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Waste Handling System	000-00C-WHS0-00600-000-00A

5. Title

Direct Radiation Dose Consequence Calculation for Category 1 and 2 Event Sequences

6. Group

Environmental & Nuclear Engineering

7. Document Status Designation

☒ Preliminary ☐ Final ☐ Cancelled

8. Notes/Comments

LP-2.14Q-BSC review was not conducted for the following reasons:

This document is an initial issue containing the dose information for Category 1 and 2 event sequences. There is no impact on any existing controlled documents. Use of the information from this document as direct input into products that are under development requires consultation with the Environmental and Nuclear Engineering organization for appropriateness and impact.

Attachments	Total Number of Pages
I. Listing of the Electronic Files Contained in Attachment II	2
II. One Compact Disc Containing Electronic Files	N/A

RECORD OF REVISIONS

9. No.	10. Reason For Revision	11. Total # of Pgs.	12. Last Pg. #	13. Originator (Print/Sign/Date)	14. Checker (Print/Sign/Date)	15. QER (Print/Sign/Date)	16. Approved/Accepted (Print/Sign)	17. Date
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FIGURES

TABLES

ACRONYMS

ALARA	as low as is reasonably achievable
ANSI	American National Standards Institute
ANS	American Nuclear Society
ATR	advanced test reactor
B&W	Babcock & Wilcox, Inc.
BSC	Bechtel SAIC Company, LLC
BWR	boiling water reactor
CHF	Canister Handling Facility
CFR	Code of Federal Regulations
CRWMS M&O	Civilian Radioactive Waste Management System Management & Operating Contractor
CSNF	commercial spent nuclear fuel
DIRS	Document Input Reference System
DOE	U.S. Department of Energy
DPC	dual-purpose canister
DTF	Dry Transfer Facility
FFTF	Fast Flux Test Facility
FHF	Fuel Handling Facility
FRR	foreign research reactor
FSVR	Fort Saint Vrain reactor
HLW	high-level (radioactive) waste
HFIR	high flux isotope reactor
HVAC	heating, ventilation, and air-conditioning
LWBR	light water breeder reactor
MCNP	Monte Carlo N-Particle transport code
MCO	multi-canister overpack
MPC	multi-purpose canister
MTU	metric tons of uranium
NAC	Nuclear Assurance Corporation
NRC	U.S. Nuclear Regulatory Commission
PDC	Project Design Criteria (Document)
PWR	pressurized water reactor

SFTM	spent fuel transfer machine
SNF	spent nuclear fuel
SRS	Savannah River Site
TCRRF	Transportation Cask Receipt/Return Facility
TEDE	total effective dose equivalent
TMI	Three Mile Island
TRIGA	training, research, isotopes General Atomics
WP	waste package

1. PURPOSE

Performance objectives for the geologic repository operations area through permanent closure in 10 CFR (Code of Federal Regulations) 63.111 [DIRS 158535] identify compliance with regulatory dose limits for workers and members of the public as a design objective. The purpose of this design calculation is to determine direct radiation dose consequences for Category 1 and 2 event sequences. It does not include the worker dose assessment for recovery operations following Category 1 event sequences.

The scope of work includes the following items:

- Evaluation of worker doses as a result of direct radiation from Category 1 event sequences
- Evaluation of dose to members of the public from direct radiation after Category 1 event sequences.
- Evaluation of dose to members of the public from direct radiation after Category 2 event sequences.
- Evaluation of doses above for each of the waste forms to be received and disposed of at Yucca Mountain. These waste forms include commercial spent nuclear fuel (CSNF), high level (radioactive) waste (HLW), naval spent nuclear fuel (SNF), multi-canister overpack (MCO), and U.S. Department of Energy (DOE) SNF.

The results of this calculation support the preclosure safety analysis that has the purpose to demonstrate that repository design and operations meet the 10 CFR Part 63 [DIRS 158535] requirements for normal operations and Category 1 and 2 event sequences. Part of the geological repository operations area includes the surface area with the Waste Handling System designated as Safety Category in *Q-List* (BSC [Bechtel SAIC Company] 2005 [DIRS 171190], Table A-1) such as Fuel Handling Facility (FHF), Canister Handling Facility (CHF), Dry Transfer Facility (DTF), Remediation Facility, and (TCRRF) Transportation Cask Receipt/Return Facility. Therefore, this calculation is subject to the requirements of the *Quality Assurance Requirements and Description* (DOE 2004 [DIRS 171539]). Development, performance, and documentation of this preliminary design calculation conform to the administrative procedure AP-3.12Q, *Design Calculations and Analyses*.

The calculations contained in this document were developed by the Design & Engineering, Environmental & Nuclear Engineering, and are intended solely for the use of the Design & Engineering in its work regarding doses from Category 1 and 2 event sequences. Yucca Mountain Project personnel from the Design & Engineering, Environmental & Nuclear Engineering, should be consulted before use of the calculations for purposes other than those stated herein or use by individuals other than authorized personnel in Design & Engineering.

2. METHOD

The method used in this calculation is based on evaluation criteria for License Application that are described *in Yucca Mountain Review Plan* (U.S. Nuclear Regulatory Commission [NRC] 2003 [DIRS 163274], Section 2.1.1.5, *Consequence Analyses*).

2.1 CONSEQUENCE ANALYSIS METHODOLOGY

The dose consequence calculation methodology consists of the following:

- Identification of applicable project requirements, design criteria, and regulatory requirements.
- Identification of Category 1 and 2 event sequences. This information is the result of a calculation that evaluates internal event sequences that may occur before permanent closure of the geologic repository operations area and categorizes event sequences according to their frequency of occurrence (BSC 2005 [DIRS 171429]).
- Consequence evaluations for Category 1 and 2 event sequences.
- Determination of on-site and off-site doses from direct exposures during Category 1 event sequences.
- Determination of off-site doses from direct exposures after Category 2 event sequences.

An acceptable shielding analysis methodology is used to determine doses from direct exposures. The methodology employs a qualified computer code for shielding calculations, a bounding radiation source term for waste forms, the flux-to-dose rate conversion factors documented in American National Standards Institute (ANSI)/American Nuclear Society (ANS) 6.1.1-1977 [DIRS 107016], and credit for shielding materials.

2.2 EVALUATION OF DIRECT DOSE FOR CATEGORY 1 EVENT SEQUENCES

Regulatory requirements include dose limits for annual total effective dose equivalent (TEDE), which is defined as the sum of deep-dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures) (10 CFR 20.1003 [DIRS 173165]). External exposures are the result of direct radiation. Internal exposures are excluded from this calculation, since the scope covers direct radiation only.

The deep-dose equivalent applies to external whole-body exposures and represents the dose equivalent at a tissue depth of 1 cm. Whole body means, for purposes of external exposure, head, trunk (including male gonads), arms above the elbow, or legs above the knee (10 CFR 20.1003 [DIRS 173165]). The estimation of the on-site direct dose to a member of the public is based on design criteria for facilities. The on-site worker dose is determined according to Equation 1 (see Section 5.1.2.3).

$$ED = T \times EDR \times \nu \qquad \text{Eq. 1}$$

where:

ED	=	Frequency-weighted annual external dose to a worker resulting from a Category 1 event sequence (mrem/year)
T	=	Duration of exposure per event (hours)
EDR	=	External dose rate at worker location (mrem/h)
ν	=	Frequency of event occurrence/year

2.3 COMPUTATIONAL METHOD

This calculation determines direct radiation dose rates from both Category 1 and 2 event sequences using the MCNP (Monte Carlo N-Particle) transport code (Briesmeister 1997 [DIRS 103897]). MCNP has the capability to accurately simulate gamma and neutron radiation interaction with matter. Flux at a point (tally type F5), flux averaged over a surface segment (tally type F2), and flux averaged over a cell volume (tally type F4), modified by the ANSI/ANS-6.1.1-1977 [DIRS 107016] flux-to-dose-rate conversion factors, were used to compute dose rates.

Electronic management of information generated from this calculation is controlled in accordance with AP-3.13Q, *Design Control*. The electronic input and output files are stored on a compact disc and included as an attachment to this document (Attachment II).

3. ASSUMPTIONS

3.1 BOUNDING ASSUMPTIONS

3.1.1 It is assumed that workers in continuous occupational access areas are exposed to direct radiation from a Category 1 event sequence for a maximum duration of eight hours.

Rationale: Eight hours exposure following the event sequences is conservative. It represents the maximum time period of exposure for a working day.

Usage: This assumption is used in Section 7.1.1.

3.1.2 It is assumed that the Babcock & Wilcox (B&W) 15x15 Mark B pressurized water reactor (PWR) fuel assembly is a bounding spent nuclear fuel (SNF) assembly for dose rate calculations.

Rationale: This assembly type has been used in a source term calculation (BSC 2004 [DIRS 169061], Section 5.2) to generate radiation source terms for shielding calculations. The uranium mass used in the source term calculation is 475 kg per assembly (BSC 2004 [DIRS 169061], Section 5.2), which bounds the heavy metal load for existing PWR assemblies (BSC 2004 [DIRS 167058], Table 24). Further, BSC (2004 [DIRS 167058], Section 6.3.2.1) has determined that the design basis gamma and neutron source intensities per metric ton of uranium (MTU) for B&W 15x15 Mark B bound the gamma and neutron source intensities for different PWR assembly types. Therefore, the use of this assembly will provide conservative dose rates.

Usage: This assumption is used in Section 6.3.1.

3.1.3 It is assumed that a CSNF assembly consists of four distinct regions, which are the active fuel, bottom end fitting, plenum, and top end fitting. The materials and radiation sources of each assembly region are homogenized inside a volume defined by the region height and the cavity radius of a canister.

Rationale: The assembly representation is consistent with source term generation calculations (BSC 2004 [DIRS 169061], Table 3) and the simplified representation of the fuel regions generates conservative dose rates. The effect of source homogenization is twofold: it decreases the mass density of the source region and places source points slightly closer to detector locations, reducing radiation attenuation. The results of the calculations provided in CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) (1998 [DIRS 102134], Section 6) showing that the effect of assembly homogenization is a slight increase in average waste-package surface dose rates also support this assumption.

Usage: This assumption is used in Section 6.3.1.

3.1.4 It is assumed that the bottom end-fitting length for a B&W 15x15 Mark B PWR fuel assembly is 4 inches.

Rationale: The available reference for the PWR assembly (DOE 1987 [DIRS 132333], p. 2A-35) provides the length of the bottom end nozzle of the fuel assembly to be 2 in. An additional length of 2 in., for a total length of 4 in., is used in this calculation to cover all materials in the bottom-end fitting region. Dose rate results are insensitive to the assumption of the length, as the mass in the bottom end-fitting region is conserved.

Usage: This assumption is used in Section 6.3.1.

3.1.5 It is assumed that the active fuel region of SNF contains the same concentrations of U-235 and U-238 as does fresh, unirradiated fuel.

Rationale: This assumption is conservative since it leads to slightly higher dose rates. The higher percentage of fissile U-235 in each assembly results in a greater number of induced fission neutrons, and consequently a stronger radiation field. It should be noted that this assumption applies to the material composition. It does not apply to the source term, which is based on spent fuel.

Usage: This assumption is used in Sections 6.3.1, 6.3.4, and 6.3.5.

3.1.6 It is assumed that a power peaking factor of 1.25 bounds the axial distribution of gamma source in the active fuel region of an SNF assembly.

Rationale: This value is based on the heat profile for a representative PWR assembly (Creer et al. 1987 [DIRS 136937], p. 3-29).

Usage: This assumption is used in Sections 6.3.1, 6.3.4, and 6.3.5.

3.1.7 It is assumed that a neutron peaking factor of 1.7 bounds the neutron source axial distribution of a CSNF assembly.

Rationale: CSNF burnup axial profiles tend to flatten with burnup, as shown in BSC (2003 [DIRS 166138], Table 32), where the ratio of maximum burnup to average burnup varies from 1.15 (10 GWd/MTU) to 1.086 (45 GWd/MTU). The neutron peaking factor for a typical PWR assembly provided in CRWMS M&O (2000 [DIRS 153872], p. S4.4.17), of approximately 1.7, is bounding for the burnup value of 80 GWd/MTU used in this calculation.

Usage: This assumption is used in Section 6.3.1.

3.1.8 It is assumed that the design-basis glass developed at Savannah River Site (SRS) Defense Waste Processing Facility is a bounding HLW form with respect to shielding evaluations.

