Design Study of Small Lead-Cooled Fast Reactor Cores Using SiC Cladding and Structure*

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Abstract

Neutronics of a reactor core with SiC cladding and structure was compared with that with steel cladding and structure analytically for small lead-cooled fast reactors. Uranium nitride fuel was used for this reactor. U\textsuperscript{235} enrichment was 11% in inner core and 13% in outer core for relatively flat neutron flux distributions and power density distribution. The core design was optimized using natural uranium blanket and nitride fuel for long life-core with reshuffling interval of 15 years. The analytical result indicated that neutron energy spectrum was slightly softer in the core with the SiC cladding and structure than that with steel cladding and structure. The SiC type reactor was designed to have criticality at the beginning of cycle (BOC), although the steel type reactor could not have criticality with the same size and geometry. In other words, the SiC type core can be designed smaller than the steel type core. The peak power densities could remain constant over the reactor operation. The consumption capability of uranium was quite high, i.e. 10% for 125 MWt reactor and 18.4% for 375 MWt reactor at the end of cycle (EOC).

Key words: Fast Reactor, SiC, Lead, Uranium Nitride, Neutron Energy Spectrum

1. Introduction

Small reactors have been proposed for installation in islands remote from electric power stations. Small reactors also have various advantages in safety and potential of future energy utilization in many new areas as reported by many researchers\(^1\). Lead-cooled small fast reactors have characteristics of proliferation-resistance, sustainability, safety, reliability, and economics. Lead coolant has several advantages such as low absorption cross sections of fast neutrons and no vigorous interaction with air and water/steam.

It has been reported that Russia has a lot of experience in installation and operation of lead alloy-cooled reactors, i.e., two prototype reactors and seven reactors using lead-bismuth coolant for “Project 645” and “Alpha” submarines\(^2\textsuperscript{-4}\). Lead-cooled reactors, i.e. BREST-300, 600 and 1200, have been designed in Russia. It has been reported that lead has advantages of higher availability, lower price and lower amount of induced polonium activity than lead-bismuth\(^5\).

Lead has a melting point of 327°C and boiling point of 1737°C. The operational temperature of a reactor with lead coolant is between 400°C and 600°C. Therefore, the temperature margin from the operation region to the boiling point is quite large (>1000°C),
which gives good level of safety. Because of the high boiling point of the coolant, the lead-cooled reactors have a potential of high coolant outlet temperature for high reactor efficiency.

One of the main issues of these types of reactors is the compatibility of core and structural materials with the coolant at high temperature for the development of lead alloy-cooled small fast reactors. Experimental studies on the compatibility of materials with lead alloy coolant have been performed by many researchers (6-8). Until now, various types of steels including surface-treated and coated ones have been chosen for candidates of core and structural materials. However, lead alloy can affect the reactor materials at high temperature because corrosion rate increases significantly with temperature.

Ceramics materials may be promising material for the high temperature reactors, since they are compatible with high temperature lead-alloys. Ceramics materials that have been proposed for a candidate material of high temperature nuclear reactors are silicon carbide (SiC) and its composites. They are expected to be used as cladding and structural materials for high temperature fission reactors because of the superior high-temperature properties, thermo-chemical stability, irradiation tolerance, inherent low activation and low after-heat-properties. Several researchers have reported good corrosion resistance of ceramics with lead alloys (9, 10) and investigation on fabrication of SiC/SiC composites for the candidates of the fuel cladding material (11).

However, reactor core with ceramics materials have not been studied from neutronics point of view so far. In the present study, the neutronic characteristics of a small lead-cooled fast reactor using SiC as cladding and structural materials are investigated analytically by comparing them with those of the same type of reactor using austenitic stainless steel SS316 as the cladding and structural materials. The neutronic characteristics considered are the neutron energy spectrum, the criticality, the neutron flux distribution, the power density distribution, the peak power density and the burn up capability of uranium.

