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ALTERNATE SHIELD MATERIAL FEASIBILITY

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**ALTERNATE SHIELD
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DOE Research and Development Report

E. R. Specht

L. B. Levitt

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Department of Energy

Division of Reactor Research and Technology,

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ABSTRACT

The feasibility and cost/benefit of using materials other than stainless steel for in-vessel neutron shielding in large LMFBRs were investigated. Canned vibratorally compacted B_4C powder shields were found to be much more economical than stainless steel (a savings of \$1.1M in loop plant designs and \$9.4M in pool plant designs).

The helium gas pressure buildup in B_4C shields placed around LMFBR in-vessel components (direct reactor heat exchangers in a loop reactor and intermediate heat exchangers in a pool reactor) would only be 0.04 atm after 40 yr of reactor operation (with 80% dense powder). The irradiation-induced swelling of the B_4C would only be 0.002%. No adverse reactor impact would occur if the B_4C escaped from the B_4C shields.

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1.0 CONCLUSIONS AND RECOMMENDATIONS

Local component B_4C shielding should be used in large commercial-sized LMFBRs instead of stainless steel neutron shielding. B_4C shielding is less costly and much lighter, as shown in Table 1. B_4C has no significant adverse effect on the reactor's performance. The largest B_4C grains are one-third the size of the pump bearing gap and one-sixth the size of the small fuel assembly gap. In addition, the grains would settle out before reaching these gaps. If all of the B_4C in the DRHX shield were to be accidentally released into a loop reactor's sodium inventory and uniformly distributed, it would only reduce the core's reactivity by 1.2%. If all of the B_4C in an IHX shield in a pool reactor were released, it would only reduce the core's reactivity by 4.5%.

TABLE 1
 B_4C AND STAINLESS STEEL SHIELD COMPARISONS FOR
 LOOP AND POOL REACTOR DESIGNS

	304 SS Shield		B_4C Shield in 304 ⁺ SS Containers (Locally Around DRACS and IHXs)
	Outside Core Barrel	Locally Around Component	
Loop reactor DRACS shield			
Thickness (in.)	30	9.6	0.56 (0.31 B_4C , 0.25 SS)
Weight (lb)	1,019,000	50,300	1208 (232 of B_4C) ^a
Cost (\$)	10,190,000	1,220,000	98,840 ^a
Pool reactor IHX shields			
Thickness (in.)	32	-	0.83 (0.32 B_4C , 0.50 SS) for IHX
Weight (lb)	1,087,000	-	26,836 (4,424 of B_4C) ^b
Cost (\$)	10,870,000	-	1,505,680

^aFor one DRHX.

^bTotals for four IHXs and two DRACS shields.

A 40-yr B_4C shield lifetime appears to be quite reasonable under the benign operating conditions that would be present at the position of the B_4C shields. Earlier concerns that a B_4C shield would build up a high gas pressure (due to helium generation) were unfounded.

Since B_4C control assembly absorbers are universally used in LMFBR cores and since control assemblies must accommodate much more severe temperature, space, and fluence requirements than in-vessel B_4C shields would experience, the use of B_4C within in-vessel shields presents no significant new concerns. Key B_4C shield and control absorber parameters are compared in Table 2.

TABLE 2
COMPARISON OF B_4C CONTROL ABSORBER
AND SHIELD PARAMETERS

Parameter	Control Absorbers	DRHX Shield
Average total neutron flux (neut/cm ² μs)	1.6×10^{15}	1.6×10^9
Neutron captures/cm ³ of B_4C	1.3×10^{22}	3.8×10^{18}
Helium gas pressure (atm)	1215 ^a	0.04 ^b
B_4C grain size (μm)	10-20	25 and 250
B_4C swelling (% ΔD)	7.6	0.002
Absorber geometry	Pellets	Powder
B_4C density (g/cm ³)	2.3	2.0
Absorber lifetime (yr)	3	40
σ_a , B-10 (barns)	3.5	138
B_4C temperature (°F)	1500	650
SS clad thickness (in.)	0.06	0.125
Sodium coolant velocity (ft/s)	5.0	0.006

^aTheoretical gas pressure level if the design used a 50% gas plenum volume for a 3-yr lifetime. The actual absorber design used vented absorbers.

^bPressure with unsintered, 80% dense B_4C , 40-yr shield lifetime.

