FCT Quality Assurance Program Document

Appendix E
FCT Document Cover Sheet

Name/Title of Deliverable/Milestone	Assess impact of source term relocation for meeting transportation regulatory requirements
Work Package Title and Number	ST—Transportation – ORNL FT-13OR081309
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(Participant/National Laboratory Name)

QA program which meets the requirements of

☒ DOE Order 414.1 ☐ NQA-1-2000

This Deliverable was subjected to:

☒ Technical Review ☐ Peer Review

Technical Review (TR)	**Peer Review (PR)**
Review Documentation Provided	**Review Documentation Provided**
☐ Signed TR Report or,	☐ Signed PR Report or,
☐ Signed TR Concurrence Sheet or,	☐ Signed PR Concurrence Sheet or,
☐ Signature of TR Reviewer(s) below	☐ Signature of PR Reviewer(s) below

Name and Signature of Reviewers

Kaushik Banerjee

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Initial Assessment of Source Term Redistribution for Meeting Transportation Regulatory Requirements

Prepared for
U.S. Department of Energy
Used Fuel Disposition Campaign

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September 4, 2013

ORNL/LTR-2013/286

EXECUTIVE SUMMARY

This technical letter report documents work performed supporting the Department of Energy Office of Nuclear Energy Fuel Cycle Technologies Used Fuel Disposition Campaign under work breakdown structure element 1.02.08.13, *ST Transportation*. In particular, this report fulfills the M4 milestone M4FT-13OR0813092, "Assess impact of source term relocation for meeting transportation regulatory requirements."

Used nuclear fuel is being stored at reactor sites for longer time intervals than originally foreseen. As storage times continue to increase, issues associated with implementing aging management strategies and the potential consequences of the deleterious effects of aging need to be assessed. The assessments are not only to demonstrate the continued efficacy of the storage system over extended storage periods but also to ensure safety for subsequent transportation following extended storage.

This letter report documents a shielding analysis of the HI-STAR 100 transportation package assuming fuel assembly reconfiguration that may be caused by fuel assembly failure during normal, off-normal, and accident storage/transportation conditions. The study analyzed 32 multipurpose canisters containing fuel assemblies with a maximum burnup value between 40 to 55.5 GWd/MTU and decay heat between 690 W to 1 kW at the time of loading to determine the timeframe for transporting used nuclear fuel assemblies based on compliance with the regulatory shielding requirements in 10 CFR 71 for transportation packages. The analysis used conservative source terms and fuel assembly models to account for the impact of fuel reconfiguration. Based on the shielding analysis, the fuel assemblies would require a total of 26 to 43 years of storage prior to be relocated to another facility. The decay heat of the fuel assemblies at the evaluated time of shipment was approximately 440 W, which is lower than the assembly decay limit of 625 W established for the HI-STAR 100 transportation cask based on intact fuel assembly configurations. On average, the effects of fuel reconfiguration have the potential to result in an extension of the storage time of approximately 22 years before being able to meet the 10 CFR 71 transportation requirements for shielding.

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ACRONYMS

ARP	automatic rapid processing
CFR	Code of Federal Regulations
DOE	U.S. Department of Energy
GWd	gigawatt-day
HAC	hypothetical accident conditions
MTU	metric tons of uranium
MPC	multipurpose canister
NCT	normal conditions of transport
NRC	U.S. Nuclear Regulatory Commission
ORIGEN	Oak Ridge Isotope Generation Code
SCALE	Standardized Computer Analyses for Licensing Evaluation
UNF	used nuclear fuel
W	Westinghouse

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1 INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) 10 CFR Part 51 Waste Confidence Decision Update (Ref. 1) states that commercial used nuclear fuel (UNF) assemblies "can be stored safely without significant environmental impacts for at least 60 years beyond the licensed life for operation (which may include the term of a revised or renewed license) of that reactor in a combination of storage in its spent fuel storage basin and either onsite or offsite independent spent fuel storage installations." Large capacity canisters may require storage for decades before being relocated to an interim storage facility because the thermal constraints on transportation packages are more stringent than those on storage casks. Used nuclear fuel in dry storage at independent spent fuel storage installations of the shut-down and operational nuclear power plants may be relocated to an interim storage facility starting in 2021 as a pilot basis and followed by a larger storage facility in 2025 (Ref. 2).

Several phenomena have been identified, such as hydride reorientation and delayed hydride cracking (Ref. 3) that could cause failure of high-burnup fuel assemblies under normal, off-normal, and accident conditions of storage and normal and hypothetical accident conditions of transportation. Insufficient data currently exist to characterize the behavior of high-burnup assemblies in dry storage over a long period of time. Depending on the severity, fuel failure and reconfiguration can have significant safety implications.

This letter report documents a shielding analysis of transportation packages assuming fuel assembly reconfiguration that may be caused by fuel assembly failure during normal, off-normal, and accident storage/transportation conditions. The purpose of the analysis is to determine the decay time that would be necessary to establish transportation package compliance with the 10 CFR 71.47 and 71.51 dose rate requirements for normal conditions of transport (NCT) and hypothetical accident conditions (HAC), respectively, assuming fuel reconfiguration during transportation.

The characteristics of the representative fuel assemblies considered in the shielding analysis are described in Section 2. Regulatory requirements with respect to the shielding of transportation packages are described in Section 3 followed by the transportation package model used in the shielding analysis in Section 4. The computer codes used in the analysis are presented in Section 5. The calculation method is described in Section 6, the results of the shielding analysis are provided in Section 7, and conclusions are presented in Section 8.

2 CASK LOADING DATA

The representative dry storage system type considered is the HI-STORM 100 dry system, which consists of a multipurpose canister (MPC)-32 and a HI-STORM 100S Version B overpack. The transportation cask system for the MPC-32 casks is Holtec's HI-STAR 100 Cask System (Ref. 4), which has been approved by the NRC as an acceptable Type B(U)F-85 packaging for transport by exclusive use shipment (Ref. 5). The approved maximum assembly average decay heat and burnup value for the MPC-32 PWR intact fuel with zircaloy clad and with zircaloy in-core grid spacers are 625 W and 44.5 GWd/MTU, respectively (Ref. 5). For this study, it is assumed that the HI-STAR 100 Cask System can be loaded with UNF assemblies of decay heat up to 625 W and burnup values beyond the current 44.5 GWd/MTU limit. The basis for expanding assembly average burnup limit for the transportation package is the recommendations in NRC Interim Staff Guidance – 8 Revision 3 (Ref. 6) that extend the assembly-average burnup value to 60 GWd/MTU for fuel irradiated in a PWR and cooled out-of-reactor for a time period up to 40 years. This study analyzed 32 MPCs containing fuel assemblies with a maximum burnup between 40 to 55.5 GWd/MTU and decay heat between 690 W to 1 kW at the time of loading. Hence, the fuel assemblies in dry storage would require additional decay time to reach the 625-W decay heat limit per assembly established for the HI-STAR 100 transportation package. Package safety requirements with respect to shielding may further extend the storage time period.

Each of the 32 MPC locations may contain either one fuel assembly or one fuel assembly and a fuel-related component, such as burnable poison rod assemblies, wet annular absorber assemblies, rod cluster control assemblies, and thimble plug devices. Typical data provided by a storage facility for each UNF assembly in a storage cask include MPC cell location and region, assembly identifier as used in the RW859 database (Ref. 7), power plant unit and cycle in which the fuel assembly has been irradiated, assembly average burnup, discharge date, heat load and cool years at the time of loading. Similar data are provided for the fuel-related component that may reside in the same cask cell as a fuel assembly. Typically, the fuel-related component have small heat load (e.g., ~50 W) at the time of loading.

3 REGULATORY REQUIREMENTS WITH RESPECT TO SHIELDING FOR TRANSPORTATION PACKAGES

Transportation packages comply with the dose rate limits in 10 CFR § 71.47 and 71.51.

Prior to each shipment of licensed material, a licensee must determine that the package with its contents satisfies the applicable requirements of 10 CFR 71. As part of routine determinations (10 CFR § 71.87), "the licensee shall determine that - -

(j) External radiation levels around the package and around the vehicle, if applicable, will not exceed the limits specified in § 71.47 at any time during transportation."

