NRIC National Reactor Innovation Center

Preliminary Criticality and Radiation Shielding Analysis for the Storage and Transfer of Irradiated MARVEL Reactor Fuel

Evans D. Kitcher

NRIC 20-SDD-0001 | 08/26/2020
INL/EXT-20-59200

Idaho National Laboratory

EXECUTIVE SUMMARY

This report documents the results of preliminary nuclear criticality and radiation shielding assessments during transfer and dry storage of irradiated Microreactor Applications Research Validation and Evaluation (MARVEL) reactor fuel at Idaho National Laboratory (INL). The assessments focus on transfer casks and storage canisters that are currently in use at INL, which may be compatible with the irradiated MARVEL reactor fuel.

The criticality assessments were performed using the radiation transport code MCNP6 with 37 MARVEL reactor fuel elements in various configurations and scenarios. All transfer and storage configurations under dry conditions were below the assumed criticality safety limit of 0.93. Some storage and transfer configurations under wet conditions exceeded the criticality safety limit. This suggests that the appropriate administrative and engineering controls, in addition to reducing the number of MARVEL reactor fuel elements per transfer cask or storage canister, can be expected to ensure criticality safety under all scenarios.

The radiation shielding assessments were performed by generating conservative neutron and photon source terms using the ORIGEN module in the SCALE suite of codes. These source spectra were used to estimate the dose equivalent rates using the radiation transport code MCNP6, both on contact and 1 m away from the fuel and transfer casks. The maximum estimated dose equivalent rate of 37 unshielded MARVEL reactor fuel elements on contact is approximately 42000 R/hr. The maximum estimated dose equivalent rates on contact to the ATR transfer cask, HFEF-5 transfer cask, and high load charger were approximately 233 mR/hr, 171 mR/hr, and 201 mR/hr, respectively. This suggests that with the appropriate administrative and engineering controls, all three transfer casks analyzed can be expected to provide sufficient radiation shielding to workers during transfer of irradiated MARVEL reactor fuel.

These calculations are performed to support the planning and strategy for the MARVEL project and will demonstrate the technical viability of the different configurations discussed and help identify where engineered or administrative controls may be necessary. A complete criticality safety analysis and radiation shielding analysis, including validation and contingency and accident analysis must be completed by licensed and authorized personnel before any transfer or storage of MARVEL reactor nuclear fuel.

CONTENTS

FIGURES

TABLES

ACRONYMS

ATR	Advanced Test Reactor
BOL	Beginning of Life
DOE	Department of Energy
HFEF-5	Hot Fuel Examination Facility
HLC	High Load Charger
INL	Idaho National Laboratory
MARVEL	Microreactor Applications Research Validation and Evaluation
SNF	Spent Nuclear Fuel
R&D	Research and Development

1. INTRODUCTION

The Microreactor Applications Research Validation and Evaluation (MARVEL) reactor is a microreactor designed for the rapid development and implementation of a small scale microreactor for the purpose of testing the unique operational aspects and applications of microreactors. The MARVEL project will foster engagement with potential microreactor end-users to investigate and facilitate the integration of end-user application technologies with microreactors. The MARVEL reactor will meet critical research and development (R&D) needs of existing developers; provide R&D infrastructure to support design, demonstration, regulatory, and safety-related tests; develop advanced technologies and concepts for next-generation microreactor applications and systems; and enable future microreactor applications (e.g., district heat, hydrogen production, and defense applications). Figure 1 shows an overview of the MARVEL project concept. The MARVEL reactor itself is fueled by 36 sodium-bonded UZrH fuel elements using high-assay low-enriched uranium (HALEU) enriched to 19.75% of U-235. The primary coolant is sodium with an operating temperature of 500–550°C. Four Stirling engines will convert the 100 KW of thermal power into 20 kW of electric power to facilitate testing of microreactor applications.

The objective of the calculations presented here is to perform preliminary criticality and radiation shielding evaluations for various transfer and storage configurations of 37 MARVEL reactor fuel elements. Configurations include various transfer casks for the transfer of spent nuclear fuel (SNF) and irradiated experiments between INL facilities, and an SNF storage canister currently in use at INL. These calculations will support the planning and strategy for the MARVEL project, demonstrate the technical viability of the different configurations discussed, and help identify where engineered or administrative controls may be necessary. A complete criticality safety analysis and radiation shielding analysis, including validation and contingency and accident analysis, must be completed by licensed and authorized personnel before any transfer or storage of irradiated MARVEL reactor fuel.

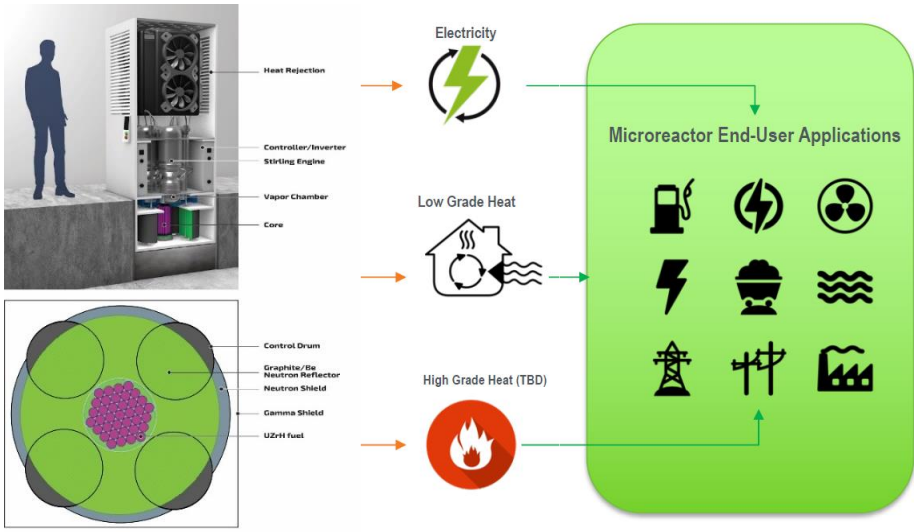

Figure 1. MARVEL project concept overview.

2. METHODS

The method to perform the criticality calculations consists of using MCNP Version 6.2 to calculate the effective neutron multiplication factor of MARVEL reactor fuel in transfer casks and storage canisters, in order to simulate the transfer and storage in a dry storage system. The calculations are performed using the ENDF/B-V continuous-energy cross section libraries. All calculations are performed with the beginning-of-life (BOL) fuel compositions which are expected to be bounding in terms of fuel reactivity.

The method to perform the radiation shielding calculations is to first generate a bounding source term and associated photon and neutron source spectra for the irradiated MARVEL reactor fuel using the ORIGEN module in the SCALE code suite (Rearden and Jessee 2018). This method is based on generic BOL isotopic composition irradiated at full power (100kW) for 730 days, using a generic thermal reactor neutron flux spectrum. Using MCNP 6.2 (Werner 2017, 2018), this source term is used to calculate the total dose equivalent rate from photon, neutron, and neutron-induced photon dose equivalent rates at various locations. All shielding calculations use the bounding source term for fuel reactivity.

3. DESCRIPTIONS

3.1 MARVEL Reactor Fuel Elements

The MARVEL reactor fuel element consists of a uranium-zirconium-hydride fuel meat with an active fuel length of approximately 51.05 cm. Above and below the fuel meat are 20-cm-long axial beryllium reflectors. The fuel pin cladding material is made of stainless-steel-316. Between the fuel meat and cladding is a very thin sodium bond. Figure 2 shows the MARVEL reactor fuel pin geometry.

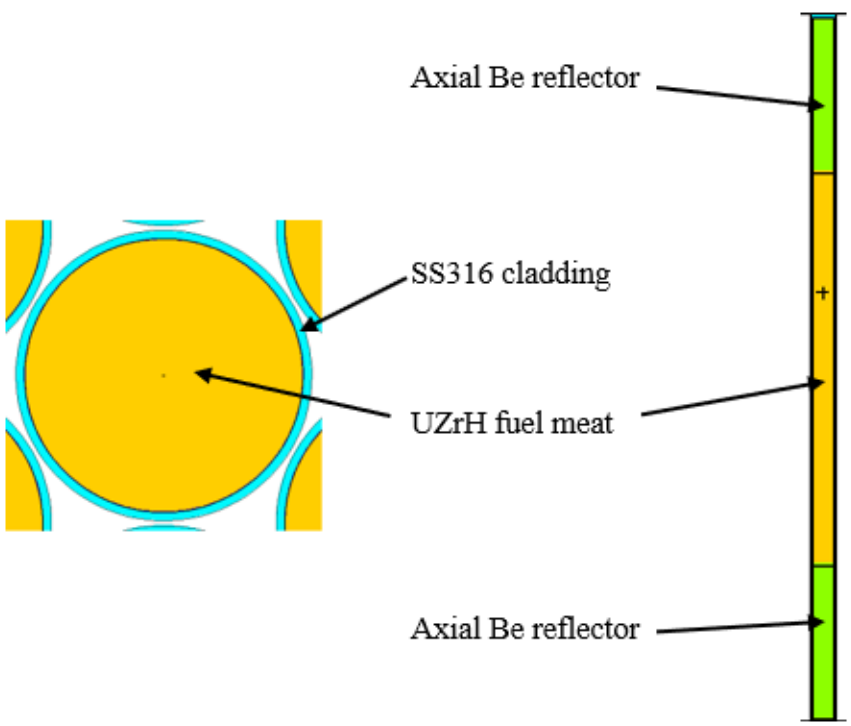

Figure 2. MARVEL reactor fuel pin geometry.

3.2 ATR Transfer Cask

The Advanced Test Reactor (ATR) Spent Fuel Element Transfer Cask is a 74.625-in.-tall by 35-in.-diameter cylindrical vessel with flanged and dished heads (Christensen 2003). Stainless steel shells surround approximately 10.5 in. of lead shielding. The top head opening is used for access to a fuel divider that holds up to eight ATR fuel elements. The top of the cask is closed with a lead-shielded plug. Trunnion structures attached near the top of the cask are used for lifting. A schematic of the ATR transfer cask is shown in Figure 3.

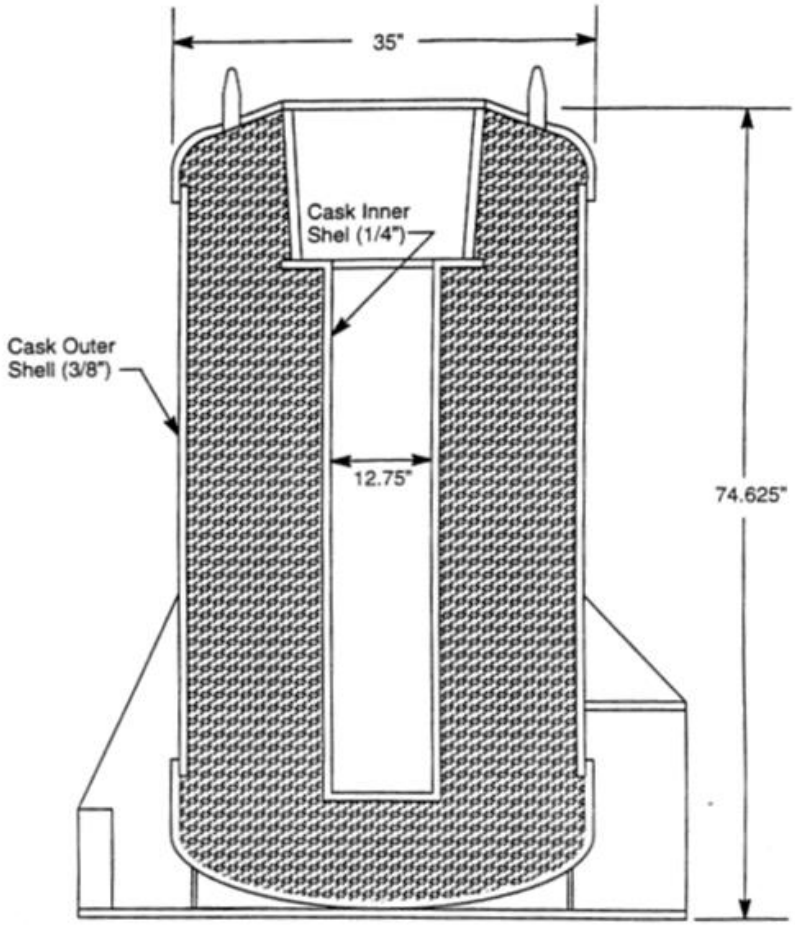

Figure 3. Schematic of the ATR transfer cask.

3.3 HFEF-5 Transfer Cask

The HFEF-5 transfer cask is a lead-filled cylindrical shipping cask with a cylindrical interior cavity (Sentieri and Casanova 2019). The HFEF-5 transfer cask is nominally 110.4-in.-tall. The outer diameter is 33 in., and the inner diameter is 14.3 in. at the top and tapers to 13.1 in. at the bottom. The lead-filled annulus is 8.5-in.-thick. The cask can be loaded and unloaded from either the top or bottom. Details of the HFEF-5 transfer cask are shown in Figure 4. The outer container serves as a containment barrier within the cask. The outer container houses an inner container, which houses the product or waste containers. The stainless-steel outer container has walls constructed of 14-gauge stainless-steel and has an outer diameter of

12.75 in. The carbon-steel inner container, which houses the actual product or waste packages, has an outer diameter of 10.0 in., with walls constructed of 14-gauge carbon steel.

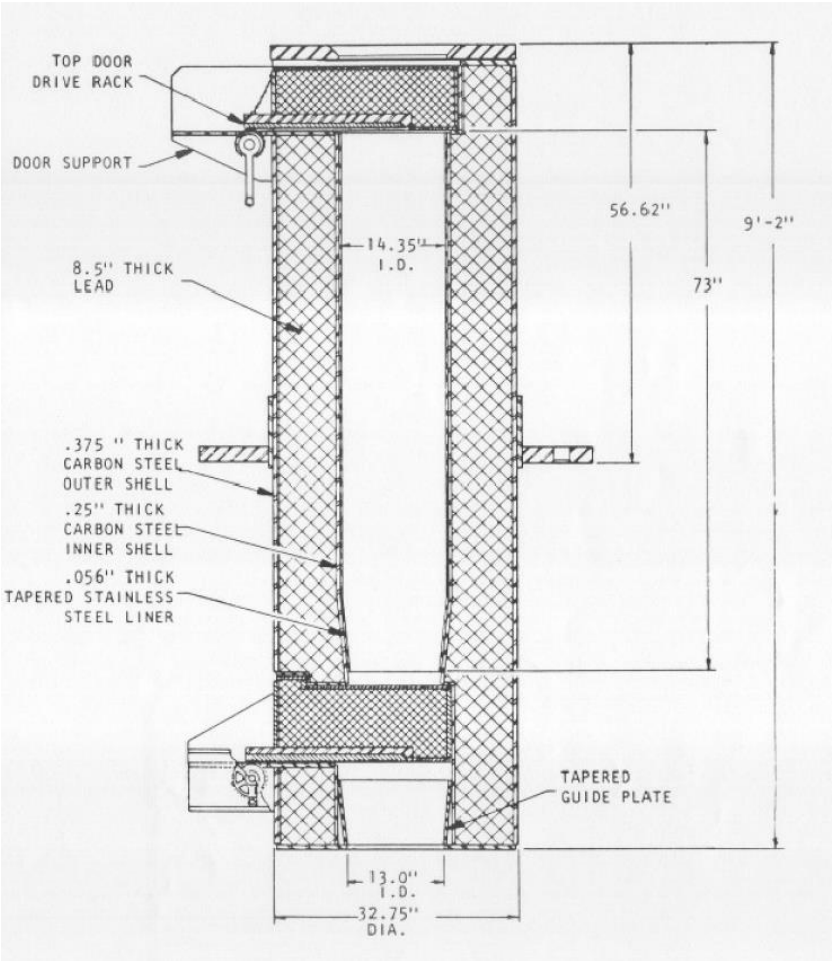

Figure 4. Schematic of the HFEF-5 transfer cask.