Although B_4C has been well qualified for use in LMFBRs as control absorbers, a small amount of R&D would be desirable to qualify it for use as a shield material. The recommended R&D tasks, such as EBR-II irradiations and neutronic experiments in the Tower Shield Facility, are described in Section 7.0.

2.0 OBJECTIVES OF INVESTIGATION

Determine the desirability and feasibility of using in-vessel shielding materials other than stainless steel in commercial-sized LMFBRs (loop and pool designs). Identify the R&D needed to qualify alternate shield materials for use in large LMFBRs.

3.0 SHIELDING DESIGN CRITERIA

The neutron activation of the secondary sodium within the Direct Reactor Heat Exchanger (DRHX) of a loop reactor and an intermediate heat exchanger (IHX) of a pool reactor must be kept to an acceptably low level in order to limit the radiation dose that plant personnel will receive from a loop reactor's auxiliary secondary sodium to about 15 mR/h in accessible areas and to 3 mR/h for the secondary sodium of pool reactor designs. At present, this is accomplished by placing neutron shielding around and near the DRHX (as shown in Figures 1 and 2) and around the core or around the IHXs and DRHXs of a pool reactor design (typical pool reactor layouts are shown in Figures 3 and 4).

The criteria of 15 mR/h and 3 mR/h in the accessible areas of loop and pool reactors, respectively, translate into doses at the surface of the DRHX and IHX secondary sodium pipes of 100 mR/h and 20 mR/h. Since an uncertainty factor of four was applied to the calculated secondary sodium activations, design values of 25 mR/h and 5 mR/h were used for the loop and pool reactors, respectively.

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Figure 1. Large-Scale Prototype Breeder

Figure 2. Local DRHX Shielding of Stainless Steel

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Figure 3. Pool Reactor Concept

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Figure 4. Top View of Pool Reactor

4.0 IN-VESSEL SHIELDING CANDIDATES

The shielding materials most attractive for use as LMFBR in-vessel shields are stainless steel, B_4C , borated stainless steel, carbon-bonded B_4C , plasma-sprayed B_4C , graphite, and resonance absorbers (Mo, Ag, Eu, Ta, Re, etc.).

Until now, in-vessel shields used stainless steel to decrease the neutron flux impinging on various ex-core components. Although it is quite compatible with sodium, it is not very efficient as a neutron shield and is quite heavy and expensive in terms of material and seismic support structure costs.

Vibratorally compacted B_4C shields were concentrated on because they appear to be the most promising from the standpoints of both cost and weight. In-vessel shields of hot pressed B_4C , borated stainless steel, carbon-bonded B_4C , plasma-sprayed B_4C , graphite, and resonance absorbers may also be useful for special applications within large LMFBRs, but they were not investigated in detail during this study. Some work will be performed on these absorbers during FY 1984. The main conclusions regarding the various in-vessel shield candidates are as follows:

- Stainless steel (SS) was previously used for in-vessel shielding because it had already been used throughout LMFBRs for many purposes.
- B_4C has the highest absorption cross section of the common poison materials, and it is light and inexpensive. These characteristics are discussed below in more detail.
- Borated SS is about as heavy as normal SS. It is a good neutron attenuator, but it becomes difficult to shape and very brittle when the boron content exceeds 4 wt. %.¹ Although 10 wt. % boron in SS has been produced, its mechanical properties are similar to those of fire brick. The cost of 2 wt. % boron in SS was \$6 to \$10/lb in 1973.² When the atomic burnup of boron is 5% in 3 wt. % boron-stainless steel, it has been shown that the thickness change is only 0.6% and that

there is essentially no helium gas release from the irradiated material.³ SS with 2 wt. % boron can, like other steels, be fabricated into various shapes and welded. Therefore, it could be used for in-vessel shielding applications and should be about 3.5 times as effective as normal SS in attenuating neutrons. Although this may be adequate for some LMFBF applications, large components such as the DRHX in loop reactor designs and the IHXs in pool reactor designs can be shielded by thin, light containers filled with less expensive B₄C.

