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The Manager
Schenectady Naval Reactors Office
United States Department of Energy
Schenectady, New York

Subject: Space Reactor Radiation Shield Design Summary, For Information

Enclosures: 1. Space Reactor Radiation Shield Design Summary
2. Shield Thermal Analyses

Dear Sir:

PURPOSE

The purpose of this letter is to provide a summary of the Prometheus space reactor radiation shield design status at the time of program restructuring.

BACKGROUND

Project Prometheus was established in 2003 with a goal of developing the first nuclear reactor-powered propulsion system for a spaceship and demonstrating that it can be operated safely and reliably for civilian deep-space exploration missions. The initial application of space fission power being evaluated was the Jupiter Icy Moons Orbiter (JIMO), a nuclear electric propulsion spaceship intended to perform deep-space scientific research. In March 2004, the Naval Reactors Program was assigned responsibility for design and delivery of the Reactor Module for Project Prometheus. In March 2005, the Naval Reactors Prime Contractor Team (NRPCT) recommended and Naval Reactors approved a gas-cooled reactor with a directly coupled Brayton energy conversion system for further development.

In September 2005, NASA priorities changed and NR Program participation in Project Prometheus was terminated by mutual agreement. Closeout activities were initiated to provide an orderly conclusion to NRPCT work, and to enable the future re-start of a nuclear space reactor project. This shield design summary was generated to document the pre-conceptual design work that was in-progress.

DISCUSSION

The reactor radiation shield is a segment of the Reactor Module that provides neutron and gamma radiation attenuation to reduce reactor radiation damage to the spaceship electronics at the mission module and to other reactor and spaceship components. Preliminary radiation limit requirements specified by NASA's Jet Propulsion Laboratory for reactor radiation to the mission module were 25 kRad equivalent silicon damage (gamma) and 5×10^{10} n/cm² 1 MeV equivalent silicon damage (neutrons).

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The reactor radiation shield mass is minimized by shielding only within a very narrow cone angle as shown in Figure 1, thus creating a locally protected shadow region, rather than shielding around the entire reactor. A 12°/6° elliptical shadow cone was based on spaceship radiator and high gain antenna configurations.

Figure 1 Reactor Radiation Shield Shadow Cone and Basic Shield Shape

Enclosure (1) summarizes the reactor radiation shield design progress based on NRPCT (KAPL and Bettis) and Oak Ridge National Laboratory (ORNL) analyses. It discusses the pre-conceptual shield configuration considered and the radiation sources, material considerations, and mass estimates. Integration with the reactor, the reactor coolant and energy conversion subsystems, and the spaceship, are also discussed. A brief discussion of future work considerations and mission extensibility is included.

Enclosure (2) contains the results of preliminary thermal analyses to determine whether each of the material temperature limits can be met for each material set under consideration, and begins to assess the overall thermal management of the shield, piping, reactor control drive mechanisms, and the energy conversion region of the Reactor Module.

SUMMARY

The shield design work done to date focused on evaluating alternative material sets, potential configurations of these material sets within the shield, and overall shield configuration alternatives. The objective was to provide the required radiation attenuation to protect spaceship equipment, minimize mass, and ensure the shield itself would perform its function throughout long duration, deep-space missions. Summary findings from this design work are as follows:

- The NRPCT had narrowed the slate of candidate shield materials to five primary materials (lithium hydride (LiH), beryllium (Be), boron carbide (B₄C), tungsten (W), and water (H₂O)) and one potential alternate material (¹⁰B metal) (see Reference (a)). Four basic shield material sets were under consideration: 1) Be/B₄C/W, 2) Be/¹⁰B/W, 3) Be/B₄C/W/LiH, and 4) Be/B₄C/W/H₂O.
- Minimizing mass of the flight shield is a primary objective. Studies by ORNL (Reference (b)) showed that the mass of the shadow shield was higher than previous estimates. This was true despite the material composition of the shield. Careful integration of reactor, piping, energy conversion equipment, and spaceship interfaces will be needed to minimize shield mass. Key methods to minimize mass include minimizing reactor size and reflector operating envelope, and maximizing the shielding benefit of other required components (e.g. energy conversion and spaceship components).
- The gas coolant piping is an important factor in designing the shield configuration and thermal design. Minimizing the number and diameter of coolant pipes will minimize shield mass. Pipe shield caps would not be necessary for piping shield penetrations if the pipes spiral azimuthally through the shield just inside the outer radial surface. Investigation of shield masses for pipes going around the shield should be completed as part of future work.
- The lifetime average release of short lived volatile fission products to the coolant must be limited (estimated at $< 10^{-3}$ release to birth ratio) so a second gamma shield segment is not required between the coolant loops and the payload, which would significantly increase the reactor radiation shield mass. This level of fission product retention is a reasonable design assumption. Fission product and corrosion/erosion product irradiation effects on the alternator require further investigation.
- Gamma radiation interactions at the outer surface of the reactor or reflector could drive electrons off the surface. This interaction needs to be further investigated as both a potential electron radiation source that could bypass the shield and as a potential spaceship electrical charging source. Resolution of this concern will require close integration with spaceship designers.

A significant next step that had not been performed prior to program closeout was the performance of spaceship level trade studies that would provide an initial balance among equipment radiation dose limits, local equipment radiation shield requirements, spaceship boom length, and reactor radiation shield requirements.

CONCURRENCE

This letter and its enclosures have been reviewed by and have the concurrence of the managers of KAPL Space Power Plant Systems (Schwartzman), KAPL Space Materials (Simonson), KAPL FSO-Shielding (Pineau), Bettis SEA-Shielding (Karnes), and KAPL Reactor Engineering (McCoy).

Very truly yours,

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Approved by:

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REFERENCES

- (a) NRPCT Letter MDO-723-0049, "Space Shielding Materials for Prometheus Applications," dated January 20, 2006.
- (b) Oak Ridge National Laboratory, ORNL/LTR/NR-PROM1/05-27, "Reactor Shield Optimization in Support of the Prometheus 1 Project Preconceptual Design Analyses," dated October 2005.

Space Reactor Radiation Shield Design Summary

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Space Reactor Radiation Shield Design Summary

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SUMMARY AND CONCLUSIONS

The reactor radiation shield segment provides neutron and gamma radiation attenuation to reduce reactor radiation damage to the spaceship electronics at the mission module. Other components (e.g. motors, alternators, cables, reactor plant sensors, radiator materials) that are not as radiation sensitive as the electronics, but are in higher radiation fields closer to the reactor must also be protected. The calculated radiation exposure from reactor radiation for the mission module was limited to 25 kRad_{Si} (gamma) and 5×10^{10} n/cm² 1 MeV equivalent silicon damage (neutrons).

Key Conclusions

The reactor radiation shield would be composed of layers of several different materials (see Figure 2) to provide the most mass-effective attenuation of gamma and neutron radiation. The NRPCT had narrowed the slate of candidate reactor shield materials to five primary materials (LiH, beryllium, boron carbide, tungsten, and water) and one potential alternate material (¹⁰B metal) for Prometheus application (see Reference (j)). Four basic shield material sets were under consideration: 1) Be/B₄C/W, 2) Be/¹⁰B/W, 3) Be/B₄C/W/LiH, and 4) Be/B₄C/W/H₂O (also referred to as the 1) Be/B₄C, 2) ¹⁰B, 3) LiH, and 4) H₂O or water shields based on the main constituents). Tungsten was the preferred gamma shield material. Be and B₄C may be necessary at the reactor side of any of these shields to accommodate thermal control, nuclear instruments within the shield, and/or an aeroshell interface. The Be/B₄C shield is a low risk (see Table 2), low complexity shield, but generally has a higher mass than other material sets. The ¹⁰B shield is similar in capability to the Be/B₄C shield, but reduces the shield mass (~100kg), while slightly increasing the risk (see Table 2) and significantly increasing the cost relative to the Be/B₄C shield. The LiH shield has the lowest mass, but has high risk (see Table 2), complexity (thermal & potentially swelling - see Reference (k)), toxicity, and cost. The H₂O shield, possibly with soluble neutron poisons, has a low cost, but has high complexity (temperature, pressure, and gas control systems), higher mass, high risk (see Table 2), and significant corrosion concerns, especially if dissolved neutron poisons are included (Reference (j)). Significant irradiated materials testing would be required for either LiH or neutron-poisoned H₂O, which increases the lead time and cost of qualifying either material.

Shield mass for all four material sets were within 150 kg of each other for analyses with pipes through the shield. For pipes routed around the shield, LiH and H₂O shield masses would be about the same, but both would require shield caps located behind where the pipes reenter the shield shadow cone after the main shield. Therefore, the water shield would need LiH for the cap shields or pay a mass penalty to use Be. The LiH shield has the least sensitivity to mass impacts due to radiation limit or dimensional changes, while the Be/B₄C and ¹⁰B shields have higher flexibility in manufacturing complex shield configurations and accommodation of design changes late in the project. Use of LiH or neutron poisoned H₂O in the shield requires significant time and resources for material development and design iterations. Therefore, a short design/development time would tend to make it more difficult to develop either material, retire the design risks, and build the shield.

Other key conclusions are:

- The mass of the flight shield is a concern. Studies by ORNL (Reference (f)) showed that the mass of the shadow shield was higher than previous estimates. This was true regardless of the material composition of the shield. Careful integration of the reactor, piping, energy conversion system, and spaceship interfaces will be needed to minimize shield mass. Shield sensitivities to reactor and spaceship configurations were planned, but not started due to project termination. Shield sensitivities to energy conversion system configurations

had not been done, and would have been done after more detailed component/arrangement information became available. Potential opportunities to minimize shield mass include: shield material selection; minimizing reactor diameter, reactor length and distance from the shield; arranging reactor components (sliders, shutdown rods, fission gas plenum) forward or aft of the reactor; radially tailoring neutron and gamma shield thickness; trimming the shadow shield based on final spaceship configuration; investigating low mass shield structural materials; maximizing the shielding benefit of energy conversion system and xenon tank placement; routing pipes outside the shield; re-evaluation of mission module dose limits; and using higher dose capable payload components to shield lower dose capable components within the mission module.

- The size, numbers, and thermal requirements of the gas coolant piping are important factors in designing the shield configuration and the thermal design, particularly for a shield with hydrogenous materials. Minimizing the number and diameter of coolant pipes will minimize shield mass. Pipe shield caps are not necessary for piping penetrations if the pipes spiral azimuthally through the shield by between 70° and 180° degrees just inside the outer radial surface. Investigation of shield masses for pipes going around the shadow shield should be performed as part of future work.
- Lifetime average release of short lived volatile fission products to the coolant must be limited (estimated at $< 10^{-3}$ release-to-birth ratio) so a second gamma shield is not required between the coolant loop and the payload, which would significantly increase the reactor shield mass. This level of fission product retention is a reasonable design assumption. Fission product irradiation effects on the alternator, which is in the primary coolant path, requires further investigation.
- Reactors with shadow shielding could potentially lead to electron radiation and reactor electrical charging concerns. Gamma radiation interactions at the outer surface of the reactor or reflector could drive electrons off the surface. This interaction needs to be further investigated as both a potential electron radiation source that could bypass the shield and as a potential spaceship charging source, requiring close integration with spaceship designers to resolve.
- Control drive shaft radiation streaming through the shield is not a significant design concern. Radiation streaming effects are localized, with negligible effects at the payload.

BACKGROUND

Shielding requirements are based on the radiation tolerance of the electrical components, sensors, and other materials in the Reactor Module and in the remainder of the spaceship. These sensitive components are subjected to radiation damage from both the space environment and the reactor. The space environment sources of radiation include charged particles (electrons, protons, heavy ions, and secondary particles, such as spallation neutrons) and gamma radiation. In general, the shielding associated with environmental sources surrounds the sensitive components and consists of aluminum plate shielding with a thickness of 300 mils or more. This space radiation shielding is not part of the reactor radiation shield, but is an interface that should be considered in the shield design.

The reactor radiation shield segment shields spaceship components from radiation emitted by the reactor, such as direct neutron and gamma radiation, neutron capture, activation of coolant

or structure, fission product release to the coolant, and electrons leaving the reactor/reflector surface.

SHIELD DESIGN

The reactor radiation shield segment mass is minimized by shielding only within a very narrow cone angle as shown in Figure 1, thus creating a locally protected shadow region, rather than shielding around the entire reactor. A 12°/6° elliptical shadow cone was based on spaceship radiator and high gain antenna configurations. A uniform 12° cylindrical cone could have been used, but would have resulted in a higher shield mass. Outside of the shadow cone radiation levels are very high. In general, all equipment is kept within the shadow cone to prevent high direct reactor radiation doses and to prevent scattering reactor neutron radiation back into the shadow cone and increasing doses to other components. Routing of the gas coolant pipes outside of the shield is an option, but additional shielding (cap shields) would be required to reduce the neutron radiation scattered by the piping.

The reactor shield attenuates gamma and neutron radiation by about a factor of 100 and 100,000, respectively. The mission module is located approximately 50m from the core. The radiation doses are reduced by about another factor of 1000 due to distance ($\sim 1/r^2$) fall off. Other equipment, such as energy conversion components and spaceship propellant tanks can also provide shielding if they can be beneficially arranged between the reactor shield and the mission module, which would reduce reactor, especially gamma, shield mass. The xenon in the propellant tank(s) would provide the shielding benefit, not the tank structure. Shielding analyses would need to account for the xenon use for propulsion over the duration of the mission.

The reactor shield size is a primary driver for shield mass. The reactor shield radius is determined by projection at a 12° (6°) angle from the forward outboard most corner of the reactor or reflector as shown in Figure 2. The reactor and reflector diameter, and reactor length are, therefore, key factors in determining shield radius (mass). Placing the shield as close to the reactor as possible minimizes the shield radius (mass). The reactor control device types (e.g., sliders, drums, etc.), and assumptions (e.g. stuck out sliders) also factor into the shield radius. The shield cone half-angles (12°/6°) affect shield radius and are controlled by required radiator area, antenna size, and payload distance. The shield thickness is primarily driven by reactor power, lifetime, core to payload distance, payload neutron dose limits, and shield material selection and optimization. Neutron attenuation by other equipment, as discussed above, can also slightly reduce shield thickness.