2. Conceptual Design

Typical types of lead alloy-cooled small fast reactors are chosen for the present study (11). The use of plutonium as a common fissile fuel for fast reactors may be difficult for some developing countries because of nonproliferation consideration. Enriched uranium may be better than plutonium as the fuel material of small reactors there (12). A neutronics calculation was performed for core design of lead alloy-cooled fast reactor with SiC cladding and structural materials. The power of the reactors under study was 125 MWt and 375 MWt. General characteristics of the reactors under study are shown in Table 1. The fuel of this reactor was uranium nitride with U^{235} enrichment of 11% (in inner core) and 13% (in outer core). The core using nitride fuel has a lot of good safety characteristics, i.e. a high melting point of approximately 2500°C and high thermal conductivity. The blanket fuel is made of natural uranium nitride. Since the coolant outlet temperature is high, this reactor has high thermal efficiency of about 40%. In this case, the electric powers of the reactors with the thermal powers of 125 MWt and 375 MWt are 50 MWe and 150 MWe, respectively. The reactors were designed for long time operation for 15 years without refueling. The use of natural uranium in blanket was optimized to have the long life fuel cycle.

The configuration of the core is shown in Fig. 1. This reactor has a central blanket for an increase in internal conversion ratio so that the initial excess reactivity can be minimized for the long time operation. The enrichment of the outer core was higher than that of the inner core so that the power density distribution is made flatter in the reactor.

The volume fractions of the fuel, the structure and the coolant in the reactor are shown in Table 2. The volume fractions of the fuel, the structure and the coolant in the blanket and the cores were 45%, 15% and 40%, respectively, which is one of typical cases of the LMFBR cores (13). In the gas plenum, the space except for the structure and the coolant was
gas, which was dealt with as empty. It was assumed that all control rods were out of the core.

Table 1 General characteristic of Pb-cooled reactors

<table>
<thead>
<tr>
<th>Characteristics</th>
<th>Specifications</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal power (MWt)</td>
<td>125, 375</td>
</tr>
<tr>
<td>Electrical power (MWe)</td>
<td>50, 150</td>
</tr>
<tr>
<td>Refueling period (yrs)</td>
<td>15</td>
</tr>
<tr>
<td>Fuel</td>
<td>Uranium Nitride (UN)</td>
</tr>
<tr>
<td>Coolant</td>
<td>Pb</td>
</tr>
<tr>
<td>Cladding and Structure</td>
<td>SiC or SS316</td>
</tr>
<tr>
<td>Enrichment in (U^{235})</td>
<td>11% and 13%</td>
</tr>
<tr>
<td>Linear heat rate (W/cm)</td>
<td>100 (125 MWt)</td>
</tr>
<tr>
<td></td>
<td>200 (375 MWt)</td>
</tr>
<tr>
<td>Inner diameter of fuel cladding (mm)</td>
<td>12.0</td>
</tr>
<tr>
<td>Outer diameter of fuel cladding (mm)</td>
<td>13.86</td>
</tr>
<tr>
<td>Pitch diameter (mm)</td>
<td>17.89</td>
</tr>
<tr>
<td>Outer diameter of active core (cm): Blanket/inner-core/outer-core (125 MWt)</td>
<td>40/106/208</td>
</tr>
<tr>
<td></td>
<td>Blanket/inner-core/outer-core (375 MWt)</td>
</tr>
<tr>
<td>Maximum length of active core (cm): Blanket/inner core/outer core (125 MWt)</td>
<td>20/70/100</td>
</tr>
<tr>
<td></td>
<td>Blanket/inner core/outer core (375 MWt)</td>
</tr>
<tr>
<td>Width of shielding (cm)</td>
<td>50</td>
</tr>
</tbody>
</table>

![Fig.1 Core configuration of reactor (r-z)](image)

