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A NEW ADVANCED SAFE NUCLEAR REACTOR CONCEPT

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Abstract. *The reactor design is based on fluidized bed concept and utilizes pressurized water reactor technology. The fuel is automatically removed from the reactor by gravity under any accident condition. The reactor demonstrates the characteristics of inherent safety and passive cooling. Here two options for modification to the original design are proposed in order to increase the stability and thermal efficiency of the reactor. A modified version of the reactor involves the choice of supercritical steam as the coolant to produce a plant thermal efficiency of about 40%. Another is to modify the shape of the reactor core to produce a non-fluctuating bed and consequently guarantee the dynamic stability of the reacor. The mixing of Tantalum in the fuel is also proposed as an additional inhibition to power excursion. The spent fuel pellets may not be considered nuclear waste since they are in the shape and size that can easily be used as a a radioactive source for food irradiation and industrial applications. The reactor can easily operate with any desired spectrum by varying the porosity in order to be a plutonium burner or utilize a thorium fuel cycle.*

Keywords: *Nuclear reactor, Fluidized bed, Inherent safety, Small reactor, Modular reactor.*

1. INRODUCTION

A small inherently safe nuclear reactor concept has been proposed (Sefidvash, 1985). This reactor is truely modular in design such that any size reactor, rather than the plant, can be constructed from the basic module. The reactor uses the light water reactor technology and promises to fulfill the objectives of design simplicity, inherent and passive safety, economy, standardization, shop fabrication, easy transportability and high availability. The inherent safety characteristic of the reactor dispenses with the need for containment; however, a simple underground containment is envisaged for the reactor in order to reduce any adverse visual impact. The purpose of the present research activities is to make a comprehensive feasibility study of the proposed reactor concept. Here a summary of such work is reported and new ideas for its development are presented.

2. DESCRIPTION OF THE REACTOR

A detailed description of the reactor is presented elsewhere (Sefidvash, 1985). Here a brief description of the main features of the reactor is given. The reactor is modular in design; therefore, any size of reactor can be made from the basic module. The total number of modules of the reactor is equal to [3N(N+1)+1], where N is the number of rings of modules surrounding the central module. The basic module has in its upper part the reactor core and a steam generator and in its lower part the fuel chamber. The core consists of a 25 cm diameter fluidizing tube in which, during reactor operation, the spherical fuel elements are fluidized. The fuel chamber is a 10 cm diameter tube which is directly connected underneath the fluidizing tube. A steam generator of the shell and tube type is integrated into the upper part of the module. A neutron absorber shell slides inside the fluidizing tube, acting similarly to a control rod, for the purposes of long term reactivity control.

The pump circulates the water coolant inside the module moving up through the fuel chamber, the core, and the steam generator and thereafter flows back down to the pump through the concentric annular passage. At the maximum or terminal fluidizing velocity, the coolant carries up the fuel elements into the core and fluidizes them . The increase in flow velocity causes higher porosity of the bed. In the shut down condition, the fuel elements leave the core and fall back into the fuel chamber by the force of gravity.

The 8 mm diameter spherical fuel elements are made of slightly enriched uranium dioxide, clad in by zircaloy for normal design, and stainless steel for modified design concept using supercritical steam. The cladding surface temperature limit in the modified design therefore is 450º C and fuel center temperature limit is 2000º C. Alternatively a ceramic cladding may be used in order to increase the cladding temperature limit.

The fresh fuel elements are fed into the reactor core from the top of the module. The spent fuel leaves the module through a valve provided at the bottom of the fuel chamber. The valve is operated by a hydraulic system allowing the spent fuel to be discharged from the fuel chamber into a permanently cooled storage tank. The module is provided with a pressurizer system to keep the pressure a constant, and a depressurizer valve which leads the steam to the condenser for reducing pressure to allow opening of the valve for refueling. A simple new concept of the pressurizer may be used in order to utilize the saturation pressure of the steam as the regulating factor.

Any hypothetical accident will cut-off power from the pump causing the fuel leave the core and fall back into the fuel chamber by its weight where remain in a highly subcritical and passively cooled condition. The fuel chambers are cooled by natural convection transferring heat to the surrounding air or water pool.

3. THE HYDRAULYCS

The reactivity of the reactor, the degree of the homogeneity of the core, and the heat transfer are all dependent on the porosity of the fluidized bed. The porosity, ε, is defined as the ratio of the moderator to total volume. Therefore, the moderator to fuel volume ratio is $\varepsilon/(1-\varepsilon)$. Consequently the study of the porosity of the bed as a function of different conditions of the flow is of great importance.

