#### Boron Use and Control in PWRs and FHRs

Anthony Monterrosa, Anagha Iyengar, Alan Huynh, Chanddeep Madaan Department of Nuclear Engineering University of California, Berkeley

#### 5/5/2012

#### **Report UCBTH-12-007**

#### ABSTRACT

Boron sees extensive use as a neutron absorber for use in nuclear reactor systems. Pressurized water reactors (PWRs) utilize boron in their control rods as solid absorbers and coolants as chemical shims. Boron is also being considered as the main component in the neutron shielding system for Fluoride-Salt Cooled High Temperature Reactors (FHRs). Through a materials review and MCNP simulations, this design project seeks to provide recommendations for a boron shielding system for FHRs, and a method to minimize tritium leakage from control rods in PWRs. Ultimately, it is recommended to use boron carbide pellets clad with stainless steel and inserted in channels drilled into the outer graphite reflector. The stainless steel cladding will be useful for several functions 1) it will provide mechanical stability 2) it will protect the coolant from boron contamination 3) it will allow for the addition of a helium vent assembly and 4) it may provide for the means to replace depleted absorber rods. In terms of the overall shielding design, it is suggested that a boron packing fraction greater than 50% and an outer graphite reflector greater than 50 centimeters thickness be used. This will ensure that there will be no more than 10% burn-up of the boron carbide through the lifetime of the reactor (80 years). To limit tritium diffusion in PWRs, it is suggested that a tritium permeation barrier of aluminide is coated onto the control rods. This will limit tritium content in the coolant and ultimately leakage to the environment.

## CONTENTS

Repo	ort l	ICBTH-12-007	1
1.0	IN	TRODUCTION	3
1.1	•	Boron Use in PWRs	
1.2	•	boi on use m firs	
2.0	Μ	ATERIALS CHARACTERIZATION	4
2.1	L	Physical Implementations of Boron	4
2.2	2	Boron Behavior Under Irradiation	5
	2.2.1	Boron Carbide Under Irradiation	5
	2.2.2	Boron Nitride Under Irradiation	8
3.0	B	DRON POISON IMPLEMENTATION AND DESIGN	9
3.1	L	Cladding Considerations	9
	3.1.1	Pellet-Clad Interactions	9
	3.1.2	Helium Vent Implementations	11
3.2	2	Tritium Permeation Barriers	
	3.2.1	Permeation Barriers and Reduction Factors	14
	3.2.2	Aluminized Steel as a Tritium Permeation Barrier	15
3.3	3	FHR Neutronic Design	
	3.3.1	Design Parameters for Modeling	17
	3.3.2	Tallies of Interest	
3.4	ŀ	MCNP Simulation Results	
	3.4.1	Boron Consumption Rates	23
	3.4.2	Poison Burn-up Time	25
	3.4.3	Damage to Core Barrel and RPV	
4.0	SU	JMMARY	
4.1	L	Boron Absorber Rod Design Suggestions	
4.2	2	Boron Shielding Design Suggestions	
4.3	3	Future Directions	
5.0	A	CKNOWLEDGEMENTS	35
6.0	REF	ERENCES	35

#### **1.0 INTRODUCTION**

Boron is used extensively as an absorbing material in reactor systems. It is a very attractive material due to its low cost of production and fabrication. Past advancements have made boron into a relatively important and cheap material to use. The main reason that boron is so appealing is due to its high thermal absorption cross section which increases the probability for neutron capture of moderated neutrons. Additionally, chemical forms of boron, such as boron carbide, are available that are very non-reactive and perform well under irradiation.

Extended cycle lengths and fuel burnups are receiving increased attention in construction of advanced light water reactor systems. Frequently, the fuel management strategies in pressurized water reactors (PWRs) require burnable poisons (shims and rods) to control power distribution and to maintain a negative moderator coefficient. Currently, boron rods are used as a burnable poison to control neutron thermal energies and limit the multiplication factor in PWRs. Boron also has prospective use in fluoride salt cooled high temperature reactor (FHR) systems for control rods and reactor vessel neutron shielding. An important potential disadvantage of using boron in these applications is the production of tritium, lithium-7 and helium. This report reviews the effects of radiation damage and helium production on the properties of boron carbide, which is an attractive solid form for boron use, control of helium from boron containing pins, as well as tritium production and control for boron compared production from other sources such as lithium-6.

#### **1.1 Boron Use in PWRs**

As cycle lengths have increased, neutron-absorbing materials, also called poisons, are now frequently inserted into PWRs in order to lower the high reactivity of their initial fresh fuel load. They can come in the form of burnable, non-burnable and soluble poisons. Soluble poisons, also called chemical shim, produce a spatially uniform neutron absorption when dissolved in the water coolant. The most common soluble poison in commercial PWRs is boric acid, which is often referred to as soluble boron. Most boiling water reactors (BWRs) typically do not use soluble neutron poisons during normal operation. Fixed burnable poisons possess some advantages over chemical shim. Fixed burnable poisons may be discretely loaded in specific locations in order to shape or control flux in the core. With chemical shim, as temperature rises and the moderator expands, some of the moderator is pushed out of the active core area. The soluble boron is also moved out and this has a positive effect on reactivity. Because some neutron reactions with boron produce tritium, the use of soluble poisons also makes it more difficult to control and contain tritium inside PWRs, giving another incentive to work with fixed burnable poisons and to implement tritium diffusion barriers to keep the resulting tritium immobilized.

In PWRs tritium is produced primarily by B-10 reactions with neutrons. Approximately 90% of the total tritium in PWR reactor coolant is produced in the coolant by the soluble boric acid reactivity shim. The remaining 10% is produced by ternary fission, diffusion of tritium from B-10 burnable poison rods, Li-6 neutron capture, and deuterium activation. Since more and more advanced reactors now require larger

lifetimes for every aspect of the core for smooth running and little shutting down, the initial concentration of boron has been increased in the reactor coolant.

## **1.2 Boron Use in FHRs**

The use of boron in FHRs is largely unknown and is still under development. A few instances where boron will predominantly be used will be to provide thermal neutron shielding for the reactor vessel and upper core support structures, and as a neutron poision in control rods and shutdown rods. The advantage that FHRs have over PWRs is that boron is not normally in direct contact with the coolant and thus, tritium (produced through neutron interactions) can be contained and its release largely reduced outside the core. However, if corrosion or materials degradation processes bring boron containing material into contact with the salt coolant, the potential that boron could dissolve into the coolant salt could have negative impacts on the FHR reliability and safety.

## 2.0 MATERIALS CHARACTERIZATION

The effectiveness of neutron absorbers depends on both the environmental conditions as well as the material form. In this section several options for boron-based shielding are considered. Their viability depends on their stability in a high-temperature, high radiation environment and also how readily available they appear to be. Boron carbide emerges asa strong candidate for its use in reactors, and due to its extensive use as a neutron poison its properties are well documented as well as its mechanical and structural response under irradiation.

## 2.1 Physical Implementations of Boron

Part of the challenge in using boron as a solid neutron absorber (rather than a soluble poison) stems from the fact that pure boron is very brittle and cannot be machined into shapes. Therefore, boron must be combined with other elements for use as solid structures in a reactor. The following paragraphs review different physical forms of boron and their potential for use in PWRs and for reactor vessel shielding in FHRs.

## Boron Carbide:

Boron carbide is currently used in solid form for control rods and as a burnable poison in PWRs. This material is extremely hard. Known for its use in tank and body armor, few materials are harder. Boron carbide is also simple and cheap to fabricate, and high concentrations of boron can be achieved. Since boron carbide has been used so extensively in PWRs as well as boiling water reactors (BWRs), its effects under irradiation are also fairly well understood. In terms of PWRs, boron carbide likely remains the most attractive physical form for solid burnable poisions. Design efforts will focus on minimizing tritium diffusion from pins containing the boron carbide.

