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Design Analysis Cover Sheet

Complete only applicable items.

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8. Checker	Mark Fortsch		1/13/00
9. Lead Design Engineer	Mark Fortsch		1/13/00
10. Department Manager	Steve Meyers		1/13/00
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ACRONYMS AND ABBREVIATIONS

AFR	active fuel region
ALARA	as low as reasonably achievable
BWR	boiling water reactor
CCPC	Canister/Cask Preparation Cell
CDA	Controlled Design Assumptions document
CDB	Characteristics Data Base
CFR	Code of Federal Regulations
CHA	Canister Handling Area
Ci	curie
cm	centimeters
CSC	canister staging cell
CSS	canister staging station
d	disintegrations
DBF	design basis fuel
DC	disposal container
DCHC	disposal container handling cell
DCLC	disposal container loading cell
DCSC	disposal container staging cell
DCTA	disposal container transporter airlock
DDR	design dose rate
DDRL	design dose rate limit
DE	design earthquakes
DHLW	defense high-level waste
DHLWCSS	defense high-level waste canister stage station
DI	Document Identifier
DOE	(U.S.) Department of Energy
DR	dose rate
EDE	effective dose equivalent
GWd	gigawatt day
g	grams
LAG	lower air gap
LDCSA	loaded disposal container staging area
LFGP	lower fission gas plenum
LTP	lower tie plate
LWR	light water reactor
MeV	million-electron-volts
MGDS	Mined Geologic Disposal System
MGR	Monitored Geologic Repository
MPC	multi-purpose canister
mrem	milli-roentgen-equivalent-man
MTIHM	metric tons initial heavy metal
MWd	megawatt day
MTRWS	multi-purpose remote work station
NBS	National Bureau of Standards

NRC	Nuclear Regulatory Commission
OG	operating gallery
PWR	pressurized water reactor
Q	Quality Affecting/Subject to QARD Requirements
QA	Quality Assurance
QAD	QAD-vCGGP-A (latest version of QAD point-kernel codes)
QAP	Quality Administrative Procedure
QARD	Quality Assurance Requirements and Description
Ref	reference
s	second
SFDH	spent fuel disassembly hardware
SFA	spent fuel assembly
SNF	spent nuclear fuel
SQR	Software Qualification Report
SRC	source
SDD	System Description Document
SSCs	structures, systems and components
TAG	top air gap
TBD	to be determined
TBV	to be verified
TFGP	top fission gas plenum
TTP	Top Tie Plate
WPRC	Waste Package Remediation Cell
WHB	Waste Handling Building
WP	waste package

1. PURPOSE

The scope of this analysis is to estimate the shielding wall, ceiling or equivalent door thicknesses that will be required in the Waste Handling Building to maintain the radiation doses to personnel within acceptable limits. The shielding thickness calculated is the minimum required to meet administrative limits, and not necessarily what will be recommended for the final design. The preliminary evaluations will identify the areas which have the greatest impact on mechanical and facility design concepts. The objective is to provide the design teams with the necessary information to assure an efficient and effective design.

2. QUALITY ASSURANCE

An activity evaluation, *Waste Handling Facility Design*, (Work Package 12462400M6) (Reference 5.31) has been performed in accordance with QAP-2-0, *Conduct of Activities*, and has determined that this analysis is subject to the requirements of the *Quality Assurance Requirements and Description* (QARD) (Reference 5.1). In addition, review of the classification of permanent items described in *Classification of the Preliminary MGDS Repository Design* (TBV-228)(Reference 5.29), prepared in accordance with QAP 2-3, *Classification of Permanent Items*, has been performed. Table 2-1 presents the QA classification assignments.

Table 2-1. QA Classification Assignments

System	SSC	QA classification ¹
SU02 - Waste Handling Facility (WHF) System	Waste Handling Building Foundations and Structures	QA-1, QA-3, QA-4, QA-5, QA-7

¹ QAP-2-3 REV 09 (07/20/98) defines only five QA classifications. However, *Classification of the Preliminary MGDS Repository Design* has not yet been revised to reflect the changes. Therefore, classifications are as listed in QAP-2-3 REV 08.

Based on Table 2-1, items addressed in this analysis are to be considered "Q" items and, as specified in NLP-3-18, *Documentation of QA Controls on Drawings, Specifications, Design Analyses, and Technical Documents*, this analysis is documented as being subject to QA controls.

The results from this analysis are preliminary and will not be used directly for procurement, fabrication, or construction. Any data from this analysis that are used as design input must be treated as unconfirmed (TBV) and tracked per NLP-3-15 *To Be Verified and To Be Determined Monitoring System*.

3. METHOD

3.1 METHODOLOGY

A mapping and shielding matrix was developed for the Waste Handling Building based on Figure 4 in the *Surface Nuclear Facilities Space Program Analysis* (Reference 5.9), which shows the floor at EL. 100+0 for the Waste Handling/Waste Treatment Building - Ground Level.

Radiation source terms for the analysis are established. Radiation source terms are an essential and necessary element of shielding and design analysis. The gamma source terms used in this analysis are in the *Repository Radiation Shielding Design Guide* (Reference 5.8) and are based on Reference 5.10, *Advanced Conceptual Design* (ACD) DBF for the design of waste packages. Neutron and secondary gamma source terms were not considered in areas outside of hot cells due to the attenuation effect of concrete (see Assumption 4.3.1.6).

Analytical models are developed and described in Section 7 to simulate the types of analyses performed. The analytical models describe the sources, configurations, and materials required for use in the analyses using QAD.

A figure showing a plan and elevation view of the source-shield configuration being assessed was then established. The figure shows the DC regions surrounded by the liner, air gap, inner and outer barriers dimensions and the distance from a concrete wall. The figure shows the materials used in each region.

Generic QAD input listings are then generated based on the source-shield configuration and materials used. Dose points in mrem per hour are then established at 1'0" (30.48cm), 2'0" (60.96cm) and 3'0" (91.44cm) intervals in the concrete wall at the package mid-plane and at two lateral distances of 4'0" (121.92cm) and 8'0" (243.84cm). Iterative calculations for each of the three regions of the source were then made to determine the shield thickness requirements. The dose rate results from the spent fuel assembly (SFA) active fuel region (AFR), top tie plate (TTP), top fission gas plenum (TFGP) and lower tie plate (LTP) regions were then tabulated.

Plots were generated for each iterative calculation. These plots are useful to assist the design team as equipment is sized and located in the facility. They are also useful in determining facility exposures.

A determination was then made as to where the shielding solution for each analysis could be applied within the WHB facility.

3.2 SHIELDING ANALYSIS

This analysis uses one qualified shielding computer code, namely, QAD-vCGGP-A (CSCI# 30010 vCGGPA)(Ref. 5.7) (hereafter referred to as QAD) computer codes to analyze the dose rate consequences on the occupied areas of the source-shield configuration.

4. DESIGN INPUT

4.1 DESIGN PARAMETERS

Table 4-1 contains the selected input parameters for all the Radiological Design Analyses in this analysis. Data/Product Input in Table 4-1 are Non-Qualified. Reference DTN MO98WSTAN00011.000 when using information from Table 4-1.

Table 4-1. Input Parameters: Non-Qualified Data/Product*

Parameter	Value	Data Source	TBV ^a
Fuel Assembly Region Length: Active Fuel Region (AFR):	11'-9.8" (360.18 cm)	See Reference 5.6 LWR Radiological PC Database - Physical Description Report	7 TBV-455
AFR Characteristics	Design Basis Fuel: 4.2% Enrichment PWR 48,086 MWd/MTIHM 10 year decay time 0.464 MTIHM per assembly	See Reference 5.8, Table 4-1, pg 6.	7 TBV-455
DHLW Canister Dimensions:	Outside diameter: 60.96 cm Inside diameter: 59.06 cm Overall height: 300 cm Percent Fill: 85% Wall material: SS Wall thickness: 0.95 cm Canister vol: 7.36e6cm ³ glass density: 2.64 g/cm ³	Reference 5.20 pg. 3.1-7, 3.1-10, 3.4-2, 3.4-4, 3.4-7.	7 TBV-455
DHLW Canister Staging Station Source: Hanford per canister 10 year decay	MeV photons/sec 1.00e-02 2.50e-02 3.75e-02 5.75e-02 8.50e-02 1.25e-01 2.25e-01 3.75e-01 5.75e-01 8.50e-01 1.25e+00 1.75e+00 2.25e+00 3.50e+00 5.00e+00 7.00e+00 9.50e+00 9.91e+14 2.07e+14 2.27e+14 1.96e+14 1.12e+14 7.43e+13 9.41e+13 4.20e+13 1.49e+15 1.05e+13 5.25e+12 2.54e+11 1.44e+09 5.48e+06 5.27e+04 6.02e+03 6.88e+02	Reference 5.6 Defense High Level Waste Characteristics Database See Output in Attachment IX, pg IX-13.	7 TBV-455
Co-60 Source (Ci) based on 60,000 MWd/MTIHM, 10 year decay time: TTP LTP TFGP	Ci: 111.5 151.0 57.64	Reference 5.6 LWR Radiological Database - Radiological Report See Output in Attachment IX, pg IX-8. See Output in Attachment IX, pg IX-10. See Output in Attachment IX, pg IX-8.	7 TBV-455

Parameter	Value		Data Source	TBV ^a
AFR Photon Report Results for 48,086 Mwd/MTIHM, 4.2% initial enrichment, and 10 years decay.	Ave. Energy (MeV)	γ's/sec/MTIHM	Reference 5.6 LWR Radiological Database - Photons Report. See Output in Attachment IX,pg IX-12.	7 TBV-455
	0.01	2.729E+15		
	0.025	5.788E+14		
	0.0375	7.127E+14		
	0.0575	5.397E+14		
	0.085	3.230E+14		
	0.125	3.146E+14		
	0.225	2.641E+14		
	0.375	1.257E+14		
	0.575	4.799E+15		
	0.85	4.673E+14		
	1.25	3.850E+14		
	1.75	4.881E+12		
	2.25	7.693E+10		
	2.75	5.941E+09		
	3.50	7.265E+08		
	5.0	2.902E+07		
	7.0	3.346E+06		
	9.5	3.844E+05		
Cask Preparation (Canister) Cell				
NAC-STC Cask Dose Rates:			Reference. 5.24, Table 5.1-10	3A1N/A
Radial Surface	42.77 mrem/hr			
Radial 1 m from cask	19.48 mrem/hr			
Radial 2 m from vehicle	7.96 mrem/hr			
NAC-STC Cask Diameter	7'-3.6" (222.504cm)		Reference. 5.24 Drawing 800	3A1N/A
Radius	3'-7.6" (111.252cm)			
NLI10/24 Cask Dose Rates			Reference. 5.23, pg. IN/A-3	3A1N/A
Radial Surface	38.2 mrem/hr			
Radial 1 m from cask	11.0 mrem/hr			
Radial 2 m from vehicle	5.48 mrem/hr			
NLI10/24 Cask Diameter	8'-0" (243.84cm)		Reference. 5.23 C of C pg. 1 of 4.	3A1N/A
Radius	4'-0" (121.92cm)			
TN-REG Cask Dose Rates			Reference. 5.26, pg. 5-21	3A1N/A
Radial Surface	38.6 mrem/hr			
Radial 1 m from cask	23.3 mrem/hr			
Radial 2 m from vehicle	9.30 mrem/hr			
TN-REG Cask Diameter	7'-6.25" (229.235cm)		Reference. 5.26 pg. 1-2	3A1N/A
Radius	3'-9.13" (114.62cm)			
HiStar ^(c) Cask Dose Rates			Reference. 5.25, pg. 5.1-3	3A1N/A
Radial Surface	37.8 mrem/hr			
Radial 1 m from cask	9.59 mrem/hr			
Radial 2 m from vehicle	8.27 mrem/hr			
HiStar Cask Diameter	7'-11.88" (243.52cm)		Reference. 5.25 Figure 1.1-4	3A1N/A
Radius	3'-11.94" (121.76cm)			
Conversion Factor: disintegrations (d) /sec(s) per curie(Ci)	3.7E+10 d/s/Ci		See Reference 5.27, pg 23	3A2N/A
Co-60 photons per disintegration	MeV 1.173 1.332		See Reference 5.27, pg 249	3A2N/A
Material Densities:			See Reference. 5.8, pg 45	3A2N/A
Air:	ρ = 0.00122 g/cm ³		Pg 45	
NBS Concrete:	ρ = 2.35 g/cm ³		Pg 44	
A825:	ρ = 8.14 g/cm ³		Pg 44	
A516:	ρ = 7.832 g/cm ³		Pg 44	
SS316B6A:	ρ = 7.745 g/cm ³		Pg 44	
SS316L Density	7.9497 g/cm ³		See Reference. 5.8	3A2N/A

