

BSC

Design Calculation or Analysis Cover Sheet

1. QA: QA

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Complete only applicable items.

3. System Monitored Geologic Repository				4. Document Identifier 000-00C-MGR0-04200-000-00B			
5. Title Aging Facility and Site Worker Dose Assessment							
6. Group Engineering / Nuclear & Radiological							
7. Document Status Designation <input type="checkbox"/> Preliminary <input checked="" type="checkbox"/> Committed <input type="checkbox"/> Confirmed <input type="checkbox"/> Cancelled/Superseded							
8. Notes/Comments Rev 00A Edward Salisbury checked the document Dan Newell checked the MCNP5 files This calculation supersedes 000-00C-MGR0-03300-000-00A <i>TLK 1-31-08</i> <i>170-00C-HA00-00200-000-00A TLK 1-31-08</i> Rev 00B This calculation supersedes 000-00C-MGR0-04200-000-00A							
Attachments							Total Number of Pages
Attachment A – List of Files on Attached CD							3
Attachment B – Inhalation and Air Submersion Doses							3
Attachment C – Seismic Restraint Analysis							1
Attachment D – Attached Compact Disc (CD)							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. EGS (Print/Sign/Date)	16. Approved/Accepted (Print/Sign/Date)
00A	Initial Issue	94	94	Matthew Brenner	Dan Newell Edward Salisbury	Norman Kahler Charlotta Sanders	David B. Darling
00B	Updated the references and made various editorial modifications, corrected math in Tables 7.2, 7.9, 7.10, 7.12, 7.14, 7.15, 7.16, 7.17 and 7.18 and the results presented in conclusions.	94	94	Tom Karl <i>TLK</i> <i>01-30-08</i>	Edward Salisbury <i>Edward Salisbury</i> <i>01-30-08</i>	Norman Kahler <i>N. Kahler</i> <i>1-30-08</i>	David B. Darling <i>DB Darling</i> <i>1-30-08</i>

DISCLAIMER

The calculations contained in this document were developed by Bechtel SAIC Company, LLC (BSC) and are intended solely for the use of BSC in its work for the Yucca Mountain Project.

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ACRONYMS

ALARA	as low as is reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
AO	aging overpack
ASME	The American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B & W	Babcock & Wilcox
BOD	Basis of Design
BOP	Balance of Plant
BSC	Bechtel SAIC Company, LLC
BWR	boiling water reactor
CCCF	Central Control Center Facility
CD	compact disc
CFR	Code of Federal Regulations
CRCF	Canister Receipt and Closure Facility
CRWMS M&O	Civilian Radioactive Waste Management System Management & Operating Contractor
DCF	dose conversion factor
DHLW	defense high-level (radioactive) waste
DIRS	Document Input Reference System
DOE	U.S. Department of Energy
DPC	dual purpose canister
DU	Depleted Uranium
EPA	Environmental Protection Agency
FRG	Federal Guidance Reports
GROA	Geologic Repository Operations Area
GWd/MTU	gigawatt days per metric ton uranium
HAM	horizontal aging module
HEMF	Heavy Equipment Maintenance Building
HLW	high-level (radioactive) waste
HP	health physics
HPT	health physics technician
ICRP	International Commission on Radiological Protection
IHF	Initial Handling Facility
LANL	Los Alamos National Laboratory
MCNP	Monte Carlo N-Particle (transport code)
MeV	Million electron Volt
mrem/hr	millirem per hour

MTHM	metric tons of heavy metal
MTU	metric tons of Uranium
NRC	U.S. Nuclear Regulatory Commission
NUREG	Nuclear Regulatory Commission technical report designation
PLC	programmable logic controllers
PDC	Project Design Criteria
PWR	pressurized water reactor
RPM	Repository Project Management
SAR	Safety Analysis Report
SNF	spent nuclear fuel
SS	stainless steel
TAD	transportation, aging, and disposal
TBD	to be determined
TC	transportation cask
TEV	transport and emplacement vehicle
TEDE	total effective dose equivalent
UF	Utilities Facility
WNNRF	Warehouse and Non-Nuclear Receipt Facility
WP	waste package
wt%	weight percent
YMP	Yucca Mountain Project

1. PURPOSE

The purpose of this calculation is to evaluate radiation doses received by personnel during cask acceptance, cask movements through the Geologic Repository Operations Area (GROA), aging pad operations, and transport and emplacement vehicle (TEV) transports to the subsurface.

The calculations contained in this document were developed by Nuclear and Radiological Engineering of the Repository Project Management (RPM) organization and are intended solely for the use of the RPM in its work regarding facility design and operation. Yucca Mountain Project (YMP) personnel from Nuclear and Radiological Engineering should be consulted before use of the calculations for purposes other than those stated herein or use by individuals other than personnel in Nuclear and Radiological Engineering.

Per Reference 2.1.1 Sections 2.3 and 3.2.2, “Assumptions that will be met by operational control of activities such as worker dose assessments, do not require verification”. “Results and conclusions from preliminary or committed calculations or analyses may be used in other preliminary or committed calculations or analyses as design input”. “Engineering sketches and studies may be used as design input in preliminary calculations and in committed calculations provided the calculations are not used for procurement, fabrication or construction purposes”.

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- 2.1.3 IT-PRO-0011, Rev. 7. *Software Management*. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20070905.0007.
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- 2.2.2 CRWMS M&O 1998. *Calculation of the Effect of Source Geometry on the 21-PWR WP Dose Rates*. BBAC00000-01717-0210-00004 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990222.0059. (DIRS 102134)
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NOTE: Reference 2.2.4 is used as a direct input, but is not QA:QA. It is justified as acceptable for the intended use, as it is directly from the DOE and considered the best available data on the subject. The referenced values are valid for use since they are reliable and reasonable based on engineering judgment.

NOTE: Reference 2.2.2 is not designated as QA:QA, though it is still considered acceptable for use as it represents the best available data on the subject.

2.3 DESIGN CONSTRAINTS

- 2.3.1 10 CFR 71. 2006. Energy: Packaging and Transportation of Radioactive Material. Internet Accessible. (DIRS 176575)
- 2.3.2 10 CFR 20. 2006. Energy: Standards for Protection Against Radiation. Internet Accessible. (DIRS 176618)

2.4 DESIGN OUTPUT

The results of this design calculation will be worker doses throughout the cask receipt, GROA, and aging pad operations. These worker doses will then be used to determine if the “as low as is reasonably achievable” (ALARA) goals as noted in the *Project Design Criteria Document* (PDC), Section 4.10.3.3.1 (Reference 2.2.27) are met in the current designs.

3. ASSUMPTIONS

3.1 ASSUMPTIONS REQUIRING VERIFICATION

3.1.1 Annual Acceptance and Throughput

The following assumptions outline the annual acceptance and throughput values.

3.1.1.1 Annual Cask Acceptance

It is assumed that 500 rail casks, represented as TS125 rail casks, will be accepted annually during peak operations.

Rationale: Per the *Basis of Design (BOD) for the transportation, aging, and disposal (TAD) Canister-Based Repository Design Concept* (Reference 2.2.47, Section 2.2.1.2, and Section 2.2.1.3) the annual acceptance during full scale operations will be:

- Accept and receive 3,000 metric tons of heavy metal (MTHM) of commercial spent nuclear fuel (SNF) and high level waste (HLW)
- Accept and receive up to 24 naval SNF canisters
- Accept and receive 179 DOE SNF canisters and 763 defense high-level (radioactive) waste (DHLW) canisters

500 TS125 rail casks are representative of all casks accepted based on dose rate limits and physical dimensions. Approximately 300 rail casks containing commercial SNF and HLW are expected along with approximately 20 DOE SNF casks (9 canisters /cask), 150 DHLW casks (5 cans/cask), and 24 naval casks. All transportation casks accepted are required to meet the dose rate limits of 10 CFR Part 71.47 (Reference 2.3.1) regardless of the waste contained or configuration. Note that about 10% of the commercial SNF could be received in truck casks, which contain only about 2 MTHM, versus about 9 MTHM for rail casks, and so require more shipments per MTHM. (Reference 2.2.47, and Section 2.2.1.3). However, for the purpose of this analysis, a TS125 rail cask is used to represent all commercial SNF transportation casks. This is acceptable due to the smaller physical size of the truck cask which should result in a much shorter time required to process than estimated for rail casks in Table 3.2. This roughly compensates for the increased number of casks received if truck casks were included.

Usage: This assumption is used in Sections 4.3.4, 4.3.5, 6.2.1, and 6.2.2.1.

3.1.1.2 Annual Waste Package Placement

It is assumed that the maximum number of waste packages transported by the TEV in one year is 316 waste packages.

Rationale: This assumption is consistent with the *Basis of Design for the TAD Canister-Based Repository Design Concept* (Reference 2.2.47) for annual subsurface receipt. The acceptance outlined is:

- 24 naval waste packages
- 162 DOE SNF and HLW waste packages
- 1,300 MTHM of commercial SNF and HLW waste packages

Approximately 130 ($162 + 24 = 186$. $316 - 186 = 130$) are expected to be commercial SNF and HLW packages, with the rest consisting of the 24 naval and 162 DOE SNF and HLW waste packages described above.

Usage: This assumption is used in Sections 4.3.5, 6.2.2.2, and 7.1.2.2.

3.1.1.3 Annual Throughput of Canisters Requiring Aging

It is assumed that during peak operations the annual maximum number of canisters requiring placement on the aging pad will be 275 while the nominal number of canisters placed is 135. It is also assumed that during peak operation the number of canisters retrieved is equal to the number placed on the aging pads.

Rationale: 275 canisters requiring aging is an acceptable approximation of the maximum number when compared with the number of canisters to be accepted. However, it is expected that the nominal number of canisters requiring placement each year will be 135. It is conservative to assume that during peak operation the number of canisters on the aging pad ready for processing will be equal to the number accepted for aging.

Usage: This assumption is used in Sections 4.3.7, 6.2.4.1, 6.2.4.2, and 7.1.3.1.

3.1.2 GROA Layout

The following assumptions outline the non-nuclear facilities and their relation to the route that the waste is transferred over as well as the travel distances for these waste movements inside the GROA. The data in these assumptions is from Reference 2.2.8.

3.1.2.1 Non-Nuclear Facilities Distance from the Waste in the GROA

Distance of facilities from transport paths:

- The minimum clear distance between any permanently manned, non-nuclear waste handling facility and the edge of any roadway anticipated to be used for transport of waste is 75 feet. This includes site transporter and semi-trailer roadways.
- The minimum clear distance between any permanently manned, non-nuclear waste handling facility and the center line of any rail anticipated to be used for transport of waste is 75 feet. This includes standard gage and transport and emplacement vehicle (TEV) rail.
- Although the Site Plans reflect Fire Water Facility A (28A) within 75 feet of a roadway anticipated to be used for transport of waste, the facility will be relocated to ensure that the clear distance is at least 75 feet.
- The Aging Overpack Staging Facility (290) borders a site transporter pathway and is approximately 60 feet from the nearest center line of any rail anticipated to be used for transport of waste.
- The Emergency Diesel Generator Facility (26D) is approximately 30 feet from the nearest site transporter roadway and semitrailer roadway anticipated to be used for transport of waste.

Source: Memorandum from Balance of Plant (BOP) to Nuclear and Radiological Engineering (Reference 2.2.8)

Rationale: The facility locations are based on a memorandum from Balance of Plant to Nuclear and Radiological Engineering (Reference 2.2.8). This is considered acceptable because it bounds the most current layout plans.

Usage: This assumption is used in Sections 4.3.5.

3.1.2.2 Transportation Cask Travel Distance

It is assumed that the average distance traveled by a transportation cask from the cask receipt security station (30B) to a nuclear facility is equal to the distance from 30B to Canister Receipt and Closure Facility (CRCF-2).

Rationale: The distance from 30B to CRCF-2 is a conservative approximation of the average distance traveled. Transportation casks will travel various distances as they move to the nuclear facilities for processing, some shorter than the distance used, and some longer than the distances used. Based on the *Geologic Repository Operations Areas North Portal Site Plan* (Reference 2.2.31) the distance from 30B to CRCF-2 is sufficient for use in the worker dose estimates.

Usage: This assumption is used in Sections 4.3.5, 6.2.2.1, 7.1.2.1, and 7.1.2.2.

3.1.2.3 TEV Travel Distance

It is assumed that the average distance traveled by the TEV during waste transport is equal to the distance traveled from CRCF-2 to the north portal entrance.

Rationale: Based on the *Geologic Repository North Portal Site Plan* (Reference 2.2.31), the distance from CRCF-2 to the north portal entrance, 3100 ft per Reference 2.2.8, is a conservative approximation of the average travel distance. The TEV will travel various distances as it travels from the three CRCFs and the Initial Handling Facility (IHF) to the north portal entrance. The distance from CRCF-2 to the north portal is sufficient for use in the worker dose assessments.

Usage: This assumption is used in Sections 4.3.5, and 6.2.2.2.

3.1.3 Transportation Cask Travel Speed

It is assumed that the site locomotives and trucks moving the transportation casks through the GROA will be limited to a maximum speed of 5 miles per hour.

Rationale: This assumption is considered conservatively slow for transporting waste inside the GROA, and results in a conservative worker dose estimate.

Usage: This assumption is used in Sections 4.3.5, 7.1.2.1, and Table 7.4.

3.1.4 Aging Pad Loading Sequence

It is assumed that the aging pad's 4x4 arrays are loaded first across the rows and then down the length of the pad beginning at the end of the pad farthest from the entrance side of the aging facility. It is also assumed that there is a fully loaded pad adjacent to the pad being loaded.

Rationale: This process of loading would reduce the amount of radiation exposure to the workers by reducing the amount of time that they are close to multiple aging overpack (AOs) on the same pad. This is a reasonable operational practice to help reach ALARA goals. It is conservative however to assume that the pad adjacent to the one being loaded is full.

Usage: This assumption is used in Sections 4.3.7, and 6.2.4.2.

3.1.5 Resuspension Factor

It is assumed that the resuspension factor for surface contamination on an AO is $2 \times 10^{-6} \text{m}^{-1}$.

Rationale: The resuspension factor expected for outdoor situations is 1/20th of what is determined for dusty operations of $4 \times 10^{-5} \text{m}^{-1}$. This is considered acceptable for use as it is based on industry data such as the International Atomic Energy Agency study on non-fixed contamination of packages and conveyances and will be confirmed at a later time. This assumption will be tracked in Caltrac.

Usage: This assumption is used in Section 4.3.2 and Attachment B.

3.2 ASSUMPTIONS NOT REQUIRING VERIFICATION

3.2.1 Cask Receipt

The following assumptions concern all operations performed by Yucca Mountain personnel before a loaded cask leaves the cask receipt security station (30B) and enters the GROA.

3.2.1.1 Receipt Process Staffing

It is assumed that five work crews are required to perform 24 hour a day, seven days a week (24/7) operations at the 30B. Each work crew consists of the following worker categories:

- Security Personnel
- Operators (includes crane operators, riggers, laborers, inspectors, and field engineers)
- Health Physics Technicians (HPTs)

One complete cask receipt work crew consists of:

- One Security Guard
- Four Operators
- One HPT

It is assumed that each of the four operators receives the same dose during all tasks.

Rationale: At least three crews are needed to support 24/7 operations. However, three crews would not allow for weekend days off, holidays, vacation or sick leave. Five crews, each working 2,000 hours per year, would provide 10,000 hours per year coverage, which covers the number of hours per year required (8,760 hours in a year). The time and distance data are the average of the four operators, thus the dose is the same for all operators, since some operators are closer to the source during certain tasks while at other times the opposite is true. The use of average distances and times is considered reasonable operationally since worker dose would be controlled administratively to reach ALARA goals by minimizing the number of people who have the potential of receiving more than 500 mrem/year total effective dose equivalent (TEDE).

Usage: This assumption is used in Sections 4.3.4, 6.2.1, and 7.1.1.

3.2.1.2 Receipt Process Time Data

It is assumed that the process time data presented in Table 3.2 accurately represents the receipt processes at 30B.

Table 3.2 Time and Motion Data for the Cask Receipt Processes

Process	Time Required (min)
Receipt documentation outside of gate	10
Initial inspection outside of gate, Radiation and contamination survey	12
Railcar or trailer towed in YMP area using YMP equipment and personnel	10
Remove personnel barrier tie downs, attach lifting devices to barrier	20
Remove and store personnel barrier	45
Second inspection, radiation and surface contamination survey	20
Railcar or trailer towed into cask receipt security station	10
Final Inspection	15

Rationale: The data presented in Table 3.2 is based on input from operations and mechanical handling as no time and motion data is yet confirmed for cask receipt processes, however the personnel barrier removal data is based on data from the *Receipt Facility Time and Motion Study* (Reference 2.2.5). This is an acceptable approximation as the removal of the barriers would be similar regardless of where they are removed. Positioning and securing of a mobile crane will be required to remove personnel barriers. This task will take approximately 50 minutes to perform. The setup will take place without the cask present and is considered a support activity. All support activities not including the cask are included in the support only dose summary presented in Section 7.1.4.

Usage: This assumption is used in Sections 4.3.4, 6.2.1, 7.1.1, and Table 7.2.

3.2.1.3 Receipt Process Task Distances

Table 3.3 displays the average distances that workers in each category will be at from the surface of a transportation cask during the receipt process.

Table 3.3 Distances to a Loaded Cask during the Cask Receipt Processes

Process	Distance (m)		
	Security	Operators	HPTs
Receipt documentation outside of gate	1	-	5
Initial inspection outside of gate, radiation and contamination survey	5	-	1
Railcar or trailer towed in YMP area using YMP equipment and personnel	-	5	5
Remove personnel barrier tie downs, attach lifting devices to barrier	-	2	5
Remove and store personnel barrier	-	2	--
Second inspection, radiation and surface contamination survey	5	-	1
Railcar or trailer towed into cask receipt security station	-	5	5
Final Inspection	5	-	-

Rationale: The distances in Table 3.3 are based on input from operations and mechanical handling as no data is yet confirmed for these tasks, however the personnel barrier data is based on data in the *Receipt Facility Time and Motion Study* (Reference 2.2.5). The Operators distance is based on the average distance reported in the study while the HPT distance is directly from the study. These are acceptable approximations as the removal of the barriers would be similar regardless of where they are removed.

Usage: This assumption is used in Sections 4.3.4, 6.2.1, Table 7.2, and 7.1.1.

3.2.2 GROA

The following assumptions concern operations occurring from the time a loaded cask leaves the cask receipt security station (30B) and enters a nuclear facility.

3.2.2.1 Transportation Cask Staffing

It is assumed that one crew consisting of two operators and one HPT will be present during all loaded transportation cask movements in the GROA (i.e. from the cask receipt security station to the buffer areas and then to the nuclear facilities).

Rationale: It is reasonable to assume that both operations and health physics (HP) personnel will be present to ensure safe transport of all transportation casks through the GROA. Two operators will allow for full view of both side and ends of the rail car while one HPT is required to accompany the loaded cask to serve the function of the individual

who takes the precautions necessary to prevent the inadvertent exposure of workers, and members of the public to radiation in excess of the limit established in 10 CFR Part 20 (Reference 2.3.2).

Usage: This assumption is used in Sections 4.3.5, 6.2.2.1, and 7.1.2.1.

3.2.2.2 TEV Operations Staffing

It is assumed that one crew consisting of two operations and one HP personnel will be present during all TEV movements in the GROA (i.e. after the TEV leaves the nuclear facilities and before it enters the subsurface).

Rationale: Though the TEV will be remotely controlled from the Central Control Center Facility (CCCF) and cameras will be present for monitoring (Reference 2.2.49), it is reasonable to assume that both operations and HP personnel will be present to ensure safe transport of all waste package movements via TEV through the GROA. Two operators will allow for full view of both side and ends of the TEV while one HPT is required to accompany the TEV to serve the function of the individual who takes the precautions necessary to prevent the inadvertent exposure of workers, and members of the public to radiation in excess of the limit established in 10 CFR Part 20 (Reference 2.3.2).

Usage: This assumption is used in Sections 4.3.5, 6.2.2.2, and 7.1.2.2.

3.2.3 Aging Facility

The following assumptions are concerned with movements of AOs from the nuclear facilities, to the aging pad for storage, and back to the nuclear facilities for packaging.

3.2.3.1 Aging Facility Operations Staffing

It is assumed that six work crews will be required to perform AO transfer, placement, and retrieval. It is also assumed that each crew used in AO placement and retrieval operations will consist of two transporter operators, two spotting/installation operators, and one HPT.

Rationale: At least three crews are needed to support 24/7 operations. However, three crews would not allow for weekend days off, holidays, vacation or sick leave. Five crews, each working 2,000 hours per year, would provide 10,000 hours per year coverage, which covers the number of hours per year required (8,760 hours in a year). One additional crew is included to perform maintenance, inspection, and any unscheduled work at the aging pads. This crew will be made up of the same number of workers in each of the categories described above.

Usage: This assumption is used in Sections 4.3.7, 6.2.4.1, 7.1.3.1, 7.1.3.2, and 7.1.3.5.

3.2.3.2 Site Transport Vehicle

It is assumed that the trip to or from the aging facility with a loaded AO will be performed at an average speed of 1.25 mph, one-half of the 2.5 mph maximum operating speed of the site transporter (Reference 2.2.7, Note 2). It is also assumed that the return trip, without an AO, can be performed at the maximum speed of 2.5 mph (Reference 2.2.7, Note 2).

Rationale: It is conservative, due to increased travel time, to assume that the transporter will be capable of traveling at least half of its maximum operating speed while loaded.

Usage: This assumption is used in Sections 3.2.3.3, 4.3.7, 6.2.4.1, and 7.1.3.1.

3.2.3.3 Aging Facility Task Parameters

It is assumed that the time and motion data in Table 3.4 is an accurate representation of AO transfer, placement, and retrieval operations.

Table 3.4 Tasks, Durations, and Distances for AO Transfer Activities

Process	Time Required (hrs)	Distance (m)		
		AO Transporter Operator	Spotting/ Installation Operator	HPT
Prepare to Transfer				
Retrieve AO transporter - Engage AO within facility	2	2	5	5
Travel to perimeter of aging facility	1.23	TBD	TBD	TBD
Transfer to aging pad				
Move to designated aging location	0.32	10	10	10
Retract AO transporter	0.5	2	5	5
Connect Monitoring Equipment				
Position mobile equipment	0.5	N/A	1	5
Connect monitoring equipment	0.1	N/A	1	5
Return from aging pad				
Travel back to perimeter of aging facility	0.16	10	10	N/A
Travel back to nuclear facility	0.62	TBD	TBD	N/A
Total	5.43			

Note: N/A = not applicable, worker not required for these tasks
TBD = to be determined

Rationale: The tasks, distances, and durations described in Table 3.4 are based on input from operations and mechanical handling since there is currently no operational protocol confirmed for these operations. It is reasonable to assume that an HPT must be present during the transport of a loaded AO. The total task time is 5.43 hours which would allow for the placement of one AO in an eight hour work shift. The distances reported as TBD are based on the specific AO being transported in order to allow for personnel to remain in a 0.25 millirem per hour (mrem/hr) or lower dose field (Assumption 3.2.7). These distances will be determined by operations at the time the operation is to be performed.