Rationale: The gamma and neutron intensities for the SRS HLW canister containing design-basis glass are more intense than those of the rest of the HLW glass forms, including HLW glass developed at the Hanford site (CRWMS M&O 2000 [DIRS 151947], Attachments V and VI).

Usage: This assumption is used in Sections 6.3.3, 7.2.3, and 7.2.4.

3.2 ASSUMPTIONS REQUIRING VERIFICATION

There are no assumptions that require further verification.

4. USE OF COMPUTER SOFTWARE

4.1 BASELINED SOFTWARE

4.1.1 MCNP

The MCNP computer code (CRWMS M&O 1998 [DIRS 154060]) was used to calculate gamma and neutron dose rates. The software specifications are as follows:

- Program Name: MCNP
- Version/Revision Number: Version 4B2LV
- Status/Operating System: Qualified/HP-UX 10.20
- Software Tracking Number: 30033 V4B2LV
- Computer Type: Hewlett Packard 9000 Series Workstations
- Central Processing Unit Number: 700887

The input and output files for the various MCNP calculations are contained on a compact disc (Attachment II) with the files documented in Attachment I.

The MCNP software was: (a) appropriate for fixed source calculations, (b) used only within the range of validation as documented throughout Briesmeister (1997 [DIRS 103897]) and CRWMS M&O (1998 [DIRS 102836], Sections 3.3 and 3.4), and (c) obtained from Software Configuration Management in accordance with appropriate procedures.

4.2 COMMERCIAL OFF-THE-SHELF SOFTWARE

4.2.1 MICROSOFT EXCEL 97 SR-2

- Title: Microsoft Excel
- Version/Revision Number: Microsoft® Excel 97 SR-2
- This software is installed on a personal computer running Microsoft Windows 2000 with Central Processing Unit Number 150423.

Standard functions of Microsoft Excel for Windows, Version 97 SR-2, are used in this calculation to display results in tabular form and to perform the mathematical operations described in the calculation. The user-defined formulas, inputs, and results are documented in sufficient detail to allow an independent repetition of computations. Microsoft Excel is an exempt software product according to the administrative procedure LP-SI.11Q-BSC, *Software Management,* (Sections 2.1.2 and 2.1.6). The Microsoft Excel files are contained on a compact disc (Attachment II) and documented in Attachment I.

5 REQUIREMENTS AND DESIGN CRITERIA

This section summarizes requirements and criteria that are relevant to this calculation, including:

- Project requirements, and design criteria and requirements
- Regulatory requirements.

5.1 PROJECT REQUIREMENTS AND DESIGN CRITERIA

5.1.1 Project Requirements

The *Project Requirements Document* (Canory and Leitner [DIRS 166275], p. 3-12) has identified the following project requirement:

Req. Number and Title: PRD-002/T-012 Performance Objectives for the Geologic Repository Operations Area Through Permanent Closure

Requirement Text: For complete requirements text, see 10 CFR 63.111 [DIRS 158535]

Organization: Design and Engineering

Rationale for Allocation: Regulation 10 CFR 63.111 [DIRS 158535] specifies the technical requirements for preclosure performance of the proposed repository in the areas of radiation protection, numerical (dose) guidelines for design objectives, preclosure safety analysis, performance confirmation, and retrievability of waste. It provides specific numerical guides for design objectives and general performance objectives. The Design and Engineering organization is responsible for designing and developing waste packages, surface facilities, and subsurface facilities that achieve the preclosure performance objectives described in this section.

Requirements source: 10 CFR 63.111 [DIRS 158535], *Performance Objectives for the Geologic Repository Operations Area Through Permanent Closure*

5.1.2 Project Design Criteria

Project design criteria have been established in the *Project Design Criteria* (PDC) *Document* (BSC 2004 [DIRS 171599]). Applicable criteria to this calculation concern shielding source terms, ALARA (as low as is reasonably achievable), and worker dose rate assessment requirements.

5.1.2.1 Shielding Source Term Criteria

The shielding source term criteria established in BSC (2004 [DIRS 171599], Section 4.9.1.4) require that radiation source terms be based on the bounding waste form and waste package type. Either a design basis or a bounding source term may be used for shielding design. The design basis source term shall cover a minimum of 95% of the total inventory, with provisions made available to accommodate the remaining 5%. The calculation uses bounding source terms.

5.1.2.2 ALARA Design Criteria and Goals

ALARA is a regulatory requirement for the geologic repository at Yucca Mountain. 10 CFR Part 63 [DIRS 158535] states that an objective of the geologic repository operations is compliance with 10 CFR Part 20 [DIRS 173165], which requires the use of procedures and engineering controls that result in occupational doses and doses to members of the public that are ALARA. The PDC document has established the following ALARA goals for workers and members of the public that can be used in the design process (BSC 2004 [DIRS 171599], Section 4.9.3.3):

- The ALARA goal for occupational worker doses is to minimize the number of individuals that have the potential of receiving more than 500 mrem/year TEDE.
- The ALARA design process is to ensure that collective dose is maintained ALARA.
- The individual doses to the onsite members of the public will be maintained ALARA below the annual TEDE limit of 100 mrem.
- The annual TEDE to any real member of the public in the general environment will be limited to the preclosure standard in 10 CFR 63.204 [DIRS 158535], as well as the annual effluent dose limit of 10 mrem in 10 CFR 20.1101(d) [DIRS 173165].

5.1.2.3 Worker Dose Assessment

Worker dose assessment for demonstration of regulatory compliance shall include annual doses for both normal operations and Category 1 event sequences in compliance with 10 CFR 63.111(b)(1) [DIRS 158535]. Annual TEDEs, including internal and external exposures, shall be calculated by summing the contribution from normal operations and frequency-weighted doses from Category 1 event sequences (BSC 2004 [DIRS 171599], Section 4.9.1.7).

5.1.3 Facility Description Documents

Facility design requirements applicable to this calculation have been identified in the facility description documents for the DTF, the FHF, and the CHF.

5.1.3.1 Dry Transfer Facility Description Document

BSC 2004 [DIRS 167232], Requirement 3.1.1.3.a:

> The shielding walls surrounding waste handling areas shall provide the same shielding functions during Category 1 event sequences as during normal operation.

BSC 2004 [DIRS 167232], Requirement 3.1.1.3.d:

> Floors and roofs over space potentially containing waste forms (bare fuel assemblies, canistered fuel, and canistered HLW) shall be designed to prevent structural failure in the event that a crane collapses or drops a load.

5.1.3.2 Fuel Handling Facility Description Document

BSC 2004 [DIRS 169630], Requirement 3.1.1.3.a:

> The shielding walls surrounding waste handling areas shall have shielding of sufficient thickness to protect workers during Category 1 event sequences.

BSC 2004 [DIRS 169630], Requirement 3.1.1.3.d:

> Floors and roofs over space potentially containing waste forms shall be designed to prevent structural failure in the event that a crane collapses or drops a load.

5.1.3.3 Canister Handling Facility Description Document

BSC 2004 [DIRS 168992], Requirement 3.1.1.3.4.11:

> Floors and roofs shall be evaluated and strengthened, if needed, to prevent any impairment to structural integrity in the event that a maintenance crane collapses or drops a load over a space potentially containing a waste form.

5.2 REGULATORY REQUIREMENTS

Regulatory requirements for the Monitored Geologic Repository at Yucca Mountain have been specified in 10 CFR Part 63 [DIRS 158535]. Radiation protection requirements have been specified in 10 CFR Part 20 [DIRS 173165].

5.2.1 10 CFR Part 63

10 CFR Part 63 [DIRS 158535] prescribes rules governing the licensing (including issuance of a construction authorization) of the DOE for disposal of HLW in a geologic repository at Yucca Mountain, Nevada. Performance objectives for the geologic repository operations area through permanent closure include compliance with requirements concerning protection against radiation exposures, releases of radioactive material, and numerical guides for design objectives (10 CFR 63.111(a) and (b) [DIRS 158535]). The rule also requires an evaluation of license application information in accordance with 10 CFR 63.21(c)(5) [DIRS 158535].

10 CFR 63.111(a)(1) [DIRS 158535] (regulatory text):

> The geologic repository operations area must meet the requirements of part 20 of this chapter (Chapter 10).

10 CFR 63.111(a)(2) [DIRS 158535] (regulatory text):

> During normal operations, and for Category 1 event sequences, the annual TEDE to any real member of the public located beyond the boundary of the site may not exceed the preclosure standard specified at §63.204 [DIRS 158535].

10 CFR 63.111 (b) (1) [DIRS 158535] (regulatory text):

> The geologic repository operations area must be designed so that, taking into consideration Category 1 event sequences and until permanent closure has been completed, the aggregate radiation exposures and the aggregate radiation levels in both restricted and unrestricted, and the aggregate release of radioactive materials to unrestricted areas, will be maintained within the limits specified in paragraph (a) of this section.

10 CFR 63.111(b)(2) [DIRS 158535] (regulatory text):

> The geologic repository operations area must be designed so that, taking into consideration any single Category 2 event sequence and until permanent closure has been completed, no individual located on, or beyond, any point on the boundary of the site will receive, as a result of a single Category 2 event sequence, the more limiting of a TEDE of 0.05 Sv (5 rem), or the sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem). The lens dose equivalent may not exceed 0.15 Sv (15 rem), and the shallow dose equivalent to skin may not exceed 0.5 Sv (50 mrem).

10 CFR 63.204 [DIRS 158535]:

DOE must ensure that no member of the public in the general environment receives more than an annual dose of 0.15 mSv (15 mrem).

10 CFR 63.21(c)(5) [DIRS 158535] (regulatory text):

The Safety Analysis Report must include a preclosure safety analysis of the geologic repository operations area, for the period before permanent closure, to ensure compliance with §63.111(a) [DIRS 158535], as required by §63.111(c) [DIRS 158535]. For the purpose of this analysis, it is assumed that operations at the geologic repository operations area will be carried out at the maximum capacity and rate of receipt of radioactive waste stated in the application.

5.2.2 10 CFR Part 20

Table 1 summarizes occupational dose limits and radiation dose limits for individual members of the public established in 10 CFR 20.1201 and 10 CFR 20.1301 [DIRS 173165], respectively. A licensee is required to conduct operations such that the values listed in Table 1 will not be exceeded. It should be noted that these requirements are included for completeness. The calculation does not use the requirements as direct input.

Table 1. Occupational Dose Limits and Radiation Dose Limits for Individual Members of the Public

Dose Type	Dose Standards	
	Occupational Dose Limits [a]	Dose Limits for Individual Members of the Public
Total effective dose equivalent	5 rem/year	100 mrem/year [b, d]
Total organ [e] dose equivalent from intake and external exposure	50 rem/year	N/A
Lens dose equivalent	15 mrem/year	N/A
Shallow-dose equivalent to the skin or to any extremity	50 rem/year	N/A
Total effective dose equivalent from radioactive material releases	N/A	10 mrem/year [c]

SOURCE: [a] 10 CFR 20.1201 [DIRS 173165], *Occupational Dose Limits for Adults*.
[b] 10 CFR 20.1301 [DIRS 173165], *Radiation Dose Limits for Individual Members of the Public*.
[c] 10 CFR 20.1101(d) [DIRS 173165].
Notes [d] Excludes radiation sources different from those generated in the geologic repository operations area.
[e] Other than the lens of the eye.

In addition to controlling radiation exposure and release, a licensee must use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA (10 CFR 20.1101 [DIRS 173165]).

6. INPUTS

This section summarized technical inputs to the calculation, including Category 1 and 2 event sequences and waste physical characteristics and radiation source terms. Waste forms that will be received for disposal at Yucca Mountain include CSNF, HLW, DOE SNF, and Naval SNF. Among these waste forms, the HLW, the DOE SNF, and the naval SNF arrive at the repository in sealed canisters that will be loaded directly into waste packages (WPs). Transportation casks may contain either CSNF (PWR and BWR [boiling water reactor] SNF) encased in a sealed dual-purpose canister (DPC) or bare CSNF assemblies. Bare CSNF assemblies and disposable canisters may undergo multiple transfer operations before packaging.

6.1 CATEGORY 1 EVENT SEQUENCES

A Category 1 event sequence is defined as an event sequence that is expected to occur one or more times before permanent closure of the geologic repository operations area (10 CFR 63.2 [DIRS 158535]). Category 1 event sequences identified in BSC (2005 [DIRS 171429], Table 51) involve drops or collisions of bare PWR and BWR SNF assemblies. The numbers of assemblies affected and the annual frequencies of drops and collisions for the repository are summarized in Table 2. It should be noted that the values have been determined for the maximum rate of receipt of radioactive waste (refer to Section 5.2.1), and accounting for mitigation of radiological releases by the HVAC (heating, ventilation, and air-conditioning) system (BSC 2005 [DIRS 171429], Section 6.3.1.3).

