

HFIR Thermal Neutron Beam Guide Hall Upgrade

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**Undergraduate Entry
Group 2
American Nuclear Society
Student Design Competition**

May 5, 1997

**Faculty Advisor
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**The University of Tennessee
Nuclear Engineering Department**

UNIVERSITY HONORS PROGRAM

SENIOR PROJECT - APPROVAL

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Beam Guide Hall Upgrade

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1996-97

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"There comes a time in the course of every project when you must shoot the engineers and begin work." - Anon.

ABSTRACT

The High Flux Isotope Reactor (HFIR) Thermal Neutron Beam Guide Hall Upgrade is currently being conducted in order to provide experimental capability that was lost when the Advanced Neutron Source project was canceled. The number of thermal neutron beam experimental stations at the HFIR HB-2 Beam Tube is proposed to increase from one to five. This increase requires an upgrade to the shielding surrounding the existing HB-2 Beam Tube. The design strategy for this upgrade is divided into five phases: preliminary design, shielding wedge determination, cavity plug composition, biological shield determination, and final dose and thermal loading calculations using Two Dimensional Discrete Ordinates Neutron/Photon Transport Code (DORT) and Monte Carlo Neutron Photon Simulation Code Version 4a (MCNP4a). A variety of shielding materials are modeled in multiple computer runs to determine the optimum compositions for the three shielding elements while at the same time meeting the design constraints. The proposed wedges are composed of borated polyethylene 1.4 meters long beginning 5 meters from the center of the core and extending out. The shielding plug is made of Oak Ridge concrete and borated polyethylene. The biological shield is composed 70 cm of Oak Ridge concrete and 45 cm of borated polyethylene. The final dose rate 30 cm from the external surface of the shield is 1.2 mRem/hr which is less than the maximum allowable dose rate of 2.5 mRem/hr.

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1.0 INTRODUCTION

1.1 History of the High Flux Isotope Reactor ¹²

The High Flux Isotope Reactor (HFIR) was built at the Oak Ridge National Laboratory (ORNL) in the mid 1960's as a 100 megawatt light-water reactor primarily intended to perform neutron-scattering research. The HFIR is currently used for neutron-scattering research, medical and industrial isotope production, materials research, transplutonium isotope production, materials irradiation, and neutron activation analyses.

The HFIR's conception occurred at the former Oak Ridge School of Reactor Technology. The flux trap research reactor was a design concept suggested to colleagues by Jimmy Lane, a former division director, as a thesis problem. During the height of the cold war the former Soviet Union announced plans for a reactor to produce transplutonium isotopes. Glenn Seaborg, then chairman of the Atomic Energy Commission, was appropriately concerned. Dick Cheverton and Lane told Seaborg they "knew exactly what to do. We had the plans for a better one."

The HFIR operated at 100 megawatts of power until 1986 when, in a wave of research reactor shutdowns following the Chernobyl reactor accident, the HFIR was shutdown due to safety concerns. Specimens of vessel material showed that there was more radiation-induced embrittlement than was at first thought to be possible. However, HFIR experts determined that the reactor's life could be extended with "periodic hydrostatic proof testing and continuation of the vessel-materials surveillance program" and operation at a lower power level of 85 megawatts.

Due to the cancellation of the Advanced Neutron Source Program, the HFIR will be refurbished with \$50 million worth of improvements and upgrades. The reactor will return to 100 megawatts of power, and two new neutron beam guide halls are to be added as well as a cold neutron source.

1.2 Benefits of the HFIR Neutron Beam Guide Hall Upgrades¹⁴

Researchers from all over the world are looking forward to the increased capacity of the HFIR to perform thermal neutron scattering research. Scientists are going to look deeper into the molecular structure of materials than ever before possible. This is because neutrons of a given energy have a specific wavelength, with thermal neutrons having a longer specific wavelength than fast neutrons. When thermal neutrons hit atoms of a certain material, they rebound off the atoms to reveal the atom's structure. The trace of an incident neutron's bounce with a certain angle in a certain direction provides a picture of the structure of the target atom. Complex materials such as polymers or organic molecules can be studied more thoroughly using this method.

Other U.S. facilities providing thermal neutron scattering reactor capabilities for experimentation are The High Flux Beam Reactor at Brookhaven National Laboratory in New York, and The National Institute of Standards and Technology reactor in Maryland. Overseas, the Institute Laue-Langevin (ILL) near Grenoble, France provides another research reactor for neutron scattering experiments. With the improvements currently scheduled for the HFIR, it

should become the "world's best" having the western world's largest thermal flux available for research.

1.3 Description of Current Neutron-Scattering Experiment Capability ¹¹

The HB-2 radial beam tube orientation with respect to the reactor core is shown in Figure 1.1. The current beam tube radial inner diameter is 10.16 cm. The length of the current beam guide in the reactor vessel extends 6.4 meters from the center of the reactor. See Tables 1.1 and 1.2 for neutron flux information for the first 5 meters of the current beam guide. The current facilities are limited to one experimental station.

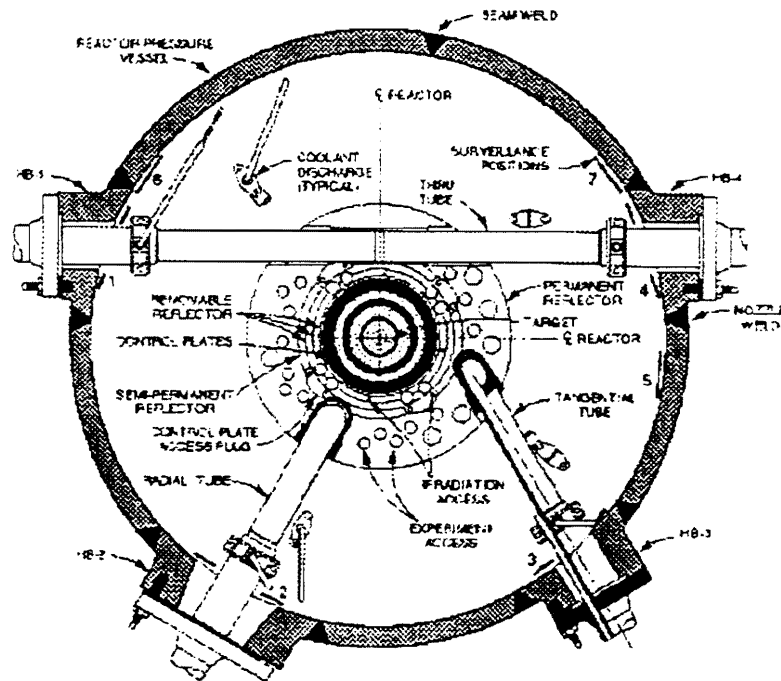


Figure 1.1 Cross sectional view of reactor core.

Table 1.1 Guide Flux as a Function of Energy for the HFIR Thermal Guide Hall Before Upgrade ¹¹

Neutron Energy (MeV)	Guide Flux TG1 (N/cm²*s)	Guide Flux TG2 (N/cm²*s)	Guide Flux TG3 (N/cm²*s)
10.0	6.636E7	6.670E7	6.668E7
20.0	3.618E7	3.855E7	3.853E7
30.0	2.552E7	2.773E7	2.772E7
40.0	1.972E7	2.209E7	2.208E7
50.0	1.605E7	1.846E7	1.846E7
60.0	1.352E7	1.577E7	1.576E7
70.0	1.182E7	1.387E7	1.386E7
80.0	1.046E7	1.242E7	1.241E7

Table 1.2 Guide Flux as a Function of the Distance from the Source to the Start of the Guide for the HFIR Thermal Guide Hall Before Upgrade ¹¹

Guide Distance (cm)	Guide Flux (N/cm²*s)
100.0	3.844E7
200.0	3.868E7
300.0	3.853E7
400.0	3.842E7
500.0	3.835E7

1.4 Description of Upgraded Neutron-Scattering Experiment Capability ¹¹

The orientation of the beam tube will not change, but its radial size will increase from a 10.16 centimeter inner diameter to a tube which expands from a 20.32 centimeter to 58.42 centimeter inner diameter. The thermal guides will extend through the current wall of the beam room into the new thermal guide hall facility as seen in Figure 1.2. The expected neutron flux

within the expanded beam port is presented in Table 1.3. Future experimental capability will be expanded from one station to five stations. Figure 1.3 shows a typical thermal beam guide.

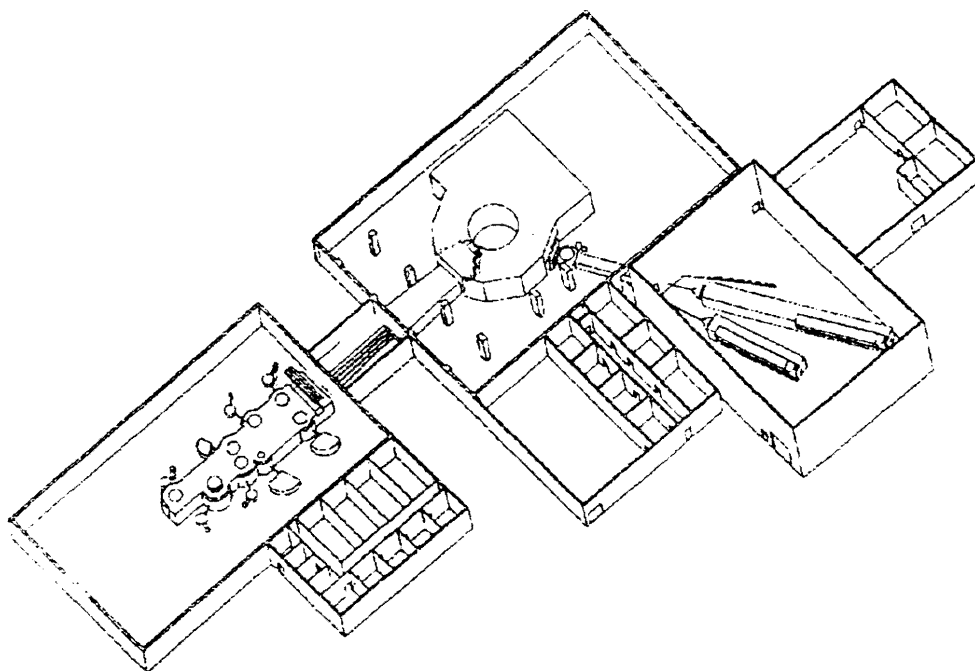


Figure 1.2 HFIR HB-2 Thermal Neutron Guide Hall

Table 1.3 Guide Flux from a Cylindrical Thimble

Thimble Diameter (cm)	Guide Label	5-15 MeV	5-15 MeV	75-85 MeV	75-85 MeV
		Guide Flux (N/cm ² *s)	%Loss	Guide Flux (N/cm ² *s)	%Loss
10.16	TG1	8.770E8	0.00	8.812E7	0.00
	TG2	1.038E9	0.00	1.246E8	0.00
	TG3	1.266E9	0.00	1.469E8	0.00
20.32	TG1	1.110E9	9.76	1.140E8	10.94
	TG2	1.325E9	0.01	1.624E8	0.00
	TG3	1.397E9	0.00	1.782E8	0.00

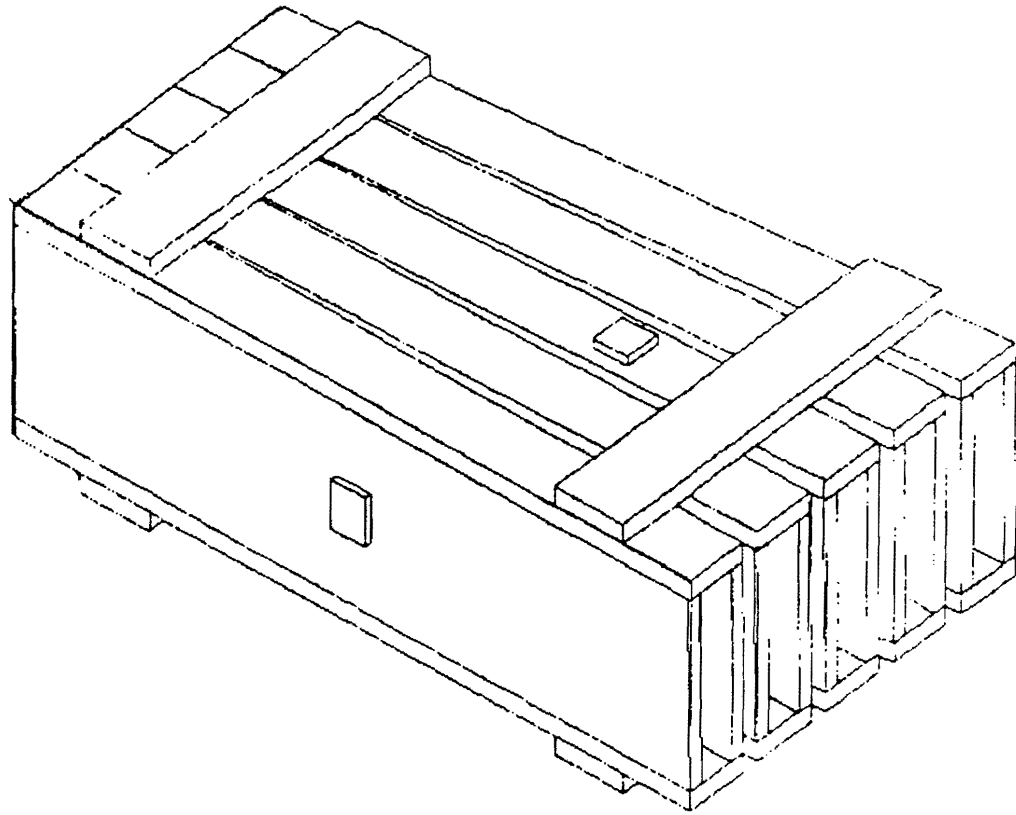


Figure 1.3 Typical Thermal Beam Guide Assembly

2.0 PROJECT DEFINITION

2.1 Design Constraints

The student design project definition (see Appendix A) for the HFIR HB-2 neutron beam tube upgrade was provided by ORNL. The shielding design for this upgrade is divided into three basic components. First, shielding is placed between the thermal neutron guides which divide the neutron flux into five separate beams. Second, the cavity plug is the shielding surrounding the beam tube inside the opening through the existing biological shield. Finally, in order to reduce the dose rate to personnel in the beam room below a maximum allowable dose rate (MADR) of 2.5 mRem/hr, biological shielding is placed around the beam tube through the existing beam room. Figure 2.1 shows the overall layout of the HB-2 upgrade.

The design of the shielding wedges is a function of composition since the wedges are limited to the space between the beam guides. Their primary objective is to minimize the fast neutron and high energy photon flux that travels between the thermal guides outside the existing biological shield. Heat loads in the wedges must not require a secondary cooling system, and the activation of the shielding material should be negligible after two days to facilitate maintenance activities.

The cavity plug assembly shield design is also a function of composition since the shape is defined by the existing opening in the biological shield. A clearance of 1/8" is required between the shield and the inside surface of the reactor vessel wall. The primary objective of the cavity

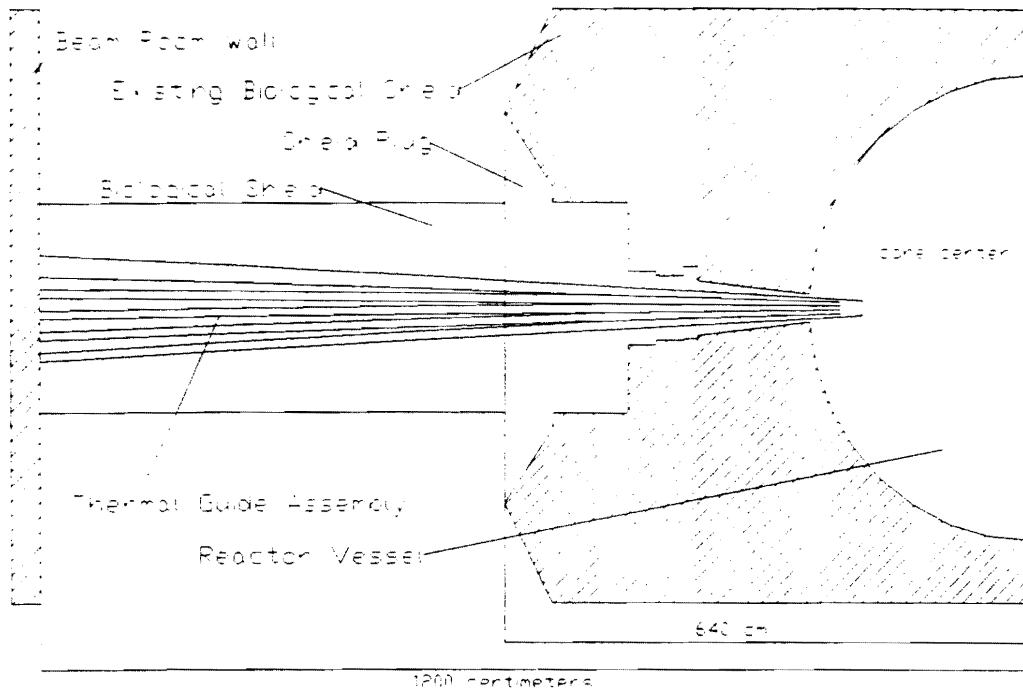


Figure 2.1 HB-2 upgrade layout.

plug is to minimize streaming of radiation from the beam guide assembly into the existing beam room and the existing biological shield. The shield plug aids in the prevention of thermal gradients in the concrete of the existing biological shield resulting from energy deposited by particle interactions in the beam tube. A temperature gradient of one degree Celsius per foot in concrete can cause thermal stresses resulting in cracking.