Transport of loaded AOs to the aging pads is separated into two parts. The first is travel from the facility to the perimeter of the aging facility and the second is travel inside the aging facility. Based on the memo from Balance of Plant (Reference 2.2.8), which gives the distance from CRCF-2 to the farthest aging spot on pad 17P (1.94 miles), the dimensions in the *Aging Facility General Arrangement, Aging Pad 17P Plan* (Reference 2.2.20), and the 0.25 mrem/hr aging pad offset (821 ft) reported in the *GROA Shielding Requirements Calculation* (Reference 2.2.30, Section 7.2.2.1), the travel distance of these two legs was determined. The distance traveled inside the aging facility on pad 17P, including the 821 ft., 0.25 mrem/hr offset, was calculated to be 2124 ft or 0.40 miles. By subtracting this from the total travel distance reported in Reference 2.2.8, the distance from the CRCF-2 to the aging facility perimeter can be calculated to be 1.54 miles. The resulting travel time for the trip from the nuclear facility to the aging facility perimeter is 1.23 hours applying the average site transporter speed reported in Assumption 3.2.3.2. The travel time inside the perimeter using the same transporter speed is 0.32 hours. These task times are presented in Table 3.4 as the second and third tasks, respectively. The last two task times reported in Table 3.4 represent the reverse of these two legs performed without an AO and at the maximum operation speed of 2.5 mph per Assumption 3.2.3.2.

Usage: This assumption is used in Sections 4.3.7, 6.2.4.2, Table 7.9, 7.10 and 7.1.3.2.

3.2.3.4 Aging Pad Unloading Sequence

The removal of AOs from the aging pad is assumed to be the same process as placing the AOs in reverse order.

Rationale: Since the process involves the same steps only in reverse, we can assume that it will take a worker approximately the same amount of time at the same distances as in the aging pad loading procedure. This includes a fully loaded pad adjacent to the pad being unloaded.

Usage: This assumption is used in Sections 4.3.7, 6.2.4.1, 6.2.4.2, and 7.1.2.3.

3.2.3.5 Aging Pad Maintenance Processes

The maintenance and repair activities associated with the aging pads are expected to occur only on an as needed basis, no regularly scheduled maintenance is planned. Any maintenance or repair activity is expected to take approximately 1-hour per event at varying distances from the AO. It is assumed that a team of 2 maintenance workers will be able to perform each task.

Rationale: These values are based on input from operations and mechanical handling because no procedure for this operation is currently available.

Usage: This assumption is used in Sections 4.3.7, 6.2.4.3, and 7.1.3.3.

3.2.3.6 Aging Pad Inspection Processes

The aging pads and AOs are expected to be inspected quarterly at an average rate of one AO every 30 seconds at a distance of 2 meters from the AOs on the outside of the arrays. It is assumed that the required inspection can be done without the inspector having to enter the array.

Rationale: These values are based on input from operations and mechanical handling because no procedure for this operation is currently available.

Usage: This assumption is used in Sections 4.3.7, 6.2.4.3, and 7.1.3.3.

3.2.3.7 Seismic Restraint Operations

It is assumed that the seismic restraints on an AO consist of 20 bolts and 20 keeper clips. It is also assumed that it take approximately 15 minutes to install one bolt and one keeper.

Rationale: Though the current aging facility design does not call for the use of seismic restraints, the dose received during this task will be included for comparison in Attachment C. Twenty bolts and keeper clips is a reasonable assumption based on the superseded drawing *Aging Overpack Outline/Interface* (170-MJ0-HAC0-00101-000-000). This drawing is the most accurate representation of seismic restraints considered as the current AO drawings do not include the seismic restraints. Therefore, it is considered acceptable as input even though it has been superseded. An installation time of 15 minutes per bolt is based on input from mechanical handling.

Usage: This assumption is used in Attachment C.

3.2.4 Source Data

The following assumptions concern the fuel assemblies used in this calculation.

3.2.4.1 Fuel Regions

It is assumed that a commercial spent nuclear fuel (SNF) assembly consists of four regions (active fuel, bottom-end fitting, plenum, and top-end fitting).

Rationale: SNF assembly representation in this calculation is consistent with source term generation (Reference 2.2.1, Section 5.4), and the simplified representation of the fuel regions generates conservative dose rates.

Usage: This assumption is used in Sections 6.1.4.3, 6.1.5.1, and 6.1.5.2.

3.2.4.2 Fuel Materials

It is assumed that the active fuel region of the SNF contains the same concentrations of uranium isotopes as the fresh, unirradiated fuel.

Rationale: This assumption is conservative because while photon attenuation properties of spent fuel and fresh fuel are similar, fresh fuel has a conservatively higher neutron dose rate, due to greater production of fission neutrons (Reference 2.2.3, Section 5.4.6). This is not due to the fission yield for neutron, but rather to the greater abundance of fissile constituents (Reference 2.2.3, Section 5.4.6). This assumption applies to the material composition only. The source term is based on spent fuel.

Usage: This assumption is used in Section 6.1.5.2.

3.2.4.3 Fuel Bottom End Fitting

It is assumed that the bottom-end fitting length for a Babcock & Wilcox (B&W) 15x15 Mark B Pressurized Water Reactor (PWR) fuel assembly is 4 inches (in).

Rationale: The available reference for the B&W 15x15 Mark B PWR fuel assembly dimensions (Reference 2.2.4, p. 2A-35) specifies a 2 inches long bottom-end nozzle. An additional 2 inches of length are added to cover all bottom-end region materials. The modeled bottom-end fitting is a total of 4 inches long. Dose rates are not significantly impacted by the length estimation, since the mass in the bottom-end fitting region is conserved.

Usage: This assumption is used in Section 6.1.5.1, and Table 6.11.

3.2.4.4 AO Sources

It is assumed that the AOs being placed onto the aging pad are loaded with the design basis fuel and the AOs being retrieved from the aging pad are loaded with the average fuel. Design basis fuel is also used to evaluate the dose rate to workers performing maintenance and inspection operations. The shielding of the AO was developed using the maximum source term to meet the surface dose rate requirements (Reference 2.2.9, Section 3.3.4).

Rationale: The main purpose of the aging facility is to reduce the thermal output of the spent fuel contained in the AOs prior to emplacement in the repository. Therefore, using the more intense source term, the design basis fuel source, for placement onto the aging pad and the less intense source term, the average fuel source, for retrieval from the aging pads is reasonable.

Usage: This assumption is used in Sections 4.3.7, 6.2.4.1, 6.2.4.2, 7.1.3.1, and 7.1.3.2.

3.2.5 Inhalation and Submersion Calculation

3.2.5.1 Airborne Radiation Usage

It is assumed that the inhalation and submersion dose to workers on the aging pads is bounding for workers in all other areas of the GROA.

Rationale: Based on the number of canisters placed on the aging pads, the airborne release calculated for workers in that area are expected to be greater than those for personnel working around loaded transportation casks or loaded TEVs. Therefore, the data calculated for the aging pads will be applied to personnel in all other areas of the GROA.

Usage: This assumption is used in Section 4.3.2.

3.2.5.2 Released Radionuclides from Aging Pads

It is assumed that the Aging Overpack handled in the Wet Handling Facility, Canister Receipt and Closure Facilities and Receipt Facility shall meet the following surface contamination limits before exiting the facility. The exterior of aging overpack and the interior of the aging overpack (or the exterior surface of the canister contained in an aging overpack), that utilize natural circulation cooling, shall not have removable contamination in excess of:

1. 1,000 dpm/100 cm² for non-fixed beta and gamma emitting radionuclide contamination
2. 20 dpm/100 cm² for alpha emitting radionuclides

The radioactive surface contamination on the AOs in the aging facility is assumed to be evenly distributed Cobalt-60 (⁶⁰Co) and Americium-241 (²⁴¹Am).

Rationale: This limit is consistent with 10 CFR Part 20 Subpart E Section 1406 (Reference 2.3.2) which requires applicants for licenses shall describe in the application how facility design and procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment and these limits are typical of release limits in the nuclear industry. Using ⁶⁰Co as the representative beta-gamma emitting isotope is conservative because it has the highest inhalation dose conversion factor (DCF) (Reference 2.2.16, Table 2.1). Using ²⁴¹Am is appropriate because it is a dose significant, long lived alpha emitting isotope that is likely to be present.

Usage: This assumption is used in 4.3.2, Table B.1, and Attachment B.

3.2.5.3 Respirability of Airborne Radioactivity

It is assumed that all released or suspended radioactive particles are respirable.

Rationale: This assumption is appropriate because it yields the maximum inhalation dose values. Furthermore, the Department of Energy (DOE) handbook (Ref. 2.2.46, Section 1) for estimating airborne release fractions/rates and respirable fractions recommends a respirable fraction of one (1) for aerodynamic entrainment of powder lying on a heterogeneous surface (indoors or outdoors) exposed to ambient conditions following an event.

Usage: This assumption is used in Section 4.3.2 and Attachment B.

3.2.5.4 Worker Breathing Rate

It is assumed that all workers breathe at a rate of 20,000 mL/min, which is equivalent to $3.33 \text{ E-}04 \text{ m}^3/\text{s}$. ($20,000 \text{ mL}/\text{min} \times 1\text{E-}06\text{m}^3/\text{mL}/\text{min}/60\text{s}$).

Rationale: This is consistent with the breathing rate contained in Reference 2.3.2 Appendix B, Footnote to Table 1 for calculating occupational doses due to inhalation intake.

Usage: This assumption is used in 4.3.2.

3.2.5.5 Worker Exposure Time to Inhalation and Submersion Doses

It is assumed that workers are exposed to inhalation and submersion doses for 428 hours per year.

Rationale: It is conservative to assume that all personnel are exposed to inhalation and submersion doses for 428 hours per year because that is the annual task time associated with the longest task, placing and retrieving 46 AOs in the aging facility (Table 6.19).

Usage: This assumption is used in 4.3.2 and Attachment B.

3.2.6 Time in Low Radiation Support Area

It is assumed that GROA personnel not directly working with waste transport will remain in support areas with dose rates of 0.05 mrem/hr. It is also assumed that all personnel, whether operations or HP, spend equal amounts of time in support areas and therefore receive the same support dose.

Rationale: When work crews are not involved with transferring casks, AOs, or TEVs they will be in either an unlimited occupancy area (R1), or a routine occupancy area (R2). It is considered reasonable based on the area classification that workers are on average residing in an environment where the dose rate is 0.05 mrem/hr or less when not involved in operations.

Usage: This assumption is used in section 6.2.5, and 7.1.4.

3.2.7 Waste Transported Through the GROA

It is assumed that for all loaded waste transport methods (i.e. transportation casks, AOs, and TEVs), all operations and HP personnel escorting them are able to remain in a dose field of 0.25 mrem/hr or less.

Rationale: It is reasonable to assume that personnel will be controlled during transport of all waste such that they will remain in a dose field of 0.25 mrem/hr or less including any streaming contributions from the bottom of the AO. Keeping workers in a low dose area is a good ALARA practice.

Usage: This assumption is used in Sections 3.2.3.3, 4.3.5, 4.3.7, 6.2.2.1, 6.2.2.2, 6.2.4.1, 7.1.2.1, 7.1.3.1, and Tables 7.4, 7.7, 7.9, 7.10.

4. METHODOLOGY

This dose assessment involves the calculation of annual individual worker exposures and annual collective workers exposures to radiation resulting from: the receipt of loaded casks; the movement of loaded casks throughout the GROA; the placement and retrieval of AOs at the aging pads; and the movement of the TEV to the subsurface. The calculated doses include the contributions from external radiation, and inhalation and submersion doses. The methodology for calculating direct external radiation doses is described in Section 4.3.1. The methodology for calculating inhalation and submersion doses is described in Section 4.3.2 and presented in Attachment B. The total effective dose equivalent (TEDE) is calculated by summing the component doses from inhalation, submersion with direct external radiation dose as described in Section 4.3.3. The description of the approach used for the dose rate calculations for the receipt of loaded casks, movement of loaded casks in the GROA, placement and retrieval of AOs, and the movement of TEVs are described in sections 4.3.4 through 4.3.7.

4.1 QUALITY ASSURANCE

This calculation is prepared in accordance with the procedure EG-PRO-3DP-G04B-00037, *Calculations and Analyses* (Reference 2.1.1). This calculation addresses worker dose evaluations and has been developed subject to requirements of the *Quality Management Directive* (Reference 2.1.2). The aging pad and transportation cask have been classified as a Safety Category item (important to safety) on the *Q-list* (Reference 2.2.24, Table A-1). Therefore, the approved version is designated as QA:QA.

4.2 USE OF SOFTWARE

Software is utilized in accordance with procedure IT-PRO-0011, *Software Management* (Reference 2.1.3). Mainly computer-based calculations were used in this document; however, several source multiplication and importance cards as well as tally surface areas in Monte Carlo N-Particle transport code (MCNP) were calculated through arithmetic by hand calculations.

4.2.1 Baseline Software

4.2.1.1 MCNP5 Code Version 1.40

The MCNP5 version 1.40 code (Reference 2.2.25) is used to calculate the dose rates (neutron, primary gamma and secondary gamma) at various locations around a transportation cask, an aging overpack, and multiple aging overpacks in the aging facility.

The software specifications are as follows:

- Program Name: MCNP5
- Version/Revision Number: Version 1.40
- Operating Systems: Windows XP

- Software Tracking Number: 11199-1.40-00
- Computer Type: Dell OPTIPLEX GX620

MCNP5, which is Level 1 software, is: (a) appropriate for three-dimensional neutron, gamma, and coupled neutron/gamma shielding calculations, (b) used within the range of validation as documented in *Software Validation Report for: MCNP5 Version 1.40* (Reference 2.2.26) and (c) obtained from Software Configuration Management and is qualified in accordance with procedure IT-PRO-0012, *Qualification of Software* (Reference 2.1.4). Therefore, MCNP5 code version 1.40 is suitable for use in this design calculation. All MCNP5 input and output files documented in this calculation are stored on a compact disc (CD) as Attachment D.

4.2.2 Commercial off the Shelf Software

4.2.2.1 MICROSOFT® Excel 2003 SP-2

The Excel software is used to calculate various input values (i.e., atom densities, weight percentages, geometrical dimensions, normalized source spectra and probabilities, etc). Standard functions of Excel are also used in this design calculation to display results (i.e., gamma and neutron dose rates) in tabular and graphic forms and to perform relative error calculations.

The software specifications are as follows:

- Program Name: Excel
- Version/Revision Number: Microsoft® Excel 2003 SP-2
- This software is installed on a personal computer running Microsoft Windows XP (central processing unit number YMP005045)

The user defined formulas, inputs, results, and graphical representations can be reproduced, were checked by hand and visual inspection, and are documented in sufficient detail to allow an independent repetition of the computations. Per *Software Management*, Microsoft® Excel, as used in this calculation, constitutes Level 2 software usage (Reference 2.1.3, Attachment 12), and is not required to be qualified in accordance with procedure IT-PRO-0012 (Reference 2.1.4). The Excel spreadsheets used to calculate input values or to display the results in graphical or tabular form are included in Attachment D.

4.3 CALCULATION APPROACH

The TEDE calculated is made up of contributions from direct radiation, inhalation, and submersion doses. The main body of this calculation is dedicated to the calculation of direct doses while Attachment B discusses and calculates the inhalation and submersion dose. TEDE will be presented in the conclusion section (Section 7.4).

4.3.1 Calculated External Dose

For all operations of cask and AO processing in this calculation, the external dose, ED_k , received by a worker for a task k is calculated as follows. The dose rates are at the locations of each operation task due to external radiation from the contained radiation sources.

$$ED_k = \frac{t_k}{60} \times EDR_{dist} \quad \text{Equation 1}$$

where

$$\begin{aligned} ED_k &= \text{external dose to a worker per task } k \text{ (mrem/task)} \\ t_k &= \text{duration of exposure per task } k \text{ (minutes)} \\ EDR_{dist} &= \text{external dose rate at the worker's distance (} dist \text{) from the source} \\ &\text{(mrem/hr)} \\ 60 &= \text{units conversion (minutes/hr)} \end{aligned}$$

The total external dose, ED_o , to a worker for a series of N different tasks per operation (e.g., cask handling) is calculated as follows:

$$ED_o = \sum_{k=1}^N ED_k \quad \text{Equation 2}$$

where

$$ED_o = \text{external dose to a worker per operation consisting of } N \text{ different tasks (mrem/operation)}$$

When not performing manual operations on a cask, the individual in a work crew is assumed to remain in the vicinity doing support activities in intermittent access lower radiation areas. This support-only time, T_n , is determined from the time available, i.e. 40 hrs/week x 50 weeks/year = 2000 hrs, minus the time performing cask operations, T_o .

$$T_o = \sum_{k=1}^{N_c} \frac{t_k}{60} \times NP_C$$

Equation 3

and

$$T_n = 2000 \left(\frac{\text{hrs}}{\text{yr}} \right) - T_o$$

where

$$\begin{aligned} T_o &= \text{Time performing cask operations for worker (hrs/year)} \\ T_n &= \text{Support-only time for worker (hrs/year)} \\ NP_C &= \text{Annual number of casks processed per crew (casks/crew-year)} = OP_C / \\ &\text{crews}_g \\ OP_C &= \text{Number of casks processed per year (casks/year)} \\ \text{crews}_g &= \text{Number of work crews} \\ 60 &= \text{Units conversion (minutes/hr)} \end{aligned}$$

The total annual external dose, ED_g , to a worker for all cask operations including support-only time is calculated as:

$$ED_g = ED_C \times OP_C + T_n \times DR \quad \text{Equation 4}$$

where

ED_g	=	external dose to a worker (mrem/year)
ED_C	=	external dose to a worker per cask during cask processing operations (mrem/cask)
OP_C	=	number of casks processed per year (casks/year)
DR	=	dose rate in areas of lower radiation (mrem/hr)

4.3.2 Inhalation and Submersion Dose

The maximum internal and external doses to a worker due to resuspension of contamination on a transportation cask (TC), waste package (WP), or AO is given in Attachment B and described below. In Attachment B, inhalation and submersion dose is calculated based on the aging facility and considered to bound all other operations in the GROA (Assumption 3.2.5.1). Given the total surface activities for beta-gamma and alpha sources (Assumption 3.2.5.2), the annual contamination release is found by:

$$C_{\beta\gamma} = A_{\beta\gamma} \times RSF \times 10,000$$

and

$$C_{\alpha} = A_{\alpha} \times RSF \times 10,000 \quad \text{Equation 5}$$

where:

$C_{\beta\gamma}$	=	beta-gamma airborne concentration ($\mu\text{Ci}/\text{m}^3$)
C_{α}	=	alpha airborne concentration ($\mu\text{Ci}/\text{m}^3$)
$A_{\beta\gamma}$	=	total beta-gamma surface activity ($\mu\text{Ci}/\text{cm}^2$)
A_{α}	=	total alpha surface activity ($\mu\text{Ci}/\text{cm}^2$)
RSF	=	re-suspension factor (m^{-1}) (Assumption 3.1.5)
10,000	=	units conversion ($10,000 \text{ cm}^2/\text{m}^2$)

$$H^{inh} = C \times DCF^{inh} \times 3.7 \times 10^9 \times RF \times BR \times t$$

Equation 6

and

$$H^{sub} = C \times DCF^{sub} \times 3.7 \times 10^9 \times t$$

H^{inh}	=	annual inhalation dose to workers (mrem/year)
H^{sub}	=	annual submersion dose to workers (mrem/year)
C	=	airborne concentration ($\mu\text{Ci}/\text{m}^3$)
DCF^{inh}	=	inhalation dose conversion factor (Sv/Bq)
DCF^{sub}	=	submersion dose conversion factor (Sv/Bq)
RF	=	respirable fraction (1 per Assumption 3.2.5.3)
BR	=	breathing rate (Reference 2.3.2, Assumption 3.2.5.4)
3.7×10^9	=	units conversion (mrem Bq/Sv μCi)
t	=	time exposed to contamination (Assumption 3.2.5.5)

The total annual inhalation and submersion dose, ID_o , in the aging facility in the presence of airborne radioactivity is calculated as follows:

$$ID_g = H^{inh} + H^{sub}$$

Equation 7

where

$$ID_g = \text{annual inhalation and submersion dose to a worker (mrem/year)}$$

4.3.3 TEDE Dose Calculation

The annual TEDE, $TEDE_g$, to a worker in a work crew is calculated by summing the component doses from inhalation, submersion and direct external radiation doses.

$$TEDE_g = ID_g + ED_g$$

Equation 8

where

$$TEDE_g = \text{annual TEDE to a worker (mrem/year)}$$

4.3.4 Cask Receipt

Dose fields around a TS125 transportation cask and the time-motion assumptions (Assumptions 3.2.1.1, 3.2.1.2, and 3.2.1.3) were used to develop a TEDE for workers performing the processes necessary during typical loaded cask acceptance. These tasks included the receipt of paperwork, initial inspections, removal of any barriers, and the final inspection in building 30B. Assumption 3.1.1.1 concerning annual cask acceptance

is used to determine dose on an annual basis. The worker category Operator includes all forms of operations personnel unless otherwise specified.

4.3.5 GROA

Dose fields around a TS125 transportation cask and the time-motion assumptions (Assumption 3.1.2.1) were used to determine any additional dose that may be a concern to personnel working in any of the non-nuclear facilities while the transportation casks move through the GROA to the buffer areas or nuclear facilities. These tasks included any movement of casks from 30B to either of the two buffer areas (33A and 33B) or directly to the nuclear facilities.

Dose to workers moving the transportation casks from the cask receipt security station to the buffer areas or nuclear facilities is also considered. Staffing and task data from Assumption 3.2.2.1 is used along with the travel distance information from Assumption 3.1.2.2 and the speed limit inside the GROA from Reference 2.2.8 per Assumption 3.1.3. The maximum worker dose rate per Assumption 3.2.7 was applied to these time and motion assumptions to determine total dose to workers per cask. The annual cask acceptance rate of Assumption 3.1.1.1 is used to evaluate the worker dose per year. The worker category Operator includes all forms of operations personnel unless otherwise specified.