Table 2. Category 1 Event Sequences

Event Sequence Identifier	Description	Annual Event Frequency for Consequence Analysis	Bounding Material at Risk
GET-03D	Drop of a CSNF assembly in the DTF or FHF	0.50 drops/y	2 PWR or 2 BWR assemblies
GET-03B	Collision involving a CSNF assembly in the DTF or FHF	0.50 collisions/y	1 PWR or 1 BWR assembly

SOURCE: BSC 2005 [DIRS 171429], Table 51.

6.2 CATEGORY 2 EVENT SEQUENCES

A Category 2 event sequence is defined as an event sequence that has at least one chance in 10,000 of occurring before permanent closure (10 CFR 63.2 [DIRS 158535]). Category 2 internal event sequences that have been identified in BSC (2005 [DIRS 171429], Table 53) involve drops or collisions in the surface facilities resulting in radioactive releases from CSNF assemblies, HLW, or naval SNF. Table 3 lists the Category 2 internal event sequences along with the estimated number of occurrences over the life of the repository and the amount of material at risk for each of the Category 2 event sequences.

Table 3. Category 2 Event Sequences

Internal Event Sequence Identifier	Description of Internal Event Sequence	Expected Number of Occurrences Before Permanent Closure	Bounding Material at Risk
GET-01A	Drop of a transportation or transfer cask without impact limiters in the TCRRF, DTF, CHF, or FHF	5.7E-01	74 BWR or 36 PWR assemblies, 5 HLW canisters, or 1 naval canister
GET-02B	Drop of inner lid of a transportation or transfer cask, site-specific cask, or WP into a transportation cask, site-specific cask, or WP in the DTF, CHF, or FHF	5.7E-01	74 BWR or 36 PWR assemblies, 5 HLW canisters, or 1 naval canister
GET-03E	Drop of a CSNF assembly during transfer in the DTF or FHF combined with HVAC failure	9.7E-02	2 BWR or 2 PWR assemblies
GET-03C	Collision involving a CSNF assembly during transfer in the DTF or FHF combined with HVAC failure	9.7E-02	1 BWR or 1 PWR assembly
GET-04B	Drop or collision of handling equipment onto a CSNF assembly in the DTF or FHF	9.7E-02	1 BWR or PWR assembly
GET-05B	Drop of a canister during transfer by crane in the CHF, DTF, or FHF	5.7E-01	2 HLW, 1 naval, or 1 DPC canister
GET-06B	Drop of handling equipment onto a canister in the CHF, DTF, or FHF	5.7E-01	1 HLW, 1 naval, or 1 DPC canister
GET-07B	Drop of an unsealed WP in DTF, CHF, or FHF	8.6E-02	44 BWR or 21 PWR assemblies, 1 naval canister, or 5 HLW canisters
GET-09C	Drop of a WP with a known closure defect in the CHF, DTF, or FHF	2.6E-02	44 BWR assemblies, 21 PWR assemblies, 1 naval canister, or 5 HLW canisters
GET-10C	Drop of a CSNF assembly in the dry remediation area of the DTF	4.9E-01	2 BWR or 2 PWR assemblies
GET-10B	Collision involving a CSNF assembly during dry remediation activities in the DTF	4.9E-01	1 BWR or 1 PWR assembly
GET-11B	Drop of a canister from a crane during WP remediation or dry remediation activities in the DTF	5.7E-02	2 HLW, 1 naval, or 1 DPC canister (74 BWR or 36 PWR assemblies)
GET-12B	Drop of CSNF assembly transfer/handling equipment onto CSNF assemblies in the DTF remediation area pool	4.9E-03	1 BWR or 1 PWR assembly

Table 3. Category 2 Event Sequences (Continued)

Internal Event Sequence Identifier	Description of Internal Event Sequence	Expected Number of Occurrences Before Permanent Closure	Bounding Material at Risk
GET-13A	Drop of a loaded transportation cask or site-specific cask in the DTF wet remediation area	1.1E-01	74 BWR or 36 PWR assemblies, 5 HLW canisters, or 1 naval canister
GET-15C	Drop of an empty or full canister for damaged CSNF assemblies (empty or full) onto or against a cask or basket in the DTF remediation pool	4.9E-02	2 BWR or 2 PWR assemblies
GET-15B	Collision of an empty or full canister for damaged CSNF assemblies (empty or full) onto or against a cask or basket in the DTF remediation pool	4.9E-02	1 BWR or 1 PWR assembly
GET-16C	Drop of a CSNF assembly in the DTF remediation pool	4.9E-01	2 BWR or 2 PWR assemblies
GET-16B	Collision involving a CSNF assembly in the DTF remediation pool	4.9E-01	1 BWR or 1 PWR assembly
GET-17B	Drop of a filled SNF basket from the spent fuel transfer machine onto the DTF remediation pool floor	5.4E-02	16 BWR or 9 PWR assemblies
GET-18B	Drop of handling equipment into or against an opened WP filled with CSNF during dry remediation activities in the DTF	4.9E-03	1 BWR or 1 PWR assembly
GET-19B	Drop of handling equipment into an open WP loaded with DOE canister or a naval canister during dry remediation activities in the DTF	3.6E-02	1 HLW canister or 1 naval canister
GET-20B	Drop of the severed DPC lid back onto the DPC in the DTF	1.0E-01	74 BWR or 36 PWR assemblies

SOURCE: BSC 2005 [DIRS 171429], Table 53.

6.3 WASTE FORMS

6.3.1 CSNF

This section describes the input information used to perform the MCNP calculations for CSNF described in Sections 7.1.1 and 7.2.1.

The PWR SNF assembly used in this calculation is B&W 15x15 Mark B (Assumption 3.1.2). For this assembly type, the number of fuel rods per assembly is 208 (DOE 1987 [DIRS 132333], p. 2A-33), the fuel pellet diameter is 0.3686 in. (0.93624 cm) (DOE 1987 [DIRS 132333], p. 2A-34), and the weight of fuel pellets per fuel rod is 5.58 lbs (2.53105 kg) (DOE 1987 [DIRS 132333], p. 2A-34). The assembly transverse dimension is 8.536 in. (21.68144 cm) (DOE 1987 [DIRS 132333], p. 2A-31). Table 4 presents the lengths of the assembly regions used in the MCNP calculations. Table 5 presents the assembly isotopic composition for unirradiated UO_2 pellets with a 5-wt% U-235 enrichment. The assembly region smeared compositions, in atoms/barn·cm, are presented in Table 6 (refer to Assumption 3.1.3).

Table 4. Assembly Source Region Lengths

Assembly Region	Length (cm)
Active Fuel	360.172
Plenum region	30.1752 [a]
Bottom end fitting	10.16 [b]
Top end fitting	20.1803 [c]

SOURCE: DOE 1987 [DIRS 132333], p. 2A-33.
Notes: [a] Fuel rod length – Active fuel length.
[b] Assumption 3.1.4.
[c] Fuel assembly length – Fuel rod length – Bottom end fitting length.

Table 5. Unirradiated UO_2 Pellet Composition

Isotope/Element	Weight/Assembly [a]	Weight Fraction [b]
U-235	23181.5000	4.40E-02
U-234	205.0600	3.90E-04
U-236	106.6349	2.03E-04
U-238	440136.8051	8.36E-01
O	62828.4	1.19E-01

SOURCE: [a] BSC 2003 [DIRS 163936], Attachment III, *DF#1_II\spreadsheets\WP_transporter\mcnp_rad_inputs.xls*, worksheet *atom_den*).
[b] Calculated in spreadsheet *calculations.xls*, worksheet *composition*.

A hypothetical radiation source consisting of 36 bare PWR SNF assemblies is used as a limiting CSNF source resulting from Category 2 event sequences listed in Table 3 (refer to Section 7.2). The radius of homogenization for the 36 PWR assemblies is 88.1253 cm, which is the inner radius of an NAC (Nuclear Assurance Corporation)-MPC (multi-purpose canister) cavity (Thompson 2000 [DIRS 171546], pp. 1.1-4, 1.2-2, and 1.2-20). This canister accommodates 36 PWR assemblies. The region smeared compositions, in atoms/barn·cm, for 36 PWR assemblies are presented in Table 6 (refer to Assumption 3.1.3).

Table 6. Atom Densities for Fuel Region Smeared Compositions

Element /Isotope	Atom Density per Smeared Region (atoms/barn·cm)							
	Active Fuel [a]		Bottom End Fitting		Plenum		Top End Fitting	
	Assembly	Cavity [b]	Assembly	Cavity [b]	Assembly	Cavity [b]	Assembly	Cavity [b]
U-235	3.5080E-04	2.4332E-04	N/A	N/A	N/A	N/A	N/A	N/A
U-234	3.1164E-06	2.1616E-06	N/A	N/A	N/A	N/A	N/A	N/A
U-236	1.6068E-06	1.1145E-06	N/A	N/A	N/A	N/A	N/A	N/A
U-238	6.5763E-03	4.5615E-03	N/A	N/A	N/A	N/A	N/A	N/A
Ni	1.5292E-04	1.0607E-04	3.2212E-03	2.2343E-03	3.8865E-04	2.6958E-04	1.9712E-03	1.3673E-03
Cr	7.2742E-05	5.0456E-05	4.4279E-03	3.0713E-03	1.7314E-04	1.2010E-04	2.4960E-03	1.7313E-03
Fe	7.0977E-05	4.9232E-05	1.2726E-02	8.8271E-03	1.6624E-04	1.1531E-04	7.0021E-03	4.8569E-03
Nb	9.6141E-06	6.6686E-06	9.0424E-05	6.2721E-05	2.4356E-05	1.6894E-05	6.3033E-05	4.3722E-05
Mo	5.5406E-06	3.8432E-06	3.2023E-04	2.2212E-04	1.4036E-05	9.7361E-06	1.7511E-04	1.2146E-04
Ti	3.2769E-06	2.2730E-06	3.0821E-05	2.1378E-05	8.3016E-06	5.7583E-06	2.1485E-05	1.4903E-05
Al	3.2297E-06	2.2402E-06	3.0377E-05	2.1070E-05	8.1820E-06	5.6753E-06	2.1175E-05	1.4688E-05
Co	2.9573E-06	2.0513E-06	2.7815E-05	1.9293E-05	7.4920E-06	5.1967E-06	1.9389E-05	1.3449E-05
Mn	1.1103E-06	7.7016E-07	2.9826E-04	2.0688E-04	3.1073E-06	2.1553E-06	1.6587E-04	1.1505E-04
Si	2.1719E-06	1.5065E-06	7.5819E-04	5.2590E-04	5.7183E-06	3.9664E-06	4.0318E-04	2.7966E-04
Cu	8.2280E-07	5.7072E-07	7.7387E-06	5.3678E-06	2.0844E-06	1.4458E-06	5.3945E-06	3.7418E-06
C	1.1609E-06	8.0521E-07	3.7878E-05	2.6273E-05	3.0419E-06	2.1100E-06	2.1977E-05	1.5244E-05
S	8.1528E-08	5.6550E-08	9.4376E-07	6.5462E-07	2.1411E-07	1.4851E-07	8.7305E-07	6.0557E-07
P	8.4403E-08	5.8544E-08	1.0686E-06	7.4121E-07	2.2557E-07	1.5646E-07	1.0791E-06	7.4850E-07
B-10	1.9249E-08	1.3351E-08	1.8104E-07	1.2557E-07	4.8764E-08	3.3824E-08	1.2620E-07	8.7536E-08
B-11	7.7478E-08	5.3741E-08	7.2871E-07	5.0545E-07	1.9628E-07	1.3615E-07	5.0797E-07	3.5234E-07
Sn	5.0015E-05	3.4692E-05	N/A	N/A	4.9736E-05	3.4499E-05	N/A	N/A
Zr	4.4033E-03	3.0542E-03	N/A	N/A	4.3788E-03	2.2072E-05	N/A	N/A
O	1.4003E-02	9.7131E-03	N/A	N/A	3.1822E-05	3.0373E-03	N/A	N/A
N	N/A	N/A	1.3507E-06	9.3689E-07	5.7741E-08	4.0051E-08	2.5840E-06	1.7923E-06
Total	2.5715E-02	1.7837E-02	2.1981E-02	1.5247E-02	5.2655E-03	3.6523E-03	1.2371E-02	8.5810E-03

SOURCE: BSC 2003 [DIRS 163936], Attachment III, *DF#1_II\spreadsheets\WP_transporter\mcnp_rad_inputs.xls*, worksheet *atom_den*.