Parameter	Value		Data Source	TBV ^a
SS316L elemental Composition & density	Element	Weight Percent	See Reference. 5.8	3A2N/A
	C	0.03		
	N	0.1		
	Si	0.753		
	P	0.045		
	S	0.03		
	Cr	17		
	Mn	2		
	Fe	65.545		
	Ni	12		
Mo	2.5			
Dose Rate Conversion Factors	(MeV)	(mrem/hr)/ (ph/cm ² -sec)	See Reference 5.21, Table 3, pg 5.	3A2N/A
	1.00e-02	2.853e-03		
	2.50e-02	5.540e-04		
	3.75e-02	6.920e-04		
	5.75e-02	2.940e-04		
	8.50e-02	2.620e-04		
	1.25e-01	3.310e-04		
	2.25e-01	5.660e-04		
	3.75e-01	9.320e-04		
	5.75e-01	1.315e-03		
	8.50e-01	1.755e-03		
	1.25e+00	2.311e-03		
	1.75e+00	2.930e-03		
	2.25e+00	3.470e-03		
	2.75e+00	3.963e-03		
	3.50e+00	4.220e-03		
	5.00e+00	5.800e-03		
	7.00e+00	7.293e-03		
	9.50e+00	9.153e-03		
DHLW Canister Staging Station				
Large 21 PWR MPC Shell Thickness	0'-1" (2.54 cm)		See Reference.5.10, Section 3 Figure 3.1-1	6N/A
Disposal Container Storage and Welding Station Cell				
Welder Airlock Location	24'-0" (731.52cm) above DC Cell EL.100'		See Reference. 5.9 Attachment II, Figure 10 and Figure 12 and Attachment I, Figure 8.	6N/A
Canister Staging Cell (CSC)				
Cart Dimensions	15'-0" (457.2cm) length 11'-0" (335.28cm) width		See Reference. 5.9 Attachment II, Figure 6.	6N/A
Fuel Assembly Region Length: Top Tie Plate (TTP): Lower (bottom) Tie Plate (LTP): Top Fission Gas Plenum :	0'-6.82" (17.31 cm) 0'-1.5" (3.81 cm) 0'-8.94" (22.71 cm)		See Reference 5.15 pg. 53.	7 TBV 368 7 TBV 368 7 TBV 368
Material Density: AFR TTP LTP Liner	$\rho = 3.043 \text{ g/cm}^3$ $\rho = 2.4701 \text{ g/cm}^3$ $\rho = 2.5402 \text{ g/cm}^3$ $\rho = 8.0038 \text{ g/cm}^3$		See Reference 5.15 pg.II-5. See pg.II-7,Table II-4. See pg.II-8,Table II-5. See Reference 5.15 pg. II-6	7 TBV 368
Radius of Fuel Assembly Region (cm): AFR: Top Tie Plate (TTP): Lower Tie Plate (LTP): SFA Basket Liner :	2'-0.02" (61.0) cm 2'-0.02" (61.0) cm 2'-0.02" (61.0) cm 2'-0.68" (62.68 cm)		See Reference 5.15 pg. 53.	7 TBV 368
DC Shield Wall (Side): Size & Material	Inner Barrier: 2 cm A825 Outer Barrier:10 cm A516		See Reference. 5.15 pg 76. See Reference 5.15 pg 20.	7 TBV 368
DC Shield Wall (Top): Size & Material	Inner Barrier:2.5 cm A825 Outer Barrier:11 cm A516		See Reference. 5.15 pg 53..	7 TBV 368
DHLW Canister Staging Station				
DHLW center to center distance in staging area.	2'-5" (76.2 cm)		See Reference. 5.16 pg. 28 of 50 Figure 7.1-6	7 TBV 368
DHLW distance to Canister Staging Station walls.	2'-0" (60.96 cm)		See Reference. 5.16 pg. 28 of 50 Figure 7.1-6	7 TBV 368
Canister Buffer Storage: DHLW	Maximum 20 DHLW Canisters		See Reference. 5.16 pg 46 Figure 7.3-3	7 TBV 368

Parameter	Value	Data Source	TBV*
Minimum crane approach	9'-0" (274.32cm)	See Reference. 5.16 Figure 7.3-4, Canister Handling Area Plan View.	7 TBV 368
Cask Cart Center to OG Wall.	11'-5" (350.52cm)	See Reference. 5.16, Figure 7.3-2.	7 TBV 368
Disposal Container Storage and Welding Station Cell			
Turntable Height	2'-0" (60.96cm)	See Reference. 5.9 Attachment II, Figure 12	7 TBV 368
Canister Staging Cell (CSC)			
Center of Cart from OG Wall:	11'-6" (350.52cm) Cart Width = 11'-0" Door Opening = 13'-0" Door - Cart = 2'-0" Cart center = 1 1/2 = 5'-6" Delta/Side = 1'-0" Shield Wall Length = 5'-0"	See Reference. 5.9 Attachment II, Figure 6.	7 TBV 368
Loaded DC Staging Area (LDCSA)			
DC Center to Center	11'-1" (336.28cm)	See Reference. 5.9 Attachment II, Figure 11.	7 TBV 368
Number of DC's per long wall face.	7	See Reference. 5.9 Attachment II, Figure 11.	7 TBV 368
<p>a - Reference states dose rates from overpack. 49 CFR 173.441 Radiation Level Limitations, section (3) states: 10 mrem per hour (0.1 millisievert per hour) at any point 2 meters (6.6 feet) from the outer lateral surfaces of the vehicle; or in the case of a flat-bed style vehicle, at any point 2 meters (6.6 feet) from the vertical planes projected by the outer edges of the vehicle.</p> <p>b: Note: This note shows the logic that was applied in determining TBV/TBD applicability.</p> <p>2A - This product is <u>not</u> for a QA-1 to QA-7 system, therefore the data is not controlled as TBV/TBD.</p> <p>3A - This information is not considered data.</p> <p>3A1N/A - The product input is obtained from a valid source, therefore the product input is not controlled as TBV/TBD.</p> <p>3A2N/A - The product input is in the realm of accepted engineering practices, therefore the product input is not controlled as TBV/TBD.</p> <p>3A3N/A - The result is <u>not</u> sensitive to the range of the product input, therefore the product input is not controlled as TBV/TBD.</p> <p>3A4N/A - The product input does not affect the system's critical characteristics, therefore the product input is not controlled as TBV/TBD.</p> <p>3A5 - The product input must be marked as TBV/TBD.</p> <p>4N/A - This data is considered qualified from origin, therefore the data are not controlled as TBV/TBD.</p> <p>5N/A - This data is considered accepted data, therefore the data are not controlled as TBV/TBD.</p> <p>6N/A - This data does not affect the systems critical characteristics, scientific results, or conclusions, therefore the data are not controlled as TBV/TBD.</p> <p>7TBV - The data/product input must be marked as TBV/TBD.</p>			

*Use DTN MO98WSTAN00011.000 when using information from Table 4-1.

4.2 CRITERIA

This section documents the design criteria that are directly applicable to these analyses. Applicable requirements such as 10 CFR 20, 10CFR835, Controlled Design Assumptions CDA) Document, Assembly Transfer System Description Document (SDD) (Reference 5.2), and/or the Canister Transfer SDD (Reference 5.3) are cited after each criteria.

Standards or other DOE-accepted documents, identified as sufficient or more appropriate for the support of design, shall be used at the discretion of the designers, provided there are no conflicts with the identified requirements of the Repository.

4.2.1 The following criterion is based on information found in the Assembly Transfer SDD and the Canister Transfer SDD.

4.2.1.1 The MGR Surface Facilities and the operation of these facilities shall be designed to control occupational radiation exposures to as low as reasonably achievable and not to exceed 500 mrem per year while performing handling operations.

Basis: Reference 5.2, Volume II, Section 1.2.2.1.11, Reference 5.3, Volume II, Section 1.2.2.1.14.

- 4.2.1.2** Areas of the MGR Surface Facilities which are full time occupied will be designed to a dose rate of 0.25 mrem/hr. The 0.25 mrem/hr will be used in shielding calculations as the design dose rate.

Basis: Reference 5.2, Volume II, Section 1.2.2.1.11, Reference 5.3, Volume II, Section 1.2.2.1.14.

- 4.2.1.3** Areas in the MGR Surface Facilities where higher than design dose rates are permitted because of the infrequent occupancy or for other reasons verified by ALARA, will be assessed on a case by case basis and the appropriate radiological protection measures (e.g. tagging, monitoring, administrative control, etc.), will be incorporated into the design.

Basis: Reference 5.2, Volume II, Section 1.2.2.1.11, Reference 5.3, Volume II, Section 1.2.2.1.14.

4.2.2 The following criteria are based on information found in the CDA (Reference 5.30)

- 4.2.2.1** The Surface Facilities that house radioactive materials or in which work is performed on radioactive materials will be designed to control occupational exposures to ALARA and less than 500 mrem (mrem) per year and include and an ALARA evaluation process to identify where further dose reduction should be made based on a cost of \$2,000.00 (1995 \$'s) per person rem averted over the life of the facility until closure.

Basis: Key 89 & Key 090 (Reference 5.30)

4.3 ASSUMPTIONS

- 4.3.1** General Assumptions - The following general assumptions will be used throughout Attachment I, II, III, IV, V, VI and VII.

Note: Refer to Note b of Table 4-1 for TBV/TBD applicability logic.

- 4.3.1.1** For dose assessment purposes it is assumed that the 500 mrem per year exposure criteria (Key 89 (Reference 5.3)) is met by limiting the exposure of an operator in the operating corridor to less than 0.25 mrem per hour. This allows an operator assigned to work in those areas for 40 hours per week and 50 weeks per year.

[TBV: 3A1 N/A]

Basis: Heavy Work Load in occupied areas.

Usage: Used throughout the analysis.

- 4.3.1.2** For modeling purposes, it will be assumed that the shielding walls contain no internal shielding materials such as the steel lattice support structure.

[TBV:3A2 N/A]

Basis: Industry practice at this stage of the design process.

Usage: Used throughout the analysis.

- 4.3.1.3** National Institute of Standards and Technology (NIST) (formerly National Bureau of Standards (NBS)) concrete will be used for all shielding calculations.