- Carbon-bonded B₄C shields are prepared as follows: Carbonized B₄C mixed with a phenolic resin is molded into complex configurations for special shielding applications. The resin is then baked out. About 50% (by volume) of the material is B₄C. This material is used by GA Technology in the Fort St. Vrian gas-cooled reactor. This material may be useful in LMFBFs but would still be more expensive than vibratorally compacted B₄C powder.
- Plasma-sprayed coatings of an alumina and boron carbide composite can be applied to metal surfaces. Coatings up to 0.1 in. thick can be applied but are more reliable in thicknesses of 0.020 to 0.040 in. A 0.020-in. coating costs about \$1/in.². Therefore, it would be too expensive for large components (about three times the cost of a vibratorally compacted B₄C powder shield). For situations where very little space is available or complex geometries need to be shielded, it may be quite useful. However, its long-term performance under LMFBF operation would need to be investigated.
- Graphite is effective as near core shielding to reduce the energy of neutrons that leak from the core of an LMFBF. This enhances the parasitic absorption of the sodium between the active core and the components of interest. Using graphite instead of SS inside the core barrel of an LMFBF only reduces the activation of the secondary sodium in an in-vessel DRHX by a factor of two. Studies were not performed with graphite placed around in-vessel components such as the DRHX since the neutron flux near the DRHX is already very soft (the average absorption energy is about 23 eV), and the neutron absorption cross section of graphite is very low. Also, graphite swells when irradiated, particularly if it is exposed to sodium. For fast fluences of 3×10^{20} neutrons/cm² and sodium soak times of 10⁴ h, graphite swells by about 8%.⁴ Recent material tests in EBR-II show similar results. Although graphite is normally sealed in steel cans when it is used in LMFBFs, leaks can occur. Therefore, it should only be used where the fluence is low or a lot of room is available for expansion.

- Resonance absorbers such as Mo, Ag, Eu, Ta, and Re were not investigated since they are much more expensive than B_4C and are generally not as effective as B_4C in absorbing neutrons. However, for situations in which the helium gas produced in B_4C when it absorbs neutrons would be a problem, these absorbers, which emit only gamma rays when they absorb neutrons, may be useful.

5.0 DESIGNS STUDIED

A 3500-MWt loop reactor with one direct reactor auxiliary cooling system (DRACS), as shown in Figures 1 and 2, was studied to determine the most cost effective way to limit the activation of the secondary sodium within the DRHX within the reactor vessel. A 3500-MWt pool reactor with four IHXs and two DRACS (as shown in Figures 3 and 4) was also studied. Earlier studies with SS placed around the core barrel showed that the cost and weight of these shields would be substantial (as shown in Table 1). These heavy shields also had an adverse effect on the seismic response of the reactor when designing to accommodate safe shutdown earthquake (SSE) and operating basis earthquake (OBE) events. For these reasons, the stainless steel shields were placed locally around the DRACS in the loop reactor design. Since the neutron flux spectrum is much softer at this location than near the core, steel is more effective as a neutron shield. Thus, the DRHX shielding is much lighter and cheaper (as shown in Table 1) than near-core steel shields. For pool reactors, it is not practical to place local steel shielding around all of the IHXs and DRHXs (shown in Figures 3 and 4) because the total amount of steel (and its cost) would be greater than the amount needed around the core barrel.

Since B_4C is much more absorptive than stainless steel, this study concentrated on its use as an in-vessel shield for loop and pool designs. Earlier studies had shown that replacing a row of SS reflectors immediately inside of the core barrel with a row of B_4C assemblies would only decrease the total fluence on the core barrel by a factor of five. Therefore, efforts were concentrated on the use of B_4C near the components of interest.

The design arrived at for a local DRHX shield is shown in Figure 5 and Table 3. The weight and cost of this shield are shown in Table 1. For a pool reactor design, shadow shields (as shown in Figure 5 for a DRHX) would not be necessary for the IHXs. A 151-in.-long annular steel container with a 0.32-in. annulus of B_4C would be welded on the lower section of each IHX (opposite the tube bundle section). These local B_4C shields would only extend halfway (180°) around each IHX. The type of IHX being thus shielded is shown in Figure 6. The total weight and cost for locally shielding the IHXs and DRHXs of a pool reactor are shown in Table 1.

TABLE 3
 B_4C SHIELD DESIGN CHARACTERISTICS
 FOR LOOP REACTOR DRHX SHIELD

B_4C thickness	0.31 in.
B_4C density	80% of theoretical
Two separate grain sizes of powder vibratorally compacted	0.001 in. and 0.01 in.
Shape of B_4C grains	Irregular
Wall of B_4C container	304 SS
Wall thickness	0.125 in.
Container inspection	Radiograph all container welds and radiographically check B_4C uniformity

Although further optimization of the local B_4C shielding for the loop and pool designs is still needed, it appears clear from Table 1 that this approach is the most desirable.