Reactor Radiation Shield Segment Functions

Primary Functions:

1. Limit radiation dose from the reactor to the spaceship, energy conversion, and reactor control equipment.

Secondary Functions:

1. Provide a thermal neutron fluence, appropriate temperature environment, and a protective housing for the nuclear instrument detectors.
2. Provide structural supports for the reactor vessel, control drive mechanisms (CDMs), and energy conversion system equipment structural supports.

3. Provide piping passage and thermal control between the reactor and energy conversion segment.

Figure 1: Reactor Shield Shadow Cone and Basic Shield Shape

Figure 2: Example (12° direction) of Shield Size Projection from the Reactor and Shield Material Laminations

Radiation Requirements Basis

Table 1 shows the radiation requirements used for shield design studies.

Table 1: Radiation Requirements

Location/Radiation Type	Component Testing Total	Calculation Limit – Radiation Design Margin (RDM) = 2.0	Reactor Contribution
Payload Plane – Total Ionizing Dose (TID) (e.g. gamma)	300 kRad _{Si}	150 kRad _{Si}	25 kRad _{Si}
Payload - Displacement Damage Dose (DDD) (e.g. neutrons)	2×10^{11} n/cm ² 1 MeV equivalent silicon damage fluence	1×10^{11} n/cm ² 1 MeV equivalent silicon damage fluence	5×10^{10} n/cm ² 1 MeV equivalent silicon damage fluence
Immediately behind shield TID	0.5 GRad material	0.25 GRad material	0.2 GRad material
Immediately behind shield DDD	TBD	TBD	TBD

Shielding design studies assumed a 1 MW_{th} reactor with a lifetime of 15 effective full power years (EFPY). A change in energy requirement, like reduced power level or lifetime, would not change the Table 1 requirements, but would change the shield mass. A flat radiation profile is

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assumed at the payload plane rather than flat at the center and increasing at the edges. A flat profile is assumed because of the large distance (~50m) from the core to the payload plane. At large payload distances, radial variations in radiation profile near the shield are smoothed out at the payload plane. Radial variations of the payload plane radiation requirements would be considered if the xenon ion thruster propellant tanks could be used to help reduce reactor shield mass. The reactor contribution to the mission module dose specified in Table 1 was based on early trade studies by JPL between reactor radiation shielding and mission module environmental shielding. The project was terminated prior to trade studies among NRPCT, JPL and Northrop Grumman Space Technology (NGST) on reactor shielding versus spaceship configuration options.

Materials

Table 2 compares shield design parameters for the four shield material options currently being considered.

Table 2: Preliminary shield design parameters for four mass competitive shield material systems

	Be/B ₄ C/W	Be/ ¹⁰ B/W	Be/B ₄ C/W/LiH	Be/B ₄ C/H ₂ O (w/ or w/o soluble poisons)
Temperature	Not Limiting	Not Limiting	~600K**-800K (620°F-980°F)	~273K-450K (32°F -350°F)
Requires containment	No	No	Yes	Yes
Pressure requirement	No	No	<< 15psia	< 150psia
Toxicity	Be LiH	Very High Medium volume	Very High Low volume	Very High Low volume
			High High volume	
Cost (Drivers)	Medium (Be manufacturing)	High (¹⁰ B enrichment)	High (LiH development, and LiH & Be manufacturing)	Low (Be manufacturing & H ₂ O w/ soluble poison development)
Corrosion Concern	Negligible	Negligible	Medium	High
Thermal expansion From room temperature to operating temperature	Small	Small	7%	12% liquid & prevent freezing
Irradiation swelling	Negligible	Negligible	< 2% within temperature range ** during operation, otherwise up to 25% (Reference (k))	Negligible (~15psi pressure contribution)
Mass (Delta)*	2436 kg (140 kg)	2347kg (51 kg)	2296 kg (0 kg)	2434 kg (138 kg)
Development	Low	Medium	High	High
Schedule Risk	Low	Medium	High	High
Lifetime Uncertainty	Low	Low	High	High
Insulation Requirement Pre-startup/ Operational	No/No	No/No	No/Yes	Yes/Yes

	Be/B ₄ C/W	Be/ ¹⁰ B/W	Be/B ₄ C/W/LiH	Be/B ₄ C/H ₂ O (w/ or w/o soluble poisons)
Heating/Cooling required	No	No	Possibly heating during coast mode	Cooling during full power operation/ coast mode TBD/ possibly heating prior to startup
Shield Micrometeoroid Protection Required	No	No	Provided by required external insulation	No – pipes through shield Yes – pipes around shield ++
Thermal design complexity	Low	Low	High (limited/high temperature range, gaps between LiH blocks, irradiation effect on conductivity)	Medium (limited/low temperature range, cooling loops from reject heat loop)
Temperature at nuclear instruments during operation (Lower is better)	460 K	490 K	680 K	580 K
Shield shape flexibility	High	High	Medium	Low
Reactor orientation flexibility (i.e. Horizontal reactor)	High	High	Medium	Low
Piping configuration flexibility	High	High	Medium	Low
Risk to CDM motion	Low	Low	High	Medium
Temperature near CDM (< ~450 K simplifies CDM thermal design	440 K	435 K	665 K	<450 K

* Based on Reference (f)– 12°/6° elliptical shield, 49m to payload, including front & back structural and containment plate, but modified to include 2x SP-100 canning thickness to eliminate single point failure. Does not include shielding benefit from reactor, energy conversion system, or spaceship components (mass effect of these is significant), nor through shield structural support. (Table 5, Case 5)

** It is possible that this lower limit could be reduced based on SNAP/ANP data and future testing.

++ If the pipes are routed through the shield, then other shielding material around the water tank would protect the water tank. If the pipes are routed around the shield, then the water tank would extend to the surface of the shield and would need micrometeoroid protection.

Many shield materials, in addition to those in Table 2, were evaluated by ORNL (References (a), (e), and (d)) and NRPCT (Reference (j)) with the goal of identifying mass competitive and less developmental materials than LiH, but were rejected as discussed in Reference (j). The three material sets identified in Table 2, in addition to the LiH material set, have the potential to meet these goals. A few other (e.g., UH₃, ZrH_x, ZrB₂, ZrC) more developmental materials considered could be used in a mass competitive shield, but not without keeping the LiH or H₂O. However, since there was no mass advantage, the addition of another significant material development program would not be warranted.

Radiation Sources

Reactor

Reactor operational radiation sources include fission neutrons, prompt and delayed fission gammas, and reactor (e.g. core, vessel, reflector, etc.) capture gammas. Most of these sources are included in coupled neutron-gamma calculations. How delayed fission product gammas are accounted-for when using a coupled neutron/gamma cross section set needs to be investigated. Delayed fission product gammas could increase reactor gamma sources by approximately 50% based on fission gamma energy yields, if they are not properly accounted-for in the coupled cross sections. The importance of including this source is also dependent on the shield composition, as demonstrated by comparing Figure 3 and Figure 4. The LiH does not produce significant capture gammas in the shield after the W, so the primary gammas are more important. However, in the Be/B₄C shield, gammas are produced in the materials after the W, so the reactor (primary) sources are less important. Methods to account for the delayed fission product gamma source must be included in future methods development plans. The portion of the delayed fission products that are volatile could also move from the fuel region to the fission gas plenum region, thus moving a portion of the gamma source. Shield analyses were focused on fast reactors. More recent thermal reactor configurations being considered would require further evaluation.

Figure 3: ORNL (Reference (a)) plots of reactor (Primary) gamma and shield capture (Secondary) gamma dose contributions through the shield for a LiH based shield.

Figure 4: ORNL (Reference (a)) plots of reactor (Primary) gamma and shield capture (Secondary) gamma dose contributions through the shield for a Be/B₄C based shield.

All portions of the reactor, reflector, structure and hardware must be shadowed by the reactor shield in order to prevent neutrons scattering around the shield from those components. Capture gamma production in the reflector, structure, etc. could also be important sources, if these components extend beyond the shadow shield cone. Another source of radiation that may not be covered by coupled cross sections is the Be(γ ,n) reaction that occurs in the reflector and shield. This is not expected to be a major source of neutrons, but must be considered.

A reactor with shadow shielding could potentially lead to electron radiation and reactor electrical charging concerns. The space reactor will not be surrounded by significant amounts of shielding, like land-based reactors. Gamma radiation is attenuated mainly by transferring energy to electrons. On the outer surface of the reactor vessel or reflector, the gamma levels may be high enough to drive a significant quantity of electrons from those surfaces creating an electron flux leaving the surface. Electrons leaving the surface of the reactor/reflector could cause a significant net electrical charge to build up on the reactor and spaceship. The buildup of a positive charge on the spaceship may quickly reduce the net current leaving the spaceship to zero. However, electrons may also leave the reactor surface and return to the spaceship at a different location behind the shield, depending on the spaceship voltage and the electrical characteristics of the spaceship components and structures. NRPCT performed conservative, simplified calculations to estimate whether this could be an issue. The estimate assumed that all of the gamma energy deposited in the outer 1 mm of the reactor vessel or reflector canning would cause electrons to leave the surface. It was assumed that one electron is released per 34 eV of energy deposited, and that there was no voltage potential to overcome. Based on these worst case

estimates, an initial current of ~10 Amps could leave a bare reactor vessel, and ~1 Amp could leave the outside of a continuous reflector. Since these were very conservative calculations, NRPCT does not expect these levels to actually occur. However, the calculations indicate that this electron source needs to be further investigated as both a potential electron radiation source that could bypass the shield, and as a potential spaceship charging source, requiring close integration with spaceship designers to resolve.

Shield

The main radiation source within the shield is capture gammas. Capture gammas are produced when neutrons are absorbed in the shielding material or structure. Minimizing the capture gamma yield, total energy released per capture, and individual gamma energy will help minimize gamma shield mass. Capturing neutrons as early in the shield as possible maximizes the gamma attenuation of the remaining portions of the neutron shielding, allowing the high density gamma shielding to be placed more forward in a smaller diameter section of the shield cone, thereby reducing mass. When a neutron is captured, it must release the binding energy for the additional neutron which is about 8 MeV for most elements, but is lower for elements with very low mass (Reference (b)). ${}^6\text{Li}$ and ${}^{10}\text{B}$ are particularly good materials to minimize the gamma source because most of the energy is released as charged particle energy, which is easily attenuated. A small amount of gamma energy is also released as low energy gammas, which require less mass to attenuate. This is one of the reasons why LiH and B_4C help minimize shield mass, especially if they can be used near the front of the shield. For water, dissolved boron and/or LiOH was being considered to minimize the gamma source. Use of these low gamma-producing materials either before and/or after other materials, like Be, W, structural and piping materials, that have high capture gamma yields and gamma energies helps reduce the gamma source production in those materials by locally depressing the thermal flux. Reference (f) showed that placing boron-containing materials adjacent to structural materials usually, but not always, reduced overall shield mass.

Coolant Activation

Activation of coolant, which then passes behind the shield, was also considered as a radiation source. Activation of the He/Xe coolant, primarily Cs from Xe activation, will not be a significant contributor of radiation dose to energy conversion system sensors, cabling, etc. and payload equipment based on Reference (c). Doses near pipes from coolant activation would be on the order of 10-100 kRad_{Si} compared to ~100 MRad_{Si} from radiation through the shield and space radiation. Doses at the payload from coolant activation would be on the order of 1-10 Rad_{Si} compared to a reactor contribution limit of 25 kRad_{Si} . The beta radiation dose to the alternator internals would need further evaluation. Chemistry/materials effects from cesium production also need to be evaluated.

Fission Products in the Coolant

Preliminary estimates of radiation doses at the payload plane were made assuming all volatile fission products (Table 3) are released to the coolant. These estimates were made to determine the level of fission product retention that would be required to prevent radiation damage to the energy conversion system equipment and to eliminate the need for a separate gamma shield behind the energy conversion system to protect payload equipment. A separate gamma shield would significantly increase the shield mass. Initial

estimates indicated that an average lifetime 10^{-3} release to birth ratio (R/B) for short lived fission products would be adequate to prevent the need for a separate gamma shield and to protect equipment outside of the pressure boundary. "Release" means release to the He/Xe coolant rather than just release from the fuel pellet. Radiation doses are affected mostly by short lived fission products that decay within a few weeks. There are many factors which can affect this release, including fuel temperature, diffusion/release rates out of fuel pellets, condensation in fission gas plenums, fuel pin initial hermeticity, fuel pin hermeticity failure rate, diffusion rate out of fuel pin, number of fuel pins, time in life of fuel or pin degradation, fission product deposition on reactor internals, etc.

The alternator may have radiation sensitive components inside the pressure boundary and requires further study. Radiation sensitive components inside the pressure boundary could be subject to beta radiation damage because the sensitive alternator materials are not covered by material that will attenuate beta particles. Additional factors that would affect the radiation exposure of alternator internals include fission product deposition in turbines and heat exchangers, alternator temperature, fission product deposition in alternators, whether the alternators are cooled with main coolant flow or localized flow. The effects of fission products in the coolant on ground test reactor maintenance and operations must also be considered and could also limit allowable fission product release.

Table 3: Volatile fission products and transuranics

Element	Boiling Point (K) (Reference (b))
Krypton	120
Xenon	165
Bromine	332
Iodine	458
Arsenic	876
Cesium	944
Selenium	958
Rubidium	961
Cadmium	1040
Zinc	1180
Tellurium	1261

Attenuation

A few shield optimization approaches were developed to improve radiation attenuation during NRPCT and ORNL shield mass optimizations for a shadow shield configuration. The minimum shield mass for hydrogenous configurations does not necessarily occur with the minimum amount of Be/B₄C before the W as seemed to be the case for the SP-100 design. In most of these hydrogenous shield configurations, ORNL determined that about 15-30 cm of Be/B₄C before the W resulted in the minimum shield mass.