Notes: [a] Fresh fuel composition (Assumption 3.1.5), 5.0 wt% U-235.
[b] Values obtained in spreadsheet *calculations.xls*, worksheet *composition*, as follows: assembly atom density X number of assemblies X assembly area / homogenization area (Harmon et al. 1994 [DIRS 154532], p. B-2).

This calculation uses a bounding radiation source term that consist of PWR SNF assemblies with a 5.0-wt% initial enrichment, a 80-GWd/MTU burnup, and a 5-year cooling time (BSC 2004 [DIRS 169061], Section 5.5). Radiation source terms for the bounding SNF are presented in Table 7, including gamma source terms for assembly bottom end fitting, active fuel, plenum, and top end fitting, and neutron source terms for the assembly active fuel.

A flat assembly axial burnup (power) profile has been used in BSC (2004 [DIRS 169061], Section 2) to generate gamma and neutron source terms. In reality, burnup is distributed non-uniformly along the fuel axial dimension with maximum values at the assembly middle section and minimum values at the fuel ends. A burnup axial profile usually is characterized by an axial peaking factor, which is defined as the ratio of assembly maximum burnup to assembly average burnup. The axial peaking factor varies with assembly type, burnup, and location in reactor, and tends to lower with increasing burnup. The peaking factors used in this calculation for the fuel axial gamma and neutron source distributions are 1.25 (Assumption 3.1.6) and 1.7 (Assumption 3.1.7), respectively.

Table 7. Gamma and Neutron Sources for the Bounding PWR SNF Assembly

Gamma Source Intensity (photons/s)					**Neutron Source Intensity (neutrons/s)**	
Upper Energy Boundary (MeV)	**Bottom End-Fitting Region [a]**	**Active Fuel Region [b]**	**Plenum Fuel Region [c]**	**Top End-Fitting Region [d]**	**Upper Energy Boundary (MeV)**	**Active Fuel Region [b]**
5.00E-02	5.94E+11	2.33E+15	5.28E+11	3.79E+11	2.00E+01	3.93E+07
1.00E-01	1.16E+11	6.44E+14	6.09E+10	7.43E+10	6.43E+00	4.43E+08
2.00E-01	2.83E+10	5.22E+14	3.52E+10	1.79E+10	3.00E+00	4.85E+08
3.00E-01	1.41E+09	1.48E+14	1.96E+09	8.91E+08	1.85E+00	2.76E+08
4.00E-01	1.90E+09	9.85E+13	5.86E+09	1.17E+09	1.40E+00	3.76E+08
6.00E-01	1.91E+09	1.53E+15	1.10E+11	7.41E+07	9.00E-01	4.11E+08
8.00E-01	4.35E+09	4.70E+15	5.95E+10	2.37E+09	4.00E-01	8.05E+07
1.00E+00	1.37E+11	7.08E+14	8.03E+09	7.66E+10	1.00E-01	0.00E+00
1.33E+00	3.38E+13	4.55E+14	1.74E+13	2.17E+13	**Total**	2.11E+09
1.66E+00	9.53E+12	1.30E+14	4.91E+12	6.12E+12	N/A	N/A
2.00E+00	1.87E+03	1.44E+12	9.19E+02	1.13E+03	N/A	N/A
2.50E+00	2.26E+08	2.49E+12	1.16E+08	1.45E+08	N/A	N/A
3.00E+00	3.51E+05	1.10E+11	1.81E+05	2.25E+05	N/A	N/A
4.00E+00	7.66E-08	1.39E+10	1.00E-08	4.16E-08	N/A	N/A
5.00E+00	0.00E+00	7.09E+07	0.00E+00	0.00E+00	N/A	N/A
6.50E+00	0.00E+00	2.85E+07	0.00E+00	0.00E+00	N/A	N/A
8.00E+00	0.00E+00	5.58E+06	0.00E+00	0.00E+00	N/A	N/A
10.00E+00	0.00E+00	1.19E+06	0.00E+00	0.00E+00	N/A	N/A
Total	4.42E+13	1.13E+16	2.31E+13	2.84E+13	N/A	N/A

SOURCE: BSC 2004 [DIRS 169061], Attachment X, Files [a] *Waste.Stream.E2.R2.B14.cut,* [b] Waste.Stream.E2.R1.B14.cut, [c] Waste.Stream.E2.R3.B14.cut, [d] Waste.Stream.E2.R4.B14.cut.

6.3.2 Naval Long Canister

The Naval Nuclear Propulsion Program has provided physical dimensions and bounding gamma and neutron sources for the naval canister at two years after fuel discharge. The long naval SNF canister has an overall maximum length of 212 in. (538.48 cm) and a maximum outer diameter of 66.5 in. (168.91 cm) (DOE 2002 [DIRS 158398], Figure C-19). The photon and neutron currents exiting the top surface of the canister are irregular due to the presence of six 3-in. (7.62-cm) diameter bolt holes and an annular cutout for seal weld access near the outer radius of the canister (Naples 1999 [DIRS 109988], Enclosure 2, Tables 1 and 3). Tables 8 and 9 present the photon and neutron currents, respectively, exiting the surfaces of a naval SNF canister. The canister surfaces include the side surface over the assembly mid section, the top surface over the 3-in. bolt holes, the top surface 18 in. (45.72 cm) from center, the top surface over seal weld cover, and the bottom surface.

Table 8. Photon Current by Energy Group Exiting the Surface of the Naval Spent Fuel Canister

Photon Current (photons/cm^{2}s)						
Characteristic Energy (MeV)	Upper Limit Energy (MeV)	Top Surface 18 in. from Center	Top Surface Over a 3-in. Bolt Hole	Top Surface Over Seal Weld Cover	Side Surface Over Assembly Mid-Section	Bottom Surface
3.000E-01	4.000E-01	4.245E-06	3.835E-02	1.090E-03	1.237E+08	1.111E+04
6.300E-01	9.000E-01	3.593E+00	1.870E+03	1.303E+02	1.612E+10	2.597E+07
1.100E+00	1.350E+00	4.253E+00	4.895E+02	5.476E+01	1.876E+08	1.445E+06
1.550E+00	1.800E+00	6.929E+01	3.707E+03	5.211E+02	2.321E+08	3.956E+06
1.990E+00	2.200E+00	2.148E+02	7.454E+03	1.165E+03	1.048E+08	2.843E+06
2.380E+00	2.600E+00	4.023E+00	1.074E+02	1.825E+01	8.328E+05	2.966E+04
2.800E+00	3.000E+00	8.458E-01	1.849E+01	3.351E+00	9.222E+04	3.996E+03
3.500E+00	4.000E+00	6.051E-03	1.062E-01	2.072E-02	3.504E+02	1.754E+01
4.500E+00	5.000E+00	1.692E-04	2.464E-03	5.109E-04	5.520E+00	3.225E-01
5.500E+00	6.000E+00	2.078E-10	2.716E-09	5.830E-10	4.685E-06	3.115E-07
6.750E+00	1.000E+01	2.147E-07	2.621E-06	5.751E-07	4.144E-03	2.812E-04

SOURCE: Naples 1999 [DIRS 109988], Enclosure 2, Table 3.

Table 9. Neutron Current by Energy Group Exiting the Surface of the Naval Spent Fuel Canister

Neutron Current (neutrons/cm^{2}s)					
Upper Energy Boundary (MeV)	**Top Surface Mid Radius From Center**	**Top Surface Over a 3-in. Bolt Hole**	**Top Surface Above Seal Weld Cover**	**Side Surface Over Assembly Mid Sections**	**Bottom Surface**
2.117E+01	2.385E-05	1.494E-04	3.449E-05	3.644E-02	2.343E-03
1.284E+01	1.098E-04	7.374E-04	1.643E-04	2.242E-01	1.302E-02
1.000E+01	3.460E-04	2.486E-03	5.350E-04	9.292E-01	4.963E-02
7.790E+00	6.072E-04	4.959E-03	1.003E-03	2.600E+00	1.204E-01
6.070E+00	1.035E-03	9.057E-03	1.771E-03	5.663E+00	2.466E-01
4.720E+00	7.116E-03	4.284E-02	8.441E-03	2.475E+01	1.140E+00
2.860E+00	2.832E-02	1.912E-01	3.985E-02	5.713E+01	3.866E+00
1.740E+00	8.722E-01	3.350E+00	8.059E-01	1.937E+02	3.043E+01
8.210E-01	9.297E+00	2.427E+01	5.985E+00	2.875E+02	9.573E+01
3.897E-01	2.202E+01	4.756E+01	1.219E+01	2.426E+02	1.225E+02
1.830E-01	1.695E+01	3.772E+01	8.947E+00	1.479E+02	8.188E+01
6.740E-02	1.071E+01	2.903E+01	5.782E+00	9.030E+01	4.982E+01
5.530E-03	3.974E+00	1.414E+01	2.261E+00	2.405E+01	1.495E+01
2.260E-05	6.532E-01	2.390E+00	3.711E-01	2.327E+00	2.089E+00
6.250E-07	4.281E-03	1.708E-02	2.458E-03	2.357E-02	1.334E-02

SOURCE: Naples 1999 [DIRS 109988], Enclosure 2, Table 1.

6.3.3 HLW Canisters

There are a short HLW canister, such as the SRS canister, and a long HLW canister, such as the Hanford canister (DOE 2002 [DIRS 158398], pp. C-20 and C-21). The SRS HLW glass possesses the largest gamma source intensity among potential HLW forms (Assumption 3.1.8). Table 10 presents the physical characteristics of HLW canisters. Tables 11, 12, and 13 present the glass elemental composition, the stainless steel 304L chemical composition, and the SRS HLW gamma and neutron source terms, respectively.

Table 10. Geometry and Material Specifications for the HLW Canister

Component	Material	Parameter	Value
HLW glass	See Table 11	Density [b]	2.65 g/cm^3
SRS canister	Stainless steel 304L [a]	Outer Diameter [c]	61 cm
		Wall Thickness [d]	0.9525 cm
		Height [c]	300 cm
		Weight	2000 kg [e]
		Glass Height [f]	275.16 cm
Hanford canister [g]	Stainless steel 304L	Outer Diameter	61 cm
		Height	450.5 cm
		Glass Height [h]	413.2 cm

SOURCE: Marra et al. 1995 [DIRS 101854], [a] p. 9, [b] p. 39, [c] p. 4, [d] p. 26.
[g] DOE 2002 [DIRS 158398], p. C-21.

Notes: [e] The SRS HLW photon and neutron source terms (see Table 13) have been generated for 2000 kg glass per canister (CRWMS M&O 2000 [DIRS 151947], Table 5-3). For consistency, this calculation uses the glass weight and chemical composition corresponding to these source terms. The maximum weight of glass is 1,950 kg (Marra et. al. 1995 [DIRS 101854], p. 39.
[f] Calculated as the ratio of weight to volume.
[h] Calculated considering that the ratios of glass height to canister height for the SRS and the Hanford canisters are identical.

Table 11. Elemental Composition of SRS HLW Glass

Element	Elemental Mass (kg) [a]	Weight Fraction [b]	Element	Elemental Mass (kg) [a]	Weight Fraction [b]
O	911.1312	0.4556	Li	41.1074	0.0206
Al	42.44698	0.0212	Mg	16.44643	0.0082
B	50.06458	0.0250	Mn	25.78786	0.0129
Ba	3.209565	0.0016	Na	132.551	0.0663
Ca	14.33668	0.0072	Ni	13.97007	0.0070
Cl	2.32847	0.0012	S	1.585873	0.0008
Cr	1.658672	0.0008	Si	472.2509	0.2361
Cs	2.286609	0.0011	Th	3.373208	0.0017
Cu	7.101018	0.0036	Ti	10.77904	0.0054
Fe	146.5277	0.0733	U	36.48976	0.0182
K	64.56705	0.0323	**Total**	2000	1.0000

SOURCE: [a] CRWMS M&O 2000 [DIRS 151947], Table 5-3.
Note: [b] Calculated in spreadsheet *calculations.xls*, worksheet *composition*.

Table 12. Chemical Composition of Stainless Steel 304L

Element [a]	Weight Percent Range [a]	Value Used
Carbon	0.030 (max)	0.03
Manganese	2.00 (max)	2.00
Phosphorus	0.045 (max)	0.045
Sulfur	0.030 (max)	0.03
Silicon	0.75 (max)	0.75
Chromium	18.00-20.00	19.00
Nickel	8.00-12.00	10.00
Nitrogen	0.10	0.10
Iron	Remainder	68.045
Density [b] = 7.94 g/cm^3		

SOURCE: [a] ASME 2004 [DIRS 171846], SEC II A SA-240, Table 1.
[b] ASTM G 1-90 [DIRS 103515], Table X1.