[TBV : 3A1 N/A]

Basis: Standard concrete for use in model development.

Usage: Used throughout the analysis.

- 4.3.1.4** It is assumed that the dose points are located at the axial mid-plane of the fuel elements (AFR + TFGP + TTP + LTP), MPCs, or DHLW Canisters.

[TBV: 3A2 N/A]

Basis: The maximum dose would occur at one-half the height of the active fuel element.

Usage: Used throughout the analysis.

- 4.3.1.5** It is assumed for modeling purposes that south is toward the DC Transporter Airlock and north is toward the Carrier Bay of the Waste Handling Building.

[TBV: 3A4 N/A]

Basis: The shift in coordinates from magnetic N-S does not impact results and is used for general orientation only. (Reference 5.9, pg I-4)

Usage: Used throughout the attachments.

- 4.3.1.6** It is assumed that there is no neutron and secondary gamma dose outside concrete shielded high radiation areas such as the DC Load and DC Handling Cell.

[TBV: 3A2 N/A]

Basis: Neutron and secondary gamma source terms were not considered in areas outside due to the attenuation effect of concrete (see Reference 5.22, pg. 458 Fig 8.12).

Usage: Used throughout the analysis.

- 4.3.1.7** It is assumed the shielding thickness required for doors is equivalent to adjacent walls.

[TBV: 3A2 N/A]

Basis: Shield doors must provide same shielding equivalency as the adjacent walls, or limit operator exposure to less than 500 mrem per year exposure criteria (DCS 003) (Reference 5.3).

Usage: Used throughout the analysis.

- 4.3.1.8** It is assumed that the active fuel region (AFR) contents are homogenized over the entire AFR of the Waste Package (WP), Disposal Container (DC), Multi-Purpose Canister (MPC), or the internal area of the Spent Fuel Assembly (SFA) Drying Vessel, thus reducing the physical density of the contents, but maintaining the source strength.
[TBV: 3A2 N/A]

Basis: This method of solution was shown in Reference 5.14 to be conservative in results, and therefore, the results are not compromised.

Usage: Used throughout the analysis.

- 4.3.1.9** It is assumed that the lower tie plate (LTP) contents are homogenized over the entire LTP region of the Waste Package (WP), Disposal Container (DC), Multi-Purpose Canister (MPC), or the internal area of the Spent Fuel Assembly (SFA) Drying Vessel, thus reducing the physical density of the contents, but maintaining the source strength.
[TBV: 3A2 N/A]

Basis: This method of solution was shown in Reference 5.14 to be conservative in results, and therefore, the results are not compromised.

Usage: Used throughout the analysis.

- 4.3.1.10** It is assumed that the top tie plate (TTP) region contents are homogenized over the entire TTP region of the Waste Package (WP), Disposal Container (DC), Multi-Purpose Canister (MPC), or the internal area of the Spent Fuel Assembly (SFA) Drying Vessel, thus reducing the physical density of the contents, but maintaining the source strength.
[TBV: 3A2 N/A]

Basis: This method of solution was shown in Reference 5.14 to be conservative in results, and therefore, the results are not compromised.

Usage: Used throughout the analysis.

- 4.3.1.11** It is assumed that the top fission gas plenum (TFGP) and lower fission gas plenum contents are homogenized over the entire TFGP region of the Waste Package (WP), Disposal Container (DC), Multi-Purpose Canister (MPC), or the internal area of the Spent Fuel Assembly (SFA) Drying Vessel, thus reducing the physical density of the contents, but maintaining the source strength.
[TBV: 3A2 N/A]

Basis: This method of solution was shown in Reference 5.14 to be conservative in results, and therefore, the results are not compromised.

Usage: Used throughout the analysis.

- 4.3.1.12** It is assumed for modeling purposes that the DC is in a vertical position.
[TBV: 3A1 N/A]

Basis: This is the present configuration as shown in the Surface Nuclear Facilities Space Program Analysis (Reference 5.9).

Usage: Used in Attachment I, II, III, IV, V, and Attachment VII, Section 4.

- 4.3.2** The following case specific assumptions are made relative to Attachment II - Canister Cask Preparation Shield Wall Analysis:

- 4.3.2.1** It is assumed that a cask is positioned 7'10.2" (239.27 cm) from the Canister Cask Preparation Area Operating Gallery wall.

[TBV: 3A1 N/A, Reference 5.9]

Basis: Cask positioned in the middle of the transfer cart.

Usage: Used in Attachment II, section 4.

- 4.3.2.2** It is assumed that dose rates are based on vendor information, where available. Where there is insufficient information provided in the vendors SAR's to determine dose rate values, values are estimated based on other systems and engineering judgement.

[TBV: 3A2 N/A]

Basis: It is intended that these estimates are performed using average dose rate values that would be characteristic of operations over a long period of time. No credit is taken for spent fuel source term decrease due to radioactive decay. Therefore, dose estimated should be conservative compared to actual operating conditions.

Usage: Used in Attachment II, Section 4.

- 4.3.2.3** It is assumed that the gamma and neutron dose rate profile at contact, 1 meter, and 2 meters from the vehicle, for the casks listed in Table 4-2, can be extrapolated dose rates out to 25 feet.

[TBV: 3A2 N/A]

Basis: Table of dose rates at contact, 1 meter (3.28 feet) and 2 meters (6'5") dose rate from the vehicle sides.

Usage: Used throughout Attachment II.

Table 4-2. Shipping Casks Dose Rates vs Distance

Feet	NAC-STC	NLI 10/24	Hi-Star 100	TN-REG
0	42.77	38.2	37.8	38.6
1	35	27.35	30	33
2	28	18.08	24.5	27.1
3	21	11.92	20	23.9
4	18	9.94	16.3	20.3
5	16	8.61	13.3	16.7
6	12	7.33	10.7	14.2
7	8.5	6.29	9.09	12.1
8	6.8	5.33	8.16	10.3
9	5.2	4.48	7.25	8.6
10	4.4	3.82	6.43	7.33
11	3.4	3.23	5.78	6.13
12	2.7	2.74	5.07	5.22
13	2.1	2.33	4.55	4.44
14	1.7	1.97	4	3.71
15	1.4	1.64	3.58	3.22
16	1.1	1.36	3.18	2.69
17	0.81	1.14	2.82	2.25
18	0.66	0.97	2.51	1.95
19	0.52	0.83	2.25	1.63
20	0.41	0.70	2	1.36
25	0.14	0.3	1.2	0.59

4.3.2.4 It is assumed for modeling purposes that casks entering the Waste Handling Building will not have a contact dose rate greater than 200 mrem per hour, or a 2 meter dose rate greater than 10 mrem per hour.

[TBV: 3A1 N/A, 49CFR173.441]

Basis: This is a conservative assumption based on the fact that 49 CFR 173.441 Radiation Level Limitations, section (3) states: 10 mrem per hour (0.1 millisievert per hour) at any point 2 meters (6.6 feet) from the outer lateral surfaces of the vehicle; or in the case of a flat-bed style vehicle, at any point 2 meters (6.6 feet) from the vertical planes projected by the outer edges of the vehicle.

Usage: Used throughout Attachment II, section 4.

4.3.3 The following case specific assumptions are made relative to Attachment VI: Spent Fuel Assembly (SFA) Drying Vessel.

4.3.3.1 It is assumed that 21 SFAs are centered inside the SFA Drying Vessel.

[TBV: 3A2 N/A]

Basis: See Assumption 4.3.1.8, Assumption 4.3.1.9, Assumption 4.3.1.10, and Assumption 4.3.1.11.

Usage: Used in Attachment VI, section 4.

- 4.3.3.2** It is assumed that the SFA are positioned 68.54 cm from the walls of the SFA Drying Vessel Cell.

[TBV: 3A1 N/A]

Basis: This would ensure adequate space and therefore adequate cooling for the spent fuel assembly. Space Requirement Figure for the Dry Assembly Handling Cell (Reference 5.9).

Usage: Used in Attachment VI, section 4.

- 4.3.3.3** It is assumed that maintenance has to be performed at the Incline Transfer Canal Opening.

[TBV: 3A2 N/A]

Basis: Space Requirement Figure for the Dry Assembly Handling Cell (Reference 5.9).

Usage: Attachment VI, Section 4.

- 4.3.3.4** It is assumed that the internal radius of the SFA Drying Vessel is 4'3" (129.54 cm).

[TBV: 3A1 N/A, Reference 5.9]

Basis: Figure 4 of Reference 5.9.

Usage: Attachment VI, section 4.

- 4.3.3.5** It is assumed that the SFA Drying Vessel material is SS316L, 0'0.5" (1.27 cm) thick.

[TBV: 3A2 N/A]

Basis: Mechanical design input based on engineering judgement.

Usage: Attachment VI, section 4.

- 4.3.3.6** It is assumed that the distance from the centerline of the SFA Drying Vessel to the transfer canal is 9'11" (283.21 cm).

[TBV: 3A1 N/A]

Basis: Assumed SFA Drying Vessel diameter and requirement to provide adequate shielding between SFA Drying Vessel and transfer canal.

Usage: Attachment VI, section 4.

- 4.3.4** The following case specific assumptions are made relative to Attachment V: Disposal Container Staging Cell:

4.3.4.1 It is assumed that a DC centerline position is 304.8 cm from the south wall. [TBV: 3A3 N/A]

Basis: This is the present configuration per the Surface Nuclear Facilities Space Program Analysis (Reference 5.9).

Usage: Used in Attachment I, section 3.

4.3.5 The following specific assumption is made relative to Attachment III: Defense High Level Waste Canister Stage Cell (DHLWCSC):

4.3.5.1 It is assumed that the DHLW is decayed 10 years.

[TBV: 3A2 N/A]

Basis: Same decay as Design Basis Fuel (DBF).

Usage: Used in Attachment III, section 3.

4.3.6 The following specific assumptions are made relative to Attachment IV: Canister Handling Area (CHA).

4.3.6.1 It is assumed for modeling purposes that one multi-purpose canister (MPC) is staged 9'0" (274.32 cm) + 2'8.07" (81.45cm) or 11'8.07" (355.77cm) from the west Operating Gallery (OG) wall.

[TBV: 3A1 N/A, Reference 5.9]

Basis: From Reference. 5.16, Figure 7.3-4 the DC with a radius of 2'8.07" (81.45cm) is on the DC Loading Station cart positioned 9'0" (274.32cm) from the Canister Handling Area wall.

Usage: Used in Attachment IV, section 4.

4.3.6.2 It is assumed for modeling purposes that the MPC is directly transferred from the MPC Cask to a DC.

[TBV: 3A2N/A, Reference 5.17]

Basis: Reference 5.17, page 45, last paragraph, second sentence states, "The major change in handling these canisters is the direct transfer of a canister from a cask to a DC."

Usage: Used in Attachment IV, Section 4.

4.3.6.3 It is assumed that MPCs and DHLW Canisters are the only waste forms handled in the CHA.

[TBV: 3A1 N/A, Reference 5.16]

Basis: Reference 5.16, Table 4.3-1 (Mechanical Design Characteristics of the Casks, Canisters and DC).

Usage: Used in Attachment IV, Section 4.

4.3.7 The following specific assumptions are made relative to Attachment VII: DC Handling Cell (DCHC):

- 4.3.7.1** It is assumed that the top outer lid of the DC has been removed to facilitate the welding on the inner barrier lid.

[TBV: 3A2 N/A]

Basis: This is a conservative assumption based on the fact that the removal of the DC top outer barrier lid to facilitate welding on the DC inner barrier would reduce the available shielding between the DC and the Weld Operating Gallery.

