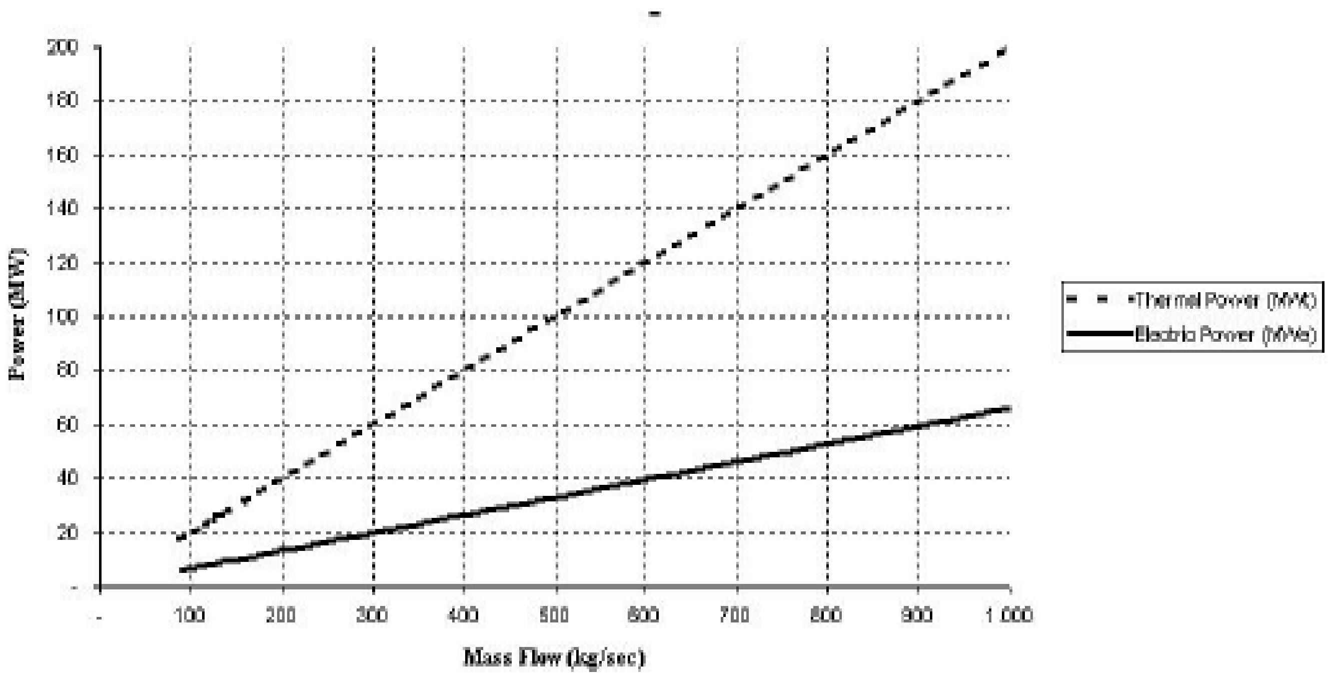

Fig 4. Reactor power generation as a function of mass flow

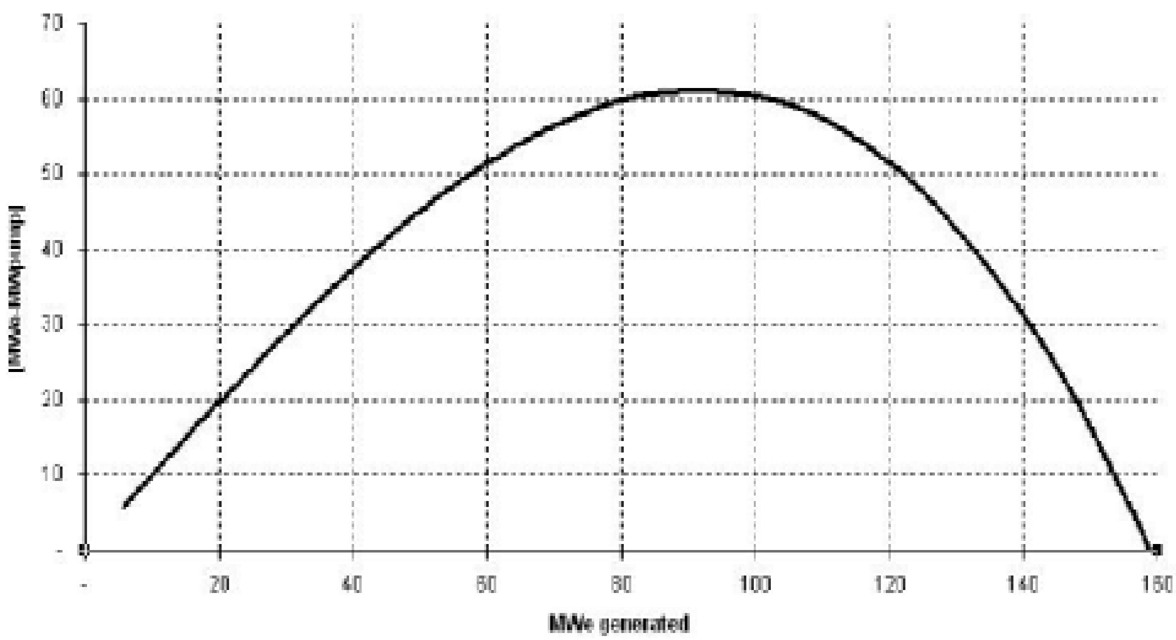

Fig. 5. The net electrical power generated as a function of reactor power

6. Passive cooling characteristic of the reactor

One of the most important merit of the FBNR is its passive cooling characteristic. Here, the passive cooling characteristic is examined and demonstrated.

Under any probable accident condition, the fuel elements fall into the fuel chamber under the force of gravity where they generate heat due to the radioactive decay of fission products. In this situation the remote controlled valves, having multiple redundancies, are opened and the water from the accumulator will enter the fuel chamber tank to remove the decay heat. In this analysis, it is assumed that the heat transfer