Table 13. Photon and Neutron Source Terms for a SRS HLW Glass Canister

Photon [a]		Neutron [b]	
Upper Energy Boundary (MeV)	Source Intensity (photons/s)	Upper Energy Boundary (MeV)	Source Intensity (neutrons/s)
0.05	1.29E+15	0.10	1.54E+05
0.10	3.89E+14	0.40	1.60E+06
0.20	3.02E+14	0.90	5.58E+06
0.30	8.58E+13	1.40	5.98E+06
0.40	6.27E+13	1.85	5.21E+06
0.60	8.55E+13	3.00	2.12E+07
0.80	1.34E+15	6.43	2.74E+07
1.00	2.08E+13	20.0	2.99E+05
1.33	2.91E+13	**Total**	6. 74E+07
1.66	6.18E+12		
2.00	4.86E+11		
2.50	2.70E+12		
3.00	1.91E+10		
4.00	2.15E+09		
5.00	5.20E+05		
6.50	2.09E+05		
8.00	4.09E+04		
10.0	8.67E+03		
Total	3.61E+15		

SOURCE: [a] CRWMS M&O 2000 [DIRS 151947], Attachment V, p. V – 1.
[b] CRWMS M&O 2000 [DIRS 151947], Attachment VI, p. VI – 1.

6.3.4 DOE SNF Standardized Canisters

A description of the DOE SNF standardized canisters is provided in DOE (2002 [DIRS 158398], Section 10.1 and pp. C-4 through C-8) and DOE (1999 [DIRS 140225], Appendix A). The DOE SNF canisters, which accommodate a large variety of DOE SNF types, occupy the central location of the 5 HLW/DOE SNF packaging configuration. The outer radius of the geometrical configuration is 94 cm (BSC 2004 [DIRS 166946]). A DOE SNF standardized canister is either 18″ (45.72 cm) or 24″ (60.96 cm) in diameter and either 10′ (300 cm) or 15′ (457 cm) long. This calculation uses the 10′-long canister with a diameter of 18″. The canister wall, which is made of SA-312, type 316L stainless steel, is 0.9525-cm thick. The canister contains a carbon steel impact plate at each end, which is neglected in calculations to produce conservative evaluations. The internal length is 254 cm with the bottom of the canister cavity starting at 22.95 cm from the canister base. Table 14 presents the chemical composition for stainless steel 316L.

Table 14. Chemical Compositions for Stainless Steel 316L

Element [a]	Weight Percent Range [a]	Value Used
Carbon	0.035 (max)	0.035
Manganese	2.00 (max)	2.00
Phosphorus	0.045 (max)	0.045
Sulfur	0.030 (max)	0.030
Silicon	1.00 (max)	1.00
Chromium	16.0-18.0	17.00
Nickel	10.0-15.0	12.50
Molybdenum	2.00-3.00	2.50
Iron	Remainder	64.89
Density [b]= 7.98 g/cm^3		

SOURCE: [a] ASME 2004 [DIRS 171846], SEC II A SA-312/SA-312M, Table 1.
[b] ASTM G 1-90 [DIRS 103515], Table X1.

Maximum gamma source terms at year 2010 for the DOE SNF are available in DOE (2004 [DIRS 169354]). Neutron source terms for representative DOE SNF types are provided in reports that describe fuel characteristics for disposal criticality analyses. Table 15 summarizes the bounding source intensities per canister for representative DOE SNF types, including ATR (advanced test reactor), Fermi, FFTF (Fast Flux Test Facility), FRR (foreign research reactor), FSVR (Fort Saint Vrain reactor), HFIR (high flux isotope reactor), Shippingport LWBR (light water breeder reactor), TMI (Three Mile Island), and TRIGA (training, research, isotopes General Atomics). Included also in Table 15 is the maximum average gamma source intensity per canister for each canister dimension.

Table 15. Maximum Source Intensity per Canister for Various DOE SNF

DOE SNF	Canister Dimensions [a]	Canister Loading Capacity [a]	Estimated Number of Canisters [a]	Gamma Source Intensity (photons/s) [a]	Neutron Source Intensity (neutrons/s)
ATR	18″x10′	20 fuel elements	197	5.567E+15	Not available
Fermi	18″x10′	3842 rods	7	3.155E+15	3842 / 140 x 1.39E+03 [b]
FFTF	18″x15′	5 assemblies	52	4.890E+15	5 x 5.532E+06 [c]
FRR	18″x15′	12 multi-pin clusters	127	4.308E+15	Not available
FSVR	18″x15′	5 elements	293	1.106E+15	5 x 2.483E+05 [d]
HFIR	24″x10′	3 fuel elements	147	2.794E+15	Not available
Shippingport LWBR	18″x15′	1 assembly	12	6.387E+15	5.496E+08 [e]
TMI-2 core debris	18″x15′	N/A	341	3.237E+14	Not available
TRIGA	18″x10′	111 elements	1	2.372E+14	111 x 1.86E+03 [f]
Maximum average	18″x10′	N/A	1399	2.971E+15	Not available
Maximum average	18″x15′	N/A	1442	2.030E+15	Not available
Maximum average	24″x10′	N/A	165	2.795E+15	Not available

SOURCE: [a] Attachment II, *CD\excel\calculations*.xls, worksheet *DOESNF_source*.
[b] DOE 1999 [DIRS 104110], Section 3.1 and Table B-1.
[c] Idaho National Engineering and Environmental Laboratory 2002 [DIRS 158820], Table B-1.
[d] Taylor 2001 [DIRS 154726], Table 2-16.
[e] DOE 1999 [DIRS 105007], Tables B-4 and B-5.
[f] DOE 1999 [DIRS 103891], Table B-3.

Among the DOE SNF types listed in Table 15, the ATR SNF and Shippingport LWBR SNF have maximum gamma source intensities per the 10′- and 15′-long DOE SNF standardized canisters, respectively. The neutron source terms are available only for certain DOE SNF types, as indicated in Table 15, and similar values are expected for any other fuel type. However, the neutron source intensity per canister is many orders (8 to 10) of magnitude smaller than the gamma source intensity. Consequently, neutron dose rate is negligible compared with the gamma dose rate and it will be neglected when calculating total dose rates (see Section 7.2.3).

A description of ATR fuel elements is provided in DOE (2004 [DIRS 169354], pp. A-103 and A-104). The ATR is a light-water reactor and an ATR fuel element contains 19 curved aluminum-clad fuel plates. The weight of Al in structural materials is 8,938.42 g/element. Uranium isotope loading per unirradiated fuel element is as follows: 1,075 g U-235, 69.93 g U-238, 13.87 g U-234, and 8.09 g U-236. Twenty fuel elements homogenized within canister cavity result in a mixture of 0.53-g/cm^3 mass density. The homogenized composition is as follows: 88.45 wt% Al, 10.64 wt% U-235, 0.69 wt% U-238, 0.14 wt% U-234, and 0.08 wt% U-236 (refer to Attachment II, *CD\excel\calculations.xls*, worksheet *composition*). Gamma source terms for the ATR are presented in Table 16. The peaking factor of the axial gamma source distribution has not been documented. In this calculation, the gamma source peaking factor is 1.25 since the ATR is a light-water reactor (Assumption 3.1.6), and the fuel composition is that of the fresh fuel (Assumption 3.1.5).

Table 16. Gamma Source Terms for ATR SNF

Average Energy (MeV)	Gamma Source Intensity (photons/s) [a]
0.015	3.897E+17
0.025	8.395E+16
0.0375	7.747E+16
0.0575	7.617E+16
0.085	4.856E+16
0.125	4.205E+16
0.225	4.116E+16
0.375	1.992E+16
0.575	2.736E+17
0.85	3.832E+16
1.25	7.130E+15
1.75	2.990E+14
2.25	6.272E+14
2.75	3.608E+12
3.5	4.003E+11
5	1.196E+06
7	1.344E+05
11	1.503E+04

SOURCE: DOE 2004 [DIRS 169354], p. C-17.
Note: [a] Gamma source intensity for 3948 fuel elements at year 2010.

6.3.5 MCO

An MCO will contain either the N Reactor SNF or the Shippingport PWR SNF (DOE 2004 [DIRS 169354], pp. C-291 and C-379). Table 17 presents gamma and neutron source intensities per MCO, which shows that an MCO containing the Shippingport PWR SNF has the highest gamma source intensity. The neutron source intensity is negligible compared with the gamma source intensity and the neutron contribution will be neglected when calculating total dose rates (see Section 7.2.4). Gamma source terms for the Shippingport PWR SNF are presented in Table 18.

Table 17. Maximum Source Intensity per MCO

DOE SNF	Canister Loading Capacity [a]	Estimated Number of Canisters [a]	Gamma Source Intensity (photons/s) [a]	Neutron Source Intensity (neutrons/s)
N Reactor	259 rods	400	4.987E+15	1.17E+07 [b]
Shippingport PWR	4 – 19 flat plates	18	7.594E+15	2.36E+06 [c]
Maximum average	N/A	419	5.093E+15	Not available

SOURCE: [a] Attachment II, *CD\excel\calculations*.xls, worksheet *DOESNF-source*.
[b] DOE 2000 [DIRS 150095], Table B-3.
[c] DOE 1999 [DIRS 104940], Table B-2.

Table 18. Gamma Source Terms for Shippingport PWR SNF

Average Energy (MeV)	Gamma Source Intensity (photons/s) [a]
0.015	4.666E+16
0.025	9.410E+15
0.0375	8.975E+15
0.0575	1.037E+16
0.085	5.221E+15
0.125	3.623E+15
0.225	4.477E+15
0.375	1.925E+15
0.575	4.478E+16
0.85	6.195E+14
1.25	6.085E+14
1.75	1.823E+13
2.25	2.935E+09
2.75	6.011E+09
3.5	6.191E+08
5	2.647E+08
7	3.050E+07
11	3.503E+06

SOURCE: DOE 2004 [DIRS 169354], p. C-379.
Note: Gamma source terms for 72 seed elements at year 2010.

A description of the MCO is provided in Garvin (2002 [DIRS 169141], pp. 1-3, F1-8, and F1-9). The MCO is a stainless steel (304L) cylindrical vessel fabricated from 24″ (60.96 cm)- diameter pipe with a minimum wall thickness of 0.42″ (1.07 cm) and an overall length of 139″ (353.06 cm). The MCO shell containing fuel sits on a 3.3″ (8.382 cm)-thick bottom plate. Additional components, including fuel baskets, a shield plug, and the bottom plate, are neglected in the MCNP calculations, which has the effect of producing conservative results. The packaging configuration consists of 2 MCO and 2 HLW canisters with an envelope outer diameter of 158.4 cm (DOE 2002 [DIRS 158398], p. C-16).

A description of the Shippingport PWR SNF is provided in DOE (1999 [DIRS 104940], pp. 5, 6, 7, C-5, and C-10). A Shippingport PWR seed element consists of four 19 flat plates, made of enriched UO_2-ZrO_2-CaO alloy, sandwiched between two Zircaloy-4 cover plates and four side strips. The uranium loading per seed element is 19.5 kg, of which 93.2 wt% is U-235. The weights of the oxides in a seed element are as follows: 23.309 kg UO_2, 30.906 kg ZrO_2, and 3.337 kg CaO. Zircaloy-4 in the cover plates, which weights 250.606 kg, is approximated with zirconium since this element composes 98% of the alloy (ASTM 811-97 [DIRS 137669], Table 2). The elemental/isotopic composition, which was determined in Attachment II, *CD\excel\calculations.xls*, worksheet *composition*, consists of 88.75 wt% Zr, 6.21 wt% U-235, 3.82 wt% O, 0.77 wt% Ca, and 0.45 wt% U-238. In this calculation, the gamma source peaking factor is 1.25 (Assumption 3.1.6), and the fuel composition is that of the fresh fuel (Assumption 3.1.5).

6.4 ENVIRONMENTAL DATA

The MCNP calculations described in Section 7.2 use the following data for dry air and tuff. Dry air has a density of 0.001204 g/cm^3 (Weast 1985 [DIRS 111561], p. F-10) and the composition presented in Table 19. Tuff chemical composition consists of SiO_2 and tuff mass density is 2.21 g/cm^3 (BSC 2003 [DIRS 166660], Table 8-7). As demonstrated in BSC (2005 [DIRS 172596], Section 6.3), the radiation component resulting from scattering processes in tuff is not sensitive to the minor tuff components, and SiO_2 adequately approximates tuff composition in shielding calculations.