3.4 High Load Charger

The high load charger (HLC) is a square-cavity, lead-shielded, shipping cask typically used for the transportation of spent fuel elements. The dimensions of a HLC cavity are 13.5-in. × 13.5-in. × 51.5-in. The sides of the HLC are comprised of a 0.5-in.-thick inner stainless-steel liner, a 9.75-in.-thick lead shielding, and a 0.5-in. thick stainless-steel outer shell. The bottom of the HLC is 6.5-in.-thick lead shielding. The lid of the HLC is 9.75-in.-thick lead sandwiched between two, 0.25-in.-thick stainless-steel sheets. The HLC is shown in Figure 5.

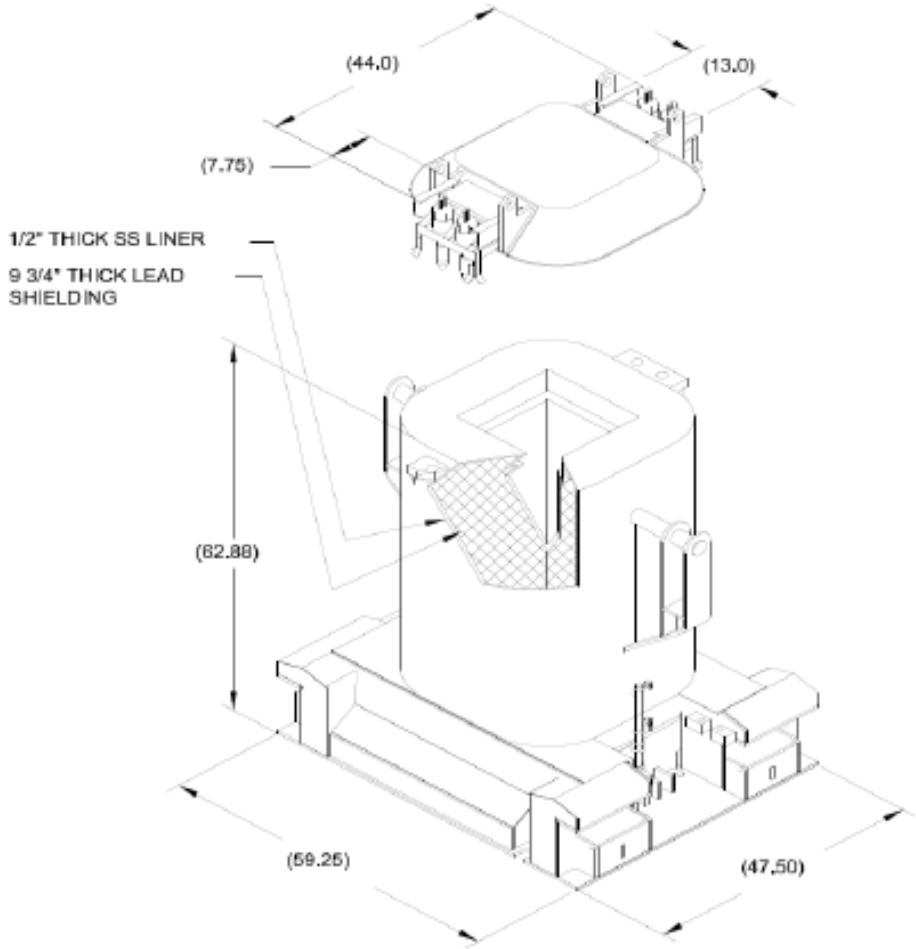

Figure 5. Schematic of the high load charger.

3.5 Storage Canisters

There are several variations of light weight storage canisters used at INL for the storage of SNF. These canisters are made of stainless-steel or carbon steel, and come in various sizes. The canisters used in the CPP603 irradiated fuel storage facility for the dry storage of various SNF have an 18-in. nominal diameter, and a length of about 132.0-in. These canisters could potentially be used for the storage of irradiated MARVEL reactor fuel. Figure 6 shows a drawing of the 18-in. canister with its lid (DWG 154872 and DWG 155808). Additionally, the DOE Standard Canister also has an 18-in. diameter variant (Morton 1998, DWG ISF-ME-S-16001). The DOE Standard Canister is designed as part of a larger system for storage, transportation, and eventual disposal of SNF. As such, an 18-in. canister configuration is of interest for storage that would be representative of eventual disposition. Figure 7 shows a drawing of the DOE Standard Canister, respectively.

Figure 6. Drawing of an 18-in. storage canister with lid.

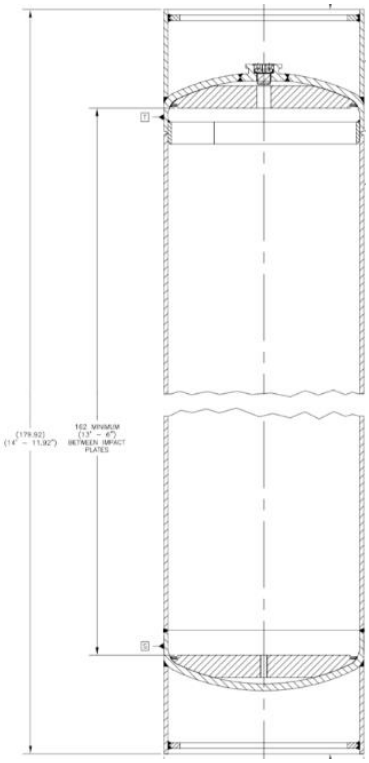

Figure 7. Drawing of the DOE standard canister.

4. ASSUMPTIONS

For the criticality and radiation shielding analysis calculations a few key assumptions must be made: (1) the criticality safety limit for the effective neutron multiplication factor in this evaluation is 0.93. Any configuration or scenario with an effective neutron multiplication factor above this limit would be unacceptable. (2) Air is modeled as void space. (3) A most reactive fissile concentration is used for the both the MARVEL reactor fuel and the ATR fuel as shown in Table 1 and (3) the MARVEL reactor fuel is enriched to 19.75% U-235, and the ATR fuel is enriched to 93% U-235. The selected fuel compositions use the BOL fuel material composition and ignore any reactivity control additives. The basis of this assumption is that the selected compositions are conservative since they maximize the fissile isotope content while minimizing the effect of neutron absorbers.

5. MATERIALS

This section gives the material compositions used in the models.

Table 1. Composition of uranium-zirconium-hydride (MARVEL reactor fuel meat).

Element	Composition (at %)
Hydrogen	60
Zirconium	35
Uranium	5
Hafnium	Trace amounts
Density = 6.83 g/cm3	

Table 2. Composition of uranium dispersed in aluminum matrix (ATR fuel meat).

Element	Composition (at %)
Aluminum	95%
Uranium	5%
Density = 2.6 g/cm3	

Table 3. Composition of stainless steel 316.

Element	Composition (at %)
Iron	65
Chromium	18
Nickel	11
Manganese	2
Silicon	2
Molybdenum	1
Density = 7.87 g/cm3	

Table 4. Composition of stainless steel 304.

Element	Composition (at %)
Iron	69
Chromium	20
Nickel	9
Silicon	1
Manganese	1
Sulphur, Carbon, Phosphorus	Trace amounts
Density = 7.82 g/cm3	

Table 5. Composition of carbon steel.

Element	Composition (at %)
Iron	98
Carbon	2
Density = 7.82 g/cm3	

Table 6. Composition of aluminum for ATR cladding.

Element	Composition (at %)
Aluminum	98%
Magnesium	1%
Silicon	1%
Iron, Copper, Chromium, Manganese, Titanium	Trace amounts
Density = 2.7 g/cm3	

Table 7. Composition of water.

Element	Composition (at %)
Hydrogen	66.6
Oxygen	33.3
Density = 1.00 g/cm3	

Table 8. Composition of liquid sodium bond.

Element	Composition (at %)
Sodium	100
Density = 0.82 g/cm^3	

Table 9. Composition of beryllium metal.

Element	Composition (at %)
Beryllium	100
Density = 1.85 g/cm^3	

Table 10. Composition of lead shielding.

Element	Composition (at %)
Lead	100
Density = 11.34 g/cm^3	

6. SCENARIOS

6.1 Criticality Scenarios

The criticality scenarios are focused on estimating the effective neutron multiplication factor for 37 MARVEL reactor fuel elements loaded in the ATR transfer cask, the HFEF-5 transfer cask, the high load charger, and loaded with ATR fuel elements inside an 18-in diameter storage canister. The scenarios include dry and flooded conditions, varying the spacing between the fuel elements, removing the axial beryllium reflectors, and the aggregation of fuel elements in one part of the casks or canister. At present, there are no basket or insert designs for the irradiated MARVEL reactor fuel, which are commonly used to maintain a known geometry or suppress criticality in transfer and storage of SNF. As such, no other structures are modeled in the casks or canister. Table 11 shows the different cases and the different monikers used for the scenarios analyzed.

Table 11. Case Naming Monikers.

Case Naming Monikers	Case Naming Moniker Meaning
M_	MARVEL reactor fuel
ATR_	ATR transfer cask
HFEF_	HFEF-5 transfer cask
HLC_	High load charger
CPP_	Storage canister
ATR4B_	Storage canister with ATR4 loaded at the bottom
ATR4T_	Storage canister with ATR4 loaded at the top
D_	Dry conditions
W_	Wet conditions (all voids filled with water)
Be_	Beryllium axial reflectors in place
NoBe_	Beryllium axial reflectors removed
Clumped	Fuel elements aggregated within geometry
tpitch	Fuel elements arranged with triangular pitch
cpitch	Fuel elements arranged with circular pitch

6.2 Shielding Scenarios

The radiation shielding scenarios are focused on estimating the dose equivalent rates, both on contact and 1 m away from the unshielded MARVEL reactor fuel with two years of cooling. The dose equivalent rates are also estimated both on contact and 1 m away from the MARVEL reactor fuel shielded inside the transfer casks. The total dose equivalent rate includes contributions from photon, neutron, and neutron-induced photon dose equivalent rates at the identified locations.

7. RESULTS AND ANALYSIS

7.1 Criticality Assessment Results

7.1.1 Criticality of MARVEL Reactor Fuel in the ATR Transfer Cask

This section presents the results of the estimated effective neutron multiplication factor for the MARVEL reactor fuel loaded in the ATR transfer cask in various configurations and scenarios. All effective neutron multiplication factors are presented with associated statistical uncertainties. Figure 8 shows an axial cross section of the MARVEL reactor fuel inside the ATR transfer cask, and a radial cross section showing the MARVEL reactor fuel aggregated to one side of the ATR transfer cask. Table 12 provides the estimated effective neutron multiplication factor results. The estimated effective neutron multiplication factor is largest for the flooded case with the axial beryllium reflectors in place.

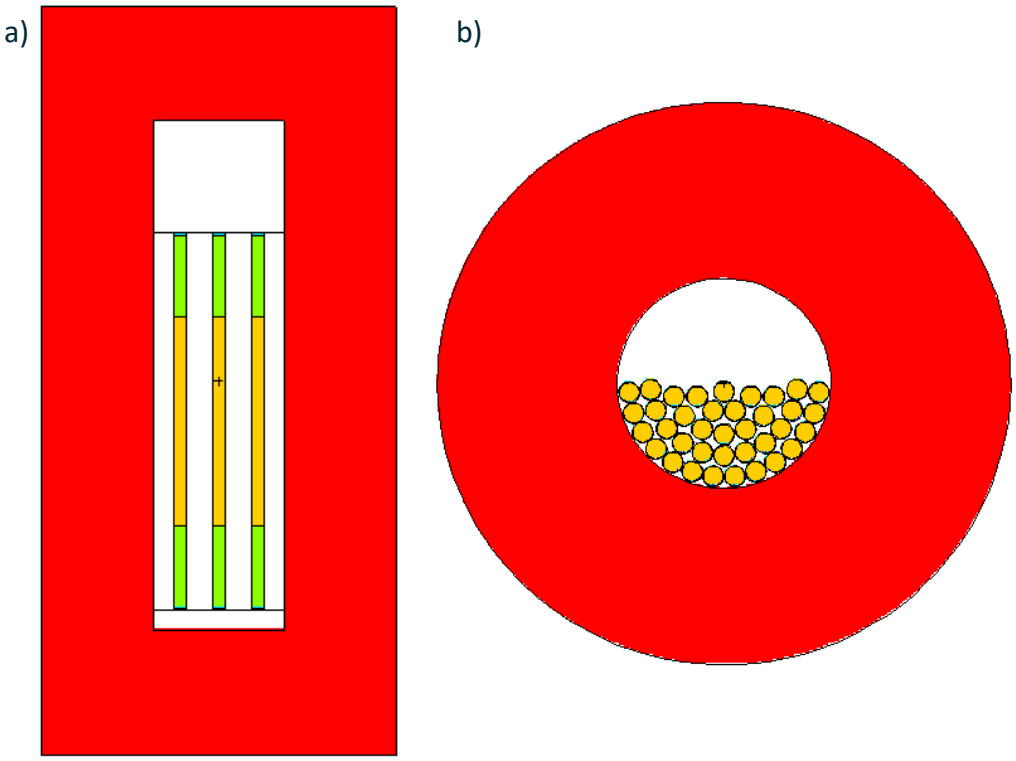

Figure 8. a) Axial cross section of MARVEL reactor fuel in the ATR transfer cask and b) radial cross section of MARVEL reactor fuel aggregated in the ATR transfer cask.

Table 12. Estimated effective neutron multiplication factor for aggregated MARVEL reactor fuel in the ATR transfer cask.

Case	Effective Neutron Multiplication Factor ($k_{eff} \pm \sigma k_{eff}$)
M_ATR_D_Be_Clumped	0.51545 ± 0.00035
M_ATR_D_NoBe_Clumped	0.50651 ± 0.00037
M_ATR_W_Be_Clumped	0.85196 ± 0.00042
M_ATR_W_NoBe_Clumped	0.84416 ± 0.00041

Figure 9 shows the MARVEL reactor fuel in a circular arrangement inside the ATR transfer cask at minimum pitch and maximum pitch. Figure 10 shows the MARVEL reactor fuel in a triangular arrangement inside the ATR transfer cask at minimum and maximum pitch.