that occurs between the room temperature cooling water and the fuel chamber tube being at about 300°C is by film boiling process. This heat transfer coefficient is calculated to be about 453 kW/°C m². This results in cooling capability of about 91 KWt/m². The real heat flux is calculated by dividing the total heat generation by the tube area and the results are presented in Fig.6.

The accumulated decay heat as a function of time is shown in Fig. 8. [7] The room temperature water to evaporate absorbs about 2592 KJ/kg of heat. Therefore, The amount of water needed to evaporate to cool the fuel chamber is shown in Fig.9. The water required to cool the reactor during the 10 days after the accident is less than 0.5 m³.

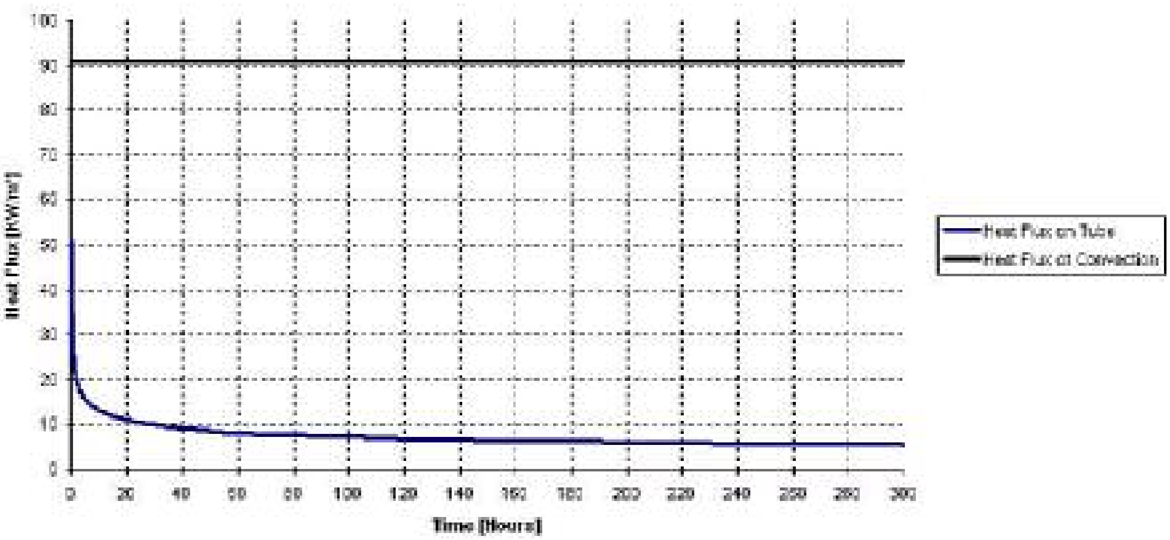

Figure 6 – The heat flux for a real case and its ability to transfer heat.