Worker dose during TEV movement is also considered based on the personnel and travel distance assumptions (Assumptions 3.2.2.2 and 3.1.2.3). The dose rate assumed for transportation cask operations is again used for the TEV operations (Assumption 3.2.7). The nominal speed of 150 ft./min for the TEV (Reference 2.2.49, Section 3.1.1) is used to determine worker doses on a per task and, along with Assumption 3.1.1.2, on a per year basis. The worker category Operator includes all forms of operations personnel unless otherwise specified.

4.3.6 TAD and AO

The amount of shielding required on an AO was determined using MCNP5, the maximum source term, and the surface dose rate criteria outlined in the *Transportation, Aging and Disposal Canister System Performance Specification* (Reference 2.2.9, Section 3.1.1(5) and 3.3.4, respectively). A parametric study was performed to determine the balance between the steel and concrete thicknesses to meet the 40 mrem/hr requirement per the *Transportation, Aging and Disposal Canister System Performance Specification* (Reference 2.2.9, Section 3.3.4). The resultant AO shielding design is presented in Section 6.1.5.1 and was used for all aging pad dose rate calculations. The dose rate and relative error data for this study are included in Attachment D, *Radial Shielding Eval.xls*.

4.3.7 Aging Facility

Dose rates outside of an individual AO and around multiple AOs were determined using MCNP5 with both the design basis and average fuel source terms to represent dose rates at various stages in the aging process (Assumption 3.2.4.4). Those dose rates along with those of Assumption 3.2.7 combine with the staffing data (Assumption 3.2.3.1), site

transporter data (Assumption 3.2.3.2), time-motion data (Assumption 3.2.3.3), and loading and unloading procedures (Assumptions 3.1.4 and 3.2.3.4) to develop a TEDE for workers performing normal operations on and around an aging pad. Throughput data (Assumption 3.1.1.3) was used to determine the annual dose. Worker dose data from the NUHOMS HD Safety Analysis Report (SAR) is used to represent horizontal aging module (HAM) operations (Reference 2.2.17) and is presented in Section 7.1.3.4. Maintenance and inspection doses were also determined based on the time and motion data in Assumptions 3.2.3.5 and 3.2.3.6. The worker category Operator includes all forms of operations personnel unless otherwise specified.

4.3.8 MCNP5 Code

MCNP5, a general Monte Carlo n-particle transport code simulates particle transport through a three-dimensional representation of the nuclear system being analyzed. During the random walk of particles through the system described by the problem geometry, the code collects information about various events (surface crossing, collisions, track length, etc.) and translates them into flux estimates displayed in tallies. It also estimates the statistical precision of the results.

Per *Project Design Criteria Document* (Reference 2.2.27, Section 4.10.1.5), the flux estimates are converted to dose rates using *Neutron and Gamma-Ray Flux-to-Dose-Rate Factors* (Reference 2.2.10). The variance reduction technique of Geometry Splitting/Russian Roulette (Section 6.1.1.4) is used to enhance computational efficiency.

The U.S. NRC recognizes the MCNP5 code for shielding calculations related to the SNF storage and transportation systems in Nuclear Regulatory (NUREG)-1567 (Reference 2.2.11, p. 7-13). Version 1.40 of the MCNP5 code is the qualified software used at BSC to run gamma and coupled neutron/gamma calculations (Reference 2.2.26).

5. LIST OF ATTACHMENTS

	Number of Pages
Attachment A. List of Files on Attached CD	3
Attachment B. Inhalation and Air Submersion Doses	3
Attachment C. Seismic Restraint Analysis	1
Attachment D. Attached Compact Disc (CD)	N/A

6. BODY OF CALCULATION

6.1 INPUTS

Technical product input and sources of the input used in the development of this calculation are documented in this section. Tables in the following sections and in the associated Microsoft® Excel spreadsheets referenced may present numerical values with high significant figures; these values are not to be considered as high precision values as the significant figures are the result of calculated values and unit conversions.

6.1.1 MCNP5 Input Data

6.1.1.1 Source Probabilities

A uniform radial distribution is used for all gamma and neutron sources. The distribution is sampled using the built-in power function $p(x)=c \cdot x^a$ with $a = 1$.

The sources axial distributions include non-uniform distributions for the active fuel region sources and uniform distributions for the bottom-end fitting, top-end fitting, and plenum activated sources. The TAD canister axial profile, provided in Table 6.16, is applied through **SI/SP** cards. The TS125 transportation cask axial profile distributions, provided in Tables 6.8 for gamma sources and Table 6.9 for neutron sources, are also applied through **SI/SP** cards. The axial uniform distribution is sampled using the built-in power function $p(x)=c \cdot x^a$ with $a = 0$.

The maximum, design basis, and average source terms for the TAD, provided in Tables 6.15a, 6.15b, and 6.15c, respectively, are used as energy probabilities, and input using **SI/SP** cards. The gamma and neutron source terms for the TS125, provided in Tables 6.6, 6.7 and 6.10, are also used as energy probabilities, and input using **SI/SP** cards.

6.1.1.2 Dose Rate Tallies

The MCNP5 dose rate calculations feature separate runs for the following problem types:

- a photon transport problem for primary gamma contribution to dose rates from active fuel, bottom-end, plenum, and top-end gamma source terms,
- a coupled neutron/photon transport problem for neutrons and secondary gamma (photons generated by neutron interactions) contributions to dose rates from the active fuel neutron source term.

The tally types F2 and F4 are used in this design calculation as flux estimators. The type F2 tally evaluates the flux averaged over a surface in particles/cm². The type F4 tally evaluates the flux averaged over a cell in particles/cm³.

To generate dose rates in terms of rem/hr, the **FM** (tally multiplier) and **DE/DF** (dose energy/dose conversion factors – refer to Section 6.1.2) cards are used in the input files to enter the tally multiplier factor (source intensity in particles/s), energy boundaries

in Million electron Volt (MeV), and corresponding dose factors in (rem/hr)/(particles/cm²/s), respectively.

In order for the F2 or F4 tallies to accumulate flux, surface area or cell volume is required. The **SD** card is used to enter surface area in cm² or cell volume in cm³ for the surface or cell that the flux will be tallied over or through.

6.1.1.3 Variance Reduction Techniques

An appropriate variance reduction technique was used to increase the efficiency of the calculations and obtain statistically meaningful results within reasonable computer time. In the Monte Carlo calculations, a fair game is always preserved regardless of the biasing scheme used, as the weight of particles is adjusted internally in the code to account for biasing. The following variance reduction technique is used in all the MCNP5 runs.

6.1.1.4 Geometry Splitting with Russian Roulette

Geometry splitting with the Russian roulette technique described in the *MCNP-A General Monte Carlo N-Particle Transport Code, Version 5* (Reference 2.2.28, p. 2-136 – 2-138) is especially useful in shielding calculations where the dose reduction factor is many orders of magnitude.

As particles migrate in an important direction (toward detector), they are increased in number, and simultaneously decreased in weight, to provide better sampling by means of geometry splitting. Russian roulette (the reverse process) is played when particles travel in an unimportant direction to allow early termination of an unimportant history. The importance factors used to apply this technique are entered in every cell card. The particular values used as importance factors for geometry splitting/Russian Roulette are estimated using test runs and then adjusting to keep the particle population approximately constant as they pass through the shielding structures toward detector locations.

6.1.2 Dose Conversion Factors

The *Project Design Criteria Document* (Reference 2.2.27, Section 4.10.1.5) requires the use of flux-to-dose rate conversion factors from American National Standards Institute (ANSI)/ American Nuclear Society (ANS)-6.1.1-1977 (Reference 2.2.10) for dose rate evaluations. Table 6.1 lists the neutron and gamma conversion factors used in the MCNP5 calculations.

Table 6.1 Gamma and Neutron Flux-to-Dose Rate Conversion Factors

GAMMA				NEUTRON	
Energy (MeV)	Conversion Factors (rem/hr)/(photon/cm ² /s)	Energy (MeV)	Conversion Factors (rem/hr)/(photon/cm ² /s)	Energy (MeV)	Conversion Factors (rem/hr)/(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	5.80E-06	1.00E+00	1.32E-04
0.4	9.85E-07	5.25	6.01E-06	2.50E+00	1.25E-04
0.45	1.08E-06	5.75	6.37E-06	5.00E+00	1.56E-04
0.5	1.17E-06	6.25	6.74E-06	7.00E+00	1.47E-04
0.55	1.27E-06	6.75	7.11E-06	1.00E+01	1.47E-04
0.6	1.36E-06	7.5	7.66E-06	1.40E+01	2.08E-04
0.65	1.44E-06	9	8.77E-06	2.00E+01	2.27E-04
0.7	1.52E-06	11	1.03E-05		
0.8	1.68E-06	13	1.18E-05		
1	1.98E-06	15	1.33E-05		

Source: Reference 2.2.10, Tables 1 and 3.

6.1.3 Design Requirements and Criteria

Calculations are executed to determine dose rates at varying distances from sources contained within transportation casks and aging overpacks. These doses are to be used to determine an estimate of worker dose during the performance of varying tasks. Also, the distances between non-nuclear facilities and the transportation casks path of travel through the GROA will be compared to determine any additional dose that might be a concern for administration personnel.

All transportation casks are designed to meet dose limits specified by 10 Code of Federal Regulations (CFR) Part 71.47 (Reference 2.3.1). As stated in 10 CFR part 71.47(b)(3), transportation casks are required to meet a dose rate limit of 10 mrem/hr at 2 m from conveyance (Reference 2.3.1).

The TAD canister and aging overpack are designed to meet the dimensional and dose requirements outlined in *Transportation, Aging and Disposal Canister System Performance Specification* (Reference 2.2.9, Sections 3.1 and 3.3). Per this specification, the loaded aging overpack combined neutron and gamma contact dose rate shall not

exceed 40 mrem/hr on any accessible exterior surface excluding the underside (Reference 2.2.9, Section 3.3.4).

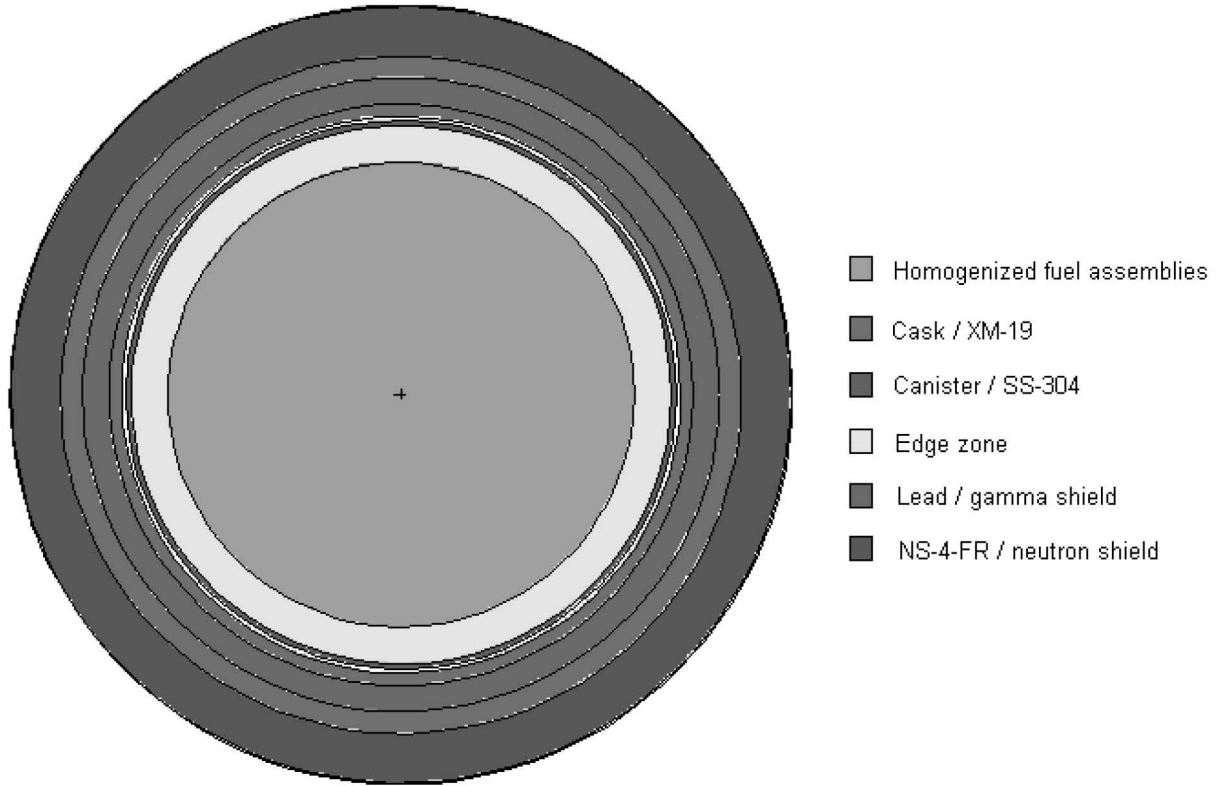
6.1.4 Transportation Cask

6.1.4.1 TS125 Transportation Cask Selection

All NRC certified casks are required to meet a dose rate limit of 10 mrem/hr at 2 m away from conveyance per 10 CFR Part 71.47(b)(3) (Reference 2.3.1). The TS125 rail cask is selected to represent both rail and truck casks throughout the receipt process and GROA operations. The TS125 cask can hold a canister containing 21 PWR fuel assemblies (Reference 2.2.13, p. 5.2-1) and it provides radiation shielding engineered to meet the regulatory requirements of 10 CFR Part 71 (Reference 2.3.1). The TS125 cask was the only cask considered for the receipt and transient dose rate calculations as it is representative of all other casks meeting 10 CFR part 71 requirements.

6.1.4.2 TS125 Transportation Cask Geometry

Each fuel assembly contained in the TS125 rail cask consists of four axial regions: top nozzle, gas plenum, active fuel and bottom nozzle. The height of the top nozzle is 4.0 in., the height of the gas plenum is 6.5 in., the height of the active fuel region is 144.0 in., and the height of the bottom nozzle is 5.0 in. (Reference 2.2.12, Table 5.3-3). Note that the fuel region is homogenized in the MCNP5 models (Reference 2.2.12, Table 5.3-3). The effective radius of the homogenized fuel region is 27.933 in. (Reference 2.2.12, Table 5.3-3). Figures 6.1 and 6.2 display the radial and axial configurations, respectively. Tables 6.2 and 6.3 feature the physical radial and axial dimensions of the various materials, respectively.



NOTE: Drawing not to scale.

Figure 6.1 TS125 Transportation Cask and Canister Radial Configuration at Mid-Plane

Table 6.2 Radial Geometry Description for TS125 Transportation Cask and Canister Model

Component	Dimensions	
	(in.)	(cm)
Canister Inner Shell IR ^a	32.375	82.233
Canister Shell Thickness ^a	0.615	1.562
Cask Inner Shell IR ^b	33.5	85.09
Cask Inner Shell Thickness ^c	1.49	3.785
Gamma Shield Thickness ^c	3.1535	8.0099
Cask Outer Shell Thickness ^c	2.64	6.71
Neutron Shield Thickness ^c	5.91	15.01
Neutron Shield Jacket Thickness ^c	0.1775	0.4509
Center Impact Limiter OR ^{b, c, f}	40.7835	103.590
Impact Limiter Divider Thickness ^b	0.74	1.88
Side Impact Limiter OR ^{b, f}	71.5	181.61
Fuel OR ^e	27.933	70.950

^a Reference 2.2.12, Figure 5.3-2

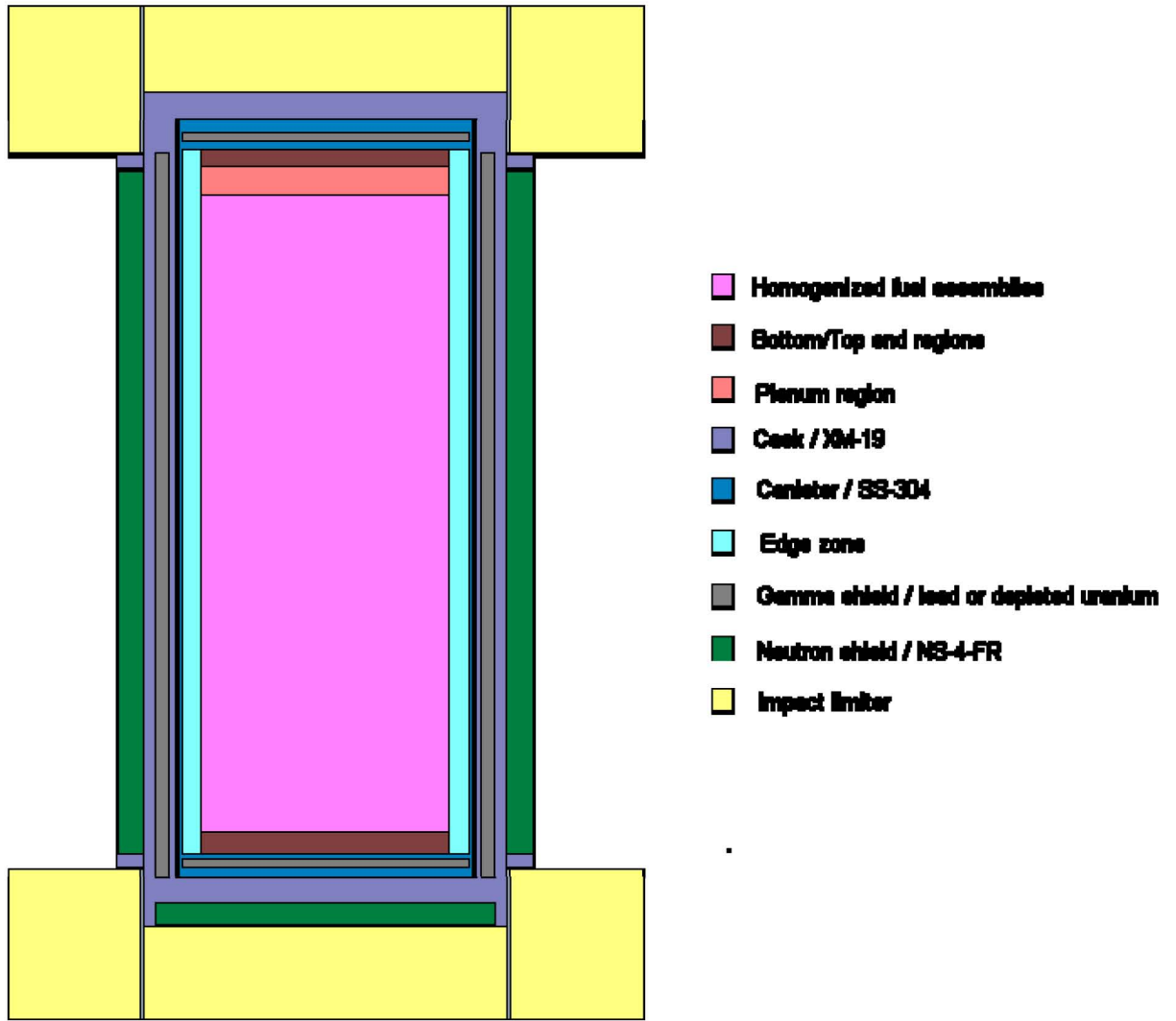
^b Reference 2.2.13, Figure 5.3-2

^c Reference 2.2.13, Table 5.3-1

^d $33.50 + 1.49 + 3.1535 + 2.64 = 40.7835$

^e Reference 2.2.13, Section 5.3.2.2, Page 5.3-6

^f Dimensions are the same for top and bottom components.



NOTE: Drawing not to scale.

Figure 6.2 TS125 Transportation Cask Axial Configuration

Table 6.3 Axial Geometry Description for TS125 Transportation Cask and Canister Model

Component	Dimensions	
	(in.)	(cm)
Canister Bottom End Plug Top Plate ^a	0.99	2.515
Canister Bottom End DU Shield Plug ^a	2.0625	5.2388
Canister Bottom End Plug Bottom Plate ^a	1.74	4.42
Cask Bottom Plate ^a	5.9375	15.081
Cask Bottom Neutron Shield ^b	4.875	12.38
Cask Bottom Neutron Shield Jacket ^b	0.24	0.61
Canister Top End Plug Bottom Plate ^a	1.615	4.102
Canister Top End DU Shield Plug ^a	2.0625	5.2388
Canister Top End Plug Top Plate ^a	3.095	7.861
Cask Top Closure Lid ^b	5.99	15.21
Center Impact Limiter ^{b, c, f}	19.5	49.5
Side Impact Limiter ^b	33.5	85.1
Canister Shell Radial Height ^{d, e}	159.5	405.1
Cask Outer Radial Ring Height ^{d, f}	3	7.62

^a Reference 2.2.12, Figure 5.3-3

^b Reference 2.2.13, Figure 5.3-2

^c $33.50 - 14.00 = 19.5$

^d Reference 2.2.13, Figure 5.3-4

^e $144.00 + 5.00 + 6.50 + 4.00 = 159.5$

^f Dimensions are the same for top and bottom components.

6.1.4.3 TS125 Transportation Cask Materials

At the center of the TS125 casks are 21 PWR fuel assemblies surrounded in the radial direction by an iron basket edge region enclosed in stainless steel (SS) 304 canister walls. The SS canister containing the PWR assemblies is situated inside an SS (XM-19) cask that consists of an inner and outer shell separated by lead gamma shielding. The final radial layer of the TS125 cask is an NS-4-FR neutron shield wrapped around the XM-19 cask. The TS125 cask also utilizes an NS-4-FR layer as the bottom neutron shield. For gamma shielding, depleted uranium (DU) is used in the top and bottom of the TS125 cask. Table 6.4 presents the composition of the various shielding materials.