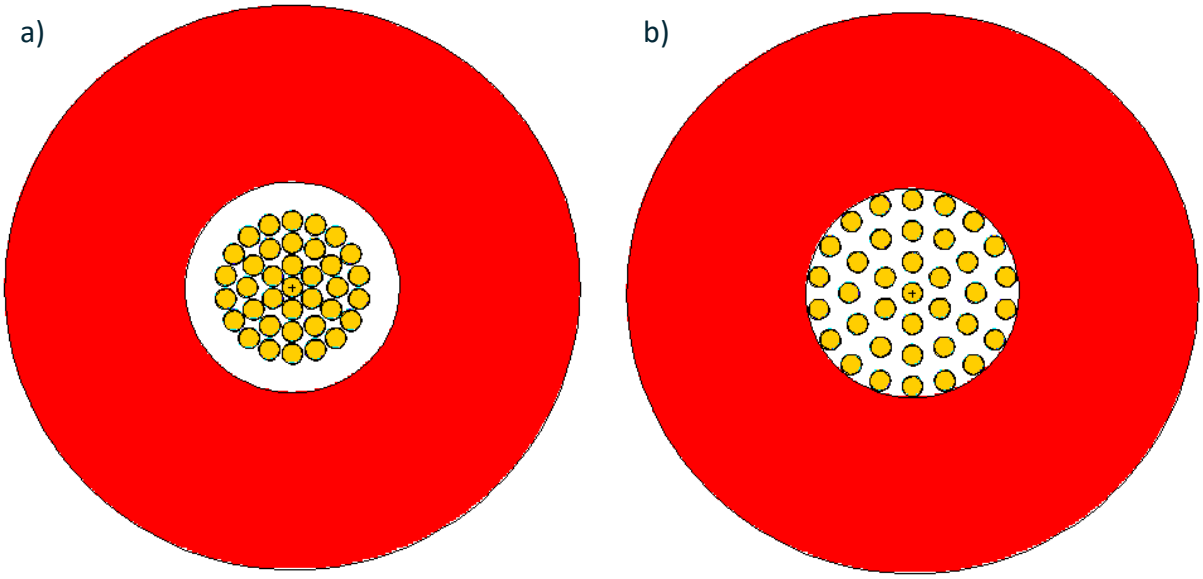

Figure 9. Radial cross sections of the MARVEL reactor fuel in the ATR transfer cask with a circular arrangement at a) minimum pitch, and b) maximum pitch.

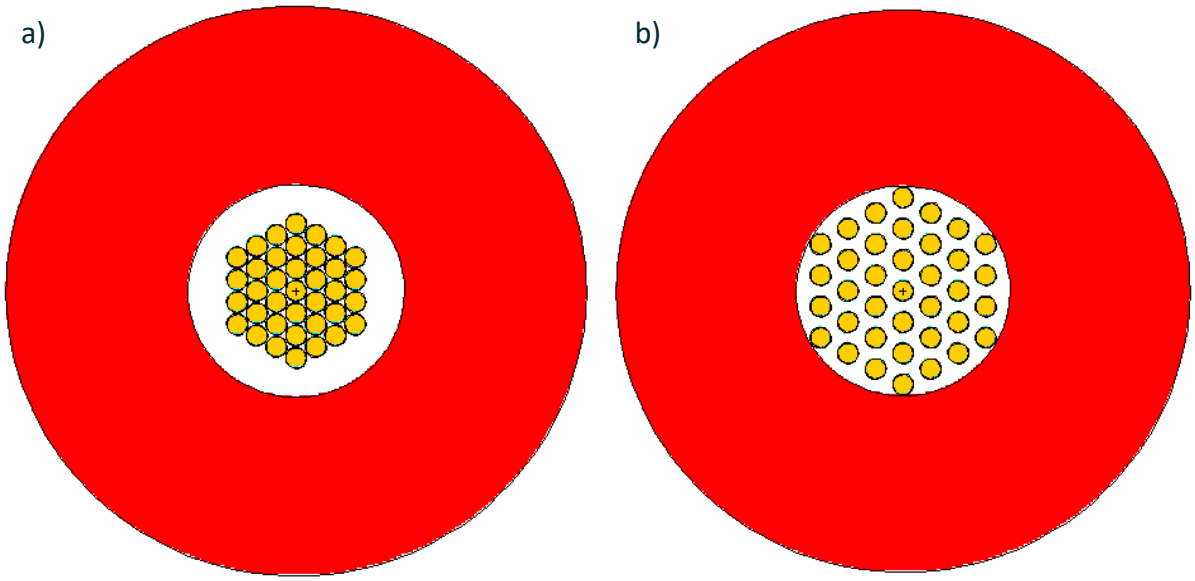

Figure 10. Radial cross sections of the MARVEL reactor fuel in the ATR transfer cask with a triangular arrangement at a) minimum pitch, and b) maximum pitch.

Figure 11 and Figure 12 show the estimated effective neutron multiplication factor values for the MARVEL reactor fuel in the ATR transfer cask, in a circular and triangular arrangement, and under dry and wet conditions, respectively. Under dry conditions, the estimated effective neutron multiplication factor decreases monotonically with increasing pitch and is significantly below the assumed criticality safety limit of 0.93. Under wet conditions, the estimated effective neutron multiplication factor shows the effect of moderation from the water with the lowest reactivity occurring at the maximum pitch of 2.35 cm for the both the triangular and

circular arrangements. At this pitch, the estimated effective neutron multiplication factor is also below the assumed criticality safety limit of 0.93.

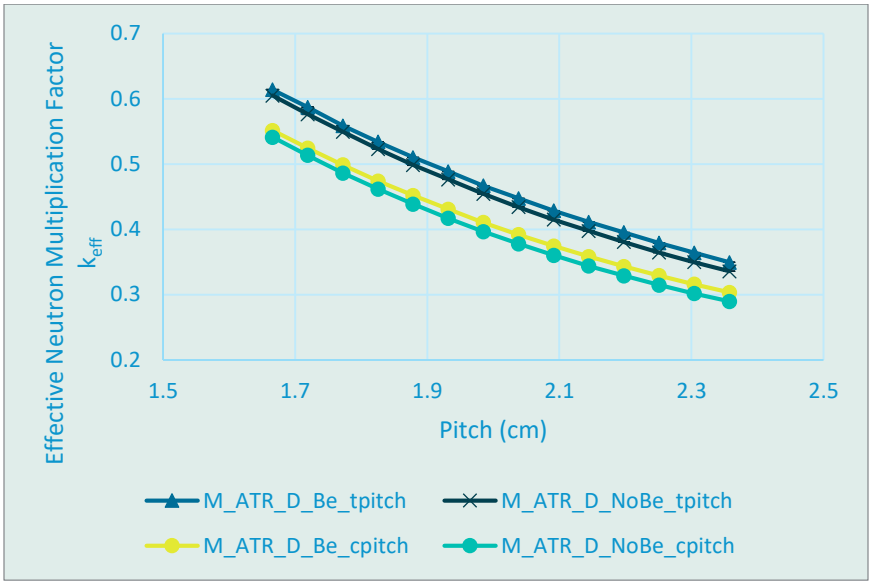

Figure 11. Estimated effective neutron multiplication factor for MARVEL reactor fuel in the ATR transfer cask in circular and triangular arrangements under dry conditions.

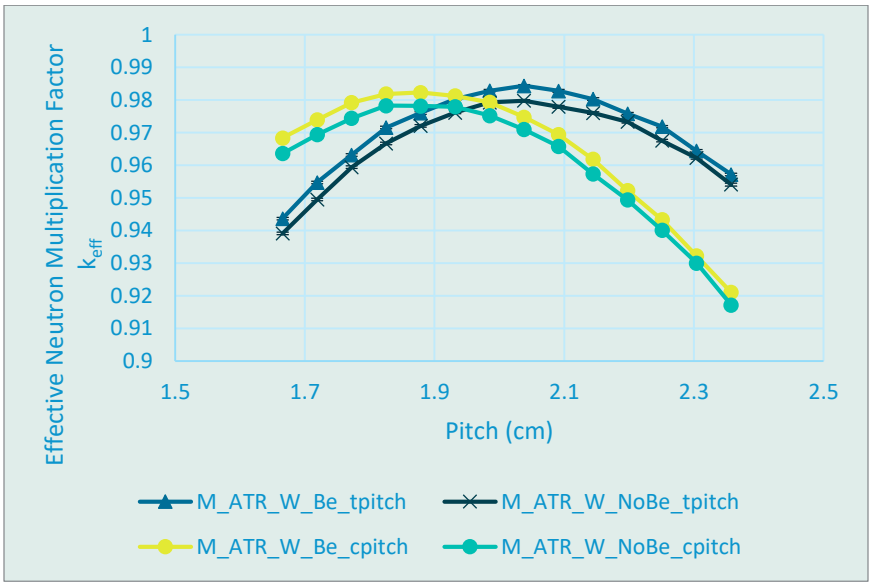

Figure 12. Estimated effective neutron multiplication factor for MARVEL reactor fuel in the ATR transfer cask in circular and triangular arrangements under wet conditions.

7.1.2 Criticality of MARVEL Reactor Fuel in the HFEF-5 Transfer Cask

This section presents the results of the estimated effective neutron multiplication factor for the MARVEL reactor fuel loaded in the HFEF-5 transfer cask in various configurations and scenarios. All effective neutron multiplication factors are presented with associated statistical uncertainties. Figure 13 shows an axial cross

section of the MARVEL reactor fuel inside the HFEF-5 transfer cask and a radial cross section showing the MARVEL reactor fuel aggregated to one side of the HFEF-5 transfer cask. Table 13 provides the estimated effective neutron multiplication factor results. The estimated effective neutron multiplication factor is largest for the flooded case with the axial beryllium reflectors in place.

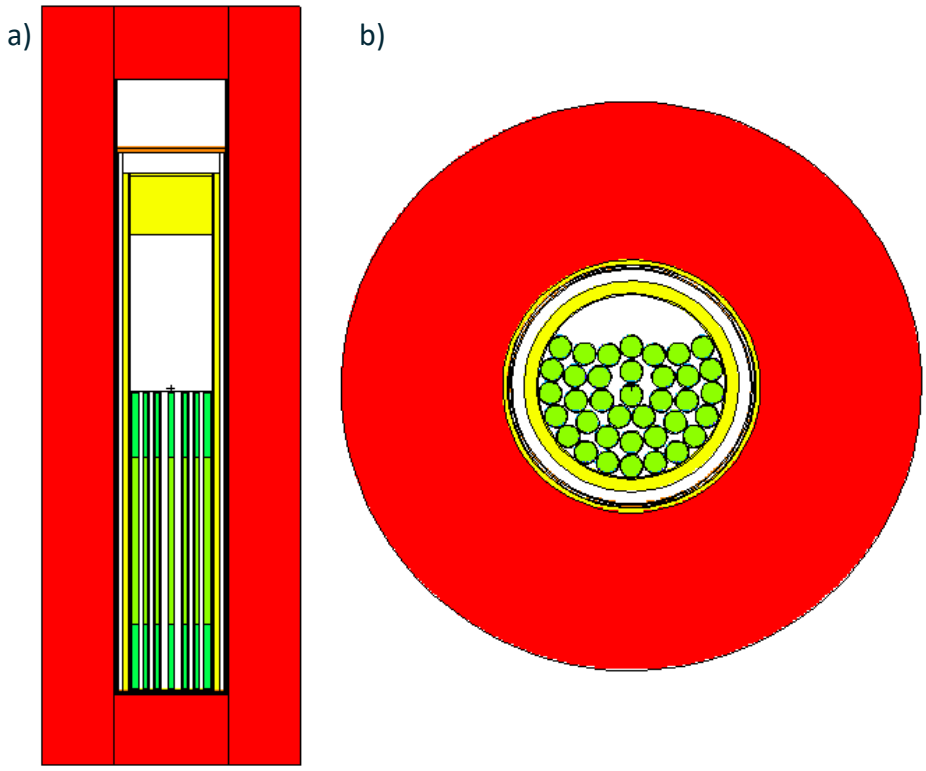

Figure 13. a) Axial cross section of MARVEL reactor fuel in the HFEF-5 transfer cask and b) radial cross section of MARVEL reactor fuel aggregated in the HFEF-5 transfer cask.

Table 13. Estimated effective neutron multiplication factor for aggregated MARVEL reactor fuel in the HFEF-5 transfer cask.

Case	Effective Neutron Multiplication Factor ($k_{eff} \pm \sigma k_{eff}$)
M_HFEF_D_Be_Clumped	0.51545 ± 0.00035
M_HFEF_D_NoBe_Clumped	0.50651 ± 0.00037
M_HFEF_W_Be_Clumped	0.97499 ± 0.0004
M_HFEF_W_NoBe_Clumped	0.9704 0 ± 0.00041

Figure 14 shows the MARVEL reactor fuel in a circular arrangement inside the HFEF-5 transfer cask at minimum pitch and maximum pitch. Figure 15 shows the MARVEL reactor fuel in a triangular arrangement inside the HFEF-5 transfer cask at minimum and maximum pitch.

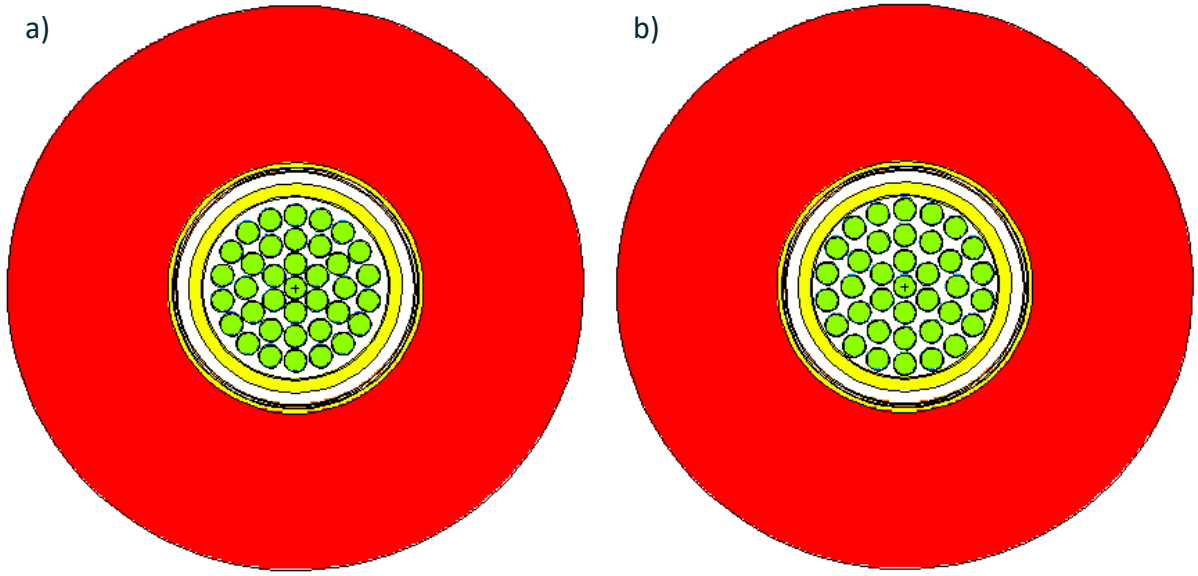

Figure 14. Radial cross sections of the MARVEL reactor fuel in the HFEF-5 transfer cask with a circular arrangement at a) minimum pitch, and b) maximum pitch.