Usage: Used in Attachment VII.

- 4.3.7.2** It is assumed that the DC Weld Station distance to the operating gallery wall is 9'0" (274.32 cm).

[TBV: 3A2 N/A]

Basis: Mechanical group engineering judgement.

Usage: Used in Attachment VII.

4.4 CODES AND STANDARDS

4.4.1 American National Standards Institute (ANSI)

ANSI/ANS 6.4.2-1985 Specification for Radiation Shielding Materials, 5/31/85

ANSI/ANS 6.4-1985 Guidelines on the Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants, 3/1/85

4.4.2 American Concrete Institute (ACI)

ACI 318-95 Building Requirements for Structural Concrete (ACI 318-95) and Commentary (ACI 318R-95), March 1, 1995, Catalog Number: 233584

ACI 349R-90 Code Requirements for Nuclear Safety Related Concrete Structures (ACI 349-90) and Commentary (ACI 349R-90) Supplement, March 1, 1990, Catalog Number: 237779

4.4.3 Code of Federal Regulations (CFR):

Title 10 Energy, Part 20-Standards for Protection Against Radiation, January 1, 1996, Catalog Number: 844.

Title 10 Energy, Part 835-Occupational Radiation Protection, 1996 Catalog Number: 846.

Note: It is recognized that 10 CFR 20 regulations supersede 10 CFR 835 regulations for the MGR. The design dose limit of 500 mrem per year in 10 CFR 835 are more ALARA for design considerations. There are no design dose limits in 10 CFR 20.

Title 49

Transportation, Part 173-Shippers -General Requirements For Shipments and Packagings, Catalog Number: 853.

4.4.4 NRC Regulatory Guides (RGs)

Regulatory Guide 8.8, *Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Reasonable Achievable (ALARA)*, Revision 3, June 1978, Catalog Number: 2887

Regulatory Guide 8.10, *Operating Philosophy for Maintaining Occupational Exposure As Low As Reasonable Achievable (ALARA)*, Revision 1-R, May 1975, Catalog Number: 2978

5. REFERENCES

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- 5.3 CRWMS M&O 1998b. *Canister Transfer System Description Document*, BCB000000-01717-1705-00001, Volume II, REV 00. Las Vegas, Nevada:Author. MOL.19980518.0158.
- 5.4 DOE 1998b. *Q-List:Yucca Mountain Site Characterization Project*, YMP/90-55Q, REV 05. Washington, D.C.: U. S. Department of Energy, Office of Civilian Radioactive Waste Management. Catalog Number: 231119.
- 5.5 CRWMS M&O 1996. *QAP-SI-0*, Rev.3, CRWMS - Computer Software Qualification, Las Vegas, Nevada:Author. MOL.19980205.0304.
- 5.6 CRWMS M&O 1993. *LWR Radiological PC Database, VA.MD.VS.1093.022*. CSCI:A000000000-02268-1200-20002 V1.1. DI: A000000000-02268-1200-20002. HQF.19940427.0022.

- 5.7 CRWMS M&O 1997a. *Software Qualification Report (SQR) for QAD VCGGP-A*, CSCI: 30010 VCGGP-1, MI: 30010-M04-001. DI: 30010-2003, Rev.00. Las Vegas, Nevada:Author. MOL.19970808.0504.
- 5.8 CRWMS M&O 1997b. *Repository Radiation Shielding Design Guide*, BC0000000-01717-2500-00001 REV 00, DCN1. Las Vegas, Nevada:Author. MOL.19980127.0312.
- 5.9 CRWMS M&O 1997c. *Surface Nuclear Facilities Space Program Analysis*, BCBD00000-01717-0200-00012 REV 01. Las Vegas, Nevada:Author. MOL.19980204.0138.
- 5.10 DOE 1994a. *Multi-Purpose Canister System Evaluation, A Systems Engineering Approach*, DOE/RW-0445, Washington,D.C.:Author. HQO.19940803.0001.
- 5.11 CRWMS M&O 1998c. *Waste Handling Building Shield Wall Analysis - QAD Computer Code Outputs*, Las Vegas, Nevada:Author. MOL.19980630.0102.
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- 5.17 CRWMS M&O 1998e. *Repository Waste Handling Integrated Model Development Report*, DI:BC0000000-01717-6700-00003, Rev.00. Las Vegas, Nevada:Author. MOL.19980220.0087.
- 5.18 DOE 1994b. *U.S.Department of Energy Radiological Control Manual*, DOE/RP/001, REV 1. Washington, DC: Author. MOL.19960206.0222.
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- 5.30 CRWMS M&O 1998e. *Controlled Design Assumptions Document (CDA)*, B000000000-01717-4600-00032, REV 05. Las Vegas, Nevada: Author. MOL.19980804.0481.
- 5.31 CRWMS M&O 1998f. *Activity Evaluation for Waste Handling Facility Design (Work Package 12462400M6)*. Las Vegas, Nevada: Author. MOL.19980223.0162.

6. USE OF COMPUTER SOFTWARE

6.1 COMPUTER HARDWARE

The computer hardware used in this analysis is a Dell OptiPlex Gxi desktop personal computer (CRWMS-M&O 113405). This computer has been qualified to perform execution of the software used in this analysis.

6.2 COMPUTER SOFTWARE

Software used in this analysis which is subject to the requirements of the QARD (Reference 5.1) includes QAD-vCGGP-A (Reference 5.7) and the *LWR Radiological PC Database* (Reference 5.6).

The QAD codes are a series of point-kernel code developed by Los Alamos Scientific Laboratories to calculate the gamma-ray penetration through various shield configurations. One of these codes, QAD-5A, was later modified by G. P. Lahti of the Lewis Research Center (Cleveland) to include gamma-ray and neutron data not in the original code, the modified code being identified as QAD-P5A. When the version of QAD with the combinatorial geometry subroutines is used, the code is identified as QAD-CG. QAD-CG uses a point kernel ray-tracing technique for gamma-ray calculations and either a Albert-Welton kernel or kernels obtained from the moments method solution of the Boltzmann equation for neutron penetration calculations. The GP version optionally makes use of the Geometric Progression (GP) fitting function for the gamma-ray buildup factor. This modification considerably reduces the time requirements for constructing the geometric zone input. QAD-vCGGP-A (CSCI #30010-vCGGP-A) is the latest version of the QAD codes. The QAD version includes the incorporation of a cubic interpolation scheme for the gamma attenuation coefficients, an added capability for source translation and rotation, correction of a potential error in evaluation of buildup factors at very deep penetration, and the adoption of complete free-format input readings routines from KENO-IV.

The *LWR Radiological PC Database* (Reference. 5.6) consists of six PC menu driven data bases. The following PC menu driven data bases were used in this analysis and are described briefly below:

1. LWR Radiological Data Base. This data base contains radionuclide compositions, heat generation rates, curies, and other information as a function of spent fuel type, enrichment, burnup, and decay time. Information in this data base is based upon a large number of runs using the ORIGEN2 computer code. The data is interpolated using curve fitting routines to allow the data base to give the user the results based upon the users input of functions.
2. LWR Assemblies Data Base. This data base contains the physical properties of the commercial fuel assemblies and fuel pins. It describes such things as the bundle configuration, the dimensions of all components, the materials of the constituents, the weights, etc.

3. High-Level Waste Data Base. This data base contains the physical and radiological descriptions of HLW, as the interim forms and as the immobilized forms.

6.3 SOFTWARE APPLICATION

In accordance with the M&O Computer Software procedure QAP-SI-0 (Reference 5.5), it was determined the software QAD-vCGGP-A (hereafter referred to as QAD) was: (1) was appropriate for the application, (2) was used only within the range of validation, and (3) was obtained from the Software Configuration Manager in accordance with appropriate procedures. The determination above was based on the Software Qualification Reports (SQRs) (Reference 5.7) that include a description of the mathematical models, the numerical methods used, testing, operating manuals and instructions, and the verification and validation performed to qualify the acquired software for each model.

In accordance with the M&O Computer Software procedure QAP-SI-0, it was determined the software LWR Radiological PC Database: (1) was appropriate for the application, (2) was used only within the range of validation, and (3) was qualified under the QAP-19 series of procedures that were in effect at that time in Vienna, Virginia, (4) was obtained from the Software Configuration Manager in accordance with appropriate procedures.. The software was installed on CPU #113405 in accordance with QAP-SI-3.

7. DESIGN ANALYSIS

7.1 INTRODUCTION

- 1) An abbreviated restatement of the objective.

Section 7.2 contains the objective and summary of the results from the Attachments. The summary(s) in Section 7.2 contains:

- 2) A statement specifying the data location and shield wall thickness required to meet the design dose rate.
- 3) A shield wall thickness recommendation statement.
- 4) A justification statement for the recommended shield wall thickness (e.g., uncertainty in assumptions, design flexibility, multiple sources/source locations).

All assumptions (General and Specific) for the shield evaluations are listed in Section 4.3 of this report. The specific assumptions listed in Section 4.3 may be reproduced in the attachments for clarity. Subsection 7.3 contains the summary or recommended wall size of the case evaluations documented in the attachments. Subsection 7.3 contains the Overall Facility Assessment.

The dose points selected are located 1 cm outside the shield wall, midplane of the source. This dose point would represent the highest dose rate from the source with respect to an operator

standing at an operator station and would establish the shielding thickness that is required to reduce to within acceptable limits, the worker radiological exposure from radiation originating during source movement operations. Establishing the dose point 1 cm outside the shield wall ensures the dose rate at the actual operator position is less than 0.25 mrem per hour and the general corridor is ALARA.

7.2 SHIELDING DESIGN CASE EVALUATIONS

7.2.1 DC Shield Wall Analysis

7.2.1.1 Objective

The objective of this analysis is to estimate the shield wall thicknesses at various locations in the Waste Handling Building (WHB) for a single disposal container (DC). This shielding is required to reduce, to within acceptable limits, the worker radiological exposure to radiation originating from single DC operations.

7.2.1.2 Summary

The objective of this analysis was to estimate the shield wall thicknesses at various locations in the Waste Handling Building (WHB) for a single DC. Table 7-1 lists the calculated shield wall thicknesses from Attachment I for single DC areas of the WHB in order to provide adequate shielding to meet the design dose rate of 0.25 mrem per hour (see Assumption 4.3.1.1) for the single DC operations using the assumptions (see Assumptions 4.3.1) and conceptual design concepts (see Attachment I) available at this time. At this phase of the design we are recommending that the shielding wall thickness for the single DC operations be 3'0" (91.44 cm) of concrete.

Table 7-1. Shield Wall Thickness for Single DC Areas of the WHB

Areas	Minimum Shield Wall Thickness feet (cm)	Recommended Thickness feet (cm)
DC Trans. Load	2'6.63" (77.8 cm)	3'0" (91.44 cm)
WP Remediation	2'9.03" (83.9 cm)	3'0" (91.44 cm)
DC Handling Cell West Wall	2'10.56" (87.78 cm)	3'0" (91.44 cm)
DC Handling Cell South Wall	2'8.7" (83.00 cm)	3'0" (91.44 cm)
DC Load	2'8.36" (83.3 cm)	3'0" (91.44 cm)
DC Decon	2'8.36" (83.3 cm)	3'0" (91.44 cm)
DC Tilting Station	2'6.63" (77.8 cm)	3'0" (91.44 cm)

7.2.2 Canister Cask Preparation Cell Shield Wall Analysis

7.2.2.1 Objective

The objective of this section of the analysis is to estimate the east shield wall thickness of the Canister Cask Preparation Cell (CCPC). This shielding is required to reduce, to within acceptable limits, the worker radiological exposure to radiation originating from the CCPC operations.