Boric Acid:

Boric Acid  $(H_3BO_4)$  is a soluble poison that also has seen use in PWRs (as well as an injectable poison for reserve shutdown in BWRs). This poison is dissolved into the PWR primary coolant where the concentration can be controlled based on the neutronics. However, the production and ultimate release of tritium cannot be prevented, seeing that any tritium produced in the coolant can be subject to leakage.

## Borated Polyethylene:

Borated polyethylene is readily available and can be produced in a many shapes and forms rather easily.<sup>1</sup> Studies have also been performed showing the effectiveness of the neutron absorption.<sup>2</sup> One significant issue with borated polyethylene (as is with any polymer) is the relatively low melting point (120°C-130°C). In a power reactor environment, some form of insulation and cooling would be necessary to use borated polyethylene.

## Boron Nitride:

Boron nitride in the form of boron nanotubes is currently under investigation for a lightweight ingredient for the hulls of spacecraft.<sup>3</sup> Using a ball-milling and annealing method these boron nanotube samples can be made in large quantities. However, since it is a new technology, radiation studies have not been conducted. Boron nitride is an extremely hard material similar to boron carbide, and it can withstand ultra-high temperatures. However, the difficulty in utilizing it comes from its relatively unexplored nature.

## Zirconium Diboride:

Zirconium Diboride ( $ZrB_2$ ) has been used as a very thin coating on fuel pellets. Helium migration through the crystal structure of  $ZrB_2$  is dependent on the crystallographic orientation, so it may be possible to manage the evolution of helium more readily than in other materials. It is likely that zirconium diboride would need to be used in conjunction with other methods of neutron absorption and shielding.

## 2.2 Boron Behavior Under Irradiation

Typical of most ceramics (excepting graphite), boron carbide experiences swelling under irradiation. Neutron irradiation initially causes displacements of primary and secondary knock-on atoms. This results in the creation of vacancies and other point defects. These defects can coalesce into voids which combined with the production of helium ultimately causes swelling. Understanding swelling, in addition to the role helium evolution plays, is vital when using boron carbide in a high radiation environment.

## 2.2.1 Boron Carbide Under Irradiation

Boron carbide in reactors is usually in the form of sintered pellets. When exposed to neutron irradiation, boron undergoes the following reactions<sup>4</sup>:

 $^{10}\text{B} + ^{1}\text{n} \rightarrow ^{7}\text{Li} + ^{4}\text{He} + 2.6 \text{ MeV}$ 

B-10 and Li-7 can occasionally produce tritium in a collision with high-energy neutrons:

$$^{10}B + {}^{1}n \rightarrow 2 {}^{4}He + {}^{5}T$$
  
 $^{7}Li + {}^{1}n \rightarrow {}^{4}He + {}^{3}T + {}^{1}n$ 

All of these reactions include the production of helium. Most of the helium produced will diffuse outside of the pellets, while the remaining gas accumulates, which leads to swelling and microcracks. Because of swelling in the absorber, there can be a strong mechanical interaction between the absorber and the tube containing it if the gap between the pellet and tube closes. The tube may also crack from a combination of swelling in the absorber and irradiation embrittlement.



Fig. 1. Outer surface of the pellets. (a) Before irradiation. After irradiation: (b) magnesothermal  $B_4C$ ; (c) carbothermal  $B_4C$ ; (d) boron-rich  $B_8C$ .



In boron carbide, atoms in CBC chains and icosahedra are knocked to interstitial positions. In the icosahedra, the vacancy left behind is highly unstable, so an atom sitting in an interstitial site will return to its original position, creating a self-healing mechanism in the icosahedra.<sup>6</sup>

Specimen No.	Experiment	B/C ratio	Isotopic <sup>10</sup> B (at %)	Burnup % of <sup>10</sup> B	$(n, \alpha)/cm^3$	Irradiation temperature (°C)	Swelling $\% (\Delta V/V)$	Gas release (%)
G-13B	ANL 'HWR' Control Rod	3.95	20	1.7	3.5 × 10 <sup>20</sup>	500b)	0.3e)	1-5
0-5T 0-5B 0-9T 0-9B	ORNL X-099	3.8 3.8 3.8 3.8	19.8 19.8 19.8 19.8	2.71 3.10 2.75 3.01	5.7 6.5 5.8 6.3	710c) 735c) 800c) 840c)	2.6 <sup>e</sup> ) 3.3 <sup>e</sup> ) 0.5 <sup>e</sup> ) 0.15 <sup>e</sup> )	49 31 41 34
31F 31S 31C 41F 41S	GGA BG-2 Sample 3685-105	2.8	1.5	7.5a)	15.8	500d) 500d) 700d) 600d) 600d)	0.5f) 0.5f) 0.5f) 0.5f) 0.5f)	
13S 13F	Sample 3685-4	5.0	19.8	48.0 81.0	101 170	550d) 550d)	6-10f) 10-13f)	

Summary of materials and irradiation variables of B4C speciment	Summary	of	materials	and	itradiation	variables	of	B4(	specimens
---	---------	----	-----------	-----	-------------	-----------	----	-----	-----------

**Figure 2.2:** Irradiated specimens of  $B_4C$  and tabulated results.<sup>7</sup> TABLE I. Characteristics of samples with compositions close to  $B_4C$  which were selected for ceramographic, SEM, and TEM observations.

Specimen number <sup>a</sup>	Atomic fraction of free carbon (at. %)	Density (% TD)	Burn-up (10 <sup>26</sup> capture m <sup>3</sup> )	Helium generation (at. %)	Irradiation temperature (°C)	Volume swelling (%)	Isotopic <sup>10</sup> B enrichment before irradiation (%)
16	?	94	≂5	≈0.4	800	< 1	20
4	0.29	96	24	1.8	620	3.2	90 *
5	0.29	96	129	9.9	1000	17.4	90
6	0.35	84	16	1.2	600	1.7	90
7	0.35	84	79	6.0	900	3.4	90
8	0.35	84	142	10.8	1500	15.7	90

\*The numbering is the same as in Ref. 16.

\*Sample 1 is a reference melted boron carbide irradiated in LMFBR Rapsodie at a very low burn-up.

Figure 2.3: Data from a study of irradiation effects on boron carbide pellets<sup>8</sup>

**Figures 2.2** and **Figure 2.3** each show the results of boron carbide neutron irradiation experiments. Using the data, it is then possible to calculate burn-up levels of B-10 as a function of swelling rate. The results of the calculations are shown in **Figure 2.4**. The data from each of the studies seem to correlate well with each other and would indicate a linear relationship of swelling and burn-up. By running simulations of FHRs with boron shielding it then becomes possible to calculate swelling rates by the simulated boron burn-up.



**Figure 2.4**: A plot of the relative swelling as a function of the relative burn-up of B-10 calculated from the data given in Figure 2.2 and Figure 2.3.

The burnup rate (i.e. the depletion rate) of <sup>10</sup>B will be one of the determining factors for the lifetime of boron carbide. However, the lifetime of boron carbide under neutron irradiation is not determined just by its burnup rate, but also by the damage induced by helium evolution through the formation of bubbles and swelling.<sup>9</sup> The effects of helium accumulation can be understood by looking at the physical damage of swelling compared with the relative burnup; furthermore, because the helium ions take a considerable amount of the kinetic energy when generated in the boron neutron-capture reaction, a review of the damage incurred by helium ions will also be conducted.