10 CFR § 71.47 external radiation standards for packages transported by exclusive use shipment:

"(b) A package that exceeds the radiation level limits specified in paragraph (a) of this section must be transported by exclusive use shipment only, and the radiation levels for such shipment must not exceed the following during transportation:

(1) 2 mSv/h (200 mrem/h) on the external surface of the package, unless the following conditions are met, in which case the limit is 10 mSv/h (1000 mrem/h):

  (i) The shipment is made in a closed transport vehicle;

  (ii) The package is secured within the vehicle so that its position remains fixed during transportation; and

  (iii) There are no loading or unloading operations between the beginning and end of the transportation;

(2) 2 mSv/h (200 mrem/h) at any point on the outer surface of the vehicle, including the top and underside of the vehicle; or in the case of a flat-bed style vehicle, at any point on the vertical planes projected from the outer edges of the vehicle, on the upper surface of the load or enclosure, if used, and on the lower external surface of the vehicle; and

(3) 0.1 mSv/h (10 mrem/h) at any point 2 meters (80 in) from the outer lateral surfaces of the vehicle (excluding the top and underside of the vehicle); or in the case of a flat-bed style vehicle, at any

point 2 meters (6.6 feet) from the vertical planes projected by the outer edges of the vehicle (excluding the top and underside of the vehicle); and

(4) 0.02 mSv/h (2 mrem/h) in any normally occupied space, except that this provision does not apply to private carriers, if exposed personnel under their control wear radiation dosimetry devices in conformance with 10 CFR 20.1502."

10 CFR § 71.51, Additional requirements for Type B packages:

Under the tests specified in:

"(2) Section 71.73 ("Hypothetical accident conditions"), there would be no escape of krypton-85 exceeding 10 A_2 in 1 week, no escape of other radioactive material exceeding a total amount A_2 in 1 week, and no external radiation dose rate exceeding 10 mSv/h (1 rem/h) at 1 m (40 in) from the external surface of the package."

4 DESCRIPTION OF THE TRANSPORTATION PACKAGE MODEL

This section describes the transportation package model and the assembly fuel characteristics used in the shielding analysis.

4.1 Transportation Package Model

Conservative package models were used in this study to account for the uncertainty associated with failed fuel configurations in dry storage casks/transportation packages. Previous shielding analysis of transportation packages containing failed fuel (Ref. 8) has identified two models that provide significant conservatism with respect to dose rate at the package top and bottom surfaces as well as at the package side surface above and below the neutron shield. The conservative models are:

a. Homogenized basket plate material and fuel rubble packed closely together into the bottom of the inner cavity assuming a 0.58 mass packing fraction based on powder mechanics (Ref. 9);
b. Homogenized basket plate material and fuel rubble assuming the homogenized mixture occupies the whole inner cavity.

Configuration (a) is consistent with a package/storage cask in the vertical orientation. Configuration (b) is consistent with a package/storage cask in the horizontal orientation. A transportation package may have both vertical and horizontal orientations during normal operations. These configurations are bounding for fuel failure scenarios resulting in fuel fragments and particulates being collected into the inner cavity regions below or above the assembly spacers or between the fuel basket outer plates and the canister radial wall. Between the two configurations, Configuration (a) generates larger dose rates at the bottom surface and Configuration (b) generates larger dose rates at the top and side surfaces. The models use homogenized radiation sources that include: (1) gamma and neutron radiation that originate in the irradiated fuel; and (2) activation gamma source in non-fuel assembly materials.

The calculation models were developed for the HI-STAR 100 transportation package based on the MPC and overpack descriptions available in Ref. 4. The impact limiters were not modeled. However, their geometric characteristics were used to establish the dose rate locations. The NCT package models with (a) fuel rubble collapsed to cavity bottom and (b) homogenized fuel rubble within the cask cavity are illustrated in Figure 1 (a) and (b), respectively. For the HAC dose rate calculations, the outer neutron shield was replaced with air.

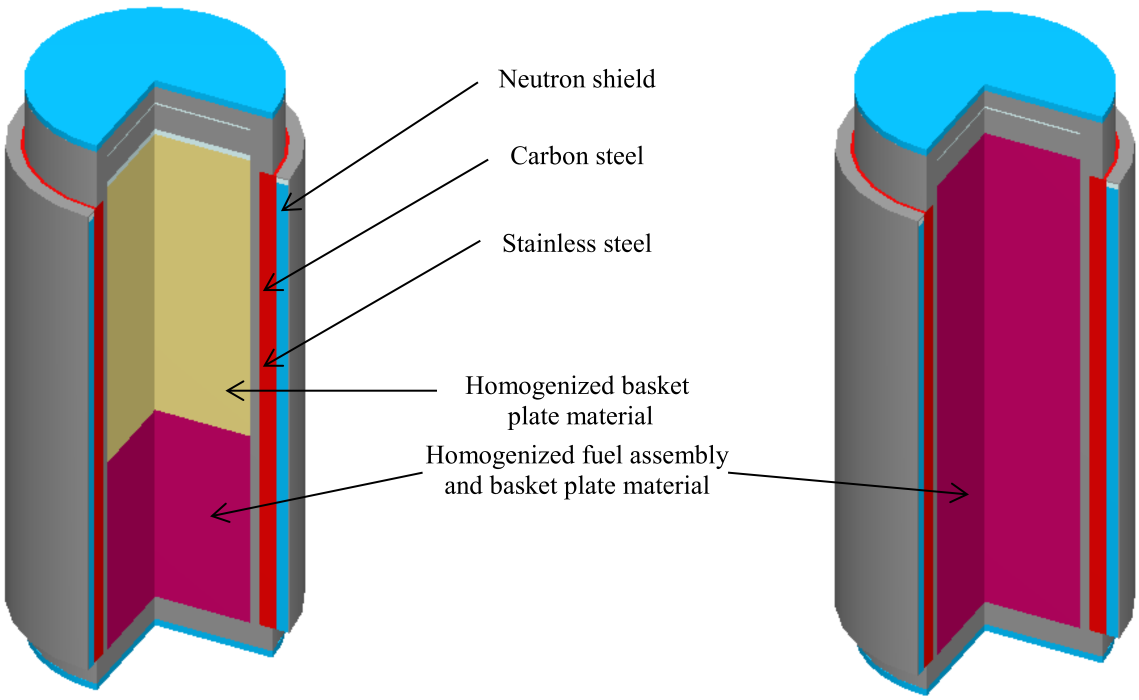

Figure 1. Transportation package models with (a) fuel rubble collapsed to cavity bottom and (b) homogenized fuel rubble within the cask cavity.

4.2 Fuel Assembly Characteristics

The fuel assembly types considered include Westinghouse (W) 17 × 17 STD, W 17 × 17 V5H, and Babcock & Wilcox 17 × 17 MkBW. Fuel assembly characteristics for depletion and shielding calculations were based on the W 17 × 17 STD assembly type (identified with the assembly type code W1717WL in the RW-859 database) because the W 17 × 17 V5H, W 17 × 17 STD, and 17 × 17 MkBW assemblies have similar neutronic characteristics. The W 17 × 17 STD fuel assembly characteristics (Refs. 10 and 11) are summarized in Table 1.

Table 1. Generic assembly design data for W 17 × 17 STD

Fuel assembly parameter	Value
Fuel assembly array size	17 × 17
Number of fuel rods per assembly	264
Typical mass of uranium in assembly (kg)	459
Active length	365.76
Fuel pellet diameter (cm)	0.81915
Clad inner diameter (cm)	0.83566
Clad outer diameter (cm)	0.94996
Clad material (cm)	Zircaloy-4
Number of guide tubes per assembly	24
Guide tube material	Zircaloy-4
Guide tube outer diameter (cm)	1.22428
Guide tube inner diameter (cm)	1.14300
Number of instrument tubes per assembly	1
Instrument tube material	Zircaloy-4
Instrument tube outer diameter (cm)	1.22428
Instrument tube inner diameter (cm)	1.14300
Upper end fitting length (cm)	15.506
Fuel plenum length (cm)	14.656
Lower end fitting length (cm)	11.951
Stainless steel (kg) in upper end fitting	6.89
Stainless steel (g) in upper fuel plenum and upper end spacer	91
Stainless steel (kg) in lower end fitting	5.9
Inconel-718 (kg) in upper end fitting	0.96
Inconel-718 (kg) in upper fuel plenum and upper end spacer	0.793

An MPC contains fuel assemblies of varying burnup values and cooling times. The fuel assembly with the largest decay heat typically has fairly high burnup and lowest decay time among the loaded fuel assemblies. For conservatism in the dose rate calculation, an MPC was assumed to contain 32 assemblies with the largest burnup value and lowest decay time among the loaded fuel assemblies. The initial enrichment, burnup, and decay time of the fuel assemblies used in the shielding analysis are tabulated in Table 2. The fuel assembly characteristics are representative of the fuel assemblies currently in dry storage at independent spent fuel storage installations.