Table 6.4 Material Compositions for TS125 Transportation Cask and Canister

Material	Density (g/cm ³)	Element	Weight Percent (%)	Atom Density (atoms/(barn-cm))	Reference
XM-19 (Cask component)	—	Si	N/A	1.29E-03	Reference 2.2.13, Table 5.3-2
		Cr		2.05E-02	
		Mn		4.40E-03	
		Fe		4.98E-02	
		Ni		1.03E-02	
		Mo		1.13E-03	
		Total		8.74E-02	
SS304 (Cask component)	—	Si	N/A	1.72E-03	Reference 2.2.13, Table 5.3-2
		Cr		1.77E-02	
		Mn		1.76E-03	
		Fe		5.89E-02	
		Ni		8.24E-03	
		Total		8.83E-02	
		DU (Canister component)		—	
U-238	4.77E-02				
Total	4.78E-02				
Side Impact Limiter	—	Al	15.59 ^{a, c}	9.40E-03	Attachment D, <i>materials.xls</i> , sheet <i>Impact Limiter</i>
End Impact Limiter			11.26 ^{a, c}	6.79E-03	
Lead (Cask shielding)	—	Pb	N/A	3.30E-02	Reference 2.2.13, Table 5.3-2
Carbon Steel (Cask component)	—	Fe	N/A	8.45E-02	Reference 2.2.13, Table 5.3-2
Edge Region	—	Fe	N/A	4.22E-03 ^b	Reference 2.2.12, Table 5.3-10
Pure NS-4-FR (Cask component)	—	H	N/A	5.48E-02	Reference 2.2.13, Table 5.3-2
		B-10		2.81E-04	
		B-11		1.13E-03	
		C		2.24E-02	
		N		1.37E-03	
		O		2.45E-02	
		Al		7.62E-03	
		Total		1.12E-01	
Neutron shield mix (Cask component)	—	H	N/A	5.19E-02	Reference 2.2.13, Table 5.3-2
		B-10		2.67E-04	
		B-11		1.07E-03	
		C		2.12E-02	
		N		1.29E-03	
		O		2.32E-02	
		Al		7.22E-03	
		Fe		4.39E-03	
Total	1.11E-01				

NOTE: ^a Weight percent of Al density. ^b Selected for conservatismSource: ^c Reference 2.2.12, Table 5.3.2, Note (1)

The fuel region material compositions are listed in Table 6.5. Note that all fuel regions are homogenized within the fuel zone outer radius (Assumption 3.2.4.1), set at 27.93 cm (Reference 2.2.12, Table 5.3-3).

Table 6.5 Material Composition for Smear Fuel Assemblies in a TS125 Cask

Material	Element	Atom Density (atoms/(barn-cm))	Material	Element	Atom Density (atoms/(barn-cm))
Top nozzle ^a	B-10	9.51E-05	Active fuel ^{b, e}	B-10	9.51E-05
	B-11	3.83E-04		B-11	3.83E-04
	C	1.08E-04		C	1.08E-04
	Al	9.84E-04		O	8.66E-03
	Si	1.79E-04		Al	9.84E-04
	Ti	2.48E-05		Si	4.73E-05
	Cr	2.26E-03		Cr	4.35E-04
	Mn	1.83E-04		Mn	4.84E-05
	Fe	6.51E-03		Fe	1.56E-03
	Ni	2.08E-03		Ni	2.72E-04
	Mo	7.65E-05		Zr	2.48E-03
	Total	1.29E-02		Total	1.95E-02
Plenum ^c	B-10	9.51E-05	Bottom end ^d	B-10	9.51E-05
	B-11	3.83E-04		B-11	3.83E-04
	C	1.08E-04		C	1.08E-04
	Al	9.84E-04		Al	9.84E-04
	Si	4.73E-05		Si	2.54E-04
	Cr	4.35E-04		Ti	1.71E-05
	Mn	4.84E-05		Cr	2.89E-03
	Fe	1.56E-03		Mn	2.60E-04
	Ni	2.72E-04		Fe	8.96E-03
	Zr	2.48E-03		Ni	2.08E-03
	Mo	3.47E-05		Mo	6.35E-05
	Sb	2.70E-05		Total	1.61E-02
Total	6.47E-03				

Source: Reference 2.2.12, ^a Table 5.3-8, ^b Table 5.3-5, ^c Table 5.3-6, and ^d Table 5.3-7
^e 4.0 wt% initial U²³⁵ enrichment

6.1.4.4 TS125 Transportation Cask Source Specification

The gamma and neutron source terms for the fuel region were taken from the FuelSolutions System SAR (Reference 2.2.12, Section 5.2). The maximum allowable source term for the TS125, 60 gigawatt days per metric ton uranium (GWd/MTU) burnup, 4.0 weight percent (wt%) initial ²³⁵U enrichment, and 18 year cooling time, was chosen for all TS125 calculations (Reference 2.2.12, Sections 5.2.2 & 5.2.3). Table 6.6 contains the primary fuel gamma source strengths by energy group as well as the gamma

energy source strength fractions. The activation gamma sources for the active fuel zone, the bottom-end zone, the gas plenum zone, and the top nozzle zone are given in Table 6.7. Note that the activations of these zones are due to Co-60, which has two energy peaks, 1.17 and 1.33 MeV, of equal probability. The axial gamma source profile for the fuel region is displayed in Table 6.8, while the neutron source term and axial profile are presented in Table 6.9. The normalized neutron energy spectrum and the energy group structure are listed in Table 6.10.

Table 6.6 TS125 Transportation Cask Fuel Gamma Source Term

Group Midpoint Energy (MeV)	Gamma Source Strength (Photons/(s-canister)) ^a	Gamma Energy Source Strength Fractions ^b
0.575	4.505E+16	9.413E-01
0.85	1.572E+15	3.284E-02
1.25	1.201E+15	2.509E-02
1.75	3.754E+13	7.843E-04
2.25	7.972E+09	1.666E-07
2.75	1.406E+10	2.938E-07
3.5	1.567E+09	3.274E-08
Total	4.786E+16	1.000E+00

NOTE: Spent fuel characteristics: 4.0 wt % initial enrichment, 60 GWd/MTU burnup and 18 years cooling time

Source: Reference 2.2.12, ^a Tables 5.2-2 and ^b Table 5.2-4

Table 6.7 TS125 Transportation Cask Hardware Regions Gamma Source Strength

Hardware Region	Gamma Source (photons/(s-canister))
Bottom-End Zone ^a	9.493E+13
Fuel Zone Hardware ^b	4.092E+14
Gas Plenum Zone ^c	5.736E+13
Top Nozzle Zone ^d	8.102E+13

NOTE: Spent fuel characteristics: 4.0 wt % initial enrichment, 60 GWd/MTU burnup and 18 years cooling time

Bottom-End, Gas Plenum, Top Nozzle Zone, and fuel activation due to Cobalt-60

Source: Reference 2.2.12, ^a Table 5.2-7, ^b Table 5.2-6, ^c Table 5.2-8, and ^d Table 5.2-9

Table 6.8 TS125 Axial Gamma Source Profile

Axial Bin (cm)	Gamma Source Fraction	Axial Bin (cm)	Gamma Source Fraction
0.00 – 20.32	0.0318	182.88 – 203.20	0.0607
20.32 – 40.64	0.0509	203.20 – 223.52	0.0605
40.64 – 60.95	0.0592	223.52 – 243.83	0.0603
60.95 – 81.27	0.0614	243.83 – 264.15	0.0601
81.27 – 101.61	0.0619	264.15 – 284.49	0.0596
101.61 – 121.93	0.0617	284.49 – 304.81	0.0584
121.93 – 142.24	0.0614	304.81 – 325.12	0.0552
142.24 – 162.56	0.0612	325.12 – 345.44	0.0462
162.56 – 182.88	0.0609	345.44 – 365.76	0.0284

Source: Reference 2.2.12, Table 5.2-10

Table 6.9 TS125 Axial Neutron Source Profile

Axial Bin (cm)	Neutron Source Fraction	Axial Bin (cm)	Neutron Source Fraction
0.00 – 20.32	0.0052	182.88 – 203.20	0.0688
20.32 – 40.64	0.0341	203.20 – 223.52	0.0678
40.64 – 60.95	0.0623	223.52 – 243.83	0.0671
60.95 – 81.27	0.0722	243.83 – 264.15	0.0659
81.27 – 101.61	0.0743	264.15 – 284.49	0.0639
101.61 – 121.93	0.0735	284.49 – 304.81	0.0588
121.93 – 142.24	0.0722	304.81 – 325.12	0.0469
142.24 – 162.56	0.0709	325.12 – 345.44	0.0231
162.56 – 182.88	0.0698	345.44 – 365.76	0.0033

Source: Reference 2.2.12, Table 5.2-14

Table 6.10 TS125 Neutron Normalized Source Spectrum

Group Upper Energy (MeV)	Neutron Source Strength Fraction	Group Upper Energy (MeV)	Neutron Source Strength Fraction
Total Neutron Source Strength = 1.44E+10 neutrons/(s-canister) ^a			
1.733E+01	3.897E-05	2.972E-01	2.716E-02
1.419E+01	1.798E-04	1.832E-01	1.407E-02
1.221E+01	1.149E-03	1.111E-01	6.833E-03
1.000E+01	2.805E-03	6.738E-02	3.282E-03
8.607E+00	6.455E-03	4.087E-02	9.268E-04
7.408E+00	1.865E-02	3.183E-02	5.299E-04
6.065E+00	3.564E-02	2.606E-02	1.613E-04
4.966E+00	9.194E-02	2.418E-02	1.890E-04
3.679E+00	8.411E-02	2.188E-02	5.036E-04
3.012E+00	4.671E-02	1.503E-02	4.519E-04
2.725E+00	4.854E-02	7.102E-03	1.471E-04
2.466E+00	2.055E-02	3.355E-03	4.782E-05
2.365E+00	4.113E-03	1.585E-03	1.948E-05
2.346E+00	2.475E-02	4.540E-04	2.383E-06
2.231E+00	7.378E-02	2.144E-04	7.738E-07
1.921E+00	7.109E-02	1.013E-04	2.890E-07
1.653E+00	8.733E-02	3.727E-05	7.028E-08
1.353E+00	1.103E-01	1.068E-05	8.597E-09
1.003E+00	5.884E-02	5.043E-06	3.211E-09
8.209E-01	2.523E-02	1.855E-06	6.228E-10
7.427E-01	4.287E-02	8.764E-07	2.022E-10
6.081E-01	3.395E-02	4.140E-07	8.566E-11
4.979E-01	3.746E-02	1.000E-07	1.153E-11
3.688E-01	1.921E-02	0.000E+00	—

Source: Reference 2.2.12, Table 5.2-13, ^a Table 5.2-11

6.1.5 TAD Canister and AO

The TAD canister and AO are designed in accordance with *Transportation, Aging and Disposal Canister System Performance Specification* (Reference 2.2.9, Sections 3.1 and 3.3). The TAD canister has the capability of containing 21 pressurized water reactor (PWR) or 44 boiling water reactor (BWR) spent fuel assemblies (Reference 2.2.9, Section 3.1.1). A TAD canister containing 21 PWR assemblies is used in this calculation because it bounds a TAD canister containing 44 BWR assemblies (Reference 2.2.14, Section 6 and Reference 2.2.15, Section 6). The representative fuel assemblies, per *PWR Source Term Generation and Evaluation*, used in this calculation are B&W Mark B PWR fuel assemblies (Reference 2.2.1, p. 11) with the following sets of characteristics:

1. Maximum Source - 5.0 wt % initial ^{235}U enrichment, 80 GWd/MTU burnup, and 5 year decay time
2. Design Basis Source - 4.0 wt % initial ^{235}U enrichment, 60 GWd/MTU burnup, and 10 year decay time
3. Average Source - 4.0 wt % initial ^{235}U enrichment, 48 GWd/MTU burnup, and 25 year decay time

6.1.5.1 TAD Canister and AO Geometry

Each fuel assembly consists of four regions: top-end fitting, plenum, fuel, and bottom-end fitting (Assumption 3.2.4.1). The assembly region height dimensions are presented in Table 6.11. The plenum length is calculated as the difference between the fuel rod and active fuel lengths (e.g. $153.68 - 141.8 = 11.88$ in). Note that the value provided for the plenum length in Reference 2.2.4, p. 2A-34 is slightly smaller than the calculated value; this small difference is not expected to have significant impact on the results. The combined length of the top and bottom end fittings is the difference between the total assembly length and fuel rod length (e.g. $165.625 - 153.68 = 11.945$ in) (Reference 2.2.4, pp. 2A-31 and 2A-33). The bottom end fitting is 4 inches long (Assumption 3.2.4.3); therefore the top end fitting is the remaining difference of 7.945 inches.

Table 6.11 Assembly Region Lengths

Assembly Region	Length (in.)	Length (cm)
Fuel Assembly Total Length ^a	165.625	420.688
Fuel Rod Length ^b	153.68	390.35
Active Fuel ^c	141.8	360.2
Top end fitting ^d	7.945	20.18
Plenum region ^d	11.88	30.18
Bottom end fitting ^e	4.00	10.16

NOTE: Reference 2.2.4, ^a p. 2A-31, ^b p. 2A-33, ^c p. 2A-33

^d Calculated per above mentioned

^e Assumption 3.2.4.3

Since the radial wall thickness of the TAD canister is currently not finalized, and was not modeled, each region is homogenized within the TAD canister volume delimited by the TAD canister outer radius and the individual region height listed in Table 6.11 (Assumption 3.2.4.1). Elimination of the TAD canister walls is conservative; this is due to an increase in the homogenized area, which lowers the fuel density and therefore the attenuation effects. It also has the added effect of moving the fuel closer to the outside of the canister, which also produces higher dose rates. In the axial direction, the TAD canister has an 11 in. shield plug on the top (Reference 2.2.30, Table 12) in order to meet the surface dose rate requirements (Reference 2.2.9, Section 3.1.4). The overall modeled dimensions of the TAD canister and AO are provided in Table 6.12. The TAD canister physical design is not finalized; therefore it was modeled as a right-circular cylinder defined in the dimensional envelope specified in *Transportation, Aging and Disposal Canister System Performance Specification* (Reference 2.2.9, Section 3.1.1).

A preliminary assessment evaluating the dose rate through a steel and concrete AO, radially and axially, was performed as part of this calculation to determine the required shielding thickness to meet the AO surface dose rate requirements per the *Transportation, Aging and Disposal Canister System Performance Specification* (Reference 2.2.9, Section 3.3.4). With the maximum fuel source term, type 1 in Section 6.1.5, 1.25 in. of steel and 37.5 in. of concrete was required to meet a surface dose rate of 40 mrem/hr on the AO. The dose rate and relative error data for this analysis can be found in Attachment D, *Radial Shielding Eval.xls*. Figures 6.3 and 6.4 show the radial and axial cross-sections of the TAD with AO, respectively.

Table 6.12 Dimensions of TAD Canister and AO

Component	Dimension	
	(in.)	(cm)
Outer length of AO ^a	258	655.32
Outer diameter of AO ^a	144	365.76
Thickness of steel liner in AO ^a	1.25	3.175
Interior length of AO / Exterior length of TAD canister ^b	212	538.48
Interior diameter of AO / Exterior diameter of TAD canister ^b	66.5	168.91
Thickness of TAD canister shield plug	11	27.94
Thickness of AO top lid ^a	10.5	26.67

^a Designed to meet dose rate limits (Section 6.1.3)

^b Reference 2.2.9, Section 3.1.1

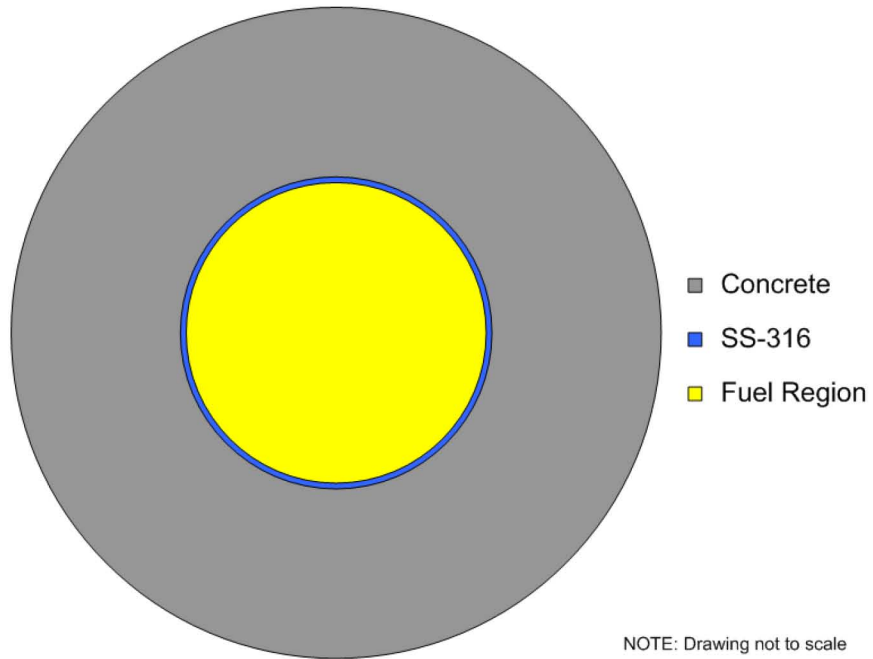


Figure 6.3 Mid-plane cross-section view of the TAD canister with steel and concrete AO radial shielding

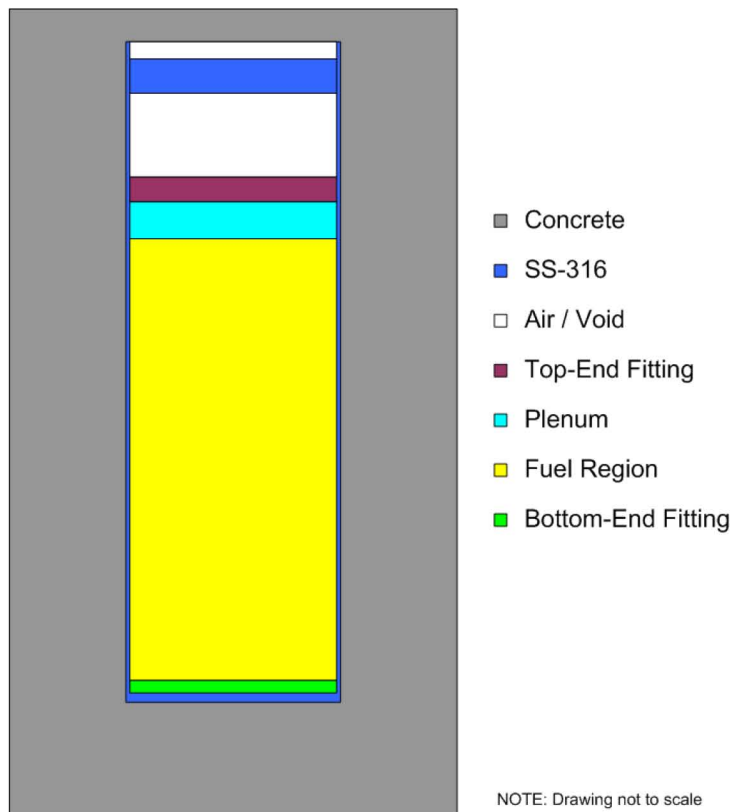


Figure 6.4 Axial cross-section of AO containing TAD canister

6.1.5.2 TAD Canister and AO Materials

The TAD is modeled with no radial walls and an SS-316 shield plug on the top. The TAD is then wrapped with SS-316 and concrete in each direction to thicknesses sufficient to meet the dose rate requirements (Reference 2.2.9, Section 3.3.4) but still fit within the dimensional requirements (Reference 2.2.9, Table 3.3.1) for the TAD AO. Table 6.13 contains the material compositions used for the AO shielding, TAD canister shield plug and the aging pad.

Table 6.13 TAD Canister with AO Material Compositions

Material	Density (g/cm ³)	Element	Weight Percent (%)	Reference
Concrete (AO)	2.35	H	0.5532	Reference 2.2.36, Table 5.2, (wt % Calculation: Attachment D, <i>materials.xls</i> , sheet <i>Concrete</i>)
		O	49.8298	
		Na	1.7021	
		Mg	0.2553	
		Al	4.5532	
		Si	31.5745	
		S	0.1277	
		K	1.9149	
SS-316 (AO and shield plug)	7.98 ^a	Ca	8.2553	Midrange values from Reference 2.2.37, SEC II A SA-240, Table 1; ^a Reference 2.2.38, Table X1
		Fe	1.2340	
		C	0.080	
		N	0.100	
		Si	0.750	
		P	0.045	
		S	0.030	
		Cr	17.000	
		Mn	2.000	
Fe	65.495			
Ni	12.000			
Mo	2.500			

^a Reference 2.2.38, Table X1

The four assembly regions of the TAD were homogenized over the radius of the TAD and their respective heights. Tables 6.14 display the maximum, design basis and average fuel material compositions of the four assembly regions calculated as homogenized regions defined by the TAD canister outer dimensions (Assumption 3.2.4.1). The fuel material compositions are calculated using fresh, unirradiated fuel (Assumption 3.2.4.2). The wt% calculations can be found in Attachment D, *materials.xls*, sheet *PWR Fuel*.

Table 6.14 Source Region Smeared Compositions for the Maximum Source

Isotope	Atom Densities per Smeared Region (radius=84.46 cm) (atoms/b-cm)				
	UO ₂ (4 wt% ²³⁵ U) ^a	UO ₂ (5 wt% ²³⁵ U) ^b	^c Bottom-End Fitting	^c Plenum	^c Top-End Fitting
²³⁵ U	1.2364E-04	1.5454E-04	----	----	----
²³⁴ U	1.0780E-06	1.3729E-06	----	----	----
²³⁶ U	5.6631E-07	7.0789E-07	----	----	----
²³⁸ U	2.9281E-03	2.8972E-03	----	----	----
Ni	6.7369E-05	6.7369E-05	1.4190E-03	1.7122E-04	8.6842E-04
Cr	3.2046E-05	3.2046E-05	1.9507E-03	7.6279E-05	1.0996E-03
Fe	3.1269E-05	3.1269E-05	5.6063E-03	7.3238E-05	3.0848E-03
Nb	4.2355E-06	4.2355E-06	3.9835E-05	1.0730E-05	2.7769E-05
Mo	2.4409E-06	2.4409E-06	1.4107E-04	6.1838E-06	7.7146E-05
Ti	1.4436E-06	1.4436E-06	1.3578E-05	3.6573E-06	9.4650E-06
Al	1.4228E-06	1.4228E-06	1.3382E-05	3.6046E-06	9.3286E-06
Co	1.3028E-06	1.3028E-06	1.2253E-05	3.3006E-06	8.5419E-06
Mn	4.8914E-07	4.8914E-07	1.3139E-04	1.3689E-06	7.3075E-05
Si	9.5683E-07	9.5683E-07	3.3401E-04	2.5192E-06	1.7762E-04
Cu	3.6248E-07	3.6248E-07	3.4092E-06	9.1830E-07	2.3766E-06
C	5.1143E-07	5.1143E-07	1.6687E-05	1.3401E-06	9.6818E-06
S	3.5917E-08	3.5917E-08	4.1576E-07	9.4325E-08	3.8462E-07
P	3.7184E-08	3.7184E-08	4.7077E-07	9.9376E-08	4.7539E-07
¹⁰ B	8.4801E-09	8.4801E-09	7.9755E-08	2.1483E-08	5.5597E-08
¹¹ B	3.4133E-08	3.4133E-08	3.2102E-07	1.7208E-08	2.2379E-07
Sn	2.2034E-05	2.2034E-05	----	2.1911E-05	----
Zr	1.9399E-03	1.9399E-03	----	1.9291E-03	----
O	6.1690E-03	6.1690E-03	----	1.4019E-05	----
N	----	----	5.9504E-07	2.5438E-08	1.1384E-06
Total	1.1328E-02	1.1329E-02	9.6835E-03	2.3196E-03	5.4501E-03

^a UO₂ composition used for design basis and average fuel

^b UO₂ composition used for maximum fuel

^c Bottom end fitting, plenum, and top end fitting data used for all three fuel compositions

Source: Attachment D, *materials.xls*, sheet *PWR Fuel*

6.1.5.3 TAD Canister and AO Source Specification

The maximum source term is used in the TAD canister within the AO, per source characteristics defined in *Transportation, Aging and Disposal Canister System Performance Specification* (Reference 2.2.9, Section 3.1.1, (5)), to determine steel and concrete shielding thicknesses to meet surface dose rate requirements (Reference 2.2.9, Section 3.3.4). In the aging pad dose rate calculations, the TAD canister will be loaded with the design basis and average fuel sources, the characteristics of which are covered in section 6.1.5. Tables 6.15a through 6.15c present the gamma and neutron source terms of the maximum, design basis, and average PWR SNF assemblies used in the MCNP5 calculations. The gamma and neutron axial source profiles for all three source terms are listed in Table 6.16. Note that the neutron source term is only present in the active fuel region and the axial gamma and neutron source profiles apply to the active fuel region sources only.