The biological shield surrounding the beam tube in the existing beam room must reduce the dose rate to personnel to 2.5 mRem/hr at a distance 30 cm from the surface of the shield. The shielding used must not exceed the maximum floor loading of 2500 pounds per square foot. The ceiling of the beam room is approximately 16 feet high, and it is assumed that the upper 8 feet of the room is reserved to allow foot traffic over the shield with 4 feet available for shielding above the beam tube and 3 feet available below the beam tube.

2.2 Design Strategy

In order to complete the goals outlined by the project definition, a design strategy consisting of five design phases was developed. The design phases are:

- Preliminary Studies
- Shielding Wedge Design
- Cavity Plug Design
- Biological Shielding Design
- Final Dose and Thermal Loading Calculations

2.2.1 Preliminary Studies

The objective of this phase is to determine the parameters to be considered in the other four phases of the design. Potential shielding materials are evaluated in a review of literature. The geometry of the beam facility and potential shielding geometries are determined.

2.2.2 Shielding Wedge Design ²

The objective of this phase is to determine the material composition and the length of the shielding material between the thermal guides. The shielding wedges are optimized using Two-Dimensional Discrete Ordinates Neutron/Photon Transport Code (DORT) calculations. The

model calculates the dose rate at the edge of the existing biological shield. The calculated dose rates are used to determine the effectiveness of each shielding material.

2.2.3 Cavity Plug Design ³

The objective of this phase is to determine the material composition of the shielding surrounding the neutron guides at the opening of the existing biological shield. Shielding effectiveness is determined by Monte Carlo Neutron Photon Transport System Version 4a (MCNP4a) calculations performed both with and without the cavity plug in place.

2.2.4 Biological Shielding Design ⁴

The objective of this phase is to determine the material composition and thickness of the biological shielding surrounding the beam guide where it passes through the current beam room. The external dose to personnel in this area is the primary concern, but room dimensions and floor loading constraints are also considered. Material selection and thickness determinations are made using One-Dimensional Shielding Analysis Sequence (SAS1) calculations.

2.2.5 Dose and Thermal Loading Calculations ³

The objective of this phase is to verify the effectiveness of the overall design. MCNP4a is used to determine the dose to personnel at various points in the facility. Additionally, the thermal loads throughout the system are calculated to insure that the shielding materials do not degrade as a result of thermal stresses imposed on the system.

3.0 SHIELDING MATERIALS

The selection of the proper shielding material is crucial to the effectiveness of each component of the system. The particle flux in the beam guide facility is large and the mass and volume of shielding material is limited. Therefore, materials used in the design must be selected to provide adequate shielding while minimizing the size and weight of the shield.

3.1 Necessary Properties

3.1.1 Reduction of Radiation Levels ¹⁶

The most important property for a shielding material is its ability to significantly reduce neutron and gamma radiation. Both the neutron and the photon fluxes are significant in the HFIR HB-2 assembly, requiring materials that attenuate both forms of radiation. Traditionally, heavier elements such as lead are used to reduce photon radiation while lighter elements like hydrogen are used to reduce neutron radiation.

Lightweight materials are desirable for this design so that they can be removed easily to perform maintenance tasks. Fast neutrons are a primary concern, so materials with high hydrogen contents and high neutron absorption cross sections are favored. The materials considered in this design are borated polyethylene, borated GESC NS-4-FR (NAC), lithiated paraffin, Krafon XP3, Krafon X4, Krafon N3, and Krafon NXP. Secondary photons produced by the neutron

interactions are also a major concern. Concrete, lead, carbon steel, and stainless steel 304 are materials that are considered to shield against these photons.

3.1.2 Maintenance and Removal Considerations ¹⁶

In order to facilitate maintenance of the HB-2 beam guide facility. All shielding components must have activation levels which allow handling with minimal protection after a seven day period. The material must allow construction of a shield which can be disassembled to allow maintenance of the facility and to allow equipment to be moved through the beam room.

3.1.3 Other Constraints

Material selection must also consider the space and mass restrictions imposed by the existing beam room. The maximum floor loading is limited to 2500 pounds per square foot which limits the use of heavier materials. The materials used in all of the shielding components must be readily available and capable of absorbing the high radiation levels without deterioration.

The materials must prevent thermal gradients in the existing concrete biological shield surrounding the HFIR. A temperature gradient of one degree Celsius per foot in the concrete wall can cause cracking. An active cooling is undesirable, so the materials must be able to transfer the heat deposited by particle interactions away from the system by conduction and natural convection.

3.2 Potential Shielding Materials

3.2.1 Concrete ¹³

Concrete is a composite shielding material, consisting of both heavy and light elements. Therefore it is an adequate shield for both neutrons and gamma particles, although the gamma radiation generally determines its thickness when used as a radiation shield. Concrete is available in many varieties with varying densities and compositions, allowing its shielding properties to be matched to specific applications. A high density concrete with a high hydrogen content such as Barytes or Oak Ridge (OR) concrete provides excellent attenuation of neutrons and gamma particles.

The physical properties of concrete limit its use in certain applications. Concrete is generally coarse, preventing its formation into the very thin wedges located between the beam guides. Concrete is highly susceptible to thermal gradients. A temperature difference of one degree Celsius per foot can induce cracking. Additionally, concrete should not be exposed to radiation levels exceeding a flux of 2×10^{11} MeV/cm²-sec.

3.2.2 Borated Polyethylene ¹³

Polyethylene is an excellent shielding material for neutrons. Polyethylene has a larger hydrogen content per unit volume than water, resulting in a high neutron scattering cross section. An advantage of polyethylene is being easily manufactured and molded into nearly any shape or form. Therefore, polyethylene is an ideal choice for the shielding wedges in this system. Polyethylene provides some photon shielding, but to reduce large gamma particle radiation levels it must be combined with other materials. Additionally, polyethylene has a melting point of 105 degrees Celsius, which is sufficient for this system.

3.2.3 Krafon Products (XP3, X4, N3, NXP) ⁹

Sanoya is a Japanese company which develops shielding materials. The Krafon series of materials have exceptional neutron and photon attenuation abilities. The melting point of Krafon products is significantly higher than that of polyethylene. Many of the Krafon products provide comparable neutron shielding to borated polyethylene while providing better photon attenuation.

Krafon N3 contains the most hydrogen of the four Krafon products, making it the most effective neutron shield. Krafon NXP contains the most lead, but depends on carbon to moderate neutrons without the help of boron poisoning. All of the Krafon products attenuate gamma particles more effectively than the same thickness of lead alone, but none attenuate gamma particles than the same thickness of concrete. Figure 3.1 shows the relative thicknesses of several

materials required to attenuate gamma radiation, and Figure 3.2 shows the relative thicknesses

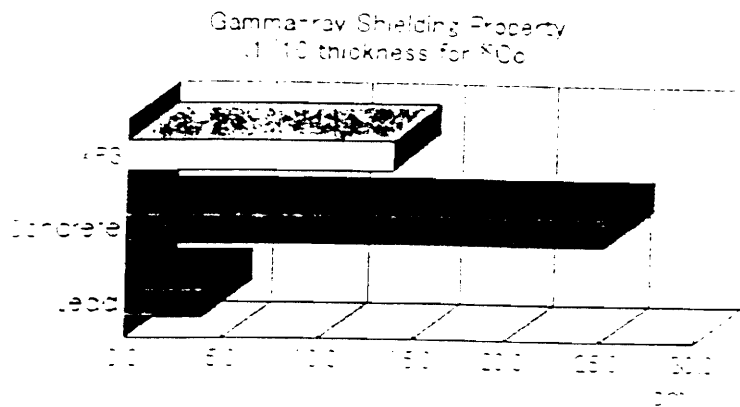


Figure 3.1 Gamma Shield Thickness

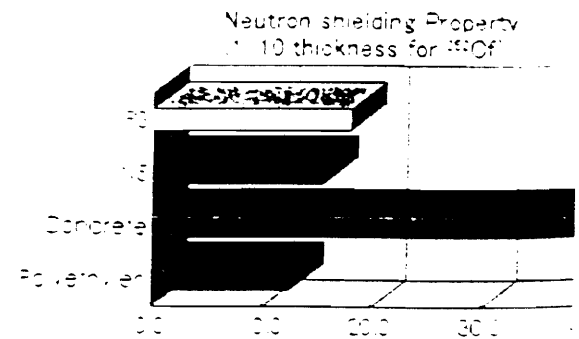


Figure 3.2 Neutron Shield Thickness

3.2.4 Lead¹³

Lead is an excellent shield against gamma radiation. It has a high density, is easily formed into any shape, and largely resists radiation damage. However, lead is a poor neutron shield and can not be used in large quantities due to floor loading constraints.

3.2.5 Steel¹³

Steel has many beneficial material characteristics such as a high melting point and high structural strength. Steel is very effective at reducing photon flux levels and is more effective at

moderating neutrons than lead. Steel is currently being used in the shield plug surrounding the current beam tube facility. Generally steel is more expensive than other shielding materials and is easily activated in neutron fields. Additionally, neutrons with certain energies remain unaffected by steel shielding due to low neutron capture cross section levels at that energy.

3.2.6 Water

Water is the most commonly used neutron moderator because of its high hydrogen content. Additionally, water has the advantage of being inexpensive and readily available. The primary problem associated with using water as shielding is that it must be held in a container. The danger associated with a container rupture and water leakage rendering the shield immediately ineffective requires active safety systems and complex operating procedures which limits its usefulness in simple shielding applications.

3.3 Conclusions

Table 3.1 summarizes the advantages and disadvantages of the shielding materials considered for the upgrade of the HFIR HB-2 beam tube. The materials are further evaluated using DORT, MCNP4a, and SAS1.

Table 3.1 Summary of Material Properties

Material	Advantages	Disadvantages
Concrete	Adequate gamma and neutron shield Widely used for similar purposes Relatively inexpensive	Probably not suitable for shaping into thin wedges for interior shield
Krafton Products	Excellent neutron and gamma shield Excellent material properties	More expensive than polyethylene and concrete
Lead	Excellent gamma shield May be needed in beam room	Poor neutron shield Relatively expensive
Polyethylene	Very good neutron shield Moldable enough for wedge fabrication Widely used and inexpensive	Not good gamma shield Boration may be necessary to reduce secondary radiation
Steel	Good gamma shield Better neutron shield than lead Good structural support	Activation problems Relatively expensive as shield material
Water	Widely used & effective neutron shield Cheap and readily available	Leakage & corrosion concerns Complex shield design

3.4 Number Densities

Number densities for the concrete are taken from the Fusion Reactor Shielding sample problem found in the MCNP4a manual. The concrete is the OR concrete (fusion reactor shield mix). The borated polyethylene, lithiated paraffin, and Stainless Steel 304 number densities are also taken from the MCNP4a Formula list in the MCNP4a manual.

The materials that make up the balance of the geometry configuration include beryllium, Stainless Steel 304, water, and air. The MCNP4a formula was used for Pyrex borated glass with

a density of 2.23 g/cm³. The nickel lining of the glass beam guides is neglected in the computer calculations since its attenuation effectiveness is negligible.

The number densities are calculated for Beryllium, Nac, Krafon X4, and Krafon N3. The calculations for Nac, Krafon X4, and Krafon N3 are based on information provided by the manufacturer using:

$$N_i = \frac{\rho \cdot wf_i \cdot N_A}{m_i}$$

where N_i = number density of the atom or molecule
 ρ = material density, g/cm³
 wf_i = weight fraction of compound molecule, unit less
 N_A = Avogadro's Number = 0.60223 atoms * cm³ / moles * barns
 m_i = atomic mass of atom, g/mol.

Material densities, weight fractions and number densities are presented in Tables 3.2 through Table 3.4.

Table 3.2 Atom Density for Beryllium

Element	Atomic Mass (g/mol)	Atom Density (a/b*cm)
Beryllium	9.0122	1.2362E-1

The material density for Beryllium used is 1.85 g/cm³.

Table 3.3 Atom Densities for Krafton-X4--Gamma Shielding Material

Element	Atomic Mass (g/mol)	Weight Fraction	Atom Density(a/b*cm)
Hydrogen	1.0079	0.0014	3.5132E-3
Aluminum	26.989154	0.0167	1.56547E-3
Oxygen	15.9994	0.1068	1.68835E-2
Iron	55.847	0.0510	2.30976E-3
Titanium	47.90	0.0028	1.47849E-4
Lead	207.2	0.5568	6.79683E-3
Sodium	22.98977	0.0128	1.40822E-3
Tungsten	183.85	0.0317	4.36106E-4
Sulfur	32.06	0.2200	1.73562E-2
Total			4.690401E-2

The material density for Krafton-X4 is 4.20 g/cm³ per the manufacturer.

Table 3.4 Atom Densities for Krafton-N3--Neutron Shielding Material

Element	Atomic Mass (g/mol)	Weight Fraction	Atom Density (a/b*cm)
Hydrogen	1.0079	0.0544	5.5255E-2
Magnesium	24.305	0.0216	9.0981E-4
Nickel	58.70	0.0261	4.5519E-4
Boron	10.81	0.0388	3.6745E-3
Carbon	12.011	0.4633	3.9489E-2
Aluminum	26.98154	0.0378	1.4342E-3
Oxygen	15.9994	0.1954	1.2503E-2
Iron	55.847	0.0108	1.9797E-4
Gadolinium	157.25	0.0344	2.2395E-4
Titanium	47.90	0.0065	1.3892E-4
Nitrogen	14.0067	0.0151	1.1036E-3
Sodium	22.98977	0.0027	1.2023E-4
Tungsten	183.85	0.0785	4.3712E-4
Total			1.1594E-1

The material density for Krafton-N3 is 1.70 g/cm³ per the manufacturer.

4.0 DESIGN METHODOLOGY

4.1 Thermal Guide Shielding Wedges

The thermal guides divide the neutron flux in the HB-2 beam tube into five separate thermal neutron beams. They extend outward from the core into the new beam room. There is an angle of 1.5 degrees between each of the thermal guides. At about 5 meters from the center of the core, the guide assembly begins to separate into individual guides as shown in Figure 4.1. At this separation, the gaps between the guides form wedges, which can be filled with shielding material.

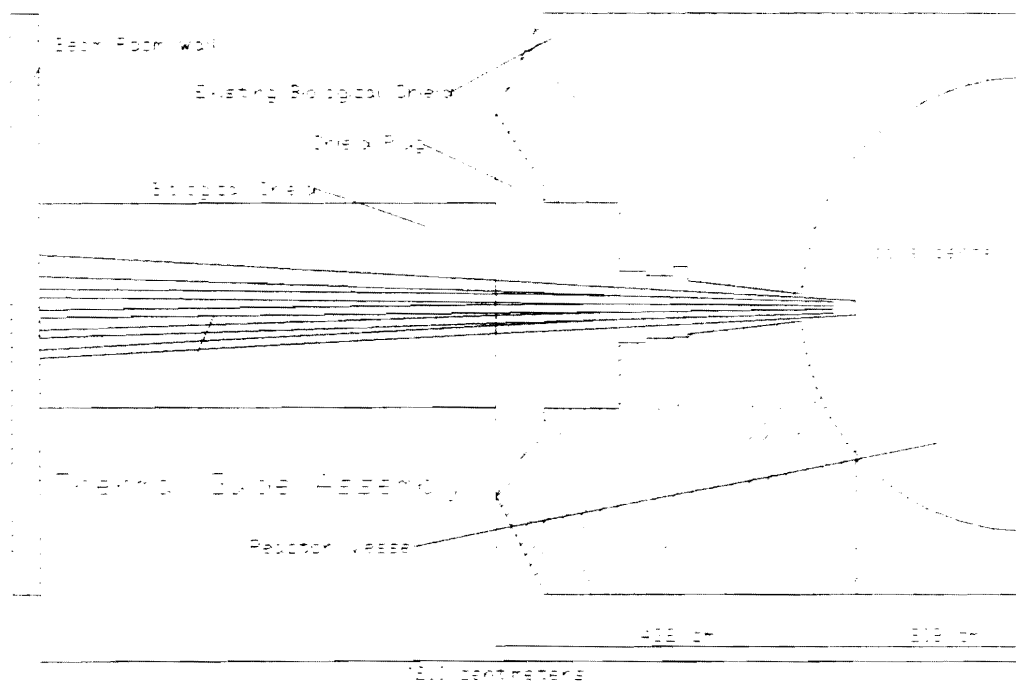


Figure 4.1. Thermal Guides Assembly

4.1.1 Shielding Wedges Function

The function of the shielding wedges between the thermal neutron guides is to reduce the flux that reaches the beam room. As outlined in the project description (Appendix A), the materials used in this area must be reasonably available, acceptable for reactor use, and cost effective.

4.1.2 Materials Considered

Several materials are considered for this part of the design, including carbon steel, Stainless Steel 304, borated polyethylene, and two variations of Krafton. Borated polyethylene is the most suitable for use in the shielding wedges, based on several modes of comparison.

First, the materials are compared based on their activation when exposed to radiation. Due to long-lived activation products, carbon steel and Stainless Steel 304 are undesirable for use in the shielding wedges. The other three materials have an acceptable activation level.

Next, the materials are evaluated on how well they attenuate radiation using a two dimensional model in DORT.

Finally, the materials are evaluated on their ease of use and availability. The two Krafton materials are specialized for either gamma particle shielding or neutron shielding. Therefore, a layering of the two materials is required to reduce the particle flux from the reactor. The borated polyethylene is not as effective at reducing the flux as the specialized Kraftons are, but it is better than the gamma Krafton at reducing neutrons and the neutron Krafton at reducing gamma

particles. For simplicity, the wedges are constructed out of one material instead of layering several together, reducing the cost of fabrication by allowing for a single material to be molded instead of several materials being blended or layered. Additionally, borated polyethylene is more widely available than the Krafon products.