In attempt to observe the hydraulic behaviour of the fluidized bed nuclear reactor, a series of full size experiments were performed. The experimental system consisted of a 1,5 meter long 25 cm in diameter transparent plexiglas tube connected to a 3 meter long 10 cm diamter tube of the same material through a 10 cm high cone. A pump circulated water from a large tank into the tubes in a closed system. The flow rates were regulated by a valve and measured by flow meters. A variety of spherical lead and steel elements of 5 to 10 mm diameters simulating the fuel elements were fluidized in the system.

At low bed porosities, the fuel elements were observed to move around smoothly and apparently homogenous cores are obtained, but at higher porosities, the bed height oscillates. The higher the porosities, the higher the amplitudes of the oscillations were observed.

There is no commonly accepted criteria to define and distinguish homogeneous particulate fluidization from heterogeneous aggregate one. In general terms, the fluidization regime will be more homogeneous and particulate the lower the particle density to fluid density ratio, the lower bed porosity, the lower the bed height to bed dimeter ratio, and the smaller the particle diameter. The study of the bed porosity is of great importance and details are presented elsewhere (Sefidvash, 1995).

The relation between the fluidizing coolant velocity, u, and the fluidized bed porosity, ε, is given by $u = u_t\, \varepsilon^{2,4}$ (Richardson, 1954). The terminal velocity u_t for cold (20º C) and hot (320 ºC) reactor conditions are 129 and 162 cm/sec.

Experimental data demonstrate that the porosity velocity relationship for a liquid fluidized bed system is independent of both the total mass of particles and fluidizing tube diameter, if the tube to particle diameter exceeds 10 to 20 (Davidson, 1985). Here the ratio is more than 30. The porosity at the minimum fluidizing velocity was assumed to be 0,40. The minimum fluidizing velocity for the cold and hot reactor is 14 and 18 cm/sec. respectively.

The fluidized bed nuclear reactor may operate as a tight lattice reactor with the bed porosity is maintained low and below 0,50. In the porosity range of 0,40-0,50, the bed is observed to be homogeneous. This would be the desired condition to have a high conversion and plutonium burning reactor.

4. PHYSICS OF THE REACTOR

The steady state reactor physics calculations have been performed using the Leopard code (Westinghouse, 1983) for its cell calculation. Since the code has been developed for the analysis of light water reactors using cylindrical fuel rods, it was necessary to determine the dimensions of a hypothetical fuel rod lattice, which is neutronically equivalent to the spherical fuel element lattice. The calculations show that the reactivity increases initially with bed porosity as the neutrons become increasingly thermalized to a maximum and decreases thereafter at higher porosity where the neutron absorption in the moderator dominates the already well thermalized reactor condition. As the bed porosity is a function of pumped coolant velocity, it is apparent that the reactor can be controlled by fluidizing the bed through the variation of pump speed. It is an interesting inherently safe feature of this reactor concept that the reactivity automatically decreases should the pump either fail or overspeed. This is due to the slightly under moderated state of the operating reactor as operating porosity is a little lower than the porosity corresponding to the peak reactivity.

The effects contributing to negative reactivity, such as depletion and fission fragment buildup, can be compensated by a combination of increased fluidization and changing the absorber shell position. This will eliminate the need to use solid burnable poison in the fuel and mix boron in the coolant, thus resulting in better neutron economy.

5. DYNAMIC STABILITY

A study of the dynamic stability of the fluidized bed nuclear reactor under such circumstances was done (Vilhena,1988; Borges,1990,1994,1995; Streck,1988). The point-kinetic model was applied to study the short time transients and bifurcation theory was used to study the long time transients. In order to include the effect of the moving boundary, first the diffusion equation considering the balance of neutrons in a time dependent volume was obtained (Vilhena 1988; Borges, 1990). The equation shows dependency on the velocity and acceleration of the bed boundary, and obviously it reverts to the conventional diffusion equation when the dislocation velocity is zero. For the short time transients, the point -kinetic model was numerically solved by the Hansen and Taylor method, considering the velocity of dislocation of the boundary constant in time. A thermo-hydraulic model of this reactor, determined by a Euler explicit method, was related to the point-kinetic model, and a study of the power behavior in response to oscillations in porosity was performed (Streck, 1988).

To treat the long time transients, the temperature and absorption feedback effects as well as small variations in the velocity of dislocation of boundary were considered. The system of nonlinear equations obtained was dealt with by bifurcation theory to define the multiplicity of solutions. The numerical method of analytic continuation combined with the Newton method was used to obtain steady solution branches. The computational package based on the arch method (Kubicek, 1976) was used. The limit points for different values of bed velocity were computed while the rest of the constants remained the same. The results showed that the dynamical stability of this reactor concept has a behavior similar to the light water reactors (Borges, 1995).