Table 6.15a Gamma and Neutron Sources for Maximum PWR SNF Assembly ^a

		Gamma Intensity (photons/sec)			Neutron Intensity (neutrons/sec)	
Upper Energy Boundary (MeV)	Bottom-End Fitting Region ^b	Active Fuel Region ^c	Plenum Fuel Region ^d	Top-End Fitting Region ^e	Upper Energy Boundary (MeV)	Active Fuel Region ^c
5.00E-02	5.94E+11	2.33E+15	5.28E+11	3.79E+11	1.00E-08	0.00E+00
1.00E-01	1.16E+11	6.44E+14	6.09E+10	7.43E+10	3.00E-08	0.00E+00
2.00E-01	2.83E+10	5.22E+14	3.52E+10	1.79E+10	5.00E-08	0.00E+00
3.00E-01	1.41E+09	1.48E+14	1.96E+09	8.91E+08	1.00E-07	0.00E+00
4.00E-01	1.90E+09	9.85E+13	5.86E+09	1.17E+09	2.25E-07	0.00E+00
6.00E-01	1.91E+09	1.53E+15	1.10E+11	7.41E+07	3.25E-07	0.00E+00
8.00E-01	4.35E+09	4.70E+15	5.95E+10	2.37E+09	4.00E-07	0.00E+00
1.00E+00	1.37E+11	7.08E+14	8.03E+09	7.66E+10	8.00E-07	0.00E+00
1.33E+00	3.38E+13	4.55E+14	1.74E+13	2.17E+13	1.00E-06	0.00E+00
1.66E+00	9.53E+12	1.30E+14	4.91E+12	6.12E+12	1.13E-06	0.00E+00
2.00E+00	1.87E+03	1.44E+12	9.19E+02	1.13E+03	1.30E-06	0.00E+00
2.50E+00	2.26E+08	2.49E+12	1.16E+08	1.45E+08	1.77E-06	0.00E+00
3.00E+00	3.51E+05	1.10E+11	1.81E+05	2.25E+05	3.05E-06	0.00E+00
4.00E+00	7.66E-08	1.39E+10	1.00E-08	4.16E-08	1.00E-05	0.00E+00
5.00E+00	0.00E+00	7.09E+07	0.00E+00	0.00E+00	3.00E-05	0.00E+00
6.50E+00	0.00E+00	2.86E+07	0.00E+00	0.00E+00	1.00E-04	0.00E+00
8.00E+00	0.00E+00	5.58E+06	0.00E+00	0.00E+00	5.50E-04	0.00E+00
1.00E+01	0.00E+00	1.19E+06	0.00E+00	0.00E+00	3.00E-03	0.00E+00
Total	4.42E+13	1.13E+16	2.31E+13	2.84E+13	1.70E-02	0.00E+00
					1.00E-01	0.00E+00
					4.00E-01	8.05E+07
					9.00E-01	4.11E+08
					1.40E+00	3.76E+08
					1.85E+00	2.76E+08
					3.00E+00	4.85E+08
					6.43E+00	4.43E+08
					2.00E+01	3.93E+07
					Total	2.11E+09

Source: Reference 2.2.1, Attachment X

^a 5.0 wt % initial enrichment, 80 GWd/MTU burnup and 5 years cooling time.

^b File Waste.Stream.E2.R2.B14.cut

^c File Waste.Stream.E2.R1.B14.cut

^d File Waste.Stream.E2.R3.B14.cut

^e File Waste.Stream.E2.R4.B14.cut

Table 6.15b Gamma and Neutron Sources for Design Basis PWR SNF Assembly ^a

Upper Energy Boundary (MeV)	Gamma Intensity (photons/sec)				Neutron Intensity (neutrons/sec)	
	Bottom-End Fitting Region ^b	Active Fuel Region ^c	Plenum Fuel Region ^d	Top-End Fitting Region ^e	Upper Energy Boundary (MeV)	Active Fuel Region ^c
5.00E-02	2.73E+11	1.21E+15	1.88E+11	1.75E+11	1.00E-08	0.00E+00
1.00E-01	5.28E+10	3.29E+14	2.77E+10	3.39E+10	3.00E-08	0.00E+00
2.00E-01	1.28E+10	2.45E+14	1.17E+10	8.19E+09	5.00E-08	0.00E+00
3.00E-01	6.39E+08	7.13E+13	6.33E+08	4.07E+08	1.00E-07	0.00E+00
4.00E-01	8.50E+08	4.55E+13	1.64E+09	5.33E+08	2.25E-07	0.00E+00
6.00E-01	4.92E+08	2.26E+14	2.69E+10	3.37E+07	3.25E-07	0.00E+00
8.00E-01	2.91E+09	2.37E+15	1.60E+10	1.86E+09	4.00E-07	0.00E+00
1.00E+00	5.40E+09	1.22E+14	2.48E+09	3.41E+09	8.00E-07	0.00E+00
1.33E+00	1.54E+13	1.95E+14	7.97E+12	9.90E+12	1.00E-06	0.00E+00
1.66E+00	4.35E+12	4.50E+13	2.25E+12	2.80E+12	1.13E-06	0.00E+00
2.00E+00	2.35E+00	1.52E+11	1.49E+02	2.15E-02	1.30E-06	0.00E+00
2.50E+00	1.03E+08	5.17E+10	5.34E+07	6.64E+07	1.77E-06	0.00E+00
3.00E+00	1.60E+05	3.79E+09	8.29E+04	1.03E+05	3.05E-06	0.00E+00
4.00E+00	9.43E-10	4.97E+08	1.55E-10	5.19E-10	1.00E-05	0.00E+00
5.00E+00	0.00E+00	2.82E+07	0.00E+00	0.00E+00	3.00E-05	0.00E+00
6.50E+00	0.00E+00	1.13E+07	0.00E+00	0.00E+00	1.00E-04	0.00E+00
8.00E+00	0.00E+00	2.22E+06	0.00E+00	0.00E+00	5.50E-04	0.00E+00
1.00E+01	0.00E+00	4.71E+05	0.00E+00	0.00E+00	3.00E-03	0.00E+00
Total	2.01E+13	4.86E+15	1.05E+13	1.29E+13	1.70E-02	0.00E+00
					1.00E-01	0.00E+00
					4.00E-01	3.16E+07
					9.00E-01	1.61E+08
					1.40E+00	1.48E+08
					1.85E+00	1.09E+08
					3.00E+00	1.91E+08
					6.43E+00	1.74E+08
					2.00E+01	1.54E+07
					Total	8.30E+08

Source: Reference 2.2.1, Attachment X

^a 4.0 wt % initial enrichment, 60 GWd/MTU burnup and 10 years cooling time.^b File Waste.Stream.E5.R2.B11.cut^c File Waste.Stream.E5.R1.B11.cut^d File Waste.Stream.E5.R3.B11.cut^e File Waste.Stream.E5.R4.B11.cut.

Table 6.15c Gamma and Neutron Sources for Average PWR SNF Assembly ^a

Gamma Intensity (photons/sec)					Neutron Intensity (neutrons/sec)	
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	3.36E+10	6.70E+14	1.86E+10	2.17E+10	1.00E-08	0.00E+00
1.00E-01	6.02E+09	1.99E+14	3.14E+09	3.87E+09	3.00E-08	0.00E+00
2.00E-01	1.46E+09	1.26E+14	8.53E+08	9.36E+08	5.00E-08	0.00E+00
3.00E-01	7.30E+07	3.89E+13	4.39E+07	4.69E+07	1.00E-07	0.00E+00
4.00E-01	9.47E+07	2.63E+13	7.16E+07	6.07E+07	2.25E-07	0.00E+00
6.00E-01	1.41E+07	2.05E+13	5.04E+08	3.83E+06	3.25E-07	0.00E+00
8.00E-01	2.08E+09	1.24E+15	1.92E+09	1.44E+09	4.00E-07	0.00E+00
1.00E+00	2.08E+09	1.10E+13	1.64E+09	1.44E+09	8.00E-07	0.00E+00
1.33E+00	1.75E+12	2.95E+13	9.09E+11	1.12E+12	1.00E-06	0.00E+00
1.66E+00	4.94E+11	5.13E+12	2.57E+11	3.18E+11	1.13E-06	0.00E+00
2.00E+00	9.73E-01	6.75E+10	6.14E+01	8.72E-03	1.30E-06	0.00E+00
2.50E+00	1.17E+07	3.53E+09	6.09E+06	7.54E+06	1.77E-06	0.00E+00
3.00E+00	1.82E+04	2.88E+08	9.44E+03	1.17E+04	3.05E-06	0.00E+00
4.00E+00	1.86E-11	1.98E+07	1.49E-11	1.27E-11	1.00E-05	0.00E+00
5.00E+00	0.00E+00	6.69E+06	0.00E+00	0.00E+00	3.00E-05	0.00E+00
6.50E+00	0.00E+00	2.69E+06	0.00E+00	0.00E+00	1.00E-04	0.00E+00
8.00E+00	0.00E+00	5.27E+05	0.00E+00	0.00E+00	5.50E-04	0.00E+00
1.00E+01	0.00E+00	1.12E+05	0.00E+00	0.00E+00	3.00E-03	0.00E+00
Total	2.29E+12	2.37E+15	1.19E+12	1.47E+12	1.70E-02	0.00E+00
					1.00E-01	0.00E+00
					4.00E-01	7.48E+06
					9.00E-01	3.82E+07
					1.40E+00	3.52E+07
					1.85E+00	2.61E+07
					3.00E+00	4.69E+07
					6.43E+00	4.18E+07
					2.00E+01	3.65E+06
					Total	1.99E+08

Source: Reference 2.2.1, Attachment X

^a Note: 4.0 wt % initial enrichment, 48 GWd/MTU burnup and 25 years cooling time.^b File Waste.Stream.E5.R2.B9.cut^c File Waste.Stream.E5.R1.B9.cut^d File Waste.Stream.E5.R3.B9.cut^e File Waste.Stream.E5.R4.B9.cut.

Table 6.16 Axial Source Profile for the PWR Fuel Region

Axial boundaries (from mid-plane) (cm)	TOP		BOTTOM	
	Neutron	Gamma	Neutron	Gamma
0	1.554	1.117	1.554	1.117
11.43	1.537	1.114	1.571	1.12
22.86	1.521	1.111	1.588	1.123
34.29	1.504	1.108	1.605	1.126
45.72	1.486	1.104	1.622	1.129
57.15	1.464	1.1	1.636	1.131
68.58	1.438	1.095	1.648	1.133
80.01	1.401	1.088	1.657	1.135
91.44	1.35	1.078	1.654	1.134
102.87	1.277	1.063	1.625	1.129
114.3	1.165	1.039	1.554	1.117
125.73	0.998	0.9995	1.414	1.091
137.16	0.769	0.9365	1.172	1.041
148.59	0.492	0.8375	0.816	0.951
160.02	0.22	0.685	0.402	0.797
171.45	0.046	0.4625	0.092	0.551
182.88	0	0	0	0

Source: Reference 2.2.18, Table S4.4.5

6.1.6 Aging Pad

The aging pad is designed to stage AOs that contain canisters loaded with SNF until the thermal output of the SNF has been reduced to a level acceptable for emplacement. The design of the aging pads can be found in the *Aging Facility General Arrangement Aging Pad Plans* (References 2.2.19, 2.2.20, and 2.2.21). The aging facility design consists of two aging pad areas (pads 17P and 17R) with a combined aging capacity of 2400 vertical AOs and 100 HAMs (Reference 2.2.20 Note 3 and Reference 2.2.21, Notes 3 and 4). Each sub-pad used to age vertical AOs is 114 ft. wide with lengths ranging from 640 to 1030 ft. The body of the pad is modeled as a 3 ft. of concrete which provides for the effects of ground scatter which are important when close to the AOs. Though this pad thickness does not agree with the *Facility General Arrangement Aging Pad Sections* (Reference 2.2.22) it is acceptable to provide maximum ground scatter. The two pads designed to hold the HAMs are both 63 ft. wide and 481 ft. long. AOs on each pad are arranged in 4x4 arrays with an 18 ft. pitch in the array and a 78 ft. pitch between arrays (References 2.2.20 and 2.2.21). HAMs are not modeled in this calculation because there was sufficient worker dose data in the *NUHOMS HD System Safety Analysis Report* (Reference 2.2.17). This data is the best representation of HAM operations until a final design is developed for the HAMs to be used.

6.1.7 Calculated Outputs

6.1.7.1 Relative Error Calculation

The estimated relative error, associated with all of the MCNP results presented in this document, of the total dose rate (from doses with more than one component) is derived from the estimated variance of the total dose rate. The estimated variance of the total dose rate, S^2_{Total} , is the sum of the estimated variances of the individual dose rates S^2_i . The estimated total dose rate, estimated variance, and relative error (Reference 2.2.29, p. 2-93) are derived according to Equations 1 through 4 and utilized in the all Excel files containing dose rate results.

$$R_i = \frac{\sqrt{S_i^2}}{T_i} \quad \text{Equation 9}$$

$$S^2_{Total} = \sum_{i=1}^n S_i^2 \quad \text{Equation 10}$$

$$T_{Total} = \sum_{i=1}^n T_i \quad \text{Equation 11}$$

$$R_{Total} = \frac{\sqrt{S^2_{Total}}}{T_{Total}} = \frac{\sqrt{\sum_{i=1}^n S_i^2}}{T_{Total}} = \frac{\sqrt{\sum_{i=1}^n (R_i \times T_i)^2}}{T_{Total}} \quad \text{Equation 12}$$

where:

- i = tally component index
- n = total number of components
- T_{Total} = total estimated tally
- T_i = tally i component
- S^2_{Total} = total estimated variance
- S^2_i = variance of the i component
- R_i = relative error of the i component
- R_{Total} = total estimated relative error

6.2 DESCRIPTION OF DOSE RATE CALCULATIONS

Dose rate fields around a TS125 transportation cask and an AO are both calculated using MCNP5. The TS125 cask, single AOs, and multiple AOs are all surrounded by void. The lack of air leads to conservative results by neglecting attenuation that air would provide.

6.2.1 Cask Receipt

A detailed dose field was calculated surrounding a single TS125 transportation cask in the radial direction. Dose rates were calculated at 0.25 meter (0.82 ft.) increments from the cask surface to a distance of 5 meters and inside axial segments of equal length (45.4 in./115.4cm). These dose fields were used with the personnel, time, and distance assumptions (Assumptions 3.2.1.1, 3.2.1.2, and 3.2.1.3, respectively) to produce worker total direct dose for cask receipt tasks. Assumption 3.1.1.1 was used to evaluate the resulting doses based on a yearly acceptance rate.

6.2.2 GROA

6.2.2.1 Transportation Cask Operations

To evaluate the transient and stationary dose fields in the GROA, results from the *GROA Shielding Requirements Calculation* were used (Reference 2.2.30). This calculation used the TS125 transportation cask to evaluate transient dose rates in the GROA and 25 TS125 transportation casks to evaluate the rail car buffer area (33A). The calculation determined that a distance of 70 ft. from a single TS125 was sufficient to reduce the dose rate to 0.25 mrem/hr (Reference 2.2.30, Section 7.2.1.2) while a distance of 165 ft. from the rail car buffer area with 25 TS125 rail casks is required to meet the same dose rate (Reference 2.2.30, Section 7.2.1.1). The location of non-nuclear facilities relative to paths of waste transport was determined based on the *Geologic Repository Operations Area North Portal Site Plan* (Reference 2.2.31) and a memo concerning the minimum distances of the non-nuclear facilities from paths of transport (Reference 2.2.8). Using this dose field and location information along with the assumptions about travel speed and facility work crew requirements from Section 3.2.2, the total direct worker doses were determined. The assumption used was 3.2.2.1.

Dose to workers transporting the cask throughout the GROA was also considered. Based on the personnel and dose rate assumptions made in Assumptions 3.2.2.1 and 3.2.7, along with the travel distance provided in a memo (Reference 2.2.8) (Assumption 3.1.2.2), a dose per cask was determined. Table 6.17 shows the distances between; the cask receipt security station (30B) and the rail car buffer area (33A), the rail car buffer area (33A) and CRCF-2, and the cask receipt security station (30B) and CRCF-2. The distance from 30B to CRCF-2 is a conservative approximation of the average distance a transportation cask would have to travel to a nuclear facility. Annual acceptance data, Assumption 3.1.1.1, was used to determine worker dose per year.

Table 6.17 Travel Distance and Time for Transportation Casks Inside the GROA

	Travel Distance (ft)	Travel Time (hrs)
30B to 33A	2250	0.085
33A to CRCF-2	4100	0.155
30B to CRCF-2	6350	0.241

6.2.2.2 TEV Operations

The TEV is designed to transport a WP from the nuclear facilities to the subsurface for emplacement. The TEV will be operated via remote controls from the Central Control Center Facility (CCCF) and programmable logic controllers (PLCs) on the TEV itself (Reference 2.2.49). Though the TEV is operated remotely and monitored by on board cameras, site personnel are still expected to accompany the TEV to ensure safe transport. Per Assumption 3.2.2.2, two operations personnel and one HP technician are expected to accompany the TEV while on the surface. It is again assumed that all personnel will be able to maintain a distance from the TEV sufficient to receive a dose rate of 0.25 mrem/hr or less while performing their duties (Assumption 3.2.7). The average distance traveled by the TEV during waste transport is based on Assumption 3.1.2.3 while the average number of waste packages emplaced per year is based on Assumption 3.1.1.2.

6.2.3 TAD Canister and AO

An AO loaded with a TAD canister containing the maximum fuel source must meet the surface contact dose rate criteria of 40 mrem/hr on all accessible surfaces except the bottom per the *Transportation, Aging and Disposal Canister System Performance Specification* (Reference 2.2.9, Section 3.3.4). To achieve this required dose rate, an analysis was run using the maximum source to determine the steel and concrete shielding thicknesses necessary. This overpack was then used to calculate a dose field around a single AO, a fully loaded 4x4 array, and a partially loaded 4x4 array.

6.2.4 Aging Facility

6.2.4.1 Transport of AOs

The first part of the worker dose assessment for the aging facility begins when the AO leaves the nuclear facility and ends when it reaches the aging pad. This portion is also split into two sections; travel from the nuclear facilities to the aging facility perimeter and travel inside the perimeter. The 0.25 mrem/hr dose rate of Assumption 3.2.7 was applied during transport outside of the aging facility perimeter. Three full 4x4 arrays provide dose during travel inside the perimeter of the aging facility (Additional arrays further away do not significantly increase worker exposures). These contributions combined with assumptions for annual aging pad throughput, AO transfer staffing requirements, and AO transporter average speed (Assumptions 3.1.1.3, 3.2.3.1, and 3.2.3.2, respectively) determine total direct worker doses for this operation. TAD AOs with design basis fuel are used to evaluate all worker doses as AOs are moved to the pad for aging while TAD

AOs with average fuel are used to evaluate worker doses as AOs are moved from the pads back to the nuclear facilities (Assumption 3.2.4.4). Design basis fuel is used for all TADs in the full 4x4 arrays. The annual throughput, AO transfer staffing requirements, and site transporter speed with a loaded AO used for the return trip are assumed to be the same as they were for the trip to the pads (Assumption 3.2.3.4) and the number of AOs retrieved during peak operation is expected to be equal to the number of AOs placed (Assumption 3.1.1.3).

Dose rates were calculated along the length and on either end of 3 fully loaded 4x4 arrays described in section 6.1.6 at distances of 1, 2, 5, and 10 meters or 3.28, 6.56, 15.24, and 32.81 ft., respectively. All of the tallies along the length of the full 4x4 arrays are 21.2 in. tall (53.848 cm) centered at the peak axial location of the TAD. The tallies run the length and width of the pad and are each 144 in. wide (365.76 cm). Figure 6.5 shows a simplified representation of the detector locations with only two full arrays.

6.2.4.2 Placement of AOs

Once the AOs have reached their aging location, personnel must lower the AO in the proper location and attach monitoring devices. Though the current plan does not call for the use of seismic restraints, an analysis was performed on the dose impact of this activity and is included as Attachment C. Dose rates were calculated at various locations inside of a partially loaded 4x4 array in order to evaluate the total direct dose to workers during placement and securing operations. The geometry of the partially loaded array was determined based on the loading sequence described in Assumption 3.1.4 and diagrammed in Figure 6.6 along with the detector locations used. This dose rate data along with assumptions on the annual aging facility throughput and AO placement task data (Assumptions 3.1.1.3 and 3.2.3.3) were used to determine direct dose rates during placement operations. It is assumed that the process for retrieving an AO from the pads includes the same tasks in reverse and would therefore require the same time and proximity to the AO (Assumption 3.2.3.4). However design basis fuel is used to calculate placement operations while average fuel is used to calculate retrieval processes (Assumption 3.2.4.4) resulting in a lower dose rate.