6.1 Decay heat generation

According to El-Wakil [7], the decay heat can be calculated by the following expression.

$$P_s = 0,095.P_0.\theta^{-0,26}$$

Where, P_0 is the nominal power of the reactor in MWt; θ is the time in seconds and P_s is the decay heat. The decay heat generation as a function of time is shown in Fig.7.

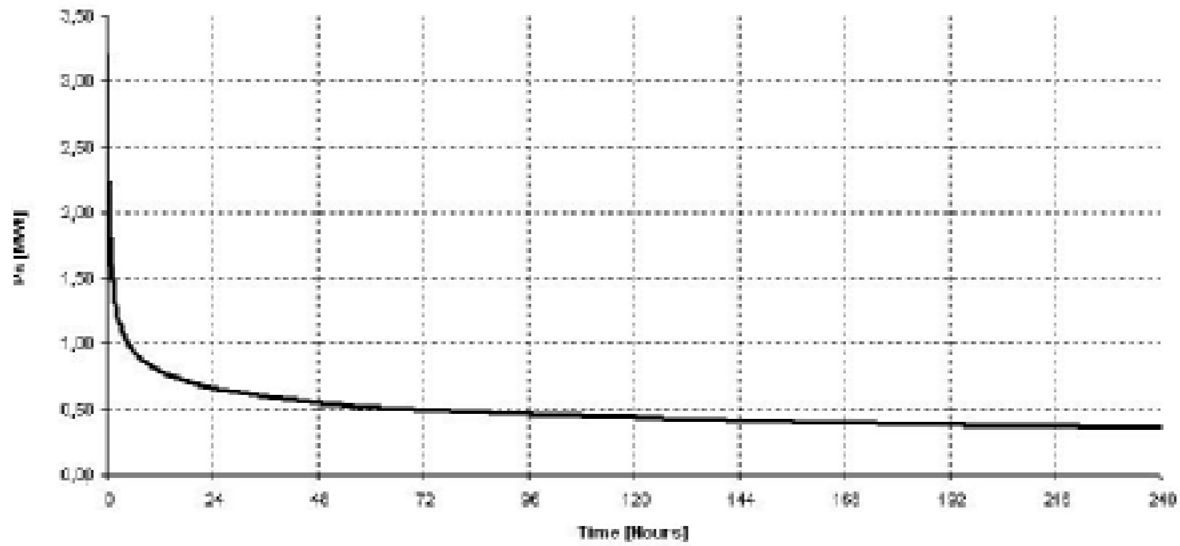

Figure 7 - Decay heat generation as a function of time

6.2 Accumulated released energy

The accumulated heat from radioactive decay of fission products as a fuction of time is calculated according to the formula [7]:
$E_s = 0.128\theta_s^{0.74}$, w*here* E_s is the realased energy and θ_s is the time in seconds. The results are presented in Fig. 7.

6.3 Heat transfer coefficient

The fuel chamber is to be cooled after a loss of coolant accident (LOCA) by water coming from the accumulator into the fuel chamber tank. It is assumed that the heat transfer will be through film boiling regime. The heat transfer coefficient α for film boiling is calculated according to John G. Collier and John R. Thome´s Convective Boiling and Condensation Third Edition, Equation 4.67 and 4.66 [8]

$$\alpha_{fb} = 0,425 \times \left(\frac{\rho_G \times (\rho_L - \rho_G) \times g \times h'_{LG} \times \lambda_G^3}{\mu_G \times (T_W - T_{SAT}) \times \left(\frac{\sigma}{g \times (\rho_L - \rho_G)} \right)^{1/2}} \right)^{1/4} \quad ; h'_{LG} = h_{LG} \left[\text{`}1 + 0,5 \left(\frac{c_{pg} . \Delta T}{h_{LG}} \right) \right]$$