Table 2. Fuel assembly irradiation and decay characteristics for depletion/decay calculations

Cask No.	Fuel initial ^{235}U enrichment	Assembly burnup (MWd/MTU)[a]	Assembly type	Decay time (years)[b]	Discharge year
1	3.6	44613	STD	12.62	1991
2	3.6	44827	STD	12.62	1991
3	3.6	44750	STD	12.62	1991
4	3.6	42432	V5H	11.24	1994
5	3.6	44805	STD	12.48	1993
6	3.6	42403	V5H	11.24	1994
7	3.6	42542	V5H	11.24	1994
8	3.6	44880	STD	12.48	1993
9	3.6	43036	V5H	9.86	1997
10	3.6	44789	V5H	8.4	1998
11	3.6	44790	V5H	8.4	1998
12	3.6	44828	V5H	8.4	1998
13	3.6	44749	V5H	9.33	1997
14	3.6	44313	STD	13.82	1993
15	3.6	44987	V5H	11.5	1997
16	3.6	44953	V5H	9.5	1999
17	3.6	43558	V5H	9.5	1999
18	3.6	42367	STD	10.1	1998
19	3.6	44826	STD	12.5	1996
20	3.6	40408	V5H	8.6	2000
21	3.6	48411	V5H	9.9	2000
22	3.6	48976	STD	10	2000
23	3.6	47772	STD	12.3	1997
24	3.6	52328	STD	16.6	1994
25	3.6	46211	V5H	7.8	2002
26	3.8	51006	STD	10	2000
27	3.8	55486	V5H	12.5	1999
28	3.8	55527	V5H	12.5	1999
29	3.8	55466	V5H	12.5	1999
30	3.8	52827	V5H	10	2001
31	3.8	52637	V5H	10	2001
32	3.8	52796	V5H	10	2001

[a]Largest assembly burnup per MPC; burnup value includes a 5% increase that accounts for burnup uncertainty.
[b]Decay time of the assembly with largest decay heat per MPC at time of loading.

5 COMPUTER CODES

The radiation source terms for the shielding analysis were determined with the SCALE 6.1.2 code system (Ref. 12) depletion and decay capabilities, ORIGEN-S (Ref. 12, Sect. F07) and ARP (Automatic Rapid Processing) (Ref. 12, Sect. D01). The neutron and gamma radiation source terms were calculated in the group structure of the SCALE 27N-19G ENDF/B-VII.0 shielding library.

The SCALE 6.1.2 shielding analysis sequence MAVRIC (Monaco with Automated Variance Reduction using Importance Calculations) (Ref. 12, Sect. S06) and the SCALE 27N-19G ENDF/B-VII shielding library were used to perform Monte Carlo transport and dose rate calculations. MAVRIC utilizes Denovo, a discrete ordinates code (Ref. 12, Sect. S06), to determine particle importance as a function of position and energy, and Monaco to perform Monte Carlo transport calculations. Radiation transport optimization is accomplished by (1) sampling more often source particles that have an ability to produce significant dose rate outside the source regions; and (2) reducing the variance of particle scores in the spatial region of interest. Meshview utility in the SCALE code system enables visualization of detailed radiation dose maps produced by MAVRIC.

The American National Standards Institute/American Nuclear Society Standard 6.1.1-1977 (ANSI/ANS-6.1.1-1977) (Ref. 13) flux-to-dose-rate conversion factors were used in the dose rate calculations, as recommended in NUREG-1617 and NUREG-1536.

6 CALCULATION METHOD

6.1 Depletion Calculation

The purpose of the depletion calculation is to generate radiation source terms as a function of decay time to be used in the shielding calculations. The radiation source term for dose rate calculations includes: (1) gamma and neutron radiation that originate in the irradiated fuel; and (2) activation gamma source in non-fuel assembly materials, primarily consisting of ^{60}Co ($t_{1/2}$ = 5.273 years) gamma rays. Typical concentrations of the cobalt impurity in fuel assembly parts made of stainless steel 304 and Inconel-718 are 800 and 4700 ppm, respectively (Ref. 14). The flux scaling factors for ^{60}Co source term calculations were 0.2, 0.1, and 0.2 for the fuel plenum and assembly top and bottom hardware regions, respectively (Ref. 15). The activation gamma source is negligible beyond approximately 50 years (i.e., ten ^{60}Co half-lives) after fuel discharge and can be neglected in shielding calculations.

An ARP library generated for the W 17×17 STD assembly type was used in all ORIGEN-ARP depletion calculations because the W 17×17 STD assembly type has similar neutronic characteristics with the V5H and MkBW assembly types. The ARP library resides within the UNF-ST&DARDS (Ref. 11), a UNF storage and transportation analysis resource and data system. A representative ORIGEN-ARP input file for the calculation of radiation source terms and decay heat is provided in Appendix A.

6.2 Dose Rate Calculation

A typical MAVRIC input file for the package model illustrated in Figure 1 (b) is provided in Appendix B. MPC-specific input data are neutron and photon energy distributions and strengths, which are provided in the "read definitions" and "read sources" data blocks. Variance reduction for the Monte Carlo calculation of the package external dose rate was accomplished by forward-weighting CADIS (consistent adjoint driven importance sampling).

6.3 Approach

A depletion calculation with ORIGEN-ARP requires several seconds of computer time whereas a Monte Carlo dose rate calculation with MAVRIC may need several days of computer time to complete. Three different approaches with respect to computer time and complexity of the calculation may be used to determine the minimum decay time assuming fuel assembly reconfiguration during transportation:

a. For each MPC, fuel depletion and package dose rate calculations are performed as a function of decay time. The minimum decay time corresponding to most stringent dose rate limit is determined by interpolation using the calculated dose rate as a function of decay time. Therefore, this approach requires significant computer time to determine the minimum decay time because multiple dose rate calculations for different decay times are necessary.

b. For each gamma and neutron energy group in the SCALE 19g-27n group structure, the package external dose rates are calculated using arbitrary source strengths that serve as normalization factors (e.g., 1 photon/s for gamma dose rate and 1 neutron/s for neutron dose rate calculations). The dose rate profiles at cask external surfaces are normalized to the source strength thus providing dose rate per particle per energy unit (e.g., rem/h/neutron/s/MeV). This approach requires a total of 46 (i.e., 19 + 27) MAVRIC calculations for a package model, the results of which can be applied to individual canisters. Depletion calculations are performed for the bounding fuel assembly with respect to shielding to calculate photon and neutron spectra as a function of decay time. The gamma and the neutron intensities in each energy group are then applied to the normalized dose rate values corresponding to each energy group to determine actual dose rate values for each individual canister. The minimum decay time corresponding to most stringent dose rate limit is determined by interpolation using the calculated dose rate as a function of decay time.

c. Photon and neutron spectra are analyzed to identify photon and neutron energy distributions that are bounding with respect to external package dose rate, as described in the next paragraph. The bounding photon and neutron spectra and arbitrary source strengths are used in MAVRIC gamma and neutron calculations, the results of which are normalized to the source strengths (e.g., rem/h/neutron/s). Depletion calculations are performed to calculate the proton and neutron spectra as a function of decay time. Then, the normalized gamma and neutron dose rate profiles at package external surfaces are multiplied by the gamma and neutron source strengths, respectively, to calculate canister-specific dose rate as a function of decay time. The minimum decay time corresponding to most stringent dose rate limit is determined by interpolation using the calculated dose rate as a function of decay time.