Table 19. Air Composition

Molecular Composition [a]		
Constituent	**Constituent (Percent) by Volume**	
N_2	78.084	
O_2	20.946	
CO_2	0.033	
Ar	0.934	
Elemental Composition		
Element	**Atomic Weight (g) [b]**	**Weight Percent [c]**
N	14.00674	0.7552
Ar	39.948	0.0129
C	12.0107	0.0001
O	15.9994	0.2318

SOURCE: [a] Weast 1985 [DIRS 111561], p. F-156.
[b] Parrington et al. 1996 [DIRS 103896].
Note: [c] Values calculated in spreadsheet *calculations.xls*, worksheet *composition.*

The MCNP calculations described in Section 7.1.1 used the concrete chemical composition presented in Table 20.

Table 20. Concrete Chemical Composition

Element	Volumetric Weight (g/cm^3)	Element	Volumetric Weight (g/cm^3)
H	0.013	S	0.003
O	1.171	K	0.045
Na	0.04	Ca	0.194
Mg	0.006	Fe	0.029
Al	0.107	Total	2.35
Si	0.742		

SOURCE: ANSI/ANS-6.4-1997, Table 5.2, Concrete type: Ordinary, Designation: 04.

6.5 FLUX-TO-DOSE-RATE FACTORS

The MCNP calculations use the neutron and gamma flux-to-dose-rate factors documented in ANSI/ANS-6.1.1-1977 [DIRS 107016], which are listed in Table 21.

Table 21. Gamma and Neutron Flux-to-Dose Rate Conversion Factors

Gamma				Neutron	
Energy (MeV)	Conversion Factors (rem/h)/(photon/cm^2·s)	Energy (MeV)	Conversion Factors (rem/h)/(photon/cm^2·s)	Energy (MeV)	Conversion Factors (rem/h)/(neutron/cm^2·s)
0.01	3.96E-06	1.4	2.51E-06	2.50E-08	3.67E-06
0.03	5.82E-07	1.8	2.99E-06	1.00E-07	3.67E-06
0.05	2.90E-07	2.2	3.42E-06	1.00E-06	4.46E-06
0.07	2.58E-07	2.6	3.82E-06	1.00E-05	4.54E-06
0.1	2.83E-07	2.8	4.01E-06	1.00E-04	4.18E-06
0.15	3.79E-07	3.25	4.41E-06	1.00E-03	3.76E-06
0.2	5.01E-07	3.75	4.83E-06	1.00E-02	3.56E-06
0.25	6.31E-07	4.25	5.23E-06	1.00E-01	2.17E-05
0.3	7.59E-07	4.75	5.60E-06	5.00E-01	9.26E-05
0.35	8.78E-07	5.0	5.80E-06	1.0	1.32E-04
0.4	9.85E-07	5.25	6.01E-06	2.5	1.25E-04
0.45	1.08E-06	5.75	6.37E-06	5.0	1.56E-04
0.5	1.17E-06	6.25	6.74E-06	7.0	1.47E-04
0.55	1.27E-06	6.75	7.11E-06	10.0	1.47E-04
0.6	1.36E-06	7.5	7.66E-06	14.0	2.08E-04
0.65	1.44E-06	9.0	8.77E-06	20.0	2.27E-04
0.7	1.52E-06	11.0	1.03E-05	N/A	N/A
0.8	1.68E-06	13.0	1.18E-05	N/A	N/A
1.0	1.98E-06	15.0	1.33E-05	N/A	N/A

SOURCE: ANSI/ANS-6.1.1-1977 [DIRS 107016], pp. 4 and 5.

7. RESULTS

7.1 CONSEQUENCE EVALUATIONS FOR CATEGORY 1 EVENT SEQUENCES

This section analyzes the consequences of Category 1 event sequences and evaluates the direct exposures to workers and members of the public caused by the events. Category 1 event sequences involve CSNF (PWR and BWR) assembly drops and collisions (refer to Section 6.1, Table 2). The initiating events are associated with modes of failure specific to the Spent Fuel Transfer Machine (SFTM), which is a component of the SNF/HLW Transfer System. The SFTM is used to move CSNF in the fuel transfer cells located in the DTF and the FHF, as well as in the wet remediation area of the Remediation Facility.

The waste handling facilities and the SNF/HLW Transfer System have an important to safety classification (BSC 2005 [DIRS 171190], Table A-1). As such, associated structures, systems, and components are credited for prevention or mitigation in a Category 1 or Category 2 event sequence. A series of requirements in description documents for the system and facilities address safety functions. The most notable are the requirements that crane collapses and load drops shall not cause facility structural failure and that facilities shall have sufficient shielding for walls surrounding waste handling areas to protect workers during Category 1 event sequences (refer to Section 5.1.3).

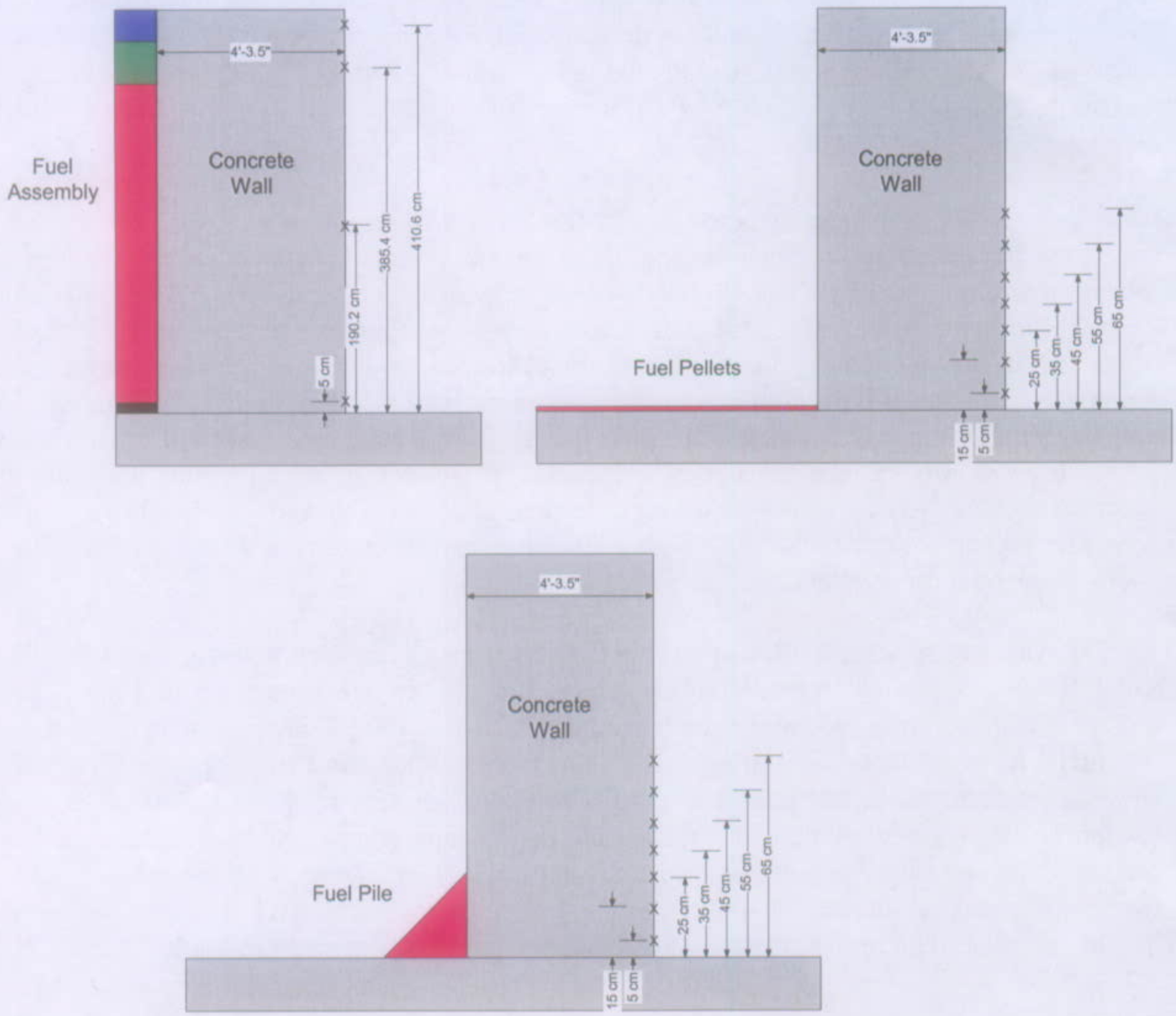
Failure modes and effects analyses were performed for the SFTM in a fuel transfer cell and the wet remediation area (Areva 2004 [DIRS 170104] and Rocher 2004 [DIRS 170261]). These analyses have identified main failures, effects, and prevention and mitigation features. The failure modes that cause fuel assembly collisions and drops include longitudinal/lateral/vertical movements that fail to stop, inadequate crane positioning, and incorrect gripping. Using low operating speeds, mechanical stops with impact limiters, and sensors and alarms, will mitigate the effects. The effect that any type of failure has on operations is immediate stop and initiation of recovery operations to restore SNF in the transfer area to a safe configuration. As stated in Section 1, this calculation does not address recovery operations following a Category 1 event sequence because recovery operational data have not been developed yet.

7.1.1 On-Site Doses during Category 1 Event Sequences

The shield thicknesses of barriers that separate workers from waste handling areas are based on normal operations and the shielding requirements established in the PDC document (BSC 2004 [DIRS 171599], Sections 4.9.1.3 and 4.9.1.4). The shielding design criterion for continuous occupational access areas, such as operating galleries, is a maximum dose rate of 0.25 mrem/h resulting from a bounding radiation source. It should be noted that the maximum dose rate in the operating gallery occurs on the inner surface of the separation wall between the operating gallery and the fuel transfer cell. However, dose rate in the operating gallery decreases with increasing distance from the wall because of geometrical attenuation. In dose rate calculations for normal operations in the DTF and FHF, the source geometrical configuration consists of a single fuel assembly at the minimum allowable distance (10′) from the separation wall between the fuel transfer cell and the operating gallery. For the wet remediation area in the Remediation Facility, the geometrical configuration consists of a fuel assembly at a minimum allowable distance from pool surface. The minimum distance is determined by the crane approach limiters. The required thickness for the operating gallery walls in the DTF and the FHF is 4′-3.5″ when considering an assembly clearance of 10′ (BSC 2004 [DIRS 171405], p. 74, and BSC 2004 [DIRS 169145], p. 41).

Category 1 event sequences include CSNF assembly collisions and drops. A collision event may cause the fuel assembly to touch the wall of an operating gallery, while a drop event may cause cladding failure that results in fuel pellets lying on the transfer cell floor. A sensitivity study is included to evaluate the effects of source geometry on dose rates and determine the maximum dose rate in the operating gallery. The study addresses the following three radiation sources: an intact fuel assembly, a thin layer of fuel pellets lying on the floor in a compact square geometry, and a pile of broken fuel pieces with a triangular cross section. The bounding geometrical configurations consist of the radiation sources in contact with the wall of the operating gallery. Drop events resulting in broken fuel pieces that form piles on the floor may generate higher dose rates due to a higher contribution from radiation scattering back from the floor of the operating gallery.

MCNP was used to determine dose rates in the operating gallery. Vertical cross sections through the MCNP geometrical representations of the three cases are illustrated in Figure 1. The compact configuration of lose fuel pellets is 360.172 cm x 152.95 cm x 0.93624 cm and the pile of fuel pieces is approximately 24 cm tall, 24 cm wide, and 180 cm long (spreadsheet *calculations.xls*, worksheet *composition*). The wall thickness in the calculations is 4′-3.5″, which is the required shielding thickness for normal operations. MCNP determined gamma, neutron, and secondary gamma fluxes averaged over surface segments on the exterior surface of the shield wall, as illustrated in Figure 1. The tally segments have a rectangular shape of a 100-cm^2 area. The results of the MCNP calculations are summarized in Table 22.



Note: Tally segment locations indicated on the drawing. Drawing not to scale.