#### 2.2.2 Boron Nitride Under Irradiation

Not much is known about the response of boron nitride under irradiation, but from what *is* known, defect formation is similar to that in boron carbide. Helium, lithium, and tritium are all produced, as well as carbon-14 through an (n,p) reaction with nitrogen-14. In addition, there are various point defects formed at low temperature from fast neutron irradiation.<sup>10</sup>

#### **3.0 BORON POISON IMPLEMENTATION AND DESIGN**

In this section, an overview of the FHR neutronic design, and MCNP model used will be described in detail. In order to assess the shielding performance of the outer reflector system, parametric studies need to be performed to determine the sensitivity of radiation damage figures of merit on the core barrel and reactor pressure vessel. Proposed designs for control rods and cladding are also discussed.

#### 3.1 Cladding Considerations

In the proposed design for FHR shielding, boron carbide pellets will be embedded in channels in the graphite reflector near its outside radius. This requires holes to be drilled into the reflector. While boron carbide does not react readily with graphite, there are several reasons to use an additional cladding for the pellets. Likely the largest concern is the evolution of helium through  $(n, \alpha)$  reactions in boron. If the graphite channels are sealed to prevent the coolant salt from contacting the boron carbide pellets, helium buildup in the channels could pose a serious pressure problem. The addition of a cladding will allow for easy implementation of a vent structure that can relieve the helium pressure. Another consideration for the addition of a cladding is in the case of an incident where the molten fluoride salt comes into contact with the boron carbide pellets. Any boron dissolved in the coolant could seriously affect the neutronics of the reactor and could create a hazardous and unstable situation. Cladding material, specifically stainless steel, would provide an effective barrier against any contact with the fluoride salts and prevent such a situation from occurring. A cladding may also increase mechanical stability and allow for a simple method of replacing depleted absorber rods As boron control rods with stainless steel cladding have already been implemented in PWRs, much work has already been done in understanding the implementation of cladding with boron carbide.

## 3.1.1 <u>Pellet-Clad Interactions</u>

As it is with fuel rods, the effects of swelling of boron carbide should be taken into account when considering the mechanical integrity of the cladding. Computer codes accounting for the production of helium and the swelling of boron carbide have predicted mechanical failure of the cladding shortly after contact is made.<sup>11</sup>Therefore it is important to predict the amount of swelling expected for the boron carbide through the entire lifetime of the reactor so an adequate gap can be provided between the absorber and the cladding.

Also similarly to fuel rods, it is important to consider any heat transfer occurring between the pellets and the cladding. However, because the rate of heat generation is far smaller than for fuel rods, the objective is to preserve mechanical integrity and not maximize heat transfer, there is more flexibility in design. Extensive experimental tests have been performed to gain insight on the thermal design of the cladding and the gap for boron pellets used in control rods for PWRs and fast spectrum reactors.<sup>12</sup> Figure 3.1 displays an example of an experimental set-up and is reminiscent of what the actual design of the FHR shield rods would be.



Figure 3.1: The dimensions and schematic of the experimental piece for heat transfer data (Table 3.1) is shown above.

It was found that the gap conductance varied as a function of the eccentricity between the pellet and the cladding. Variations of +10% to -5% were found. The expanded gap width as varied with temperature is shown in Table 3.1. Because the shielding pins in FHRs receive very low neutron dose rates, it is not expected that thermal conduction across the gap will pose a problem, but this is one issue that must be addressed during detailed design.

Run	Power	Surface	ð,		Angle	
No.	(W)	(°C)	(mm)	Ø <sub>0</sub>	$\theta_1$	02
1	83	$T_1 = 96.1$ $T_2 = 96.0$ $T_3 = 95.8$	011 012	0.510 0.510	0.510 0.510	0.510 0.510
2	172	$T_1 = 173.3$ $T_2 = 173.2$ $T_3 = 172.9$	$\hat{a}_{t1}$ $\hat{a}_{t2}$	$0.517 \\ 0.516$	$0.517 \\ 0.516$	0.517 0.516
3	271	$T_1 = 247.7$ $T_2 = 247.5$ $T_3 = 247.2$	0 <sub>11</sub> 0 <sub>12</sub>	0.524 0.523	$0.524 \\ 0.522$	0.524 0.523
4	380	$T_1 = 319.2$ $T_2 = 319.1$ $T_3 = 318.8$	ο <sub>ι1</sub> ο <sub>ι2</sub>	0.532 0.530	0.531 0.530	0.531 0.529
5	542	$T_1 = 411.2$ $T_2 = 411.1$ $T_3 = 410.6$	011 012	0.541 0.539	0.541 0.538	0. 541 0. 538
6	704	$T_1 = 492.9$ $T_2 = 492.4$ $T_3 = 492.4$	ο <sub>ι1</sub> ο <sub>ι2</sub>	$0.550 \\ 0.547$	0.549 0.546	0.550 0.547
7	850	$T_1 = 554.2$ $T_2 = 553.5$ $T_3 = 553.7$	ο <sub>ι1</sub> ο <sub>ι2</sub>	0.557 0.554	0,556 0,553	0.556 0.553
8	1,010	$T_1 = 611.5$ $T_2 = 610.5$ $T_3 = 610.9$	011 012	0.563 0.560	0.563 0.559	0. 563 0. 559

**Table 3.1**: The expanded gap width is tabulated along with temperature at various eccentricities.

 $\delta_{t_1}$ : Gap width for free axial expansion

 $\delta_{t_2}$ : Gap width for no axial expansion

The data in Table 3.1 will be helpful in determining an initial gap width for the boron carbide pellets. However, when irradiation is present, the eccentricity will of course change due to swelling. Additionally, the gap will begin to close which will create more complexities to the problem. Because FHR shielding pins operate at low dose rates, the use of a conservatively safe lower bound of initial gap width is suggested.

#### 3.1.2 Helium Vent Implementations

As previously mentioned, the production of helium in boron carbide could pressurize cladding or channels drilled in graphite blocks leading to damage, if they are hermetically sealed. It is known that helium generation is dependent on the temperature, porosity, burn-up, and material form.<sup>13</sup> Figure 3.2 plots gas release against porosity for a variety of temperatures. Figure 3.3 compares gas release between powders and pellets of different densities as a function of temperature.



Figure 3.2: Gas release is plotted against theoretical density for various temperatures



Figure 3.3: Gas Release is plotted against temperature with varying densities of boron carbide powders and pellets

As a general trend Figure 3.2 shows that gas release tends to increase with increasing temperature and with decreasing density. This could be the result of enhanced helium diffusion at higher temperatures and more pathways of escape in a more porous structure. Figure 3.3 again shows more gas release at higher temperatures. A curious result is that the pellet form seems to release more helium than the powdered form.

With significant amounts of helium production it is important to consider designs for a helium vent to relieve this pressure. A 1976 patent proposed such a design.<sup>14</sup> This vent assembly is designed to be attached to the top of a stainless steel control pin and vent helium while preventing any back-flow of coolant. A schematic of the design is shown in Figure 3.4.



Figure 3.4: The proposed helium vent assembly is shown above.

The entire module (labeled A) would be attached to the top of the FHR shielding pins. Helium flow is directed up the middle tube (26) where it is eventually vented into a gas chamber (36). A porous metal baffle (16) allows for the flow of helium out of the vent holes (24). This assembly was designed for pins fully immersed in coolant. The gas chamber allows for a slight pressurization to prevent molten salt back-flow. Implementing a helium vent chamber will effectively allow for control of the helium and pressure while protecting the coolant from boron contamination.