An analysis of the gamma and neutron energy distributions for low- and high-burnup assemblies was performed in this study. As illustrated in Figure 2, within the decay time interval 25 to 60 years, the photon energy distribution of a 43.6-GWd/MTU burnup assembly has a larger peak between 0.6 and 0.8 MeV than the 55.5-GWd/MTU GWd/MTU burnup assembly. The dominant photon source in a low-burnup assembly is the 0.661 MeV gamma ray of ^{137m}Ba ($t_{1/2}$ = 2.552 m), which is the beta decay product of ^{137}Cs ($t_{1/2}$ = 30.07 years). In addition to the 0.661- MeV gamma ray, the high-burnup assembly gamma source contains a large number of low-energy photons from the decay of higher actinides (e.g., ^{241}Am). Hence, the photon spectrum of a low-burnup assembly and a 25-year cooling time is harder (i.e., generates higher gamma dose rate) than the photon spectrum of a high-burnup assembly and may be used as a bounding photon spectrum with respect to external package dose rate. Confirmatory calculations showed very small differences, within the statistical uncertainty, between the gamma dose rates based on the actual and the bounding fuel assembly gamma spectra. The neutron dose rates within the decay time interval 25 to 60 years of low- and high-burnup assemblies are very similar, as illustrated in Figure 3. The neutron spectrum is relatively insensitive to the decay time because neutrons are primarily produced from spontaneous fission and (alpha,n) reaction due to ^{244}Cm ($t_{1/2}$ = 18.1 years) and ^{238}Pu ($t_{1/2}$ = 87.7 years).

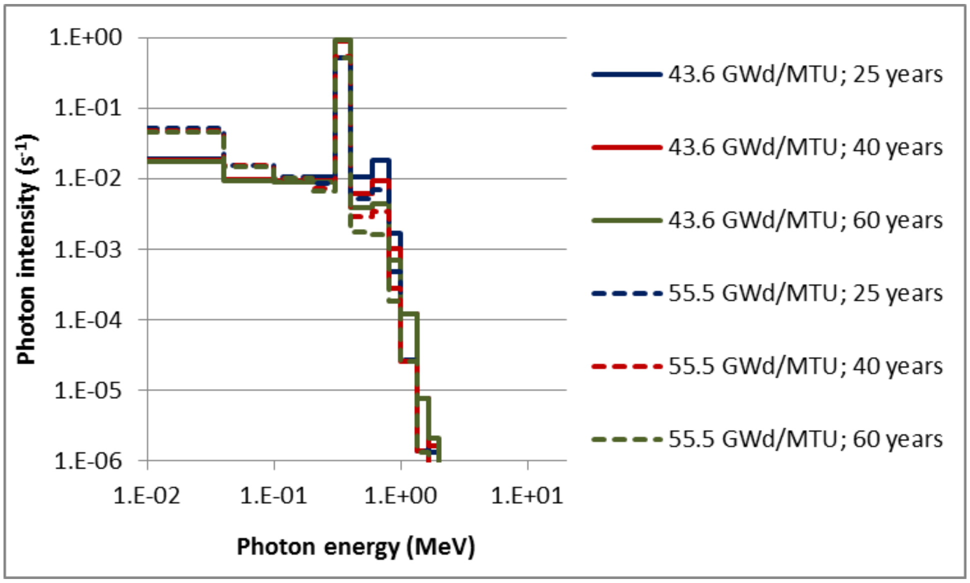

Figure 2. Photon energy distribution as a function of assembly average burnup and decay time.

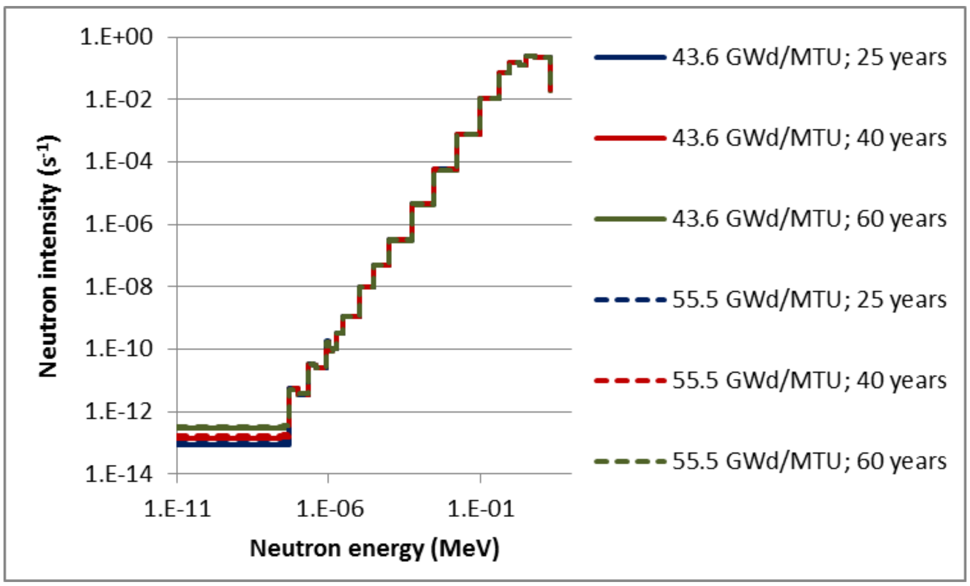

Figure 3. Neutron energy distribution as a function of assembly average burnup and decay time.

7 EXTERNAL CASK DOSE RATES

HI-STAR 100 has been approved by the NRC as an acceptable Type B(U)F-85 packaging for transport by exclusive use shipment (Ref. 4). Hence, the NCT dose rate was calculated at package external surface, at the transport vehicle outer surface, which was assumed to be defined by the outer surface of the impact limiters (Ref. 4), and at two meters from the outer surface of the vehicle. The HI-STAR 100 CoC (Ref. 5) indicates that the edge of the vehicle must be at least 9 ft from the bottom impact limiter. Typical dose rate maps obtained for the NTC package models using fuel rubble homogenized within the canister cavity and fuel rubble collapsed to the canister cavity bottom are illustrated in Figure 4 and Figure 5, respectively. The unit is rem/h for the dose rate values (i.e., “Response 3”) illustrated in the figures. The years of required storage prior to MPC transportation and the calculated dose rates at the package external surfaces of interest are tabulated in Table 3. The shipment year for the intact fuel was estimated based on the 625-W assembly decay heat limit for the HI-STAR 100. The most limiting dose rate was the dose rate at 2 meters from the side surface of the transportation vehicle. The shielding analysis used Approach (c), which is described in Section 6.3.

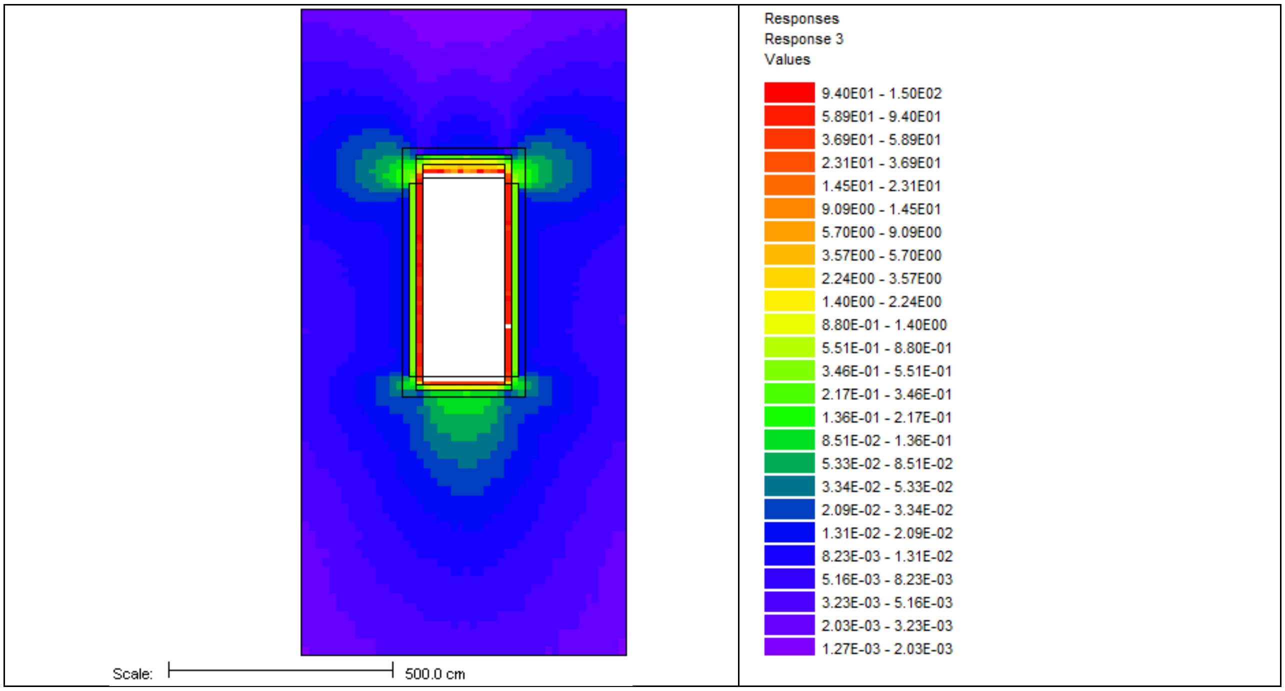

Figure 4. NCT cask external dose rate map assuming fuel rubble homogenized within cask cavity.