6. HEAT TRANSFER

6.1 Steady State Condition

A detailed heat transfer analysis of the fuel elements has shown that due to high convective heat transfer coefficient and large heat transfer surface, the maximum power extracted from the reactor core is not limited to the material temperature limits, but to the maximum mass flow of the coolant corresponding to the desired operating porosity.

Assuming entering and exit coolant temperatures of 290º C, and 325º C, and making an energy balance, the thermal power production in Mwt may be calculated by the expression $P=14{,}5\ \varepsilon^{3,4}$. For operating porosities of 0,43 and 0,5 the power production is about 0,8 and 1,4 MWt per module which makes it a truely small reactor (Sefidvash, 1995).

The collapsed core height of 70 cm requires 145 kg of UO2 per module leading to a power density of about 100 MW/m^3 of fuel. The power per unit core volume of the reactor is 33,5 MW/m^3 compared to the 60 and 100 for BWR and PWR respectively. This power density can be increased by increasing the fuel enrichement and decreased by increasing the collapsed core height to be comparable to 3 and 6 MW/m^3 for modular and standard HTGR respectively. The reactor power increases slightly with burnup as the operating porosity increases to compensate for the loss of reactivity.

In treating the problem of heat removal from the zircaloy clad spherical fuel elements, the following assumptions are made: (1) There is no heat generation except in the fuel; i.e., none is generated in cladding or coolant. (2) The resistance to heat transfer in the contact areas between fuel and cladding is negligible. (3) The radial neutron distribution in the reactor is uniform; i.e., the volumetric heat generation source is radially constant everywhere at specific height of the reactor. (4) The thermal and physical properties of the fuel and coolant can be adequately approximated to a three term polynomials.

As heat is transferred from the fuel elements in the core to the coolant, the temperature of the coolant varies from a minimum at the inlet to a maximum at the exit of the core. The coolant bulk temperature at any axial position can be determined by equating the energy added by heat transfer to the enthalpy rise of the coolant. The height of the reactor is divided into 100 divisions and the average temperature in every section is calculated. The coolant temperature rise from a lower axial section to a higher one is evaluated by assuming a cosine axial neutron flux distribution. Since the bed is suspended in the coolant, the pressure drop in the core is considered to be equal to the buoyant weight of the fluidized bed.

Two important checks on coolant conditions were made to ensure that the reactor is adequately cooled. First, it is determined whether the bulk fluid is indeed subcooled. As the coolant flows through the reactor, its enthalpy is increased due to heat transfer and its pressure is decreased as a result of flow losses. Both of these effects reduce the subcooling of the coolant. To ensure bulk subcooling, the local bulk temperature must be less than the saturation temperature at the local pressure. The critical location is at the core exit. Secondly, it was determined whether there is any local boiling near surface of the fuel element. The criterion is whether the maximum coolant temperature at any cross section is greater than the local saturation temperature.

The results under exaggerated operating conditions showed a maximum difference in fuel and clad surface temperature of 5º C. The temperature drop from clad surface to coolant varies between 2º C at the bottom and 5º C at the top of the reactor. The maximum fuel center and clad temperatures of less than 400º C are far below the reactor safety limits (Sefidvash, 1982). Thus due to a high convective heat transfer coefficient and large heat transfer surface, the maximum power extracted from the reactor core is not limited to the material temperature limits, but to the mass flow of the coolant which corresponds to the desired operating porosity.

6.2 Transient Conditions

The heat transfer from the fuel elements during operation but under transient conditions was studied by two methods of the lumped capacity technique and the exact one. A thermal analysis of the fuel chamber under transient condition is made assuming that after a hypothetical loss of coolant accident (LOCA), all of the water vaporizes and the fuel elements fall into the fuel chamber in a dry condition. The decay heat is transferred to the chamber walls by conduction and radiation, and the chambers are cooled by natural convection. The results under adverse heat transfer conditions show that the integrity of the fuel and fuel chamber are maintained.(Vilhena, 1990; Sefidvash, 1987).

As a preliminary analysis of the 8 mm diameter spherical fuel elements, under LOCA condition, the lumped-capacity technique was used. It is assumed that the internal conductive resistance of the fuel element is small compared to the external resistance to heat transfer to the surrounding steam. For the preliminary calculations, the method is expected to be adequate since under accident conditions the vapor is assumed to surround the fuel element and the Biot number to be very small namely less than 0,1.