#### 3.2 Tritium Permeation Barriers

As previously mentioned, tritium is produced in the control rods of PWRs. To prevent this tritium from contaminating the coolant, it would be effective to prevent the diffusion of tritium through the control pins. As such, coating the control rods with tritium permeation barriers will likely limit tritium leakage in PWRs. There exist two options in the realm of tritium permeation: 1) application of an oxide layer or 2) application of a metallic/ceramic material. Each option has its own advantages and disadvantages. The permeation reduction of the former is heavily dependent on the defects formed. However Figure 3.5 shows three different mechanisms of permeation through composite materials. The "Composite Permeation Model" expects the diffusing species to diffuse through each material of the composite. In the "Areal-Defect Model", the diffusing species reaches an impermeable barrier and diffuses preferentially through defects. The "Surface Limited" relies on the desorption of material. Recombination effects on the downstream side control the rate of permeation through the composite.<sup>15</sup>



**Figure 3.5**: Representative figures of the Composite Permeation Model, Areal-Defect Model, and Surface Limited case are shown. Notice that there is a P<sup>1/2</sup> dependence for two of the models but a P dependence for the Surface Limited case.<sup>15</sup>

It is common knowledge that the stainless steels are corrosion resistant because of their passive oxide layers. However, these oxide layers tend to crack under thermal stress and are prone to damage; on the other hand, these oxide layers also have self-healing mechanisms in which the passivating layer of the material reforms after they have been destroyed.<sup>16</sup>

#### 3.2.1 <u>Permeation Barriers and Reduction Factors</u>

The reduction factor of a material is defined as the ratio of the permeation rate before application of a permeation barrier to the permeation rate after the application of a permeation barrier. Tables 3.2 and Table 3.3 list composite materials explored in two separate studies for tritium barrier development. As shown, aluminum oxide appears to be the material of choice. It has a high oxidation rate (for self-repairing mechanisms) as well as a high permeation reduction factor (PRF). The scattered PRFs shown in each table are attributed to the Areal-Defect Model, as several factors in each respective study indicated that the permeation rates were strongly related to defects.

## **Table 3.2:** A table from Perujo et al. showing several composite materials selected for tritium permeation barriers

Coating material	Coating method	Coating thickness (µm)	Substrate material	Substrate thickness (mm)	Reduction factor
TiC *	CVD	1-2	TZM	0.1	10
TiN + TiC	CVD	3	AISI 316L	1.0	10
$Al_2O_3 + TiN + TiC$	CVD	6	AISI 316L	1.0	10
Al <sub>2</sub> O <sub>3</sub> *	CVD	3	AISI 316L	1.0	10
Al <sub>2</sub> O <sub>3</sub> b	Pack cementation	80	AISI 316L	1.0	100
Al <sub>2</sub> O <sub>3</sub>	VPS	100	MANET	2.0	100 - 1000
Cr + Mn oxides	Oxidation on H <sub>2</sub> (1 ppm O <sub>2</sub> ) at 810 K.	≈0.04	MANET	Cylinder of diameter 6 mm and length 60 mm	100

Effectiveness of different applied coatings as permeation barriers, with the permeation reduction factors given as approximate values

<sup>a</sup> No difference was found when the layer was in contact with Pb-17Li.

<sup>b</sup> Not efficient when in contact with Pb-17Li, owing to cracks (see text).

# **Table 3.3**: Another table from Hollenberg et al. showing several composite materials selected for tritium permeation barriers

Systems of	considered	as	tritium/hydrogen	barriers
------------	------------	----	------------------	----------

Barrier coating	Base metals	Permeation reduction	Source
Aluminide (or Al <sub>2</sub> O <sub>3</sub> )	SS316, MANET, TZM, Hastelloy-X, Ni	10 to above 10000	[1-9]
TiC, TiN, TiO <sub>2</sub>	SS316, MANET, TZM, Ti	Less than 10 to above 10000	[10-13]
Cr (or Cr <sub>2</sub> O <sub>3</sub> )	SS316	10	[14]
Si	Steels	10	[15]
BN	304SS	100	[16]
Sn	Ferritic steel	Rapidly degraded	[17]
H <sub>3</sub> PO <sub>4</sub> glass	304SS	100 (unstable)	[1]
N	Iron	10-20	[18]

#### 3.2.2 Aluminized Steel as a Tritium Permeation Barrier

Aluminized steel exhibits many aspects of a good tritium permeation barrier. As the above studies have shown, aluminum oxide has a high PRF. It also has good corrosion resistance, good resistance to thermal cycling, a self-healing mechanism, and is easily manufactured as aluminizing is already an established commercial technique.

#### **3.3 FHR Neutronic Design**

In order to understand the implementation of boron in shielding and damage received in the FHR graphite reflector and reactor pressure vessel, neutronic models were developed using MCNP. The design for neutronic implementation is shown in Figure 3.7 below. The MCNP model includes modeling the entire reactor core, the graphite

reflector, and the reactor pressure vessel. Graphite blocks establish the internal geometry of the core, and provide neutron reflection, shielding, and some neutron moderation. The fuel system is housed inside the graphite blocks, and the coolant flow-paths are provided by channels in the graphite blocks. Boron rods are inserted in the graphite blocks at their radial extremity, and wherever else necessary and practical, to ensure neutron shielding of the reactor vessel.

To implement the simulations, the Pebble bed Input Maker for Parametric Studies (PIMPS2) was used with MCNP5 to set up the code design and equilibrium depletion analysis. There are four different design variables that can be used to complete this analysis: Graphite reflector thickness, shield thickness, pitch of poison rods, and the diameter of the poison rods. However, to complete the analysis in a simpler manner, it should be noted that several assumptions were made. Since the 4-dimensional design space is hard to implement, a reduced set of variables was chosen as the factors by which to gauge analysis; the thickness of the reflector and shield, and the packing fraction of boron carbide poison.

Realistically, the boron poison rods will be in a hexagonal pattern in the shielding section, however, as the packing fraction is increased, the number of control rods within the shielding cell is accordingly increased. Figure 3.6 below shows the close-up view of the hexagonal absorbers, support mechanism and the RPV.



**Figure 3.6**: Hexagonal absorber rods surrounded by RPV and restraint mechanism in the Japanese HTTR reactor.<sup>17</sup>





and fuel active region are labeled. The blanket active region contains graphite pebbles.

The goals that we plan to address by presenting the implemented model are to assess the shielding and reflector performance in FHRs and assess the role of boron carbide as an absorber. Finally, these parametric studies and their results will be used to determine an 'optimal' solution that satisfies various conditions.

## 3.3.1 Design Parameters for Modeling

The design parameters used here to gage the effectiveness of the FHR shielding and damage incurred due to radiation damage during operation can be divided into two main categories: radiation damage and poison lifetime analysis. Carbon structures such as the reflector graphite and CFRC core barrel are damaged fast neutrons, while the reactor vessel and other metallic structures are damaged by both thermal and fast neutrons (in particular due to the generation of helium from neutron reactions with nickel). Radiation damage was investigated by looking at the thermal neutron flux at different locations within the reactor internals, and by calculating a DPA as a function of the reflector thickness at the core barrel and reactor pressure vessel. The lifetime goal for the poison rods is to achieve 10% burn up within 75-80 years. The levels of boron were varied, and sensitivity to radiation damage as a function of this and change in thickness of reflector were investigated. In all cases that were considered, there were three parameters that remained constant throughout. One was the source strength, which was 7.02 x  $10^{19}$  [n/sec]. This number was derived from the reactor power of 900 MWth. The other values that remained constant throughout were the radius of the central graphite reflector (90 cm), and the outer radius of the blanket active region (240 cm).

In order to obtain information necessary, there are 4 design parameters that need to be considered. The variables that need to be considered are the thickness of the graphite reflector, shield region thickness, the diameter of  $B_4C$  poison rods, and the pitch of the  $B_4C$  poison rods. Since a 4-parameter 4-dimention design space is too hard to implement and interpret, it is possible to simplify the number of parameters down to just two values we need to alter to obtain necessary information; the thickness of reflector and shield, and the packing fraction of the poison rods.