Properties used are: $\rho_G = 7,78$; $\rho_L = 1000$; $g = 9,81$; $h'_{LG} = 22,235 \times 10^5, \lambda_G = 0,0363$;
$\mu_G = 1,56 \times 10^5$, $T_W = 300$, $T_{SAT} = 100, T_{film} = 200$; $\sigma = 3,85 \times 10^{-2}$, results in $\alpha_{fb} = 453.66 \ \mathrm{W} / \mathrm{m}^2 . K$

where, α_{fb} is the convection heat transfer coefficient[W/m²K]; ρ_G vapor specific mass at film temperature [kg/m³]; ρ_L Water specific mass at saturation temperature [kg/m³]; g aceleration of gravity [m/s²] c_{pg} steam specific heat [$J/kg.K$]; h_{LG} vaporization heat [J/kg]; h'_{LG} corrected heat [J/kg]; λ_G = steam thermal conductivity[$W/m.K$]; μ_G steam viscosity [N/s.m²]; T_W cylinder wall temperature [^oC], T_{SAT} Water Saturation temperature[^oC]; σ Water Superficial Tension [N/m]

6.4 Loss of coolant accident (LOCA)

An attempt is made to adopt a simple methodology to analyze and demonstrate the passive cooling characteristic of the FBNR in such a manner that be simple enough to be understood even by non-technical persons. The maximum temperature in the fuel chamber is calculated using code TRANSCAL v. 11 [9] developed in the Federal University of Santa Catarina, Brazil. It uses the finite volume method of solving the general two dimensional heat conduction equation with constant heat generation. It was first used to evaluate the sensitivity of the temperatures in the fuel chamber to various heat transfer parameters. The parameters used were thermal conductivity K of 18.3, density of 2424.5, specific heat of 840, water temperature of 100 °C, initial chamber temperature of 300 °C, and the heat generation of $8x10^5$ MWt/m³ corresponding to 200 seconds after accident. It was concluded that the density and specific heat only affects the time needed to reach the steady state condition. 1% increase in these values results in approximately 1% increase in cooling time. The higher the cooling water temperature, the higher is the maximum temperature. In these calculations, to be conservative, the maximum possible water temperature of 100 °C was assumed. The maximum fuel temperature was found to be proportional to the heat generation. However, the thermal conductivity was found to have the highest influence on the maximum temperatures. The higher the thermal conductivity K, resulted in lower the maximum temperature. As seen in Fig. 10, for K higher than 10, the variation in the temperature of the centre of the fuel chamber tube is small. The value of K used being 18, the maximum tube temperature at its center was found to be 378 °C. This is much below the design temperature even though a highly conservative assumption is made , that is considering a constant maximum heat generation.

6.5 Emergency core cooling system after a LOCA

In the case of a loss of coolant accident, the fuel elements fall into the fuel chamber under the force of gravity. The accumulator valves become activated allowing cooling water flow into the fuel chamber tank. The water will absorb the heat and turn into steam flowing out of the tank through the film boiling regime. The water needed to cool the fuel chamber as a function of time is shown in Fig.9.