Minimizing the > 1 MeV equivalent silicon damage neutron fluence and the resulting gamma dose does not minimize shield mass. Lower energy (i.e., epithermal and thermal) neutrons must also be attenuated and absorbed to prevent creating a gamma source further aft in the

shield where gamma shield mass would be higher. It is also preferable to capture these neutrons in low gamma energy and low gamma yield materials like ^{10}B or ^6Li . Thick regions with little neutron absorption (e.g. Be) should be avoided. ORNL studies (Reference (e)) indicated that laminations of 5cm Be and 0.5cm of B_4C was as effective as Be doped with 2% B, which means already developed materials (i.e. Be and B_4C) can be used with negligible mass penalty relative to pure Be.

ORNL performed mass sensitivities (Reference (e)) to determine whether pure ^{10}B had a shield mass advantage over natural boron ($^{\text{nat}}\text{B}$) in the Be/B/W shield concept and relative to Be+2%B (Be/ B_4C laminate). The study indicated that ^{10}B would save more than 125kg relative to $^{\text{nat}}\text{B}$ and about 100kg relative to Be+2%B (Be/ B_4C laminate). This 100kg savings relative to Be/ B_4C laminate was seen in all later studies with ^{10}B even if Be/ ^{10}B laminates are used. However, use of Be/ $^{10}\text{B}_4\text{C}$ did not show a significant benefit relative to Be/ $^{\text{nat}}\text{B}_4\text{C}$.

ORNL performed mass sensitivities (Reference (e)) to determine whether stainless steel, borated stainless steel, nickel alloy 230, or ZrB_2 were competitive with W as a gamma shield in a non-hydrogenous shield. The steel and nickel alloys were about 50 kg heavier than the shield using tungsten, and the shield that used ZrB_2 was about 100kg heavier than with W. Borated stainless steel showed no mass benefit relative to non-borated stainless steel with boron on faces of the steel. Reference (p) indicates that ZrC is slightly less effective as a neutron and gamma shield material than ZrB_2 , therefore ZrC would not be effective either.

ORNL performed mass sensitivities (Reference (e)) to determine whether stainless steel, borated stainless steel, nickel alloy 230, or ZrB_2 were competitive with W as a gamma shield in a hydrogenous ($\text{H}_2\text{O} + ^6\text{LiOH}$) shield. These alternate gamma shielding material shields were all significantly higher mass (166-300kg) than with W.

ORNL performed mass sensitivities (Reference (e)) to determine whether the type of dissolved poison used in water shields affected shield mass for $^6\text{LiOH}$, ^{10}B , or $^6\text{LiOH} + ^{10}\text{B}$ and there was no difference. ORNL also performed mass sensitivities to determine the poison concentration level needed in water to minimize shield mass. The addition of poison had only a minor effect on neutron levels, but the gamma mass savings was estimated to be about 100kg. A $^6\text{LiOH}$ concentrations of 4200 ppm was adequate to provide the gamma source reductions and higher concentrations provided no significant additional mass benefit. ORNL also optimized a case which added Al_2O_3 insulation just before the water simulating a need to reduce heat input to the water portion of the shield. The addition of 5cm of Al_2O_3 increased shield mass by 40kg.

ORNL performed mass sensitivities (Reference (e)) to determine the effect of changing the gamma dose requirements for one hydrogenous ($\text{H}_2\text{O}/\text{W}$) and one non-hydrogenous (Be/ $^{10}\text{B}/\text{W}$) material set. Results indicated that for an increase of a factor of 10 in the allowable dose at the payload, which is equivalent to accounting for xenon tank attenuation, both the hydrogenous and non-hydrogenous shield masses decreased by an equivalent mass (~325kg) and all of the W mass was eliminated. Neither case would benefit from additional gamma attenuation. This study also indicates the significant benefit from component shielding. However, for the non-hydrogenous shield, the W was eliminated and significantly more Be was used to obtain that reduction. This means that a non-hydrogenous material set of just $^{10}\text{B}/\text{W}$ will not provide the adequate design space for high gamma attenuation from other components and Be must be kept as a possible shield material for the ^{10}B based shield. The goal with trying to use ^{10}B was to try to eliminate the toxic Be from the shield material set, but this cannot be done. Once the W has been eliminated and the ^{10}B replaced with Be, the amount of mass savings is limited for further increases in allowed gamma dose or alternate component attenuation. This conclusion is true for the Be/ B_4C based shield when the W has been eliminated. If the allowable gamma

dose requirement is reduced by a factor of 10, the increase in shield mass is much more significant, 722kg for a hydrogenous shield, and 642kg for a non-hydrogenous shield. Therefore, further restrictions of the gamma dose are definitely not desirable due to the high mass penalty.

Penetrations

Pipes

NRPCT piping penetration analyses focused on pipes that would go through the shield. Evaluations of pipes routed around the shield had not yet been done and should be further evaluated. Pipes routed around the shadow shield could simplify the shield configuration, assembly, and thermal design, particularly for LiH or H₂O shield concepts.

Pipe penetration shield analyses were performed by both ORNL and NRPT (References (f) & (i), respectively). Figure 6 and Figure 7 show an example of gamma and neutron radiation doses for various piping configurations through a non-hydrogenous shield from Reference (i). For non-hydrogenous shields, the lowest neutron radiation doses through the shield due to piping penetrations occurred when each pipe made a smooth spiral nearly 180° around the shield (see Figure 8). Both reports determined that the best (i.e. simplest and lowest mass) penetration configuration for non-hydrogenous shield configurations was a large angle spiral. The reason for this is that the large angle spiral, especially the smooth spiral, significantly reduces any straight through paths compared to shorter angle dog-leg, slanted or smooth spiral penetrations (See Figure 5). The low neutron streaming levels through the large angle spiral pipes can be compensated-for by a modest thickening of the shield and shield caps over the region where the pipe exits the back of the shield are not required. The large angle spiral also displaces more of the shield material than shorter spirals and is sufficient to offset the mass added by the required shield thickness increase. Although gamma doses at the payload do not increase significantly (Figure 6) due to adding piping penetrations, further investigation into the mass effects is warranted because of the high sensitivity of shield mass with changes in gamma shielding. Gamma flux plots at the back of the shield, reported in Reference (i), also indicate that radial tailoring of the gamma shielding should be investigated to reduce gamma shield mass.

Figure 5: Examples of various ways to route piping through the shield

Figure 6: Pipe penetration gamma streaming for various piping penetration configurations through the shield

Figure 7: Pipe penetration neutron streaming for various piping penetration configurations through the shield

Figure 8: Piping penetration configuration with a 180 degree smooth spiral inside the outer edge of the shield

For hydrogenous shields, it was assumed that the pipes would be routed through a Be cup region, as shown in Figure 9, rather than directly through the LiH or H₂O regions to avoid excessive temperature spikes in the shield around the pipes and because the hot leg pipe requires heat transfer to the shield to keep the pipe wall material below design limits (~900K/1160°F). Note that the Be cup design for the LiH is partial, i.e., two "C-shaped" regions to minimize the Be mass, while for the H₂O shield it is continuous to maintain a round pressure tank. Because the Be (1.85 g/cc) cup region displaces LiH, the shield mass for a very large angle spiral would be higher than for a lower angle spiral. Since the cold leg pipe does not need external cooling and could have external thermal insulation, the Be following the pipe may be able to be eliminated for the cold leg. For the hot leg, it may be possible to tailor the Be to follow the pipe rather than covering the entire region. This would complicate the Be and LiH region designs, but could reduce the shield mass. More detailed thermal analyses would be required to ensure that the pipe could be cooled via radiation to space with a reduced Be area. This type of Be cup region reduction is not as feasible for the H₂O shield because the higher pressure (~1 MPa/ 150psi) of the water tank is not compatible with the more angular geometry. In the water shield model, Figure 9, the water tank had flat front and back faces for ease of modeling and future analyses would have to investigate the shielding and mass effects of making these ends ellipsoidal to accommodate the tank pressure.

Analyses were performed to evaluate the need for shield caps at the pipe penetrations on the rear end of the shield. Shield caps would not be required for any of the shield material concepts. Elimination of the need for shield caps benefits the overall plant design, for example: 1) the reactor shield is simpler and has less mass, 2) the number of piping bends is reduced which reduces coolant pressure drop, and 3) the amount of piping near the control drive mechanisms (CDMs) is reduced.

The large diameter of the coolant pipes (~16cm) for the gas reactor concept makes routing the pipes through the shield a challenge. There are three locations in the shield that limit pipe routing. The first is whether there is sufficient space between the CDM shafts and the side of the shield for the pipe to enter the front face of the shield. The width of this space is largest at the 12° half cone angle direction. The amount of space is affected by the reflector outer radius, the reactor control type (drums, sliders, etc.), radial location of slider shafts on the reflector, drum radius, distance to front end of sliders/drums, slider support structure, reactor vessel to shield distance, etc. The second limiting location is the space

between the CDM shafts and the 6° radial side of the shield. The space along the 6° radial side is affected by the same factors as the front face space, but is also affected by shield length which is affected by reactor power level, reactor lifetime, amount of neutron attenuation from the energy conversion system components and Xe tank, payload distance, and payload dose requirements. This dimension determines whether there is sufficient space between the CDM shafts and the edge of the shield to route the pipe around inside the narrow edge of the shield for large azimuth bends within the shield like Figure 8. The third limiting location is the space between the CDM motors and the outer edge of the shield where the pipe exits the shield. This limitation at the back of the shield is similar to that at the front of the shield. Although the shield radius has increased (~7/15cm for 6°/12°) the CDM motor located on the back of the shield covers more of the back face of the shield, causing a limiting area near the 6° edge of the shield where there is insufficient space between the CDM motor and the edge of the shield for the pipe to exit. Also, it is not desirable to have the pipe exit the shield too close to the CDM motor where it could overheat the motor.

Additional sensitivity studies with regard to piping design are also needed. Shield mass sensitivities to varying pipe diameters are needed for pipes routed inside and outside the shield. Internal insulation in the hot piping will reduce neutron streaming for pipes routed through the shield, but increase neutron scattering for piping routed around the shield. Sensitivity studies for pipe wall thickness, particularly for pipes routed around the shield, are needed.

Figure 9: ORNL pipe routings for the hydrogenous shields.

Pipes are shown routed through a 70° azimuthal angle for the LiH shield (top) and the water shield (bottom). In both cases a dog-leg spiral is modeled. The Be cup in each case is shown as translucent and the LiH or water tank is opaque. (Reference (f)).

Control Drive Mechanisms

Reference (h) documents preliminary shielding analyses for control drive mechanism (CDM) penetrations through the shield. These results were based on a Be/B₄C shield, straight CDM penetrations through the shield, and narrow gaps (< 0.5 cm) between the shield and the solid shaft. This report concluded that gamma streaming at the back of the shield was minor (20% increase relative to no penetration). Neutron streaming at the back face of the shield was only a factor of 6 above levels with no streaming through the shield and the peaks were very narrow. Because the peaks were narrow, a non-radiation sensitive CDM component, such as the ball nut or other structure at the back face of the shield should scatter the narrow peaks and reduce neutron streaming peaking to much lower levels before reaching more radiation sensitive components in the CDM motor. Neutron and gamma radiation doses at other energy conversion system components would not be significantly affected by CDM streaming. Radiation doses at the payload were not affected by the CDM penetration streaming. At this point, CDM shaft sleeves with step changes in radius through the shield are not required to reduce radiation streaming, as long as CDM-to-sleeve gaps can be kept small. The gap size along the CDM shaft must be large enough to prevent obstructing CDM shaft movement.

The ability to maintain small CDM gap sizes is dependent on the shield materials. Non-hydrogenous shields (i.e. Be/B₄C or ¹⁰B based shields) can maintain smaller gaps because the thermal and radiation induced expansion/swelling would be small and canning along the shaft may not be required. For a H₂O based shield, a sleeve will be needed along the shaft, increasing streaming, and water pressure effects on the sleeve may require increased gap size. For a LiH based shield, a sleeve will be needed along the shaft, and thermal expansion (~7%) and potential irradiation induced swelling will require larger gap sizes. For these cases, a step change in shaft radius is more likely, which would complicate manufacturing.

SP-100 testing indicated CDM failure rates increased significantly above 450K (350°F) and even lower temperatures are preferred. The shield temperature is a significant factor in determining CDM temperatures because the CDMs are mounted on the back of the shield. Enclosure (2) results indicate that shield material temperatures at the back of the shield for the Be/B₄C, B, and H₂O based shields could be maintained below 450K (350°F), which would mean that CDM temperatures would be even cooler. The LiH based shield thermal design was configured to ensure LiH temperatures above ~600K (620°F) to minimize irradiation induced swelling. This temperature could be reduced by modifying the thermal configuration, if irradiation testing of LiH allowed reducing the minimum temperature below 600K (620°F) (Reference (k)). However, it was expected that LiH temperatures would always need to be higher than for non-LiH based shield concepts. Therefore, extra thermal design measures, such as insulating shims between the shield and the CDM, would be required to keep the CDMs cool if a LiH shield is used. The contour plots in Enclosure (2) were used to determine shield temperatures at the back of the shield for the Be/B₄C, B, H₂O and LiH based shields. The temperatures of the shield aft surface given in the Enclosure (2) tables are the peak shield temperature that could provide radiation heating of the energy conversion region, rather than the shield material temperatures near the CDM foundations. For example, in the LiH case the shield aft surface temperature in the Enclosure (2) table was on the outside of the multi-layer insulation used to keep the LiH hot and reduce radiation heating of the energy conversion subsystem components by the shield. The CDM foundation would be attached to the shield rather than to the insulation.

Nuclear Detectors

One of the primary locations for placing neutron detectors for reactor start-up and monitoring was in the front side of the shield. The neutron detectors are most sensitive to a thermal neutron flux. Figure 10 shows thermal neutron fluence (proportional to flux) in the first 10 cm of the shield for various materials considered to generate the thermal neutron signal for the detectors. Most of these materials produce a reasonable thermal neutron signal. The ^{11}B containing materials analyzed were enriched to an assumed 99.9% ^{11}B , but they may be more feasible with higher enrichment. For the hydrogenous materials, the need for additional canning would depress the thermal neutron signal further and would complicate the design. The hydrogenous materials (except possibly water which can recombine more easily) are also not viable due to high radiolytic decomposition at the front of the shield. Because Al_2O_3 is a thermal insulator, the detector would heat up due to nuclear heating, which would make it more difficult to meet detector temperature requirements. Keeping the temperature below 800K (980°F) is desirable to allow the use of off-the-shelf materials for the neutron detectors. Be provides the best thermal neutron signal and the best heat removal. Based on preliminary thermal model results discussed in Enclosure (2), all of the shield material concepts (i.e. LiH, H_2O , Be/ B_4C , and ^{10}B based shields) being considered could maintain temperatures at the front of the shield below 800K (980°F), even with pipes passing through the shield and with micrometeoroid shielding (insulation) which may be required outside the shield.