4.1.3 Design Geometry

The thermal guides begin 280 cm from the center of the reactor core and continue into the new beam room. The wedges begin at the point where the guides divide and continues to the edge of the biological shield (see Figure 4.2).

The edge of the biological shield is 640 cm from the center of the reactor, and the thermal guides begin to divide 500 cm from the center of the reactor core, allowing for 1.4 meters of wedge material to be placed between each of the five thermal guides.

4.1.4 Discrete Ordinates Radiation Transport Model

The DORT model fills the beam tube with shielding material from the point where the wedges begin to the edge of the existing biological shield. Therefore, all of the shielding materials are evaluated equally. A typical input file is shown in Appendix C.

A boundary source generated from 30 cm to 640 cm from the center of the reactor core to 30 cm outside of the existing beam tube was provided by ORNL. The source was

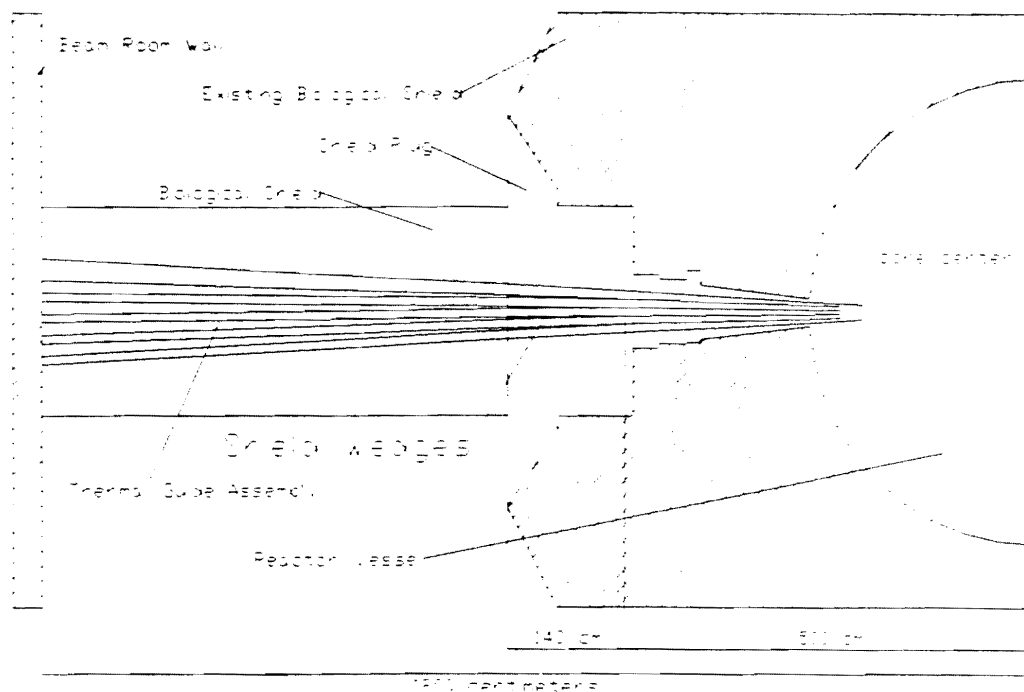


Figure 4.2 Wedge Shielding Starting Point

generated using a geometry scheme that consisted of 35 rectangles overlaid to construct 15 regions of interest. Materials are assigned to each of these regions to represent the HFIR. Figure 4.3 shows the geometry of the DORT model provided by ORNL. Figure 4.4 shows the modified geometry with the shielding materials in place.

Several assumptions are made in the design of the shielding wedges. First, the geometry in the source files supplied by ORNL is assumed to be correct. Additionally, the wedges are assumed to be placed at the point that the guide tube assembly begins to split into separate guides.

Figure 4.3

Figure 4.4

4.2 Cavity Plug Design

The opening in the biological shield surrounding the thermal neutron guides provides a radiation streaming path which must be considered. This cavity must be filled with shielding material to reduce radiation streaming and the subsequent increased dose rate to personnel in the current beam room. The cavity plug shielding must attenuate both neutrons and photons, and be able to accommodate any heat generation as a result of particle interactions.

4.2.1 Materials Considered

In order to reduce the photon flux, the material used in the cavity plug shield must have a high photon attenuation coefficient, which is generally associated with a high material density. However, neutrons are best attenuated by materials with a higher hydrogen content. Therefore, the composition of the cavity plug shielding must consist of layers of several materials or a composite material.

The materials considered for use in the cavity plug must also be able to accommodate heat generated as a result of particle interactions. Therefore, materials which are highly susceptible to thermal stresses, experience large changes in volume with thermal energy deposition, or that have low melting points will not be considered.

Several materials and combinations of materials meet the design requirements. Concrete (Oak Ridge Mix) provides excellent gamma particle shielding when used in significant thickness, and also provides good neutron shielding as a result of the high hydrogen content. A second

material choice incorporates Borated polyethylene as a neutron shield, a heavy metal such as lead or steel for gamma ray attenuation, and concrete for biological isolation. However, due to floor loading considerations, the use of heavy metals will be avoided.

4.2.2 Design Geometry

The cavity plug must completely fill the opening in the biological shield from the steel bellows assembly, which separates the reactor pool water from the cavity plug, to the outer wall of the current biological shield. The cavity has an overall length of the 2.5 meters with a maximum width of 2.8 m and a minimum width of 1 m. A tolerance of 0.35 cm must be allowed between the cavity plug and the biological shield wall to allow for removal of the cavity plug for maintenance of the facility. Figure 4.5 shows the location of the cavity plug.

4.2.4 Description of the model used

The cavity plug model is an MCNP4a model of the HB-2 beam tube. MCNP4a is a neutron and photon transport code which uses stochastic Monte Carlo modeling to simulate the system. The advantages of MCNP4a over discrete ordinates codes are that three dimensional effects are considered, particle streaming can be accounted for, and secondary photons are accurately tracked. The model includes the neutron guide facility from the center of the core to the outside wall of the beam room, approximately 12 meters from the center of the core. The cylindrical steel beam tube is modeled as a series of step increments ranging from 9.5 cm i.d. to

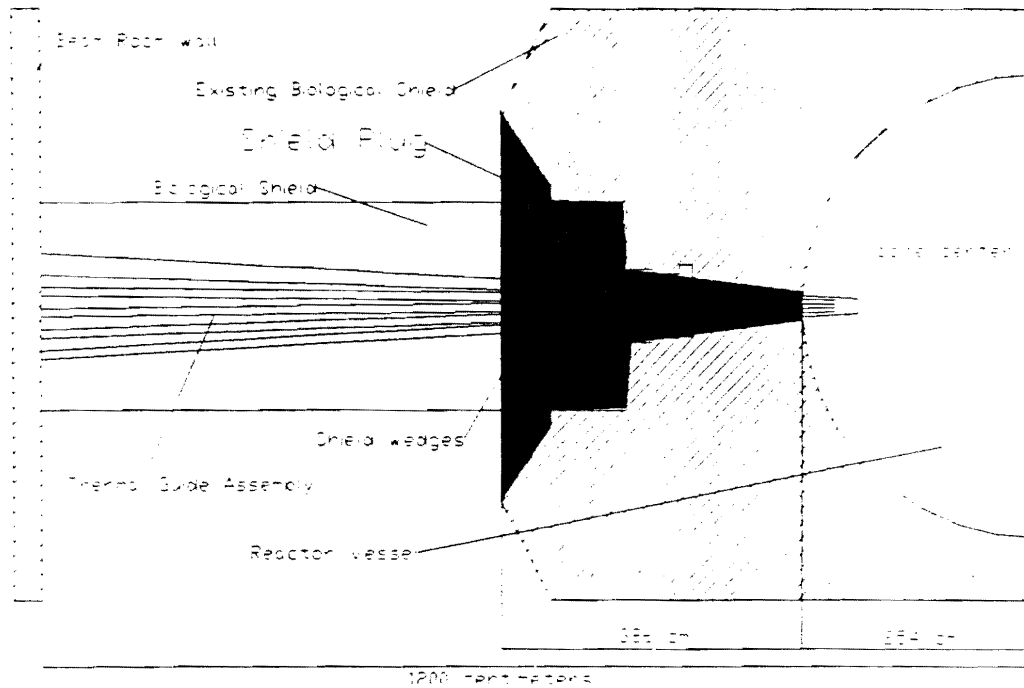


Figure 4.5 Cavity Plug Location

19.6 cm i.d. At 2.9 meters, from the core centerline the five borosilicate glass thermal neutron beam guides begin. The guides have inner dimensions of 17.9 cm tall by 5.08 cm wide, with a wall thickness of 2.54 cm. The super mirror coatings lining the thermal guides are neglected because its thickness renders it useless as a shield. Additionally, the biological shield surrounding the reactor is modeled to a distance of 2.5 m from center of the beam tube allowing for a more complete modeling of the cavity shielding.

This model assumes that the reactor vessel does not have significant effect on the shielding design and it is not modeled. The vessel should attenuate both neutrons and gamma rays, to some extent. However, in comparison to the large volumes of concrete and water, the attenuation due to the vessel is insignificant.

The cavity plug is modeled by completely filling the cavity in the biological shield with the material to be considered. Therefore, streaming through the clearance surrounding the cavity plug is not accounted for. However, streaming is negligible since the 0.3 cm clearance between the cavity plug and the biological shield changes direction in five locations. The streaming through the clearance between the shield plug and the thermal guides is attenuated by the beam room shielding.

Two point sources, derived from the boundary source supplied by ORNL using a nonlinear regression, is used by the MCNP4a model. The neutron source has a strength of 6×10^{18} particles per second and is located 24 cm behind the supplied boundary source. Similarly, the primary photon source has a strength of 8×10^{18} particles per second and is located 24 cm behind the boundary source. The point sources maintain energy group structure and angular characteristics of the original source. The source is further discussed in Appendix E.

4.3 Biological Shield Design

The one dimensional SAS1 sequence of SCALE 4.2 is used to model the movement of particles through the biological shield. The point kernel method is employed, using an isotropic point source calculated in MCNP4a, even though the true source is actually multi-directional. The model uses 27 neutron and 18 gamma particle coupled energy group cross sections. Only the last neutron energy group is included in these calculations because 97 percent of the neutrons reaching the beam room are thermal. By taking the photon energy of the majority of the flux traveling down the tube, the energy groups 1.0 MeV to 1.5 MeV are included in these

calculations. The source term for SAS1 is based upon the MCNP4a calculations in the center beam guide. This value is multiplied by 5 to form a conservative estimate of the source from the thermal guide assembly. As a result, this isotropic source is larger than the actual source term.

The materials used in the SAS1 models include various thickness of OR concrete, Borated polyethylene, lead, and carbon steel. For each material cited in the input file (shown in Appendix F) SAS1 uses the Material Information Processor (MIP) to prepare the cross sections and the mixing table for the shielding calculations. MIP takes the input file and redirects it to the BONAMI-S, NITAWL-S, and XSDRNPM-S Scale modules for further consideration. BONAMI-S is used to calculate resonance self-shielding for certain materials, while NITAWL-S calculates the resonance self-shielding corrections for the other materials. The XSDRNPM-S module is included to analyze the radiation transport data by performing the fixed source calculation to determine the neutron and photon fluxes. These fluxes are converted to dose rates to determine the relative effectiveness of the materials. The location of the Biological Shield is shown in figure 4.6.

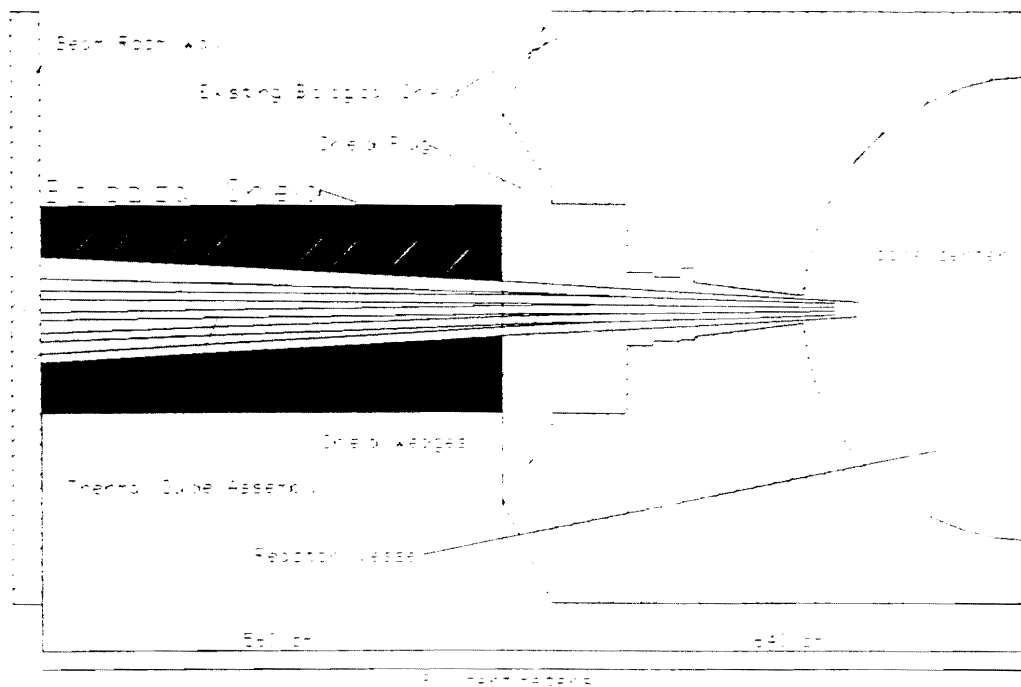


Figure 4.6 Biological Shield Location

4.4 Dose and Thermal Loading Calculations

In order to verify the effectiveness of the overall design, dose rates in the beam room are calculated with all of the shielding components in place. In order to effectively model the system, it is treated three dimensionally. Additionally, the energy deposition in the shielding components is determined in order to insure material stability.

An MCNP4a model of the HB-2 thermal neutron guide facility is used to evaluate the total dose rate in the beam room surrounding the thermal neutron guide facility. The facility is

modeled from the reactor core to the inner wall of the current beam room, approximately 12 m. All shielding is in place, and neutrons, primary photons, and secondary photons are considered.

The MCNP4a simulation models the energy deposition as a result of particle interactions on materials accurately. The energy deposition, determined for all the shielding materials used, is used to calculate the temperature gradients in the materials of interest, to insure that material melting and degradation are not a concern.

The shielding wedges are modeled by completely filling the space between the thermal guides with borated polyethylene with an over all length of 1.4 m. The clearance between the borosilicate glass of the guides and the borated polyethylene is not considered. While this clearance has a significant effect on the shielding at the end of the thermal guides in the new guide hall since it provides a significant shielding path, the clearance will affect radial attenuation very little.

The biological shield in the beam room is modeled as a constant thickness of borated polyethylene and OR concrete. The maximum thickness is required near the cavity plug, and less shielding is required as the guides cross the beam room. However, to simplify the problem and the construction of the final design, a uniform thickness is modeled.

5.0 SUPPORTING CALCULATIONS

5.1 DORT Calculations Supporting Wedge Shield Design

The DORT model is used to determine the optimal thermal guide wedge shielding material by comparing the calculated dose rates with each material in place. The input file is shown in Appendix C.

Several calculations are performed for each of the different materials and the dose rates are compared to determine the optimal shield material and thickness. Flux determination is taken at 640 cm from the center of the reactor core. The doses are averaged for the beam tube width.

The first scenario consisted of air as the shielding material. The neutron dose rate with air is $5.16\text{E}9$ mRem/hr. The photon dose rate is $9.07\text{E}8$ mRem/hr. The other materials are compared against the results for air to determine the most effective shield.

The second scenario uses borated polyethylene as the shielding material and is placed 500 cm from the center of the reactor core and continued to the biological shield edge at 640.08 cm. The neutron flux is $2.35\text{E}2$ mRem/hr. The photon flux is $7.65\text{E}5$ mRem/hr.

The third and fourth scenarios use Krafton- X_4 and Krafton- N_3 as the shielding material respectively. The material is placed at the same location as the borated polyethylene. For Krafton- X_4 , the neutron flux is $4.63\text{E}3$ mRem/hr and the photon flux is $1.96\text{E}2$ mRem/hr. For Krafton- N_3 , the neutron flux is $1.99\text{E}1$ mRem/hr and the photon flux is $9.95\text{E}6$ mRem/hr.

A comparison of the dose rates are shown in Table 5.1. From this table it is evident that the single best material for reducing both neutrons and gamma particles is borated polyethylene even though Krafon-N₃ is the best Neutron shield, and Krafon-X₄ is the best photon shield.

Table 5.1 Dose rates calculated from DORT and their rankings.

Material	Neutron Dose Rate	Gamma Dose Rate	Neutron Ranking	Gamma Ranking	Overall Ranking
Air	5.16E9	9.07E8	4	4	4
Polyethylene	2.35E2	7.65E5	2	2	1
K-X ₄	4.62E3	1.96E2	3	1	3
K-N ₃	1.99E1	9.95E6	1	3	2

5.2 MCNP4a Calculations Supporting Shield Plug Design

The MCNP4a model provides flux estimates in the cavity plug. These fluxes are used to determine the effectiveness of the particular cavity plug design. In order to simplify the source used in the model, primary photons and neutrons are treated as separate cases. A sample input file is found in Appendix D, and a more detailed discussion of the source can be found in Appendix E.