Figure 1. Vertical Cross Sections of the MCNP Representations for Category 1 Event Sequences

Table 22. Dose Rates during Category 1 Event Sequences

Fuel Assembly								
	Gamma [a]		Neutron [a]		Secondary Gamma [a]		Total [b]	
Distance (cm) [c]	Dose Rate (rem/h)	Relative Error [e]	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Relative Error
5	5.28E-04	0.065	4.25E-05	0.113	2.39E-05	0.113	5.95E-04	0.058
190.2	6.91E-04	0.085	7.01E-05	0.085	5.03E-05	0.099	8.11E-04	0.073
385.4	5.78E-04	0.064	2.14E-05	0.124	1.97E-05	0.174	6.19E-04	0.060
410.6	4.33E-04	0.065	9.47E-06	0.179	3.06E-05	0.742	4.73E-04	0.077
Square Arrangements of Pellets on the Floor of Operating Gallery								
	Gamma [a]		Neutron [a]		Secondary Gamma [a]		Total [b]	
Distance (cm) [d]	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Relative Error
5	2.50E-04	0.058	3.68E-05	0.066	2.05E-05	0.090	3.07E-04	0.048
15	4.05E-04	0.113	3.70E-05	0.071	3.02E-05	0.183	4.73E-04	0.098
25	4.41E-04	0.051	4.00E-05	0.065	2.69E-05	0.164	5.07E-04	0.045
35	5.19E-04	0.061	3.86E-05	0.066	2.54E-05	0.122	5.83E-04	0.055
45	5.60E-04	0.098	3.74E-05	0.059	2.20E-05	0.095	6.19E-04	0.089
55	4.70E-04	0.039	3.84E-05	0.072	3.07E-05	0.192	5.39E-04	0.036
65	4.40E-04	0.053	3.18E-05	0.069	2.18E-05	0.138	4.94E-04	0.047
Triangular-Cross Section Fuel Pile								
	Gamma [a]		Neutron [a]		Secondary Gamma [a]		Total [b]	
Distance (cm) [d]	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Relative Error
5	8.81E-04	0.062	1.90E-04	0.0386	1.70E-04	0.0943	1.24E-03	0.046
15	8.47E-04	0.067	1.71E-04	0.0435	1.81E-04	0.1094	1.20E-03	0.050
25	7.09E-04	0.041	1.45E-04	0.0415	1.78E-04	0.1634	1.03E-03	0.040
35	6.64E-04	0.062	1.41E-04	0.0492	1.52E-04	0.1207	9.56E-04	0.048
45	4.86E-04	0.049	1.05E-04	0.0533	1.22E-04	0.1297	7.12E-04	0.041
55	3.33E-04	0.062	7.75E-05	0.0755	9.05E-05	0.128	5.01E-04	0.049
65	2.20E-04	0.056	6.72E-05	0.0745	6.16E-05	0.0947	3.49E-04	0.041

SOURCE: [a] Attachment II, *CD\mcnp\category1\outputs*.
Notes: [b] Values determined in spreadsheet *calculations.xls*, worksheet *results*.
[c] Distance from the assembly bottom where tally segments are located.
[d] Distance from the floor top where tally segments are located.
[e] Relative error is the ratio of the MCNP estimated standard deviation to sample mean.

The results of the MCNP calculations indicate that the maximum dose rates in an operating gallery from an intact assembly and a pile of broken fuel pieces are 0.81 mrem/h and 1.24 mrem/h, respectively.

The bounding source configurations addressed in the calculation are very unlikely and probabilities of occurrence for these configurations have not been determined. The resulting position of an affected fuel assembly will be more likely away from the wall and the fuel cladding will more likely suffer breaches only. Therefore, worker dose evaluations based on the bounding source configurations will be overly conservative, when associating the probabilities of

Category 1 event sequences consisting of generic drop or collision events that usually result in much lower dose rates. As described in Section 6.1, the frequencies of a drop event and of a collision event are 0.5 and 0.5 events per year, respectively; therefore, the frequency of a drop or a collision event is one event per year.

Although the placement of an intact assembly against the shield wall (see Figure 1) would provide sufficient conservatism for the calculated dose rate (i.e., 0.81 mrem/h), this calculation uses 1.24 mrem/h as a basis for the incremental dose rate due to the Category 1 event sequences to provide additional conservatism. Therefore, operators may be exposed to a maximum dose rate of 1.24 mrem/h for a maximum duration of eight hours (Assumption 3.1.1), which results in an annual worker dose of approximately 10 mrem (refer to Section 2.2, Equation 1). This value is small in comparison with the 100-mrem annual dose for the operating galleries for normal operations evaluated in BSC (2004 [DIRS 171772], Section 7.1.5). The total dose from normal operations and Category 1 event sequences is well below the ALARA goal of 500 mrem per year (refer to Section 5.1.2.2). Therefore, there is no impact on the existing shielding design, and no additional shielding is recommended.

7.1.2 Doses to Members of the Public during Category 1 Event Sequences

Category 1 event sequences may increase temporarily the dose rates in the controlled and unrestricted areas where members of the public, such as construction workers, have access or the off-site dose rates where a real member of the public may be located. Regulatory limits concerning annual doses to members of the public from normal and Category 1 event sequence consequences are 100 mrem within the site boundaries and 15 mrem for the off-site locations (refer to Section 5.2).

The shielding design criterion for controlled and unrestricted areas is a dose rate of 0.05 mrem/h when considering a bounding radiation source (BSC 2004 [DIRS 171599], Sections 4.9.1.3 and 4.9.1.4). As described in Section 7.1.1, a Category 1 event sequence may increase the dose rates in operating galleries to a maximum of 0.811 mrem/h. However, more than 4′ (BSC 2004 [DIRS 169317] and BSC 2004 [DIRS 171860]) of additional building interior and/or exterior walls separate an operating gallery from exterior. The additional walls reduce dose rate more than 1.3E+06 times, as determined in BSC (2004 [DIRS 171405], Table 19), to a negligible level. Therefore, Category 1 event sequences do not affect dose rates to members of the public located on site or beyond the site boundaries.

7.2 CONSEQUENCE EVALUATIONS FOR CATEGORY 2 EVENT SEQUENCES

Category 2 event sequences involve a drop and a breach of a canister, transportation cask, or waste package containing CSNF, HLW, or naval SNF (refer to Section 6.2, Table 3), which may effect an increase in external radiation levels. This section evaluates direct radiation dose to members of the public during a Category 2 event sequence. A member of the public may be located at the minimum distance from the geologic repository operations area, which is approximately 11 km (Data Tracking Number MO0001YMP00001.000 [DIRS 143142]).

Most of the Category 2 event sequences take place in fuel or canister transfer areas, which usually occupy inner locations in a facility general arrangement. Waste handling facilities at Yucca Mountain have been designated as Safety Category in the *Q-List* (BSC 2005 [DIRS 171190], Table A-1). As such, structures, systems, and components associated with waste handling facilities are credited for prevention or mitigation in a Category 1 or Category 2 event sequence. Further, facility design prevents structural failure of floors and roofs following crane collapses or load drops (refer to Section 5.1.3). Although a Category 2 event sequence may result in elevated radiation levels that exceed the shielding design capacity of the waste-transfer cell walls, floor, or ceiling, the exterior facility structures provide additional shielding. For instance, structural analyses have determined exterior wall thickness that exceeds the shielding requirements. Consequently, dose levels outside a facility are maintained ALARA for these Category 2 event sequences.

Category 2 event sequences involving a drop and a breach of a transportation cask may result in a maximum external radiation dose rate of 1 rem/h at 1 m from the external surface of the cask (10 CFR 71.51(2) and 71.43(f) [DIRS 171308]). For cask operations that take place in an unshielded area, such as the TCRRF, the event sequences may result in elevated radiation levels at distant locations. However, for a Category 2 event sequence to result in significant direct radiation dose increase at distant locations, damage of shield structures must be so extensive that these structures lose their shielding function. Such shield structures include cask and waste package shells and/or facility shield walls. This calculation evaluates direct radiation dose to a member of the public from Category 2 event sequences using a hypothetical event that bounds every Category 2 event sequence. The hypothetical event would result in complete impairment of the waste outer shield and exposure to a maximum radiation source. The maximum radiation sources for affected waste forms are defined as follows: 36 PWR bare assemblies, one naval long canister, five HLW canisters and one DOE SNF canister in a 5 HLW/DOE SNF packaging configuration, and two MCOs and two HLW glass canisters in a 2 HLW/2 MCO packaging configuration. It should be noted that the DOE SNF canisters and the MCOs are included in this calculation for completeness. Drops and collisions involving these waste forms result in Beyond Category 2 event sequences (BSC 2005 [DIRS 171429], Section 7.2.4).

7.2.1 CSNF

The bounding CSNF radiation source resulting from Category 2 event sequences consists of 36 bare PWR assemblies, where 36 is the maximum number of assemblies affected by a Category 2 event sequence. This source may occupy the cavity volume of a NAC-MPC that accommodates 36 PWR SNF assemblies. The radiation source, material composition, and source volume dimensions for the MCNP calculations are described in Section 6.3.1. In the calculations, the radiation source is placed on the ground and is surrounded by a large volume of air that extends 2 km skywards and 4 km in the radial direction. The effect of surrounding air on radiation level at distant locations is a contribution from the skyshine radiation. The calculations specified point detectors located at 1 m, 100 m, and 500 m from the source, and volume detectors located at 1,000 m, 1,500 m, 2,000 m, and 3,000 m from the source. The volume detectors consist of annular air volumes (up to 2 m from the ground) and the elevation of the point detectors corresponds approximately to the source midplane.

Table 23 presents dose rates as a function of distance from 36 PWR SNF assemblies. The results show that the dose rate from direct radiation is negligible at distant locations from the bounding radiation source. However, dose depends on the time a member of the public is exposed to elevated levels of radiation. If a member of the public were located at 3 km from the source, which is much closer that in reality (~11 km), the direct radiation doses would be 2.88E-07 mrem, 6.91E-06 mrem, 2.07E-04 mrem, and 2.52E-03 mrem for exposure time of one hour, one day, one month, and one year, respectively. The results demonstrate that the direct radiation dose to members of the public is negligible for a reasonable recovery time, even in the case of a hypothetical event with the potential of creating elevated levels of radiation. The annual dose limit for members of the public is 15 mrem/h within the Yucca Mountain site (refer to Section 5.2.1).

Table 23. Dose Rate versus Distance from 36 PWR SNF Assemblies

	Gamma		Neutron		Secondary Gamma		Total [b]	
Distance (m) [a]	**Dose Rate (rem/h)**	**Relative Error**	**Dose Rate (rem/h)**	**Relative Error**	**Dose Rate (rem/h)**	**Relative Error**	**Dose Rate (rem/h)**	**Relative Error**
1 [c]	4.18E+04	0.002	2.90E+01	0.002	6.43E-02	0.011	4.18E+04	0.002
100	1.44E+01	0.021	1.60E-02	0.014	5.89E-05	0.043	1.44E+01	0.021
500	5.93E-02	0.028	1.34E-04	0.047	1.47E-06	0.108	5.94E-02	0.028
1,000	5.74E-04	0.023	2.29E-06	0.025	6.51E-08	0.087	5.76E-04	0.023
1,500	9.59E-06	0.039	7.24E-08	0.034	4.62E-09	0.208	9.67E-06	0.038
2,000	2.19E-07	0.059	2.52E-09	0.043	4.13E-10	0.442	2.22E-07	0.058
3,000	2.82E-10	0.094	6.31E-12	0.104	1.21E-13	0.294	2.88E-10	0.092

SOURCE: Attachment II, *CD\mcnp\category2\outputs*.
Notes: [a] Distance from the center of the source.
[b] Values determined in spreadsheet *calculations.xls*, worksheet *results*.
[c] Distance from source outer surface.

7.2.2 Naval Canister

A bounding radiation source for Category 2 event sequences involving a naval canister consist of an unshielded naval long canister above the ground that is surrounded by air. In the MCNP calculations, the canister was represented as a cylindrical body with reflective surfaces that contain uniformly distributed surface sources. Canister dimensions and the photon and neutron currents exiting the canister surfaces at two years after fuel discharge are described in Section 6.3.2. As seen from the values presented in Tables 8 and 9, the photon current exiting the canister radial surface is several order of magnitude greater that the photon currents exiting the other surfaces or the neutron currents. Consequently, only the photon current exiting the radial surface is specified in the MCNP calculations. The calculated dose rates at selected distances from a naval canister are presented in Table 24, and they bound dose rate values from 36 PWR assemblies (refer to Section 7.2.1). It should be noted that the doses for the naval canister at five years after fuel discharge are expected to be comparable to those of the CSNF.