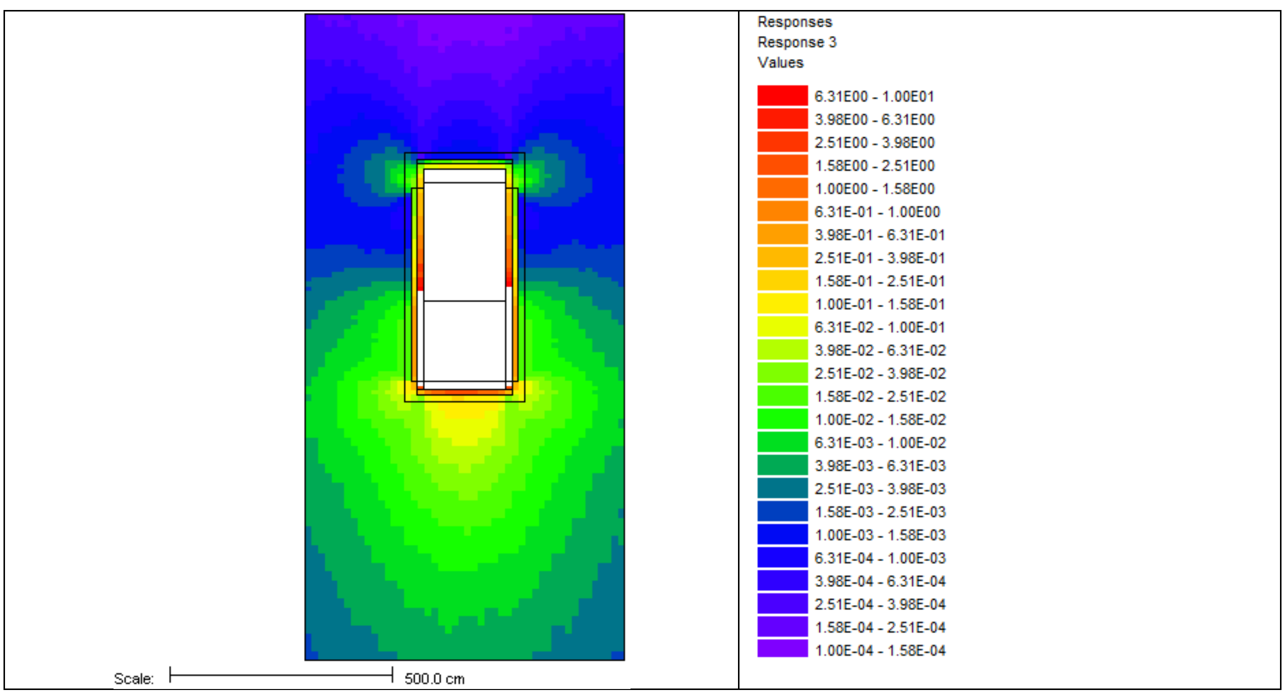

Figure 5. NCT cask external dose rate map assuming fuel rubble homogenized within bottom of cask cavity.

Table 3. Package shipment year and dose rates

					NCT dose rate (mrem/h)							HAC dose rate (mrem/h)		
	Intact Fuel	Reconfigured Fuel			Package outer surfaces[a]			Vehicle[b]	Surface located at 2 meters from the vehicle[c]			Surface located at 1 meter from the package[a]		
Cask No.	Shipment year[d]	Storage years	Shipment year	Decay heat (W)[e]	Top	Side	Bottom	Side	Top	Side[f]	Bottom	Top	Side	Bottom
1	2012	26	2031	438.4	11.8	389.8	123.7	66.7	1.8	9.8	4.1	46.3	166.4	215.6
2	2012	26	2031	438.4	11.8	389.8	123.7	66.7	1.8	9.8	4.1	46.3	166.4	215.6
3	2012	26	2031	438.4	11.8	389.8	123.7	66.7	1.8	9.8	4.1	46.3	166.4	215.6
4	2012	26	2031	439.0	11.4	385.4	124.9	66.6	1.8	9.8	4.2	44.9	161.6	212.7
5	2014	26	2033	438.4	11.8	389.8	123.7	66.7	1.8	9.8	4.1	46.3	166.4	215.6
6	2012	26	2031	439.0	11.4	385.4	124.9	66.6	1.8	9.8	4.2	44.9	161.6	212.7
7	2012	26	2031	439.0	11.4	385.4	124.9	66.6	1.8	9.8	4.2	44.9	161.6	212.7
8	2014	24	2033	438.4	11.8	389.8	123.7	66.7	1.8	9.8	4.1	46.3	166.4	215.6
9	2015	27	2034	439.0	11.4	385.4	124.9	66.6	1.8	9.8	4.2	44.9	161.6	212.7
10	2019	29	2038	438.4	11.8	389.8	123.7	66.7	1.8	9.8	4.1	46.3	166.4	215.6
11	2019	31	2038	438.4	11.8	389.8	123.7	66.7	1.8	9.8	4.1	46.3	166.4	215.6
12	2019	29	2038	438.4	11.8	389.8	123.7	66.7	1.8	9.8	4.1	46.3	166.4	215.6
13	2018	29	2037	438.4	11.8	389.8	123.7	66.7	1.8	9.8	4.1	46.3	166.4	215.6
14	2014	26	2033	438.4	11.8	389.8	123.7	66.7	1.8	9.8	4.1	46.3	166.4	215.6
15	2018	29	2037	438.4	11.8	389.8	123.7	66.7	1.8	9.8	4.1	46.3	166.4	215.6
16	2020	29	2039	438.4	11.8	389.8	123.7	66.7	1.8	9.8	4.1	46.3	166.4	215.6
17	2017	25	2036	439.0	11.4	385.4	124.9	66.6	1.8	9.8	4.2	44.9	161.6	212.7
18	2016	27	2035	439.0	11.4	385.4	124.9	66.6	1.8	9.8	4.2	44.9	161.6	212.7
19	2017	28	2036	438.4	11.8	389.8	123.7	66.7	1.8	9.8	4.1	46.3	166.4	215.6
20	2018	28	2037	439.0	11.4	385.4	124.9	66.6	1.8	9.8	4.2	44.9	161.6	212.7
21	2026	38	2050	414.9	12.8	395.2	117.3	65.5	1.9	9.6	3.8	50.1	177.7	220.0
22	2026	34	2050	414.9	12.8	395.2	117.3	65.5	1.9	9.6	3.8	50.1	177.7	220.0
23	2021	30	2043	428.8	12.2	390.0	120.0	65.7	1.8	9.7	3.9	47.8	170.7	216.4
24	2024	37	2048	423.3	12.6	388.9	115.4	64.4	1.9	9.5	3.7	49.3	174.9	216.5
25	2023	32	2045	428.3	11.7	383.0	120.2	65.2	1.8	9.6	4.0	46.1	165.0	212.1
26	2027	36	2052	424.7	13.1	401.8	118.7	66.4	1.9	9.8	3.8	51.1	181.3	223.8
27	2033	46	2057	423.9	13.6	405.9	116.7	66.2	2.0	9.8	3.7	52.9	186.7	226.6
28	2033	46	2057	423.9	13.6	405.9	116.7	66.2	2.0	9.8	3.7	52.9	186.7	226.6
29	2033	46	2057	423.9	13.6	405.9	116.7	66.2	2.0	9.8	3.7	52.9	186.7	226.6
30	2031	40	2055	423.3	12.6	388.9	115.4	64.4	1.9	9.5	3.7	49.3	174.9	216.5
31	2031	40	2055	423.3	12.6	388.9	115.4	64.4	1.9	9.5	3.7	49.3	174.9	216.5
32	2031	40	2055	423.3	12.6	388.9	115.4	64.4	1.9	9.5	3.7	49.3	174.9	216.5

Dose rate limit is [a]1000 mrem/h; [b]200 mrem/h; [c]10 mrem/h.
[d]Shipment year for intact fuel based on the 625-W assembly decay heat limit for the HI-STAR 100 package (Ref. 5).
[e]Maximum assembly decay heat per cask corresponding to the shipment year.
[f]Statistical error at the 95% confidence level is ~2%. The statistical error for dose rate on other surfaces is <6%.