6.2.4.3 Maintenance and Inspection Operations

While the AOs are on the pad they will be visually inspected on a regular basis and could require maintenance for various reasons. It is assumed that inspections would be performed once every 3 months (quarterly) and it would take the inspector approximately 30 seconds to visually inspect each AO at a distance of 2 meters from the outer AO (Assumption 3.2.3.5). Though it is hard to approximate any maintenance operations that might be required, it is assumed that each event will take one hour for a two-man team to perform at varying distances from the AOs (Assumption 3.2.3.6). Dose rate data from inside a fully loaded 4x4 array will be used to determine worker doses for the maintenance operations. In both the inspection and maintenance analysis design basis fuel will be used.

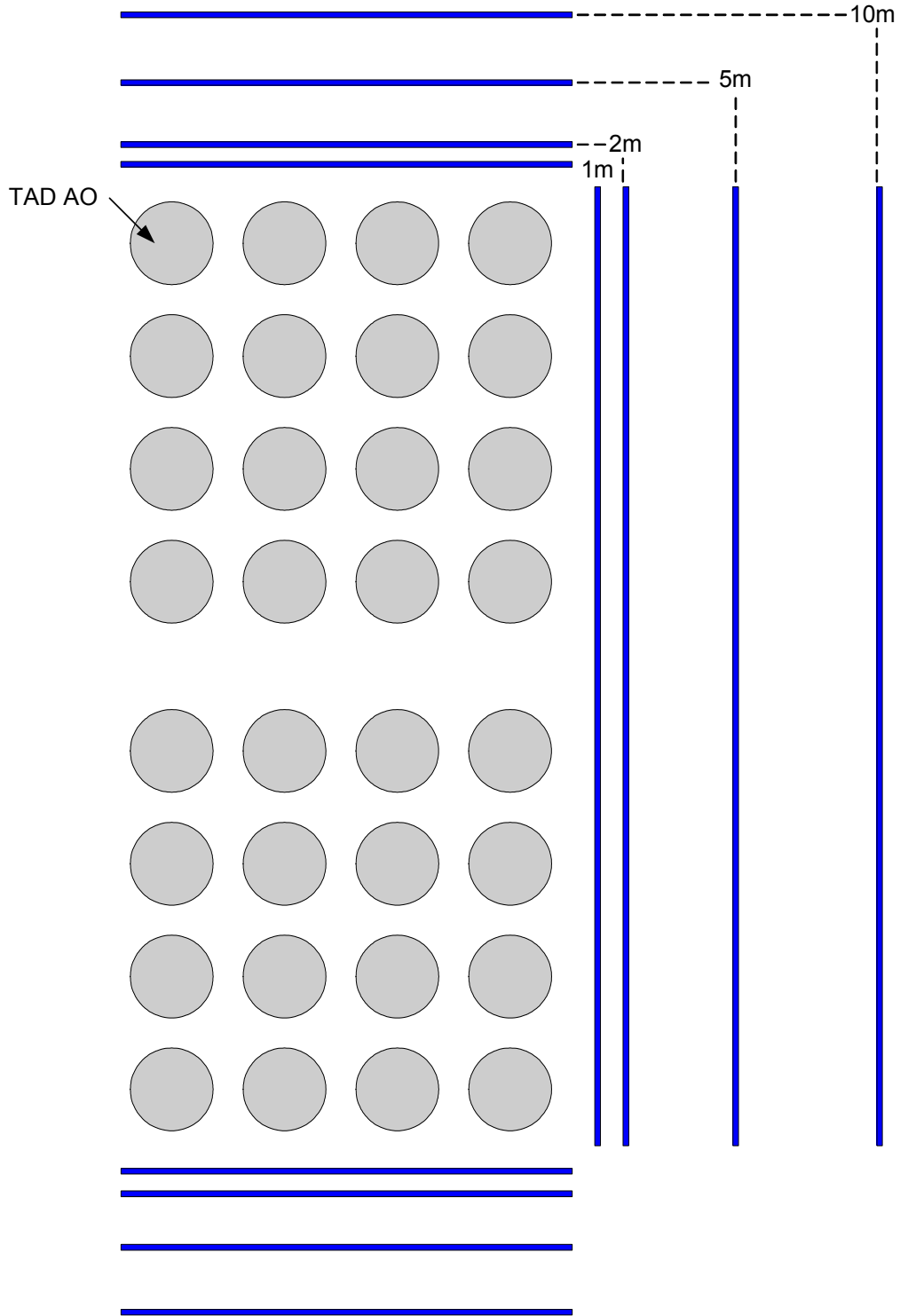


Figure 6.5 Detector Location Diagram for a Fully Loaded 4x4 Array

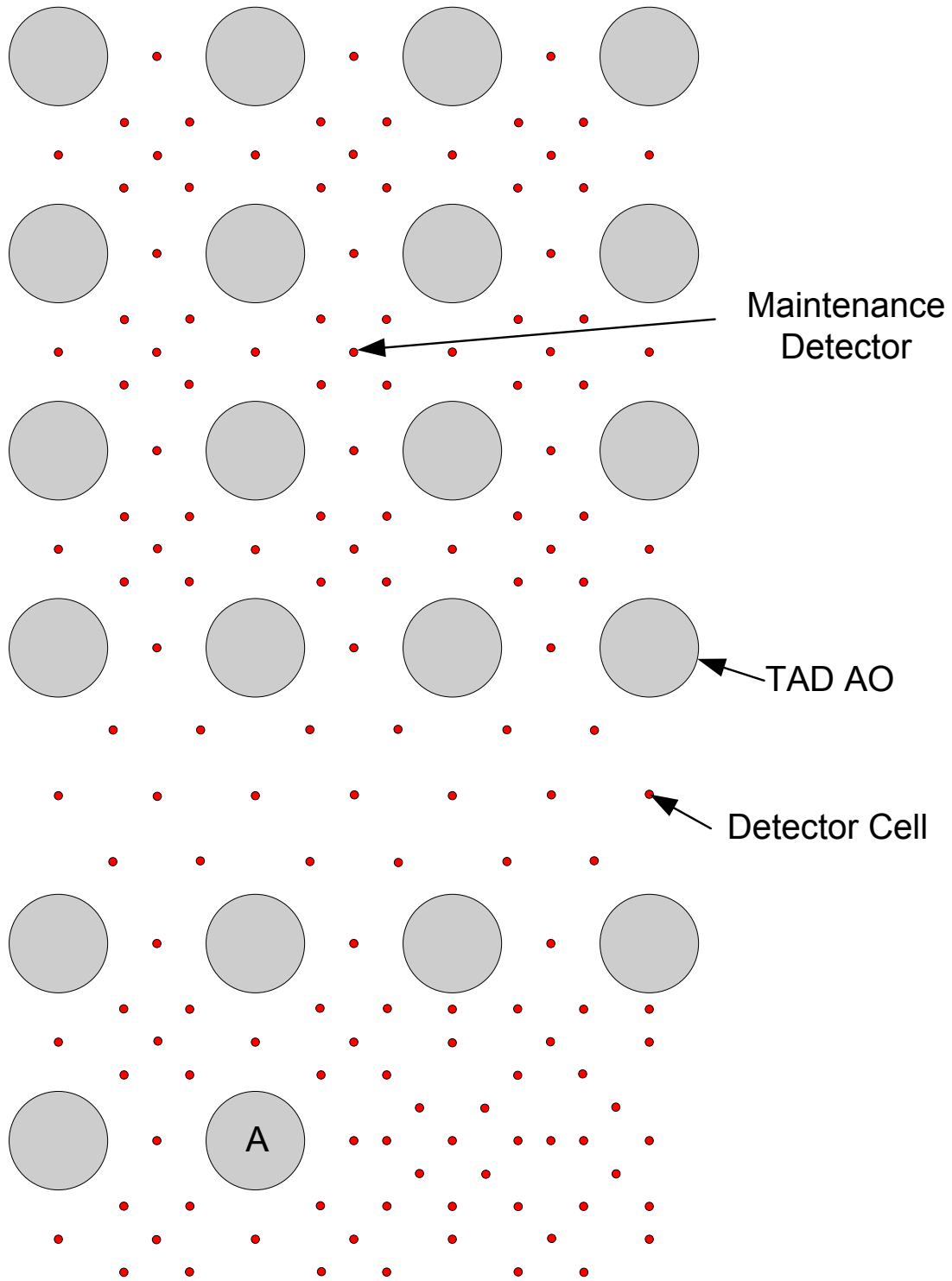


Figure 6.6 Detector Locations for a Partially Loaded Array

6.2.4.4 Horizontal Aging Modules

Horizontal aging modules will be used to house horizontal dual purpose canister (DPCs) for aging. The HAM was not modeled in this calculation; rather information from the *NUHOMS HD System Safety Analysis Report* (Reference 2.2.17) on worker doses for HAM operations will be reported. Table 6.18 gives a summary of the time and motion data found in the SAR (Reference 2.2.17, Table 10-1). The source term used to generate these worker doses is 32 BWR assemblies with: 60 GWd/MTU burnup, 4% initial enrichment, and 7 years cooling time (Reference 2.2.17, Section 5.2).

Table 6.18 Time and Motion Data for the NUHOMS HD Horizontal Aging Module

Task Description	No. of workers	Duration (hr)
Prepare the HSM-H and hydraulic ram	2	2
Transport the Cask to ISFSI	6	1
Position the Cask in Close proximity with the HSM-H	3	1
Remove the Cask Lid	2	1
Align and Dock the Cask with the HSM-H	2	0.25
Position and Align Ram with Cask	2	0.5
Remove Ram Access Cover Plate	1	0.25
Transfer the DSC from the Cask to the HSM-H	3	0.5
Lift the Ram Back onto the Trailer and Un-Dock the Cask from the HSM-H	2	0.083
Install HSM-H Access Door	2	0.5
Totals	N/A	7.08

Source: Reference 2.2.17, Table 10-1

6.2.5 Time in Low Radiation Support Areas

Assumption 3.2.6 states that any time GROA personnel are not performing transfer operations they will remain in a dose field of 0.05 mrem/hr performing support operations. Equation 3 of Section 4.3.1 is used to determine this support only time for each operation. Table 6.19 list the total task times (T_o) and total support only time (T_n) for each work crew. The support only times (T_n) are calculated based on a 2000 hour work year.

Table 6.19 Total Task and Support Only Times

Operation	Task Time per Cask (hrs/cask)	Task Time per Year (hrs/year)	Support Only time (hrs/year)
Cask Receipt	2.37	237 ^a	1763
Transportation Cask Operations	0.24	120 ^b	1880
TEV Operations	0.35	111 ^c	1889
AO Operations (275) ^f	9.3	428 ^d	1572
AO Operations (135) ^f	9.3	214 ^e	1786

^a Cask receipt task time based on 100 casks processed per year [500 casks / 5 crews = 100 casks/crew]

^b Transportation cask operation task time based on 500 casks processed per year

^c TEV operation task time based on 316 WPs processed

^d AO placement operations based on 46 AOs placed per year

^e AO placement operations based on 23 AOs placed per year

^f AO Operations includes placement and retrieval in one round trip

7. RESULTS AND CONCLUSIONS

7.1 RESULTS

7.1.1 Cask Receipt

As casks are received at the entrance of the repository, they are visually and radiologically inspected by operations and health physics personnel, respectively, and any personnel barriers are removed (Reference 2.2.32). Assumptions 3.2.1.1, 3.2.1.2, and 3.2.1.3 are used to estimate staffing requirements, task times, and distance information for the receipt processes. Table 7.1 lists the peak dose rates as a function of distance away from a TS125 transportation cask to be used as dose rate input for the direct dose evaluation. These dose rates were evaluated using MCNP5 with the characteristics described in Section 6.1.4. Note that the dose rates presented in Table 7.1 are based on a distance from the surface of the cask as opposed to the edge of the conveyance as dictated by 10 CFR Part 71.47 (Reference 2.3.1), therefore the 10.64 mrem/hr dose rate reported in the table is acceptable. Figure 7.1 shows a graphic representation of the data in Table 7.1. Application of these dose rates to the task times for each worker on a per task and total operation basis is shown in Table 7.2. The dose data presented uses Equations 1 and 2 to determine dose per task and dose per operation, respectively, as described in Section 4.3.1. Dose rate results and relative error calculations for the data presented in Table 7.1 and Figure 7.1 can be found in Attachment D, *TS125 Dose Rates.xls*, sheet *Radial Dose Rate*.

Table 7.1 Radial Dose Rate for a TS125 Transportation Cask

Distance From TS125 Surface (m)	Peak Dose Rate (mrem/hr)
0.25	30.20
0.50	24.81
0.75	20.95
1.00	17.92
1.25	15.56
1.50	13.63
1.75	12.01
2.00	10.64
2.25	9.48
2.50	8.49
2.75	7.63
3.00	6.89
3.25	6.24
3.50	5.68
3.75	5.18
4.00	4.74
4.25	4.35
4.50	4.00
4.75	3.70
5.00	3.42

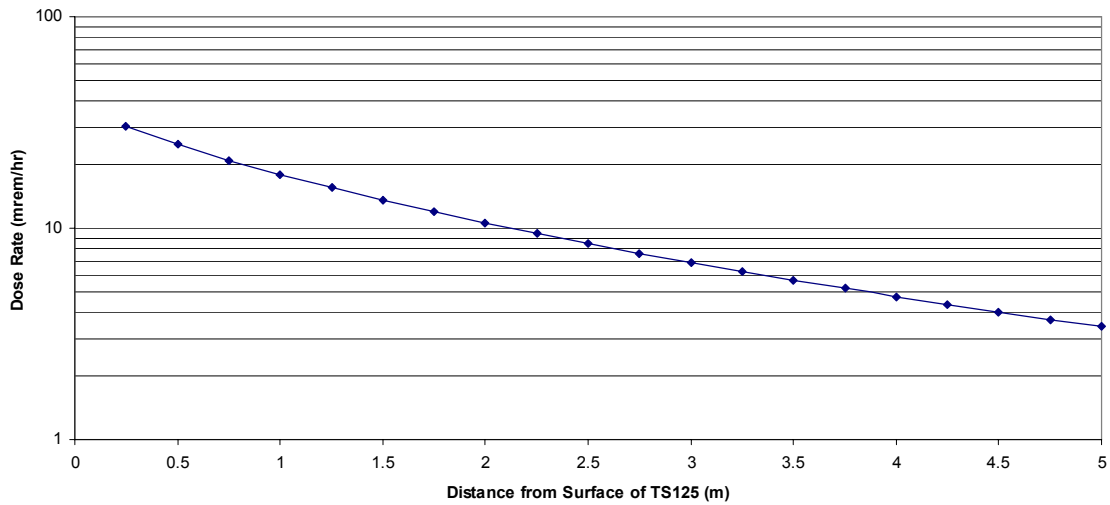


Figure 7.1 Peak Radial Dose Rate as a Function of Distance from a TS125

Table 7.2 Direct External Dose Data for Cask Receipt Operations

	[A]	[B]	[C]	[D]	[E]	[F]	[G]	[H]	[I]	[J]
Task	Time Required (min)	Security			Operators			HPTs		
		Distance from Cask (m)	Dose Rate at Distance (mrem/hr)	Dose/Task (mrem/task)	Distance from Cask (m)	Dose Rate at Distance (mrem/hr)	Dose/Task (mrem/task)	Distance from Cask (m)	Dose Rate at Distance (mrem/hr)	Dose/Task (mrem/task)
Receipt documentation outside of gate	10	1	17.92	2.99	--	--	--	5	3.42	0.57
Initial inspection outside of gate, Radiation and contamination survey	12	5	3.42	0.68	--	--	--	1	17.92	3.58
Railcar or trailer towed into YMP area using YMP equipment and personnel	10	--	--	--	5	3.42	0.57	5	3.42	0.57
Remove tie downs, attach lifting devices to personnel barrier	20	--	--	--	2	10.64	3.55	5	3.42	1.14
Remove and store personnel barrier	45	--	--	--	2	10.64	7.98	--	--	--
Second inspection, Surface contamination survey	20	5	3.42	1.14	--	--	--	1	17.92	5.97
Railcar or trailer towed into security building	10	--	--	--	5	3.42	0.57	5	3.42	0.57
Final Inspection, Damage	15	5	3.42	0.86	--	--	--	--	--	--
Individual Dose per Cask (person-mrem/cask)				5.67^a			12.67^b			12.41^c

Source: Attachment D, *TS125 Dose Rates.xls*, sheet *Radial Dose Rate*

[A] Assumed time data per task (Assumption 3.2.1.2)

[B] Assumed task distances for Security (Assumption 3.2.1.3)

[C] Dose rate at distance data from Table 7.1

[D] Column C X (Column A / 60 min/hr), Equation 1

[E] Assumed task distances for Operators (Assumption 3.2.1.3)

[F] Dose rate at distance data from Table 7.1

[G] Column F X (Column A / 60 min/hr), Equation 1

[H] Assumed task distances for HPTs (Assumption 3.2.1.3)

[I] Dose rate at distance data from Table 7.1

[J] Column H X (Column A / 60 min/hr), Equation 1

^a Sum of doses in Column D, Equation 2

^b Sum of doses in Column G, Equation 2

^c Sum of doses in Column J, Equation 2

Table 7.2 shows that the individual dose for security personnel is 5.67 mrem/cask, the individual dose for each operations personnel is 12.67 mrem/cask, and the dose for HPT personnel is 12.41 mrem/cask. Computation of these dose results can be found in Attachment D, *Receipt Dose.xls*, sheet *Receipt Dose*.

7.1.2 GROA

Operations in the GROA include movement of transportation casks from the cask receipt security station (30B) to the nuclear facilities and transporting loaded waste packages inside a TEV to the subsurface. In some cases the transportation casks will have to go to either the rail or truck buffer area (33A and 33B, respectively).

7.1.2.1 Transportation Cask Operations

Data from the *GROA Shielding Requirements Calculation* (Reference 2.2.30, Sections 7.2.1.1 and 7.2.1.2) using TS125 rail casks is used along with Assumption 3.1.2.2 and 3.1.3 to determine dose impact to workers in the non-nuclear facilities for both transportation cask and TEV movement. The calculation determined that a distance of 70 ft. from a single TS125 was sufficient to reduce the dose rate to 0.25 mrem/hr (Reference 2.2.30, Section 7.2.1.2) while a distance of 165 ft. from the rail car buffer area was sufficient to reduce the dose rate to 0.25 mrem/hr (Reference 2.2.30, Section 7.2.1.1). Table 7.3 summarizes the non-nuclear facilities and their proximity to any transient or stationary source in the GROA (Reference 2.2.8).

Table 7.3 Separation of Non-Nuclear Facilities and Waste Transport Paths in the GROA

Area Number	Description	Normal Occupational Status	Distance to Closest Transport Path
220	Heavy Equipment Maintenance Building (HEMF)	Continuously	≥ 75 ft
230	Warehouse and Non-Nuclear Receipt Facility (WNNRF)	Continuously	≥ 75 ft
240	Central Control Center Facility (CCCF)	Continuously	≥ 75 ft
25A	Utilities Facility (UF)	Continuously	≥ 75 ft
25B	Cooling Tower	Never	≥ 75 ft
25C	Evaporation Pond	Never	≥ 75 ft
26B	Standby Diesel Generator Facility	Rarely	≥ 75 ft
26D	Emergency Diesel Generator Facility	Rarely	~ 30 ft
27A	Switchyard (138 kV)	Rarely	≥ 75 ft
27B	13.8 kV Switchgear Facility	Rarely	≥ 75 ft
28A	Fire Water Facility	Rarely	≥ 75 ft
28B	Fire Water Facility	Rarely	≥ 75 ft
28F	Fire Water Facility	Rarely	≥ 75 ft
290	Aging Overpack Staging Facility	Intermittently	~ 60 ft
30A	Central Security Station	Continuously	≥ 75 ft
30B	Cask Receipt Security Station	Continuously	N/A
30C	North Perimeter Security Station	Continuously	≥ 75 ft
33A	Rail Car Buffer Area	Rarely	N/A
33B	Truck Buffer Area	Rarely	N/A
35A	Septic Tank and Leach Field	Never	≥ 75 ft
620	Administration Facility	Continuously	≥ 75 ft
63A	Fire, Rescue, and Medical Facility	Continuously	≥ 75 ft
65A	Administration Security Station	Rarely ^a	≥ 75 ft
65B	Administration Security Station	Rarely ^a	≥ 75 ft
68A	Warehouse/Central Receiving	Continuously	≥ 75 ft
68B	Materials/Yard Storage	Intermittently	≥ 75 ft
690	Vehicle Maintenance and Motor Pool	Continuously	≥ 75 ft
70A	Diesel Fuel Oil Storage	Rarely	≥ 75 ft
70B	Fueling Stations	Continuously	≥ 75 ft
71A	Craft Shops	Continuously	≥ 75 ft
71B	Equipment/Yard Storage	Intermittently	≥ 75 ft

^a These facilities will only be manned in times of high alert.

The cask receipt security station (30B) and the cask buffer areas (33A and 33B) are marked as N/A because they will have transportation casks in close proximity and have been analyzed in the *GROA Shielding Requirements Calculation* (Reference 2.2.30) for dose and shielding. The current location for the emergency diesel generator facility (26D) puts it within approximately 30 ft of a waste path. This facility will not be permanently manned and is only expected to have personnel present during test startups and maintenance procedures. The time and motion data for these tasks should be analyzed once system requirements are determined. It is a reasonable assumption that any scheduled maintenance or testing can be performed at a time when a loaded cask is not present. The aging overpack staging facility is also within 75 ft. of a waste path, approximately 60 ft. This area will not be permanently manned either. The only time

personnel are expected to be present is when an empty AO is being retrieved for use in a nuclear facility. It is a reasonable assumption that this task could be performed during a time when a loaded cask is not present. All other non-nuclear facilities in the GROA are at least 75 ft. from the path of a single loaded cask (References 2.2.8 and 2.2.31) and at least 165 ft. from the truck and rail cask buffer areas (References 2.2.8 and 2.2.31). Therefore, these facilities would receive a dose rate of no greater than 0.25 mrem/hr (Reference 2.2.30, Sections 7.2.1.1, and 7.2.1.2).