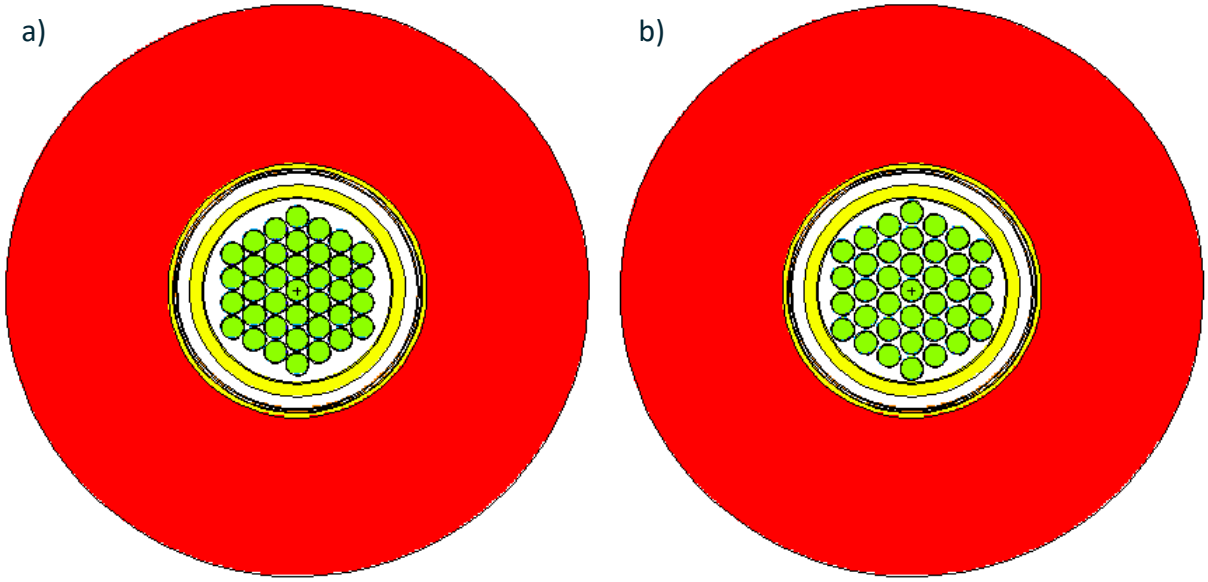

Figure 15. Radial cross sections of the MARVEL reactor fuel in the HFEF-5 transfer cask with a triangular arrangement at a) minimum pitch, and b) maximum pitch.

Figure 16 and Figure 17 show the estimated effective neutron multiplication factor values for the MARVEL reactor fuel in the HFEF-5 transfer cask, in a circular and triangular arrangement, and under both dry and wet conditions, respectively. Under dry conditions, the estimated effective neutron multiplication factor decreases monotonically with increasing pitch and is significantly below the assumed criticality safety limit of 0.93. Under wet conditions, the estimated effective neutron multiplication factor shows the effect of moderation from the water with lowest reactivity occurring at a pitch of approximately 1.66 cm for both the triangular and circular

arrangements due to the confined space within the HFEF-5 transfer cask. All HFEF-5 transfer cask scenarios under flooded conditions exceed the assumed criticality safety limit of 0.93.

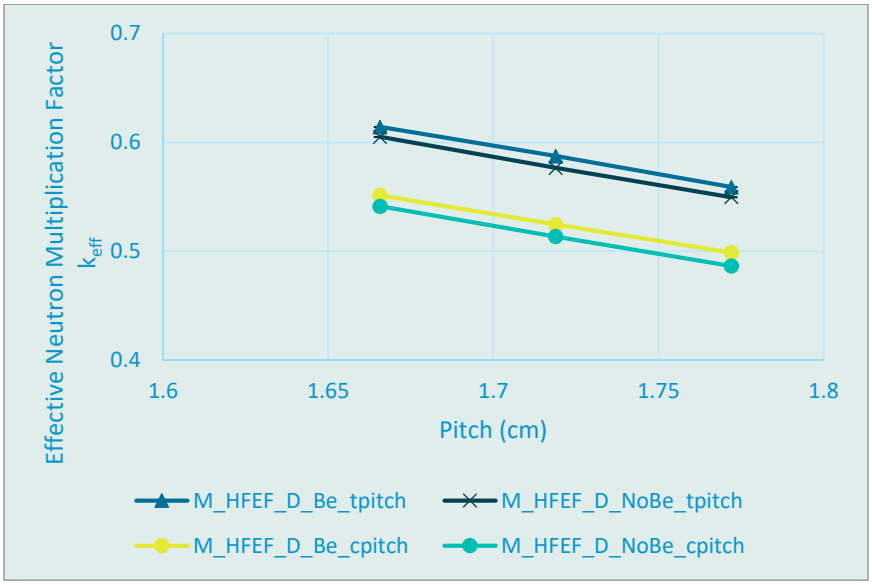

Figure 16. Estimated effective neutron multiplication factor for MARVEL reactor fuel in the HFEF-5 transfer cask in circular and triangular arrangements under dry conditions.

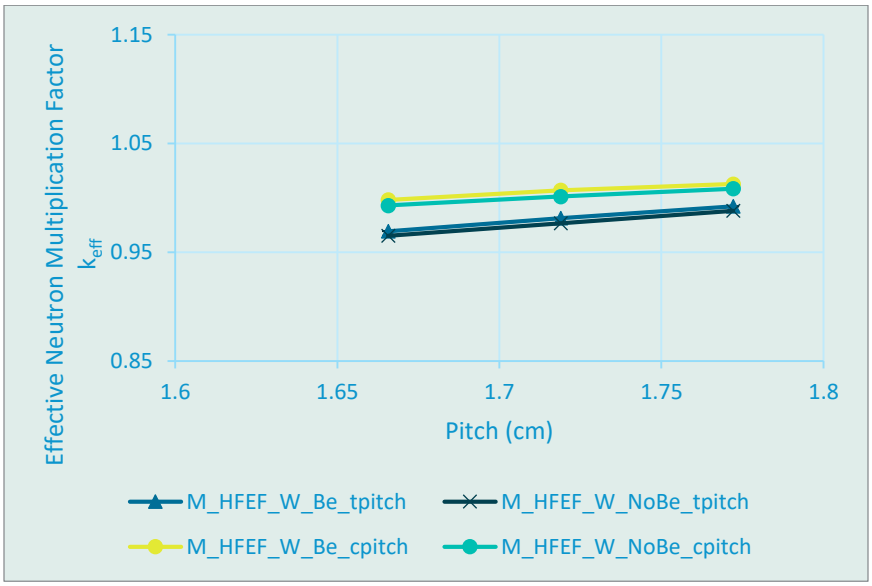

Figure 17. Estimated effective neutron multiplication factor for MARVEL reactor fuel in the HFEF-5 transfer cask in circular and triangular arrangements under wet conditions.

7.1.3 Criticality of MARVEL Reactor Fuel in the High Load Charger

This section presents the results of the estimated effective neutron multiplication factor for the MARVEL reactor fuel loaded in the HLC in various configurations and scenarios. All effective neutron multiplication factors are presented with associated statistical uncertainties. Figure 13 shows an axial cross section of the

MARVEL reactor fuel inside the HLC and a radial cross section showing the MARVEL reactor fuel aggregated to one side of the HLC. Table 14 provides the estimated effective neutron multiplication factor results. The estimated effective neutron multiplication factor is largest for the flooded case with the axial beryllium reflectors in place.

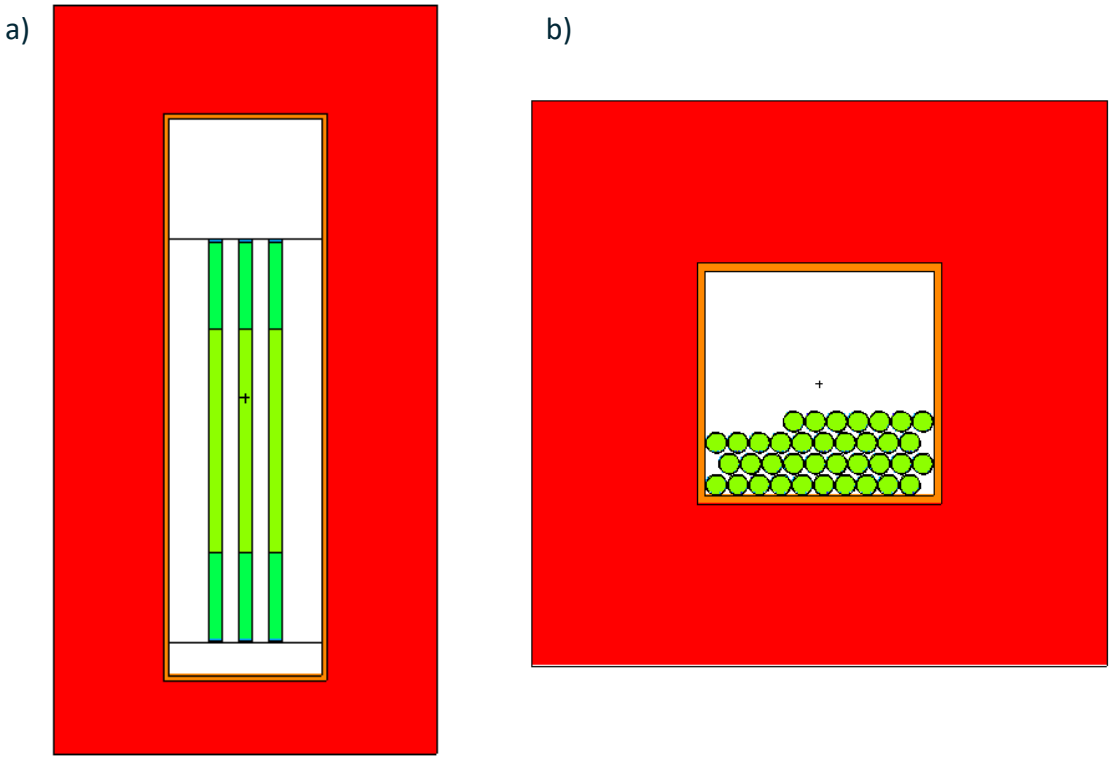

Figure 18. a) Axial cross section of MARVEL reactor fuel in the HLC and b) radial cross section of MARVEL reactor fuel aggregated in the HLC.

Table 14. Estimated effective neutron multiplication factor for aggregated MARVEL reactor fuel in the HLC.

Case	Effective Neutron Multiplication Factor ($k_{eff} \pm \sigma k_{eff}$)
M_HLC_D_Be_Clumped	0.72922 ± 0.00036
M_HLC_D_NoBe_Clumped	0.71086 ± 0.00037
M_HLC_W_Be_Clumped	0.95640 ± 0.00037
M_HLC_W_NoBe_Clumped	0.95114 ± 0.00038

Figure 19 shows the MARVEL reactor fuel in a circular arrangement inside the HLC at minimum pitch and maximum pitch. Figure 20 shows the MARVEL reactor fuel in a triangular arrangement inside the HLC at minimum and maximum pitch.

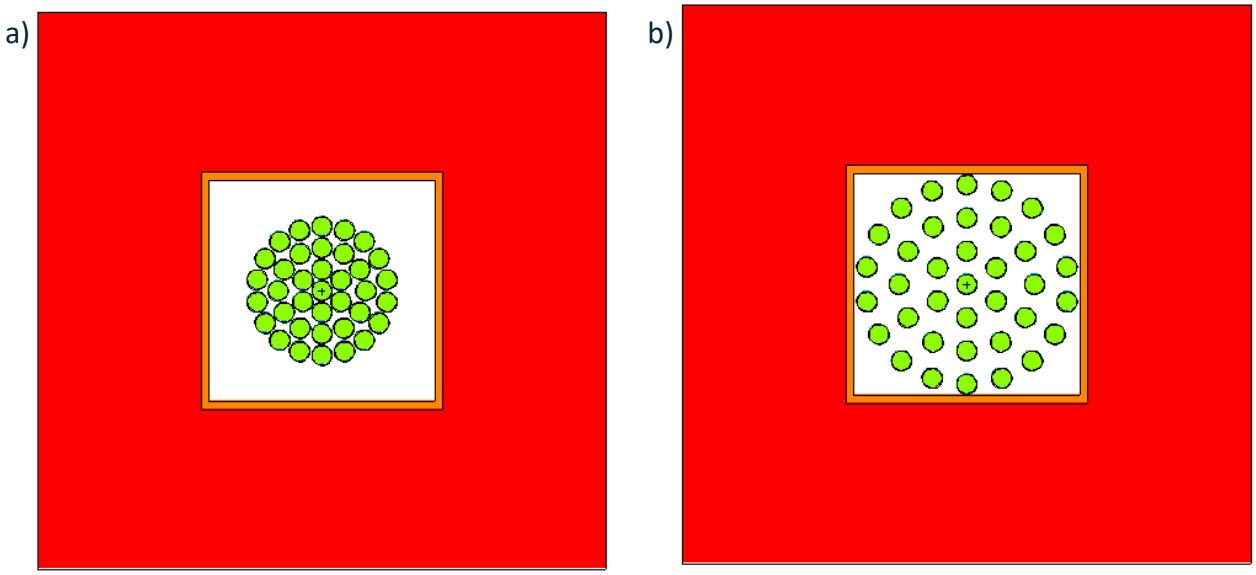

Figure 19. Radial cross sections of the MARVEL reactor fuel in the HLC with a circular arrangement at a) minimum pitch, and b) maximum pitch.

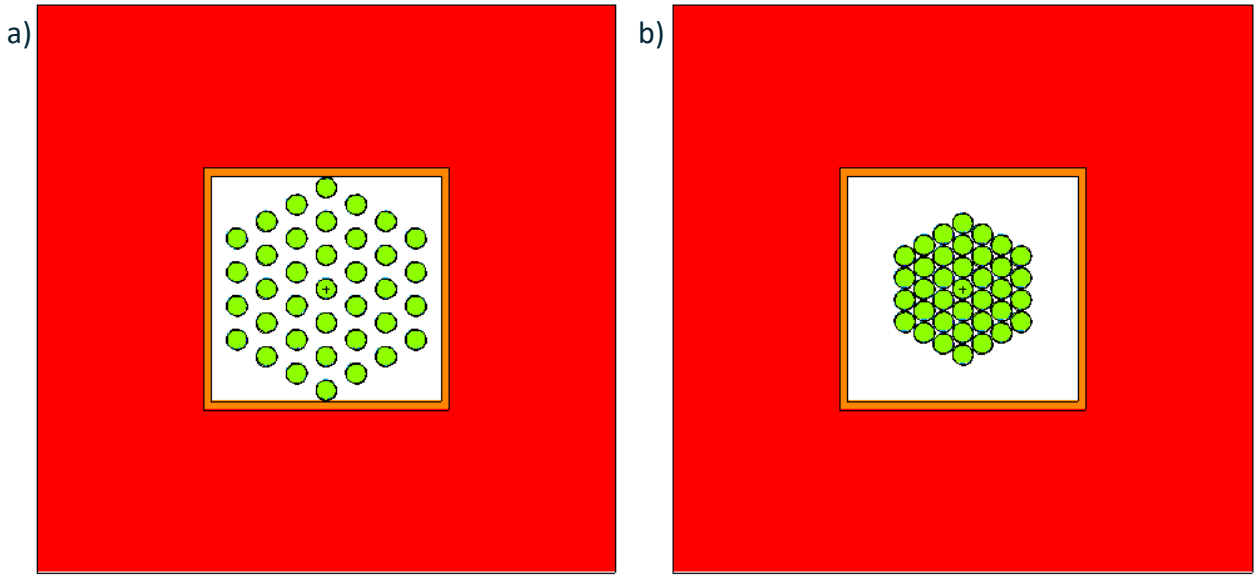

Figure 20. Radial cross sections of the MARVEL reactor fuel in the HLC with a triangular arrangement at a) minimum pitch, and b) maximum pitch.