## 3.3.2 Tallies of Interest

In the MCNP code, four key tallies were used to accumulate data of interest: Tally 2 (Figure 3.8) gives the fluence on the first wall of the outer graphite reflector, and hence the flux and number of source particles can be determined, Tally 34 (Figure 3.9) gives the tallies within the boron shielding which will help in calculating time for burn up of 10% of boron, Tally 44 (Figure 3.10) gives information regarding damage that occurs within the core barrel, and finally Tally 54 (Figure 3.11) gives information regarding damage in the RPV.

The total flux from the core was calculated using the power of the reactor, which was assumed to be 900 MWth and converting this to number of source neutrons per second produced.  $7.02*10^{19}$  n/s. The flux from at the first wall was calculated using the formula:

(1)

Flux = Fluence x Source neutrons (Strength) per second

1tally 2 2001445 nps = tally of fluence on first wall of outer graphite reflector + tally type 2 particle flux averaged over a surface. units 1/cm\*\*2 tally for neutrons 1000000.00 number of histories used for normalizing tallies = areas surface: 932 segment 1 1.00000E+00 2 1.00000E+00 4.52389E+05 3 surface 932 -802 segment: energy 3.72101E-01 0.0104 1.0000E-01 2.0000E+01 9.98741E-04 0.0890 total 3.73099E-01 0.0104 surface 932 802 803 segment: energy 1.0000E-01 3.05154E-01 0.0114 7.77689E-04 0.1051 2.0000E+01 total 3.05932E-01 0.0114 surface 932 802 -803 segment: energy 2.17040E-06 0.0058 1.0000E-01 2.0000E+01 2.69367E-08 0.0252 total 2.19734E-06 0.0058

**Figure 3.8**: Sample output deck for thickness of 90 mm and packing fraction of 50% shows total value of fluence on the outer wall of the reflector normalized per source neutron.

Table 3.4 lists the values for flux and fluence for various packing fractions of boron and varied thicknesses of reflector. It is expected the flux should be  $\sim 10^{14} [n/cm^2s]$ . The first wall of the outer reflector is shown in Figure 3.7 as surface 932, between the blanket and the outer graphite reflector.

<b>Packing</b> Fraction	Thickness of Reflector (cm)	Fluence on first wall of outer reflector (1/cm <sup>2</sup> )	Flux outer reflector (1/ s*cm <sup>2</sup> )
0	95	2.78E-06	1.96E+14
0	80	2.69E-06	1.89E+14
0	85	2.69E-06	1.89E+14
0	90	2.68E-06	1.88E+14
0.14	40	1.96E-06	1.38E+14
0.14	50	2.17E-06	1.52E+14
0.14	60	2.40E-06	1.69E+14
0.14	70	2.54E-06	1.78E+14
0.14	80	2.57E-06	1.80E+14
0.14	85	2.61E-06	1.83E+14
0.14	90	2.66E-06	1.87E+14
0.14	95	2.71E-06	1.91E+14
0.3	40	2.00E-06	1.40E+14
0.3	50	2.21E-06	1.55E+14
0.3	60	2.37E-06	1.67E+14
0.3	70	2.45E-06	1.72E+14
0.3	95	2.70E-06	1.90E+14
0.3	80	2.58E-06	1.81E+14
0.3	85	2.65E-06	1.86E+14
0.3	90	2.64E-06	1.86E+14
0.4	40	1.96E-06	1.38E+14
0.4	50	2.22E-06	1.56E+14
0.4	60	2.33E-06	1.64E+14
0.4	70	2.51E-06	1.77E+14
0.4	95	2.74E-06	1.93E+14
0.4	80	2.57E-06	1.81E+14
0.4	85	2.59E-06	1.82E+14
0.4	90	2.63E-06	1.85E+14
0.5	40	1.96E-06	1.38E+14
0.5	50	2.20E-06	1.54E+14
0.5	60	2.35E-06	1.65E+14
0.5	70	2.48E-06	1.74E+14
0.5	95	2.71E-06	1.90E+14
0.5	80	2.61E-06	1.83E+14
0.5	85	2.69E-06	1.89E+14
0.5	90	2.69E-06	1.89E+14
0.6	40	2.01E-06	1.41E+14

0.6	50	2.16E-06	1.52E+14
0.6	60	2.36E-06	1.66E+14
0.6	70	2.48E-06	1.74E+14
0.6	95	2.16E-06	1.52E+14
0.6	80	2.62E-06	1.84E+14
0.6	85	2.69E-06	1.89E+14
0.6	90	2.65E-06	1.86E+14
0.75	40	1.96E-06	1.38E+14
0.75	50	2.21E-06	1.55E+14
0.75	60	2.35E-06	1.65E+14
0.75	70	2.49E-06	1.75E+14
0.75	95	2.67E-06	1.87E+14
0.75	80	2.57E-06	1.81E+14
0.75	85	2.65E-06	1.86E+14
0.75	90	2.68E-06	1.89E+14

**Table 3.4**: Shows values of flux and fluence on the outer reflector wall for varied packing fraction and thickness values. Flux values correspond to what is expected to be seen.

```
1tally 34
                          2001445
                nps =
                               tallies for boron shielding
+
          tally type 4 track length estimate of particle flux.
          tally for neutrons
 number of histories used for normalizing tallies =
                                                    1000000.00
          volumes
                 cell:
                           400
                      1.33394E+06
 cell (400<9041)
 multiplier bin: 1.00000E+00
               5.34320E-10 0.0469
 cell (400<9041)
 multiplier bin: -1.00000E+00 400
                                       103 :
                                                    104 :
                                                                 105 :
                                                                              106
                                                                                   :
                                                                                           107
               1.22012E-08 0.0096
 cell (400<9041)
 multiplier bin: -1.00000E+00 400
                                         -2
               1.22029E-08 0.0096
```

**Figure 3.9**: Sample output deck for thickness of 90 mm and packing fraction of 50% shows total value of reaction rate density of Boron Carbide interacting with neutrons from which consumption rates can be calculated.

```
1tally 44
                           2001445
                 nps =
                                   tallies for radiation damage in core barrel
+
                        track length estimate of particle flux.
          tally type 4
          tally for neutrons
number of histories used for normalizing tallies =
                                                      1000000.00
          volumes
                  cell:
                            905
                        4.69566E+05
cell 905
 multiplier bin: 1.00000E+00
     energy
   1.0000E-01 1.38962E-09 0.0411
   2.0000E+01 5.27085E-11 0.2198
     total
               1.44233E-09 0.0414
cell 905
 multiplier bin: 1.00000E+00
                                         444
                              1
     energy
   1.0000E-01 7.62422E-13 0.2738
   2.0000E+01 2.70155E-12 0.2250
     total
                3.46398E-12 0.2007
```

**Figure 3.10**: Sample output deck for thickness of 90 mm and packing fraction of 50% shows total value of tallies for damage in core barrel. From this total (3.46398E-12), the cross section for damage and DPA were calculated.

```
1tally 54
                         2001445
                nps =
                               tallies for radiation damage in RPV
+
          tally type 4 track length estimate of particle flux.
          tally for neutrons
 number of histories used for normalizing tallies =
                                                    1000000.00
          volumes
                           907
                 cell:
                       4.98898E+05
 cell 907
 multiplier bin: 1.00000E+00
     energy
   1.0000E-01 2.07062E-10 0.0837
   2.0000E+01 1.47152E-11 0.6741
               2.21777E-10 0.0906
     total
 cell 907
 multiplier bin: 1.00000E+00 31
                                       444
     energy
   1.0000E-01 2.55233E-13 0.2980
   2.0000E+01 1.03481E-12 0.5527
               1.29005E-12 0.4512
     total
 cell 907
                                       103 :
                                                                105 :
 multiplier bin: -1.00000E+00 31
                                                    104 :
                                                                             106 :
                                                                                           107
     energy
   1.0000E-01 6.97734E-17 0.4868
   2.0000E+01 6.55728E-14 0.7450
               6.56426E-14 0.7442
     total
```



#### **3.4 MCNP Simulation Results**

As stated in the previous section, two particular parameters were varied in this project to gain a better understanding of the FHR shielding and damage; packing fraction of boron carbide and the thickness of the reflector and shield. The following sections detail the results from various aspects of the FHR analysis, including the boron consumption rate, the resulting time required to reach 10% depletion, and damage rates to the core barrel and reactor vessel.