An advantage of MCNP over discrete ordinates methods is that it provides an estimate of the quality of the calculation. Along with each flux or dose rate, a percentage error is calculated. In general, an error of less than 10 percent is desired, and less than 5 percent is preferred. Results with associated errors greater than 10 percent but less than 35 percent can only be used to

establish an order of magnitude. Results are considered meaningless if they have an associated error of 35 percent or more.

Primary flux calculations evaluate the particle flux at four positions in the cavity plug itself. Additional secondary flux calculations evaluate the particle flux at the same four positions in the concrete biological shield surrounding the cavity plug. The primary calculations have an associated error of less than ten percent, and secondary calculations have an associated error of less than 25 percent. In order to achieve this level of accuracy, 2 million particle histories were tracked for each case, and mean free path based importance biasing was utilized.

The primary objective of this portion of the design project is to determine the composition of the design plug. Initially, borated polyethylene, OR concrete, carbon steel and air were modeled in the cavity plug separately. The neutron flux in the cavity plug itself as a function of distance from the core centerline is shown in figure 5.1.

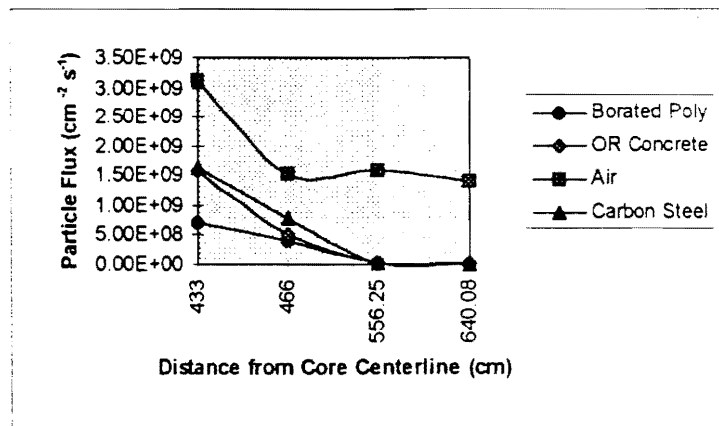


Figure 5.1 Neutron flux in the cavity plug for various compositions

The corresponding photon flux (both primary and secondary) is shown as a function of distance from the core centerline in Figure 5.2. The flux pattern in the concrete biological shield is similar but much lower in magnitude. The borated polyethylene plug provides the best neutron flux reduction, and steel provides the best photon shielding. However, steel is undesirable as a shielding material because it has an activation time greater than 7 days. In order to adequately shield both neutrons and photons, a combination of materials must be used.

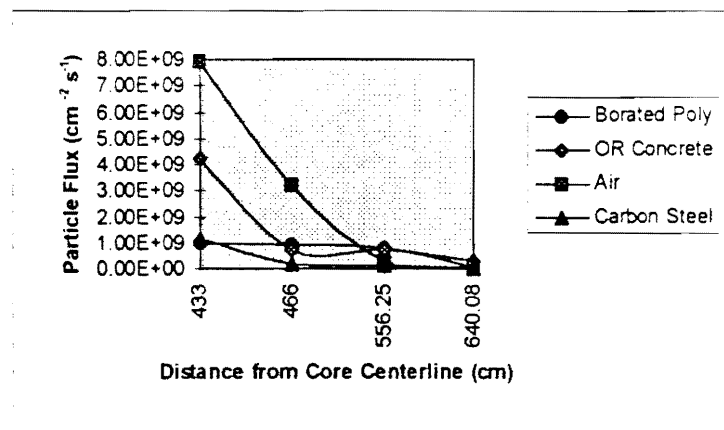


Figure 5.2 Photon flux in the cavity plug for various compositions

The final cavity plug design utilizes 41.275 cm of borated polyethylene in the first two sections of the cavity plug to reduce the neutron flux followed by 207 cm of OR concrete to attenuate the photon flux. The dose rate in the cavity plug as a function of distance from the centerline of the core is shown in Figure 5.3. The neutron dose rate at the surface of the cavity

plug is 0.8 mRem/hr, and the photon dose rate is 1.2 mRem/hr. Therefore the cavity plug design provides adequate shielding.

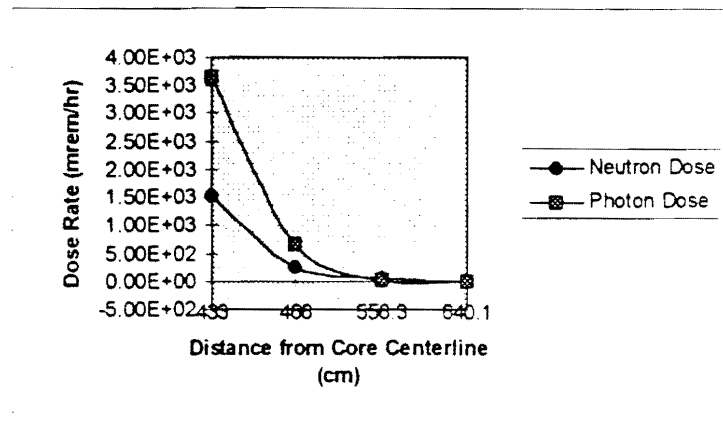


Figure 5.3 Dose rate in the cavity plug

5.3 SAS1 Calculations Supporting Biological Shield Design

A concrete biological shield is first modeled using a 1-d sphere of OR concrete with the source placed in the center of the sphere. The radius of the sphere varies from 50 to 500 cm as shown in Table 5.2 and Figure 5.4. The neutron and photon dose rates decrease with increasing thickness, but neither fall below the MADR of 2.5 mRem/hr.

Secondly, a borated polyethylene shield is modeled using a variable radius sphere from 50 to 500 cm. The neutron dose rate is considerably lower than the gamma as shown in Table 5.3 and Figure 5.5. The neutron dose rate falls below the MADR with only 100 cm of borated polyethylene in place, whereas the gamma dose rate never falls below the MADR.

Carbon steel is modeled next with a varying sphere radius of 50 to 500 cm. Neutron dose rates drop steadily and fall below the MADR using a 400cm shield thickness. However, the gamma dose rate never falls below the MADR. Results are shown in Table 5.4 and Figure 5.6.

After single materials are modeled, 50 cm of borated polyethylene and various thickness of concrete are modeled in an attempt to find the optimum mix of neutron and gamma shields. Again, neutron dose rates fall quickly using 50 cm borated polyethylene and 150 cm OR concrete, but gamma dose rates remain high as shown in Table 5.5 and Figure 5.7. The overall dose rate results are similar even when the borated polyethylene is increased to 100 cm as shown in Table 5.6 and Figure 5.8.

Finally, the 1-d sphere is modeled using 50 cm borated polyethylene with increasing thickness of lead and concrete in an attempt to better attenuate the gamma particles at the outside surface of the biological shield. With the addition of 300 cm of lead and 400 cm of OR concrete to the 50 cm of borated polyethylene, the gamma dose rate falls below the MADR. Table 5.7, Table 5.8, Figure 5.9, and Figure 5.10, shows the results.

The source term for the SAS1 model of particle travel through the biological shield is very conservative. The source used in this model produces much higher dose rates than the actual source since it was assumed to be isotropic. A shield that utilizes 50 cm of borated polyethylene and 100 cm of concrete optimizes the shield effectiveness with a minimal volume of shielding material.

Table 5.2 Dose Rate Reduction as a Function of OR Concrete Thickness

Concrete Thickness	Neutron Dose Rate	Gamma Dose Rate	Total Dose Rate
cm	mRem/hr	mRem/hr	mRem/hr
50	3.28E+03	3.41E+03	6.69E+03
100	8.72E+02	9.01E+02	1.77E+03
150	4.03E+02	4.18E+02	8.22E+02
200	2.33E+02	2.44E+02	4.77E+02
250	1.52E+02	1.60E+02	3.12E+02
300	1.07E+02	1.14E+02	2.20E+02
350	7.87E+01	8.49E+01	1.64E+02
400	6.04E+01	6.58E+01	1.26E+02
450	4.77E+01	5.25E+01	1.00E+02
500	3.84E+01	4.28E+01	8.13E+01

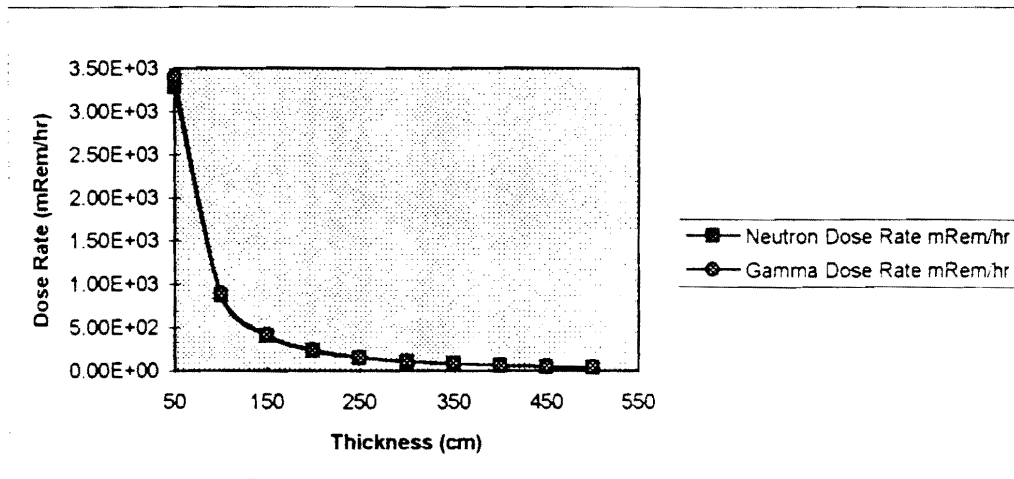


Figure 5.4 Dose Rate Reduction as a Function of OR Concrete Thickness

Table 5.3 Dose Rate Reduction as a Function of Borated Polyethylene Thickness

BPoly Thickness	Neutron Dose Rate	Gamma Dose Rate	Total Dose Rate
cm	mRem/hr	mRem/hr	mRem/hr
50	42.5067	3979.54	4022.0467
100	0.423307	975.412	975.835307
150	0.008499	423.439	423.447499
200	2.29E-04	232.417	232.417229
250	7.27E-06	145.028	145.028007
300	2.60E-07	98.1471	98.1471003
350	1.03E-08	70.2228	70.2228
400	4.54E-10	52.32375	52.32375
450	2.22E-11	40.2146	40.2146
500	1.22E-12	31.6627	31.6627

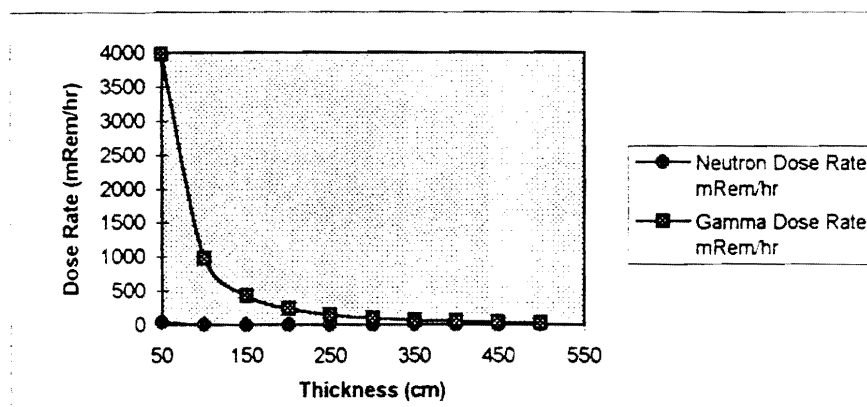


Figure 5.5 Dose Rate Reduction as a Function of Borated Polyethylene Thickness

Table 5.4 Dose Rate Reduction as a Function of Carbon Steel Thickness

Carbon Steel Thickness	Neutron Dose Rate	Gamma Dose Rate	Total Dose Rate
cm	mRem/hr	mRem/hr	mRem/hr
50	2480	5366.74	7846.74
100	469.964	1724.59	2194.554
150	147.317	857.597	1004.914
200	55.7855	496.342	552.1275
150	23.2441	310.141	333.3851
300	10.2791	203.374	213.6531
350	4.73315	138.126	142.85915
400	2.2463	96.6534	98.8997
450	1.0923	60.923	62.0153
500	0.54235	50.2304	50.77275

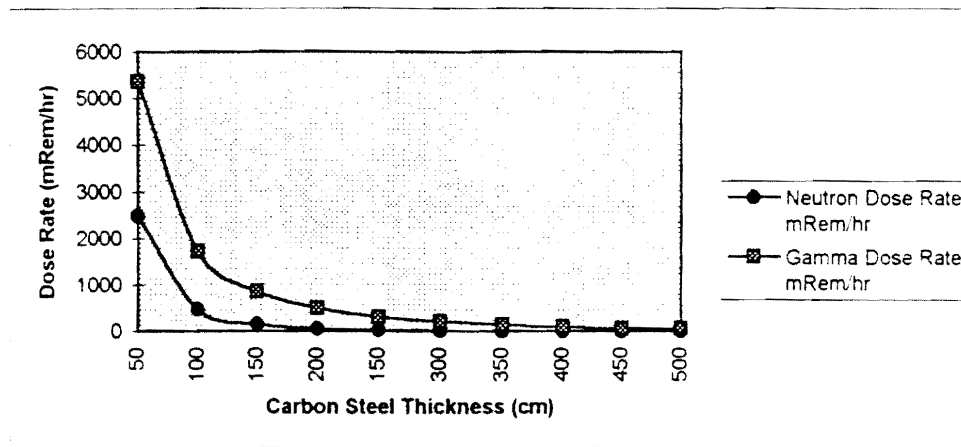


Figure 5.6 Dose Rate Reduction as a Function of Carbon Steel Thickness

Table 5.5 Dose Rate Reduction of as a Function of OR Concrete and 50 cm of Borated Polyethylene

BPoly Thickness	Concrete Thickness	Neutron Dose Rate	Gamma Dose Rate	Total Dose Rate
cm	cm	mRem/hr	mRem/hr	mRem/hr
50	100	3.997	423.66	427.657
50	150	2.30388	232.852	235.15588
50	200	1.5	145	146.5
50	250	1.06462	98.458	99.52262
50	300	0.792667	70.4378	71.230467
50	350	0.61268	52.4549	53.06758
50	400	0.487016	40.2683	40.755316
50	450	0.395624	31.65511	32.050734
50	500	0.326983	25.3618	25.688783

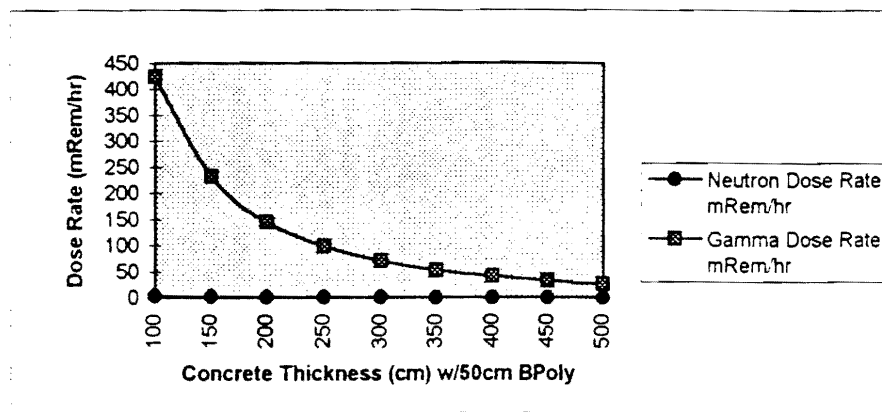


Figure 5.7 Dose Rate Reduction as a Function of OR Concrete Thickness and 50 cm of Borated Polyethylene

Table 5.6 Dose Rate Reduction as a Function of OR Concrete Thickness and 100 cm of Borated Polyethylene

BPoly Thickness	Concrete Thickness	Neutron Dose Rate	Gamma Dose Rate	Total Dose Rate
cm	cm	mRem/hr	mRem/hr	mRem/hr
100	100	0.0867053	232.747	232.8337053
100	150	0.055851	145.33	145.385851
100	200	0.039138	98.3609	98.400038
100	250	0.028991	70.3484	70.377391
100	300	0.0223343	52.3715	52.3938343
100	350	0.0177149	40.1902	40.2079149
100	400	0.0143702	31.5818	31.5961702
100	450	0.0118659	25.2926	25.3044659
100	500	0.00993987	20.5718	20.58173987

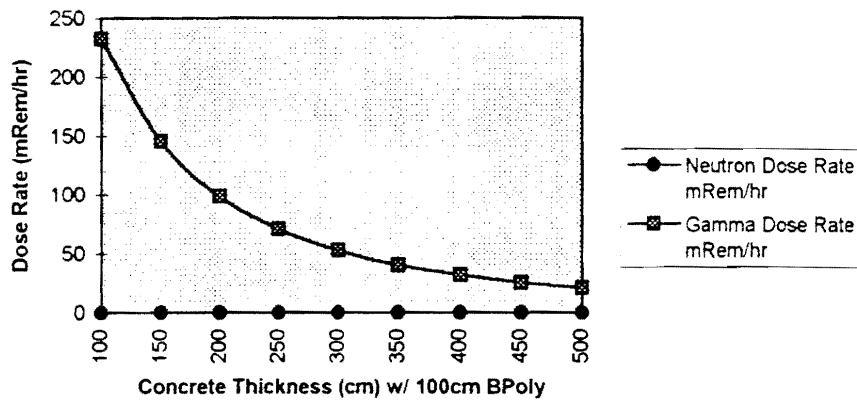


Figure 5.8 Dose Rate Reduction as a Function of OR Concrete Thickness and 100 cm of Borated Polyethylene