Table 2 Volume fractions of fuel, structure and coolant

<table>
<thead>
<tr>
<th>Region</th>
<th>Blanket</th>
<th>Inner core</th>
<th>Outer core</th>
<th>Gas plenum</th>
<th>Shielding</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel</td>
<td>Natural UN</td>
<td>11% (U^{235})N</td>
<td>13% (U^{235})N</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Volume fractions</td>
<td>Fuel, UN</td>
<td>45%</td>
<td>45%</td>
<td>45%</td>
<td>0%</td>
</tr>
<tr>
<td></td>
<td>Structure, SiC or SS316</td>
<td>15%</td>
<td>15%</td>
<td>15%</td>
<td>15%</td>
</tr>
<tr>
<td></td>
<td>Coolant, Pb</td>
<td>40%</td>
<td>40%</td>
<td>40%</td>
<td>40%</td>
</tr>
</tbody>
</table>
It was assumed that the configuration was two-dimensional in the cylindrical coordinates of $r$ and $z$. The calculation was done for the quarter of the reactor since the reactor was symmetrical axially and vertically. SRAC (14) neutronics computer code was used for the calculation. Macroscopic cross sections for 74 condensed fast energy groups were calculated with the collision probability method (PIJ) from nuclear library JENDL-3.3 (14) with 74 fast energy groups. Thermal calculation was not performed in this analysis. The neutronic properties of the materials were obtained assuming that the temperatures of the fuel, the cladding, and the coolant were 610°C, 500°C and 500°C, respectively, and the plenum and shielding temperatures were assumed to be 400°C.

Effective multiplication factor and neutron flux distributions were obtained by CITATION module with diffusion method. The core physics analysis has been carried out for neutron energy spectrum, criticality, neutron flux distribution, burn-up capability of uranium, power distribution and peak power density.

3. Results and Discussion

3.1 Neutron Energy Spectrum and Criticality

The spectrum of neutron energy has a large influence on the neutronic behavior. The distributions of neutron energy spectra in the cell of inner core of the SiC and steel SS316 types of reactors obtained with the collision probability method (PIJ) are shown in Fig. 2. It is found that the neutron energy spectra of reactor using the SiC was not much different from that using the SS316, but it was slightly softer than that using the SS316. This was because of higher moderation property of SiC than the steel. The distributions of neutron energy spectra in the blanket and outer core were quite similar to those in the inner core.

![Fig. 2 Neutron energy spectra in the cell of inner core of SiC and steel SS316 type reactors at initial condition obtained with the collision probability method (PIJ)](image)

Fig. 3 shows the effective multiplication factor during operation for the reactors using SiC as cladding and structural material with thermal powers of 125 MWt and 375 MWt. It is found that the criticality of the reactors could be maintained for 15 years without refueling or reshuffling of fuel. In the case of the reactor with the power of 125 MWt, the initial excess reactivity was less than 0.1% at the beginning of a cycle (BOC) and excess reactivity reached its maximum values at around 6 years and then decreased to about 0.05% at the end of the cycle (EOC). In the case of the reactor with the power of 375 MWt, the initial excess reactivity was less than 0.09% at BOC and excess reactivity reached its maximum values at around 6 years and then decreased to about 0.03% at the EOC. It is an important result that U$^{235}$ is feasible as initial fissile fuel of small lead-cooled fast reactor for proliferation resistance. For the practical reactor, it may be necessary to design the core to have a little more excess reactivity for start up of the reactor at the BOC.
3.2 Neutron Flux Distribution

The neutron flux distributions in the radial direction of 125 MWt and 375 MWt at the middle of the cycle (MOC) after 8 year-operation are shown in Figs. 4 and 5, respectively. Neutron flux in lower energy range is lower than that in high energy range for both of the 125 MWt power reactor and the 375 MWt power reactor. However, the neutron flux in lower energy range was still higher in the core with SiC cladding and structure than in the core with steel cladding and structure. Fast neutrons with high energy are major for fission reaction than neutrons with lower energy in fast reactors. Nevertheless, the neutron capture cross section of SiC that is lower than steel has more influence on the criticality of the reactor than the neutron flux in lower energy region. It is also found in Figs. 4 and 5 that neutron flux distributions in the both energy ranges were made relatively flatter by using two core regions with different enrichment, which suggests that the power distributions of these reactors are also flat.

