COOLANT OPTIONS FOR A WATER-COOLED REACTOR WITH A HARD NEUTRON SPECTRUM

H. Takahashi, J. Zhang, D. Cokinos, U. Rohatgi
Brookhaven National Laboratory
Upton, New York.11973
takahash@bnl.gov ; zhang@bnl.gov, cokinos@bnl.gov ; rohatgi@bnl.gov

T. Downar
Purdue University
School of Nuclear Engineering
West Lafayette, IN 47907-1290

ABSTRACT

A thorium-plutonium-fueled, water-cooled reactor with hard neutron spectrum has been analyzed by using the Monte Carlo (MC) burn-up code MCBURN developed at BNL. Preliminary analyses have shown that thorium fertile fuel has better neutronics characteristics than U-238. Reasonable computer time is required to analyze, by the MC method, a whole reactor consisting of many lattice cells. It is demonstrated that the MCBURN code can be a useful tool for a reactor with a complicated geometry. The application of the MC method to RELAP and PARCS code systems to analyze the kinetic behavior in the case of an accident is discussed.

1. INTRODUCTION

A proliferation-resistant, hexagonal, tight-lattice BWR fuel core is proposed as part of the Nuclear Energy Research Initiative (NERI) program [1]. Basic objectives of this program include non-proliferation, advancement of nuclear technology, and the maintenance of the computational tools and scientific and engineering talent required for the future of nuclear engineering. Our objective is to advance the well-established technology of the water-cooled reactor to develop a design for a proliferation resistant, economically competitive, high burnup boiling water reactor (HBBWR), fueled with fissile plutonium and fertile thorium-oxide fuel elements. The proposed HBBWR has a very tight lattice with a relatively small water-volume fraction, and a fast-reactor neutron spectrum. Design goals are to achieve a high conversion of thorium to U-233, and reduce the accumulated national inventory of plutonium while producing useful energy. Through a combination of high concentration of plutonium and a large rate of production of U-233, the tight-lattice core can achieve high fuel burn-up. The consequent reduction in the required fuel reshuffling will increase the plant's operating factor and lower the cost of electricity.

2. CORE CONFIGURATION

To safely operate a light water cooled reactor with a hard neutron energy spectrum [2.3,4], prone to positive void coefficient, the incorporation of a neutron-leaky core, similar to that in the Na-cooled fast reactor, was proposed. One such type of reactor, has a very flat core. Another type consists of fuel assemblies with and without void channels through which neutrons can stream out when there is extensive boiling of the coolant. Thus, the increase in reactivity due to hardening of the neutron-energy spectrum can be reduced by neutron leakage. To get a high powered LWR, the pancake-type flat core must have a large diameter relative to its height. This requirement results in the need for a thick pressure vessel with a large radius.

Neutronic characteristics for various fuel and coolant options for a hexagonal tight lattice using light water as coolant and their corresponding initial k-inf values and neutron energy spectra have been calculated by the Monte Carlo method. Tables 1A and 1B list the fuel rod assembly characteristics and neutronic characteristics, respectively. Table 1B shows that the nitride fuel has higher multiplication rates than the oxide fuel due to its higher density.

TABLE IA
FUEL ROD ASSEMBLY GEOMETRY - HEXAGONAL LATTICE

Fissile and fertile fuel	PuN-ThN	PuO2-ThO2
Fissile enrichment (cm)	10%	10% and 20%
Fuel rod diameter(cm)	0.7068	0.484
Gap (cm)	0.02	0.02
Clad (cm)	0.046	0.046
Pitch (cm)	0.572	1.1945
Hexagonal can width (cm)	0.3	0.3
Distance between rod and can(cm)	0.061	0.061
No. of fuel rods in assembly	61	231
Density of H20 (gr/cc)	0., 0.3, 1.0, 3.0	0., 0.3,

TABLE 1B
NEUTRONICS CHARACTERISTICS OF PuN-ThN, PuN-UN, PuO2-ThO2 FUEL ASSEMBLIES

Case	Fuel Fissile, Fertile, fissile %	Coolant density (gr/cc)	k-inf	Number weighted energy	Flux weighted energy	Capture in Th or U	Fission in Th or U
1	PuN,ThN 10%	H20 1.0	1.4611	2.2720e-2	9.2262e-1	8.2369e-2	3.053e-3
2	PuN,ThN,10%	H20 0.3	1.6679	5.6494e-2	7.8791e-1	1.0322e-1	4.856e-3
3	PuN,UN,10%	H20 0.3	1.2583	3.0613e-2	7.245e-1	2.2430e-1	3.8939e-2
4	PuN, ThN,10%	H20 0.0	2.1489	2.6506e-1	6.1005e-1	9.7822e-2	6.7340e-3
5	PuN,ThN,10%	H20 3.0	1.3398	9.9413e-3	9.9788e-1	5.7222e-2	1.4736e-3
6	PuO2,ThO2,10%	H20 0.3	0.9310	1.7809e-2	6.7822e-1	3.0573e-1	1.1077e-2
7	PuO2,ThO2,20%	H20 0.3	1.2067	2.9658e-2	7.3128e-1	2.3174e-1	9.8173e-3
8	PuO2,ThO2,20%	D20 .33	1.1451	6.1732e-2	4.6313e-1	3.4269e-1	9.9600e-3
9	PuO2,ThO2,20%	H20 0.0	1.3446	1.4584e-1	4.7803e-1	3.3497e-1	1.3365e-2

3. D2O COOLED REACTOR

A tight lattice limits the ability to remove heat, thus jeopardizing operation. A hard neutron spectrum can also be achieved by using D2O coolant that has a lower slowing-down power for neutrons than H20 coolant. Thus, a higher amount of heavy water coolant than light water can be used for heat removal.