**Table 5.7 Dose Rate Reduction as a Function of Lead Thickness and 100 - 400 cm of OR
Concrete and 50 cm of Borated Polyethylene**

BPoly Thickness	Concrete Thickness	Lead Thickness	Neutron Dose Rate	Gamma Dose Rate	Total Dose Rate
cm	cm	cm	mRem/hr	mRem/hr	mRem/hr
50	100	10	3.54326	343.553	347.09626
50	150	20	1.93241	164.298	166.23041
50	200	50	0.07527	68.6088	68.68407
50	300	50	0.619163	36.5699	37.189063
50	400	100	0.332525	12.5175	12.850025
50	400	250	0.2939193	4.26292	4.5568393
50	400	300	0.179368	1.54063	1.719998

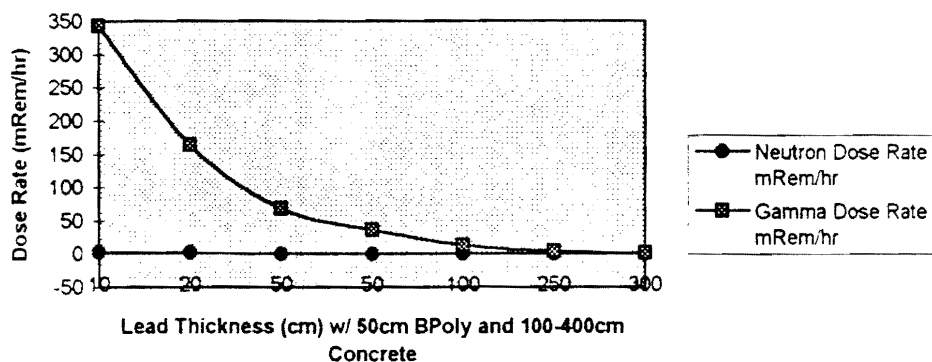


**Figure 5.9 Dose Rate Reduction as a Function of Lead Thickness and 100 - 400 cm of OR
Concrete and 50 cm of Borated Polyethylene**

Table 5.8 Dose Rate Reduction as a Function of OR Concrete Thickness and 10 - 300 cm of Lead and 50 cm of Borated Polyethylene

BPoly Thickness	Lead Thickness	Concrete Thickness	Neutron Dose Rate	Gamma Dose Rate	Total Dose Rate
cm	cm	cm	mRem/hr	mRem/hr	mRem/hr
50	10	100	3.54326	343.553	347.09626
50	20	150	1.93241	164.298	166.23041
50	50	200	0.07527	68.6088	68.68407
50	50	300	0.619163	36.5699	37.189063
50	100	400	0.332525	12.5175	12.850025
50	250	400	0.2939193	4.26292	4.5568393
50	300	400	0.179368	1.54063	1.719998

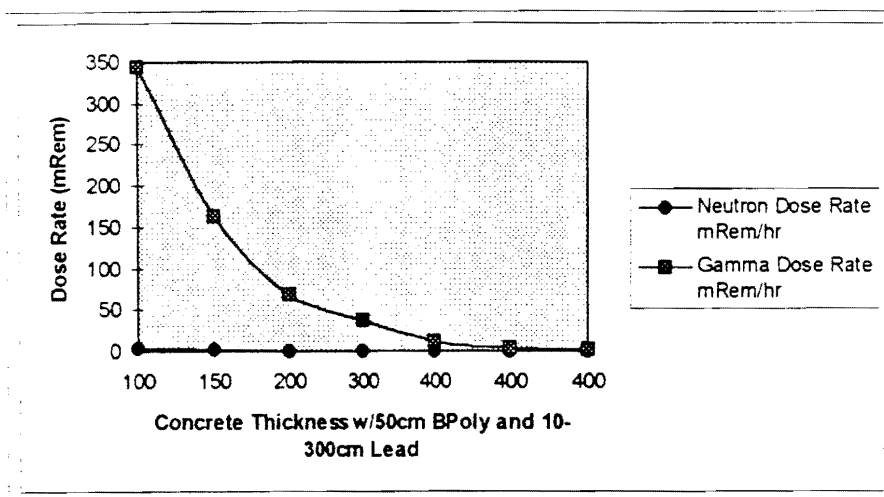


Figure 5.10 Dose Rate Reduction as a Function of OR Concrete Thickness and 10 - 300 cm of Lead and 50 cm of Borated Polyethylene

5.4 MCNP4a Calculations Supporting Overall Design

The MCNP model provides dose rate estimates in the beam room as well as kerma energy deposition estimates in the shielding material. As in the development of the cavity plug, neutrons and primary photons are treated separately. Dose rates are calculated using ANSI/ANS standard 6.1.1 flux to dose conversions.

The dose rate is calculated in the air of the beam room, 30 cm from the surface of the beam room biological shielding. The error associated with the dose rates calculated is less than ten percent, therefore results can be considered to be reasonable. Accuracy is achieved by tracking 2 million particle histories, and mean free path based importance biasing is utilized.

The primary objective of this portion of the design is to verify shielding effectiveness by calculating the dose rate in the beam room. The total dose rate 30 cm from the surface of the biological shielding is 1.2 mRem/hr. Therefore the shielding design is acceptable.

The second objective of this portion of the design is to demonstrate that energy deposition is not large enough to cause material instability. The kerma detectors calculated the largest volumetric energy deposition in the wedge shielding. The average energy input for all shielding materials is less than 10 Watts per m^3 . A volumetric heat source of less than 100 Watts per m^3 is not large enough to cause significant temperature gradients in any portion of the shielding design. The associated error of the kerma response calculations is less than 20 percent. Therefore they only serve to establish the order of magnitude of the heat source.

6.0 FINAL DESIGN SUMMARY

The final design consists of three components: the shielding wedges, the cavity plug shielding, and the beam room biological shielding. Each component serves to reduce the particle flux reaching the beam room air and, thus, to reduce the dose rate in the beam room. The design of each component is validated using discrete ordinates and Monte Carlo models.

The shielding wedges are fabricated of borated polyethylene. Each wedge is 140 cm long by 17.9 cm tall and extends from the existing biological shield wall at 640.08 cm from the core centerline toward the center of the reactor, allowing for a 0.3 cm clearance between the borosilicate glass and the wedge material.

The cavity plug contains two components. The first component is fabricated of borated polyethylene block, and it is 41.275 cm thick and completely fills the first two sections of the cavity with 0.3 cm of clearance between the cavity plug and the existing biological shield. The second component is fabricated of OR concrete block and is 207 cm thick and fills the remainder of the cavity, with the same clearances allowed.

The beam room biological shielding is also a two component system. Approximately 45 cm of borated polyethylene followed by 70 cm of OR concrete completely surrounds the beam guides. A portion of the borated polyethylene is inside the vacuum casing surrounding the guides and begins at the point where the shielding wedges begin (~500 cm from the center of the reactor) and ends at the wall of the beam room. For more detailed dimensions of the beam room shield, see Figure 6.1. The biological shield design causes an average floor loading of 1126 lb/ft², which is far below the design constraint of 2500 lb/ft².

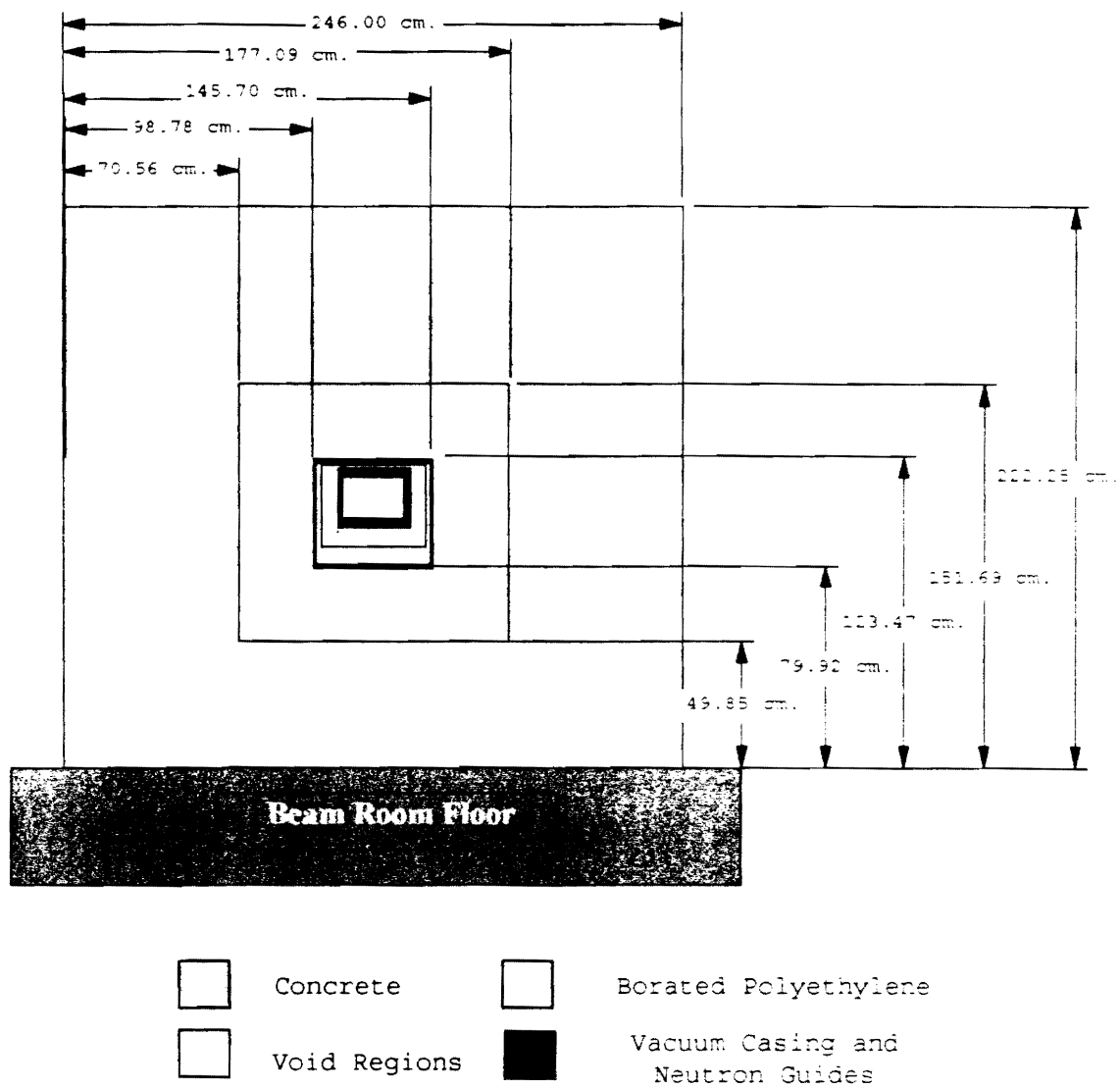


Figure 6.1 Cross-sectional View of Beam Room Shielding

7.0 CONCLUSIONS

The two-dimensional DORT model calculates dose rates at the outside wall of the existing biological shield, 640.08 cm from the core centerline. With the shielding wedges in place, the dose rate is reduced from $5.1\text{E}9$ mRem/hr to $2.35\text{E}2$ mRem/hr. The MCNP4a model calculates dose rates at the outside wall of the existing biological shield. With the cavity plug and wedges in place the dose rate is reduced from $2.35\text{E}2$ mRem/hr to 2.1 mRem/hr. Finally, with all shielding in place, the dose rate at 30 cm from the surface of the biological shield is 1.1 mRem/hr, which satisfies the design constraints. Additionally, the average floor loading is 1126 pounds per square foot which meets the limitation of 2500 pounds per square foot. An average energy deposition of 35 W/m^3 indicates that thermal gradients in the concrete are insignificant.

8.0 HUMAN RESOURCE ALLOCATION

Due to the complexity of this project, all group members could not be heavily involved in each phase of the design process. Each group member was assigned tasks, based on their experience or interests. All major tasks involved at least two group members. Figure 8.1 charts the progress toward completion of major design elements until completion of the design. Table 2.1 shows which group members were the most involved in the various tasks.

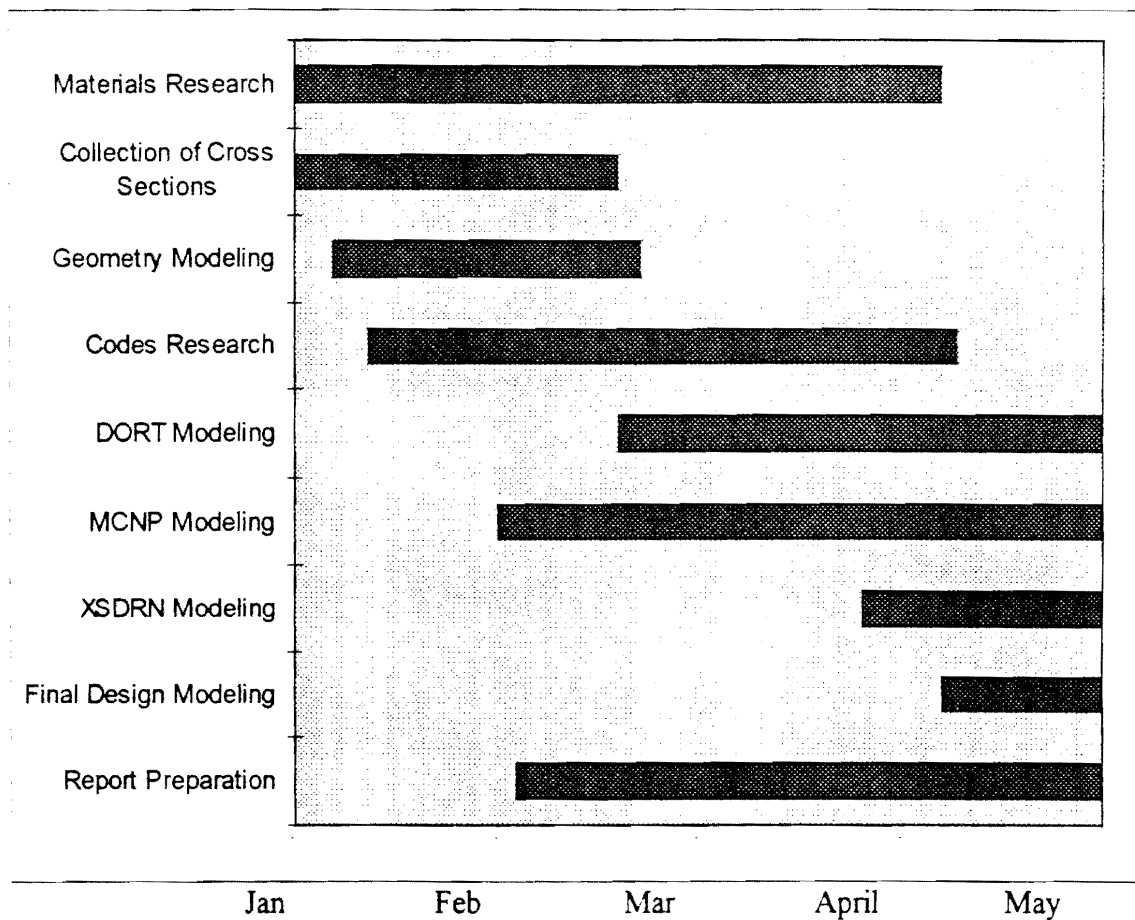


Figure 8.1 Milestone Chart

Table 8.1 Work Breakdown Structure

Task	Dan Evans	Dax Jolly	Robert Page	David Pointer	Susan Turner
Materials Research	xx		xx		xx
Cross Section Collection					xx
Codes Research	xx	xx	xx	xx	xx
Geometry Modeling				xx	
DORT Modeling (Wedges)	xx	xx			
MCNP4a Modeling				xx	xx
SCALE 4.2 Modeling (Bio Shield)			xx		xx
Final Design Modeling	xx	xx	xx	xx	xx
Floor Loading Calculation			xx	xx	
MCNP Final Dose Estimate				xx	xx
Report Preparation	xx	xx	xx	xx	xx

9.0 FUTURE WORK

Additional calculations should be done to verify the accuracy of the source used in MCNP4a calculations. The source should be modeled in DORT to demonstrate that the source accurately reproduces the boundary source supplied by ORNL.

A complete cost analysis should be performed to evaluate the cost effectiveness of the design. Material costs as well as construction costs should be considered.

MCNP4a calculations should be repeated, tracking more particle histories and using better source biasing techniques such as forced collision cards and improved weighting functions to reduce the error associated with each result. An error of less than ten percent was obtained in primary calculations. While an error of ten percent is acceptable in shielding calculations, an error less than five percent is preferred.