Figure 21and Figure 22 show the estimated effective neutron multiplication factor values for the MARVEL reactor fuel in the HLC, in a circular and triangular arrangement, and under dry and wet conditions, respectively. Under dry conditions, the estimated effective neutron multiplication factor decreases monotonically with increasing pitch and is significantly below the assumed criticality safety limit of 0.93. Under wet conditions, the estimated effective neutron multiplication factor shows the effect of moderation from the water with lowest reactivity occurring at a pitch of approximately 1.66 cm for the triangular arrangement and

a pitch of approximately 2.57cm for the circular arrangement. All HLC scenarios under flooded conditions exceed the assumed criticality safety limit of 0.93.

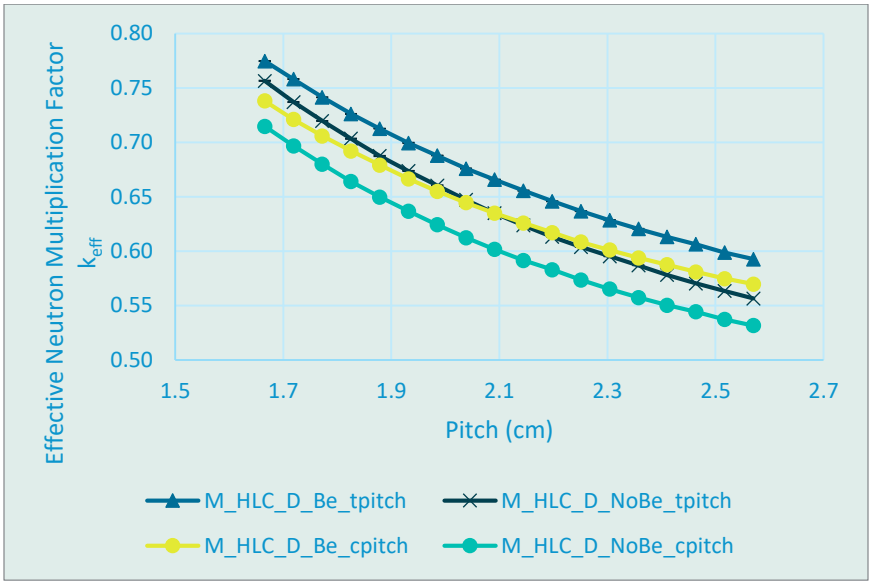

Figure 21. Estimated effective neutron multiplication factor for MARVEL reactor fuel in the HLC in circular and triangular arrangements under dry conditions.

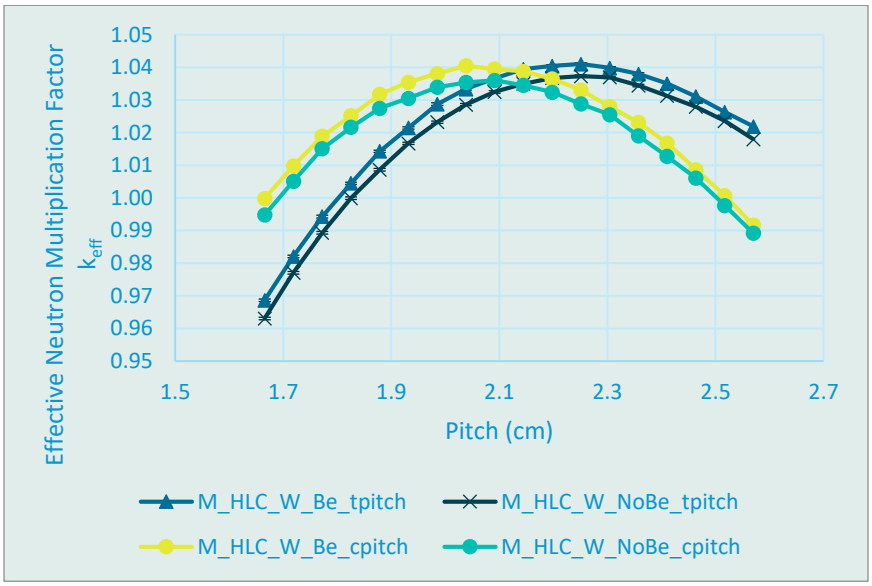

Figure 22. Estimated effective neutron multiplication factor for MARVEL reactor fuel in the HLC in circular and triangular arrangements under wet conditions.

7.1.4 Criticality of MARVEL Reactor Fuel in the Storage Canister

This section presents the results of the estimated effective neutron multiplication factor for the MARVEL reactor fuel loaded in the 18-in.-diameter storage canister in various configurations and scenarios. These canisters are used to store various types of SNF, including aluminum clad SNF, primarily ATR SNF. All effective

neutron multiplication factors are presented with associated statistical uncertainties. The ATR storage configurations feature ATR SNF in either the ATR4 bucket or ATR8 bucket at the bottom and upper sections of the canister, with a shorter top bucket reserved for shorter aluminum clad SNF. An option to store irradiated MARVEL reactor fuel with ATR SNF may be attractive as an efficient storage configuration. The MARVEL reactor fuel element can be placed in either the lower or upper sections. Figure 23 shows radial cross sections of the ATR fuel elements loaded in the ATR8 bucket and ATR4 bucket, respectively. Figure 24 shows an axial cross section of the MARVEL reactor fuel inside the storage canister and a radial cross section showing the MARVEL reactor fuel aggregated to one side of the storage canister. Figure 25 shows the possible configurations of irradiated MARVEL reactor fuel stored with ATR SNF. The scenarios involving the ATR4 bucket are chosen as bounding for the ATR8 buckets, and so the ATR8 geometries are not analyzed.

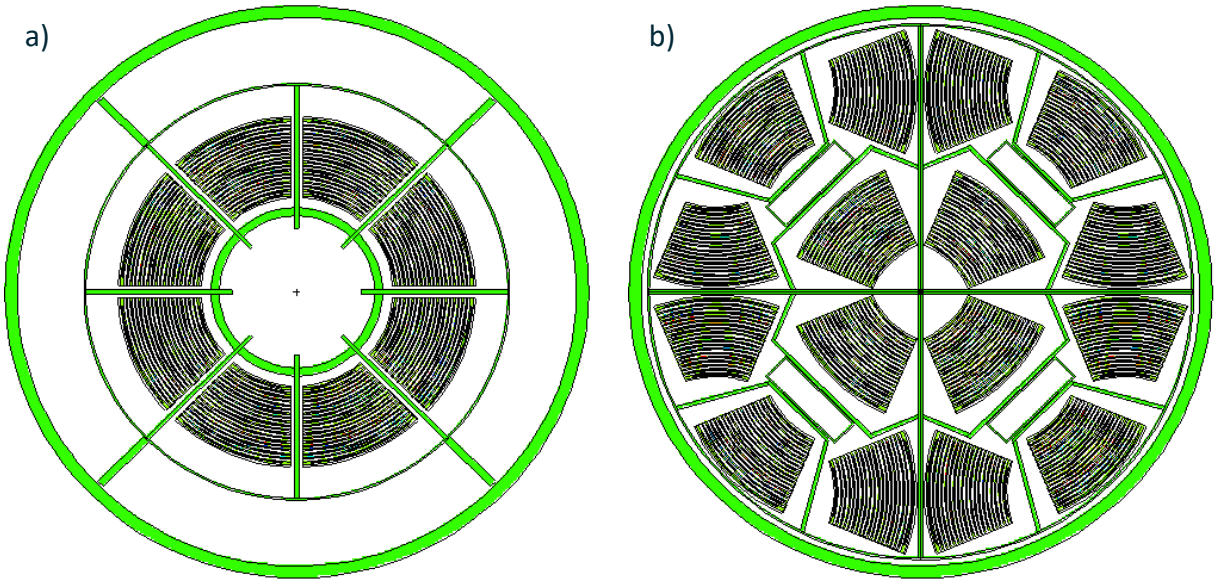

Figure 23. Radial cross section of ATR fuel elements in a) the ATR8 bucket and b) the ATR4 bucket.

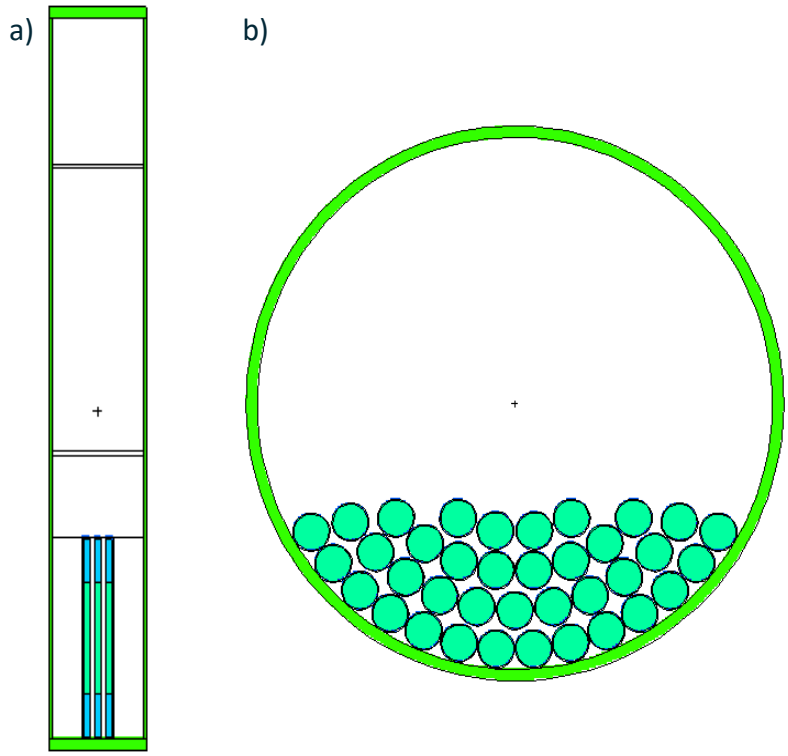

Figure 24. a) Axial cross section of MARVEL reactor fuel in the storage canister, and b) radial cross section of MARVEL reactor fuel aggregated in the storage canister.

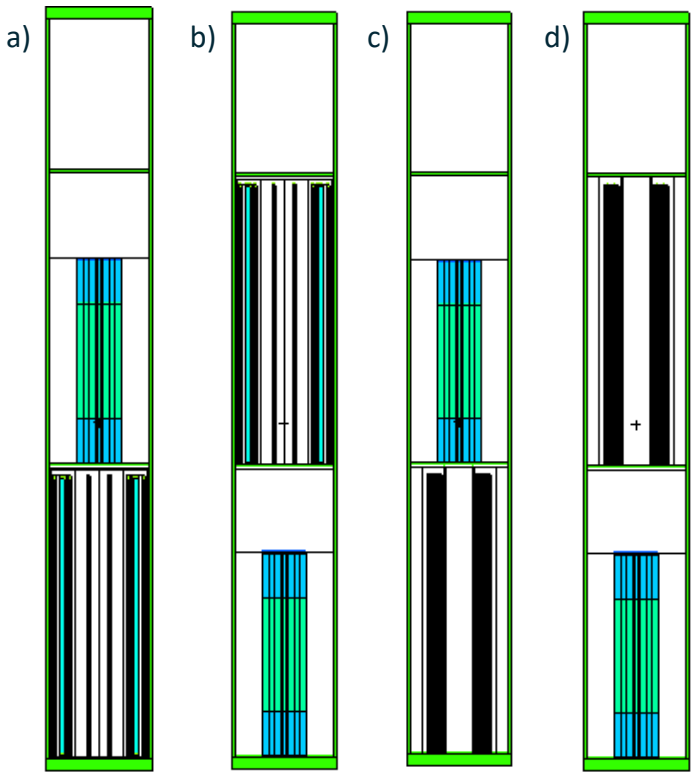

Figure 25. MARVEL reactor fuel in the 18-in. storage canister with a) ATR4 bucket in lower position, b) ATR4 bucket in upper position, c) ATR8 bucket in lower position, and d) ATR8 bucket in upper position.

Table 15 provides the estimated effective neutron multiplication factor results for aggregated MARVEL reactor fuel in the various storage canister configurations shown above. The estimated effective neutron multiplication factor for the MARVEL reactor fuel stored independently is significantly below the assumed criticality safety limit of 0.93 for dry and wet conditions. The estimated effective neutron multiplication factor for the MARVEL reactor fuel stored with the ATR SNF in the ATR4 bucket is significantly below the assumed criticality safety limit of 0.93 for dry conditions, but exceeds this limit under wet conditions. The estimated effective neutron multiplication factor is larger for the flooded cases with the axial beryllium reflectors in place.

Figure 26 and Figure 27 show the estimated effective neutron multiplication factor results for the MARVEL reactor fuel stored independently in the storage canister, in a circular and triangular arrangement, and under dry and wet conditions, respectively. Under dry conditions, the estimated effective neutron multiplication factor decreases monotonically with increasing pitch and is significantly below the assumed criticality safety limit of 0.93. Under wet conditions, the estimated effective neutron multiplication factor shows the effect of moderation from the water with lowest reactivity occurring at a pitch of approximately 3.31 cm for both the triangular and circular arrangements. At this pitch, the estimated effective neutron multiplication factor is below the assumed criticality safety limit of 0.93. The same trends are observed for MARVEL reactor fuel co-loaded with the ATR SNF, as shown in Figure 28 and Figure 29 for the ATR fuel in the lower position within the canister and Figure 30 and Figure 31 for the ATR fuel in the upper position within the canister.

Table 15. Estimated effective neutron multiplication factor for aggregated MARVEL reactor fuel in the storage canister.

Case	Effective Neutron Multiplication Factor ($k_{eff} \pm \sigma k_{eff}$)
M_CPP_D_Be_Clumped	0.49180 ± 0.00035
M_CPP_D_NoBe_Clumped	0.48467 ± 0.00037
M_CPP_W_Be_Clumped	0.83795 ± 0.00040
M_CPP_W_NoBe_Clumped	0.83584 ± 0.00038
M_ATR4B_D_Be_Clumped	0.49172 ± 0.00035
M_ATR4B_D_NoBe_Clumped	0.48479 ± 0.00037
M_ATR4B_W_Be_Clumped	0.94399 ± 0.00063
M_ATR4B_W_NoBe_Clumped	0.94286 ± 0.00083
M_ATR4T_D_Be_Clumped	0.49184 ± 0.00034
M_ATR4T_D_NoBe_Clumped	0.48576 ± 0.00035
M_ATR4T_W_Be_Clumped	0.83802 ± 0.00040
M_ATR4T_W_NoBe_Clumped	0.83584 ± 0.00038

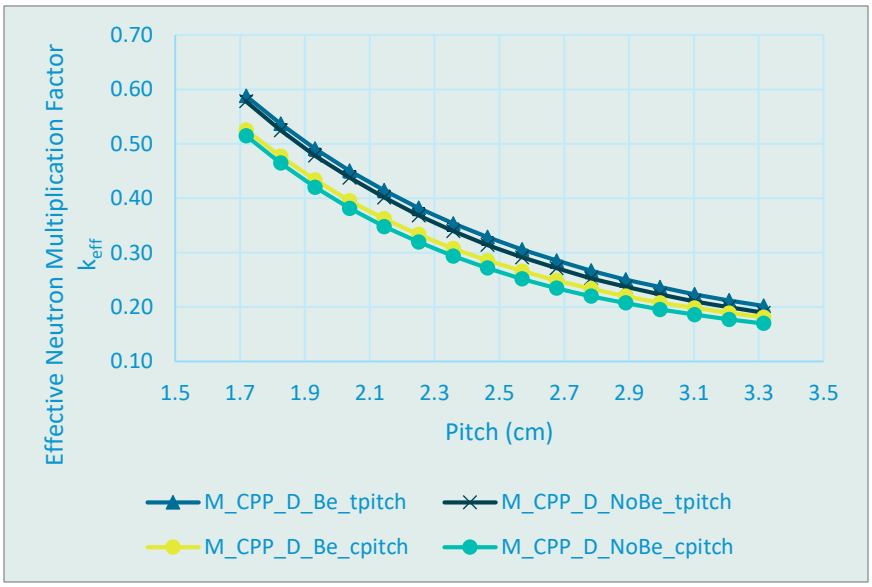

Figure 26. Estimated effective neutron multiplication factor for MARVEL reactor fuel in the storage canister in circular and triangular arrangements under dry conditions.