Personnel will accompany transportation casks as they move throughout the GROA. Per Assumption 3.2.2.1, two Operators and one HPT will be required to accompany the loaded casks at all time to ensure safe transport. It is assumed that during movement of a transportation cask, escorting personnel are able to remain in a 0.25 mrem/hr or less dose field (Assumption 3.2.7). The distance from the cask receipt security station (30B) to the center of the rail car buffer area is approximately 2250 ft. (Reference 2.2.8) and the distance from the center of the rail car buffer area to CRCF-2 is approximately 4100 ft. (Reference 2.2.8). This results in a total travel distance from the cask receipt security station to CRCF-2 of 6350 ft. (4100 ft. + 2250 ft. = 6350 ft.) which is assumed to be the average distances traveled by a transportation cask inside the GROA (Assumption 3.1.2.2). It is also assumed that the maximum speed a transportation cask can travel inside the GROA is 5 mph (Assumption 3.1.3). Table 7.4 summarizes the personnel dose during transportation cask operations inside the GROA. In this case Operations and HP personnel receive the same dose. Equation 1 was used to determine the dose for each leg of the trip while equation 2 was used to determine the total trip dose. Computation of these doses can be found in Attachment D, *Transportation Inside GROA*, sheet *TS125 Transport*.

Table 7.4 Direct External Dose Data for Transportation Cask Operations

	30B to 33A	33A to CRCF-2	30B to CRCF-2
Travel Distance ft [m]	2250 [685.8]	4100 [1250]	6350 [1936]
Travel Time ^a hrs	0.085	0.155	0.241
Individual Dose per Cask ^b person-mrem/cask	0.021	0.039	0.060
Individual Dose per Year ^c Person-mrem/year	10.63	19.38	30.13

Source: Attachment D, *Transportation Inside GROA*, sheet *TS125 Transport*

^a Maximum travel speed of 5 mph applied over the distance presented (Assumption 3.1.3)

^b Travel Time X 0.25 mrem/hr, Equation 1 (Assumption 3.2.7)

^c Annual Dose Based on Processing 500 Transportation Casks (Assumption 3.1.1.1)

7.1.2.2 TEV Operations

The TEV is escorted by two operations and one HP (Assumption 3.2.2.2). The TEV travels at a speed of no greater than 150 ft./min (1.70 mph) (Reference 2.2.49, Section 3.3.4) for an average distance of 3100 ft. (945 m) based on the distance from CRCF-2 to the north portal (Reference 2.2.8 and Assumption 3.1.2.2). The resulting dose to individual workers and crews was used to determine an annual worker dose based on the

placement of an average of 316 waste packages per year (Assumption 3.1.1.2). In this case Operations and HP personnel receive the same dose.

At 150 ft./min (1.7 mph) it would take the TEV almost 21 minutes to travel from CRCF-2 to the north portal. Each worker would receive a 0.086 mrem dose per operation (See below).

$$\left(\frac{3100 \text{ ft}}{150 \frac{\text{ft}}{\text{min}} \times 60 \frac{\text{min}}{\text{hr}}}\right) \times 0.25 \frac{\text{mrem}}{\text{hr}} = 0.086 \text{ mrem}$$

Therefore, the annual individual dose for TEV operations would be 27.18 mrem/year.

$$0.086 \frac{\text{mrem}}{\text{operation}} \times 316 \frac{\text{operations}}{\text{year}} = 27.18 \text{ mrem / year}$$

7.1.3 Aging Facility

The aging facility operations have been divided into four sections. The first section looks at the transport of AOs to and from the aging pads and the second looks at the placement and retrieval operations on the pad. The third section looks at the inspection and maintenance operations of the aging facility while the fourth considers dose during HAM placement and retrieval operations.

7.1.3.1 Transport of AOs

Radial dose rates from a single AO and dose rate during transport (Assumption 3.2.7) are used along with assumptions for annual throughput, transfer staffing, and AO transporter speed (Assumptions 3.1.1.3, 3.2.3.1, and 3.2.3.2, respectively) to determine worker doses during transport. The same procedure in reverse is assumed for returning AOs to the nuclear facilities for processing with the only difference being the fuel source (Assumption 3.2.4.4). It is assumed that design basis fuel is used for the trip to the pad while average fuel is used for the return trip (Assumption 3.2.4.4). Three fully loaded 4x4 arrays of AOs with design basis fuel are used to evaluate doses to workers as they travel in close proximity to the pads for AO placement. Table 7.5 shows the dose rate as a function of distance from the surface of a single AO for the three fuel types described in Section 6.1.5 of this calculation while Figure 7.2 shows a graphical representation of this data. Table 7.6 summarizes the dose rate as a function of distance from three fully loaded 4x4 arrays containing design basis fuel. Dose rate results and relative error calculations for the data presented in Table 7.5 and Figure 7.2 can be found in Attachment D, *AO Radial Dose Rates.xls*, sheet *Comparison*. Dose rate results and relative error calculations for the data presented in Table 7.6 can be found in Attachment D, *4x4 Dose Rate.xls*, sheet *Design*.

Table 7.5 Radial Dose from a Single AO

	Maximum Fuel	Design Basis Fuel	Average Fuel
Distance from AO (m)	Dose Rate (mrem/hr)	Dose Rate (mrem/hr)	Dose Rate (mrem/hr)
0.25	30.71	10.24	1.87
0.50	26.24	8.77	1.58
0.75	22.82	7.68	1.38
1.00	20.12	6.79	1.20
1.25	17.77	6.04	1.06
1.50	15.95	5.42	0.95
1.75	14.33	4.91	0.85
2.00	12.99	4.46	0.77
2.25	11.80	4.07	0.70
2.50	10.79	3.75	0.64
2.75	9.85	3.45	0.58
3.00	9.06	3.17	0.53
3.25	8.32	2.92	0.48
3.50	7.68	2.71	0.44
3.75	7.07	2.50	0.41
4.00	6.53	2.33	0.38
4.25	6.04	2.16	0.35
4.50	5.64	2.02	0.33
4.75	5.27	1.89	0.30
5.00	4.95	1.77	0.28
10.00	1.71	0.62	0.09
20.00	0.49	0.18	0.03
30.00	0.22	0.09	0.01
40.00	0.14	0.05	0.01
50.00	0.09	0.03	0.01

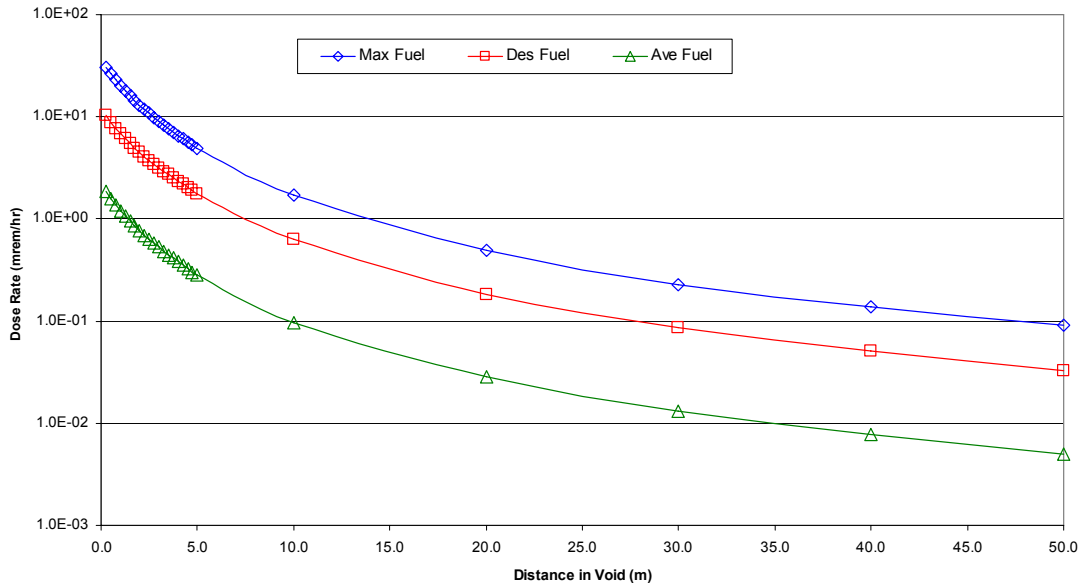


Figure 7.2 Dose Rate as a Function of Distance from a Single AO

Table 7.6 Peak Dose from Three Fully Loaded 4x4 Arrays

Distance from AO (m)	Dose Rate (mrem/hr)
1	11.25
2	9.05
5	5.66
10	3.31

Table 7.7 combines the dose rate data presented in Table 7.6 along with the dose rate assumption for waste transport inside the GROA (Assumption 3.2.7) to determine the dose to workers transporting an AO to the aging pad. The dose received by workers transporting an aged AO from the pad to the nuclear facilities is also represented by the dose reported in Table 7.7. These doses are the same because the total dose is based on the 0.25 mrem/hr limit of Assumption 3.2.7 and fully loaded arrays of design basis fuel in both cases.

Table 7.7 Worker Dose Data for Transport of AOs

	CRCF-2 to Aging Facility Parimeter	Aging Facility Parimeter to Aging Spot	Total Trip
Travel Distance miles	1.54	0.40	1.94
Travel Time hrs	1.23	0.32	1.55
Individual Dose per Cask person-mrem/cask	0.31 ^a	1.14 ^b	1.45 ^c

^a Based on 0.25 mrem/hr maximum dose rate (Assumption 3.2.7) [1.23 hrs X 0.25 mrem/hr = 0.31 mrem]

^b Based on 10 m dose rate from Table 7.6 plus the 0.25 mrem/hr maximum dose rate (Assumption 3.2.7) [(3.31 mrem/hr + 0.25 mrem/hr) X 0.32 hrs = 1.14 mrem]

^c Sum of doses for two legs of trip [0.31 mrem + 1.14 mrem = 1.45 mrem]

7.1.3.2 Placement of AOs

Dose rates were calculated in and around a partially loaded array of TAD AOs (Figure 6.6) and along with Assumptions 3.2.3.1 and 3.2.3.3 were used to determine direct dose on a per task and operation basis. It is assumed that design basis fuel is used for placement operations while average fuel is used for retrieval operations (Assumption 3.2.4.4) and that the same procedure used for placement in reverse is used for retrieval (Assumption 3.2.3.4). Table 7.8 gives a summary of the dose as a function distance from overpack “A” depicted in Figure 6.6. Placement dose rates are the result of a curve-fit of the data produced by the partially loaded array pictured in Figure 6.6. The dose rate values used to generate the curve were calculated at points 3, 6.7, and 10.5 meters from the surface of the AO and averaged from four dose areas, one in each quadrant around the AO. Retrieval dose rates presented are scaled down from those calculated for the placement operations using a 0.18 scaling factor determined by comparing the radial dose rate around single AOs containing design basis and average fuel (see Attachment D, *Radial Dose Rates.xls*, sheet *Comparison*). The data and curves used to generate these dose rates from the MCNP5 output can be found in Attachment D, *Odd Dose Rates.xls*, sheet *Dose for AP Opps*. Table 7.9 gives a summary of these worker doses for placement operations while Table 7.10 summarizes the dose for retrieval operations.

Table 7.8 Dose Rate Data Around a Partially Loaded 4x4 Array

Distance from AO (m)	Placement Dose Rates (mrem/hr)	Retrieval Dose Rates (mrem/hr)
0.25	15.85	2.85
1	15.55	2.80
2	15.16	2.73
3	14.78	2.66
5	14.05	2.53
7	13.35	2.40
10	12.36	2.23

Table 7.9 Dose Data for Placement of AOs

Task	Time Required (hr)	AO Transporter Operator			Spotting/Installation Operator			HPT		
		Distance from AO (m)	Dose Rate at Distance (mrem/hr)	Dose at Distance/Task/Person (mrem/task/operator)	Distance from AO (m)	Dose Rate at Distance (mrem/hr)	Dose at Distance/Task/Person (mrem/task/operator)	Distance from AO (m)	Dose Rate at Distance (mrem/hr)	Dose at Distance/Task/Person (mrem/task/technician)
Prepare to Transfer:										
Retrieve AO transporter - Engage AO within facility	0.33	2	4.46*	1.47	5	1.77*	0.58	5	1.77*	0.58
Transfer to aging pad spot: From facility to aging location (Table 7.7).				1.45			1.45			1.45
Place AO:										
Retract AO transporter	0.08	2	15.16**	1.21	5	14.05**	1.12	5	14.05**	1.12
Connect Monitoring Equipment:										
Position mobile equipment	0.25	N/A	--	--	1	15.55**	3.89	5	14.05**	3.51
Connect monitoring equipment	1.00	N/A	--	--	1	15.55**	15.55	5	14.05**	14.05
Individual Dose per AO Placed (mrem/person)				4.13			22.59			20.71

Source: Attachment D, Odd Dose Rates.xls, sheet Dose for AP Opps

- *Dose rate based on a single AO with Design Basis Fuel (Table 7.5)
- **Dose rate based on a partially loaded array with Design Basis Fuel (Table 7.8)
- ***Dose rate based on fully loaded 4x4 array with Design Basis Fuel (Table 7.6)
- [A] Assumed time data per task (Assumption 3.2.3.3)
- [B] Assumed task distances for AO Transporter Operators (Assumption 3.2.3.3)
- [C] Dose rate at distance data from Tables 7.6, 7.8 and Assumption 3.2.7
- [D] Column C X (Column A / 60), Equation 1
- [E] Assumed task distances for Spotting/Installation Operators (Assumption 3.2.3.3)
- [F] Dose rate at distance data from Tables 7.6, 7.8 and Assumption 3.2.7
- [G] Column F X (Column A / 60), Equation 1
- [H] Assumed task distances for HPTs (Assumption 3.2.3.3)
- [I] Dose rate at distance data from Tables 7.6, 7.8 and Assumption 3.2.7
- [J] Column I X (Column A / 60), Equation 1
- a Sum of doses in Column D, Equation 2
- b Sum of doses in Column G, Equation 2
- c Sum of doses in Column J, Equation 2

Table 7.10 Dose Data for Retrieval of AOs

Task	[A] Time Required (hr)	[B] AO Transporter Operator			[E] Spotting/Installation Operator			[H] HPT		
		[C] Distance from AO (m)	[C] Dose Rate at Distance (mrem/hr)	[D] Dose at Distance/Task/Person (mrem/task/operator)	[E] Distance from AO (m)	[F] Dose Rate at Distance (mrem/hr)	[G] Dose at Distance/Task/Person (mrem/task/operator)	[H] Distance from AO (m)	[I] Dose Rate at Distance (mrem/hr)	[J] Dose at Distance/Task/Person (mrem/task/technician)
Disconnecting Monitoring Equipment:										
Position mobile equipment	0.25	N/A	--	--	1	2.80**	0.70	5	2.53**	0.63
Disconnect monitoring equipment	1.00	N/A	--	--	1	2.80**	2.80	5	2.53**	2.53
Retrieve AO:										
Engage AO transporter	0.08	2	2.73**	0.22	5	2.53**	0.20	5	2.53**	0.20
Transfer to facility: From aging location to facility (Table 7.7)				1.45			1.45			1.45
Prepare to Transfer:										
Disengage AO within facility - Return AO transporter	0.33	2	0.77*	0.25	5	0.28*	0.09	5	0.28*	0.09
Individual Dose per AO Placed (mrem/person)				1.92			5.24			4.90

Source: Attachment D, Odd Dose Rates.xls, sheet Dose for AP Opps
 *Dose rate based on a single AO with Average Fuel (Table 7.5)
 **Dose rate based on a partially loaded array with Average Fuel (Table 7.8)
 ***Dose rate based on fully loaded 4x4 array with Design Basis Fuel (Table 7.6)
 [A] Assumed time data per task (Assumption 3.2.3.3)
 [B] Assumed task distances for AO Transporter Operators (Assumption 3.2.3.3)
 [C] Dose rate at distance data from Tables 7.6, 7.8 and Assumption 3.2.7
 [D] Column C X (Column A / 60), Equation 1
 [E] Assumed task distances for Spotting/Installation Operators (Assumption 3.2.3.3)
 [F] Dose rate at distance data from Tables 7.6, 7.8 and Assumption 3.2.7
 [G] Column F X (Column A / 60), Equation 1
 [H] Assumed task distances for HPTs (Assumption 3.2.3.3)
 [I] Dose rate at distance data from Tables 7.6, 7.8 and Assumption 3.2.7
 [J] Column I X (Column A / 60), Equation 1
 a Sum of doses in Column D, Equation 2
 b Sum of doses in Column G, Equation 2
 c Sum of doses in Column J, Equation 2

7.1.3.3 Inspection and Maintenance Operations

The data produced by the fully loaded 4x4 arrays diagramed in Figure 6.5 along with Assumption 3.2.3.6 were used to evaluate worker doses during inspection operations. Based on the data presented in Table 7.6, the dose rate received by the inspector at 2 meters from the AOs would be 9.05 mrem/hr. The aging facility can hold a maximum of 2400 vertical AOs and 100 HAMs (Reference 2.2.20 Note 3 and Reference 2.2.21, Notes 3 and 4). At 30 seconds per AO or HAM for inspection it would take one inspector almost 21 hours to inspect each aging location [$2500 \text{ locations} \times 0.5 \text{ min/location} / 60 \text{ min/hr} = 20.8 \text{ hours}$]. Therefore the inspector would receive 188.5 mrem per quarterly inspection. The individual annual dose for this task would be 754 mrem/year, as shown below.

$$188.5 \frac{\text{mrem}}{\text{quarter}} \times 4 \frac{\text{quarters}}{\text{year}} = 754 \text{ mrem / year} / 24 \text{ operators} = 31.42 \text{ mrem/yr}$$

The dose point used to represent maintenance operations is pictured in the partially loaded arrays diagramed in Figure 6.6 labeled as “Maintenance Detector”. The dose rate at this point is 25.35 mrem/hr. Based on this dose rate, and the 1-hour per operation described in Assumption 3.2.3.5, the individual dose per maintenance operation would be 25.35 person-mrem.

$$25.35 \frac{\text{mrem}}{\text{hr}} \times 1 \text{ hour} = 25.35 \text{ mrem}$$

Per Assumption 3.2.3.5, two operations personnel are required to perform maintenance operations and approximately 20 maintenance operations are expected per year. Based on these assumption, the collective annual dose for the 20 maintenance operations would be 1014 person-mrem/year, as shown below.

$$25.35 \frac{\text{mrem}}{\text{operation}} \times 20 \frac{\text{operations}}{\text{year}} \times 2 \text{ Operators} = 1014 \text{ person - mrem / year} / 24 \text{ operators} \\ = 42.25 \text{ mrem/yr}$$

7.1.3.4 Horizontal Aging Modules

Table 7.11 lists the expected dose to personnel during HAM operations on a per cask basis. The data is directly from the *NUHOMS HD System Safety Analysis Report* (Reference 2.2.17, Table 10-1) and considered the best dose information on HAM operations at this time. The HAM dose is not considered to be a major contributor to the overall TEDE for this calculation due to the small amount that will be processed.

Table 7.11 Dose Data for HAM Operations

Task Description	No. of workers	Duration (hr)	Total exposure (person-mrem)	Fraction of the Total Time	Fraction of the Total Dose
Prepare the HSM-H and hydraulic ram	2	2	0	0.282	0.000
Transport the Cask to ISFSI	6	1	0	0.141	0.000
Position the Cask in Close proximity with the HSM-H	3	1	0	0.141	0.000
Remove the Cask Lid	2	1	68	0.141	0.170
Align and Dock the Cask with the HSM-H	2	0.25	87	0.035	0.218
Position and Align Ram with Cask	2	0.5	173	0.071	0.434
Remove Ram Access Cover Plate	1	0.25	21	0.035	0.053
Transfer the DSC from the Cask to the HSM-H	3	0.5	0	0.071	0.000
Lift the Ram Back onto the Trailer and Un-Dock the Cask from the HSM-H	2	0.083	29	0.012	0.073
Install HSM-H Access Door	2	0.5	21	0.071	0.053
Totals	N/A	7.08	399	1	1

Source: Reference 2.2.17, Table 10-1

7.1.3.5 Summary of External Dose for Aging Pad Operations

The data presented in Section 7.1.3.1 and 7.1.3.2 represent the dose to aging pad workers on a per task basis. If 6 crews are required for aging facility operations (Assumption 3.2.3.1) and the maximum and nominal aging facility throughputs are 275 and 135 AOs, respectively, each crew will process 46 and 23 AOs, respectively. The following tables combine the individual dose results for transport, placement, retrieval, support only time, inspection, and maintenance operations to determine an annual individual external dose. In all of the summary tables below the AO transporter operator and the spotting/installation operator doses were added and represent one operator. Table 7.12 shows the annual external individual dose for both placement and retrieval operations in one work shift. Therefore, the dose from “Return from Aging Pad” and “Unloaded Trip to Aging Pad” presented in tables 7.9 and 7.10 are not included because the transporter would go from placement directly to retrieval without returning to the facility.

Table 7.12 Individual External Dose Summary for Aging Facility Operations

	Operator	HPT	Source
Placement of AO (mrem/cask)	13.36 ^a	20.71	Table 7.9
Retrieval of AO (mrem/cask)	3.58 ^a	4.90	Table 7.10
Total Individual Dose per Cask (mrem/cask)	16.94	25.61	N/A
Individual Dose per Year for 46 AOs Processed (mrem/year)	779	1179	N/A
Individual Dose per Year for 23 AOs Processed (mrem/year)	390	589	N/A

Source: Attachment D, GROA *TEDE Summary.xls*, Aging

^a The values presented are the combined average of AO Transporter and Spotting/Installation Operators doses.

7.1.4 Dose to Workers in Low Radiation Support Areas

Assumption 3.2.6 assumes that any time operations or HP personnel are not directly participating in waste transfer operation, they remain in a support capacity in a dose field of 0.05 mrem/hr. Equation 3 was used to determine this support only time and presented in Table 6.19 of Section 6.2.5. Table 7.13 summarizes this support only time for the year and applies the 0.05 mrem/hr dose rate to determine the total support only dose per individual and per crew for each operation. The crew dose is based on the assumption that all personnel (Operators and HPTs) spend an equal amount of time in support areas.