The transients of any nature are assumed to occur when the reactor is in normal operating condition, when the fuel and coolant temperatures are at 320º C and 310º C respectively. The fuel temperature rise assuming the burnout condition of $h=1000$ W/m^2 ºK and the extreme situation of fuel element finding itself in a minimum heat transfer condition, i.e., conduction through static vapor, such that $h=5$ W/m^2 ºK. It is found that even under such improbable adverse conditions, the fuel pellet must stay under such situations for too long of a time in order to become damaged.(Sefidvash, 1995). Therefore, the transients will not cause sufficient increase in temperatures to damage the fuel elements.

An attempt is also made to perform an exact calculation (Vilhena, 1990). Therefore, in order to calculate the fuel, cladding, and chamber wall temperatures heated by decay heat after a probable accident, a multilayer heat conduction equation was solved. An approximate closed-form solution by Laplace transform with Gaussian quadrature inversion technique was found. The method of solution was tested for slab geometry, considering three regions. The results were shown to be accurate within five significant digits .

In the analysis of the condition of fuel in the fuel chamber, after a loss-of-coolant accident, it is assumed that the fuel element immediately loses contact with the coolant and becomes surrounded by steam. This causes reduction of its convecting heat transfer coefficient for which values of 1000, and 100 W/m^2 ºK. are assumed. The decay heat is supposed only to be transported to the fuel chamber tube and moreover, only by convection, making the conservative assumption that the contact area between the spheres and the tube is small. The fuel chamber is 10 cm in diameter having a 1 cm thick wall. The results show that the fuel temperature at the center of the tube rise to a maximum of 890ºC and decreases thereafter. This demonstrates the passive cooling characteristics of this nuclear reactor concept.

The fuel elements during operation and under transient conditions will maintain their integrity. A thermal analysis of the fuel chamber under transient condition even assuming that after a hypothetical loss of coolant accident, all of the water vaporizes and the fuel elements fall into the fuel chamber in a dry condition, shows that the reactor is passively cooled.

7. NUCLEAR FUEL

The sherical fuel elements are fabricated either from compacting the UO2 powder or grinding the cylindrical fuel pellets to spherical shape. A successful technique for fuel clad fabrication has been developed (Schaeffer,1986). It involves producing hemispheres by pressing the cladding sheets, then the two hemispheres containing the fuel pellet are brought into contact and pressed together under a defined temperature and pressure. Consequently the two hemispheres containing the fuel pellet weld together perfectly. The feasibility of leak testing the fuel elements using helium gas detection has been demonstrated even though that the quantities of the helium involved are relatively small.(Sefidvash, 1987).

Another possibility is to fabricate the fuel elements from UO2 microsheres which are used in High Temperature Gas Cooled Reactors. The particles of about 1mm in diameter are made of UO2 microspheres coated with 4 layers of porous pyrolytic carbon (PYC) called buffer layer, inner dense PYC, silicon carbide (SiC), and outer dense PYC. The densities are 10,63, 1,10, 1,85, 3,20, and 1,85 respectively. The fuel elements can be fabricated by compacting the microsheres in a spherical matrix of porous pyrolytic carbon and then clad it by dense silicon carbide. Such a fuel design will make a high burnup reactor possible.

In order to create additional inhibition to power excursion, the isoptopes of Lu^{175} /Lu^{176} or Ta^{181}/Ta^{182} may be added to the fuel. Harms, et.al (Harms , 1991) argue that introducing a large Doppler effect isotope such as tantalum,Ta^{181} , into the fuel, in the case of a power excursion, it absorbs much neutrons producing Ta^{182} which in turn having a large neutron capture cross section, absorbs still more neutrons. This tandem effect must further be studied.

In this reactor concept, the small fuel elements are in such convenient form and size that they may be utilized as a source of radiation without the necessity of reprocessing or manipulations. They can be used for irradiatin of food and agricultural products and the numerous applications in industry.

8. EVALUATIONS OF THE REACTOR

The proposed reactor design has been analyzed and evaluated by numerous scientists and scientific forums internationally appreciating the claims made . Some of the known 'published' reviews are as follows: Two full articles published by the Russian scientists (Artamkin, 1986; Legchilin,1987) dedicated to the analysis of the proposed reactor. Also the Japanese scientists (Mizuno,1987) reviewed this along with other new nuclear reactor concepts . In 1989 Oak Ridge National Laboratory prepared a report for the US Department of Energy in which W.J. Reich (Forsberg,1989) reviewed this reactor concept . Later on

Mizuno, et.al. (Mizuno, 1990), took the proposed reactor concept a step further and proposed "The Inherently-Safe fluidized-bed Boiling Water Reactor Concept" which they describe to be "The combination of fluidized-bed concept and the density-lock mechanism of PIUS". Edlund proposed the fluidized-bed nuclear reactor as a potential for a high burn-up reactor concept (Edlund, 1991).