Boron Packing Fraction	Thickness of Outer Reflector	B4C reaction rate density [dB/dt cm^3]	Volume Boron [cm^3]	Reaction Rate
0	95	1.19E-11	8.14E+05	6.81E+14
0	90	1.26E-11	8.02E+05	7.11E+14
0	85	1.42E-11	7.89E+05	7.85E+14
0	80	1.54E-11	7.77E+05	8.38E+14
0.14	40	2.52E-08	6.80E+05	1.20E+18
0.14	50	2.14E-08	7.04E+05	1.06E+18
0.14	60	1.86E-08	7.28E+05	9.52E+17

3.4.1	Boron	Consumption	Rates

0.14	70	1.59E-08	7.53E+05	8.38E+17
0.14	80	1.34E-08	7.77E+05	7.29E+17
0.14	85	1.25E-08	7.89E+05	6.92E+17
0.14	90	1.14E-08	8.02E+05	6.42E+17
0.14	95	1.04E-08	8.14E+05	5.95E+17
0.3	40	1.85E-08	9.97E+05	1.30E+18
0.3	50	1.56E-08	1.03E+06	1.13E+18
0.3	60	1.33E-08	1.07E+06	9.97E+17
0.3	70	1.10E-08	1.10E+06	8.56E+17
0.3	95	7.75E-09	1.19E+06	6.49E+17
0.3	80	9.61E-09	1.14E+06	7.69E+17
0.3	85	9.01E-09	1.16E+06	7.32E+17
0.3	90	8.41E-09	1.18E+06	6.94E+17
0.4	40	1.59E-08	1.15E+06	1.29E+18
0.4	50	1.39E-08	1.29E+06	1.25E+18
0.4	60	1.15E-08	1.23E+06	1.00E+18
0.4	70	1.02E-08	1.28E+06	9.09E+17
0.4	95	6.88E-09	1.38E+06	6.66E+17
0.4	80	8.75E-09	1.32E+06	8.09E+17
0.4	85	7.80E-09	1.34E+06	7.32E+17
0.4	90	7.21E-09	1.36E+06	6.87E+17
0.5	40	1.45E-08	1.29E+06	1.31E+18
0.5	50	1.22E-08	1.33E+06	1.14E+18
0.5	60	1.04E-08	1.38E+06	1.01E+18
0.5	70	9.05E-09	1.43E+06	9.06E+17
0.5	95	6.21E-09	1.54E+06	6.72E+17
0.5	80	7.68E-09	1.47E+06	7.94E+17
0.5	85	7.10E-09	1.50E+06	7.45E+17
0.5	90	6.56E-09	1.52E+06	6.99E+17
0.6	40	1.36E-08	1.41E+06	1.35E+18
0.6	50	1.13E-08	1.46E+06	1.16E+18
0.6	60	9.71E-09	1.51E+06	1.03E+18
0.6	70	8.26E-09	1.56E+06	9.06E+17
0.6	95	1.13E-08	1.46E+06	1.16E+18
0.6	80	7.08E-09	1.61E+06	8.02E+17
0.6	85	6.68E-09	1.64E+06	7.69E+17
0.6	90	6.12E-09	1.66E+06	7.15E+17
0.75	40	1.19E-08	1.58E+06	1.32E+18
0.75	50	1.03E-08	1.63E+06	1.18E+18
0.75	60	8.79E-09	1.69E+06	1.04E+18
0.75	70	7.45E-09	1.75E+06	9.14E+17

0.75	95	4.98E-09	1.89E+06	6.60E+17
0.75	80	6.31E-09	1.80E+06	7.99E+17
0.75	85	5.85E-09	1.83E+06	7.52E+17
0.75	90	5.41E-09	1.63E+06	6.21E+17

**Table 3.5**: Shows calculated values of consumption rate of boron for various cases where both packing fraction and thickness were varied.

As seen above in Table 3.5, boron consumption does vary as a function of reflector thickness, and as a function of packing fraction. Boron consumption is defined as the average reaction rate of all boron present. The pattern observed is that the consumption rate of boron increases as the packing fraction and volume are increased, and rate goes down as thickness of the reflector and shield increases. Figure 3.12 shows a plot that depicts the results. Figure 3.13 shows the consumption rate for a given radius increases as packing fraction increases. These are confirmed from theoretical expectations since these studies have been performed beforehand.

## 3.4.2 Poison Burn-up Time

Another key aspect of this project was to look at the amount of time it would take to burn up 10% of the total boron burnable poison. Since boron carbide has both isotopes B-10 and B-11, for simplicity it will be assumed that burn-up of B-10 is the most important since the reaction cross section for B-10 is highest.

**Table 3.6**: Shows the values from the input deck of the atomic content of enriched boron-10 that will be used to calculate time to reduce B-10 amounts by 10%.

Isotope	Material ID	Atomic weight	Weight	
		[at /cm-b]	[at/cm^3]	
Boron 10	5010.72c	2.20E-02	$2.196 \times 10^{22}$	

**Table 3.7**: Shows results of time to consume 10% of the boron in seconds and years.

Boron Packing Fraction	Outer Reflector Thickness (cm)	Boron 10 time in years	
0	95	83234.315	
0	90	78561.919	
0	85	70005.923	
0	80	64565.402	
0.14	40	39.301	
0.14	50	46.425	
0.14	60	53.264	
0.14	70	62.585	

0.14	80	74.201
0.14	85	79.494
0.14	90	86.950
0.14	95	95.316
0.3	40	53.497
0.3	50	63.689
0.3	60	74.613
0.3	70	89.815
0.3	95	127.987
0.3	80	103.207
0.3	85	110.065
0.3	90	117.882
0.4	40	62.314
0.4	50	71.589
0.4	60	85.963
0.4	70	97.724
0.4	95	144.101
0.4	80	113.389
0.4	85	127.155
0.4	90	137.591
0.5	40	68.609
0.5	50	81.290
0.5	60	95.329
0.5	70	109.659
0.5	95	159.815
0.5	80	129.080
0.5	85	139.803
0.5	90	151.242
0.6	40	72.762
0.6	50	87.769
0.6	60	102.158
0.6	70	120.090
0.6	95	87.769
0.6	80	140.104
0.6	85	148.456
0.6	90	162.048
0.75	40	83.054
0.75	50	96.645
0.75	0.75 60 112.81	
0.75	70	133.108
0.75	95	199.362

0.75	80	157.272
0.75	85	169.693
0.75	90	183.242

As it can be seen from the graph below, the time for 10% burn up of the boron increases as the packing fraction of boron is increased, and as the thickness of the reflector and shielding is increased. The goal was to make it possible for the boron poison to last the lifetime of the reactor, which we assumed to be around 80 years. The results show that this is in fact possible with certain combinations of packing fractions and reflector and shielding thicknesses.



**Graph 3.1:** Shows time it takes to achieve 10% depletion of Boron-10 in boron carbide as a function of packing fraction. Each series represents points for different reflector thicknesses.



**Graph 3.2:** Shows time it takes to achieve 10% depletion of Boron-10 in boron carbide as a function of reflector and shielding thickness. Each series represents points for different boron packing fractions.