Vibratorally compacted B_4C powder was selected for these shield designs because it is much less expensive than hot-pressed B_4C shapes (which usually require expensive grinding operations). A B_4C density of about 80% can be

Figure 5. DRHX Shield Design

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Figure 6. Pool Reactor Intermediate Heat Exchanger

easily obtained for vibratorally compacted material by using two grain sizes that are a factor of 10 apart (such as 25 and 250 μm).⁵ The coarse material is first poured in, then the container is vibrated while the finer material is being added. B_4C densities of 80 to 83% can be readily achieved by using three grain sizes. By using spherically rather than irregularly shaped particles, densities of 85 to 88% can be obtained. The stainless steel containers should be filled in several steps (axially) to assure good compaction over their whole height.

For most of the shield sections (shown in Figure 5), a small amount of B_4C settling during the 40-yr lifetime of the reactor would not significantly affect its shielding effectiveness. However, it would be undesirable to have a large gap at the top of the shadow shield shown in Figure 5 since this region is almost opposite the sodium outlet "windows" of the DRHX. Therefore, a small reservoir of B_4C was placed at the top of the shadow shield.

6.0 BEHAVIOR AND REACTOR SYSTEM IMPACT OF B_4C SHIELDS

6.1 Analytical Techniques

The analyses for this study were performed with the MORSE multigroup Monte Carlo code. The multigroup library used was the ORNL 51 neutron group shielding library. Deep penetration experiments through SS and sodium at the Tower Shield Facility (TSF) at ORNL have shown that the total neutron flux calculated with this library should be accurate to within about a factor of two for the sodium and steel configurations of interest in this study. Since B_4C shields behind large amounts of sodium were not studied in the TSF experiments, the uncertainty factor of four that was applied to the calculated results appears to be reasonable.

The main results of the MORSE code calculations for the DRHX shield are shown in Table 2. B_4C average atomic burnup was only 0.024%, the peak heating rate in the B_4C shield was only 0.0044 W/cm^3 , and the total heat generated in the shadow shield was 28.5 W. The energy at which the average boron neutron absorption took place was only 23 eV.

6.2 B_4C Helium Gas Pressure and Swelling

The helium gas generation rate in the B_4C is the same as the neutron captures/ cm^3 shown in Table 2. Most of the helium is retained within the B_4C , particularly at low burnups and temperatures.

Extensive gas release and B_4C swelling experiments have been performed for an LMFBR environment, mainly in EBR-II. These are summarized in Ref. 6. These data indicate that, for hot pressed B_4C pellets, only 0.0038% of the helium would be released at 3.8×10^{18} captures/ cm^3 in B_4C , and the diametral swelling would be 0.0008%.

However, the gas release and swelling of B_4C is somewhat different for compacted powder samples and thermal neutron environments. These data were best summarized in Refs. 7 and 8. For the conditions of interest (0.024% average burnup and 650°F operating temperature), the helium gas release fraction for the vibratorially compacted B_4C shields previously described would be 6.5% (with an uncertainty factor of 1.5).

From Refs. 7 and 8 (as well as older literature), it was determined that the most probable linear swelling for the B_4C in the shields of interest would be 0.002% over the 40-yr operating life of the reactor.

The helium gas release fraction and B_4C swelling that would occur for the flux spectrum of interest (23 eV for the average B_4C absorption) would probably be closest to that found in thermal reactors but might lie between the values found in fast and thermal reactor environments.

If sodium enters the B_4C through a break in the SS cladding, little if any swelling should occur. Some data indicate the possibility of 1% swelling due to sodium logging.

6.3 B_4C -Stainless Steel Compatibility

If the containers holding the B_4C were to fail accidentally resulting in B_4C entering the sodium, little B_4C would remain in the sodium since it is fairly insoluble. Most of the B_4C would settle to the bottom of the reactor vessel and remain there.

Experiments with B_4C held in contact with stainless steel at various temperatures were reported in Ref. 9. From these data, it was determined that the B_4C would only cause minor surface diffusion of the reactor vessel to a depth of 3.2×10^{-6} in. in 40 yr.