8 CONCLUSIONS

This letter report describes a shielding analysis of a representative PWR transportation package (HI-STAR 100) assuming fuel assembly reconfiguration. The study analyzed 32 multipurpose canisters containing fuel assemblies with a maximum burnup value between 40 to 55.5 GWd/MTU and decay heat of 690 W to 1 kW at the time of loading to determine the timeframe for transporting used nuclear fuel assemblies based on compliance with the regulatory shielding requirements in 10 CFR 71 for transportation packages. The fuel assembly characteristics are representative of the fuel assemblies currently in dry storage at independent spent fuel storage installations. The shielding analysis used conservative radiation source terms and fuel assembly models to account for the impact of fuel reconfiguration that may be caused by fuel assembly damage during normal, off-normal, and accident storage/transportation conditions. Based on this analysis, an MPC may require between 26 to 43 years of storage prior to being transported to another facility. The decay heat of the fuel assemblies at the evaluated time of shipment was approximately 440 W, which is lower than the assembly decay limit of 625 W established for the HI-STAR 100 transportation package containing intact fuel assemblies. Analysis results show that fuel reconfiguration has the potential to result in an extension of the average storage time of 22 years before being able to meet the 10 CFR 71 transportation requirements for shielding (e.g., 10 CFR 71.87). The shielding analysis may be updated in the future to reflect more realistic fuel reconfigurations based on high burnup fuel failure data that may become available.