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APPENDIX A

PROBLEM DEFINITION

1997 UTK STUDENT DESIGN PROJECT DEFINITION

by

D. L. Selby

Oak Ridge National Laboratory

January 2, 1997

Develop a nuclear shielding concept for the High Flux Isotope Reactor (HFIR) HB-2 beam tube upgrade. The shielding concept as presently perceived should address three items

- 1) Shielding wedges between thermal guides within the biological shield
Objective - Propose wedge shield composition and physics design to maximize the reduction in source term at the outside surface of the biological shield. Constraints - Materials must be reasonably available, acceptable for reactor environment use, and cost effective. Heat loads in shield should be low enough to allow conduction cooling. Shield activation levels should be relatively low two days after shutdown to simplify maintenance and removal.
- 2) Cavity Plug Assembly shielding
Objective - Propose shield composition and step-out locations to minimize streaming around the cavity plug assembly and to minimize neutron and gamma sources impinging upon the concrete.
Constraints - Materials must be reasonably available, acceptable for reactor environment use, and cost effective. A 1/8-inch gap between shield and concrete must be maintained for clearance. Heat loads in concrete must be low enough to prevent thermal gradients that will cause cracking of the concrete.
- 3) Beam Room shielding
Objective - Propose shield composition and physics design that reduces the source term to 2.5 mRem/hr at 30 cm from the external shield surface. Constraints - Materials must be reasonably available, acceptable for reactor environment use, and cost effective. Shield must fit within available volume. Floor loading cannot exceed maximum loading of 2500 lb./sq. ft. Shield activation must be low enough to handle with minimum protection 7 days after reactor shutdown. Shield must account for streaming under the shield via scattering within the floor. Shield must not require an active cooling system.

The preparation, review, and documentation of calculations shall be performed in accordance with Research Reactors Division procedure RTP-1 (Preparation, Review, and Handling of Calculations) and RRAP-3.9.1 (Performance of Software Quality Assurance). This ensures that the information generated in this design project can be used later by ORNL.

APPENDIX B

EQUIPMENT LIST

SOLARIS 2 Workstations Used For MCNP4a Calculations

boyle.engr.utk.edu
carnot.engr.utk.edu
chadwick.engr.utk.edu
maxwell.engr.utk.edu
nestor.engr.utk.edu
pauli.engr.utk.edu
reynolds.engr.utk.edu
ritz1.engr.utk.edu
ritz2.engr.utk.edu
ritz3.engr.utk.edu
ritz4.engr.utk.edu
stokes.engr.utk.edu
verdi.engr.utk.edu

UTCC UNIX Cluster Machines Used for DORT Calculations

utkux1.utcc.utk.edu
utkux4.utcc.utk.edu
dogwood1.utcc.utk.edu
dogwood2.utcc.utk.edu
dogwood3.utcc.utk.edu
dogwood4.utcc.utk.edu
gis1.utcc.utk.edu
gis2.utcc.utk.edu
gis3.utcc.utk.edu
gis4.utcc.utk.edu

APPENDIX C

DORT INPUT FILE

```

\
\ DORT Input File with Borated Polyethylene as the Wedge Material
\ March 10, 1997
\
\ University of Tennessee, Knoxville
\ Nuclear Engineering 472
\ Spring Semester 1997
\ Senior Design Project
\
\ Written by Dax Jolly and Daniel Evans
\ With Assistance from: David Pointer
\                      Susan Turner
\                      Bobby Page
\
\
\ Set up Environment
\
= '/bin/ksh
set -xv
date
rmdir='pwd'
mytmpdir=/tmp/daxjolly
tempdir=/tmp/$USER.$$
sedatdir=/u03/sedat/design97
mkdir $tempdir
cd $tempdir
ln -s /u03/sedat/codes/dort/dort
\
\ Begin Dort Input
\
cat >dort1 << 'EOF'
    hfir 84 grp - beam tube 2 thru reactor face - dtd srce          9/27/96
\
\ Control Input Arrays
\
61$$ 44 0 8 48 e
61$$ a8 47 all 49 e
62$$ 0 3 15 88 125 84 3 4 97 0 35 0 40 0 224
    1 1 4 4 0 1 -1 0 4 0 2 4z
    4z 0 -1 0 2r1 0 5z e
62$$ a33 1 e
62$$ a62 -1 800 800 10 e
63** 0.0 0.0 1.0-4 1.0-4 1.0-4 e
63** a17 5.0-3 e
    t
    t
\
\ Directional Weighting Array
\
81** 0 2r49504-10 0 2r49504-10 0 2r49504-10 0 2r49504-10 0 2r49504-10 0
    2r17327-9 0 2r17327-9 0 2r17327-9 0 2r17327-9 0 2r17327-9 0 2r43316-9 0
    2r43316-9 0 2r23037-8 0 2r23037-8 0 2r72319-8 0 2r98414-8 0 2r12451-7 0
    2r15035-7 0 2r1762-6 0 2r2018-6 0 2r22715-7 0 2r2525-6 0 2r27735-7 0
    24297-6 2r12844-6 24297-6 0 2685-5 84992-7 2r19096-6 84992-7 2685-5 0
    24296-6 84992-7 27304-6 2r68163-7 27304-6 84992-7 24296-6 0 83635-7
    64835-7 96393-7 34407-7 2r91447-7 34407-7 96393-7 64835-7 83635-7 0
    85642-7 66391-7 98706-7 35233-7 2r93642-7 35233-7 98706-7 66391-7
    85642-7 ql12
\
\ Mu's X,R Directions
\
82** -44752-7 -29356-7 29356-7 -77358-7 -50745-7 50745-7 -99829-7 -65486-7
    65486-7 -1181-5 -77471-7 77471-7 -1339-5 -87835-7 87835-7 -16403-6
    -1076-5 1076-5 -20209-6 -13257-6 13257-6 -23406-6 -15354-6 15354-6
    -26214-6 -17196-6 17196-6 -28751-6 -1886-5 1886-5 -32717-6 -21461-6

```

```

21461-6 -37667-6 -24709-6 24709-6 -50191-6 -32924-6 32924-6 -66105-6
-43363-6 43363-6 -89576-6 -5876-5 5876-5 -12192-5 -79977-6 79977-6
-15413-5 -10111-5 10111-5 -18621-5 -12215-5 12215-5 -21808-5 -14306-5
14306-5 -24971-5 -16381-5 16381-5 -2811-4 -18439-5 18439-5 -31216-5
-20477-5 20477-5 -34291-5 -22494-5 22494-5 -50167-5 -4334-4 -14888-5
14888-5 4334-4 -73377-5 -67942-5 -4334-4 -14888-5 14888-5 4334-4 67942-5
-90121-5 -86507-5 -67941-5 -4334-4 -14887-5 14887-5 4334-4 67941-5
86507-5 -97371-5 -95899-5 -95182-5 -669-3 -42676-5 -14659-5 14659-5
42676-5 669-3 95182-5 95899-5 -99709-5 -98201-5 -87227-5 -68507-5 -437-3
-15011-5 15011-5 437-3 68507-5 87227-5 98201-5 q112
\
\ Eta's Z, theta Directions
93** 3r-99999-5 3r-99997-5 3r-99995-5 3r-99993-5 3r-99991-5 3r-99987-5
3r-99984-5 3r-99973-5 3r-99966-5 3r-99959-5 3r-99946-5 3r-99929-5
3r-99974-5 3r-99981-5 3r-99998-5 3r-99254-5 3r-98805-5 3r-98251-5
3r-97593-5 3r-96832-5 3r-95968-5 3r-95003-5 3r-93937-5 3r-86506-5
7r-6794-4 9r-43339-5 11r-22778-5 11r-7625-5
3r-99999-5 3r-99997-5 3r-99995-5 3r-99993-5 3r-99991-5 3r-99987-5
3r-99984-5 3r-99973-5 3r-99966-5 3r-99959-5 3r-99946-5 3r-99929-5
3r-99974-5 3r-99981-5 3r-99998-5 3r-99254-5 3r-98805-5 3r-98251-5
3r-97593-5 3r-96832-5 3r-95968-5 3r-95003-5 3r-93937-5 3r-86506-5
7r-6794-4 9r-43339-5 11r-22778-5 11r-7625-5
\
\ Data Arrays
1** 50.0
1** 11 30.48 11 33.1216 11 34.0741 11 53.6575 11 54.61
11 119.38 91 127.0 11 180.975 41 181.9275 91 208.28
11 262.99 11 263.9425 274.0025 11 274.32 11 274.955
280.035 191 283.1216 11 394.6525 11 395.605 11 400.685
11 401.6375 11 411.7975 11 412.75 11 417.83 11 419.08
31 421.64 191 436.98 640.08
4** 41 0.0 41 5.08 11 11.43 11 12.3825 11 12.70
11 13.6525 91 14.6050 29.865
11 30.1625 11 31.115 11 33.675 11 34.925 11 36.195
21 37.1475 11 40.005 41 40.9575 11 47.3075 21 48.260
11 51.1175 52.07 11 53.34 21 54.2925 11 57.15
21 58.1025 51 60.96 141 76.20 152.40
9ss 509 608 520 520 498 498 619 542 531 619 619 542 542 487 498
564
\ 487 \ Borated Glass
\ 498 \ Water
\ 509 \ Air
\ 520 \ Beryllium
\ 531 \ Concrete
\ 542 \ SS304
\ 553 \ Lithiated Paraffin
\ 564 \ Borated Polyethylene
\ 575 \ NAK
\ 586 \ Krafon - X4
\ 597 \ Krafon - N3
\ 608 \ Aluminum
\ 619 \ Carbon Steel
10ss -509 -509 -608 -520 -498 -498 -619 -619 -542 -542 -542 -542
-531 -531 -531 -531 -531 -531 -531 -531 -531 -487 -487 -487
-487 -487
-564 -564 -564 -564
\ -487 -487 -487 -487 -487
\ -498 -498
\ -509 -509
\ -520
\ -531 -531 -531 -531 -531 -531 -531 -531 -531
\ -542 -542 -542 -542
\ -553 -553 -553 -553 -553

```

```

'      -564 -564 -564 -564
'      -575 -575 -575 -575 -575
'      -586 -586 -586 -586 -586 -586 -586 -586 -586
'      -597 -597 -597 -597 -597 -597 -597 -597 -597 -597
'      -597 -597 -597
'      -608
'      -619 -619
118$ 79 85 109 43 1 85 73 181 169 175 181 193 1 85 79 103
      109 115 145 151 181 55 109 97 55 115
      1 73 55 61
      55 109 97 85 115
      1 85
      79 85
      43
      1 85 79 103 109 115 145 151 181
      169 175 181 193
      1 73 85 31 37
      1 73 55 61
      73 85 1 79 109
      1 109 85 181 167 355 97 305 107
      1 103 193 55 73 109 85 181 265 157 79 97 325
      109
      73 181
10** 3.64-5 9.74-6 6.03-2 1.2362-1 6.68-2 3.34-2 1.96-3 8.39-2 1.77-2
      1.77-3 6.02-2 7.83-3 7.86-3 4.39-2 1.05-3 1.40-4 2.39-3 1.58-2
      6.90-4 2.92-3 3.10-4 4.6-3 5.0-4 2.4-3 4.491-2 1.803-2
      7.13-2 3.41-2 4.87-4 1.97-3
      4.6-3 5.0-4 2.4-3 4.491-2 1.803-2
      6.68-2 3.34-2
      3.64-5 9.74-6
      1.2362-1
      7.86-3 4.39-2 1.05-3 1.40-4 2.39-3 1.58-2 6.90-4
      2.92-3 3.10-4
      1.77-2 1.77-3 6.02-2 7.83-3
      5.926-2 3.339-2 1.125-2 5.565-4 6.944-3
      7.13-2 3.41-2 4.87-4 1.97-3
      2.333-2 2.706-2 6.0227-2 8.700-5 8.0617-3
      3.5132-3 1.56547-3 1.68835-2 2.30976-3 1.47849-4
      6.79683-3 1.40822-3 4.36106-4 1.73562-2
      5.5255859-2 9.098190-4 4.5519689-4 3.674539-3
      3.948935-2 1.4342403-3 1.2503101-2 1.979797-4
      2.2395701-4 1.3892318-4 1.1036668-3 1.20233647-4
      4.3712224-4
      6.03-2
      1.96-3 8.39-2
318$ 4 5 6 7 8 9 5r1
      10r10 11 3r12 3r13 3r14 2r15 3 2
      2r16
32** 6r0.0 5r0.0
      0.0 2r11.43 2r12.70 2r30.1625 2r47.3075 51.1175
      13.6525 2r36.195 40.005 2r53.34 57.150
      2r29.865 33.675 48.260 52.07 2r0.0
      0.0 40.9575
33** 6r152.40 11.43 12.70 30.1625 47.3075 51.1175
      2r12.3825 2r13.6525
      2r31.115 2r48.26 2r52.07 14.605 37.1475 2r40.9575
      54.2925 2r58.1025 31.115 2r34.925 53.34 57.15 2r11.43
      33.68 51.1175
34** 30.48 54.61 127.0 119.38 274.32 274.955
      34.0741 54.61 181.9275 263.8425 401.6375
      2r33.1216 2r53.6575 2r180.975 2r262.89
      2r400.685 127.0 280.035 2r411.7975 274.955
      2r394.6525 283.1216 2r417.83 274.955 395.605
      30.48 33.1216
      508.0 401.64
35** 54.61 119.38 274.32 127.0 274.955 6r640.08

```

```

34.0741 2r54.61 2r191.9275
2r263.8425 2r401.6375 640.08 180.975 2r412.75 640.08
2r395.605 640.08 2r419.08 640.08 400.685 640.08
33.1216 34.0741
2r640.08
1a** 640.08
28SS fl
28SS 2r11 2r31 2r51 2r81 2r111
30SS 1 66 4q2
t
,
, End Dort Input
,
EOF
ln -s $sedatdir/hfrvelm.lib fort.9
ln -s $sedatdir/224/hb22dtd.srce fort.48
ln -s $sedatdir/hb22rl1.p3flx fort.44
time dort < dort1 > dorto
cat dorto >hb22rl2.out
mv hb22rl2.out $mytmpdir
mv fort.48 $mytmpdir/hb22rl2.dsout
mv fort.47 $mytmpdir/hb22rl2.zbflx
rm f* g* d*
cd $rtnadir
rm -rf $tempdir
date

```

APPENDIX D

MCNP4A INPUT FILE

HB-2 HFIR Shielding Design - design effectiveness verification

C *****

C Cell Cards

C

20 12 -7.82 25 -26 96 -27 imp n=128 imp p=128 \$ Tube inc 7

21 14 1e-15 -25 96 -27 imp n=256 imp p=256 \$ Void

22 3 1002 26 96 -32 -2 3 97 -98 imp n=64 imp p=64 \$ Water

C Bore around beam tube in bioshield

23 3 1002 -28 29 -30 31 32 24 -27 26 96 imp n=32 imp p=32 \$ Water

C Concrete outside Bore (Beam Tube)

24 5 07506 32 -27 29 -28 30 -2 96 imp n=16 imp p=16 \$ Top

25 5 07506 32 -27 29 -28 3 -31 96 imp n=16 imp p=16 \$ bottom

26 5 07506 32 -27 -29 97 -2 3 96 imp n=16 imp p=16 \$ Left

27 5 07506 32 -27 28 -98 -2 3 96 imp n=16 imp p=16 \$ Right

C Guide Box

28 12 -7.82 -33 34 -37 -39 27 -42 imp n=1024 imp p=1024 \$ Top

29 12 -7.82 -36 35 -37 -39 27 -42 imp n=1024 imp p=1024 \$ Bottom

30 12 -7.82 -37 38 -34 36 27 -42 imp n=1024 imp p=1024 \$ right

31 12 -7.82 -39 40 -34 36 27 -42 imp n=1024 imp p=1024 \$ left

C Water outside guide box in bore

32 3 1002 -30 33 -28 29 27 -48 imp n=256 imp p=256 \$ top

33 3 1002 31 -35 -28 29 27 -48 imp n=256 imp p=256 \$ bottom

34 3 1002 -33 35 39 29 27 -48 imp n=256 imp p=256 \$ left

35 3 1002 -33 35 -28 37 27 -48 imp n=256 imp p=256 \$ right

C Concrete outside bore (guide box)