To remove heat from the tight lattice, a high pressure pump is needed and this arrangement might incur difficulties in removing heat during an accident. A D2O-cooled pressurized water reactor might possibly offer a harder neutron-energy spectrum without using a tight lattice.

Figure 1 shows the neutron energy spectra for the fuel assemblies listed as cases 1 through 9 in Table 1B. We note that the neutron flux of the D20-cooled reactor is negligibly small in the low-energy region of 20 eV. By contrast, the neutron flux in the H2O-cooled reactor in this energy range is still substantial.

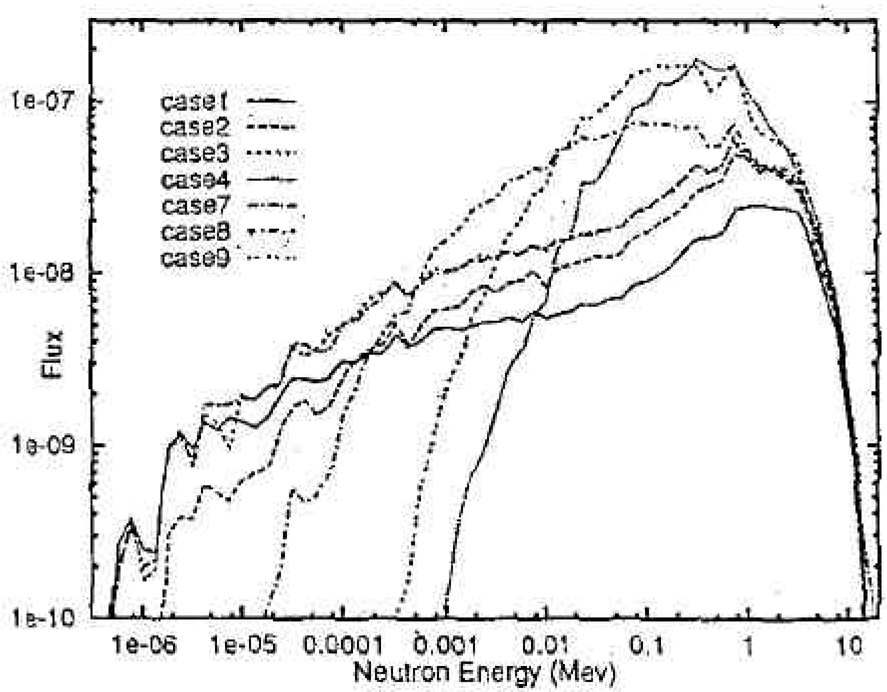

Figure 1. Neutron Energy Spectra of Various Fuel Assemblies

4. CORE BURNUP WITHOUT VOID CHANNEL

Table 2 shows the change of the multiplication factor as a function of burnup for the reactor whose dimensions are tabulated in the second type core without void channel (Table 3). This figure shows that the thorium fertile fuel needs a higher Pu-239 fissile content than the Pu-239 content in the U-238 fertile fuel due to the larger neutron capture cross-section. In the burnup region from 0 to ~50 MWD/ton, the reactivity increases with burnup. After this burn-up level, k-inf decreases rather rapidly. In the case of theU238 fertile material, k-inf decreases monotonically as the burn-up increases. Neutron capture by control rods is required for criticality control. This requirement, however, worsens the burn up characteristics of the U-238 fertile fuel.

TABLE 2
K-INF AS FUNCTION OF THE BURN UP FOR THE REACTOR

Burn Up in Units of 30MWD/ton	qqq11 U238 O2 H20 0.3 gr/cc (1000, 50)*	Qqq16 U-238 O2 H20 0.3 gr/cc (200,35)*	qqq14 Th O2 H20 0.3 gr/cc (200, 35)*
0	1.1495	1.1777	1.0096
	0.0031	0.0068	0.0067
0.2	1.1183	1.1357	1.0533
	0.0031	0.0069	0.0062
0.5	1.0756	1.0810	1.0607
	0.0029	0.0073	0.0064
1.0	1.0048	1.0177	1.0022
	0.0030	0.0072	0.0064
2.0	0.8654	0.8751	0.8583
	0.0025	0.0053	0.0052
3.0	0.7221	0.7210	0.6856
	0.0021	0.0047	0.0042
4.0	0.5233	0.5221	0.4795
	0.0014	0.0033	0.0032

* No. of sampling for initial neutrons source (No. of particle , No of Batches)

The use of thorium does not create Np-237 or the other minor actinides created when U-238 fertile material is used. This feature offers a great advantage in the use of thorium over U-238 as fertile material.