Shield features included to accommodate neutron detectors were to use layers of 0.5cm B_4C , 5cm Be, 0.5cm B_4C , 5cm Be, and 0.5cm B_4C immediately after the shield structural plate at the reactor end of the shield. The detectors would be inserted from the side of the shield into holes in the second layer of Be with a separate Be piece filling the outer portion of this hole not occupied by the detector or the cable. The B_4C laminations are there to minimize overall shield mass due to captures in structure and Be. The second and possibly third layer of B_4C would be replaced with Be near the detectors to maximize the thermal flux in the detectors. The neutron detector cables would be routed to the outer edge of the shield and along channels to the back of the shield. These channels would be just inside the outer radial structure or micrometeoroid shielding to allow parallel neutron detector and shield construction with installation of the neutron detectors late in the overall shield assembly process.

Figure 10: Thermal Neutron Fluence in the First 10 cm of Shield Materials Considered for Around Nuclear Instruments (Reference (g))

Shield Mass

Basis for Shield Mass Estimates for Reactor and Plant Mass Comparisons

Shield mass estimates to compare different reactor (Reference (t)) and plant (Reference (s)) concepts were made using the Quick Mass Analysis for Reactor Shields (QMARS) code (Reference (d)), developed by ORNL. QMARS scales shield mass based on reactor power, reactor size, distance to payload, and neutron and gamma radiation requirements. QMARS shield masses and sizes were normalized to the most recent shield mass estimates for one reactor design in Reference (f). However, the energy conversion system and xenon tank designs and locations were not set at the time the Reference (f) shielding analyses were started. Therefore, radiation attenuation provided by these components was estimated (3x for neutrons and 5x for gammas) and was the same for all shield mass estimates. The radiation limits at the payload were increased to account for this assumed attenuation. The above estimate assumes that the energy conversion subsystem components and xenon tank(s) are well-placed between the reactor shield and the payload electronics. Shield mass is significantly higher if the xenon tank cannot be used for shielding. Radiation attenuation from the energy conversion subsystem could also vary significantly depending on the number of components, mass of components, and component materials (e.g. amount of water in the gas cooler).

A few reactor and shield assumptions were made to ensure a fair mass comparison of the many different reactor concepts, but these do not reflect a specific design recommendation. The payload was assumed to be 49m from the core. A Be/B4C/W elliptical shield with 12° and 6° cone half angles was assumed. The effect of piping was

not included in QMARS mass estimates, but for a Be/B₄C/W shield the shield mass does not change significantly if pipes are routed through the shield. Fission gas plenums were assumed to be forward in the cold plenum. Sliders, if applicable, were assumed to slide open in the forward direction. It was assumed that sliders could be reduced in thickness by 25% at the forward end, which controls shield size but that drums could not be trimmed. Because the trimmed (void) section of a drum could be rotated toward the core, the trimming of the drums would have a larger effect on core neutron leakage than trimming the outside of a slider.

The core nuclear design basis assumed that one slider could be stuck full-open. To reduce shield mass, shield mass estimates assumed that all sliders could be moved toward the shield to the hot, beginning of life, critical position and could be interlocked to prevent moving any further back out. This assumption means that there could be no sliders stuck full-out and that slider operational modes with some sliders full-out and others further in would not be used. Allowing stuck full-out sliders, per the nuclear design basis, would significantly increase shield mass (~190±25kg) for slider controlled reactor cases because shield radii would increase by about 4cm/2cm in the 12°/6° directions, respectively. The relative shield mass differences between slider cases reported in References (s) and (t) would still be about the same since applying the one stuck full-out design basis criteria increases all slider case shield masses by about the same amount (±25kg). The small variation in shield mass between slider cases is due to changes in shield radius resulting from differences in slider stroke, which varies with core length. However, the shield mass of slider cases would increase by ~190kg relative to drum controlled cases in References (s) and (t) because a stuck drum does not affect shield mass.

Optimized Shield Masses

ORNL optimized four final shield material sets with pipes routed through the shield (References (e) and (f)). Table 4 shows the ORNL optimized shield material thicknesses starting at the front of the shield for each set of materials. As discussed above, the hydrogenous cases have Be regions in the vicinity of the pipes as shown in Figure 9.

Table 5 cases 1 through 4 provide ORNL mass results at various stages of the optimization process, and compares each alternative material set to the LiH case. Table 5 cases 5-6 are NRPCT mass estimates for incorporating design features to address design details and are based on the ORNL results and SP-100, plus extra canning for LiH and water shields to meet the single point failure criteria. The Table 5 masses should be used as a basis for comparison of material combinations rather than as absolute shield masses. The shield masses shown in Table 5 are significantly larger than would actually be expected. Shield masses decreased from ~2400kg (Be/B₄C case Reference (f)) to ~1600kg (average of current slider based reactor configurations) because more recent reactor models have decreased reactor size relative to the Reference (o) reactor model used for the ORNL shield optimization study (Reference (f)) and due to the increased energy conversion system and xenon tank attenuation assumptions discussed in the previous section. The ORNL optimizations were set up to get the uncertainty in the optimizations down to about 100kg.

The Table 5 case 1 optimization used only the basic shield materials. Each subsequent case added additional design detail to each case as specified. H₂O based shields with piping routed through the shield are significantly higher mass than similar LiH cases. Also, as additional design details are added to the LiH case, the difference in mass between

non-hydrogenous material sets and the LiH based shield goes down and non-hydrogenous shields become more competitive, particularly if the reduced development time/costs and reduced risk are considered. The need for internal structure for LiH (i.e. stainless steel "egg-crate") as used on SNAP, but not SP-100, has not been determined, but could add a small amount of additional mass to the LiH shield. Additional evaluations are required to define which design details are required, whether shields with pipes around the shield are mass competitive for hydrogenous shields (see below), and to compare the mass for different material sets if additional gamma attenuation is provided by energy conversion subsystem components and the xenon tank(s).

Case 6 was developed to try to assess shield mass differences if the pipes were to be routed around the outside of the shield, because the mass of a required shield cap behind where the pipes reenter the main shield shadow cone should be about the same for all material combinations. Based on cases 1 and 2, H₂O based shields are probably competitive with LiH based shields for piping routed around the shields. The lowest mass cap shield would be LiH. For a H₂O based shield, either LiH would be required for the cap shield, or a mass penalty would be incurred to use Be. LiH in a cap shield would be in a low dose region which would not be susceptible to irradiation induced swelling, but would make pipe and CDM cooling more difficult. Be in a cap shield could be used for dual benefit as a radiation shield and to help remove heat from the pipes and CDMs. For pipes routed around the shield, non-hydrogenous shields would be about 200-300kg (case 6 of Table 5) higher mass than hydrogenous shields. If there is sufficient space in the shield, routing pipes through the shield is better from a mass perspective than around the shield for non-hydrogenous shields. Since the mass of a cap shield still needs to be investigated, mass comparisons for routing pipes around versus through the shield cannot be made.

When comparing case 1 to case 2 within the same materials set, the mass did not change significantly for most of the material sets, which means that addition of thick front and back structural plates and incorporation of regions to place nuclear instruments did not significantly affect shield mass. This is because addition of the structural plates displaced approximately an equal amount of W mass. The H₂O shield was an exception and increased by about 140kg when the structure was added.

Table 4: ORNL optimized shield layers for the four material sets

Be/B ₄ C/W/LiH		W/ H ₂ O or Be/B ₄ C/W/H ₂ O		Be/B ₄ C/W		B ¹⁰ /W or Be/B ¹⁰ /W	
Material	Thickness (cm)	Material	Thickness (cm)	Material	Thickness (cm)	Material	Thickness (cm)
SS316	1.5	SS316	1.5	SS316	1.5	SS316	1.5
B ₄ C	1.0	B ₄ C	1.0	B ₄ C	1.0	¹⁰ B	1.0
3 layers [5 cm Be + 0.5 cm B ₄ C]	16.5	2 layers [5 cm Be + 0.5 cm B ₄ C]	11.0	9 layers [6.5 cm Be + 0.5 cm B ₄ C]	67.5	11 layers [5 cm Be + 0.5 cm B ₄ C]	60.5
W	1.75	SS316	1.0	1 layer [5 cm Be + 2.0 cm B ₄ C]	7.0	¹⁰ B	12.0
LiH	56.75	Water + ⁶ LiOH	73.5	SS316	2.0	SS316	2.0
SS316	2.5	SS316	2.25				
Total	80	Total	90.25	Total	79	Total	77

Table 5: ORNL Shield Mass (kg) Comparisons for Four Material Sets

(Δ LiH is the mass increase relative to the same LiH case)
(Pipes routed through shield for cases above 3)

Design features	Be/B ₄ C/W/LiH	W/ H ₂ O or Be/B ₄ C/W/H ₂ O		Be/B ₄ C/W		B ¹⁰ /W or Be/B ¹⁰ /W	
	Total	Total	Δ LiH	Total	Δ LiH	Total	Δ LiH
1. Only given shield materials *	2023	1921	-102	2442	+419	2317	+294
2. Add NI regions and SP-100 front & back structural plate to 1.	2048	2057	+9	2473	+425	2364	+316
3. Add piping to 2. **	2152	2312	+160	2436	+284	2347	+195
4. Add radial containment canning @ SP-100 thickness, as applicable to 3. ** +	2224	2373	+149	2436	+212	2347	+123
5. Add 2 nd layer of canning to meet single point failure criteria, as applicable to 4. ** +	2296	2434	+138	2436	+140	2347	+51
6. Approximation of pipes around the shield (Case 5-3+2) excluding cap shields	2192 +cap	2179 +cap	-13	2473 +cap	+281	2364 +cap	+172
* Scaled up 15% for change in reactor/shield size per Reference (f) ** Pipes routed through the shield + Fore/aft canning is assumed to be off set an equivalent mass of W or steel. Containment canning is not required for non-hydrogenous cases							

Preliminary Thermal Analyses

Preliminary thermal finite element analyses (FEA) were performed for an early concept representative of the space reactor radiation shield and are documented in Enclosure (2). The objective of the analyses was to evaluate four shield material configurations with reactor coolant piping routed through the shield. The shield configurations assessed were primarily composed of water (H₂O), boron, beryllium/B₄C, and lithium hydride (LiH). The analyses utilized preliminary shield geometry and radiation heating data. The thermal analyses included the radiation heating, conduction, and thermal radiation effects to determine the approximate temperature ranges of the shield configurations. These temperature operating ranges were compared to material temperature goals.

The temperature and heat flux results were used to assess potential thermal management issues related to temperature control of the Reactor Module. Temperature results were also

used to approximate the temperatures of Instrumentation and Control components located in or on the shield.

The Enclosure (2) results indicated that preliminary LiH, Be/B₄C, boron or water shield configuration temperatures would meet the temperature goals for these materials. The analyses showed that all four configurations were feasible with respect to meeting shield material temperature goals, but additional thermal management evaluations would be needed to show that hot coolant pipe temperatures could be controlled. The water option may have had limited design space due to competing requirements to meet shield water temperature and pressure goals, and to reject heat to cool pipe walls cool.

Structure

Shield structural design was not considered a limiting design area at this stage relative to material selection, shield configurations, and thermal design. Structural material candidates initially considered were stainless steel or a nickel alloy. Stainless steel was used in the SNAP and SP-100 shield designs. Nickel alloys were considered by NRPCT because it may be possible to minimize material development costs by using the same material that would be developed for the reactor vessel and piping. Other potential structural materials that could be investigated to reduce mass include titanium alloys, Be, AlBeMet (Brush Wellman aluminum beryllium compound), carbon or SiC composites. When comparing shield masses for different shield material combinations, the shield support structure was assumed to be the same, but any containment canning for LiH or water was added.

NRPCT started investigating different techniques for canning hydrogenous materials and not allowing for a single point failure. A leak in a single-layer can containing either LiH or H₂O would be a mission ending failure. Possible failure mechanisms include micrometeoroid punctures, corrosion, shield swelling (LiH), and over-pressurization (H₂O). Three canning configurations that avoid single point failure are shown in Figure 11. A version of the double walled canning (A), with rounded ends, would probably be required for a water shield due to the high pressure (~1 MPa/150 psi) needed to allow water temperatures up to 450K (350°F). Above 450K (350°F), water saturation pressures and, therefore, structural requirements increase rapidly. For LiH any of the methods are possible. Configuration "A" was envisioned as using a honeycomb structure between the two walls and prevents single point failure by providing a double barrier to hydrogen loss. Configurations "B" and "C" prevent single point failure by adding an extra separately canned section of LiH beyond that needed to meet radiation dose requirements so that if one can leaks, mission radiation dose requirements would still be met. However, the potential for detrimental interactions between possible leaking Li and H and other spaceship materials would need to be assessed and may limit the value of multiple single walled containment volumes.

Further investigation of the mass, manufacturability, reliability, and structural and thermal capabilities of these configurations is required, including compatibility with LiH formation methods (e.g. pressed or cast). The primary canning material candidate for LiH was stainless steel. For water, additional material possibilities include nickel and titanium alloys. Canning thicknesses for both LiH and water would be controlled by launch loads, containment pressure, or H retention.

Figure 11: Hydrogenous material canning configurations that avoid single point failure

Other Notional Shield Configurations

Figure 12 shows a notional shield configuration with concepts intended to improve the manufacturability of the shield and piping, assuming pipes passing through the shield. The shield has diverging flat faces rather than a diverging ellipse. The $12^{\circ}/6^{\circ}$ shield cone half-angle was maintained. The pipe routing uses regular toroidal bends and straight sections rather than following the continuously varying curvature of the diverging ellipse. Although Figure 12 shows the pipes exiting normal to the reactor axis, rotating them down toward the shield would improve pipe routing and the shield configuration. The pipe is routed through a "U" shaped channel in the outer edge of the shielding material to simplify shield manufacturing and shield/pipe assembly. The assembly sequence would be to build the bulk shield and pipes in parallel, insert the pipes into the "U" channels in the bulk shielding, then insert that assembly into a structural sheath or can. A similar process would be used for the neutron detectors. The Figure 12 configuration has the same mass as a similar conical ellipse.