7.2.2.2 Summary

The objective of this section of the analysis is to estimate the east shield wall thickness of the Canister Cask Preparation Cell (CCPC). The data from Attachment II, Section 4, indicates that 1'1.2" (33.53cm) concrete would provide adequate shielding in the CCPC to meet the design dose rate criteria of 0.25 mrem per hour (see Assumption 4.3.1.1) using the assumptions (see Assumptions 4.3.1 and 4.3.2) and conceptual design concepts (see Attachment II) available at this time. At this phase of the design we are recommending that the shielding wall thickness for the CCPC be 1'1.2" (33.53cm) of concrete. This will provide flexibility for pursuing the layout of the CCPC relative to occupied areas during further design.

7.2.3 DHLW/DOE SNF Canister Staging Station Shield Wall Analysis

7.2.3.1 Objective

The objective of this section of the analysis is to estimate the shield wall thickness of the DHLW/DOE SNF Canister Staging Station (DHLWCSS). This shielding is required to reduce, to within acceptable limits, the worker radiological exposure to radiation originating from the DHLWCSS operations.

7.2.3.2 Summary

The objective of this section of the analysis is to estimate the shield wall thickness of the DHLW/DOE SNF Canister Staging Station (DHLWCSS). The graph of the data (see Attachment III, Section 4) indicates that 3'5" (104.14 cm) of concrete for the east and west walls of the DHLWCSS, 3'0" (91.44 cm) of concrete for the north and south walls of the DHLWCSS, and 2'0" (60.96 cm) for the DHLWCSS ceiling would provide adequate shielding to meet the design dose rate criteria of 0.25 mrem per hour (see Assumption 4.3.1) using the assumptions (Assumptions 4.3.1 and 4.3.5) and conceptual design concepts (see Attachment III) available at this time. This will provide flexibility for pursuing the layout of the DHLWCSS relative to occupied areas during further design.

7.2.4 Canister Handling Area Shield Wall Analysis

7.2.4.1 Objective

The objective of this analysis is to estimate the west and south shield wall thickness for the Canister Handling Area (CHA). This shielding is required to reduce, to within acceptable limits, the worker radiological exposure to radiation originating from the CHA operations .

7.2.4.2 Summary

The objective of this analysis was to estimate the shield wall thickness for the CHA. The graph of the data (see page 7 of Attachment IV) indicates that 3'10.46" (118 cm) of concrete would provide adequate shielding in the CHA to meet the design dose rate of 0.25 mrem per hour (see

Assumption 4.3.1.1) using the assumptions (see Assumption 4.3.1 and 4.3.6) and conceptual design concepts (see Attachment IV) available at this time. At this phase of the design we are recommending that the shielding wall thickness for the west wall be 121.92 cm. At this phase of the design we are recommending that the shielding wall thickness for the south wall be 4'0" (121.92cm) of concrete due to the fact that the MPC can be moved closer to the south wall as is the case with the Performance Confirmation Cell. This will provide flexibility for pursuing further design of the CHA.

7.2.5 DC Staging Cell (DCSC) Shield Wall Analysis

7.2.5.1 Objective

The objective of this analysis is to estimate the shield wall thickness for the DC Staging Cell (DCSC). This shielding is required to reduce, to within acceptable limits, the worker radiological exposure to radiation originating from the DCSC operations.

7.2.5.2 Summary

The objective of this analysis was to estimate the shield wall thickness for the DCSC. The graph of the data (see page 6 of Attachment V) indicates that 2'8.67" (83.00 cm) of concrete would provide adequate shielding in the DCSC to meet the design dose rate of 0.25 mrem per hour (see Assumption 4.3.1.1) using the assumptions (see Assumption 4.3.1 and 4.3.4) and conceptual design concepts (see Attachment V) available at this time. At this phase of the design we are recommending that the shielding wall thickness for the DCSC be 3'0" (91.44 cm) of concrete.

7.2.6 Drying Vessels/Transfer Canal Shield Wall Analysis

7.2.6.1 Objective

The objective of this analysis is to estimate the shield wall thickness between the SNF Drying Vessels and the Incline Transfer Canal. The shielding is required to meet the design dose rate criteria of 0.25 mrem per hour.

7.2.6.2 Summary

The objective of this analysis was to estimate the shield wall thickness required between the SNF Drying Vessels and the Incline Transfer Canal. The data (see Attachment VI, Table VI-1) indicates that 5'0" (152.4 cm) of concrete would provide more than adequate shielding in the SNF Drying Vessels and the Incline Transfer Canal to meet the design dose rate of 0.25 mrem per hour (see Assumption 4.3.1.1) using the assumptions (see Assumption 4.3.1 and 4.3.3) and conceptual concepts (see Attachment VI) available at this time. At this phase of the design we are recommending that the shielding wall thickness between the SNF Drying Vessels and the Incline Transfer Canal be 5'0" (152.4 cm) of concrete.

7.2.7 Disposal Container Handling Cell Shield Wall Analysis

7.2.7.1 Objective

The objective of this analysis is to estimate the shield wall thicknesses in the Disposal Container Handling Cell (DCHC) Welding Operating Gallery floor and walls. This shielding is required to reduce, to within acceptable limits, the worker radiological exposure to radiation originating from the DCHC operations.

7.2.7.2 Summary

The objective of this analysis was to estimate the shield wall thicknesses at various locations in the Disposal Container Handling Cell (DCHC) Welding Operating Gallery floor and walls. The graph of the data (see page 6 of Attachment VII) indicates that 4'0" (121.92 cm) of concrete on the floor and side of the Welding Operating Gallery would provide adequate shielding in the DCHC to meet the design dose rate (see Assumption 4.3.1.1) using the assumptions (see Assumption 4.3.1 and 4.3.8) for a single DC using the assumptions and conceptual design concepts (see Attachment VII) available at this time. At this phase of the design we are recommending that the shielding wall thickness for the DCHC be 4'0" (121.92 cm) of concrete on the floor and walls.

8. CONCLUSIONS

Due to the unqualified/unconfirmed input data used in this analysis the output/conclusions from this analysis can not be used as input into documents supporting procurement, fabrication, or construction unless they are controlled and tracked as TBV or TBD in accordance with NLP-3-15.

The objective of this analysis was to estimate the minimum shielding wall or door thicknesses that will be required in the Waste Handling Building to maintain the radiation doses to personnel within acceptable limits. The preliminary evaluations identified the areas which would have the greatest impact on mechanical and facility design concepts. The initial calculations which have been performed were selected on the basis of need by the design team. The purpose was to provide the design teams with the necessary information to assure an efficient and effective design.

The calculational models are the building blocks for the more sophisticated calculations that will follow when the design is developed to a point where the geometric dimensions, the materials, and other influencing factors are known.

At this phase of the design we are recommending that the shielding wall thicknesses for the Waste Handling Building be as shown in Table 8-1. The recommended shield wall thicknesses will ensure that, "The Surface Facilities that house radioactive materials or in which work is performed on radioactive materials will be designed to control occupational exposures to 500 mrem per year (Criteria:4.2.1.1) and will be used as input to more detailed analysis which will determine if the facility meets the ALARA criteria.." This will provide flexibility for pursuing the further design of the Waste Handling Building.

The models were established in such a way that other configurations in the WHB could be estimated by comparison of similar parameters. Areas of the Waste Handling Building ground elevation not evaluated during this analysis [e.g., Assembly Handling Line (DC Load Cell, DC Inner Lid Weld, Decontamination Cell) and Dry Handling Line (Handling Airlock, Cask Prep/Lid Unbolt Cell)] will be evaluated during the ongoing design process. At this phase of the design we are recommending that the shield wall thicknesses for these areas be made on the basis of areas that have been evaluated.

The Assembly Handling Line (DC Load Cell, DC Inner Lid Weld, and Decontamination Cell) contain the same source strength. By comparison with other cases, we are recommending that the minimum shield wall thicknesses for these areas be 5'0" (152.4 cm).

As the design of the Waste Handling Building progresses, further calculation will be required to recommend required shield wall or door thicknesses in areas not addressed in this release and to re-evaluate those areas changed due to the evolving Waste Handling Building design.

Table 8-1. Waste Handling Building Shielding Summary

Attachment #	Case Title	Shield Thickness Recommended
I	DC Transfer Load	3'0" (91.44 cm)
	WP Remediation	3'0" (91.44 cm)
	DC Handling Cell West Wall	3'0" (91.44 cm)
	DC Handling Cell South Wall	3'0" (91.44 cm)
	DC Load	3'0" (91.44 cm)
	DC Decontamination	3'0" (91.44 cm)
	DC Tilting Station	3'0" (91.44 cm)
II	Canister Cask Preparation Cell	1'1.2" (33.53 cm)
III	DHLW Canister Staging Cell	
	East and West Walls	3'5" (104.14 cm)
	North and South Walls	3'0" (91.44 cm)
	Ceiling	2'0" (60.96 cm)
IV	Canister Handling Area	4'0" (121.92 cm)
V	DC Staging Cell	3'0" (91.44 cm)
VI	Drying Vessels/Transfer Canal	5'0" (152.4 cm)
VII	DC Handling Cell - Operator Gallery Floor and Side	4'0" (121.92 cm)

9. ATTACHMENTS

<u>Attachment #</u>	<u>Attachment Title</u>
I	Disposal Container Shield Wall Analysis
II	Canister Cask Preparation Cell Shield Wall Analysis
III	DHLW/DOE SNF Canister Staging Station Shield Wall Analysis
IV	Canister Handling Area Shield Wall Analysis
V	DC Staging Cell Shield Wall Analysis
VI	Drying Vessels/Transfer Canal Shield Wall Analysis
VII	Disposal Container Handling Cell Shield Wall Analysis
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Disposal Container Shield Wall Analysis**Table of Contents**

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Disposal Container Shield Wall Analysis**1. Introduction**

The objective of this analysis is to estimate the shield wall thicknesses at various locations in the Waste Handling Building (WHB) for a single Disposal Container (DC). This shielding is required to reduce, to within acceptable limits, the worker radiological exposure to radiation originating from the DC operations.

Note: Due to the unqualified/unconfirmed input data used in this attachment the output/conclusions from this attachment cannot be used as input into documents supporting procurement, fabrication, or construction unless they are controlled and tracked as TBV or TBD in accordance with NLP-3-15.

2. Summary

The objective of this analysis was to estimate the shield wall thicknesses at various locations in the Waste Handling Building (WHB) for a single Disposal Container (DC). Table I-1 lists the results from Section 4 of Attachment I for a single DC in the areas specified using the assumptions (see Assumptions 4.3.1) and conceptual design concepts presented in this attachment. At this point in the design, the recommendation is that the shielding wall thickness for the areas listed in Table I-1 be 3'-0" (91.44 cm) of concrete.

Table I-1. Shield Wall Thickness for Single DC Areas of the WHB

Areas	Minimum Shield Wall Thickness
DC Transporter Load Cell	2'-6.63" (77.8 cm)
WP Remediation	2'-9.03" (83.9 cm)
DC Handling Cell West Wall	2'-10.56" (87.78 cm)
DC Handling Cell South Wall	2'-8.7" (83.00 cm)
DC Load	2'-8.36" (83.3 cm)
DC Decon	2'-8.36" (83.3 cm)
DC Tilting Station	2'-6.63" (77.8 cm)

3. Analysis Method**3.1 Single Disposal Container Model**

The calculation performed in this section determined the dose rate in mrem per hour from a single DC positioned in the areas shown in Table I-1.