The results show that the dose rate from direct radiation is negligible at distant locations from the bounding radiation source. However, dose depends on the time a member of the public is exposed to elevated levels of radiation. If a member of the public were located at 3 km from the source, which is much closer that in reality (~11 km), the direct radiation doses would be 7.87E-07 mrem, 1.89E-05 mrem, 5.67E-04 mrem, and 6.89E-03 mrem for exposure time of one hour, one day, one month, and one year, respectively. The results demonstrate that the direct radiation dose to members of the public is negligible for a reasonable recovery time, even in the case of a hypothetical event with the potential of creating elevated levels of radiation. The annual dose limit for members of the public is 15 mrem/h within the Yucca Mountain site (refer to Section 5.2.1).

Table 24. Dose Rate versus Distance from a Naval Canister

Distance (m) [a]	Dose Rate (rem/h)	Relative Error
1,000	7.35E-04	0.009
1,500	1.18E-05	0.018
2,000	3.44E-07	0.028
3,000	7.87E-10	0.044

SOURCE: Attachment II, *CD\mcnp\category2\outputs*.
Note [a] Distance from the center of the source.

7.2.3 5 HLW/DOE SNF Packaging Configuration

The bounding radiation source for HLW and DOE SNF standardized canisters consist of five HLW canisters and one DOE SNF canister arranged in a 5 HLW/DOE SNF packaging configuration (BSC 2004 [DIRS 166946]), which is illustrated in Figure 2. The MCNP representation consists of the 5 HLW/DOE SNF packaging configuration placed on the ground and surrounded by air. Further, shielding structures that may include waste package shell, lids, and basket, were neglected. Physical parameters for the radiation sources and the HLW and DOE SNF canisters are described in Sections 6.3.3 and 6.3.4, respectively, and the environmental data are provided in Section 6.4. In addition, dose rates from a single DOE SNF canister were determined.

The maximum radiation sources for the HLW canisters, the 10′-long DOE SNF canister, and the 15′-long DOE SNF canister are the SRS HLW glass, the ATR SNF, and the Shippingport LWBR SNF (refer to Section 6.3.4). The calculations were performed only for the short (10′) canister configuration because similar results would be obtained for the 15′-long configuration. This configuration would consist of five Hanford HLW canisters surrounding a 15′-long DOE SNF standardized canister containing Shippingport LWBR, which have a slightly lower gamma intensity per volume unit than the short 5 HLW/DOE SNF packaging configuration (refer to Assumption 3.1.8 and Table 15).

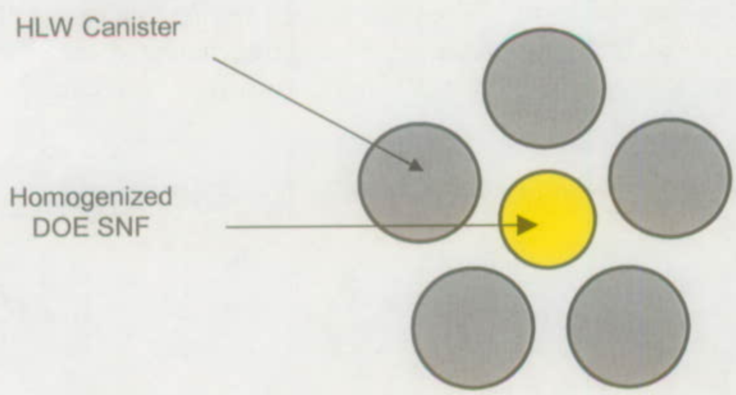


Figure 2. Illustration of the 5 HLW/DOE SNF Packaging Configuration

Doses rates at selected distances from the 5 HLW/DOE SNF packaging configuration are presented in Table 25. The values are bounded by the dose rates from a naval long canister listed in Table 24. Consequently, a Category 2 event sequence involving the HLW and/or DOE SNF canisters does not affect a member of the public.

Table 25. Dose Rate versus Distance from the 5 HLW/DOE SNF Packaging Configuration

	5 HLW/DOE SNF Packaging Configuration		Single DOE SNF canister	
Distance (m) [a]	**Dose Rate (rem/h)**	**Relative Error**	**Dose Rate (rem/h)**	**Relative Error**
1 [b]	2.71E+03	0.001	1.55E+04	0.001
100	1.09E+00	0.007	2.58E+00	0.003
500	3.23E-03	0.019	8.23E-03	0.018
1,000	2.25E-05	0.038	6.22E-05	0.015
1,500	3.98E-07	0.095	1.21E-06	0.032
2,000	9.62E-09	0.120	4.09E-08	0.042
3,000	3.12E-11	0.117	1.17E-10	0.063

SOURCE: Attachment II, *CD\mcnp\category2\outputs*.
Notes: [a] Distance from the center of the source.
[b] Distance from source outer dimension.

7.2.4 2 HLW/2 MCO Packaging Configuration

The hypothetical bounding radiation source for the MCO and HLW canisters consist of two Hanford canisters containing SRS glass (refer to Assumption 3.1.8) and two MCOs arranged in a 2 MCO/2 HLW waste package configuration (DOE 2002 [DIRS 158398], p. C-16), which is illustrated in Figure 3. Similarly to the MCNP dose rate calculations for the CSNF, this radiation source is placed on the ground and is surrounded by air. Further, all waste-package shielding structures, which include waste package shell, lids, and basket, are neglected. Physical parameters for the HLW and MCO are described in Sections 6.3.3 and 6.3.5, respectively, and the environmental data are provided in Section 6.4. Doses rate at selected distances from this radiation source are presented in Table 26. The values are bounded by the dose rates from a naval long canister listed in Table 24. Consequently, a Category 2 event sequence involving the MCO does not affect a member of the public.

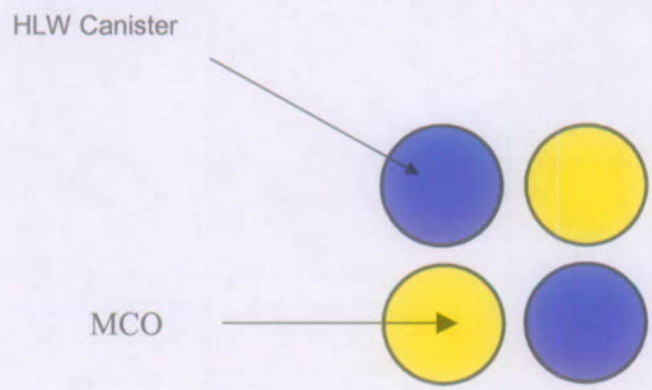


Figure 3. Illustration of the 2 HLW/2 MCO Packaging Configuration

Table 26. Dose Rate versus Distance from 2 HLW/2 MCO Canisters

Distance (m) [a]	Dose Rate (rem/h)	Relative Error
1 [b]	1.33E+04	0.001
100	5.80E+00	0.003
500	1.61E-02	0.012
1,000	8.08E-05	0.019
1,500	9.75E-07	0.044
2,000	2.13E-08	0.072
3,000	2.70E-11	0.126

SOURCE: Attachment II, *CD\mcnp\category2\outputs*.
Notes: [a] Distance from the center of the source.
[b] Distance from source outer surface.

8. CONCLUSIONS

The outputs of the calculations presented in this document are reasonable compared to the inputs, and the results are suitable for the intended use. The uncertainties were taken into account by consistently using a conservative approach that is the result of the methods and assumptions documented in Sections 2 and 3, respectively.

This document meets the objectives delineated in Section 1 that are based on *Yucca Mountain Review Plan* (NRC 2003 [DIRS 163274], Section 2.1.1.5).

The results of the calculation are the following:

- The maximum dose rate in continuous occupational access areas, such as the operating galleries in the DTF and the FHF, during Category 1 event sequences is 1.24 mrem/h. The direct radiation from Category 1 event sequences may result in a maximum worker dose of 10 mrem. The combined worker dose from normal operations and Category 1 event sequences is well below the ALARA goal (refer to Section 7.1.1).

- On-site dose to members of the public, such as construction workers, from direct radiation after a Category 1 event sequence is inconsequential (refer to Section 7.1.2).

- Dose from direct radiation after a Category 1 event sequence to members of the public located beyond the boundary of the site is inconsequential (refer to Section 7.1.2).

- Dose to members of the public from direct radiation after a Category 2 event sequence is inconsequential (refer to Section 7.2). This is demonstrated for CSNF, the naval canister, the HLW and the DOE SNF standardized canisters, and the MCO.

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10. ATTACHMENTS

This calculation document includes the following attachments:

ATTACHMENT I Listing of the Electronic Files Contained in Attachment II.

ATTACHMENT II One compact disc containing electronic files.

ATTACHMENT I

Listing of Electronic Files Contained in Attachment II

File attributes in the following list include file directory, date, time, size, and name.

Directory: CD\excel

Date	Time	File Size (bytes)	File Name
04/13/2005	04:20p	83,456	calculations.xls

Directory: CD\mcnp\category1\inputs

MCNP gamma (-p) and neutron (-n) input files for Category 1 event sequences involving a fuel assembly (a-), fuel pellets lying on the floor of the operating gallery (uo2-), and a pile of broken fuel pieces (uo3-).

Date	Time	File Size (bytes)	File Name
04/01/2005	10:01a	5,013	an
04/01/2005	10:01a	5,922	ap
04/13/2005	08:12a	2,872	uo2n
04/12/2005	02:28p	2,710	uo2p
04/12/2005	02:28p	2,913	uo3n
04/12/2005	03:09p	2,750	uo3p

Directory: CD\mcnp\category1\outputs

Date	Time	File Size (bytes)	File Name
04/01/2005	10:01a	79,250	an.io
04/01/2005	10:01a	55,375	ap.io
04/13/2005	08:12a	100,646	uo2n.io
04/12/2005	02:29p	60,997	uo2p.io
04/12/2005	03:09p	137,747	uo3p.io
04/13/2005	08:38a	330,369	uo3n.io

Directory: CD\mcnp\category2\inputs

MCNP input files for calculation of dose rate as a function of distance from PWR SNF (pwrn.i and pwrp.i), the 2 HLW/2 MCO packaging configuration (mco.i), a naval canister (navy.i), the 5 HLW/DOE SNF packaging configuration (hlw.i), and a DOE SNF canister (doe.i).

Date	Time	File Size (bytes)	File Name
03/07/2005	10:36a	7,155	pwrn.i
03/21/2005	11:55a	7,037	pwrp.i
03/21/2005	11:56a	8,306	mco.i
03/08/2005	09:04a	4,422	navy.i
03/21/2005	11:55a	9,798	hlw.i
03/21/2005	11:55a	9,143	doe.i

Directory: CD\mcnp\category2\outputs

Date	Time	File Size (bytes)	File Name
03/07/2005	10:36a	411,340	pwrn.io
03/21/2005	11:56a	343,875	pwrp.io
03/21/2005	11:56a	299,526	doe.io
03/08/2005	09:04a	34,518	navy.io
03/21/2005	11:56a	283,569	mco.io
03/21/2005	11:56a	310,664	hlw.io

OCRWM	SPECIAL INSTRUCTION SHEET file list 4-26-05 MBC	1. QA: QA Page 1 of 1

This is a placeholder page for records that cannot be scanned.

2. Record Date 04/19/2005	3. Accession Number Att: ENG.20050419.0013
4. Author Name(s) Georgeta Radulescu	5. Authorization Organization BSC

6. Title/Description
Direct Radiation Dose Consequence Calculation for Category 1 and 2 Event Sequences

7. Document Number(s) 000-00C-WHS0-00600-000-00A	8. Version Designator 00A
9. Document Type Calculation; Data	10. Medium CD

11. Access Control Code
PUB

12. Traceability Designator
000-00C-WHS0-00600-000-00A

13. Comments
~~CD - Proprietary Information~~ 4/21/05

The file list for this CD is located in Attachment I of this document.

Validation of Complete File Transfer.
All files copied.

This is an electronic attachment.

Software used is Excel.

THIS IS AN ELECTRONIC
ATTACHMENT

ATTACHMENT I

Listing of Electronic Files Contained in Attachment II

File attributes in the following list include file directory, date, time, size, and name.

Directory: CD\excel

Date	Time	File Size (bytes)	File Name
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Directory: CD\mcnp\category1\inputs

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04/12/2005	03:09p	2,750	uo3p

Directory: CD\mcnp\category1\outputs

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04/13/2005	08:38a	330,369	uo3n.io

Directory: CD\mcnp\category2\inputs

MCNP input files for calculation of dose rate as a function of distance from PWR SNF (pwrn.i and pwrp.i), the 2 HLW/2 MCO packaging configuration (mco.i), a naval canister (navy.i), the 5 HLW/DOE SNF packaging configuration (hlw.i), and a DOE SNF canister (doe.i).

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