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Appendix A

The ORIGEN-ARP input file for the SQN Unit 2 V40 assembly is listed in this section.

```
=shell
cp /home/raq/unf/data/arplibs/bounding/arpdata.txt arpdata.txt
cp /home/raq/unf/data/arplibs/bounding/W1717WL-e1.0.arplib W1717WL-e1.0.arplib
cp /home/raq/unf/data/arplibs/bounding/W1717WL-e1.5.arplib W1717WL-e1.5.arplib
cp /home/raq/unf/data/arplibs/bounding/W1717WL-e2.0.arplib W1717WL-e2.0.arplib
cp /home/raq/unf/data/arplibs/bounding/W1717WL-e3.0.arplib W1717WL-e3.0.arplib
cp /home/raq/unf/data/arplibs/bounding/W1717WL-e4.0.arplib W1717WL-e4.0.arplib
cp /home/raq/unf/data/arplibs/bounding/W1717WL-e5.0.arplib W1717WL-e5.0.arplib
cp /home/raq/unf/data/arplibs/bounding/W1717WL-e6.0.arplib W1717WL-e6.0.arplib
end
=arp
W1717WL
3.8
17
80.3
80.3
80.3
80.3
80.3
80.3
88.1
88.1
88.1
88.1
88.1
84.9
84.8
84.9
84.9
84.8
84.75
38.79886
38.79886
38.79886
38.79886
38.79886
38.79886
38.79886
38.79886
38.79886
38.79886
38.79886
38.79886
38.79886
38.79886
38.79886
38.79886
38.79886
1
1
1
1
1
1
1
1
1
```

```
1
1
1
1
1
1
1
1
0.6905
ft33f001
end
=origens
' W17x17V5H; 462.9 kg U; 3.8 wt% U-235; 55.527 GWd/MTU; fuel only
0$$ a4 33 a11 71 e t
W1717WL
3$$ 33 a3 1 27 a16 2 a33 18 e t
35$$ 0 t
56$$ 10 10 a6 3 a10 0 a13 4 a15 3 a19 1 e
57** 0 a3 1e-05 0.056103 e t
cycle 1
0.462 mtu
58** 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96
60** 8.1 16.1 24.1 32.1 40.2 48.2 56.2 64.2 72.3 80.3
66$$ a1 2 a5 2 a9 2 e
73$$  922340 922350 922360 922380
74**  156 17556 81 444207
75$$ 2 2 2 2
t
W1717WL
3$$ 33 a3 2 27 a16 2 a33 19 e t
35$$ 0 t
56$$ 10 10 a6 3 a10 10 a15 3 a18 1 e
57** 80.3 a3 1e-05 0.056103 e t
cycle 1
0.462 MTU
58** 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96
60** 88 96 104.4 112.4 120.5 128.5 136.5 144.5 152.6 160.6
66$$ a1 2 a5 2 a9 2 e t
W1717WL
3$$ 33 a3 3 27 a16 2 a33 19 e t
35$$ 0 t
56$$ 10 10 a6 3 a10 10 a15 3 a18 1 e
57** 160.6  a3 1e-05 0.056103 e t
cycle 1
0.462 MTU
58** 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96
60** 168.6 176.4 184.4 192.4 200.5 208.5 216.5 224.5 232.6 240.9
66$$ a1 2 a5 2 a9 2 e t
W1717WL
3$$ 33 a3 4 27 a16 2 a33 19 e t
35$$ 0 t
56$$ 10 10 a6 3 a10 10 a15 3 a18 1 e
57** 240.9  a3 1e-05 0.056103 e t
cycle 1
0.462 MTU
58** 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96
60** 248.9 257 265 273 281 289 297 305 313 321.2
66$$ a1 2 a5 2 a9 2 e t
W1717WL
3$$ 33 a3 5 27 a16 2 a33 19 e t
35$$ 0 t
56$$ 10 10 a6 3 a10 10 a15 3 a18 1 e
57** 321.2  a3 1e-05 0.056103 e t
```

```
cycle 1
0.462 MTU
58** 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96
60** 329.3 337.3 345.3 353.4 361.4 369.4 377.4 385.5 393.5 401.5
66$$ a1 2 a5 2 a9 2 e t
W1717WL
3$$ 33 a3 6 27 a16 2 a33 19 e t
35$$ 0 t
' 56$$ 10 10 a6 3 a10 10 a15 3 a18 1 e
56$$ 10 10 a10 10 a15 3 a18 1 e
57** 401.5  a3 1e-05 0.056103 e t
cycle 1
0.462 MTU
58** 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96
60** 409.5 417.6 425.6 433.6 441.7 449.7 457.7 465.7 473.8 481.8
66$$ a1 2 a5 2 a9 2 e t
54$$ a8 1 a11 0 e
56$$ a2 10 a6 3 a10 10 a15 3 a17 4 e
57** 0 a3 1e-05 e
95$$ 0 t
downtime after cycle 1
0.462 mtu
60** 0.01 0.03 0.08 0.2 0.6 1 3 10 30 50
61** f0.05
65$$
'Gram-Atoms   Grams   Curies   Watts-All   Watts-Gamma
 3z   1   0   0   3z   3z   3z   6z
 3z   1   0   0   3z   3z   3z   6z
 3z   1   0   0   3z   3z   3z   6z
t
W1717WL
3$$ 33 a3 7 27 a16 2 a33 19 e t
35$$ 0 t
56$$ 10 10 a6 3 a10 10 a15 3 a18 1 e
57** 481.8 a3 1e-05 0.061525 e t
cycle 2
0.462 MTU
58** 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96
60** 490.6 499.4 508.2 517 525.8 534.6 543.4 552.3 561.1 569.9
66$$ a1 2 a5 2 a9 2 e t
W1717WL
3$$ 33 a3 8 27 a16 2 a33 19 e t
35$$ 0 t
56$$ 10 10 a6 3 a10 10 a15 3 a18 1 e
57** 569.9  a3 1e-05 0.061525 e t
cycle 2
0.462 MTU
58** 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96
60** 578.7 587.5 596.3 605.1 614 622.8 631.6 640.4 649.2 658.0
66$$ a1 2 a5 2 a9 2 e t
W1717WL
3$$ 33 a3 9 27 a16 2 a33 19 e t
35$$ 0 t
56$$ 10 10 a6 3 a10 10 a15 3 a18 1 e
57** 658.0  a3 1e-05 0.061525 e t
cycle 2
0.462 MTU
58** 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96
60** 666.8 675.6 684.4 693.2 702.1 710.9 719.7 728.5 737.3 746.0
66$$ a1 2 a5 2 a9 2 e t
W1717WL
3$$ 33 a3 10 27 a16 2 a33 19 e t
35$$ 0 t
```

```
56$$ 10 10 a6 3 a10 10 a15 3 a18 1 e
57** 746.0  a3 1e-05 0.061525 e t
cycle 2
0.462 MTU
58** 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96
60** 754.8 763.6 772.4 781.2 790.1 798.9 807.7 816.5 825.3 834.1
66$$ a1 2 a5 2 a9 2 e t
W1717WL
3$$ 33 a3 11 27 a16 2 a33 19 e t
35$$ 0 t
56$$ 10 10 a10 10 a15 3 a18 1 e
'56$$ 10 10 a6 3 a10 10 a15 3 a18 1 e
57** 834.1  a3 1e-05 0.061525 e t
cycle 2
0.462 MTU
58** 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96
60** 842.9 851.7 860.5 869.3 878.2 887 895.8 904.6 913.4 922.1
66$$ a1 2 a5 2 a9 2 e t
54$$ a8 1 a11 0 e
56$$ a2 9 a6 3 a10 10 a15 3 a17 4 e
57** 0 a3 1e-05 e
95$$ 0 t
downtime after cycle 2
0.462 mtu
60** 0.01 0.03 0.1 0.3 1 3 6 15 30
61** f0.05
65$$
'Gram-Atoms   Grams   Curies   Watts-All   Watts-Gamma
 3z   1   0   0   3z   3z   3z   6z
 3z   1   0   0   3z   3z   3z   6z
 3z   1   0   0   3z   3z   3z   6z
t
W1717WL
3$$ 33 a3 12 27 a16 2 a33 19 e t
35$$ 0 t
56$$ 10 10 a6 3 a10 9 a15 3 a18 1 e
57** 922.1  a3 1e-05 0.059293 e t
cycle 3
0.462 MTU
58** 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96
60** 930.6 939.1 947.6 956.1 964.6 973.1 981.6 990.1 998.6 1007
66$$ a1 2 a5 2 a9 2 e t
W1717WL
3$$ 33 a3 13 27 a16 2 a33 19 e t
35$$ 0 t
56$$ 10 10 a6 3 a10 10 a15 3 a18 1 e
57** 1007  a3 1e-05 0.059293 e t
cycle 3
0.462 MTU
58** 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96
60** 1015.5 1024 1032.5 1041 1049.5 1058 1066.5 1075 1083.5 1091.8
66$$ a1 2 a5 2 a9 2 e t
W1717WL
3$$ 33 a3 14 27 a16 2 a33 19 e t
35$$ 0 t
56$$ 10 10 a6 3 a10 10 a15 3 a18 1 e
57** 1091.8  a3 1e-05 0.059293 e t
cycle 3
0.462 MTU
58** 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96
60** 1100 1108.8 1117 1125.8 1134 1142.8 1151 1159.8 1168 1176.7
66$$ a1 2 a5 2 a9 2 e t
W1717WL
```

```
3$$ 33 a3 15 27 a16 2 a33 19 e t
35$$ 0 t
56$$ 10 10 a6 3 a10 10 a15 3 a18 1 e
57** 1176.7  a3 1e-05 0.059293 e t
cycle 3
0.462 MTU
58** 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96
60** 1185 1193.7 1202 1210.7 1219 1227.7 1236 1244.7 1253 1261.6
66$$ a1 2 a5 2 a9 2 e t
W1717WL
3$$ 33 a3 16 27 a16 2 a33 19 e t
35$$ 0 t
56$$ 10 10 a6 3 a10 10 a15 3 a18 1 e
57** 1261.6  a3 1e-05 0.059293 e t
cycle 3
0.462 MTU
58** 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96
60** 1270 1278.6 1287 1295.6 1304 1312.6 1321 1329.6 1338 1346.4
66$$ a1 2 a5 2 a9 2 e t
W1717WL
3$$ 33 a3 17 27 a16 2 a33 19 e t
35$$ 0 t
56$$ 10 10 a10 10 a15 3 a18 1 e
'56$$ 10 10 a6 3 a10 10 a15 3 a18 1 e
57** 1346.4  a3 1e-05 0.059293 e t
cycle 3
0.462 MTU
58** 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96 17.96
60** 1355 1363.4 1372 1380.4 1389 1397.4 1406 1414.4 1423 1431.15
66$$ a1 2 a5 2 a9 2 e t
54$$ a8 1 a11 0 e
56$$ a2 10 a6 1 a10 10 a14 5 a15 3 a17 2 e
57** 0 a3 1e-05 e
95$$ 0 t
decay calculation
0.462 mtu
60** 0.01 0.03 0.1 0.3 0.9 2.7 8 25 46 58
61** f0.05
65$$
'Gram-Atoms   Grams   Curies   Watts-All   Watts-Gamma
 3z   1   0   0   1    2z   3z   3z   6z
 3z   1   0   0   1    2z   3z   3z   6z
 3z   1   0   0   1    2z   3z   3z   6z
81$$ 2 0 26 1 a7 200 e
82$$ 0 0 0 0 0 0 2 2 2 2 e
83**
 2.0000000e+07 1.0000000e+07 8.0000000e+06 6.5000000e+06 5.0000000e+06
 4.0000000e+06 3.0000000e+06 2.5000000e+06 2.0000000e+06 1.6600000e+06
 1.3300000e+06 1.0000000e+06 8.0000000e+05 6.0000000e+05 4.0000000e+05
 3.0000000e+05 2.0000000e+05 1.0000000e+05 4.0000000e+04 1.0000000e+04 e
84**
 2.0000000e+07 6.4340000e+06 3.0000000e+06 1.8500000e+06
 1.4000000e+06 9.0000000e+05 4.0000000e+05 1.0000000e+05 1.7000000e+04
 3.0000000e+03 5.5000000e+02 1.0000000e+02 3.0000000e+01 1.0000000e+01
 3.0499900e+00 1.7700000e+00 1.2999900e+00 1.1299900e+00 1.0000000e+00
 8.0000000e-01 4.0000000e-01 3.2500000e-01 2.2500000e-01 9.9999850e-02
 5.0000000e-02 3.0000000e-02 9.9999980e-03 1.0000000e-05 e
 t
56$$ 0 0 a10   7 e t
56$$ 0 0 a10   8 e t
56$$ 0 0 a10   9 e t
56$$ 0 0 a10  10 e t
56$$ f0 t
```

```
end
=opus
libunit=33
units=watts
libtype=all
time=years
nposition= 1 2 3 4 end
end
```