6.4 Possible System Effects of B_4C

If the grains of B_4C entered the sodium through a break in the SS container, the most serious consequences would occur if they plugged small clearance passages. The smallest clearance passages in the primary sodium circuit are the stellite-coated pump bearing gap (710 μm radially) and the gaps between fuel pins in the active core (1700 μm wire wrap OD). The largest grain size of B_4C powder is only 250 μm . Therefore, no plugging problems are anticipated.

If all of the B_4C in the shielding in the loop reactor design was released into the primary sodium coolant inventory of 5.5×10^6 lb and uniformly suspended, there would be about 53 ppm of B_4C in the sodium. This would decrease the reactivity of the core by only 1.2%. In a pool reactor design, releasing all of the B_4C in one of the IHX shields would only decrease the reactivity of the core by 4.5%. In reality, the shields must be constructed in several sections. Therefore, the reactivity effect would be much lower.

Since an excess carbon content of 3% can be present in B_4C , the carbon impurity level in the sodium could theoretically be raised by 1.7 ppm. The impact on carborization of steel from the excess carbon would be negligible since the carbon impurity level in the sodium can be as high as 30 ppm during normal operation.

The lithium generated during irradiation of the B_4C could also be released to the primary sodium. However, even if all of the lithium could be released to the primary sodium, the lithium content of the sodium would increase by 1.3 ppm, compared to the allowable limit in the sodium of 5 ppm.

6.5 Impact of Shield Loss on Secondary Sodium Activation

Loss of all of the B_4C from the shield would increase the radiation level from the secondary sodium by a factor of 17. However, the inner DRHX shield is attached to the DRHX and could be removed for repair, as could the IHX shield. Loss of the B_4C from the shadow shield of the DRHX would increase the radiation level from the secondary sodium by a factor of 2.7. If repair or replacement of the shadow shield proved to be too difficult, this higher dose could be accommodated by placing lead blankets around the DRACS piping or by increasing the exclusion distance around the DRACS piping.

6.6 Higher Design Reliability Options

Several approaches for further assuring a 40-yr integrity of the B_4C shield (particularly the shadow shield of the DRHX) would be to use a thicker steel container, double-canning the B_4C so that it has an outer guard can around the canning that holds the B_4C , or putting the B_4C in several smaller containers. None of these measures would greatly increase the cost of the B_4C shields.

7.0 POTENTIAL R&D NEEDS

A list of potential R&D needs was developed for in-vessel shields composed of vibratorally compacted B₄C powder. This list is given below.

These are as follows:

<u>Concern</u>	<u>Proposed Test</u>
No irradiation data exist for B ₄ C powder samples in intermediate neutron spectra.	Perform EBR-II tests with samples in intermediate spectrum.
No neutronics measurements exist with B ₄ C regions placed in intermediate neutron spectra.	Perform TSF tests with B ₄ C slabs beyond a thick region of sodium.
Vibratorally compacted B ₄ C powder shield may further compact after reactor operation due to thermal cycling and from reactor vibration.	Perform out-of-reactor temperature cycling of a typical B ₄ C shield component as well as shake table tests which simulate normal operating and seismic conditions.
No vibratorally compacted B ₄ C powder components have been made that are as large as needed. The density achievable is somewhat uncertain.	Fabricate a typical B ₄ C shield container and run B ₄ C powder densification experiments with two mesh and three mesh sized combinations of B ₄ C powder.
No experiments exist for a faulted container of B ₄ C powder in a sodium tank. Some swelling may occur or the B ₄ C powder may leak out.	Use out-of-pile tests in a stagnant tank of sodium to determine the behavior of the B ₄ C if the SS container fails.
Large amounts of B ₄ C in the sodium coolant may have undesirable long-term effects, but little data exists on B ₄ C solubility in Na.	Run B ₄ C solubility tests in Na versus temperature.
Must verify that B ₄ C particles that are entrained before settling can occur do not build up in fuel and bearing passages.	Run sodium test under prototypic flow conditions and introduce B ₄ C particles simulating worst expected situation.

8.0 SUMMARY

Canned, vibratorally compacted B_4C powder (natural boron) shields placed locally around LMFBF in-vessel components are much lighter and less expensive than stainless steel shields. A loss of integrity of the B_4C container would not significantly affect the performance or safety of the reactor.

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