Figure 12: Notional Shield Configuration for Improved Manufacturability

Another shield configuration that should be investigated (Figure 13) would orient the reactor axis parallel to the shield front face with the reflector (control drums) directly adjacent to the shield. The reactor longitudinal axis would be in the plane of the radiators (12° cone half angle direction). This configuration would have cold coolant enter the vessel at one end; flow through the core and out the other end of the vessel with bypass flow to cool the vessel within the core length. The point on the reflector or reactor for projecting the shield radii for the horizontal configuration is closer to the shield than in the normal longitudinal orientation, which limits the shield size. Elliptical heads on the reactor may be possible if the pipes enter at an angle and that would limit reactor and shield length. But turning the long reactor axis parallel to the shield and increasing the reflector thickness to use drums instead of sliders tends to increase the shield size. The net result is that the horizontal configuration has approximately the same shield areas as the normal longitudinal reactor-shield configuration. However, the reflector in the horizontal configuration directly attenuates radiation in the direction of the shield, whereas the reflector does not provide significant attenuation for the normal longitudinal arrangement. Therefore, the shield thickness for the horizontal reactor will be smaller and shield mass should be less. Because the reactor axis is parallel to the shield face and the reactor is immediately adjacent to the shield, the shield mass sensitivity to shield cone half angle and heat rejection radiator area is reduced. The CDM shield penetrations could pass through both ends of the shield. Routing pipes around the outside of the shield in the horizontal configuration would minimize conflicts with CDM penetrations. CDM linkage to the control drums has not yet been evaluated, but CDM linkage to drums forward of the reactor vessel (opposite the shield) would be a particular challenge, if drums are needed there. The aeroshell design for the horizontal configuration may be more difficult due to the proximity of the reactor and shield, and the change in orientation.

Figure 13: "Horizontal" Reactor/Shield Configuration with reactor axis parallel to the shield face

Methods Development

Radiation Transport Programs

Radiation transport program development consisted of NRPCT investigations into which radiation transport program(s) should be used for NRPCT calculations and ORNL development of a system of codes to be used with TORT (ORNL's three dimensional discrete ordinates transport program). Three dimensional results would be required to model penetrations, energy conversion subsystem components, etc. The two basic methods being investigated were MCNP (Monte Carlo Null Particle) and discrete ordinates programs. The status of preliminary NRPCT transport program investigations is documented in Reference (i). NRPCT had not selected a program or set of programs. Multiple programs or splicing between programs may have been required to provide all of the information and capability required to design the shield. Typically, discrete ordinates transport programs have trouble calculating the effect of large, low scattering regions like the voids between the reactor radiation shield and the payload. Reference (i) also discusses preliminary NRPCT investigation into charged particle transport codes. ORNL has a two dimensional set of programs based on the DORT discrete ordinates program which could help resolve the discrete ordinates problem of transport across a void and was in the process of converting these other programs to three dimensions(3D) for use with the 3D TORT discrete ordinates program. The status of the ORNL transport code development work is documented in Reference (v). MCNP was being used by both ORNL and NRPCT for most of the Prometheus radiation transport analyses to date.

Benchmark Testing

Benchmarking of shield design methods allows better understanding of method uncertainties which could then be used to minimize mass and mission risk by ensuring more accurate radiation predictions. Reference (l) provides the status of ORNL investigations of existing and proposed benchmark tests at the time of project termination. Reference (l) identifies that the experimental program should be designed to anticipate the need for measurement of the following quantities: cross-sections, gamma production rates, heating, and gas generation. Based on preliminary review, NRPCT considers that benchmark testing should be performed to obtain data to more directly quantify the accuracy of the shielding computational predictions. The measured quantities given above would not directly provide this. Accordingly, any future benchmark development should be structured to measure three responses: (1) total neutron flux > 1 MeV exiting the shield, (2) total silicon gamma dose exiting the shield and, (3) the temperature profile internal to the shield to confirm radiation heating rates. The benchmark testing should be structured to perform these measurements for bulk shield configurations with and without a representative piping penetration (to benchmark penetration streaming predictions).

Cross Section Development

Reference (n) provides the status of ORNL work to develop radiation cross sections for use in shielding programs. This analysis was still in progress at the time of project termination. Reference (n) indicated and NRPCT agrees that the next steps were to incorporate the NRPCT SPACE05A cross section set (Reference (r)) and to develop few group cross sections. Definition of cross section energy group structure would need to incorporate both bulk attenuation and scattering (e.g. piping penetrations, pipes around the shield, control drive shaft penetrations, etc.) models to ensure shielding programs provide accurate results for these limiting conditions.

Sensitivity and Uncertainty Analysis

Reference (m) provides the status of ORNL cross section sensitivity and uncertainty analyses intended to identify shielding materials which would be most sensitive to cross section uncertainties. As discussed under cross section development, the latest cross section set (SPACE05A) and few group cross sections would need to be incorporated in future work. Continuation of these analyses would need to incorporate the latest shield configurations including penetrations. Future work should become more focused on what data is needed to define the shield design method uncertainty factor and help define the benchmark tests.

SHIELD INTEGRATION

This section discusses reactor radiation shield interfaces with other Reactor Module and spaceship components, and with the ground test reactor.

Reactor Interface

The reactor and shield have many interfaces. Some of which are listed below:

- Reactor is the primary radiation source for neutron and gamma radiation, and the neutrons produce secondary radiation in structures and shielding.

- The shield is heated by the reactor neutron and gamma radiation, thermal radiation, and conduction through structure and control mechanisms.
- Shield size is projected from the forward most outer radius of the reactor or reflector, so reactor size and distance from the shield, directly affect shield size.
- The control type, assumption on stuck controllers (sliders), and control support structures affect shield size and mass.
- The length and location of fission gas plenums may affect shield size.
- Attenuation provided by the reactor internals, like fission gas plenums at the shield end of the core and shutdown control rods, if applicable, could reduce shield thickness. But fission gasses that move into a fission gas plenum at the shield end would become a more significant source, which would partially offset the shield benefit.
- The core type (e.g. block, or open lattice) and materials affect the amount of radiation leakage toward the shield.
- In slider controlled reactors, the power can move axially within the core over life, which can cause about a factor of two change in the radiation flux and heating in the shield.
- The slider stroke length affects shielding size by affecting either the distance from the core to shield (sliders moving out aft) or the location of the forward end of the slider.
- The location of the power peak in the core is further from the shield for sliders that slide out forward vs. aft.
- The reactor is supported by the shield.
- The neutron detectors could be located in the shield or would have cables that must pass through the shield.
- Reactor control shafts must pass freely through the shield and motors are mounted on the back of the shield.
- Neutron radiation streams along control shaft gaps and the type of motion, translation or rotation, of the shaft may affect the ability to reduce radiation streaming along the shaft.
- Reactor gamma radiation may drive electrons off of the surface of the reactor/reflector causing spaceship electrical charging and the electrons may go around and come back in behind the shield.
- Sliders that move out forward allow higher neutron levels out to the edge of the shield, compared to sliders that slide aft, and drums may be somewhere in between. The amount of radiation reaching the outer edge of the shield will affect the amount of neutron scatter from piping around the shield, thus affecting the mass of cap shields over the pipes.
- The reactor materials affect the gamma source strength and neutron spectrum entering the shield.
- Any parameter that affects reactor size also affects shield size.
- Piping exiting the reactor must penetrate or go along the outside of the shield.
- The reactor size, control shaft radial location, and control shaft spacing affect how the piping can enter and be routed through the shield, and how far the reactor must be from the shield to allow the pipe to bend, which affects shield size/mass.
- The potential for fission product release from the reactor may affect both the flight shield and ground test reactor shield design.
- The configuration of the ground test reactor shielding must be designed to minimize neutron reflection effects that could impact reactor performance prototypically in addition to meeting facility shield design requirements.

Plant and Energy Conversion System Interface

The shield mass can be significantly affected by the system architecture (the number of components) and arrangement of the energy conversion system (Reference (s)). The energy

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conversion system components, especially the heat exchangers, can provide significant gamma radiation attenuation for the payload, which would allow reduction of the reactor gamma radiation shield thickness. As the mass and number of energy conversion system components increases, the shield mass will decrease, although not by an equal amount. Gaps between components and plating parallel to the spaceship's axis are not as effective as a continuous plate normal to the spaceship's axis. Also, because the shadow cone is larger at the location of the energy conversion system components than at the shield, it takes more mass to provide the same attenuation. A small amount of neutron attenuation is provided by the energy conversion system components. However, if the heat rejection heat exchanger contains a significant amount of water, the amount of neutron attenuation would increase. Note if the reactor shield is thinned because more of the radiation attenuation is provided by the energy conversion system components, the radiation dose to sensitive components (e.g. motors, cabling, sensors, lubricants, etc.) immediately behind the shield is increased.

Shield mass is only one of many factors to be considered when arranging the energy conversion system components. The main shielding constraint on the energy conversion system is the need to keep all components within the shield shadow cone. The following techniques are provided as additional guidelines to minimize shield mass when arranging energy conversion system components. Try to keep the components, especially the heat exchangers, as close to the shield as possible. Try to arrange components across the spaceship's axis rather than parallel to it. Try to cover as much of the shadow cone as possible, but if that is not possible, mass near the spaceship's axis will be more effective than mass near the perimeter of the shield shadow cone. Minimize gaps parallel to the spaceship's axis between components.

The effect on shielding of various piping features was discussed in the piping penetrations section. However, decisions that benefit the shield design would also affect plant performance and arrangements. The performance of the gas Brayton plant is extremely sensitive to system pressure drops (Reference (s)). Small diameter pipes, which would favor shield mass reductions, would increase coolant system pressure drops. The bend radius and total bend angle within the shield would affect coolant pressure drop. The need for neutron radiation cap shields behind where pipes come around the shield requires at least two 90° bends in each pipe, which affects coolant pressure drop (Reference (s)). Impacts on plant performance would result in the need for a higher thermal reactor power rating, which could more significantly impact overall shield size and mass.

Spaceship Interface

Investigation of spaceship and shielding interfaces were about to commence when the program was terminated. Interactions with JPL and Northrop Grumman were needed to identify areas which affected both the reactor shield design and spaceship design and to identify trade studies that would minimize overall spaceship mass. Examples of areas identified to be investigated or discussed are listed below.

- Payload Shielding vs. Reactor Shielding
 - Dose split
 - Space radiation shielding materials
- Boom length, straightness and oscillations
- Shield shape
 - Conical, elliptical, asymmetric elliptical
 - Cone angle
- Payload dose requirements, shapes, locations, variations in sensitivity

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- Radiation sensitivity of non-payload components (e.g. cables, lubricants, motors, radiators) and locations/arrangements
- Space radiation effects on SNPP components
- Xenon tank location/shape
- Radiation Scatter
 - High gain antenna and whip antenna locations/orientations
 - Radiator extent and materials
- Reactor shield gamma vs. neutron shield shape
- Piping penetration design, location, and orientation effects on the spaceship.
- Launch configuration, load reaction, etc. interfaces with the shield
- Spaceship electrical charging

Micrometeoroid Protection Interface

LiH based shield concepts require external thermal insulation to maintain the required temperature range, and this insulation was assumed to also provide micrometeoroid protection for the shield. The non-hydrogenous shield concepts could accommodate micrometeoroid shielding, but it may not be necessary because significant degradation of shielding properties is unlikely to occur. Elimination of micrometeoroid shielding would allow the shield to operate at a lower temperature, which benefits hot leg coolant pipe, nuclear instrument, and CDM motor thermal design. For water shield concepts with the reactor coolant pipes routed through the shield, the outer surface of the water tank would be surrounded by non-hydrogenous shielding and may also not require micrometeoroid protection. For water shield concepts with the pipes routed outside the shield, the water tank would be at the outer surface of the shield and may require micrometeoroid protection unless the tank configuration can provide that function (e.g. configuration "A" of Figure 11). For water shield concepts, it is desirable to maximize the capability to radiate heat to space to limit the water temperature and micrometeoroid protection would tend to act as insulation. At water temperatures above 450K (350°F), water saturation pressure and shield design pressure increase rapidly. Increased pressure requirements could quickly become structurally limiting and increase structural mass requirements. Alternately, active cooling could be required, but increased heat rejection system capacity would also increase mass and complexity.

Aeroshell Interface

An aeroshell was being considered to prevent reactor disintegration and slow the reactor velocity as it re-enters the atmosphere during a launch casualty (Reference (w)). The aeroshell interface with the shield was not well defined before program termination. If the aeroshell back face was located between the reactor and the shield to ensure shield separation from the reactor during a potential re-entry casualty, the shield design still needed to incorporate this plate into the shield design models. The shield designs included Be at the front of the shield which could be incorporated as part of the aeroshell, if an integrated aeroshell/shield were pursued. Other functions of Be at the front of the shield are removal/distribution of high heat loads, neutron shielding material capable of withstanding high neutron fluence, and thermal neutron flux production for nuclear instruments.

Ground Test Reactor (GTR)

In addition to the benchmark test discussed above, a prototypical flight shield was planned to be included in the GTR facility being evaluated (Reference (u)). Shielding related measurements in the GTR would include shield temperature, neutron fluence and gamma dose, but may also include shield pressure and dimensional measurements, particularly with LiH or H₂O based

shields. The GTR would confirm the final flight shield design for thermal and radiation performance, and the effects of the shield on the reactor control system.

Some of the more challenging design issues for shielding include minimizing neutron reflection to the core, design of a collar shield around the flight shield to be prototypical of the cooling provided in space and minimize scatter around the shield, and activation of reactor plant components with high levels of tantalum and cobalt, if those materials are selected for use. The preliminary GTR radial facility shield thickness was estimated to be about three meters of concrete. Care must be taken in selecting the vacuum vessel cooling gas and the related system. Selection of Ar or N, for instance, could result in high Ar^{41} radiation levels outside the shield or high C^{14} in the system (which is not measurable without sampling) because of high neutron fluxes at the vacuum vessel wall.