Appendix B

A typical MAVRIC input file for HI-STAR 100 package dose rate calculation is listed in this section. MPC-specific input data are neutron and photon spectra and strengths provided in the “read definitions” and “read sources” data blocks.

```
=mavric
gbc-32 cask; W17x17-OFA; 55.5 GWd/MTU; 46-year decay time; intact assemblies
v7-27n19g
read comp
' homozenized contents
' homozenized contents
 wtptRubble   1 2.4945 18  8016  7.513 92238 56.620
                        40000 16.744 28000  2.003 24000  3.633 26000 12.889
                        41000  0.011 42000  0.007 22000  0.002 13000  0.001
                        27000  0.016 25000  0.379
                        14000  0.143 29000  0.001  6000  0.006
                        16000  0.006 15000  0.009  7000  0.017
              1.0 293.0 end
' homogenized ss basket plates
 wtptss 60 0.4242  10 28000 10.000 24000 19.000 26000 67.785
                        27000  0.260 25000  2.000
                        14000  0.750  6000  0.030
                        16000  0.030 15000  0.045  7000  0.100
              1.0 293.0 end
'
carbonsteel  6   1.0 293.0 end
ss304  8   1.0 293.0 end
wtpthol 10 1.61 7 13000 21.285 1001 5.92 6000 27.66039 8016 42.372 7014 1.98
                    5010 0.14087 5011 0.64174 1.0 293.0 end

dry-air   11 1.0 293.0 end
dry-air   20 1.0 293.0 end
dry-air   12 1.0 293.0 end
'-----------------------------------
end comp
read geometry
'
global unit 5
' cylinder 22 85.725 202.58 21.59
' cylinder 23 85.725 469.9 202.58
' cylinder 24 85.725 474.98 469.9
'
 cylinder  1 85.725 474.98 21.59
 cylinder  2 85.725 499.11 0
 cylinder  3 85.725 500.6975 499.11
 cylinder  4 105.7275 515.9375 0
 cylinder 10 93.424375 461.9625 22.225
 cylinder  5 108.664 461.9625 22.225
 cylinder  7 119.586 455.9525 22.225
 cylinder  6 119.586 460.6925 22.225
 cylinder  8 120.856 461.9625 22.225
' 2.5 in. top and bottom neutron shield
 cylinder 50 105.7275 522.2875 515.9375
 cylinder 60 105.7275 -6.35 0
'
 cylinder  9 141     542.3 -26.35
'boundary 215 cm from cask external surfaces
' edge of vehicle at leat 9' from bottom of bottom impact limiter
' bottom impact limiter is 60" thick; 130 cm below bottom surface
' total distance between package bottom and edge of vehicle ~405 cm
```

```
' personnel barrier ~ 160 cm from cask cylindrical axis
' add 2 m for the 10 mrem/h dose rate
 cuboid 11 360 -360 360 -360 852 -605

' media 1  1 22
' media 60 1 23
' media 11 1 24
 media 1  1 1

 media 8 1 10 -1
 media 6 1 5 -10 -2
 media 11 1 6 -5 -7
 media 10 1 7 -5
 media 8 1 8 -5 -6
 media 8 1 2 -10 -1 -5
 media 11 1 3
 media 8 1 4 -3 -2 -8
 media 10 1 50
 media 10 1 60
 media 20 1 9 -4 -8 -50 -60
 media 12 1 11 -9
 boundary 11

end geometry

read definitions
 response 1  specialDose=9029  end response
 response 2  specialDose=9504  end response
 response 3  specialDose=9729  end response
'ANSI standard (1977) flux-to-dose-rate (rem/h)/(particle/cm2/s)
'9029=neutron; 9504=photon; 9729=coupled neutron photon

 gridGeometry 1
   title="for discrete ordinates calculations"
   xplanes -360 -330 -300 -270 -240 -210 -180 -150 -135

           -121 -118 -116.5 -114 -112.5 -111 -109.5
           -108 -106.5 -105 -103.5 -102 -100.5 -99
           -97.5 -96 -94.5 -93 -91.5 -90 -88.5 -87

           -85.5 -83.4 -80.4 -77.4 -74.4 -71.4 -68.4 -65.4
           -62.4 -59.4 -55 -50 -40 -25 0 25 40 50
            55 59.4 62.4 65.4 68.4 71.4 74.4 77.4 80.4 83.4 85.5

            87 88.5 90 91.5 93 94.5 96 97.5 99 100.5 102 103.5
            105 106.5 108 109.5 111 112.5 114 116.5 118 121

            135 150 180 210 240 270 300 330 360 end

   yplanes -360 -330 -300 -270 -240 -210 -180 -150 -135

           -121 -118 -116.5 -114 -112.5 -111 -109.5
           -108 -106.5 -105 -103.5 -102 -100.5 -99
           -97.5 -96 -94.5 -93 -91.5 -90 -88.5 -87

           -85.5 -83.4 -80.4 -77.4 -74.4 -71.4 -68.4 -65.4
           -62.4 -59.4 -55 -50 -40 -25 0 25 40 50
            55 59.4 62.4 65.4 68.4 71.4 74.4 77.4 80.4 83.4 85.5

            87 88.5 90 91.5 93 94.5 96 97.5 99 100.5 102 103.5
            105 106.5 108 109.5 111 112.5 114 116.5 118 121

            135 150 180 210 240 270 300 330 360 end
```

```
zplanes -605 -561 -531 -501 -471 -441 -411 -381 -351 -321
        -291 -171 -141 -111 -81 -51 -21.35 -6.35 -4.2 -2.1 0

        1.5 3 4.5 6 7.5 9 10.5 12 13.5 15 16.5 18 19.5
        21 23 25 27.5 30.5 34 38 43

        50 60 70 80 90 100 110 120 130 140 150
        160 170 180 190 200 210 220
        230 240 250 260 270 280 290
        300 310 320 330 340 350 360
        370 380 390 400 410 420 430 440 450
        455 460 465 469 472 475 476.5

        478 479.5 481 482.5 484 485.5 487 488.5 490 491.5
        493 494.5 496 497.5 499 501 502.5 504 505.5 507
        508.5 510 511.5 513 514.5 516 518.1 520.2 522.35

        537.35 567 597 632 667 702 732 762 792 822 852 end

end gridGeometry

 gridGeometry 2
    title="for mesh tallies"
     xplanes -360 -121 121 360 end
     xLinear 14 -345 -136
     xLinear 14 -106 106
     xLinear 14  136 345

     yplanes -360 -121 121 360 end
     yLinear 14 -345 -136
     yLinear 14 -106 106
     yLinear 14  136 345

     zplanes -605 -6.35 522.2875 852 end
     zLinear 28 -585 -21.35
     zLinear 48 4.25 512.4
     zLinear 20 538 837

 end gridGeometry

 cylGeometry 3
     radii 10 20 30 40 50 60 70 80 90 100 110 120.856 136 151 166
           181 196 211 226 241 256 271 286 301 316 331 346 360 end
     degreeLinear 30 0 360
     zLinear 96 -605 852
 end cylGeometry

 distribution 1
     title="neutron energy distribution for the fuel assembly"
     neutronGroups
     truepdf 2.02E-02 2.15E-01 2.38E-01 1.27E-01 1.58E-01 1.58E-01 7.24E-02
             1.07E-02 7.65E-04 5.67E-05 4.46E-06 3.13E-07 4.95E-08 9.79E-09
             1.11E-09 3.26E-10 1.05E-10 9.10E-11 9.11E-11 1.74E-10 2.58E-11
             3.01E-11 3.43E-11 3.43E-12 5.36E-12 1.50E-13 1.39E-13 end
 end distribution

 distribution 2
     title="photon energy distribution for the fuel assembly"
     photonGroups
     truepdf 1.81E-12 9.87E-11 4.50E-10 2.30E-09 5.72E-09 1.70E-08 2.98E-07
             1.37E-06 2.64E-05 3.69E-04 4.47E-03 3.40E-03 5.27E-01 7.49E-03
             1.04E-02 1.53E-02 4.91E-02 1.14E-01 2.69E-01 end
```

```
    end distribution


end definitions

'------------------------------------------------------------------------------
' Sources Block
'------------------------------------------------------------------------------
read sources
    src 1
        title="neutron source"
        neutrons
        strength=7.35E+09
        cuboid 4p85.725 474.98 21.59
        mixture=1
        eDistributionID=1
    end src
    src 2
        title="active fuel photon source"
        photons
        strength=5.09E+16
        cuboid 4p85.725 474.98 21.59
        mixture=1
        eDistributionID=2
    end src
end sources

'------------------------------------------------------------------------------
' Tallies Block - look at neutron, photon, and total dose all around the cask
'------------------------------------------------------------------------------
read tallies
    meshTally 1
        gridGeometryID=2
        responseIDs 1 2 3 end
        noGroupFluxes
    end meshTally
    meshTally 2
        cylGeometryID=3
        responseIDs 1 2 3 end
        noGroupFluxes
    end meshTally
end tallies

'------------------------------------------------------------------------------
' Parameters Block
'------------------------------------------------------------------------------
read parameters
    randomSeed=8675309385
    noFissions

    perBatch=3500000  batches=100

end parameters

'------------------------------------------------------------------------------
' Importance Map Block - optimize the MC calculation for calculation of
'     total (neutron + photon) dose in a mesh tally outside of the cask.
' Use the macroMaterials (mmSubCells) to homogenize materials in the
'     Denovo model to produce a more accurate importane map.
'------------------------------------------------------------------------------
read importanceMap
    gridGeometryID=1
    adjointSource 1
```

```
        boundingBox  360  -360 360  -360  852 -605
        responseID=1
        mixture=20
    end adjointSource

    adjointSource 2
        boundingBox  360  -360 360  -360  852 -605
        responseID=2
        mixture=20
    end adjointSource

    adjointSource 3
        boundingBox  360  -360 360  -360  852 -605
        responseID=3
        mixture=20
    end adjointSource
    respWeighting
    reduce
    subCells=3
    mmSubCells=3

end importanceMap

end data
end
```