Refer to Attachment VIII, Waste Handling Building (functional layout) for orientation of the areas listed in Table I-1 with respect to the rest of the WHB. For orientation purposes, the Carrier Bay is considered north and the DC Transporter Airlock is considered south. The DC Transporter Load and DC Transporter Airlock are located south of the DC Handling Area. The WP Remediation Cell is south and west of the DC Handling Area. The Electrical Distribution Room, el 100+ is located south of the DC Handling Area. The Canister DC Load/Decon Cells are located north of the DC Handling Area. The DC Tilting Station is located on the west side of the DC Handling Area, north of the DC Transporter Load Area and east of the DC Staging Area.

Disposal Container Shield Wall Analysis

3.1.1 Geometric Model

Figure I-1 depicts the geometric model for the QAD calculations. The DC was initially positioned 1 cm from the concrete shield wall. For modeling purposes it will be assumed that the shielding walls contain no internal shielding materials such as the steel lattice support structure (see Assumption 4.3.1.2).

The DC portion of the model was taken directly from Reference 5.10, page 53..

Figure I-1. DC Geometric Model

Disposal Container Shield Wall Analysis**3.1.2 Homogenized Material Densities**

As shown in Figure I-1, the fuel assembly region is divided into five axial zones for the limiting B&W 15x15 Mark B fuel assembly type, including the top tie plate (TTP), top fission gas plenum (TFGP), active fuel region (AFR), lower fission gas plenum (LFGP) and lower tie plate (LTP). As per Assumption 4.3.1.8, it was assumed that the active fuel contents are homogenized over the entire AFR region of the DC, thus reducing the physical density of the contents, but maintaining the source strength. As per Assumption 4.3.1.9, it was assumed that the LTP contents are homogenized over the entire LTP region of the DC, thus reducing the physical density of the contents, but maintaining the source strength. As per Assumption 4.3.1.10, it was assumed that the TTP contents are homogenized over the entire TTP region of the DC, thus reducing the physical density of the contents, but maintaining the source strength. As per Assumption 4.3.1.11, it was assumed that the TFGP and LFGP contents are homogenized over the entire TFGP region of the DC, thus reducing the physical density of the contents, but maintaining the source strength. This method of solution was shown in reference 5.14 to be conservative in results, and therefore, the results were not compromised. Air is used for the upper and lower fission gas plenums, which ignore Zircaloy tubing for conservatism.

3.1.3 Single DC Radiation Analysis

For this analysis, the QAD code was run for a single DC, using the model in Figure I-1. The dose rates were determined at 1'-0" (30.48 cm) intervals inside a 5'-0" (152.4 cm) thick concrete shield wall midplane of the DC. The DC was subsequently moved to 6'-6.74" (200.0 cm), 10'-0" (304.8 cm), 15'-0" (457.2 cm), 20'-0" (609.6 cm) and 25'-0" (762.0 cm) and the QAD runs repeated and the mid-plane dose rates were determined for each distance at the established intervals. The dose rates from all the QAD runs were then tabulated and dose rates summed.

3.1.3.1 Active Fuel Region Gamma Source Terms

The first step in the calculation was to determine the source spectrum for the DC-AFR. The DC-AFR source was obtained from the LWR Radiological Database (Reference 5.6). The photons report was run with the characteristics of 4.2% initial enrichment, 48,086 MWd/MTU, and 10 years decay. The photon report produced the source strength (Table I-2, column 3) in photons per second at the energy (MeV) levels shown in Table I-2, column 2. Table I-2, column 4 contains the Metric-Tons-Initial-Heavy-Metal (MTIHM) per Assembly from Table 4-1 of the main report. The source strength (column 3) is multiplied by the MTIHM per Assembly (column 4) to produce the source strength per assembly (column 5). The source strength per assembly column 5 is multiplied by the assemblies per DC (column 6) to produce the AFR Gamma Source (column 7). Table I-2 contains the AFR Gamma Source Terms used in the QAD shielding code.

Disposal Container Shield Wall Analysis

Table I-2. AFR Gamma Source Terms

Group	Energy (MeV)	Source Strength (γ's/sec per MTIHM)	MTIHM per Assembly	Source Strength per Assembly (γ's/sec)	Assemblies per DC	AFR Gamma Source (γ's/sec)
1	0.01	2.729E+15	0.464	1.27e+15	2.10e+01	2.66e+16
2	0.025	5.788E+14	0.464	2.69e+14	2.10e+01	5.64e+15
3	0.0375	7.127E+14	0.464	3.31e+14	2.10e+01	6.94e+15
4	0.0575	5.397E+14	0.464	2.50e+14	2.10e+01	5.26e+15
5	0.085	3.230E+14	0.464	1.50e+14	2.10e+01	3.15e+15
6	0.125	3.146E+14	0.464	1.46e+14	2.10e+01	3.07e+15
7	0.225	2.641E+14	0.464	1.23e+14	2.10e+01	2.57e+15
8	0.375	1.257E+14	0.464	5.83e+13	2.10e+01	1.22e+15
9	0.575	4.799E+13	0.464	2.23e+13	2.10e+01	4.68e+14
10	0.85	4.673E+13	0.464	2.17e+13	2.10e+01	4.55e+14
11	1.25	3.850E+13	0.464	1.79e+13	2.10e+01	3.75e+14
12	1.75	4.881E+12	0.464	2.26e+12	2.10e+01	4.76e+13
13	2.25	7.693E+10	0.464	3.57e+10	2.10e+01	7.50e+11
14	2.75	5.941E+09	0.464	2.76e+09	2.10e+01	5.79e+10
15	3.50	7.265E+08	0.464	3.37e+08	2.10e+01	7.08e+09
16	5.0	2.902E+07	0.464	1.35e+07	2.10e+01	2.83e+08
17	7.0	3.346E+06	0.464	1.55e+06	2.10e+01	3.26e+07
18	9.5	3.844E+05	0.464	1.78e+05	2.10e+01	3.75e+06

3.1.3.2 Co-60 Sources in Non-Fuel Components

The top tie plate, gas plenum and lower tie plate regions of each fuel assembly contain Co-60 sources, which are due to neutron activation of non-fuel components during reactor operation. For non-fuel components, Co-60 is the only activation product considered. Table 4-1 of the main report contains the CDB Co-60 sources in PWR non-fuel components for the parameters of 4.73% initial enrichment, 60,000 MWd/MTIHM and 10 years decay. The Design Basis Fuel (DBF) in Table 4-1 is defined as 4.2% initial enrichment, 48,086 MWd/MTIHM and 10 year decay. The CDB Co-60 source activity for 60,000 MWd/MTIHM is scaled down by the burnup ratio 48,086/60,000 or 0.801, as the Co-60 activity is fairly proportional to fuel burnup. The DBF of 4.2% initial enrichment, which is different from the CDB 4.73% initial enrichment, poses only a minor effect on the Co-60 source activity, as confirmed by the data given in Reference 5.28 for different enrichments.

Table I-3 contains the Co-60 source terms in the end fittings. The photon source strength in photons per second (s) per assembly (column 6) is equal to the curie content (column 2) multiplied by the burnup ratio (column 3) which is multiplied by the photons per disintegration (d) for Co-60 (column 4) that is then multiplied by the conversion factor d/s per curie (column 5). Column 6 is multiplied by column 7 to obtain the photons per second (sec) per DC (column 8).

Disposal Container Shield Wall Analysis

Table I-3. Co-60 End Fittings Source Terms

Region	Curies ^a per Assembly	Burnup Ratio 48.06/60	photons per disintegration ^b	d/s per curie ^b	photons/s per assembly	Assemblies per DC	Photon per sec per DC
Top tie plate	111.5	8.01e-01	2	3.70e+10	6.61e+12	21	1.39E14
Gas Plenum	57.64	8.01e-01	2	3.70e+10	3.42e+12	21	7.18E13
Lower Tie Plate	151.0	8.01e-01	2	3.70e+10	8.95e+12	21	1.88E14

a - See reference 5.10 page 45 Table 7.2-1

b - See Table 4-1 of the main report

3.1.3.3 QAD Input Established

The next step in the design process was to establish the geometry as shown in Figure I-1 for the AFR. The dimensions used in this calculation were obtained from Table 4.1 - Input Parameters of the main report. Using the source-shield geometry shown in Figure I-1, a QAD input was established with the DC initially positioned 1 cm from a concrete shield wall.

4. Analysis

Geometries were established for the AFR, TTP, LTP and TFGP regions. The dose rates were determined at 1'-0" (30.48 cm) intervals inside a 5'-0" (152.4 cm) thick concrete shield wall midplane of the DC. The DC was subsequently moved to 6'-6.74" (200.0 cm), 10'-0" (304.8 cm), 15'-0" (457.2 cm), 20'-0" (609.6 cm) and 25'-0" (762.0 cm) and the QAD runs repeated. The QAD input/outputs for the TTP, LTP and TFGP are located in Reference 5.11. The following is a list of the QAD input/output files:

<u>File</u>	<u>Format</u>	<u>Bytes</u>	<u>Date Created</u>	<u>Time Created</u>	<u>Disk#</u>
AFR200	WPD	55,645	06-04-98	2:12pm	1
AFR304	WPD	55,556	06-04-98	2:14pm	1
AFR457	WPD	55,556	06-04-98	2:16pm	1
AFR609	WPD	55,643	06-04-98	2:46pm	1
AFR762	WPD	55,643	06-04-98	2:51pm	1
AFR82	WPD	55,714	06-04-98	2:40pm	1
LO200	WPD	40,387	04-15-98	7:59am	1
LOW304	WPD	40,216	04-15-98	8:00am	1
LOW457	WPD	40,216	04-15-98	8:00am	1
LOW609	WPD	40,216	04-15-98	8:01am	1
LOW762	WPD	39,908	04-15-98	8:01am	1
LOW82	WPD	40,316	04-15-98	8:02am	1
PLEN200	WPD	38,891	04-15-98	8:06am	1
PLEN304	WPD	38,791	04-15-98	8:06am	1
PLEN457	WPD	38,791	04-15-98	8:06am	1
PLEN609	WPD	38,791	04-15-98	8:06am	1
PLEN762	WPD	38,791	04-15-98	8:06am	1
PLEN82	WPD	38,891	04-15-98	8:06am	1
UP200	WPD	39,095	04-15-98	8:11am	2

Disposal Container Shield Wall Analysis

UP304	WPD	38,995	04-15-98	8:11am	2
UP457	WPD	38,927	04-15-98	8:11am	2
UP609	WPD	38,791	04-15-98	8:11am	2
UP762	WPD	38,791	04-15-98	8:11am	2
UP82	WPD	30,095	04-15-98	8:11am	2

The dose rates from all the QAD runs were tabulated in Table I-4 and dose rates summed as shown in column 6 and column 11.

The dose rate data versus shield wall thickness from Table I-4 was then plotted for each of the centerline distances of 2'-8.46" (82.45 cm), 6'-6.74" (200.0 cm), 10'-0" (304.8 cm), 15'-0" (457.2 cm), 20'-0" (609.6 cm) and 25'-0" (762.0 cm). A line at the Design Dose Rate (DDR) of 0.25 was also included. The resultant plot is shown in Figure I-2.