36 5 07506 27 -48 29 -28 30 -2 imp n=64 imp p=64 \$ top

37 5 07506 27 -48 29 -28 3 -31 imp n=64 imp p=64 \$ bottom

38 5 07506 27 -48 -29 97 -2 3 imp n=64 imp p=64 \$ left

39 5 07506 27 -48 28 -98 -2 3 imp n=64 imp p=64 \$ right

C Bellows

40 12 -7.82 -43 33 45 -46 48 -47 imp n=512 imp p=512 \$ top

41 12 -7.82 44 -35 45 -46 48 -47 imp n=512 imp p=512 \$ bottom

42 12 -7.82 -33 35 45 39 48 -47 imp n=512 imp p=512 \$ left

43 12 -7.82 -33 35 -46 37 48 -47 imp n=512 imp p=512 \$ right

C Opening in concrete A2

44 7 1115 -43 33 45 -46 47 -49 imp n=512 imp p=512 \$ top

45 7 1115 44 -35 45 -46 47 -49 imp n=512 imp p=512 \$ bottom

46 7 1115 -33 35 45 39 47 -49 imp n=512 imp p=512 \$ left

47 7 1115 -33 35 -46 37 47 -49 imp n=512 imp p=512 \$ right

C Concrete surrounding bellows and A2

48 5 07506 48 -49 43 -2 97 -98 imp n=128 imp p=128 \$ top

49 5 07506 48 -49 -44 3 97 -98 imp n=128 imp p=128 \$ bottom

50 5 07506 48 -49 -43 44 97 -45 imp n=128 imp p=128 \$ left

51 5 07506 48 -49 -43 44 46 -98 imp n=128 imp p=128 \$ right

C Opening B

52 7.1115 49 -54 -50 33 52 -53 imp.n=1024 imp.p=1024 \$ top

53 7.1115 49 -54 51 -35 52 -53 imp.n=1024 imp.p=1024 \$ bottom

54 7.1115 49 -54 -33 35 52 39 imp.n=1024 imp.p=1024 \$ left

55 7.1115 49 -54 -33 35 -53 37 imp.n=1024 imp.p=1024 \$ left

C Concrete outside B

56 5.07506 49 -54 50 -2 -98 97 imp.n=256 imp.p=256 \$ top

57 5.07506 49 -54 -51 3 -98 97 imp.n=256 imp.p=256 \$ bottom

58 5.07506 49 -54 -50 51 -52 97 imp.n=256 imp.p=256 \$ left

59 5.07506 49 -54 -50 51 -98 53 imp.n=256 imp.p=256 \$ right

C Opening C in Concrete

60 5.07506 54 -57 55 -56 -50 33 imp.n=2048 imp.p=2048 \$ top

61 5.07506 54 -57 55 -56 51 -35 imp.n=2048 imp.p=2048 \$ bottom

62 5.07506 54 -57 55 39 -33 35 imp.n=2048 imp.p=2048 \$ left

63 5.07506 54 -57 -56 37 -33 35 imp.n=2048 imp.p=2048 \$ right

C Concrete outside C

64 5.07506 54 -57 -98 97 50 -2 imp.n=512 imp.p=512 \$ top

65 5.07506 54 -57 -98 97 -51 3 imp.n=512 imp.p=512 \$ bottom

66 5.07506 54 -57 -55 97 -50 51 imp.n=512 imp.p=512 \$ left

67 5.07506 54 -57 -98 56 -50 51 imp.n=512 imp.p=512 \$ right

C Opening in concrete D

68 5.07506 57 -62 -61 60 -58 33 imp.n=4096 imp.p=4096 \$ top

69 5.07506 57 -62 -61 60 59 -35 imp.n=4096 imp.p=4096 \$ bottom

70 5.07506 57 -62 60 39 -33 35 imp.n=4096 imp.p=4096 \$ left

71 5.07506 57 -62 -61 37 -33 35 imp.n=4096 imp.p=4096 \$ right

C Concrete outside D

72 5.07506 57 -62 -98 97 58 -2 imp.n=8192 imp.p=8192 \$ top

73 5.07506 57 -62 -98 97 -59 3 imp.n=8192 imp.p=8192 \$ bottom

74 5.07506 57 -62 -60 97 -58 59 imp.n=8192 imp.p=8192 \$ left

75 5.07506 57 -62 -98 61 -58 59 imp.n=8192 imp.p=8192 \$ right

C Opening E in concrete

76 5.07506 62 -64 65 66 -63 33 imp.n=8192 imp.p=8192 \$ top

77 5.07506 62 -64 65 66 59 -35 imp.n=8192 imp.p=8192 \$ bottom

78 5.07506 62 -64 65 39 -33 35 imp.n=8192 imp.p=8192 \$ left

79 5.07506 62 -64 37 66 -33 35 imp.n=8192 imp.p=8192 \$ right

C Concrete outside E

80 5.07506 62 -64 -98 97 63 -2 imp.n=32768 imp.p=32768 \$ top

81 5.07506 62 -64 -98 97 -59 3 imp.n=32768 imp.p=32768 \$ bottom

82 5.07506 62 -64 -65 97 59 -63 imp.n=32768 imp.p=32768 \$ left

83 5.07506 62 -64 -98 -66 59 -63 imp.n=32768 imp.p=32768 \$ right

C Air in beam room

84 4.4.614e-5 -2 135 -98 97 64 -42 imp.n=131072 imp.p=131072 \$ top

86 4.4.614e-5 -135 59 97 139 64 -42 imp.n=131072 imp.p=131072 \$ left

87 4.4.614e-5 -135 59 138 -98 64 -42 imp.n=131072 imp.p=131072 \$ right

C Concrete floor of beam room
 88 5 07506 -42 64 -59 3 -98 97 imp:n=16384 imp:p=16384
 C Void outside problem
 90 14 1e-15 2 -100 -102 103 -104 105 imp:n=0 imp:p=0 Stop
 91 14 1e-15 -3 101 -102 103 -104 105 imp:n=0 imp:p=0 Sbottom
 92 14 1e-15 -97 103 -2 3 -104 105 imp:n=0 imp:p=0 Sleft
 93 14 1e-15 98 -102 -2 3 -104 105 imp:n=0 imp:p=0 Sright
 94 14 1e-15 105 -96 -2 3 97 -98 imp:n=0 imp:p=0 Sback
 95 14 1e-15 42 -104 -2 3 97 -98 imp:n=0 imp:p=0 Sfront
 C Beam guide outer box
 96 6 07071 27 -42 70 71 -67 72 imp:n=512 imp:p=512 Stop
 97 6 07071 27 -42 70 71 -73 68 imp:n=512 imp:p=512 Sbottom
 98 6 07071 27 -42 70 -74 -72 73 imp:n=512 imp:p=512 Sleft
 99 6 07071 27 -42 71 -75 -72 73 imp:n=512 imp:p=512 Sright
 C void surrounding guides in guide box
 100 14 1e-15 27 -42 -109 -110 -107 67 imp:n=1024 imp:p=1024 Stop
 101 14 1e-15 27 -42 -109 -110 108 -68 imp:n=1024 imp:p=1024 Sbottom
 102 14 1e-15 27 -42 -110 -70 -67 68 imp:n=1024 imp:p=1024 Sleft
 103 14 1e-15 27 -42 -109 -71 -67 68 imp:n=1024 imp:p=1024 Sright
 C Beam guide 3
 104 6 07071 -42 84 -86 -72 73 91 imp:n=512 imp:p=512 S left
 105 6 07071 -42 -85 87 -72 73 82 imp:n=512 imp:p=512 S right
 C void inside 3
 106 14 1e-15 69 -42 86 -87 -72 73 imp:n=256 imp:p=256
 C beam guide 2
 107 6 07071 -42 -79 82 -72 73 85 imp:n=512 imp:p=512 Sleft
 108 6 07071 -42 -80 -83 -72 73 78 imp:n=512 imp:p=512 Sright
 C void in guide 2
 109 14 1e-15 69 -42 83 -82 -72 73 imp:n=256 imp:p=256
 C Beam guide 1
 110 6 07071 -42 -76 78 -72 73 80 imp:n=512 imp:p=512 Sleft
 C Beam guide 4
 111 6 07071 -42 -89 -92 -72 73 95 imp:n=512 imp:p=512 Sleft
 112 6 07071 -42 -88 91 -72 73 -84 imp:n=512 imp:p=512 Sright
 C Void inside 4
 113 14 1e-15 69 -42 -91 92 -72 73 imp:n=256 imp:p=256
 C beam guide 5
 114 6 07071 -42 -93 95 -72 73 89 imp:n=512 imp:p=512
 C void inside guide 5
 157 14 1e-15 69 -42 -95 74 -72 73 imp:n=256 imp:p=256 Sleft
 C Void inside guide 1
 158 14 1e-15 69 -42 -78 75 -72 73 imp:n=256 imp:p=256 Sright
 C shielding between guides
 159 7 1115 -64 85 79 -72 73 imp:n=2048 imp:p=2048 \$ btwn 3 & 2

160 7 1115 80 76 -64 -72 73 imp n=2048 imp p=2048 \$ btwn 1&2
 161 7 1115 -64 -84 88 -72 73 imp n=2048 imp p=2048 \$btwn 3&4
 162 7 1115 -64 93 89 -72 73 imp n=2048 imp p=2048 \$btwn 4&5
 C Void in plenum
 163 14 1e-15 27 -69 75 74 -72 73 imp n=256 imp p=256
 C Borated poly inside guide box
 164 7 1115 27 -42 -34 107 -38 -40 imp n=1024 imp p=1024 \$ top
 165 7 1115 27 -42 36 -108 -38 -40 imp n=1024 imp p=1024 \$ bottom
 166 7 1115 27 -42 -107 108 -38 109 imp n=1024 imp p=1024 \$ right
 167 7 1115 27 -42 -107 108 -40 110 imp n=1024 imp p=1024 \$ left
 C Borated poly outside guide box step 1
 168 7 1115 64 -42 -113 37 112 -111 imp n=2048 imp p=2048 \$ right
 169 7 1115 64 -42 -114 39 112 -111 imp n=2048 imp p=2048 \$ left
 170 7 1115 64 -42 -37 -39 -111 33 imp n=2048 imp p=2048 \$ top
 171 7 1115 64 -42 -37 -39 112 -35 imp n=2048 imp p=2048 \$ bottom
 C Borated Poly Step 2
 172 7 1115 64 -42 -115 113 112 -111 imp n=4096 imp p=4096 \$ right
 173 7 1115 64 -42 -116 114 112 -111 imp n=4096 imp p=4096 \$ left
 C Borated Poly Step 3
 176 7 1115 64 -42 -117 115 112 -111 imp n=8192 imp p=8192 \$ right
 177 7 1115 64 -42 -118 116 112 -111 imp n=8192 imp p=8192 \$ left
 C Void After wedges end
 180 14 1e-15 64 85 79 -72 73 -42 imp n=1024 imp p=1024 \$ btwn 3 & 2
 181 14 1e-15 80 76 64 -72 73 -42 imp n=1024 imp p=1024 \$ btwn 1&2
 182 14 1e-15 64 -84 88 -72 73 -42 imp n=1024 imp p=1024 \$btwn 3&4
 183 14 1e-15 64 93 89 -72 73 -42 imp n=1024 imp p=1024 \$btwn 4&5
 C Concrete outside guides step 1
 184 5 .07506 64 -42 117 -120 -119 59 imp n=8192 imp p=8192 \$ right
 185 5 .07506 64 -42 118 -121 -119 59 imp n=8192 imp p=8192 \$ left
 186 5 .07506 64 -42 -117 -118 -119 111 imp n=8192 imp p=8192 \$ top
 187 5 .07506 64 -42 -117 -118 59 -112 imp n=8192 imp p=8192 \$ bottom
 C Concrete step2
 188 5 .07506 64 -42 120 -122 -119 59 imp n=16384 imp p=16384 \$ right
 189 5 .07506 64 -42 121 -123 -119 59 imp n=16384 imp p=16384 \$ left
 C step3
 190 5 .07506 64 -42 122 -124 -119 59 imp n=16384 imp p=16384 \$ right
 191 5 .07506 64 -42 123 -125 -119 59 imp n=16384 imp p=16384 \$ left
 C step 4
 192 5 .07506 64 -42 124 -126 -119 59 imp n=32768 imp p=32768 \$ right
 193 5 .07506 64 -42 125 -127 -119 59 imp n=32768 imp p=32768 \$ left
 C step 5
 194 5 .07506 64 -42 126 -128 -119 59 imp n=32768 imp p=32768 \$ right
 195 5 .07506 64 -42 127 -129 -119 59 imp n=32768 imp p=32768 \$ left
 C step 6

196 5 07506 64 -42 128 -130 -119 59 imp.n=65536 imp.p=65536 S right
 197 5 07506 64 -42 129 -131 -119 59 imp.n=65536 imp.p=65536 S left
 C step 7
 198 5 07506 64 -42 130 -132 -119 59 imp.n=65536 imp.p=65536 S right
 199 5 07506 64 -42 131 -133 -119 59 imp.n=65536 imp.p=65536 S left
 C Air outside concrete
 200 4 4.614e-5 64 -42 -132 -133 -134 119 imp.n=131072 imp.p=131072 S top
 201 4 4.614e-5 64 -42 133 -137 -134 59 imp.n=131072 imp.p=131072 S left
 202 4 4.614e-5 64 -42 -136 132 -134 59 imp.n=131072 imp.p=131072 S right
 C Air for detection
 203 4 4.614e-5 64 -42 -137 -136 -135 134 imp.n=262144 imp.p=262144 S top
 204 4 4.614e-5 64 -42 137 -139 -135 59 imp.n=262144 imp.p=262144 S left
 205 4 4.614e-5 64 -42 136 -138 -135 59 imp.n=262144 imp.p=262144 S right

c surface cards

2 pz 200 Stop
 3 pz -150 Sbottom
 \$ define beam tube - vessel to beam guide plenum
 22 cx 11.43 S sixth increment
 23 cx 13.335
 24 px 187.96
 25 cx 19.606 S seventh increment
 26 cx 20.241
 27 px 290.3855
 28 py 29.884 S bore in concrete
 29 py -29.884
 30 pz 29.884
 31 pz -29.884
 32 cz 274.32 S bio-shield wall - inner
 \$ vacuum box around beam guides
 33 pz 15.24 \$ upper boundary
 34 pz 14.2875
 35 pz -27.432 \$ lower boundary
 36 pz -26.48
 \$ right boundary
 37 p 289.56 23.368 15.24 289.56 23.368 -27.432 1297.94 77.74 15.24
 38 p 289.56 22.415 15.24 289.56 22.415 -27.432 1297.94 76.787 15.24
 \$ left boundary
 39 p 289.56 -23.368 15.24 289.56 -23.368 -27.432 1297.94 -77.74 15.24
 40 p 289.56 -22.415 15.24 289.56 -22.415 -27.432 1297.94 -76.787 15.24
 41 px 289.56 \$ back boundary
 42 px 840 \$ front boundary
 \$ define bellows location opening
 43 pz 38.735