#### 3.4.3 Damage to Core Barrel and RPV

Another important aspect of this project was to look at the damage incurred at the core barrel and the reactor vessel (RPV). Radiation damage to the corresponding areas was simulated and calculated for the 900 MWth reactor using MCNP. This was modeled for a total of 1,000,000 particles. Of course, one cannot simply use MCNP's output for the dpa reaction. The MCNP output is in MeV-cm<sup>2</sup> per source neutron, so it is necessary to obtain the source strength before one can find the dpa. The source strength was determined using the reactor power (900 MWth), burn-up for each time step, and an average neutron production rate, and was found to be about 7.02E+19 neutrons per second at each time step. Further, the dpa reaction has an efficiency of 0.8 and it is necessary to convert from barns to cm<sup>2</sup>. The dpa over a time step can be found using the formula:

$$DPA = 0.8 \frac{\sigma}{2E_D} T * S * 10^{-24}$$

Where  $\sigma$  is the MCNP output,  $E_D$  is carbon or iron's displacement energy (31 eV for C (Core barrel) or 40 eV (RPV) respectively), T is the time step length (1 year), S is the source strength, and 10<sup>-24</sup> is the barns to cm<sup>2</sup> conversion factor. The final results for the radiation damage in the core barrel and the RPV are shown in Tables 3.8 and 3.9 below.

Packing	Outer	DPA	DPA	DPA	DPA
Fraction	Reflector	(Core	(Core	(RPV)	(RPV)
	Thickness	Barrel)	Barrel)	per year	over 80
		per year	over 80		years
			years		
0	95	7.64815E-06	0.000611852	0.000148827	0.011906151
0	80	9.088E-06	0.00072704	0.000194129	0.015530283
0	85	4.88326E-06	0.00039066	0.000178687	0.014294934
0	90	1.01321E-05	0.000810565	0.000161852	0.012948177
0.14	40	0.000337886	0.027030884	5.86667E-05	0.004693333
0.14	50	7.95919E-05	0.006367353	5.09489E-05	0.004075916
0.14	60	3.2698E-05	0.00261584	3.68091E-05	0.002944731
0.14	70	1.21116E-05	0.000968925	3.07482E-05	0.002459856
0.14	80	5.96001E-06	0.000476801	1.94573E-05	0.001556582
0.14	85	1.43292E-06	0.000114633	2.00027E-05	0.001600215
0.14	90	1.54535E-06	0.000123628	1.70491E-05	0.001363925
0.14	95	9.31275E-07	7.4502E-05	1.7691E-05	0.001415283
0.3	40	0.000256986	0.02055891	3.94252E-05	0.003154014
0.3	50	0.000139586	0.01116688	2.38994E-05	0.001911952
0.3	60	5.30316E-05	0.004242525	1.75368E-05	0.001402942
0.3	70	5.01495E-05	0.004011957	1.0702E-05	0.000856158
0.3	95	1.01191E-07	8.09525E-06	6.66583E-06	0.000533266
0.3	80	3.46915E-05	0.00277532	8.2264E-06	0.000658112
0.3	85	9.0748E-06	0.000725984	8.12766E-06	0.000650213
0.3	90	1.5391E-07	1.23128E-05	6.78025E-06	0.00054242
0.4	40	0.000242566	0.01940527	3.95986E-05	0.003167885
0.4	50	0.000251251	0.020100083	2.62415E-05	0.002099322
0.4	60	5.40805E-05	0.004326436	8.5879E-06	0.000687032
0.4	70	1.85001E-05	0.00148001	9.55245E-06	0.000764196
0.4	95	3.91059E-06	0.000312847	3.03071E-06	0.000242457
0.4	80	1.19611E-05	0.000956886	4.21666E-06	0.000337332
0.4	85	8.22512E-06	0.000658009	4.0553E-06	0.000324424
0.4	90	8.85148E-06	0.000708118	3.46922E-06	0.000277538
0.5	40	0.000251251	0.020100083	2.62415E-05	0.002099322
0.5	50	9.89743E-05	0.007917944	2.85664E-05	0.002285311
0.5	60	5.02718E-05	0.00402174	4.05084E-08	3.24068E-06
0.5	70	4.71114E-05	0.003768908	5.24602E-06	0.000419682
0.5	95	4.95552E-06	0.000396441	1.4909E-06	0.000119272

**Table 3.8**: Shows values of DPA per year calculated in the carbon-fiber reinforced composite core barrel and the reactor pressure vessel.

0.5	80	3.37715E-05	0.002701717	2.21509E-06	0.000177207
0.5	85	7.37074E-07	5.89659E-05	1.86983E-06	0.000149586
0.5	90	3.82116E-06	0.000305693	1.4909E-06	0.000119272
0.6	40	0.000237421	0.01899369	1.26303E-05	0.001010423
0.6	50	0.000168515	0.013481224	7.5733E-06	0.000605864
0.6	60	6.62706E-05	0.005301647	1.25851E-05	0.001006805
0.6	70	2.34736E-05	0.001877892	2.75907E-06	0.000220726
0.6	95	0.000168515	0.013481224	7.5733E-06	0.000605864
0.6	80	1.805E-05	0.001444	8.71044E-07	6.96835E-05
0.6	85	1.18842E-06	9.50737E-05	1.31974E-06	0.000105579
0.6	90	5.6176E-06	0.000449408	4.10738E-07	3.2859E-05
0.75	40	0.000270921	0.021673715	1.20612E-05	0.000964895
0.75	50	9.63634E-05	0.007709068	2.96397E-06	0.000237118
0.75	60	4.69648E-05	0.003757182	4.4142E-06	0.000353136
0.75	70	3.7575E-06	0.0003006	2.0089E-07	1.60712E-05
0.75	95	1.20366E-06	9.62927E-05	2.98219E-07	2.38575E-05
0.75	80	5.07409E-06	0.000405927	6.88766E-07	5.51013E-05
0.75	85	5.34261E-06	0.000427409	8.95898E-08	7.16718E-06
0.75	90	9.21172E-07	7.36938E-05	6.56266E-07	5.25013E-05

Graphs 3.3 and 3.4 show graphical representations of the DPA calculations conducted. As expected, it can be seen that the DPA decreases as the reflector thickness is increased and as the boron packing fraction is reduced. The trend with regards to the packing fractions is not immediately obvious in the core barrel, but it can be clearly seen in the reactor pressure vessel, where it is clear that as packing fraction goes up, the DPA received goes down.

Boron absorbs thermal neutrons, so does not affect the fast neutron flux significantly, hence we are using thermal values. The reflector thickness should have a stronger effect on the fast neutron flux, however you still see a huge difference in flux at the RPV versus the flux in the core barrel since the key point of the boron is to reduce the neutron flux and damage seen.



**Graph 3.3**: Shows DPA measurements in the Core Barrel over the time period of 80 year.



Graph 3.4: Shows DPA measurements in the RPV over the time period of 80 year.

Highest DPA values seen by the RPV and Core Barrel don't prove themselves to be a factor of concern since they are so low. DPA at the RPV is a whole order of magnitude below the DPA seen by the core barrel. In order to make sure that the DPA values did in fact change logically, the thermal neutron flux seen at the RPV was plotted as a function of reflector thickness. This is shown in Figure 3.5. As it can be seen, the thermal neutron flux goes down as the reflector thickness increases.



**Graph 3.5:** Thermal neutron flux  $(n / s^*cm^2)$  as a function of reflector thickness for various boron packing fractions. A null case was run (0), and as it can be seen, the flux when we replaced the boron with graphite, while maintaining a packing fraction of 14% had a higher neutron flux. This proves that the boron carbide acts as an efficient absorber.

#### 4.0 SUMMARY

Given the materials review and the simulation results, general recommendations can be made for the design of the boron absorber rods and for the design of the shielding system itself.