FUTURE CONSIDERATIONS

This section discusses the next steps in the reactor radiation shield design, other possible shield configurations that could be considered, extensibility of the shield design to potential surface missions

Future Directions

The next step in the shield design would be to investigate the mass effects of the different shield material sets for pipes routed around the shield. Evaluations of shield mass effects for direction of slider movement, stuck slider, drum control, and shutdown rod location are needed. Evaluations of shield mass effects from energy conversion components and the xenon tank should be performed. Evaluations are needed to determine which design features are required to mitigate thermal, structural, and single point failure criteria (see section on Optimized Shield Masses) concerns for H_2O and LiH based shields. Based on these results and those in References (e) and (f), plus any additional material data that becomes available, a final material selection should be made. The projected shield need date and available resources play a significant role in the material selection. A very short term need date makes it more difficult to select a material that requires a lot of development and has many design concerns. LiH falls into this category, but is a very valuable material to have available. To better understand LiH irradiation performance and mitigate the risks of LiH development for a project with an aggressive schedule like JIMO, an advanced development effort for LiH could be considered to support future space reactor projects. If either H_2O , LiH, or B^{10} based shield concepts were selected as the primary material set, then a Be/ B_4C shield design could be carried as a backup.

Evaluation of the mass efficiency and reactor design practicability of the horizontal reactor/shield concept should be pursued.

Extensibility

The primary shield material list identified for Prometheus is expected to be feasible for a manned surface mission. Substantial shielding would be required to protect people from reactor radiation if they are to be located within a reasonable distance to benefit from the electrical power produced. The need for very low density neutron shielding material would be required if the mission design is to launch the entire shield from earth. A Lunar/Mars reactor shield would be much heavier than the Prometheus shield and the lightest neutron materials would be required for the bulk of that shield. LiH would be a prime candidate for this type of shield because it can take the higher temperatures and most of the shield would be low dose and not susceptible to irradiation swelling concerns. Water could be used, but it may require a higher

design temperature/pressure due to higher reject heat temperatures on the moon compared to space, which drives up structural mass and possible heat rejection system mass. Water could be carried in a separate launch and pumped into a tank around the reactor placed during an earlier launch.

However, some Lunar mission concepts provide a low shield mass option, such as use of regolith (and water if available on Mars) as a shield, or placement of the reactor in a hole. For these concepts, the need for a low mass shield material in the immediate vicinity of the reactor is not as critical. It is expected that the Prometheus type shadow shield would still be required to protect the reactor control drive mechanisms, reactor plant sensors, and the energy conversion equipment. The regolith would be less effective at attenuating neutrons than likely shadow shield materials, which would allow the regolith adjacent to the shadow shield to scatter neutrons around the shadow shield. Therefore, the shadow shield concept would need to be revised to reduce neutron scattering around the shadow shield.

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Preliminary Reactor Shield Thermal Analyses

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Preliminary Reactor Shield Thermal Analyses

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Introduction and Conclusion

Preliminary thermal finite element analyses (FEA) have been performed of representative Prometheus Space Reactor radiation shields. The objective of the analyses was to evaluate the configuration and identify potential concerns related to the thermal performance of four shields made from different materials: Water (H_2O), boron (B), beryllium/boron carbide (Be/B_4C), and lithium hydride (LiH). Figures 1 through 4 show the four shield material configurations that were evaluated. The analyses utilized preliminary shield geometry and radiation heating data that were ORNL documented in Reference (a). Table 6 describes the material and geometry and Table 7 describes the thermal loads and boundary conditions of the four shield configurations.

The thermal analyses include the radiation heating, conduction, and thermal radiation effects to determine the approximate temperature ranges of the shield. These calculated temperature operating ranges have been compared to material temperature goals as shown in Table 1.

Table 1: Shield Material Temperature Goals

Shield Configuration	Temperature Goals	Reference(s)
LiH	600K – 800K range	References (k) and (o)
Be/ B_4C	823K Maximum for Be structure 1500K Maximum for B_4C	References (d) and (f)
Boron	1174K Maximum	Reference (m)
Water	454K Maximum Design Temperature; 1.0 MPa Design Pressure.	

Be: Reference (d) discusses the 823K limit for beryllium as a structural material, but higher beryllium temperatures may be feasible for non-structural uses.

B_4C : Reference (f) discusses the 1500K limit for B_4C .

B: Boron is limited to half of the melting temperature of 2348K which is obtained from reference (m).

H_2O : For the water configuration a maximum design temperature of 454K is set. This corresponds to the saturation temperature at a pressure of 1.0MPa. This temperature is selected as the initial design point for the pre-conceptual design based on a desire to keep the design pressure low (1.0MPa) to limit water tank structural mass. However, higher water temperatures and pressures could be considered, especially if water tank wall thicknesses are determined by other design factors like launch loads.

LiH: As discussed in References (k) and (o), at this time it is not known if the LiH lower temperature goal of 600K is applicable to mixed neutron and gamma irradiated LiH. As discussed in Reference (l) the temperature design space of LiH is directly dependent on the fluence level that the material is exposed to and the purity of the material.

The results indicate that preliminary LiH, Be/ B_4C , boron or water shield configuration temperatures are within the goals listed in Table 1. However, the H_2O option may have a more limited design space due to temperature and pressure limits. More detailed analysis would be

necessary to verify that the H₂O shield with the restrictions of the micrometeoroid protection and hot leg piping heat load rejection is feasible. Also, additional thermal management evaluation which includes the entire Reactor Module is needed to ensure that all component temperatures can be met, especially reactor coolant hot leg piping.

Analysis Results

Table 2, Table 3, Table 4, and Table 5 show the results of the thermal FEA for the Be/B₄C, boron, LiH, and H₂O shield configurations respectively. Net surface heat fluxes are shown for each configuration and include conduction and radiation heat flux. The net heat flux values indicate the direction of the heat flux as either IN or OUT of the shield. The front surface of the shield refers to the flat surface facing the reactor and the aft surface is the flat surface facing the plant. The shield side surfaces refer to the conical shaped surface facing space. The hot and cold pipe surfaces refer to the heat transfer through the penetration surface(s) and are the total for the hot and cold pipe(s) respectively.

Approximately 2.1kW of heat needs to be rejected per meter of hot leg piping in order to reduce the temperature of the piping outer wall to 900K (Reference (n)). The shield thickness used in this analysis is 0.595m. Based on this shield thickness and running the pipes straight through the shield, the required heat loss for the hot piping routed straight through the shield is 1.3kW and for piping routed around the shield could be 2 – 4 kW. The results from the analyses documented here show that the heat from the hot leg piping varies for the different cases, but for all cases is less than the required 1.3 kW. From this, it can be concluded that an additional heat rejection path would most likely be required in order to maintain the section of hot leg piping within the shield less than 900K.

There are a number of uncertainties and deviations from updated shield designs that are not accounted for in this analysis. The known deviations and uncertainties are described in Table 14. These individual effects are difficult to quantify individually. Reference (p) is an analysis which utilizes updated shield geometry parameters and more accurate hot leg piping heat losses. These analyses show that the shield temperature increases moderately on average relative to the results documented herein, but the overall conclusions remain unchanged.

The internal heat generation results listed under the body heat flux category correspond to the total heat generated as a direct result of the neutron and gamma radiation heating within the shield.

The *Shield Temp* ranges listed in the results tables correspond to the range of temperature in the shield. The *shield aft surface temperature* is given to provide information on the aft surface radiation to the rest of the reactor control and plant components behind the shield.

In general the heat flux from the shield to space, which are listed in the results tables, increases with higher shield temperatures.

Table 2: Analysis Results: Be/B₄C Shield

Case Name	Shield Configuration Name	Surface Heat Fluxes & Temperatures	
mod_8	Layered Be/B ₄ C	Surface Heat Flux	Heat Flux (Q) kW
		IN From Uninsulated Cold Pipe:	0.4
		IN From Hot Pipe:	0.4
		Net IN shield through Front Shield Surface:	0.5
		Net OUT shield through Aft Shield Surface:	0.7
		Net OUT Shield Side Surfaces:	2.5
		Body Heat Flux	
mod_8b	Layered Be/B ₄ C (Insulated) effective emissivity=0.02	Total Internal Heat Generation	2
		Temperatures:	
		Shield Temp: 430-465K	
		Shield Aft surface temp: 430 - 440K

008: mod_8.sdb

Step: 39 of 100
Increment

1: Step Time 100.0

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*Note: See Table 7 for description of loads and boundary conditions

Table 3: Analysis Results: Boron Shield

Case Name	Shield Configuration Name	Surface Heat Fluxes & Temperatures	
mod_10	Boron	Surface Heat Flux	Heat Flux (Q) kW
		IN From Cold Pipe:	0.6
		IN From Hot Pipe:	0.6
		Net IN shield through Front Shield Surface:	0.2
		Net OUT shield through Aft Shield Surface:	0.5
		Net OUT Shield Side Surfaces:	2.9
		Body Heat Flux	
		Total Internal Heat Generation	2
		Temperatures:	
		Shield Temp: 410K - 495K	
		Shield Aft surface temp: 410 - 435K	

008: mod10.odb ABAQUS/STANDARD Version 6.3-6 Wed Oct 26 12:49:11

Step: 38 of 1
Increment: 1 Step Time = 100.0
Primary Var: Wtill
Deformed Var: not set Deformation Scale Factor: not set

008: mod10.odb ABAQUS/STANDARD Version 6.3-6 Wed Oct 26 12:49:11

Step: 38 of 1
Increment: 1 Step Time = 100.0
Primary Var: Wtill
Deformed Var: not set Deformation Scale Factor: not set

*Note: See Table 7 for description of loads and boundary conditions

Table 4: Analysis Results: LiH Shield

Case Name	Shield Configuration Name	Surface Heat Fluxes & Temperatures	
mod_13	LiH (Explicitly Modeled Insulation: 1cm thick low conductive layer)	Surface Heat Flux	Heat Flux (Q) kW
		IN From Cold Pipe:	0.4
		IN From Hot Pipe:	0.9
		Net OUT shield through Front Shield Surface:	2
		Net OUT shield through Aft Shield Surface:	0.4
		Net OUT Shield Side Surfaces:	0.9
		Body Heat Flux	
mod_13b	LiH (No IHG)	Total Internal Heat Generation	2
		Temperatures:	
		Shield Temp: 650K - 690K / Shield Aft Insulation exterior surface temp: 480K - 535K	
mod_13c	LiH (Insulation modeled with effective emissivity=0.02)	Surface Heat Flux	Heat Flux (Q) kW
		IN From Cold Pipe:	0.4
		IN From Hot Pipe:	0.9
		Net OUT shield through Front Shield Surface:	2.3
		Net OUT shield through Aft Shield Surface:	0.2
		Net OUT Shield Side Surfaces:	0.7
		Body Heat Flux	
mod_13c	LiH (Insulation modeled with effective emissivity=0.02)	Total Internal Heat Generation	2
		Temperatures:	
		Shield Temp: 675K - 695K / Shield Aft Surface Temperature: 675K-695K

*Note: See Table 7 for description of loads and boundary conditions

Table 5: Analysis Results: H₂O Shield

Case Name	Shield Configuration Name	Surface Heat Fluxes & Temperatures	
mod_14	Water	Surface Heat Flux	Heat Flux (Q) kW
		IN From Cold Pipe:	2.1
		IN From Hot Pipe:	2.2
		Net OUT shield through Front Shield Surface:	0.4
		Net OUT shield through Aft Shield Surface:	0.8
		Net OUT Shield Side Surfaces:	5.1
		Body Heat Flux	
		Total Internal Heat Generation	2
		Temperatures:	
		Shield Temp: 420K - 454K (H ₂ O region) 560K – 590K (Be/B ₄ C region)	
		Shield Aft surface temp: 420- 454K (H ₂ O region); 560K-570K (Be region)	

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*Note: See Table 7 for description of loads and boundary conditions

Analysis Description

Configurations Analyzed

Four primary shield material configurations have been analyzed. Figure 1, Figure 2, Figure 3, and Figure 4 show the Be/B₄C, boron, LiH, and H₂O configurations respectively.

Figure 1: Be/B₄C Shield

Figure 2: Boron Shield

Figure 3: LiH Shield

Figure 4: Water Shield

Table 6 lists the key component dimensions and materials used in the analyses. The number of pipes used for the LiH and H₂O configurations is greater than the Be/B₄C and Boron configurations, because it was believed at the time the analysis was started to be necessary to have more pipes to maintain a more uniform temperature distribution in the LiH and water configurations. The pipe surface area of the two large pipes and the four small pipes is approximately the same. The *mod_8b*, *mod_13b*, and *mod_13c* cases are sensitivity studies on boundary conditions. The material and geometry configurations are the same as their respective base cases.