Table I-4. Dose Rate (mrem/hour) from DC

Thickness (cm)	82.45 cm from DC Centerline					200 cm from DC Centerline				
	AFR	LTP	TTP	TFGP	Total	AFR	LTP	TTP	TFGP	Total
30.48	4.05e+02	1.27e-10	6.77e-09	6.72e-11	4.05e+02	2.59e+02	4.72e-01	1.00e-01	7.31e-02	2.60e+02
60.96	7.90e+00	6.32e-10	2.18e-08	3.67e-10	7.90e+00	4.91e+00	9.45e-03	1.76e-03	1.35e-03	4.92e+00
91.44	1.56e-01	4.69e-10	1.27e-08	2.95e-10	1.56e-01	1.01e-01	2.17e-04	3.81e-05	3.09e-05	1.01e-01
Thickness (cm)	304.8 cm from DC Centerline					457.2 cm from DC Centerline				
	AFR	LTP	TTP	TFGP	0	AFR	LTP	TTP	TFGP	Total
30.48	1.72e+02	1.45e+00	4.23e+00	1.15e+00	1.79e+02	1.02e+02	3.29e+00	7.88e+00	3.09e+00	1.16e+02
60.96	3.42e+00	2.45e-02	8.04e-02	2.09e-02	3.55e+00	2.14e+00	6.45e-02	1.68e-01	6.53e-02	2.44e+00
91.44	7.92e-02	4.48e-04	1.59e-03	4.07e-04	8.16e-02	4.75e-02	1.25e-03	3.49e-03	1.36e-03	5.36e-02
Thickness (cm)	609.6 cm from DC Centerline					762 cm from DC Centerline				
	AFR	LTP	TTP	TFGP	Total	AFR	LTP	TTP	TFGP	Total
30.48	6.43e+01	3.34e+00	7.67e+00	3.75e+00	7.91e+01	3.83e+01	2.43e+00	6.43e+00	3.28e+00	5.04e+01
60.96	1.41e+00	7.32e-02	1.78e-01	8.70e-02	1.75e+00	8.63e-01	5.60e-02	1.56e-01	8.10e-02	1.16e+00
91.44	3.22e-02	1.53e-03	3.89e-03	1.92e-03	3.95e-02	2.02e-02	1.22e-03	3.54e-03	1.87e-03	2.68e-02

The intersection of the DDR curve and each centerline distance dose rate (DR) curve represents the minimum shield wall thickness required to meet a dose rate of 0.25 mrem per hour from a single DC positioned at that centerline distance. Figure I-3 is a plot of Shield Wall Thickness (cm) required to meet the DDR at each centerline distance. Figure I-3 was then used to determine the shield wall thicknesses for the areas of the WHB shown in Table I-5.

Disposal Container Shield Wall Analysis

Figure I-2 Dose Rate vs Shield Wall Thickness at Various Centerline Distances

Disposal Container Shield Wall Analysis

Figure I-3. Shield Wall Thickness (cm) Required to Meet the Distance Dose Rates

Table I-5. Minimum Shield Wall Thickness for Single DC Areas of the WHB

Areas	DC Position	Ref.5.10 Att.2	Shield Wall Thickness
DC Trans. Load	18'0" (548.64 cm)	Figure 13	2'6.63" (77.8 cm)
WP Remediation	7'8.1" (233.85 cm)	Figure 15	2'9.03" (83.9 cm)
DC Handling Cell West Wall	2'8.46" (82.45 cm)	Figure 10	2'10.56" (87.78 cm)
DC Handling Cell South Wall	9'0" (274.32 cm)	Figure 10	2'8.7" (83.00 cm)
DC Load	8'6" (259.08 cm)	Figure 5	2'8.36" (83.3 cm)
DC Decon	8'6" (259.08 cm)	Figure 5	2'8.36" (83.3 cm)
DC Tilting Station	18'0" (548.64 cm)	Figure 10&11	2'6.63" (77.8 cm)

Canister Cask Preparation Cell Shield Wall Analysis**Table of Contents**

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Canister Cask Preparation Cell Shield Wall Analysis**1. Objective**

The objective of this section of the analysis is to estimate the east shield wall thickness Canister Cask Preparation Cell (CCPC). This shielding is required to reduce, to within acceptable limits, the worker radiological exposure to radiation originating from the CCPC operations.

Note: Due to the unqualified/unconfirmed input data used in this analysis the output/conclusions from this analysis can not be used as input into documents supporting procurement, fabrication, or construction unless they are controlled and tracked as TBV or TBD in accordance with NLP-3-15.

2. Summary

The objective of this section of the analysis is to estimate the east shield wall thickness Canister Cask Preparation Cell (CCPC). The data from section 4 indicates that 1'1.2" (33.53cm) concrete would provide adequate shielding in the CCPC to meet the design dose rate criteria of 0.25 mrem per hour (see Assumption 4.3.1.1), using the assumptions (see Assumption 4.3.1 and Assumption 4.3.2) and the conceptual design concepts presented in this attachment. At the current phase of the design it is recommended that the shielding wall thickness for the CCPC be 1'1.2" (33.53cm) of concrete.

3. Analysis Method

Radiation source terms for the analyses are established. Radiation source terms are an essential and necessary element of shielding and design analysis. For this analysis, the dose rate data (neutron and gamma) from cask Safety Analysis Reports (SAR) were used.

The analytical models describe the sources, configurations, and materials required for use in the analyses using QAD.

A figure showing a plan and elevation view of the source-shield configuration being assessed was then established. The figure shows the DC regions surrounded by the liner, air gap, inner and outer barriers dimensions and the distance from a concrete wall. The figure shows the materials used in each region.

Dose points in mrem per hour are then established at 1'0", 2'0" and 3'0" intervals in the concrete wall at the package mid-plane and at two lateral distances of 4' (121.92cm) and 8'0" (243.84cm). Iterative calculations for each of the three region of the source were then made to determine the shield thickness requirements. The dose rate results from the spent fuel assemble (SFA) active fuel region (AFR), top tie plate (TTP), top fission gas plenum (TFGP) and lower tie plate (LTP) regions were then tabulated.

Plots were generated for each iterative calculation. These plots are useful to assist the design team as equipment is sized and located in the facility. They are also useful in determining facility exposures.

Canister Cask Preparation Cell Shield Wall Analysis

4. Analysis

Note: The cask dose rate data (see Table II-1, column 2,3 and 4) and the diameter of the cask (see Table II-1, column 5) are from the respective Safety Analysis Reports (SAR) as shown in section 4.1 of the main report.

A cask is positioned as shown in Figure II-1 and Figure II-2. From Table II-1, the distance from the cask edge to the CCPC Inner Wall is calculated by:

- 1) Subtracting the diameter of the applicable cask (column 5 and column 6) from the cart diameter (column 7 and column 8).
- 2) The results from step 1 are then divided by 2 and are shown in column 9 and column 10.
- 3) From Figure II-1 CCPC Plan View and Table 4-1, the cart is 6'-0" (182.88cm) from the OG wall.
Add 72 to column 9 and place answer in column 11.
Add 182.88 to column 10 and place answer in column 12.
- 4) Convert column 11 from inches to ft and place answer in column 13.

Table II-1. CCPC Parameters												
Cask	Dose Rates (mrem per hour)			Diameter of Cask ^(Note 1)		Cart Width ^(Note 2)		Delta per Side		Distance to CCPC OG Inner Wall		
	Contact	1 m (Note 3)	2 m (Note 4)	(in)	(cm)	(in)	(cm)	(in)	(cm)	(in)	(cm)	(ft)
NAC-STC	42.77	19.48	7.96	87.60	222.50	132.00	335.28	22.20	56.39	94.20	239.27	7.85
TN-REG	38.60	23.30	9.30	90.25	229.24	132.00	335.28	20.88	53.02	92.88	235.90	7.74
NLI-10/24	38.20	11.00	5.48	96.00	243.84	132.00	335.28	18.00	45.72	90.00	228.60	7.50
HI-STAR	37.80	9.59	8.27	95.88	243.52	132.00	335.28	18.06	45.88	90.06	228.76	7.51

1 - Diameter from Ref 5.23, 5.24, 5.25 and 5.26

2 - From Reference 5.16, Figure 7.3-2 Cask Preparation/Decontamination Plus Air Lock Area (Area 1)

3 - From Cask Surface

4 - From Transport Vehicle

- 5) Next, the dose rate (see Table II-1, column 2, 3 & 4) for each cask type was plotted (at contact, 1 meter and 2 meters) and the results of the plot is shown in Figure II-3. A vertical line was then established at the distance (see Table II-1, column 13) to the CCPC Inner Wall (cask side) for each cask type. The dose rate at contact with the CCPC Inner Wall (Dose Point 1) could then be determined for each cask type listed. For example, the distance to CCPC Inner Wall (see Table II-1, column 13) for the TN-REG Cask is 6.04 feet. The dose rate at 6.04 feet for the TN-REG cask is 13.8 mrem per hour.

Canister Cask Preparation Cell Shield Wall Analysis

Figure II-1 Cask Preparation/Decontamination - Plan View

Canister Cask Preparation Cell Shield Wall Analysis

- 6) A plot (see Figure II-4) of dose rate in mrem per hour verses shield wall thickness in feet and centimeters (cm) was generated using the data from Table I-4 for the AFR, TTP, LTP and TFGP at a distance of 200 cm. All 4 regions were summed together at 1'-0" (30.48cm), 2'-0" (60.96cm) and 3'-0" (91.44cm) and the results were applied to the Figure II-4 below. This plot establishes the dose rate verses shield wall thickness (similar to Figure I-2).

Note: The slope of all the curves are the same.

- 7) The design dose rate limit (DDRL) of 0.25 mrem per hour was placed on the Figure II-4. A vertical line was then set where the DDRL and the curve intersected. This vertical line represents the concrete wall thickness required to meet the DDRL at contact with the CCPC inner wall.
- 8) Vertical lines (see Figure II-4) were then established at for each cask type at the dose rate determined in step 5 above. These vertical lines represent the dose rate on the inside of the CCPC OG inner wall for each cask type.
- 9) The amount of shielding required for each cask type is the difference between the DDRL vertical line and the specific cask is shown in Table II-2. For example the difference between the DDRL verticle and the TN-REG vertical line or $2.74 \text{ ft} - 1.68 \text{ ft} = 1'-0.72"$ (32.31 cm).

Table II-2. CCPC Shield Wall Thickness vs Cask Type					
Cask	Distance ^(a) to CCPC Inner Wall (ft)	Dose Rate ^(b) at CCPC Inner Wall Distance (mrem/hr).	DDRL Shield Wall Required Thickness (ft).	Cask Vertical Line Position on X-axis (ft).	Shield Thickness ^(c) Required to meet DDRL (ft).
NAC-STC	7.85	7.50	2.74	1.83	0.91
TN-REG	7.74	10.00	2.74	1.77	0.97
NLI-10/24	7.50	5.81	2.74	1.9	0.84
HI-Star 100	7.51	8.62	2.74	1.8	0.94

a: From Table II-1

b: Determined by linear interpolation.

c: Column 4 - Column 5

Canister Cask Preparation Cell Shield Wall Analysis

Figure II-2 Cask Preparation/Decontamination - Elevation

Canister Cask Preparation Cell Shield Wall Analysis

Figure II-3. Cask Dose Rate vs Distance

Canister Cask Preparation Cell Shield Wall Analysis

Figure II-3.4: CCPC Shield Wall Thickness (cm)

Figure II-4. CCPC Shield Wall Thickness (cm)

DHLW/DOE SNF Canister Staging Station Shield Wall Analysis**Table of Contents**

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DHLW/DOE SNF Canister Staging Station Shield Wall Analysis**1. Objective**

The objective of this section of the analysis is to estimate the shield wall thickness of the DHLW/DOE SNF Canister Staging Station (DHLWCSS). This shielding is required to reduce, to within acceptable limits, the worker radiological exposure to radiation originating from the DHLWCSS operations.