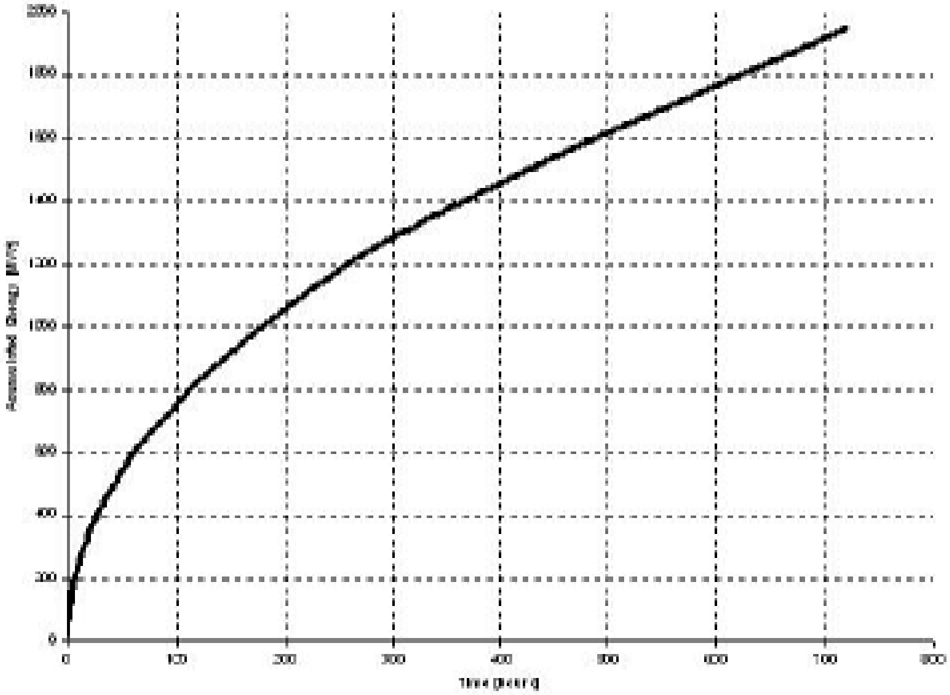

Figure 8 – Accumulated heat as a function of time

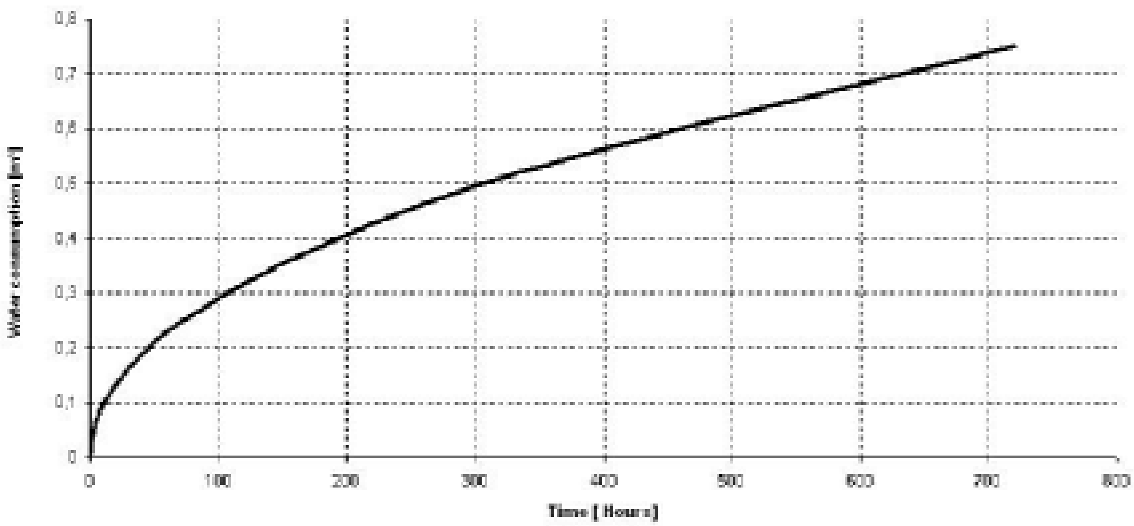

Figure 9 - Water consumption vs. Time

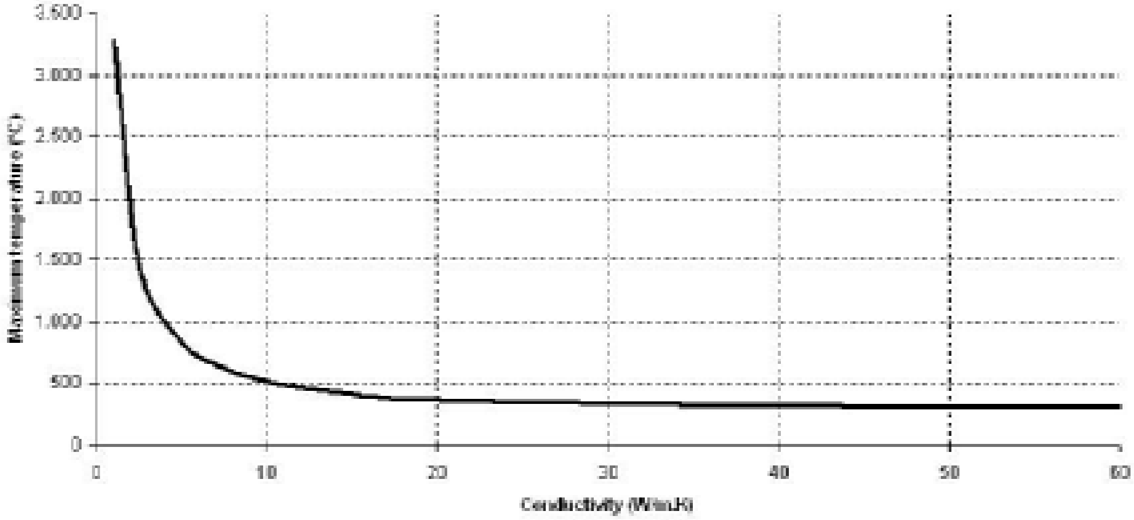

Fig. 9: Maximum fuel temperature as a function of thermal conductivity

6.6 Loss of flow accident (LOFA)

Should there occur a power cut to the pump or for any other unforeseen reason the coolant flow should stop, it is demonstrated that the safety of reactor is maintained.