The nuclear community is invited to do further evaluation of this reactor concept in order that their critical analysis help further development of the concept. There is also a need for the definition of an appropriate safety case for this reactor concept.

9. MODIFICATION OF THE DESIGN

9.1 Conical Shape

A simple solution is found to eliminate the oscillations in the fluidized bed nuclear reactor. As the porosity is a function of coolant flow velocity, it is proposed to construct the fluidizing tube in a slighltly conical shape. In this way the coolant velocity as well as porosity decreases along the height of the reactor. The slow and continuous reduction of porosity has a compacting effect on the bed not allowing the oscillations to set in. The bed will have a porosity distribution along the height.

The porosity along the height $\varepsilon(z)$ can be calculated by

$$(1-\varepsilon) / (1-\varepsilon_0) = f / Z \tag{1}$$

Where $Z = z / H_0$ and H_0 is the collapsed height corresponding to its porosity ε_0 .

$$f = [(3a^2/H_0^2)+(3a \tan\alpha /H_0)+\tan^2\alpha]/[(3a^2/H_0^2+3Z\, a \tan\alpha / H_0 + Z^2 \tan 2\, \alpha)] \tag{2}$$

Where a, is the cone's base radius, α, the small vertical cone angle, and z, the fluidized bed height.

In the case of a cylindrical tube, the factor, f, becomes unity and the expression relating porosity to fluidized bed height simplifies to

$$z/H_0 = (1-\varepsilon) / (1-\varepsilon_0) \tag{3}$$

The moderator volume to fuel ratio along the height, z, of the fluidized bed may be calculated by

$$V_m/V_f = \varepsilon(z) / [1-\varepsilon(z)] \tag{4}$$

At the collapsed condition where porosity is about 0,4, the moderator to fuel volume ratio is 0,67. As any porosity can easily be obtained by manipulating the coolant flow rate, the reactor may operate with any desired spectrum. This reactor can serve as a plutonium burner for consuming the materials coming from the desmantling of the nuclear weapons. Also the reactor can easily operate with thorium fuel cycle which is of interst to Brazil since the country is a great repository of thorium.

9.2 Supercritical Steam

The concept of a direct cycle reactor operating at supercritical pressure is attractive for improving the thermal efficiency drastically to enhance the resulting environmental protection. The reactor combines the fluidized bed concept with the idea of using direct cycle reactor operating at supercritical pressure proposed by Oka (Oka, 1995; Keyfitz,1964). The supercritical steam is used as the reactor coolant. The critical pressure of water is 221 bar. When the reactor operates at 250 bar, the supercritical water does not exhibit a change in phase and the concept of boiling does not exist.. The water density decreases continuously with temperature.

The coolant entering temperature , on the lower part of the bed, is 310 ºC and the exit temperature, on the upper part of the bed, is 416 ºC. Therefore, the water density decrease continuously from 0,725 to 0,137 g/cm^3 along the bed. This is an important factor in causing the fluidized bed to become a non-fluctuating bed and have a stable reactor as the upper part porosities are lower than the lower ones due to the fact that the bed porosity is a function of fluid density. The recommended pressure of 250 bars is due to the smooth and mild variation of density with pressure in this region resulting in stability of flow in the core. The power production is much higher in this modified concept as the difference in inlet and outlet enthalpy is much higher than a simple pressured or even boiling reactor. The plant thermal efficiency is estimated to be about 40% which is about 20% higher than the conventional PWR's. The turbines will be smaller compared with the light water reactors by adopting the supercritical water as the coolant. The superheated steam is fed directly into the turbine. The steam-water separation is not needed for direct cycle reactor. Some other advantages of such a choice besides the high thermal efficiency, will have smaller turbine, no steam generators, and reduced waste heat.(Oka, 1995, Sefidvash,1995)

10. CONCLUSIONS

This reactor concept is demonstrating its potential as a simple design using pressurized water reactor technology obtaining the desired characteristics of inherent and passive safety, integral plant having once-through type steam generator inside the module, controllable neutron spectrum, tight lattice, no soluble or burnable poison leading to increased neutron economy and reduced activity level, on load refueling, flexible fuel cycle choice, modular, shop fabrication, possibility of eliminating operators, ease of decommissioning tasks, underground containment, and a host of other positive features including the possibility of using its spent fuel as a source of radiation for food irradiation, and industrial applications without the need for reprocessing.

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