Table 6: Analysis Case Material and Geometric Configurations

Case Name	Shield Configuration Name	Component Materials & Dimensions
<i>mod_8</i> (Figure 1)	Layered Be/B4C	Shield: Be/B4C layers (12 layers; 4.5cm each) ; Total 59.5cm thickness Front face diameter: 1.148m; Aft Face diameter: 1.40m
		Piping: Haynes 230 18cm OD (1Hot 1Cold)
		Reactor & Spacecraft support structure: Haynes230 (6) 1cm OD cylinders to plant; (6) 1cm OD cylinders to Reactor
		Reactor: Haynes 230; 40cm diameter; 2cm thick vessel
<i>mod_8b</i>	Layered Be/B4C (Insulated)	Same As <i>mod_8</i>
<i>mod_10</i> (Figure 2)	Boron	Shield: All Boron; Total 59.5cm thickness Front face diameter: 1.148m; Aft Face diameter: 1.40m
		Piping: Haynes 230 18cm OD (1Hot 1Cold)
		Reactor & Spacecraft support structure: Haynes230 (6) 1cm OD cylinders to plant; (6) 1cm OD cylinders to Reactor
		Reactor: Haynes 230; 40cm OD; 2cm thick vessel
<i>mod_13</i> (Figure 3)	LiH	Shield: 5cm Be/B4C/Be/B4C/Be layers; 34.5cm LiH Total 59.5cm thickness Front face diameter: 1.148m; Aft Face diameter: 1.40m Be Cylindrical Ring outside of layers
		Piping: Haynes 230 8cm OD (2Hot 2Cold)
		Reactor & Spacecraft support structure: Haynes230 (6) 1cm OD cylinders to plant; (6) 1cm OD cylinders to Reactor
		Reactor: Haynes 230; 50cm OD; .58cm thick vessel
<i>mod_13b</i>	LiH (No IHG)	Same as <i>mod_13</i>
<i>mod_13c</i>	LiH (effective emissivity)	Same as <i>mod_13</i>
<i>mod_14</i> (Figure 4)	Water	Shield: 5cm Be/B4C/Be/B4C/Be layers; 34.5 Insulated Water Can Be Cylindrical Ring outside of layers. Total 59.5cm thickness Front face diameter: 1.148m; Aft Face diameter: 1.40m
		Piping: Haynes 230 8cm OD (4Hot 4Cold)
		Reactor & Spacecraft support structure: Haynes230 (6) 1cm OD cylinders to plant; (6) 1cm OD cylinders to Reactor
		Reactor: Haynes 230; 50cm OD; .58cm thick vessel

Boundary Conditions

Table 7 lists the boundary conditions used for all of the analyses. Only the modified boundary conditions for the *mod_8b*, *mod_13b*, and *mod_13c* are listed for those cases.

Table 7: Analysis Case Loads and Boundary Conditions

Case Name	Shield Configuration Name	Loads & Boundary Conditions
<i>mod_8</i> (Figure 1)	Layered Be/B4C	<p>Aft face of Shield: Radiation to Ambient @ Plant Temperature of 400K ($\epsilon=0.75$)</p> <p>Sides Surface of Shield: Radiation to Ambient @ 4K ($\epsilon=0.75$ B4C; $\epsilon=0.35$ Be)</p> <p>Fwd face of Shield (inner circle): Cavity Radiation with Rx and ambient (500K to account for anticipated cover over reactor)</p> <p>$\epsilon=0.35$ Be; $\epsilon=0.1424$ H230. Inner circle is located radially inward of the pipes.</p> <p>Fwd face of Shield (Outer Ring): Radiation to ambient (4K) $\epsilon=0.35$ Be. The Outer Ring is the region of the fwd face not included in the inner circle.</p> <p>Reactor Inner surface Temperature: 891K (885K Outer Surface)</p> <p>Conduction Path to Plant: End at 400K.</p> <p>Pipe Temperatures: Cold Pipe 890K; Hot Pipe 900K</p> <p>Radiation From Pipes: ($\epsilon=.1424$) to Shield Penetrations ($\epsilon=0.75$ B4C; $\epsilon=0.35$ Be)</p> <p>Insulation: No Insulation on Pipes; No insulation on Exterior Shield Surface</p> <p>Internal Heat Generation: 10 times ORNL base case IHG function</p>
<i>mod_8b</i>	Layered Be/B4C (Insulated)	<p>Same As <i>mod_8</i> EXCEPT:</p> <p>Insulation: Insulation on Cold Pipe ($\epsilon=0.02$);</p> <p>Insulation on Exterior Shield Surface ($\epsilon=0.02$)</p>
<i>mod_10</i> (Figure 2)	Boron	<p>Aft face of Shield: Radiation to Ambient @ Plant Temperature of 400K ($\epsilon=0.75$)</p> <p>Sides Surface of Shield: Radiation to Ambient @ 4K ($\epsilon=0.75$)</p> <p>Fwd face of Shield (inner circle): Cavity Radiation with Rx and ambient (500K to account for anticipated cover over reactor)</p> <p>$\epsilon=0.75$ B; $\epsilon=0.1424$ H230. Inner circle is located radially inward of the pipes.</p> <p>Fwd face of Shield (Outer Ring): Radiation to ambient (4K) $\epsilon=0.75$ B. The Outer Ring is the region of the fwd face not included in the inner circle.</p> <p>Reactor Inner surface Temperature: 891K (885K Outer Surface)</p> <p>Conduction Path to Plant: End at 400K.</p> <p>Pipe Temperatures: Cold Pipe 890K; Hot Pipe 900K</p> <p>Radiation From Pipes: ($\epsilon=.1424$) to Shield Penetrations ($\epsilon=0.75$)</p> <p>Insulation: No Insulation on Pipes; No insulation on Exterior Shield Surface</p> <p>Internal Heat Generation: 10 times ORNL base case IHG function</p>
<i>mod_13</i> (Figure 3))	LiH	<p>Aft face of shield: Radiation to Ambient @ Plant Temperature of 400K ($\epsilon=0.11$)</p> <p>Sides Surface of Shield: Radiation to Ambient @ 4K ($\epsilon=0.35$)</p> <p>Fwd face of Shield (inner circle): Cavity Radiation with Rx and ambient (500K to account for anticipated cover over reactor)</p> <p>$\epsilon=0.35$ Be; $\epsilon=0.1424$ H230. Inner circle is located radially inward of the pipes.</p> <p>Fwd face of Shield (Outer Ring): Radiation to ambient (4K) $\epsilon=0.35$ Be. The Outer Ring is the region of the fwd face not included in the inner circle.</p> <p>Reactor Inner surface Temperature: 891K (890K Outer Surface)</p> <p>Conduction Path to Plant: End at 400K.</p> <p>Pipe Temperatures: Cold Pipe 890K; Hot Pipe 900K</p> <p>Radiation From Pipes: ($\epsilon=.1424$) to Shield Penetrations ($\epsilon=0.35$)</p> <p>Insulation: Insulation on Cold Pipes; None on Hot Pipes; Insulation on Exterior Shield Surface (Modeled explicitly with 1cm thickness and a low conductivity of insulation)</p> <p>Internal Heat Generation: 10 times ORNL base case IHG function</p>
<i>mod_13b</i>	LiH (No IHG)	Same as <i>mod_13</i> , EXCEPT: Internal Heat Generation = Zero
<i>mod_13c</i>	LiH (effective emissivity)	Same as <i>mod_13</i> , EXCEPT: Insulation on Exterior Shield Surface: Modeled as effective emissivity: $\epsilon=0.02$ and conductivity equals that of Be
<i>mod_14</i> (Figure 4)	Water	<p>Aft face of shield: Radiation to Ambient @ Plant Temperature of 400K ($\epsilon=0.35$ Be; $\epsilon=0.35$ SS Can)</p> <p>Sides Surface of Shield: Radiation to Ambient @ 4K ($\epsilon=0.35$)</p> <p>Fwd face of Shield (inner circle): Cavity Radiation with Rx and ambient (500K to account for anticipated cover over reactor)</p> <p>$\epsilon=0.35$ Be; $\epsilon=0.1424$ H230. Inner circle is located radially inward of the pipes.</p> <p>Fwd face of Shield (Outer Ring): Radiation to ambient (4K) $\epsilon=0.35$ Be. The Outer Ring is the region of the fwd face not included in the inner circle.</p> <p>Reactor Inner surface Temperature: 891K (890K Outer Surface)</p> <p>Conduction Path to Plant: End at 400K.</p> <p>Pipe Temperatures: Cold Pipes 890K; Hot Pipes 900K</p> <p>Radiation From Pipes: ($\epsilon=.1424$) to Shield Penetrations ($\epsilon=0.35$)</p> <p>Insulation: No Insulation on Cold Pipes.</p> <p>No Insulation on Exterior Shield Surface.</p> <p>Internal Heat Generation: 10 times ORNL base case IHG function</p>

Material Properties

Thermal conductivity material properties are required to assess the conduction aspect of the steady state thermal analyses and emissivity properties are required for the surfaces which are modeled with thermal radiation.

Table 8 shows the B₄C thermal properties used in the analysis. The thermal conductivity properties are obtained from Reference (f). The thermal emissivity value is taken from Reference (e).

Table 8: Boron Carbide Thermal Conductivity and Emissivity

Temp(K)	k(W/m-K)	Emissivity
293	28.412	0.75
300	28.210	
350	26.808	
400	25.480	
450	24.228	
500	23.050	
550	21.948	
600	20.920	
650	19.968	
700	19.090	
750	18.288	
800	17.560	
850	16.908	
900	16.330	
950	15.828	
1000	15.400	
1050	15.048	
1100	14.770	
1150	14.568	
1200	14.440	
1248	14.388	

Table 9 shows the boron thermal properties used in the analysis. The thermal conductivity properties are obtained from Reference (h). The thermal emissivity value is taken from Reference (e).

Table 9: Boron Thermal Conductivity and Emissivity

Temp(K)	k(W/m-K)	Emissivity
100.6	159.5	0.75
116.2	130.4	
126.9	109.5	
140.9	84	
170	71.5	
291	29	
470	21	
630	17	
1473	17	

Table 10 shows the LiH thermal properties used in the analysis. The thermal conductivity properties are obtained from Reference (d).

Table 10: LiH Thermal Conductivity

Temp(K)	k(W/m-K)
200	10.5
300	7.9
400	6.5
500	5.5
600	4.9
700	4.4
800	4.0
900	3.7
1000	3.4
1100	3.2
1200	3.0

Table 11 shows the beryllium thermal properties used in the analysis. The thermal conductivity properties are obtained from Reference (d). The thermal emissivity value is taken from Reference (a).

Table 11: Be Thermal Conductivity and Emissivity

Temp(K)	k(W/m-K)	Emissivity
293	184.5	0.35
400	159.5	
500	140.6	
600	125.4	
700	113.3	
800	103.7	
900	95.9	
1000	89.5	
1100	83.8	
1200	78.1	
1300	71.9	
1400	64.5	
1500	55.4	

Table 12 shows the Haynes 230 Nickel Superalloy thermal properties used in the analysis. The properties are obtained from Reference (g).

Table 12: Haynes 230 Thermal Conductivity and Emissivity

Temp(K)	k(W/m-K)	Emissivity
293	8.9	0.14
373	10.4	
473	12.4	
573	14.4	
673	16.4	
773	18.4	
873	20.4	
973	22.4	
1073	24.4	
1173	26.4	
1273	28.4	

Multi-foil insulation (MFI) is located on the outside surface of the LiH shield configuration. The properties for the MFI are obtained using a correlation from Reference (i) located in the OSTI database. The correlation equation used for the Molybdenum MFI is:

$$\frac{Q}{A} = \frac{1.06 \times 10^{-12} (T_{HOT}^4 - T_{COLD}^4)}{0.778N + 0.0111N^2} \quad [W/cm^2]$$

$$\text{Solving for } k: \quad k = \frac{Q \Delta x}{A \Delta T} = \frac{Q(N * t)}{A(T_{HOT} - T_{COLD})} \quad [W/cm-K]$$

$$\text{Substituting in values:} \quad k = \frac{1.06 \times 10^{-12} (T_{HOT}^4 - T_{COLD}^4)(N * t)}{(0.778N + 0.0111N^2)(T_{HOT} - T_{COLD})} \times 100 \quad [W/m-K]$$

Where:

- T_{HOT} = Hot side Foil Temperature (K)
- T_{COLD} = Cold side Foil Temperature (500K)
- N = Number of foils (35)
- t = Foil spacing (0.0127 cm)

Table 13 shows the resulting temperature dependant thermal properties for the MFI insulation. The N variable was iterated until a reasonable temperature range was obtained in the shield. T_{COLD} was set to match the approximate outer insulation surface temperature of 500K. The thermal conductivity of the insulation on the cold piping in the LiH shield configuration is 0.25 W/m-K as taken from reference (j).

Table 13: MFI Insulation Thermal Properties

$T_{Hot}(K)$	$k(W/m-K)$	Emissivity
100	0.00032	0.11
150	0.00037	
200	0.00042	
250	0.00049	
300	0.00056	
400	0.00077	
499.9	0.001	
600	0.0014	
700	0.0018	
800	0.0024	
900	0.0031	
1000	0.0039	
1100	0.0049	
1200	0.006	

Uncertainties

The shield thermal analyses were based on preliminary shield parameters, material properties, and material temperature goals. There are a number of differences from the analysis to the current pre-conceptual design concepts. These differences are shown in Table 14 with a qualitative assessment of the differences from the model results. None of the differences are expected to change the conclusion that all of the material configurations are feasible with respect to material temperature goals. One of the geometrical differences is related to the reactor structural support to the shield. It is expected that the shield would be slightly hotter due to the increased conduction area of the support skirt compared to this analysis which uses cylindrical rod supports.

The internal heat generation (IHG) function used in the FEA to simulate the neutron and gamma heating was assumed to be ten times the function used in reference (a). This factor of ten was applied to account for the reactor placement being closer to the shield than in the reference (a) analysis. More accurate results became available in reference (k) for the four shield material configurations, and the radiation heating results are close to this assumption. Figure 5 compares the IHG function used in ABAQUS to the four heating functions for each of the four shield configurations in reference (k). The figures in the table show that the assumed function is an approximate representation of the updated total radiation heating functions. Thicker shields than what was modeled are not expected to have significantly greater IHG since most of the radiation heating will occur near the reactor and the energy deposition rate at the aft of the shield is significantly less than at the front.

The shield sizes listed under the pre-conceptual design column are based on reference (k) designs. The actual shield design would most likely be slightly smaller than the reference (k) dimensions because the designs in this reference do not account for the shielding benefits of the plant components and reductions in the reactor size.

The emissivity values for the materials modeled were selected to be reasonably close to the temperatures of the shield. There are a number of uncertainties related to the thermal emissivity of external surfaces modeled in this analysis. As discussed in reference (p), the emissivity of surfaces degrades over time in space. The emissivity of surfaces is also dependent on the surface temperature. These effects were not modeled in this analysis. In addition to the radiation heat transfer from the piping, there may be phenomenon associated

with modeling radiation in cavities that aren't modeled with a straightforward emissivity and cavity relationship as captured in the ABAQUS analysis. More in-depth studies of this heat transfer would be necessary to validate the modeling methods.