In the first hypothesis, it is assumed that the pump is locked in such a manner that the flow is totally blocked in the pump. There is about 1.6 m^3 of water in between the fuel elements in the fuel chamber. It will absorb about 1046 MJ of heat to change into steam. This corresponds to more than the heat generation due to decay heat during 200 hours after incident.

This amount of water changed into steam, will increase the volume from 1.6 m^3 to 8.7 m^3 of steam. The increase in volume will force circulation in the loop, thus more water will enter in the fuel chamber and a natural circulation is established that will cool the fuel chamber without any problem.

In the real situation, the centrifugal pump can never be blocked up and the natural circulation in the loop occurs. Considering the 12 m^3 of water existing in the loop, the increase in the coolant temperature will be less than 1°C even several weeks after the incident.

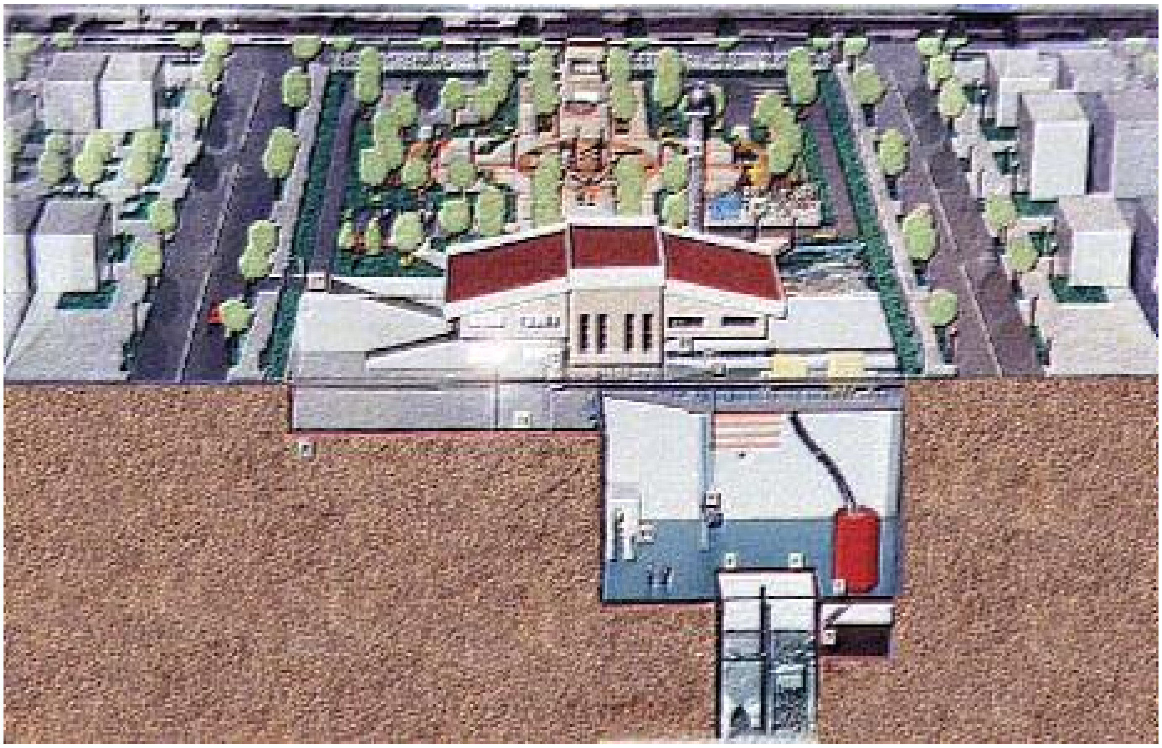

Fig.11: General View of FBNR Nuclear Power Plant with Underground Containment.

8. Acknowledgement

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