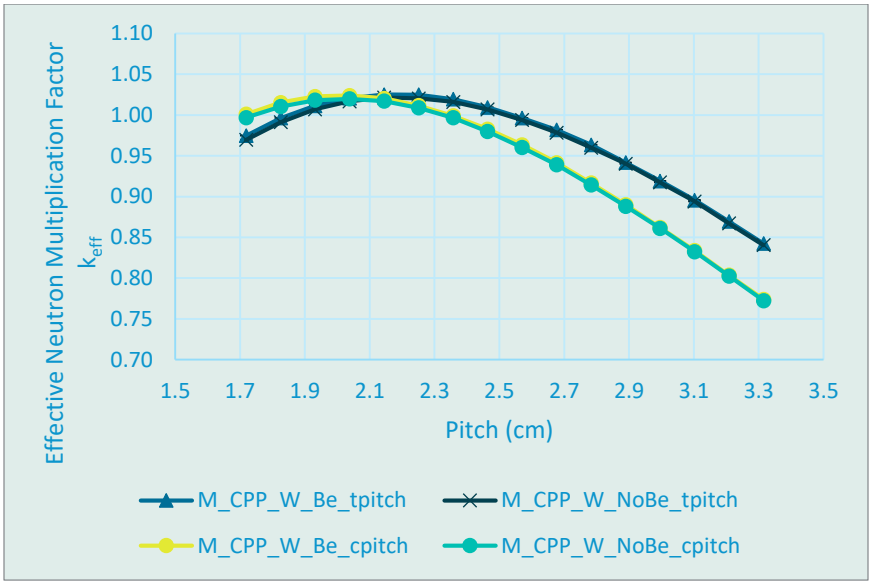

Figure 27. Estimated effective neutron multiplication factor for MARVEL reactor fuel in the storage canister in circular and triangular arrangements under wet conditions.

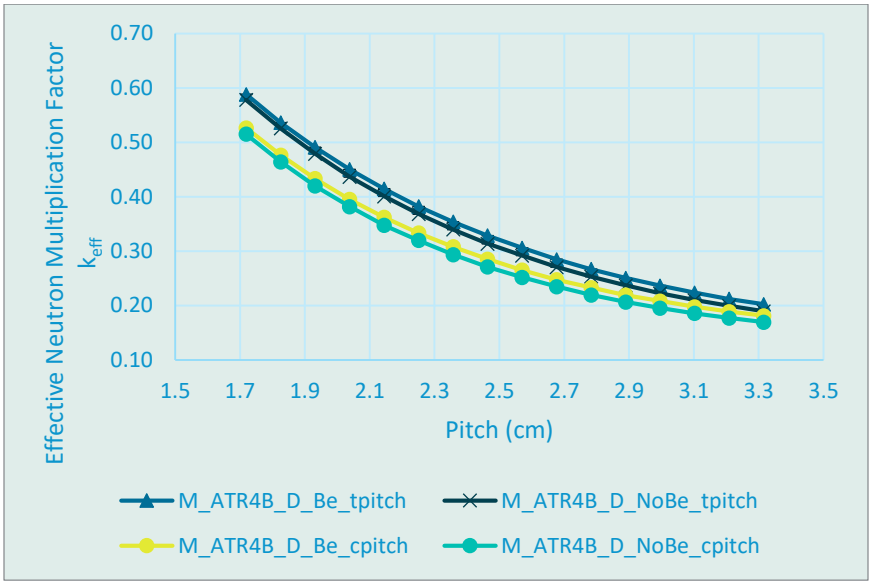

Figure 28. Estimated effective neutron multiplication factor for MARVEL reactor fuel in the storage canister with the ATR4 bucket in the lower location in circular and triangular arrangements under dry conditions.

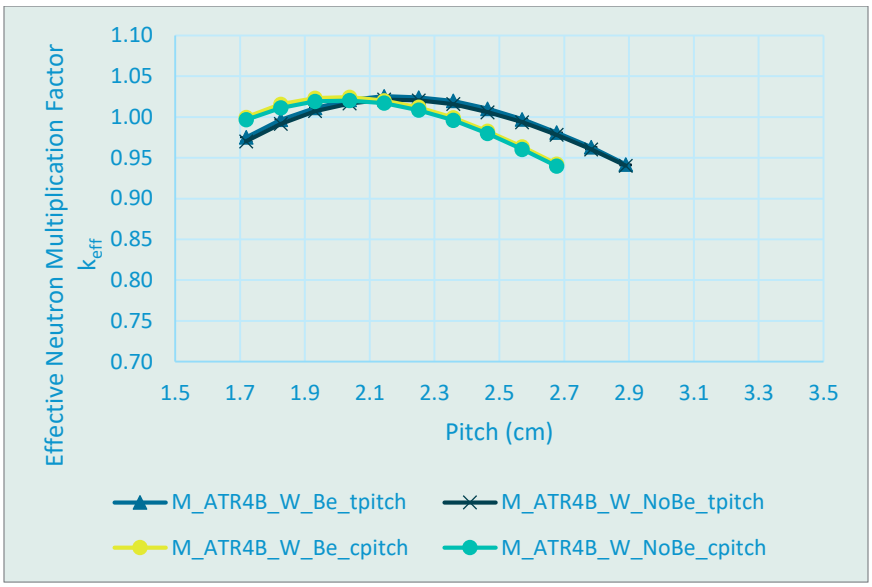

Figure 29. Estimated effective neutron multiplication factor for MARVEL reactor fuel in the storage canister with the ATR4 bucket in the lower location in circular and triangular arrangements under wet conditions.

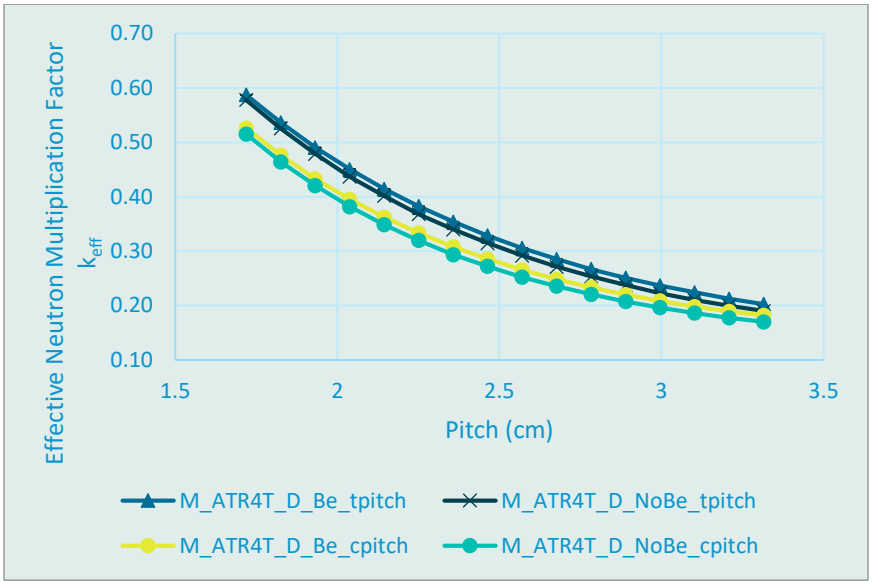

Figure 30. Estimated effective neutron multiplication factor for MARVEL reactor fuel in the storage canister with the ATR4 bucket in upper location in circular and triangular arrangements under dry conditions.

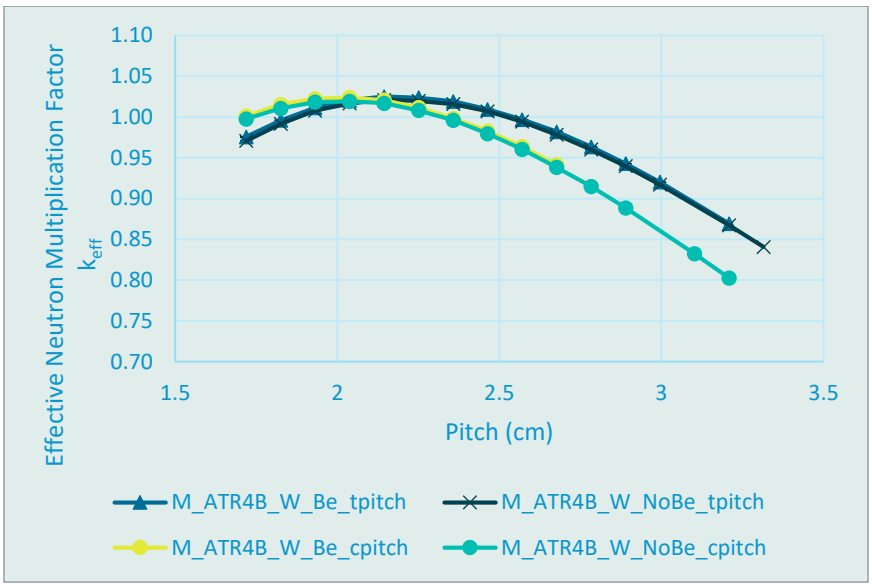

Figure 31. Estimated effective neutron multiplication factor for MARVEL reactor fuel in the storage canister with the ATR4 bucket in upper location in circular and triangular arrangements under wet conditions.

7.2 Radiation Shielding Assessment Results

SCALE 6.2.3 was used to generate the photon and neutron source spectra. Table 16 and Table 17 show the photon and neutron source spectra for a single MARVEL reactor fuel element, respectively. Using the source term generated in SCALE, the MCNP6 radiation transport code was used to estimate the total dose equivalent rate at various locations within the geometry. Dose equivalent rates were estimated on contact, and 1 m away from the unshielded MARVEL reactor fuel and on contact, and 1 m away from the various transfer cask geometries. The estimated total dose equivalent rate includes contributions from photon, neutron, and neutron-induced photon dose equivalent rate estimates. Figure 32 shows the axial heights for which radial contact and 1 m dose equivalent rates were estimated. Figure 33 shows angles at which circumferential dose equivalent rates were estimated for the aggregated configurations.

Table 16. Photon source term per assembly for irradiated MARVEL reactor fuel.

Group No.	Energy Group Upper Bound (MeV)	Energy Group Lower Bound MeV)	Photon Source Intensity (photons/s)
1	2.00E+01	1.00E+01	1.30E+01
2	1.00E+01	8.00E+00	1.78E+02
3	8.00E+00	6.50E+00	8.29E+02
4	6.50E+00	5.00E+00	4.25E+03
5	5.00E+00	4.00E+00	1.04E+04
6	4.00E+00	3.00E+00	3.68E+08
7	3.00E+00	2.50E+00	3.90E+09
8	2.50E+00	2.00E+00	1.74E+11
9	2.00E+00	1.66E+00	2.82E+10
10	1.66E+00	1.33E+00	3.97E+11
11	1.33E+00	1.00E+00	7.36E+11
12	1.00E+00	8.00E-01	1.10E+12
13	8.00E-01	6.00E-01	2.36E+13
14	6.00E-01	4.00E-01	4.41E+12
15	4.00E-01	3.00E-01	1.39E+12
16	3.00E-01	2.00E-01	1.96E+12
17	2.00E-01	1.00E-01	8.00E+12
18	1.00E-01	4.50E-01	8.95E+12
19	4.50E-02	1.00E-02	2.62E+13

Table 17. Neutron source term per assembly for irradiated MARVEL reactor fuel.

Group No.	Energy Group Upper Bound (MeV)	Energy Group Lower Bound MeV)	Neutron Source Intensity (neutrons/s)
1	2.00E+01	6.38E+00	4.59E+03
2	6.38E+00	3.01E+00	4.48E+04
3	3.01E+00	1.83E+00	5.12E+04
4	1.83E+00	1.42E+00	2.42E+04
5	1.42E+00	9.07E-01	3.49E+04
6	9.07E-01	4.08E-01	3.41E+04
7	4.08E-01	1.11E-01	1.57E+04
8	1.11E-01	1.50E-02	2.74E+03
9	1.50E-02	3.04E-03	1.34E+02
10	3.04E-03	5.83E-04	1.23E+01
11	5.83E-04	1.01E-04	1.05E+00
12	1.01E-04	2.90E-05	6.95E-02
13	2.90E-05	1.07E-05	9.79E-03
14	1.07E-05	3.06E-06	2.38E-03
15	3.06E-06	1.86E-06	2.27E-04
16	1.86E-06	1.30E-06	8.42E-05
17	1.30E-06	1.13E-06	2.32E-05
18	1.13E-06	1.00E-06	1.56E-05
19	1.00E-06	8.00E-07	2.29E-05
20	8.00E-07	4.14E-07	3.62E-05
21	4.14E-07	3.25E-07	6.53E-06
22	3.25E-07	2.25E-07	6.33E-06
23	2.25E-07	1.00E-07	6.05E-06
24	1.00E-07	5.00E-08	1.65E-06
25	5.00E-08	3.00E-08	4.82E-07
26	3.00E-08	1.00E-08	3.38E-07
27	1.00E-08	1.00E-11	8.06E-08

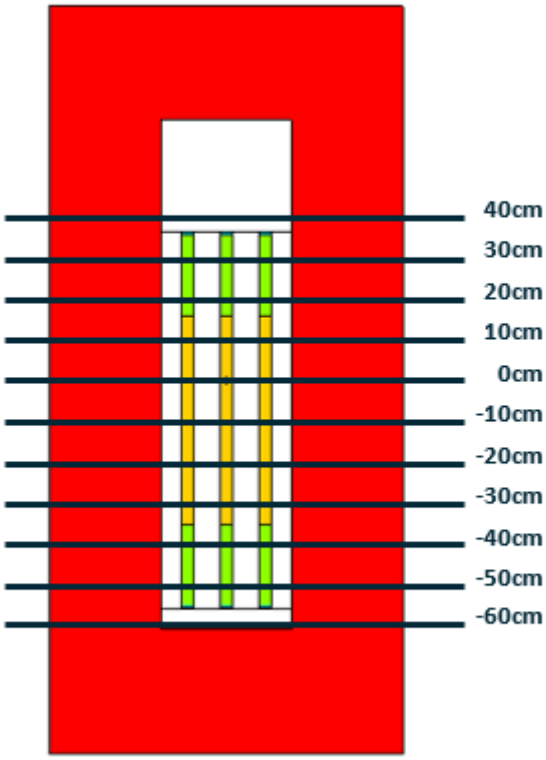

Figure 32. Axial heights for which radial contact and 1 m dose equivalent rates were estimated.