The MFI used in this analysis would likely not be the specific insulation used on the shield. Studies would be necessary to better understand the variability of the insulating capability of the insulation due to the many different environmental effects including fabrication variability and time of exposure to the space environment.

Table 14: Deviations from Pre-conceptual Designs

Case	As Modeled	Pre-conceptual Design	Qualitative Effect(s) of Deviations
mod_8 Be/B₄C	(6) 1cm diameter cylinder structural connections to Rx	Support skirt	Under calculated conduction area from Rx to shield and thermal flux into shield. Reference (p) has calculations with updated geometry. ~ 200 - 300 W more heat conducts into shield.
	(1) hot & (1) cold leg OD 18cm Uninsulated cold pipe	(1) hot & (1) cold leg OD 16cm Insulated cold pipe	Over calculated heat transfer area from piping to shield by 12.5%. Over calculated cold pipe heat flux into shield by 0.4 kW. Under calculated heat flux from hot pipe by 0.9kW or more.
	10X IHG curves from reference (a)	ORNL final shield document	Negligible: See Figure 5 Preconceptual shield is thicker, but IHG in aft section is insignificant compared to front.
	Be/B ₄ C layers (12 layers; 4.5cm each)	8X(5cm Be/0.5cm B ₄ C)+1.5cm W +4X(5cm Be/0.5cm B ₄ C) + 4cm Be	Negligible: See Figure 5
	Total 59.5cm thickness	Total 74cm thickness	Under calculated thermal radiation surface area and flux out of shield
	front face diameter: 1.148m Conical Shaped shield	Ellipsoidal: Major radii: 70.335cm Minor radii: 56.795cm	Under calculated thermal radiation surface area and flux out of shield
	Reactor diameter=40cm	Reactor Diameter=85cm	Under calculated heat flux into shield by ~350 - 450 W
	No insulation on shield surfaces	Insulation on shield surfaces	Over calculated thermal radiation out of shield
mod_8b Be/B₄C with insulated cold Pipes & shield surfaces	(6) 1cm diameter cylinder structural connections to Rx	Support skirt	Under calculated conduction area from Rx to Shield and thermal flux into shield. Reference (p) has calculations with updated geometry. ~ 200 - 300 W more heat conducts into shield.
	(1) hot & (1) cold leg OD 18cm Insulated cold pipe	(1) Hot & (1) cold leg OD 16cm Insulated cold pipe	Over calculated heat transfer area from Piping to Shield by 12.5% and flux into shield. Under calculated heat flux from hot pipe by 1kW or more.
	10X IHG curves from reference (a)	ORNL final shield document	Negligible: See Figure 5 Preconceptual shield is thicker, but IHG in aft section is insignificant compared to front.
	Be/B ₄ C layers (12 layers; 4.5cm each)	8X(5cm Be/0.5cm B ₄ C)+1.5cm W +4X(5cm Be/0.5cm B ₄ C) + 4cm Be	Negligible: See Figure 5
	front face diameter: 1.148m Conical Shaped shield	Ellipsoidal: Major radii: 70.335cm Minor radii: 56.795cm	Under calculated thermal radiation surface area and flux out of shield
	Reactor diameter=40cm	Reactor Diameter=85cm	Under calculated heat flux into shield by ~350 - 450W
	Total 59.5cm thickness	Total 74cm thickness	Under calculated thermal radiation surface area and flux out of shield

Table 14 (cont'd): Deviations from Pre-conceptual Designs

Case	As Modeled	Pre-conceptual Design	Qualitative Effect(s) of Deviations
<i>mod_10</i> <i>Boron</i>	(6) 1cm diameter cylinder structural connections to Rx	Support skirt	Under calculated conduction area from Rx to Shield and flux into shield. Reference (p) has calculations with updated geometry. ~ 200 - 300 W more heat conducts into the shield.
	(1) hot & (1) cold leg OD 18cm Uninsulated cold pipes	(1) Hot & (1) cold leg OD 16cm Insulated cold pipes	Over calculated heat transfer area from Piping to Shield by 12.5% and thermal flux into shield. Over calculated cold pipe heat flux into shield by 0.4kW. Under calculated heat flux from hot pipe by .7kW or more.
	10X IHG curves from reference (a)	ORNL final shield document	Negligible: See Figure 5 Preconceptual shield is thicker, but IHG in aft section is insignificant compared to front.
	59.5cm boron	12cm B / 0.5cm W/ 57cm B	Negligible: See Figure 5
	Front face diameter: 1.148m Conical Shaped shield	Ellipsoidal: Major radii: 70.335cm Minor radii: 56.795cm	Under calculated thermal radiation surface area and flux out of shield
	Total 59.5cm thickness	Total 72cm thickness	Under calculated thermal radiation surface area and flux out of shield
	Reactor diameter=40cm	Reactor Diameter=85cm	Under calculated heat flux into shield by ~350 - 450W
	No insulation on shield surfaces	Insulation on shield surfaces	Over calculated thermal radiation out of shield
<i>mod_13</i> <i>LiH</i>	(6) 1cm diameter cylinder structural connections to Rx	Support Skirt	Under calculated conduction area from Rx to Shield and thermal flux into shield. Reference (p) has calculations with updated geometry. ~ 200 - 300 W more heat conducts into the shield.
	(2) Hot & (2) Cold leg OD 8cm Insulated cold pipe	(1) Hot & (1) Cold leg OD 16cm Insulated cold pipe	Same heat transfer surface area. Under calculated heat flux from hot pipe by .4kW or more.
	10X IHG curves from reference (a)	ORNL Final Shield Document	Negligible: See Figure 5 Preconceptual shield is thicker, but IHG in aft section is insignificant compared to front.
	5cm Be/B4C/Be/B4C/Be layers; 34.5cm LiH	5X(5cm Be/0.5cm B4C)+3cm W + 39cm LiH	Negligible: See Figure 5
	front face diameter: 1.148m Conical Shaped shield	Ellipsoidal: Major radii: 70.335cm Minor radii: 56.795cm	Under calculated thermal radiation surface area and flux out of shield
	Reactor diameter=40cm	Reactor Diameter=85cm	Under calculated heat flux into shield by ~350 - 450W
	Total 59.5cm thickness	Total 80cm Thickness	Under calculated thermal radiation surface area and flux out of shield
<i>mod_14</i> <i>Water</i>	(6) 1cm diameter cylinder structural connections to Rx	Support Skirt	Under calculated conduction area from Rx to Shield and thermal flux into shield. Reference (p) has calculations with updated geometry. ~ 200 - 300 W more heat conducts into the shield.
	(4) Hot & (4) Cold leg OD 8cm Uninsulated Cold Pipes	(1) Hot & (1) Cold leg OD 16cm Insulated Cold Pipes	Over calculated heat transfer area from Piping to Shield by 100% and flux into shield. Over calculated cold pipe heat flux into shield by 0.8 kW. Heat flux from hot pipe approximately correct or possibly under-calculated.
	10X IHG curves from reference (a)	ORNL Final Shield Document	Negligible: See Figure 5 Preconceptual shield is thicker, but IHG in aft section is insignificant compared to front.
	5cm Be/B4C/Be/B4C/Be layers; 34.5 Insulated Water Can	(19cm H2O w/ LiOH) + 2cm W + (66cm H2O w/ LiOH)	Negligible: See Figure 5
	front face diameter: 1.148m Conical Shaped shield	Ellipsoidal: Major radii: 70.335cm Minor radii: 56.795cm	Under calculated thermal radiation surface area and flux out of shield
	Total 59.5cm thickness	Total 87.25cm Thickness	Under calculated thermal radiation surface area and flux out of shield
	Reactor diameter=40cm	Reactor Diameter=85cm	Under calculated heat flux into shield by ~350 - 450W
	No Insulation on Shield Surfaces	Insulation on Shield Surfaces	Over calculated thermal radiation out of shield; underestimated shield temperatures

In general the heat flux into the shield from the reactor, the heat flux from the shield to the plant and space was under predicted for all shield configurations. The required heat flux from the hot leg piping into the shield was under calculated for the Be/B₄C, Boron, and LiH configurations. The specific thermal management issues associated with the hot leg piping would need to be addressed in more detailed thermal analyses. It is believed that the heat flux from the reactor and from the shield can be controlled by means of controlling surface emissivity, conductivity or blanketing.

Figure 5: Internal Heat Generation used in Analyses compared to Reference (k)

Figure 5a: Be/B₄C Shield Configuration

Figure 5b: H₂O Shield Configuration

Figure 5c: LiH Shield Configuration

Figure 5d: B Shield Configuration

Sensitivity Studies

Three sensitivity studies have been performed on the base cases. The first study was a modification of the Be/B₄C (*mod_8*) case. The modification (*mod_8b*) added insulation to the cold pipe and also to the sides of the shield and the aft surfaces of the shield to represent effects of minimizing heat loss from the cold pipe, isolating the shield from the plant, and reducing heat loss prior to start-up. As expected, most of the heat flux from the cold pipe was eliminated and the heat flux out of the side and aft of the shield was reduced, but more heat was rejected out of the front shield surface. The insulation on the exterior surface was the dominating effect and resulted in an increase in shield temperature of 150K. The effects would be similar for the Boron shield case.

The *mod_13b* study on the LiH (*mod_13*) base case eliminated the neutron and gamma radiation heating effect in the model. The total energy added to the shield from this phenomenon is 2 kW. As a result of neglecting this effect, the temperature of the shield was reduced by about 90K. The purpose of this study was to quantify the effects of the neutron and gamma heating of the shield.

The *mod_13c* study on the LiH (*mod_13*) base case changed the method of modeling the shield exterior insulation. The base case models the insulation explicitly by applying the conductivity of insulation to elements that represent the insulation. The more commonly used and straightforward method is to model the insulation with an effective emissivity for the thermal radiation from the surface of the shield. The *mod_13c* case uses an effective emissivity of 0.02 and made the insulation have the same conductivity of the beryllium. This study showed that there was small change in the results from the base case LiH shield for this analysis.

Analysis Software

ABAQUS Standard v6.5.4 has been used for the thermal FEA. The analyses are all steady state thermal analyses which use 3D heat transfer continuum elements.

References

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- (c) Bettis Letter, B-MT(SRME)-36, "Updated Refractory Metal Properties", dated October 25, 2005.
- (d) KAPL Letter, MDO-723-0042, "Reflector and Shield Material Properties for Project Prometheus", dated November 2, 2005.
- (e) KAPL Letter, Enclosure 2 of SPP-67210-0001, "SNPP Candidate Material Capabilities and Liabilities", dated November 2, 2004.
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- (g) Haynes International Property Data Sheet H3000H, "HAYNES 230 Alloy", dated February 5, 2004.
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- (i) General Electric: Space Systems Division Letter, 1444L, VP Haley, "Insulation for TEM Pump", dated September 23, 1987.
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- (k) Oak Ridge National Laboratory, ORNL/LTR/NR-PROM1/05-27, "Reactor Shield Optimization in Support of the Prometheus 1 Project Preconceptual Design Analyses", dated October 25, 2005.
- (l) KAPL Letter MDO-723-0048, "The Evaluation of Lithium Hydride for Use in a Space Nuclear Reactor Shield, Including a Historical Perspective, for NR Information", dated December 9, 2005.
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- (p) KAPL Letter SPP-SPPS-0035, "Reactor Module Thermal Zone Scoping Analyses", dated TBI

CONCURRENCE/DESIGN CHECK FORM FOR DOCUMENT NO. SPP-67210-0011Date: 2/17/06DOCUMENT
TITLE:

Space Reactor Radiation Shield Design Summary, For Information

REFERENCES All Encl. 1 & 2ENCLOSURES: 1, 21. ADSARS: PERMANENT RECORD: Yes x No Repository MFLIB Corporate Author: KAPL NR PROGRAM RKey Words: SHIELD, PIPES, CDM, PROMETHEUS, JIMONeed to Know Categories REPAvailable Sites: BETT KAPL NRHQ PNRO SNRO Design File Location(s) BETT KAPL

2. DESIGN CHECK

Type of Check	Signature(s)	Comments: (Including Reference to Check Document if Appropriate)
A. No check considered necessary		
B. Check vs. previous results/issues		
C. Checked calculations made	<u>AD Brady</u>	<u>checked fusion production c, d, e, h, i</u>
D. Checked computer input and/or output		
E. Computer Programs approved/qualified		
F. Performed independent audit	<u>APL Data</u>	<u>Reviewed Enclosure 2 for reasonableness</u> <u>Order of magnitude accuracy</u>
G. Spot checked significant points		
H. Reviewed methods used		
I. Reviewed results for reasonableness	<u>MG Collins</u>	<u>Encl. 2 + cover letter c, d, e, h, i, m, n</u>
J. Comparison with test data		
K. Reviewed vs. drawings		
L. Verified procedures		
M. Technical content reviewed		
N. Management verification of adequate review by others		
O. Performed Lessons Learned Search		
P. Used Measurement Uncertainty Methods		
Q. Other Checks (Describe)		

3. CONCURRENCE REQUIREMENTS:

Indicate signatures required by X:

SPP MANAGER
x DEMLAY
NUCLEAR ENGINEERING
REACTOR TH/MECH DESIGN
REACTOR EQUIPMENT
x DEMLAY
SPP MECHANICAL
SPP ELECTRICAL
FINANCE
PROJECT OFFICE
ENERGY CONVERSION
x DEMLAY
Cognizant Manager

ADVANCED CONCEPTS
x DEMLAY
SHIELDING Bettis
REACTOR SAFETY
TO
RSO
FSO
x DEMLAY
MDO SPACE MATERIALS
x DEMLAY
x DEMLAY

FLUID DYNAM
STRUC. ENGRG
DRAFTING
QA
OTHER
BETTIS
BPMI
ADMIN REVIEW

2/16/06
(Must Be Subsection or Higher for External Letters)

4. AUTHORIZED CLASSIFIER:

Reviewed By: MG CollinsCLASSIFICATION: unclassified

5. RELATED SUBJECTS:

	Commitment Made (Y/N)	Commitment Complete (Y/N)
UTRS Implication (Y/N)	<u>N</u>	<u>N</u>
Safety Council Review (Y/N)	<u>N</u>	<u>N</u>

6. Distribution: see next page

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