3.3 Power Density

The power density distributions of the 125 MWt and 375 MWt power reactors using SiC after 8 year-operation (MOC) are shown in Fig. 6. The discontinuity between the inner and outer cores is a direct consequence of the higher fissile fuel content in the outer core region. It is found that the power distribution was made flatter because the outer core enrichment was higher than inner core.

Figure 7 shows the change in peak power densities which depend on the power level and the configuration of the reactor. In the case of 125 MWt power reactor, the peak power density was initially 108.7 W/cc and finally 112.2 W/cc at the EOC. In the case of 375 MWt power reactor, the peak power density was initially 236.6 W/cc and finally 274.4 W/cc at the EOC. In both reactors, the profiles of peak power density were almost constant during the operation. This indicates the stability and controllability of operation of the reactors and also indicates the safety of the reactors. It was found that the power density of the 125 MWt power reactor was lower than that of the 375 MWt power reactor because the blanket size has to be designed large for long life core. The merit of the 125 MWt power reactor with such low power density is the feature of the long life core.
Fig. 4 Neutron flux distribution of SiC type reactor with 125 MWt at 8 year-operation (radial) at the half level of core height

Fig. 5 Neutron flux distribution of SiC type reactor with 375 MWt at 8 year-operation (radial) at the half level of core height

Fig. 6 Power density of 125 MWt and 375 MWt of SiC type reactors after 8 year-operation (radial) at the half level of core height
3.4 Burnup of Uranium

In this reactor, initial fissile nuclide in the fuel is only uranium-235. Except convert to U236, fission reactions convert U235 to fission products. Absorption of neutrons converts U238 to Pu239, Pu240, Pu241, and Pu242, and then fission reactions of Pu239 and Pu241 produce smaller nuclides. The amount of consumption of U235 and U238 during the operation is equal to the sum of the amount of U235 that burned and the amount of U238 converted to Pu239 which burned during the operation. The burnup is defined by

$$\text{burnup of uranium (at. %)} = \frac{(U_{\text{235}} + U_{\text{238}})_{\text{ended}} - (U_{\text{235}} + U_{\text{238}})_{\text{ended-operation}}}{(U_{\text{235}} + U_{\text{238}})_{\text{ended-operation}}} \times 100\%$$

The burnup of the uranium in the 125 MWt and 375 MWt reactors is shown in Figs. 8 and 9, respectively. In the case of the 125 MWt power reactor, the consumption of uranium, or burnup, was 10.3% in the inner core, 9.8% in the outer core, and 13.3% in the blanket at the end of the cycle (EOC). The total consumption of uranium was 9.9% at EOC. In the case of the 375 MWt power reactor, the consumption of uranium was 18.7% in the inner core, 18.2% in the outer core, and 20.5% in the blanket at EOC. The total consumption of uranium was 18.4% at EOC.
These results showed that burnup of uranium of these reactors during operation were quite high. It is also found that U\(^{235}\) can be used in place of plutonium as a fissile fuel of the present small reactors.

It was found that the burnup of the 125 MWt power reactor was lower than that of the 375 MWt power reactor because of low power density. The merit of the 125 MWt power reactor with such low burnup is the feature of the long life core.

4. Conclusions

A lead-cooled small fast reactor with SiC as cladding and structure has good neutronic performance, i.e.,
1. A long life core can be designed for 15 year-operation without refueling or reshuffling of fuel.
2. The spectrum of the reactor using SiC is slightly softer than that using steel, although it is not much different from that using steel.
3. The core can be designed to have relatively flat neutron flux distribution and power density with nearly constant peak power densities over the reactor operating period.
4. Burnup of uranium of the reactors is evaluated from the consumption of uranium. The total consumption of uranium is 10% and 18.4% at EOC in 125 MWt and 375 MWt power reactors, respectively. The uranium consumption capability is quite high for the lead-cooled fast reactor using SiC cladding and structure.

References


(14) Okumura, K., Kugo, K., Kaneko, K., Tsuchihashi, K., SRAC (ver. 2002); the comprehensive neutronics calculation code system, to be published in JAERI-Data/Code, (2002).