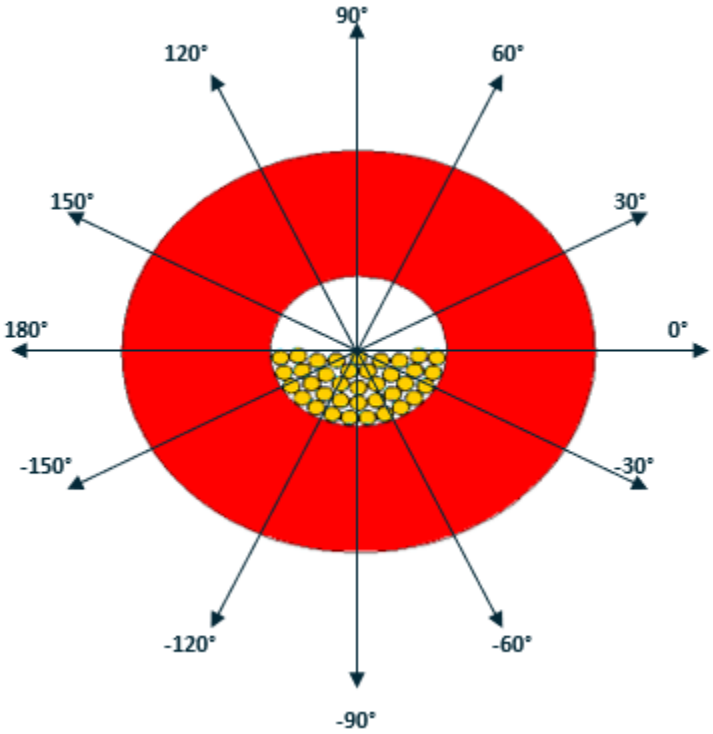

Figure 33. Angles at which circumferential dose equivalent rates were estimated for the aggregated configurations.

7.2.1 Dose Equivalent Rate of Unshielded MARVEL Reactor Fuel

This section presents the results of the dose rate equivalent rates for the unshielded MARVEL reactor fuel. All dose equivalent rates are presented in rem/hr with the associated statistical uncertainties. Table 18 shows the estimated dose equivalent rate on contact with the unshielded MARVEL reactor fuel. Table 19 shows the estimated dose equivalent rate 1 m away from the unshielded MARVEL reactor fuel. Figure 34 shows the estimated circumferential dose rate equivalent rates on contact from the unshielded MARVEL reactor fuel in the aggregated configuration.

Table 18. Estimated contact radial dose equivalent rate at various heights for unshielded MARVEL reactor fuel.

Estimate height (cm)	**Estimated contact dose equivalent rate (rem/hr)**			
	Photon	Neutron	Neutron Induced Photon	Total
-60	2.46E+03 (0.83%)	4.25E-03 (0.85%)	1.57E-04 (0.92%)	2.46E+03 (0.83%)
-50	8.49E+03 (0.68%)	1.73E-02 (0.59%)	4.51E-04 (0.74%)	8.49E+03 (0.68%)
-40	3.25E+04 (0.53%)	8.00E-02 (0.38%)	1.31E-03 (0.75%)	3.25E+04 (0.53%)
-30	9.80E+04 (0.45%)	2.39E-01 (0.29%)	3.47E-03 (0.61%)	9.80E+04 (0.45%)
-20	1.14E+05 (1.03%)	3.05E-01 (0.25%)	4.78E-03 (0.46%)	1.14E+05 (1.03%)
-10	1.15E+05 (1.65%)	3.19E-01 (0.25%)	5.23E-03 (1.68%)	1.15E+05 (1.65%)
0	1.11E+05 (0.62%)	3.05E-01 (0.26%)	4.82E-03 (0.51%)	1.11E+05 (0.62%)
10	9.78E+04 (0.41%)	2.37E-01 (0.29%)	3.48E-03 (0.54%)	9.78E+04 (0.41%)
20	3.34E+04 (2.86%)	7.97E-02 (0.38%)	1.29E-03 (0.60%)	3.34E+04 (2.86%)
30	8.48E+03 (0.62%)	1.72E-02 (0.62%)	4.55E-04 (0.86%)	8.48E+03 (0.62%)
40	2.41E+03 (0.90%)	4.23E-03 (0.81%)	1.53E-04 (0.97%)	2.41E+03 (0.90%)

Table 19. Estimated 1 m radial dose equivalent rate at various heights for unshielded MARVEL reactor fuel.

Estimate height (cm)	Estimated 1 m dose equivalent rate (rem/hr)			
	Photon	Neutron	Neutron Induced Photon	Total
-60	2.12E+03 (0.23%)	5.86E-03 (0.14%)	9.29E-05 (0.33%)	2.12E+03 (0.23%)
-50	2.29E+03 (0.23%)	6.43E-03 (0.14%)	1.01E-04 (0.33%)	2.29E+03 (0.23%)
-40	2.46E+03 (0.36%)	6.92E-03 (0.14%)	1.09E-04 (0.31%)	2.46E+03 (0.36%)
-30	2.59E+03 (0.31%)	7.32E-03 (0.14%)	1.15E-04 (0.30%)	2.59E+03 (0.31%)
-20	2.67E+03 (0.40%)	7.57E-03 (0.14%)	1.19E-04 (0.27%)	2.67E+03 (0.40%)
-10	2.70E+03 (0.31%)	7.65E-03 (0.14%)	1.20E-04 (0.26%)	2.70E+03 (0.31%)
0	2.69E+03 (0.39%)	7.56E-03 (0.14%)	1.20E-04 (0.44%)	2.69E+03 (0.39%)
10	2.61E+03 (0.43%)	7.32E-03 (0.14%)	1.15E-04 (0.31%)	2.61E+03 (0.43%)
20	2.46E+03 (0.31%)	6.91E-03 (0.14%)	1.09E-04 (0.29%)	2.46E+03 (0.31%)
30	2.31E+03 (0.45%)	6.42E-03 (0.14%)	1.02E-04 (0.39%)	2.31E+03 (0.45%)
40	2.13E+03 (1.06%)	5.86E-03 (0.14%)	9.27E-05 (0.27%)	2.13E+03 (1.06%)

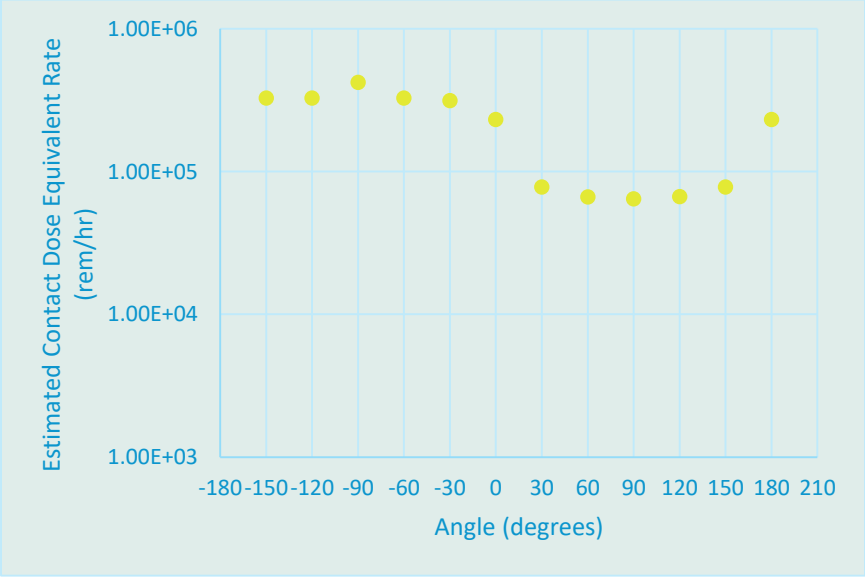

Figure 34. Estimated circumferential contact dose equivalent rate for the unshielded aggregated MARVEL reactor fuel.

7.2.2 Dose Equivalent Rate of MARVEL Reactor Fuel in the ATR Transfer Cask

This section presents the results of the dose rate equivalent rates for the MARVEL reactor fuel. loaded in the ATR transfer cask. All dose equivalent rates are presented in rem/hr with the associated statistical uncertainties. Table 20 shows the estimated dose equivalent rate on contact with the ATR transfer cask. Table 21 shows the estimated dose equivalent rate 1 m away from the ATR transfer cask. Figure 35 shows the estimated circumferential dose rate equivalent rates on contact with the ATR transfer cask with the MARVEL reactor fuel in the aggregated configuration.

Table 20. Estimated contact radial dose equivalent rate at various heights for MARVEL reactor fuel in the ATR transfer cask.

Estimate height (cm)	**Estimated contact dose equivalent rate (rem/hr)**			
	Photon	Neutron	Neutron Induced Photon	Total
-60	2.07E-18 (7.26%)	2.18E-02 (1.29%)	4.81E-06 (3.18%)	2.18E-02 (1.29%)
-50	2.09E-17 (11.35%)	3.51E-02 (0.96%)	8.50E-06 (3.25%)	3.51E-02 (0.96%)
-40	9.81E-17 (19.90%)	5.30E-02 (0.90%)	1.38E-05 (3.49%)	5.33E-02 (0.90%)
-30	1.82E-16 (15.87%)	7.18E-02 (0.87%)	1.87E-05 (1.84%)	7.23E-02 (0.87%)
-20	3.02E-16 (14.13%)	8.64E-02 (0.85%)	2.26E-05 (1.34%)	8.73E-02 (0.85%)
-10	2.83E-16 (10.94%)	9.19E-02 (0.85%)	2.53E-05 (5.33%)	9.27E-02 (0.85%)
0	3.43E-16 (12.64%)	8.65E-02 (0.85%)	2.44E-05 (6.85%)	8.75E-02 (0.85%)
10	3.28E-16 (24.21%)	7.22E-02 (0.87%)	1.82E-05 (1.49%)	7.31E-02 (0.91%)
20	1.25E-16 (21.95%)	5.27E-02 (0.89%)	1.29E-05 (1.78%)	5.30E-02 (0.90%)
30	3.65E-17 (15.80%)	3.51E-02 (0.94%)	8.05E-06 (2.28%)	3.52E-02 (0.94%)
40	2.30E-18 (10.13%)	2.19E-02 (1.02%)	4.91E-06 (3.72%)	2.19E-02 (1.02%)

Table 21. Estimated 1 m radial dose equivalent rate at various heights for MARVEL reactor fuel in the ATR transfer cask.

Estimate height (cm)	Estimated 1 m dose equivalent rate (rem/hr)			
	Photon	Neutron	Neutron Induced Photon	Total
-60	3.96E-17 (15.00%)	5.43E-03 (0.81%)	9.53E-07 (0.57%)	5.55E-03 (0.85%)
-50	2.77E-17 (15.88%)	5.93E-03 (0.81%)	1.03E-06 (0.43%)	6.01E-03 (0.83%)
-40	1.00E-16 (12.01%)	6.32E-03 (0.81%)	1.12E-06 (0.56%)	6.60E-03 (0.93%)
-30	1.40E-16 (15.87%)	6.63E-03 (0.81%)	1.17E-06 (0.62%)	7.02E-03 (1.18%)
-20	1.14E-16 (14.38%)	6.84E-03 (0.81%)	1.21E-06 (0.62%)	7.17E-03 (1.01%)
-10	1.38E-16 (17.19%)	6.92E-03 (0.80%)	1.22E-06 (0.47%)	7.31E-03 (1.20%)
0	1.18E-16 (13.35%)	6.86E-03 (0.80%)	1.21E-06 (0.42%)	7.20E-03 (0.98%)
10	1.09E-16 (11.03%)	6.67E-03 (0.80%)	1.18E-06 (0.54%)	6.98E-03 (0.91%)
20	1.07E-16 (17.76%)	6.38E-03 (0.80%)	1.13E-06 (0.95%)	6.69E-03 (1.11%)
30	8.25E-17 (16.55%)	5.98E-03 (0.81%)	1.05E-06 (0.62%)	6.22E-03 (1.00%)
40	1.31E-16 (50.19%)	5.51E-03 (0.81%)	9.56E-07 (0.41%)	5.88E-03 (3.27%)

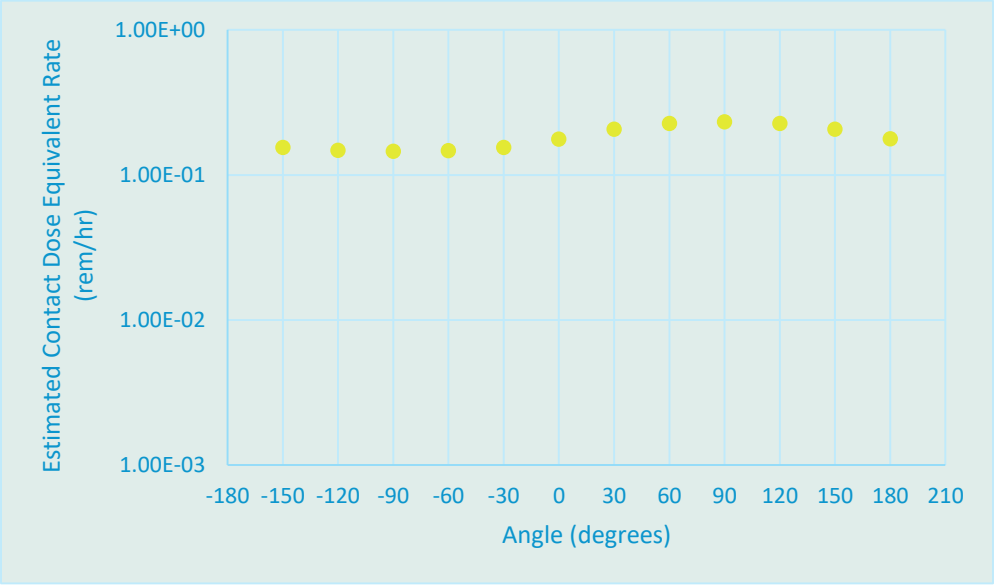

Figure 35. Estimated circumferential contact dose equivalent rate for the ATR transfer cask with aggregated MARVEL reactor fuel.

7.2.3 Dose Equivalent Rate of MARVEL Reactor Fuel in the HFEF-5 Transfer Cask

This section presents the results of the dose rate equivalent rates for the MARVEL reactor fuel loaded in the HFEF-5 transfer cask. All dose equivalent rates are presented in rem/hr with the associated statistical uncertainties. Table 22 shows the estimated dose equivalent rate on contact with the HFEF-5 transfer cask. Table 23 shows the estimated dose equivalent rate 1 m away from the HFEF-5 transfer cask. Figure 36 shows the estimated circumferential dose rate equivalent rates on contact HFEF-5 transfer cask with the MARVEL reactor fuel in the aggregated configuration.

Table 22. Estimated contact radial dose equivalent rate at various heights for MARVEL reactor fuel in the HFEF transfer cask.