Table 7.13 Annual Dose Summary for Support Only Time

Operation	Support Only time (hrs/year)	Individual Dose (mrem/year)
Cask Receipt	1763 ^a	88.2
Transportation Cask Operations	1880 ^b	94.0
TEV Operations	1889 ^c	94.5
AO Operations (275)	1572 ^d	78.6
AO Operations (135)	1786 ^e	89.3

^a Cask receipt task time based on 100 casks processed per year

^b Transportation cask operation task time based on 500 casks processed per year

^c TEV operation task time based on 316 WPs processed

^d AO placement operations based on 46 AOs placed per year per crew

^e AO placement operations based on 23 AOs placed per year per crew

7.2 ALARA DESIGN AND WORK PROCESS REVIEWS

Two ALARA reviews were performed to analyze the operations in the GROA. The reviews focused on the operations at the cask receipt security station (30B) and operations at the aging pads (17P and 17R) as these are the two areas of the GROA in which worker dose is expected to be the highest. The first of the reviews (AR-07-014, Reference 2.2.33) was focused on cask receipt at 30B with personnel barrier removal, movement of casks to buffer areas, movement of casks from buffer areas to nuclear facilities, and moving and placing AOs on the aging pads. This review produced action items concerned with personnel barriers, specifically; at what point the barriers will be

removed, how they will be removed, and personnel required to remove the barriers. A resolution for this action item was distributed via memorandum (Reference 2.2.35). The resolution stated that the barriers would be removed before the cask entered 30B but exact methods and personnel requirements could not be determined at the time. As a result, assumptions had to be made as input for this calculation (Section 3.1.1). The second ALARA review (AR-07-015, Reference 2.2.34) was focused more on the operations at the aging pads. Specifically, security escorts for AOs, remote control operation for transporters, personnel requirements for AO movement, procedure for placing AO on the pads, HAM operations, and maintenance operations were discussed. This meeting produced one action item concerning the placement procedures once the AO had reached the pad. The use of an air pallet for final positioning vs. direct placement by the transporter was discussed. A resolution for this action item was distributed via memorandum (Reference 2.2.35). The resolution concluded that the AO transporter was able to move across the pad and place the AO in the desired location therefore the use of an air pallet for movement of AOs would not be needed. ALARA Design Review AR-07-014 can be found as Reference 2.2.33 and ALARA Design Review AR-07-015 can be found as Reference 2.2.34. A memo concerning the resolution of action items produced by these ALARA Design Reviews can be found as Reference 2.2.35.

7.3 TEDE RESULTS

Inhalation and air submersion doses are included in the TEDE determination. The methodology and calculation of inhalation and air submersion doses are included in Attachment B.

7.3.1 Cask Receipt

Table 7.2 presents the individual dose to security, operations, and HP personnel. Table 7.14 summarizes the annual collective dose to security, operations, and HP personnel during cask acceptance operations assuming each individual and crew processes 100 transportation casks annually. This TEDE includes the contribution from direct external radiation during tasks and support operations and inhalation and submersion doses. The TEDE calculation can be found in Attachment D, *GROA TEDE Summary.xls*, sheet *Cask Receipt*.

Table 7.14 Annual Individual and Collective Dose for Cask Receipt Operations

	Security	Operator	HPT
Individual Dose per Cask ^b (mrem/cask)	5.67	12.67	12.41
Individual Dose per Year ^c (mrem/year)	567	1267	1241
Support Time Dose per Year ^d (mrem/year)	88.2	88.2	88.2
Inhalation and Submersion ^e (mrem/year)	0.42	0.42	0.42
Individual TEDE per Year ^f (mrem/year)	656	1356	1330
Collective Crew Dose per Year ^a (person-mrem/year)	7410		
Collective Facility Dose per Year ^g (person-mrem/year)	37050		

Source: Attachment D, *GROA TEDE Summary.xls*, sheet *Cask Receipt*

^a Crew dose based on 1 security, 4 operations, and 1 HP personnel

^b From Table 7.2

^c Individual dose per cask times 100 casks annually

^d From Table 7.13

^e From Table B.2

^f Sum of column

^g Collective crew dose per year times 5 crews

7.3.2 GROA

The results presented in Section 7.1.2.1 and Table 7.3 indicate that there is no significant dose rate to non-nuclear facilities during the transport of waste throughout the GROA. All non-nuclear facilities regardless of occupational requirements are at least 75 ft. from any path of waste transport except for the emergency diesel generator facility and the aging overpack staging facility (Reference 2.2.8). These two facilities however are not expected to have personnel present continuously and any operation at these facilities can be scheduled during a time when no waste is present.

The other dose consideration inside the GROA is the dose to escorting personnel during waste transport inside the GROA, both transportation casks and TEV movements. Based on the results of Section 7.1.2.1 and Table 7.4 the dose for the trip from the cask receipt security station (30B) to CRCF-2 resulted in an external dose of 0.06 mrem per cask. Table 7.15 summarizes the annual collective dose to both operations and HP personnel for transportation cask movements assuming one crew processes 500 transportation casks annually. According to the dose results presented in Section 7.1.2.2, the dose to each worker during TEV movement operations is 0.086 mrem per operation. Table 7.16 summarizes the annual collective dose to both operations and HP personnel for TEV operations assuming each crew processes 316 WPs annually. The TEDE results presented in both tables include the contribution from direct external radiation during tasks and support operations and inhalation and submersion doses. The TEDE calculation can be found in Attachment D, *GROA TEDE Summary.xls*, sheet *GROA*.

Table 7.15 Annual Individual and Collective Dose for Transportation Cask Operations

	Operator Individual Dose (mrem/year)	HPT Individual Dose (mrem/year)	Crew Collective Dose (person-mrem/year) ^a
Annual Transport Operations Dose ^b	30.13	30.13	90.39
Annual Support Operations Dose ^c	94	94	282
Annual Inhalation and Submersions Dose ^d	0.42	0.42	1.26
Annual TEDE^e	125	125	375

Source: Attachment D, *GROA TEDE Summary.xls*, sheet *GROA*

^a Crew dose based on 2 Operators and 1 HPT per crew.

^b From Table 7.4

^c From Table 7.13

^d From Table B.2

^e Sum of Column

Table 7.16 Annual Individual and Collective Dose for TEV Operations

	Operator Individual Dose (mrem/year)	HPT Individual Dose (mrem/year)	Crew Collective Dose (person-mrem/year) ^a
Annual Transport Operations Dose ^b	27.18	27.18	81.54
Annual Support Operations Dose ^c	94.5	94.5	283.5
Annual Inhalation and Submersions Dose ^d	0.42	0.42	1.26
Annual TEDE^e	122	122	366

Source: Attachment D, *GROA TEDE Summary.xls*, sheet *GROA*

^a Crew dose based on 2 Operators and 1 HPT per crew.

^b From Section 7.1.2.2

^c From Table 7.13

^d From Table B.2

^e Sum of Column

7.3.3 Aging Facility

Aging facility operations is made up of several components which were summarized in the tables of Section 7.1.3.5. The TEDE presented for aging facility operations is based on transporting a loaded AO to the pad for aging and returning with a loaded AO. Tables 7.17 and 7.18 summarize the annual collective dose to both operations and HP personnel for the placement and retrieval of 275 and 135 AOs, respectively, without the use of seismic restraints. If six crews are used for aging facility operations, each crew will process 46 AOs in the maximum throughput case [$275 \text{ AOs} / 6 \text{ crews} = 46 \text{ AOs}_{/\text{crew}}$] and 23 AOs in the nominal case [$135 \text{ AOs} / 6 \text{ crews} = 23 \text{ AOs}_{/\text{crew}}$]. The TEDE calculation can be found in Attachment D, *GROA TEDE Summary.xls*, Sheet *Aging*.

Table 7.17 Annual Individual and Collective Dose for Aging Facility Operations with Maximum Throughput
(275 AOs Placed and Retrieved by 6 crews)

	Operator Dose (mrem)	HPT Dose (mrem)
Annual Aging Operations Dose ^b	779	1179
Annual Support Operations Dose ^c	78.6	78.6
Annual Inhalation and Submersions Dose ^d	0.42	0.42
Annual Inspection Operations ^e	31.42	N/A
Annual Maintenance Operations ^f	42.25	N/A
Annual Individual TEDE^g	932	1258
Annual Crew TEDE ^a	4986	
Collective Facility Dose (mrem)^h	29916	

Source: Attachment D, GROA *TEDE Summary.xls*, Aging

^a Crew dose based on 4 Operators and 1 HPT per crew

^b From Table 7.12

^c From Table 7.13

^d From Table B.2

^e From Section 7.1.3.3

^f From Section 7.1.3.3

^g Sum of Column

^h Annual Crew TEDE times 6 Crews

Table 7.18 Annual Individual and Collective Dose for Aging Facility Operations with Nominal Throughput
(135 AOs Placed and Retrieved by 6 crews)

	Operator Dose (mrem)	HPT Dose (mrem)
Annual Aging Operations Dose ^b	390	589
Annual Support Operations Dose ^c	89.3	89.3
Annual Inhalation and Submersions Dose ^d	0.42	0.42
Annual Inspection Operations ^e	31.42	N/A
Annual Maintenance Operations ^f	42.25	N/A
Annual Individual TEDE^g	553	679
Annual Crew TEDE ^a	2891	
Total Collective Facility Dose (mrem)^h	17346	

Source: Attachment D, GROA *TEDE Summary.xls*, Aging

^a Crew dose based on 4 Operators and 1 HPT per crew

^b From Table 7.12

^c From Table 7.13

^d From Table B.2

^e From Section 7.1.3.3

^f From Section 7.1.3.3

^g Sum of Column

^h Annual Crew TEDE times 6 Crews

7.4 CONCLUSIONS

7.4.1 Cask Receipt

The individual TEDE received by security personnel performing cask receipt at 30B is **0.656 rem/year** while the individual TEDE received by operations personnel is **1.36 rem/year**. The individual TEDE to HP personnel for these operations is **1.33 rem/year**. These values are reported in Table 7.14. These doses are based on five crews comprised of one security officer, four operators, and one HPT processing a total of 500 casks annually or 100 casks annual for each crew. All of these workers remain under the regulatory limit of 5 rem/year. The collective facility dose for cask receipt operations at 30B is **37.1 person-rem/year**.

7.4.2 GROA

Worker dose in the GROA is primarily from two different activities; escorting movements of transportation casks and TEVs. The TEDE for transportation cask operations is the same for both operations and HP personnel (Table 7.15). The resulting individual TEDE is **0.125 rem/year** for both. This results in a collective dose of **0.375 rem/year** for all transportation cask operations in one year based on a crew of two operators and one HPT. The individual TEDE for TEV operations is again the same for both operations and HP personnel (Table 7.16). The resulting TEDE is **0.122 rem/year** for both. This results in a collective dose of **0.366 person-rem/year** for all TEV operations in one year based on a crew of two operators and one HPT. The individual TEDE for all personnel performing both transportation cask and TEV movement are below the 5 rem/year regulatory requirement.

7.4.3 Aging Facility

The aging facility was considered with two throughput values; one is considered a maximum case, 275 AOs per year, while the other is considered a nominal case, 135 AOs per year. For the maximum case, the TEDE for the operations and HP personnel were **0.932 rem/year** and **1.26 rem/year**, respectively (Table 7.17). In the nominal case, the TEDE for the operations and HP personnel were **0.553 rem/year** and **0.679 rem/year**, respectively (Table 7.18). All of these TEDE fall under the 5 rem/year regulatory limit and are therefore acceptable. The collective dose for the facility in the maximum and nominal throughput cases are approximately **29.9 person-rem/year** and **17.3 person-rem/year**, respectively. This collective dose is based on six crews performing operations in the aging facility.

7.4.4 Compliance

These doses are in compliance with the requirements of the limit of 5 rem/yr for occupational workers (Reference 2.3.2, Article 1201(a)(1)(i)). These estimated annual doses to cask receipt, GROA, and aging facility Operators and HPTs do not individually support the ALARA design goal of minimizing the number of workers exposed to more than 0.5 rem/yr (Reference 2.1.5).

7.4.4.1 Regulations

The regulation applicable to worker doses is provided in Reference 2.3.2, Article 1201 (as specified in Reference 2.2.27, Section 4.10.1):

“The licensee shall control the occupational dose to individual adults to the following dose limits:

- (1) An annual limit, which is the more limiting of:
 - (i) The total effective dose equivalent being equal to 5 rems; or
 - (ii) 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 being equal to 50 rems.
- (2) The annual limits to the lens of the eye, to the skin, and to the extremities, which are: A lens dose equivalent of 15 rems, and a shallow-dose equivalent of 50 rems to the skin or to any extremity.”

7.4.4.2 ALARA Design goals

ALARA design goals (Reference 2.1.5, Section 4.10.3.3.1) for occupational workers ensure that individual annual doses are maintained at ALARA levels during normal operations or during potential Category 1 event sequence. The following ALARA design goal is established for the design process.

- (1) Individual Dose:

“The ALARA design goal for individual radiation worker doses is to minimize the number of individuals that have the potential of receiving more than 500 mrem/yr TEDE. That goal is 10 percent of the annual TEDE limit in 10 CFR 20.1201 (Reference 2.3.2), and includes both internal and external exposures.”

7.4.4.3 Discussion

This worker dose estimate is based on best available estimates or projections of annual cask receipt, GROA, and aging facility processing. Dose rates are based on representative cask configurations and estimated time and motion data.

One of the secondary uses of the dose assessment is to identify and prioritize cask receipt, GROA, and aging facility design areas that should consider additional ALARA design features to reduce individual worker doses to achieve the ALARA goals. In this respect, attention should be focused on means by which doses to workers involved with cask receipt, GROA, and aging facility operations can be reduced.

7.5 RECOMMENDATIONS

- It is recommended that cask receipt operations be re-evaluated when more detailed throughput and processing task data is available.
- It is recommended that movement operations of both transportation casks as TEVs be re-evaluated when more detailed throughput and process task data is determined.
- It is recommended that aging facility operation be re-evaluated when detailed aging facility throughput and process task data is determined.
- It is recommended that seismic restraints, as assumed by this calculation, not be used due to the dose received by personnel as demonstrated in Attachment C.

ATTACHMENT A

List of Files on Attached CD (Attachment D)

All MCNP5 input and output files documented in this calculation were stored on a compact disc (CD) as Attachment D. Also, all Microsoft® Excel spreadsheets used to calculate input values or to display the results in graphical or tabular form are included on the CD.

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ATTACHMENT B

Inhalation and Air Submersion Doses

To evaluate individual TEDE for aging facility and site personnel, inhalation and submersion dose must be considered. Airborne concentrations, and inhalation and submersion doses, are determined based on the methodology in Section 4.3.2. The airborne concentration of radioactive contamination from a release of surface contamination is based on the allowable surface contamination for an AO before it leaves the nuclear facility in order to reduce exposure to the environment (Assumption 3.2.5.2). The allowable surface contamination for beta-gamma emitters is 1,000 dpm/100 cm² or 4.5x10⁻⁶ μCi/cm² represented as ⁶⁰Co (Assumption 3.2.5.2). The allowable surface contamination for alpha emitters is 20 dpm/100 cm² or 9.0x10⁻⁸ μCi/cm² represented as ²⁴¹Am (Assumption 3.2.5.2). To determine the airborne concentration, a resuspension factor of 2x10⁻⁶ m⁻¹ was chosen (Assumption 3.1.5). Equation 5 was then applied for both beta-gamma and alpha emitters as shown below:

$$C_{\beta\gamma} = 4.5 \times 10^{-6} \frac{\mu\text{Ci}}{\text{cm}^2} \times 2 \times 10^{-6} \text{m}^{-1} \times 10,000 \frac{\text{cm}^2}{\text{m}^2} = 9.0 \times 10^{-8} \frac{\mu\text{Ci}}{\text{m}^3}$$

and

$$C_{\alpha} = 9.0 \times 10^{-8} \frac{\mu\text{Ci}}{\text{cm}^2} \times 2 \times 10^{-6} \text{m}^{-1} \times 10,000 \frac{\text{cm}^2}{\text{m}^2} = 1.8 \times 10^{-9} \frac{\mu\text{Ci}}{\text{m}^3}$$

The annual inhalation and submersion dose to a worker resulting from resuspension of contamination over the entire 2000 hour work year is determined by Equation 6 below:

$$H^{inh} = C \times DCF^{inh} \times 3.7 \times 10^9 \times RF \times BR \times t$$

and

$$H^{sub} = C \times DCF^{sub} \times 3.7 \times 10^9 \times t$$

In order to determine the inhalation dose a respirability factor and breathing rate must be determined. It is assumed that all of the released or suspended particles are respirable resulting in a respirability factor of 1 (Assumption 3.2.5.3). This is acceptable because it is conservative to assume that all of the available particles are respirable. The worker breathing rate is 3.3x10⁻⁴ m³/s taken directly from 10 CFR Part 20, Appendix B (Reference 2.3.2).

Dose conversion factors (DCFs) are used to convert the airborne contamination into dose in both the inhalation and submersion cases. The purpose of the Federal Guidance Reports (FGR) and International Commission on Radiological Protection 68 (ICRP) in part is to set forth derived guides that are consistent with current Federal radiation protection guidance. They are intended to serve as the basis for regulations setting upper bounds on the inhalation and ingestion of, and submersion in, radioactive materials in the workplace. They also include tables of exposure-to-dose conversion factors, for general

use in assessing average individual committed doses in any population that is adequately characterized by Reference Man.

FGR 11 (Limiting Values of Radionuclide Intake And Air Concentration and Dose Conversion Factors For Inhalation, Submersion, And Ingestion)(Reference 2.2.16) and ICRP 68 (Dose Coefficients for Intakes of Radionuclides by Workers)(Reference 2.2.44) contain the inhalation dose conversion factors (DCF^{inh}). FGR 12 (Eternal Exposure to Radionuclides in Air, Water, and Soil)(Reference 2.2.45) and FGR 13 (Cancer Risk Coefficients for Environmental Exposure to Radionuclides)(Reference 2.2.39) contain the submersion dose conversion factors (DCF^{sub}). Table B.1 contains the DCFs used in Equation 6 to determine dose.

Table B.1 Dose Conversion Factors for Co-60 and Am-241

Isotope [A]	FGR 11	FGR 12	ICRP 68	FGR 13
	DCF^{inh}	DCF^{sub}	DCF^{inh}	DCF^{sub}
(Sv/Bq) [B]	(Sv/Bq-s-m ³) [C]	(Sv/Bq) [D]	(Sv/Bq-s-m ³) [E]	
⁶⁰ Co	5.91E-08	1.26E-13	2.90E-08	1.19E-13
²⁴¹ Am	1.20E-04	8.18E-16	3.90E-05	6.77E-16

Sources: [A] Isotopes (Assumption 3.2.5.2)
[B] FGR11 (Reference 2.2.16, Table 2.1)
[C] FGR12 (Reference 2.2.45, Table III.1)
[D] ICRP68 (Reference 2.2.44)
[E] FGR13 (Reference 2.2.39, Table B.1)

Table B.2 summarizes the inhalation and submersion dose to workers by applying Equation 6 along with the DCFs from FGR-11 and FGR-12 and results of Equation 5 reported above. Table B.3 summarizes the inhalation and submersion dose to workers also by applying Equation 6 but using the DCFs from ICRP-68 and FGR-13.

Table B.2 Inhalation and Submersion Doses using FGR-11 and FGR-12

	Airborne Concentration ($\mu\text{Ci}/\text{m}^3$) [A]	Inhalation DCFs (Sv/Bq) [B]	Inhalation Dose (mrem/year) [C]	Submersion DCFs ($\text{Sv}/\text{Bq}\cdot\text{s}\cdot\text{m}^3$) [D]	Submersion Dose (mrem/year) [E]	Total Dose (mrem/year) [F]
Beta-Gamma						
Co-60	9.00E-08	5.91E-08	1.00E-02	1.26E-13	6.46E-05	1.01E-02
Alpha						
Am-241	1.80E-09	1.20E-04	4.06E-01	8.18E-16	8.39E-09	4.06E-01
					TOTAL	0.42

- [A] Product of Equation 5 above
- [B] Dose Conversion Factors from FGR-11 (Table B.1, Column B)
- [C] Product of Equation 6 as described above for inhalation dose
- [D] Dose Conversion Factors from FGR-12 (Table B.1, Column C)
- [E] Product of Equation 6 as described above for submersion dose
- [F] Sum of columns C and E

Table B.3 Inhalation and Submersion Doses using ICRP-68 and FGR-13

	Airborne Concentration ($\mu\text{Ci}/\text{m}^3$) [A]	Inhalation DCFs (Sv/Bq) [B]	Inhalation Dose (mrem/year) [C]	Submersion DCFs ($\text{Sv}/\text{Bq}\cdot\text{s}\cdot\text{m}^3$) [D]	Submersion Dose (mrem/year) [E]	Total Dose (mrem/year) [F]
Beta-Gamma						
Co-60	2.00E-06	2.90E-08	4.91E-03	1.19E-13	6.11E-05	4.97E-03
Alpha						
Am-241	2.00E-07	3.90E-05	1.32E-01	6.77E-16	6.95E-09	1.32E-01
					TOTAL	0.14

- [A] Product of Equation 5 above
- [B] Dose Conversion Factors from ICRP-68 (Table B.1, Column B)
- [C] Product of Equation 6 as described above for inhalation dose
- [D] Dose Conversion Factors from FGR-13 (Table B.1, Column C)
- [E] Product of Equation 6 as described above for submersion dose
- [F] Sum of columns C and E

Equation 7 is used to determine the total annual dose to workers from inhalation and submersion by summing their contributions. When applying the DCFs from FGR-11 and FGR-12, the resulting dose is 0.42 mrem/year. When applying the DCFs from ICRP-68 and FGR-13 however, the resulting dose is only 0.14 mrem/year. Therefore, the results of FGR-11 and FGR-12 will be included in the TEDE calculation of Section 7 of this calculation.

ATTACHMENT C

Seismic Restraint Analysis

Though the current plan does not call for the use of seismic restraints, a dose analysis was performed in order to evaluate the impact of this activity on overall worker doses. Assumption 3.2.3.7 gives a total task time of 5 hours to install the 20 bolts and keepers assuming an installation rate of 15 min/bolt. An average distance of 0.25 meters from an AO in a partially loaded pad is given for operations personnel and a distance of 5 meters for HP personnel. During AO placement operations, design basis fuel is used for modeling while for retrieval; average fuel is used for modeling keeping the task times and distances the same. Table C.1 shows the dose to Operations personnel during seismic restraint operations while Table C.2 shows the dose to HP personnel for the same operations.

Table C.1 Dose to Operations Personnel for Seismic Restraint Installation and Removal

	Dose Rate (mrem/hr)	Dose (person-mrem)
AO Placement	15.85	79.29
AO Retrieval	2.85	14.25
Combined	18.7	93.54

Table C.2 Dose to HP Personnel for Seismic Restraint Installation and Removal

	Dose Rate (mrem/hr)	Dose (person-mrem)
AO Placement	14.05	70.25
AO Retrieval	2.53	12.65
Combined	16.58	82.90

Based on the results of Tables C.1 and C.2 above and the maximum throughput of 46 AOs/crew [275 AOs / 6 crews = 46 AOs/crew], the annual individual dose for operations and HP personnel would be 4.3 rem and 3.8 rem, respectively. This is significantly greater than the annual TEDE for aging pad operations reported in Table 7.17 of this calculation. Combining this activity to the operations would increase the annual TEDE over the 5 rem/year regulatory limit. For this reason, the use of seismic restraints is not recommended.