## 4.1 Boron Absorber Rod Design Suggestions

For the absorber material it is an easy choice to use boron carbide as the main absorber. Considering its popular use in PWRs, boron carbide is well understood and has been studied extensively. As evident from the material review, there are studies observing the behavior of boron carbide under irradiation, its behavior inside and interaction with cladding, and its heat transfer properties. Boron carbide easily fabricated into a pellet form for direct use in a rod design. Pellets are preferable over powder because again, their behavior has been extensively studied. While the pellet form of boron carbide may in fact release more helium than the powder form, the helium release can be controlled through the addition of a vent assembly. A cladding for the boron rods is highly suggested for several reasons. First, it will allow for the implementation of the helium vent assembly as mentioned above. A stainless steel cladding will also protect the fluoride salt coolant from being contaminated with boron carbide, preventing a potentially hazardous situation. Additionally, a cladding will provide mechanical stability to the rods and potentially allow for simple replacement of depleted boron carbide rods.

Recommendations for specific dimensions of cladding thickness, gap width, and pellet dimensions cannot be made at this time until more general shielding specifications, such as the packing fraction and reflector size, have been made.

#### 4.2 Boron Shielding Design Suggestions

For a reactor lifetime, it is desired to have a burn-up of boron carbide no greater than 10%. Based on the MCNP simulation results, a 10% or less burn-up of boron carbide (in 80 years or greater) occurred for conditions in which the boron packing fraction was greater than 50% and the reflector was greater than 50 centimeters thick.

In terms of damage to the RPV and core barrel, every simulation case resulted in a DPA value less than  $10^{-2}$  after 80 years. This is a very small amount of damage and would may result in some slight embrittlement or swelling. The main mechanism for embrittlement in stainless steels is helium generation due to neutron interactsion with nickel. For Alloy N, a potential RPV material, a lifetime thermal and fast neutron dose limit of  $10^{21}$  neutrons/cm<sup>2</sup> has been recommended to prevent any degradation of the material properties.<sup>18</sup>

For PWRs, to limit the amount of tritium diffusion, it is suggested to explore the use of a tritium permeation barrier. Specifically, aluminide has been shown to reduce tritium permeation by a factor of 100-1000. This aluminide coating can be used on the cladding of the control rods in PWRs (potentially on the inside and outside) to dramatically reduce the diffusion of tritium and ultimately its leakage into the coolant.

#### 4.3 Future Directions

To fully implement the boron carbide rods in an FHR shielding system, there is still several more areas which require further exploration. The simulations which were run did not account for the stainless steel cladding as proposed. To be entirely accurate, this cladding would need to be implemented. Additionally, it would be helpful to determine a radial distribution of depletion. The boron carbide rods closer to the core may be subject to higher burn-ups. MCNP also allows for a tally on the evolution of helium. This can be used to calculate a volume of helium generated in both the absorber rods and in the RPV/cladding. Under irradiation, nickel in the steel could potentially create a significant amount of helium as well. This will also provide a better justification for the need of helium vent assembly. If the amount of helium generated is negligible, then perhaps a helium vent assembly would not be needed.

Ultimately the optimal design specifications will be based on a variety of factors that are beyond the scope of this project. The optimal design solution will based on minimizing the evolution of helium, the cost of production, the size of the reactor, and maximizing the lifetimes of the RPV, core-barrel, and shielding. A decision considering all of these factors will ultimately determine the final design for the FHR shielding system.

## **5.0 ACKNOWLEDGEMENTS**

We would like to thank Professor Per Peterson for guidance and aid throughout the course of this design project. We would also like to thank Tommy Cisneros for helping extensively with the MCNP simulations.

#### 6.0 REFERENCES

<sup>10</sup> Atobe, Kozo, Makoto Honda, Munetoshi Ide, Hiromichi Yamaji, Tokuo Matsukawa, Noboru Fukuoka, Moritami Okada, and Masuo Nakagawa. "Point Defects in Cubic Boron Nitride after Neutron Irradiation." *Japanese Journal of Applied Physics* 32.Part 1, No. 5A (1993): 2102-104.

<sup>&</sup>lt;sup>1</sup> Marshield, Custom Radiation Products. http://www.marshield.com/page/borated-polyethylene-neutron-shielding

<sup>&</sup>lt;sup>2</sup> Roy,S., Sandison, G. Shielding for neutron scattered does to the fetus in the patients treated with 18 MV x-ray beams. *The International Journal of Medical Physics Research and Practice*. **27**:8, June 2000.

<sup>&</sup>lt;sup>3</sup> Yu J., Chen Y., Elliman R., Petravic M. Isotopically Enriched <sup>10</sup>B Nanotubes. *Advanced Materials*. **18**, 2157-2160 (2006).

<sup>&</sup>lt;sup>4</sup> Thevenot, F. "A Review on Boron Carbide." Key Engineering Materials 56-57 (1991): 59-88.

<sup>&</sup>lt;sup>5</sup> Froment, K., D. Gosset, M. Guery, B. Kryger, and C. Verdeau. "Neutron Irradiation Effects in Boron Carbides: Evolution of Microstructure and Thermal Properties." *Journal of Nuclear Materials* 188 (1992): 185-88.

<sup>&</sup>lt;sup>6</sup> Ashbee, K. "Defects in Boron Carbide before and after Neutron Irradiation." *Acta Metallurgica* 19.10 (1971): 1079-085.

<sup>&</sup>lt;sup>7</sup> Jostsons, A., C. Dubose, G. Copeland, and J. Stiegler. "Defect Structure of Neutron Irradiated Boron Carbide." *Journal of Nuclear Materials* 49.2 (1973): 136-50.

<sup>&</sup>lt;sup>8</sup> Stoto, T., N. Housseau, L. Zuppiroli, and B. Kryger. "Swelling and Microcracking of Boron Carbide Subjected to Fast Neutron Irradiations." *Journal of Applied Physics* 68.7 (1990): 3198.

<sup>&</sup>lt;sup>9</sup> Jostsons, A., and C.k.h. Dubose. "Microstructure of Boron Carbide after Fast Neutron Irradiation." *Journal of Nuclear Materials* 44.1 (1972): 91-95.

<sup>&</sup>lt;sup>11</sup> Homan, F.J. "Performance Modeling of Neutron Absorbers." ANS Nuclear Society Meeting, April 24-27, 1972.

<sup>&</sup>lt;sup>12</sup> Kaminaga, F., Sato S., Okamoto Y. "Evaluation of Gap Heat Transfer between Boron Carbide Pellet and Cladding in Control Rod of FBR." *Journal of Nuclear Science and Technology*. 29:12 (2012), 121-130

<sup>&</sup>lt;sup>13</sup> Fuels and Materials Development Program Quarterly Progress Report, Oakridge National Laboratory. June 1972.

<sup>&</sup>lt;sup>14</sup> McGuire, Joseph. "Vent assembly including an air lock." US Patent No. 3948628. April 6, 1976.

<sup>&</sup>lt;sup>15</sup> Hollenberg, G., E. Simonen, G. Kalinin, and A. Terlain. "Tritium/hydrogen Barrier

Development." Fusion Engineering and Design 28 (1995): 190-208.

<sup>&</sup>lt;sup>16</sup> Forcey, K.s., D.k. Ross, and C.h. Wu. "The Formation of Hydrogen Permeation Barriers on Steels by Aluminising." *Journal of Nuclear Materials* 182 (1991): 36-51.

<sup>&</sup>lt;sup>17</sup> J. Sumita et. al. "Reactor Internals Design" Nuclear Engineering and Design 233 (2004) 81–88

<sup>&</sup>lt;sup>18</sup> "Two-Fluid Molten-Salt Breeder Reactor Design Study (Status as of January 1. 1968) Oak Ridge National Laboratory, ORNL-4528, Section 3.3. August 1970.