Estimate height (cm)	Estimated contact dose equivalent rate (rem/hr)			
	Photon	Neutron	Neutron Induced Photon	Total
-60	6.19E-05 (19.52%)	1.96E-02 (1.10%)	3.55E-06 (3.58%)	1.96E-02 (1.10%)
-50	5.03E-04 (25.83%)	3.55E-02 (0.97%)	7.71E-06 (4.79%)	3.60E-02 (1.02%)
-40	2.21E-03 (20.31%)	5.86E-02 (0.90%)	1.33E-05 (2.06%)	6.08E-02 (1.14%)
-30	4.03E-03 (12.77%)	8.43E-02 (0.86%)	1.98E-05 (1.33%)	8.84E-02 (1.01%)
-20	5.98E-03 (10.95%)	1.04E-01 (0.84%)	2.59E-05 (1.91%)	1.10E-01 (0.99%)
-10	3.50E-02 (53.89%)	1.12E-01 (0.83%)	2.79E-05 (2.13%)	1.47E-01 (12.87%)
0	2.07E-02 (44.84%)	1.04E-01 (0.84%)	2.63E-05 (3.85%)	1.25E-01 (7.47%)
10	1.69E-02 (56.68%)	8.37E-02 (0.86%)	1.36E-05 (3.10%)	1.01E-01 (9.54%)
20	2.97E-03 (33.89%)	5.85E-02 (0.90%)	7.01E-06 (2.34%)	6.14E-02 (1.85%)
30	5.19E-04 (23.15%)	3.55E-02 (0.97%)	3.46E-06 (3.69%)	3.60E-02 (1.01%)
40	5.71E-05 (16.96%)	1.98E-02 (1.10%)	8.91E-07 (0.63%)	1.99E-02 (1.10%)

Table 23. Estimated 1 m radial dose equivalent rate at various heights for MARVEL reactor fuel in the HFEF-5 transfer cask.

Estimate height (cm)	Estimated 1 m dose equivalent rate (rem/hr)			
	Photon	Neutron	Neutron Induced Photon	Total
-60	7.81E-04 (11.76%)	5.69E-03 (0.78%)	9.76E-07 (0.42%)	6.47E-03 (1.58%)
-50	9.86E-04 (11.61%)	6.21E-03 (0.78%)	1.06E-06 (0.84%)	7.20E-03 (1.73%)
-40	9.76E-04 (8.33%)	6.65E-03 (0.78%)	1.11E-06 (0.61%)	7.62E-03 (1.26%)
-30	1.36E-03 (11.77%)	7.03E-03 (0.78%)	1.15E-06 (0.54%)	8.39E-03 (2.02%)
-20	1.58E-03 (14.94%)	7.25E-03 (0.78%)	1.17E-06 (0.44%)	8.84E-03 (2.75%)
-10	1.46E-03 (12.42%)	7.35E-03 (0.78%)	1.16E-06 (0.61%)	8.81E-03 (2.16%)
0	1.43E-03 (11.36%)	7.26E-03 (0.77%)	1.16E-06 (0.61%)	8.69E-03 (1.97%)
10	1.46E-03 (14.02%)	7.06E-03 (0.78%)	1.13E-06 (1.22%)	8.52E-03 (2.49%)
20	1.14E-03 (10.07%)	6.72E-03 (0.78%)	1.05E-06 (0.52%)	7.86E-03 (1.60%)
30	1.01E-03 (12.72%)	6.29E-03 (0.78%)	9.87E-07 (0.50%)	7.30E-03 (1.88%)
40	9.54E-04 (13.53%)	5.78E-03 (0.78%)	8.96E-07 (0.44%)	6.73E-03 (2.03%)

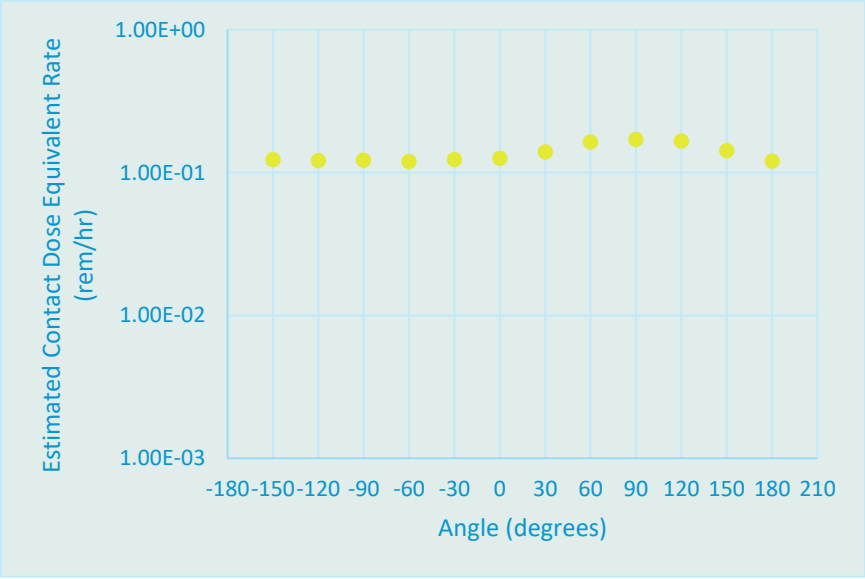

Figure 36. Estimated circumferential contact dose equivalent rate for the HFEF-5 transfer cask with aggregated MARVEL reactor fuel.

7.2.4 Dose Equivalent Rate of MARVEL Reactor Fuel in the High Load Charger

This section presents the results of the dose rate equivalent rates for the MARVEL reactor fuel loaded in the HLC. All dose equivalent rates are presented in rem/hr with the associated statistical uncertainties. Table 24 shows the estimated dose equivalent rate on contact with the HLC. Table 25 shows the estimated dose equivalent rate 1 m away from the HLC. Figure 37 shows the estimated circumferential dose rate equivalent rates on contact from the HLC with the MARVEL reactor fuel in the aggregated configuration.

Table 24. Estimated contact radial dose equivalent rate at various heights for MARVEL reactor fuel in the HLC.

Estimate height (cm)	**Estimated contact dose equivalent rate (rem/hr)**			
	Photon	Neutron	Neutron Induced Photon	Total
-60	2.60E-04 (10.89%)	3.90E-02 (1.49%)	7.78E-06 (8.75%)	3.93E-02 (1.48%)
-50	1.23E-03 (16.55%)	6.04E-02 (1.19%)	1.41E-05 (7.07%)	6.17E-02 (1.21%)
-40	4.15E-03 (8.36%)	8.77E-02 (1.06%)	2.11E-05 (5.24%)	9.19E-02 (1.08%)
-30	1.16E-02 (26.46%)	1.16E-01 (1.00%)	3.04E-05 (6.58%)	1.28E-01 (2.57%)
-20	1.19E-02 (8.14%)	1.39E-01 (0.95%)	3.37E-05 (4.35%)	1.51E-01 (1.08%)
-10	2.90E-02 (37.54%)	1.42E-01 (0.87%)	4.06E-05 (5.91%)	1.71E-01 (6.42%)
0	1.36E-02 (14.57%)	1.35E-01 (0.91%)	3.10E-05 (3.40%)	1.49E-01 (1.57%)
10	4.34E-02 (71.06%)	1.16E-01 (0.99%)	2.94E-05 (5.63%)	1.60E-01 (19.35%)
20	6.45E-03 (36.82%)	8.77E-02 (1.07%)	1.97E-05 (5.05%)	9.42E-02 (2.71%)
30	1.26E-03 (13.02%)	6.15E-02 (1.22%)	1.39E-05 (8.60%)	6.28E-02 (1.22%)
40	7.78E-04 (65.87%)	4.12E-02 (1.41%)	7.54E-06 (7.69%)	4.20E-02 (1.84%)

Table 25. Estimated 1 m radial dose equivalent rate at various heights for MARVEL reactor fuel in the HLC.

Estimate height (cm)	Estimated 1 m dose equivalent rate (rem/hr)			
	Photon	Neutron	Neutron Induced Photon	Total
-60	7.67E-05 (20.47%)	3.85E-03 (0.54%)	5.74E-07 (2.38%)	3.93E-03 (0.66%)
-50	1.19E-04 (23.19%)	4.19E-03 (0.53%)	6.48E-07 (2.33%)	4.31E-03 (0.82%)
-40	1.70E-04 (28.79%	4.49E-03 (0.53%)	6.83E-07 (4.39%)	4.66E-03 (1.17%)
-30	1.81E-04 (33.21%)	4.73E-03 (0.53%)	7.45E-07 (5.10%)	4.91E-03 (1.33%)
-20	1.78E-04 (22.56%)	4.89E-03 (0.53%)	7.52E-07 (1.90%)	5.07E-03 (0.94%)
-10	1.91E-04 (29.21%)	4.95E-03 (0.53%)	7.46E-07 (2.09%)	5.14E-03 (1.20%)
0	1.80E-04 (20.41%)	4.92E-03 (0.53%)	7.27E-07 (1.48%)	5.10E-03 (0.88%)
10	1.66E-04 (29.28%)	4.79E-03 (0.53%)	8.07E-07 (10.42%)	4.96E-03 (1.10%)
20	1.94E-04 (29.18%)	4.58E-03 (0.53%)	8.07E-07 (10.42%)	4.77E-03 (1.29%)
30	1.69E-04 (41.14%)	4.29E-03 (0.54%)	6.63E-07 (1.34%)	4.46E-03 (1.64%)
40	1.89E-04 (40.29%)	3.96E-03 (0.54%)	6.55E-07 (2.76%)	4.15E-03 (1.91%)

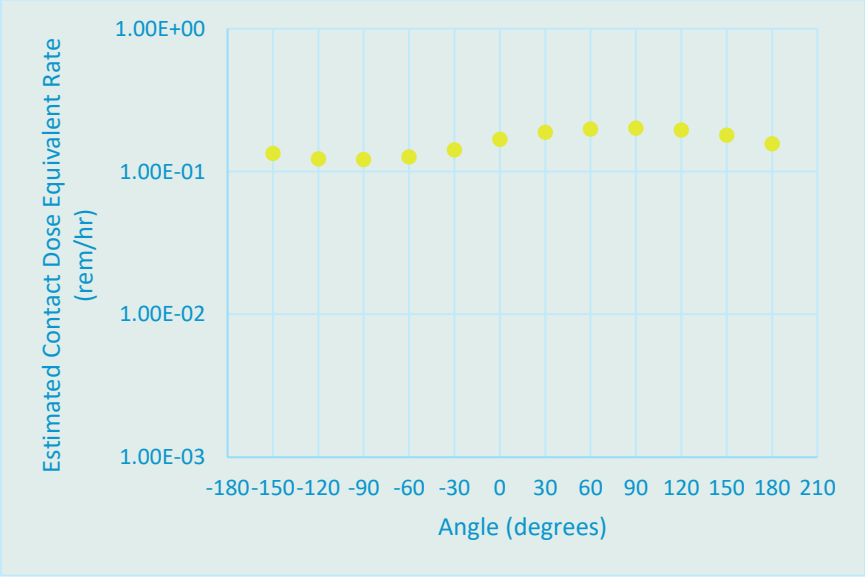

Figure 37. Estimated circumferential contact dose equivalent rate for the HLC with aggregated MARVEL reactor fuel.

8. CONCLUSIONS AND RECOMMENDATIONS

The ATR transfer cask, HFEF-5 transfer cask, and the HLC are used for the transfer of irradiated fuel between INL facilities. As such, these casks could potentially be used for the transfer of irradiated MARVEL reactor fuel between INL facilities. The criticality results show that under dry conditions, all configurations of 37 MARVEL reactor fuel elements loaded in the casks are subcritical and below the assumed limit of 0.93. For the ATR transfer cask under wet conditions, some configurations resulted in an estimated effective neutron multiplication factor that exceeded the assumed criticality safety limit of 0.93. All HFEF-5 transfer cask scenarios under flooded conditions exceed the assumed criticality safety limit of 0.93. All HLC scenarios under flooded conditions exceed the assumed criticality safety limit of 0.93. This suggests that administrative and engineering controls could be used, in addition to reducing the number of MARVEL reactor fuel elements per transfer, to ensure criticality safety under all transfer scenarios.

The estimated effective neutron multiplication factor for the MARVEL reactor fuel stored independently is significantly below the assumed criticality safety limit of 0.93 for dry and wet conditions. The estimated effective neutron multiplication factor for the MARVEL reactor fuel stored with the ATR SNF in the ATR4 bucket is significantly below the assumed criticality safety limit of 0.93 for dry conditions but exceeds this limit under wet conditions. This suggests that administrative and engineering controls could be used, in addition to reducing the number of MARVEL reactor fuel elements per storage canister, to ensure criticality safety under all storage scenarios.

The maximum estimated dose equivalent rates of 37 unshielded MARVEL reactor fuel element on contact is approximately 42,000 R/hr, resulting from the aggregated configuration. The ATR transfer cask reduced this dose equivalent rate to approximately 233 mR/hr on contact and 8 mR/hr, 1-m away from the cask surface. The HFEF-5 transfer cask reduced this dose equivalent rate to approximately 171 mR/hr on contact and 9 mR/hr, 1-m away from the cask surface. The HLC reduced this dose equivalent rate to approximately 201 mR/hr on contact and 6 mR/hr, 1-m away from the cask surface. This suggests that all three transfer casks analyzed can be expected to provide sufficient radiation shielding to workers during transfer of the MARVEL reactor fuel.

These calculations are performed to support the planning and strategy for the MARVEL project and will demonstrate the technical viability of the different configurations discussed and help identify where engineered or administrative controls may be necessary. A complete criticality safety analysis and radiation shielding analysis, including validation and contingency and accident analysis must be completed by licensed and authorized personnel before any transfer or storage of irradiated MARVEL reactor fuel.

9. REFERENCES

Christensen, A. B., “Estimate of Maximum ATR Fuel and Cask Temperature from a Heat Balance,” EDF-3510, April 2, 2003.

Morton, K., “Preliminary Design Specification for Department of Energy Standardized Spent Nuclear Fuel Canisters”, Idaho National Engineering and Environmental Laboratory. Volume I - Design Specification, 1998.

Rearden B. T. and Jessee M.A. (editors) “SCALE Code System” Version 6.2.3, Oak Ridge National Laboratory, ORNL/TM-2005/39, 2018.

Sentieri, P.J. and Casanova K. M., “Criticality Safety Evaluation for the Transport of HUP’s, UPS’s and Waste Containers in the HFEF-5 Cask” EDF-6631, 2019.

Stuart III C. E., “Criticality Safety Evaluation for the ATR Fuel Elements within the High Load Charger Cask” ECAR-4031, 2018.

Werner, C.J., Bull J.S., Solomon C.J., Brown F.B., McKinney G.W., Rising M.E., Dixon D.A., Martz R.L., Hughes H.G., Cox L.J., Zukaitis A. J., Armstrong J.C., Forster R. A., and Casswell L. "MCNP6.2 Release Notes", Los Alamos National Laboratory, LA-UR-18-20808, 2018.

Werner C.J. (editor), "MCNP User’s Manual - Code Version 6.2", Los Alamos National Laboratory, report LA-UR-17-29981, 2017.

Drawing 154872, “CPP-603 Phase II – BLDG 603 Irradiated Fuels Storage Facility (HTGR) Fuel Element Carbon Steel Storage Can CAN-GSF-101,” Rev 6, FDC XXXXXX, Idaho Cleanup Project Core, March 27, 2019

Drawing 155808, “BLDG 603 Irradiated Fuels Storage Facility (HTGR) Fuel Element Storage Can-Dust Tight Lid,” Rev 3, FDC XXXXXX, Idaho Cleanup Project Core, March 27, 2019.