44 pz -39.37
 45 py -48.895
 46 py 48.895
 47 px 391.795
 48 px 383.54
 49 px 400.05 \$ opening in conc. a2 front
 50 pz 167.64 \$ opening in conc. b
 51 pz -39.37
 52 py -38.1
 53 py 38.1
 54 px 433.07
 55 py -43.18 \$ opening in conc. c
 56 py 48.18
 57 px 466.09
 58 pz 99.06 \$ opening in conc. d
 59 pz -106.68
 60 py -123.825
 61 py 123.825
 62 px 556.26
 63 pz 106.68 \$ opening in conc. e
 64 px 640.08
 65 p 556.26 -140.97 106.68 556.26 -140.97 -106.68 640 -210.82 106.68
 66 p 556.26 140.97 106.68 556.26 140.97 -106.68 640 210.82 -106.68
 67 pz 11.7475 \$ beam guide plenum - outer
 68 pz -11.24
 69 px 294.925
 70 p 289.56 -14.53 11.7 289.56 -14.53 -11.24 1297.94 -67.818 -11.24
 71 p 289.56 14.53 11.7 289.56 14.53 -11.24 1297.94 67.818 -11.24
 72 pz 9.2075 \$ beam guide plenum - inner
 73 pz -8.70
 74 p 289.56 -11.987 9.20 289.56 -11.987 -8.7 1297.94 -65.275 -8.7
 75 p 289.56 11.987 9.20 289.56 11.987 -8.7 1297.94 65.275 -8.7
 \$ beam guide 1 - outer ang=3 deg
 76 p 294.925 5.522 11.75 294.925 5.522 -11.24 1297.94 57.695 -11.24
 \$ plane separating guide 1 and 2 at ang=2.25 deg
 77 p 294.925 7.622 11.75 294.925 7.622 -11.24 1297.94 48.578 -11.24
 \$ beam guide 1 - inner
 78 p 294.925 7.622 9.2075 294.925 7.622 -8.70 1297.94 60.195 9.2075
 \$ beam guide 2 - outer ang=1.5 deg
 79 p 294.925 0 11.745 294.925 0 -11.24 1297.94 26.265 11.7475
 80 p 294.925 10.163 11.75 294.925 10.163 -11.2 1297.94 36.428 -11.2
 \$ plane dividing guides 2 and 3 ang=.75 deg
 81 p 294.925 2.54 11.75 294.925 2.54 -11.24 1297.94 15.67 11.75
 \$ beam guide 2 - inner

```

82 p 294.925 2.54 9.2075 294.925 2.54 -8.70 1297.94 28.805 9.2075
83 p 294.925 7.622 9.2075 294.925 7.622 -8.70 1297.94 33.887 -8.70
    S beam guide 3 - outer ang=0 deg
84 py -5.08
85 py 5.08
    S beam guide 3 - inner
86 py -2.54
87 py 2.54
    S beam guide 4 - outer
88 p 294.925 0 11.745 294.925 0 -11.24 1297.94 -26.265 11.745
89 p 294.925 -10.163 11.7 294.925 -10.163 -11.2 1297.94 -36.428 11.7
    S plane dividing guides 3 and 4 ang= 75 deg
90 p 294.925 -2.54 11.75 294.925 -2.54 -11.24 1297.94 -15.67 11.75
    S beam guide 4 - inner
91 p 294.925 -2.54 9.2 294.925 -2.54 -8.7 1297.94 -28.805 9.2075
92 p 294.925 -7.622 9.2 294.925 -7.622 -8.7 1297.94 -33.887 -8.7
    S beam guide 5 - outer
93 p 294.925 -5.522 11.75 294.925 -5.522 -11.24 1297.94 -57.695 -11.24
    S plane separating guide 4 and 5 at ang=2.25 deg
94 p 294.925 -7.622 11.75 294.925 -7.622 -11.24 1297.94 -48.578 -11.24
    S beam guide 5 - inner
95 p 294.925 -7.622 9.2075 294.925 -7.622 -8.70 1297.94 -60.195 9.2075
    S define boundaries of problem
96 px 187.96 $ back boundary
97 py -250 $ left boundary
98 py 250 $ right boundary
C void outside problem
100 pz 400 $ top boundary
101 pz -400 $ bottom boundary
102 py 550 $ right boundary
103 py -550 $ left boundary
104 px 1000 $ front boundary
105 px -200 $ back boundary
C beginning of shielding wedges
106 px 510 $ not used!
C poly in vacuum box
107 pz 13.02
108 pz -18.86
109 p 289.56 20.42 15.24 289.56 20.42 -27.432 1297.94 72.3 15.24
110 p 289.56 -20.42 15.24 289.56 -20.42 -27.432 1297.94 -72.3 15.24
C Poly outside vacuum box - step1
111 pz 45.24
112 pz -56.8
113 p 289.56 33.368 15.24 289.56 33.368 -27.432 1297.94 87.74 15.24

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114 p 289.56 -33.368 15.24 289.56 -33.368 -27.432 1297.94 -87.74 15.24
 C step2
 115 p 289.56 43.368 15.24 289.56 43.368 -27.432 1297.94 97.74 15.24
 116 p 289.56 -43.368 15.24 289.56 -43.368 -27.432 1297.94 -97.74 15.24
 C step3
 117 p 289.56 53.368 15.24 289.56 53.368 -27.432 1297.94 107.74 15.24
 118 p 289.56 -53.368 15.24 289.56 -53.368 -27.432 1297.94 -107.74 15.24
 C Conc bioshielding - step1
 119 pz 115.24 \$ bottom = surf 59
 120 p 289.56 63.368 15.24 289.56 63.368 -27.432 1297.94 117.74 15.24
 121 p 289.56 -63.368 15.24 289.56 -63.368 -27.432 1297.94 -117.74 15.24
 C step 2
 122 p 289.56 73.368 15.24 289.56 73.368 -27.432 1297.94 127.74 15.24
 123 p 289.56 -73.368 15.24 289.56 -73.368 -27.432 1297.94 -127.74 15.24
 C step 3
 124 p 289.56 83.368 15.24 289.56 83.368 -27.432 1297.94 137.74 15.24
 125 p 289.56 -73.368 15.24 289.56 -73.368 -27.432 1297.94 -127.74 15.24
 C step 4
 126 p 289.56 93.368 15.24 289.56 93.368 -27.432 1297.94 147.74 15.24
 127 p 289.56 -93.368 15.24 289.56 -93.368 -27.432 1297.94 -147.74 15.24
 C step 5
 128 p 289.56 103.368 15.24 289.56 103.368 -27.432 1297.94 157.74 15.24
 129 p 289.56 -103.368 15.24 289.56 -103.368 -27.432 1297.94 -157.74 15.24
 C step 6
 130 p 289.56 113.368 15.24 289.56 113.368 -27.432 1297.94 167.74 15.24
 131 p 289.56 -113.368 15.24 289.56 -113.368 -27.432 1297.94 -167.74 15.24
 C step 7
 132 p 289.56 123.368 15.24 289.56 123.368 -27.432 1297.94 177.74 15.24
 133 p 289.56 -123.368 15.24 289.56 -123.368 -27.432 1297.94 -177.74 15.24
 C setup air in beam room for detectors
 134 pz 145.24 \$top lower
 135 pz 147.24 \$top upper
 136 p 289.56 153.368 15.24 289.56 153.368 -27.432 1297.94 207.74 15.24 \$i
 137 p 289.56 -153.368 15.24 289.56 -153.368 -27.432 1297.94 -207.74 15.24
 138 p 289.56 155.368 15.24 289.56 155.368 -27.432 1297.94 209.74 15.24 \$o
 139 p 289.56 -155.368 15.24 289.56 -155.368 -27.432 1297.94 -209.74 15.24

c Data Cards

mode n p

ssr psc=1

de5 2.5e-8 1.0e-7 1.0e-6 1.0e-5 1.0e-4 1.0e-3

1.0e-2 1.0e-1 5.0e-1 1.0 2.5 5.0 7.0 10.0

14.0 20.0

df5 3.67e-6 3.67e-6 4.46e-6 4.54e-6 4.18e-6 3.76e-6

3.56e-6 2.17e-5 9.26e-5 1.32e-4 1.25e-4 1.56e-4
 1.47e-4 1.47e-4 2.08e-4 2.27e-4
 de15 2.5e-8 1.0e-7 1.0e-6 1.0e-5 1.0e-4 1.0e-3
 1.0e-2 1.0e-1 5.0e-1 1.0 2.5 5.0 7.0 10.0
 14.0 20.0
 df15 3.67e-6 3.67e-6 4.46e-6 4.54e-6 4.18e-6 3.76e-6
 3.56e-6 2.17e-5 9.26e-5 1.32e-4 1.25e-4 1.56e-4
 1.47e-4 1.47e-4 2.08e-4 2.27e-4
 de25 2.5e-8 1.0e-7 1.0e-6 1.0e-5 1.0e-4 1.0e-3
 1.0e-2 1.0e-1 5.0e-1 1.0 2.5 5.0 7.0 10.0
 14.0 20.0
 df25 3.67e-6 3.67e-6 4.46e-6 4.54e-6 4.18e-6 3.76e-6
 3.56e-6 2.17e-5 9.26e-5 1.32e-4 1.25e-4 1.56e-4
 1.47e-4 1.47e-4 2.08e-4 2.27e-4
 de35 2.5e-8 1.0e-7 1.0e-6 1.0e-5 1.0e-4 1.0e-3
 1.0e-2 1.0e-1 5.0e-1 1.0 2.5 5.0 7.0 10.0
 14.0 20.0
 df35 3.67e-6 3.67e-6 4.46e-6 4.54e-6 4.18e-6 3.76e-6
 3.56e-6 2.17e-5 9.26e-5 1.32e-4 1.25e-4 1.56e-4
 1.47e-4 1.47e-4 2.08e-4 2.27e-4
 de45 2.5e-8 1.0e-7 1.0e-6 1.0e-5 1.0e-4 1.0e-3
 1.0e-2 1.0e-1 5.0e-1 1.0 2.5 5.0 7.0 10.0
 14.0 20.0
 df45 3.67e-6 3.67e-6 4.46e-6 4.54e-6 4.18e-6 3.76e-6
 3.56e-6 2.17e-5 9.26e-5 1.32e-4 1.25e-4 1.56e-4
 1.47e-4 1.47e-4 2.08e-4 2.27e-4
 de55 2.5e-8 1.0e-7 1.0e-6 1.0e-5 1.0e-4 1.0e-3
 1.0e-2 1.0e-1 5.0e-1 1.0 2.5 5.0 7.0 10.0
 14.0 20.0
 df55 3.67e-6 3.67e-6 4.46e-6 4.54e-6 4.18e-6 3.76e-6
 3.56e-6 2.17e-5 9.26e-5 1.32e-4 1.25e-4 1.56e-4
 1.47e-4 1.47e-4 2.08e-4 2.27e-4
 de65 2.5e-8 1.0e-7 1.0e-6 1.0e-5 1.0e-4 1.0e-3
 1.0e-2 1.0e-1 5.0e-1 1.0 2.5 5.0 7.0 10.0
 14.0 20.0
 df65 3.67e-6 3.67e-6 4.46e-6 4.54e-6 4.18e-6 3.76e-6
 3.56e-6 2.17e-5 9.26e-5 1.32e-4 1.25e-4 1.56e-4
 1.47e-4 1.47e-4 2.08e-4 2.27e-4
 de75 2.5e-8 1.0e-7 1.0e-6 1.0e-5 1.0e-4 1.0e-3
 1.0e-2 1.0e-1 5.0e-1 1.0 2.5 5.0 7.0 10.0
 14.0 20.0
 df75 3.67e-6 3.67e-6 4.46e-6 4.54e-6 4.18e-6 3.76e-6
 3.56e-6 2.17e-5 9.26e-5 1.32e-4 1.25e-4 1.56e-4
 1.47e-4 1.47e-4 2.08e-4 2.27e-4

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de85  2.5e-8 1.0e-7 1.0e-6 1.0e-5 1.0e-4 1.0e-3
      1.0e-2 1.0e-1 5.0e-1 1.0 2.5 5.0 7.0 10.0
      14.0 20.0
df85  3.67e-6 3.67e-6 4.46e-6 4.54e-6 4.18e-6 3.76e-6
      3.56e-6 2.17e-5 9.26e-5 1.32e-4 1.25e-4 1.56e-4
      1.47e-4 1.47e-4 2.08e-4 2.27e-4
de4   2.5e-8 1.0e-7 1.0e-6 1.0e-5 1.0e-4 1.0e-3
      1.0e-2 1.0e-1 5.0e-1 1.0 2.5 5.0 7.0 10.0
      14.0 20.0
df4   3.67e-6 3.67e-6 4.46e-6 4.54e-6 4.18e-6 3.76e-6
      3.56e-6 2.17e-5 9.26e-5 1.32e-4 1.25e-4 1.56e-4
      1.47e-4 1.47e-4 2.08e-4 2.27e-4
de14  0.01 0.03 0.05 0.07 0.1 0.15 0.2 0.25 0.3 0.35 0.4 0.45 0.5
      0.55 0.6 0.65 0.7 0.8 1.0 1.4 1.8 2.2 2.6 2.8 3.25 3.75 4.25
      4.75 5. 5.25 5.75 6.25 6.75 7.5 9. 11. 13. 15.
df14  3.96e-6 5.82e-7 2.90e-7 2.58e-7 2.83e-7 3.79e-7 5.01e-7
      6.31e-7 7.59e-7 8.78e-7 9.85e-7 1.08e-6 1.17e-6 1.27e-6
      1.36e-6 1.44e-6 1.52e-6 1.68e-6 1.98e-6 2.51e-6 2.99e-6
      3.42e-6 3.82e-6 4.01e-6 4.41e-6 4.83e-6 5.23e-6 5.6e-6
      5.8e-6 6.01e-6 6.37e-6 6.74e-6 7.11e-6 7.66e-6 8.77e-6
      1.03e-5 1.18e-5 1.33e-5
f5:n 650 154 -100 0    $ section e right top
f15:n 650 154 0 0      $ section e right center
f25:n 650 154 100 0    $ section e right bottom
f35:n 650 25 146 0     $ section e top right
f45:n 650 0 146 0      $ section e top middle
f55:n 650 -25 146 0    $ section e top left
f65:n 650 0 -146 0     $ section e bottom center
f75:n 650 25 -146 0    $ section e bottom left
f85:n 650 -25 -146 0   $ section e bottom right
f4:n 183 184 185       $ 30cm from surface
f14:p 183 184 185      $ same
f6:n 59 83 159 184 166 168 $ Kerma detectors in various materials
f16:p 59 83 159 184 166 168 $ ditto
m1 4009 1.2362e-1 $be
m2 24000 1.77e-2 $ss304
    25055 1.77e-3
    26000 6.02e-2
    28000 7.83e-3
m12 06000 1.96e-2 $Carbon Steel
    26000 8.39e-2 $ total 7.8212
m3 1001 6.68e-2 $water
    8016 3.34e-2
m4 7014 3.64e-2 $air

```

8016 9.74e-6
 m14 7014 3.64e-2 Sair (void mix
 8016 9.74e-6 \$total 1.0E-15
 m5 1001 7.86e-3 Sconcrete (fusion reactor shield mix)
 8016 4.39e-2
 11023 1.05e-3
 12000 1.40e-4
 13027 2.39e-3
 14000 1.58e-2
 19000 6.90e-4
 20000 2.92e-3
 26000 3.10e-4
 m6 5010 4.60e-3 SPyrex borated glass mcnp formula
 13027 5.00e-4
 11023 2.40e-3
 8016 4.49e-2
 14000 1.80e-2
 m7 1001 7.13e-2 Sborated polyethylene mcnp formula
 6000 3.41e-2
 5010 4.87e-4
 5011 1.97e-3
 m8 1001 5.926e-2 Slithiated paraffin
 6000 3.338e-2
 8016 1.125e-2
 3006 5.565e-4
 3007 6.944e-3
 m9 6000 2.333e-2 Snak
 8016 2.706e-2
 1001 6.0227e-2
 7014 8.700e-5
 13027 8.0617e-3 \$total = 0.1187657
 m10 1001 3.5132e-3 \$krafton-x4
 13027 1.56547e-3
 8016 1.68835e-2
 26000 2.30976e-3
 22000 1.47849e-4
 82000 6.79683e-3
 11023 1.40822e-3
 74000 4.36106e-4
 16032 1.73562e-2 \$total = 4.690401e-2
 m11 1001 5.5255e-2 \$krafton-n3
 12000 9.0982e-4
 28000 4.5520e-4
 5010 3.6745e-3

```
6000 3.9489e-2
13027 1.4342e-3
8016 1.2503e-2
26000 1.9798e-4
64000 2.2396e-4
22000 1.3892e-4
7014 1.1037e-3
11023 1.2023e-4
74000 4.3712e-4 Stotal = 0.1159439
prdmp
print
```

APPENDIX E

MCNP4A SOURCE TERM

The source supplied by ORNL is in the form of a DORT boundary source located in the thimble section of the beam tube. The plane source is located 30.4 cm from the center of the core. The source specifies a 244 angle quadrature set, 61 neutron energy groups and 27 photon energy groups.

If the source is converted directly to a MCNP4a source, the angular distribution of the supplied source is lost. It was suggested by Dr. Pevey that a point source extrapolated from the boundary source using a non-linear regression would better approximate the angular distribution of the supplied source. Additionally a point source some distance behind the plane source approximates the flux distribution well.

A least-squares analysis of the source was performed and a coupled set of equations to determine the source strength and the source location.

$$\frac{s z^2}{16 \pi^2} \left(\frac{1}{4 z^2} - \frac{1}{4 (R^2 + z^2)^2} \right) = \frac{z}{4 \pi} \left(\frac{1}{z} - \frac{1}{(R^2 + z^2)^{1/2}} \right) \sum f_i$$

$$s z \int_0^R \frac{r (r^2 + z^2) - 3 z^2 r}{(r^2 + z^2)^4} dr = \sum f_i \int_0^R \frac{r (r^2 + z^2) - 3 z^2 r}{(r^2 + z^2)^{5/2}} dr$$

where,

s = source strength

z = distance from original plane to point source

r = radius of the disk source

f_i = value of the current in interval i

A code was developed to solve this system of equations for the source strength and source position. Two sources, one neutron and one photon, were created. The photon source has a strength of 7×10^{18} photons per second at a distance of 24 cm from the initial disk source. The photon has a strength of 6×10^{18} neutrons per second and a position 24 cm from the surface. The energy group structure is maintained for both sources.

Energy group limits (in eV)

Neutron energy boundaries (62)

```
0.14918E-08 0.12214E+08 0.10000E+08 0.81873E-07 0.67032E-07 0.54881E-07
0.44933E-07 0.36788E+07 0.30119E+07 0.24660E+07 0.23457E-07 0.22313E-07
0.20190E-07 0.16530E+07 0.13534E+07 0.11080E-07 0.90718E-06 0.74273E-06
0.60810E-06 0.52340E+06 0.49787E+06 0.38774E-06 0.30197E-06 0.29849E-06
0.29721E-06 0.29452E+06 0.27324E+06 0.22371E-06 0.18316E-06 0.14996E-06
0.12277E-06 0.86517E+05 0.56562E+05 0.52475E-05 0.34307E-05 0.28501E-05
0.27000E-05 0.26058E+05 0.24788E+05 0.23579E-05 0.15034E-05 0.91188E+04
0.55308E+04 0.37074E+04 0.30354E+04 0.26126E-04 0.22487E-04 0.20347E-04
0.12341E+04 0.74852E+03 0.45400E+03 0.27536E+03 0.16702E-03 0.10130E+03
0.47851E+02 0.22603E+02 0.10677E+02 0.50435E+01 0.23824E-01 0.11253E+01
0.41399E+00 0.10000E-04
```

Gamma energy boundaries (24)

```
0.14000E+08 0.10000E+08 0.80000E+07 0.75000E-07 0.70000E-07 0.60000E-07
0.50000E+07 0.40000E+07 0.30000E+07 0.25000E-07 0.20000E+07 0.15000E+07
0.10000E+07 0.70000E+06 0.60000E+06 0.51000E-06 0.40000E+06 0.30000E+06
0.15000E+06 0.10000E+06 0.70000E+05 0.45000E+05 0.20000E+05 0.10000E+05
```

Probability of particle falling into a particular energy group

Neutron probabilities by group (61)

0.80189E-05 0.37309E-04 0.11933E-03 0.31025E-03 0.62364E-03 0.10316E-02
0.13556E-02 0.11821E-02 0.11558E-02 0.66082E-03 0.77707E-03 0.19610E-02
0.52507E-02 0.50982E-02 0.41717E-02 0.35445E-02 0.34236E-02 0.25210E-02
0.23842E-02 0.91289E-03 0.47099E-02 0.43541E-02 0.19762E-03 0.73102E-04
0.15443E-03 0.12636E-02 0.32653E-02 0.30608E-02 0.28927E-02 0.27953E-02
0.46282E-02 0.53934E-02 0.93405E-03 0.52048E-02 0.22834E-02 0.66382E-03
0.43535E-03 0.61321E-03 0.61325E-03 0.55208E-02 0.61582E-02 0.62167E-02
0.50269E-02 0.25312E-02 0.19056E-02 0.19138E-02 0.12799E-02 0.64666E-02
0.65524E-02 0.66380E-02 0.67221E-02 0.68038E-02 0.68827E-02 0.10488E-01
0.10641E-01 0.10781E-01 0.10908E-01 0.11048E-01 0.11261E-01 0.15485E-01
0.76870E-00

Photon probabilities by group (23)

0.19859E-05 0.99421E-04 0.48431E-02 0.52301E-03 0.87171E-02 0.59108E-02
0.11588E-01 0.23686E-01 0.17951E-01 0.42067E-01 0.32880E-01 0.52394E-01
0.45832E-01 0.18809E-01 0.23024E-01 0.31687E-01 0.40531E-01 0.14959E-00
0.10523E+00 0.11346E+00 0.16552E+00 0.10389E+00 0.17591E-02

APPENDIX F

SCALE 4.2 INPUT FILES