The results of the calculations shown in the first column of Table 2 are based on sampling 1000 initial source neutrons and 50 batches. Results in the next column have been obtained by using a smaller number of particles: 200 neutrons and 33 batches. Due to the smaller sampling number in the latter case, the statistical errors of the results are relatively larger, but even this small number of sampling provides a useful trend. It thus shows that this Monte Carlo burn-up calculation can be used for parameter study of the conceptual design.

Due to the need for many cells in the core, the calculation time (125 min) for sampling numbers of 1000 particles * 50 batches is five times longer than the single cell calculation time (25 min) for using the same sampling numbers.

TABLE 3
SPECIFICATION OF THE WATER-COOLED REACTOR WITH A HARD NEUTRON SPECTRUM

Design Parameters	Specification with Void Tube	Specification Without Void Tube
Rod Outer Diameter	11.91 mm	14.5 mm
Rod Inner Diameter	11.11 mm	14.mm
Pitch Of Rod	13.21 mm	15.8mm
Number Of Rods In Bundle	469	217(271)
Smeared Fuel Density	93 %	88%=10/11.3
Inner Distance Of Channel Box	298 mm	206.3 mm
Outer Distance Of Channel Box	299 mm	213.mm
Pitch Of Fuel Assemblies	304.2 mm	219.7 mm
No Of Fuel Assembly	252	974(924)
No Of Void Tube Assemblies	61	0
Core Diameter	5651 mm	7200 mm
Core Height	1600 mm	1000 mm
Upper Blanket	300 mm	No
Lower Blanket	300 mm	No

5. MONTE CARLO BURN-UP CODE DEVELOPED AT BNL

The neutronic characteristics of the reactor without void channel were calculated using the Monte Carlo burnup code (MCBURN) [5]. The MCBURN code was developed at BNL, first by Yang combining the MCNP [6] and ORIGEN [7] codes. This code was then revised by Cetnar using the Unix system and C-language to facilitate handling of the large number of files created during the calculations and their reuse. This code was further developed by Cetnar in collaboration with a Swedish university. Reference 8 describes recent developments with this code.

In the analysis of this reactor, void channels have been introduced to suppress the positive void-coefficient by neutron streaming when coolant boiling occurs. It is very crucial to properly analyze neutron transport, using the Monte Carlo method, although this requires a longer computer time than using analytical methods based on the first-flight collision probability method [9], which we developed in 1960, to analyze the heterogeneity of the fuel assembly configuration. The regular first-collision probability method assumes isotropic neutron scattering. To take anisotropic scattering into account, one of the authors developed a generalized collision probability method [10] for a cylindrical lattice system, which can be easily extended to other geometries such as square or hexagonal lattice. However, extension of this method to three-dimensional problems becomes a rather complicated operation. The collision-probability methods are analytical deterministic methods, so that the uncertainty associated with the probabilistic Monte Carlo method, due to the statistically small number of samples, does not exist. Since an energy group approximation has to be used, the calculation is inexact for the analysis of a core system composed of material that has resonance cross sections, although the H2O coolant can provide some relief for this approximation.

The Monte Carlo method can handle continuous-energy cross section representation and is also free from any approximation in geometry. Thus, we adopted the Monte Carlo method for our burn-up calculation. In carrying out this reactor calculation, the core has been segmented into many separate regions. In each region, the change of composition due to burnup is calculated by the ORIGEN code. Hence, while this approach is not completely approximation-free, the streaming of neutrons can be treated without any approximation.

6. DEVELOPMENT OF THE KINETICS CODE

To evaluate the safety of a tight lattice reactor, the neutronics and thermo-hydraulics codes RELAP [11] and PARCS [12] will be used. The PARCS code, developed for a square lattice by Downar, was recently extended to enable calculations of hexagonal lattice geometries, so that it can be used to analyze the Russian-designed PWRs. The PARCS code uses a diffusion approximation for neutron transport. To implement strong anisotropic neutron flux distributions, it can use the Sn type transport method, but this calculation takes a long time, and its deterministic methodology is not as powerful as the Monte Carlo method.

Instead of calculating the dynamics of the whole reactor with the Monte Carlo method which requires a substantial amount of computer time, we can provide the diffusion coefficient which will be used for the PARCS assessment by calculating the segmented core regions with the Monte Carlo method, with suitable boundary conditions. We have successfully applied this approach to calculate the one-dimensional kinetics problem [13]. This approach shortens the kinetics calculation while properly treating the neutron streaming through the voided tubes by tabulating the diffusion coefficient or by formulating it in an analytical form.

CONCLUSIONS

Preliminary studies of a water-cooled reactor with a hard neutron spectrum indicates that thorium fertile material yields higher burn ups than U-238, without producing Pu and minor actinides. This approach meets the objective of a proliferation-resistant design, as required in the NERI program. The Monte Carlo method is a useful tool for the analysis of a reactor with complicated geometry, although the computer time is longer than that required for the analytical method. Even when analyzing the whole reactor with many lattice cells and small uncertainties, computing does not require prohibitively long times.

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