Note: Due to the unqualified/unconfirmed input data used in this attachment the output/conclusions from this attachment can not be used as input into documents supporting procurement, fabrication, or construction unless they are controlled and tracked as TBV or TBD in accordance with NLP-3-15.

2. Summary

The objective of this section of the analysis is to estimate the shield wall thickness of the DHLW/DOE SNF Canister Staging Station (DHLWCSS). The data from section 4 of Attachment III indicates that 3'-5" (104.14 cm) of concrete for the east and west walls of the DHLWCSS, 3'-0" (91.44 cm) of concrete for the north and south walls of the DHLWCSS, and 2'-0" (60.96 cm) for the DHLWCSS ceiling would provide adequate shielding to meet the design dose rate criteria of 0.25 mrem per hour (see section 4.3.1.1 of the main report) using the assumptions (see Assumption 4.3.1 and 4.3.5 of the main report) and conceptual design concepts presented in this attachment. This will provide flexibility for pursuing the layout of the DHLWCSS relative to occupied areas during further design.

3. Analysis Method

Radiation source terms for the analyses are established. For this analysis, the data input from section 4.1 of the main report for the DHLW Canister Staging Station will be used. The radiation source used for the DHLW Canister is the Hanford DHLW Canister. Neutron and secondary gamma source terms were not considered in areas outside of hot cells due to the attenuation effect of concrete (see Assumption 4.3.1.6).

Analytical models are developed and described in Section 7 to simulate the types of analyses performed. The analytical models describe the sources, configurations, and materials required for use in the analyses using QAD.

A figure showing a plan and elevation view of the source-shield configuration being assessed was then established. The figure shows the DHLW Canister in the Staging Station and the distance from a concrete wall. The figure shows the materials used in each region.

Generic QAD input listings are then generated based on the source-shield configuration and materials used. Dose points in mrem per hour are then established at intervals in the concrete wall at the package mid-plane and at 1'-0" (30.40 cm), 2'-0" (61.96 cm) and 3'-0" (91.44 cm) intervals in the concrete wall and at the package mid-plane and at two lateral distances of 4'-0" (121.92 cm) and 8'-0" (243.84 cm). Iterative calculations were then made to determine the shield thickness requirements. The dose rate results from DHLW Canister were then tabulated.

DHLW/DOE SNF Canister Staging Station Shield Wall Analysis

Plots were generated for each iterative calculation. These plots are useful to assist the design team as equipment is sized and located in the facility. They are also useful in determining facility exposures.

4. Analysis

4.1 Single DHLW Canister Dose Analysis

A preliminary dose assessment was performed using a single DHLW canister with no associated shielding at various distances (cm) and height (cm) from the base of the DHLW canister. A QAD run was performed using input parameters from section 4.1 of the main report. The results (see Ref 5.11, disk#6, file CANISO77.WPD for the QAD input/output file) from the QAD run are shown in Table III-1.

Table III-1 shows the dose rate in milli-rem (mrem) per hour at various vertical and horizontal distances from an unshielded DHLW Canister. This data was also plotted to produce Figure III-1 DHLW Canister Isodose Curves.

4.2 Defense High Level Waste Canister Staging Station (DHLWCSS)

Table III-1. DR Profile from Unshielded DHLW Canister

Refer to Attachment VIII, Waste Handling Building (functional layout) for orientation of the DHLWCSS with respect to the rest of the WHB. The DHLWCSS is west of the Canister Handling Area. DHLW canisters are positioned in a 2 by 10 array in a cell 37'-0" (1127.76 cm) long, 11'-0" (335.28 cm) wide with the east wall 22'-0" (670.56 cm) high and the south wall 15'-0" (457.2 cm). The DHLW canisters are positioned 2'-0" (60.96 cm) from the DHLWSS east and south walls as shown in Figure III-2. The DHLW Canisters are staged 2'-5" (73.66 cm) center-to-center.

Next, the geometry for the DHLW/DOE SNF Canister Staging Station (see Figure III-2) was established based upon the data input from section 4.1. This geometry was used as input for the QAD shielding program. A plan and elevation view was included. The manipulator maintenance

DHLW/DOE SNF Canister Staging Station Shield Wall Analysis

bay was also simulated. Dose points were established at the mid-plane of the east wall (see Figure III-2, Plan View). Dose points were established outside the manipulator maintenance bay in order to determine the dose rate from the DHLWCSS.

The dose rate was determined at each DHLW Canister Staging matrix location by translating or moving the source to each matrix location and then adding the dose rates together. The results of the QAD runs are shown in Table III-2. The QAD input/output results are from Reference 5.11, disk number 5&6, files:CANHLW1.WPD, CANHLW2.WPD, and CANHLW3.WPD.

The dose rate data from Table III-2, column 5 is shown in Figure III-4.

Figure III-1. DHLW Canister Isodose Curves

DHLW/DOE SNF Canister Staging Station Shield Wall Analysis

Figure III-2. DHLW/DOE SNF Canister Staging Station QAD Geometry

Table III-2. DHLWCSS QAD-VCGGP-A Dose Rate Results				
Location	X	Y	Z	Dose Rate (mrem/hour)
East Wall 22'-0" (670.56 cm)	331.47	-219.71	150	1.87e-02
	331.47	-189.23	150	2.14e+00
	331.47	-158.75	150	5.04e+02
	331.47	-128.27	150	2.31e+05
South Wall 15'-0" (457.2 cm)	-219.71	36.83	150	1.94e-03
	-189.23	36.83	150	2.27e-01
	-158.75	36.83	150	5.05e+01
	-128.27	36.83	150	1.86e+04
Manipulator Maintenance Bay	-829.31	36.83	853.44	5.31e+05
	-829.31	36.83	883.92	6.02e+05
	-829.31	36.83	914.4	6.64e+05
	-829.31	36.83	944.88	7.15e+05
DHLWCSS Ceiling	331.47	36.83	2011.68	4.01e+01
	331.47	36.83	2042.16	2.11e-01
	331.47	36.83	2072.64	2.14e-03

DHLW/DOE SNF Canister Staging Station Shield Wall Analysis**4.3 Discussion / Recommendations**

Based on the dose rate data presented in Table III-2 and as plotted in Figure III-4, the following changes are recommended (as explained below):

- 1) The east and west shield wall thickness is 3'-5" (104.14 cm).
 - 2) The south shield wall thickness is 3'-0" (91.44 cm).
 - 3) Consider the addition of a keyway to the DHLWCSS as shown in Figure II-3.
 - 4) Increase the height of the east, west and south walls to the maximum height possible.
 - 5) Establish a keyway access on the east end of the Maintenance Bay as shown in Figure III-3.
- or
- 6) Establish an enclosed vault design for the SNF Canister Staging Station with block shield or shield plugs over vault storage locations.

Figure III-3 Modified DHLWCSS Cell

DHLW/DOE SNF Canister Staging Station Shield Wall Analysis

The stated shield wall thickness above would ensure dose rates during any required maintenance activities in the Canister Transfer Cell would be ALARA. The change in the height of the east and south DHLWCSS shield walls would reduce the direct dose rate from the DHLWCSS at the Crane and Manipulator Maintenance Bays.

The Crane Bay Keyway as shown on Figure III-3 would:

- 1) Provide access to the Crane / Manipulator Maintenance Bay at a point farthest away from the DHLWCSS Keyway and DHLW Staging.
- 2) Reduce the requirement for a Crane Maintenance Bay shield door that extends the total width of the Canister Transfer Cell.

From Table III-2, the dose rate outside the Manipulator Maintenance Bay Door at 28'-0" (853.44 cm) is $4.09\text{e}+05$ mrem per hour from the exposed DHLWCSS matrix.

Figure III-4: DHLW Shield Wall Thickness vs Dose Rate

Canister Handling Area Shield Wall Analysis

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Canister Handling Area Shield Wall Analysis**1. Introduction**

The objective of this analysis is to estimate the west and south shield wall thickness for the Canister Handling Area (CHA). This shielding is required to reduce, to within acceptable limits, the worker radiological exposure to radiation originating from the CHA operations.

Note: Due to the unqualified/unconfirmed input data used in this analysis the output/conclusions from this analysis can not be used as input into documents supporting procurement, fabrication, or construction unless they are controlled and tracked as TBV or TBD in accordance with NLP-3-15.

2. Summary

The objective of this analysis was to estimate the shield wall thickness for the CHA. The graph of the data (see page 7 of Attachment IV) indicates that 3'10.46" (118 cm) of concrete would provide adequate shielding in the CHA to meet the design dose rate of 0.25 mrem per hour using the assumptions (see Assumption 4.3.1 and 4.3.6) and conceptual design concepts presented in this attachment. At the current phase of the design the recommendation is that the shielding wall thickness for the south and west walls be 4'0" (121.92 cm). This will provide flexibility for further design.

3. Analysis Method

Radiation source terms for the analyses are established. Radiation source terms are an essential and necessary element of shielding and design analysis.

Analytical models are developed to simulate the MPC transfer operation. The analytical models describe the sources, configurations, and materials required for use in the analyses using QAD.

A figure showing a plan and elevation view of the source-shield configuration being assessed was then established. The figure shows the MPC regions surrounded by the liner dimensions and the distance from a concrete wall. The figure shows the materials used in each region.

Generic QAD input listings are then generated based on the source-shield configuration and materials used. Dose points in mrem per hour are then established at 1'0" (30.40cm), 2'0" (61.96cm), and 3'0" (91.44cm) intervals in the concrete wall at the package mid-plane and at two lateral distances of 4' (121.92cm) and 8'0" (243.84cm). Iterative calculations for each of the three region of the source were then made to determine the shield thickness requirements. The dose rate results from the spent fuel assemble (SFA) active fuel region (AFR), top tie plate (TTP), top fission gas plenum (TFGP) and lower tie plate (LTP) regions were then tabulated.

Plots were generated for each iterative calculation. These plots are useful to assist the design team as equipment is sized and located in the facility. They are also useful in determining facility exposures.

Canister Handling Area Shield Wall Analysis**4. Analysis****4.1 Dose Rate Contribution From MPC Inside Cell**

The calculation performed in this section determined the dose rate in mrem per hour at contact with the outside of the CHA west wall in the operating corridor.

The dimensions used in this calculation were obtained from Section 4.1 - Input Parameters of the main report. Figure IV-1 contains a plan view of the Canister Handling Area (CHA). Figure IV-2 contains an elevation view of the CHA. Refer to Reference 5.9, pg I-1 and I-4, for orientation of the CHA with the rest of the Waste Handling Building (WHB). For orientation, the Carrier Bay of the WHB is orientated north and the DC Handling Cell oriented south. The CHA is located south of the Canister Cask Preparation/Decon Cell, north of the DC Handling Cell, and east of the DHLWCSS. This information was used to establish the geometry as shown in Figure IV-3.

Figure IV-1. Canister Handling Area (CHA) Plan View

Canister Handling Area Shield Wall Analysis

Figure IV-2. Canister Handling Area (CHA) Elevation View

Based upon Figures IV-1 and Figure IV-2 a QAD input geometry was established with the MPC centered in the universe and shield walls positioned on an XYZ axis (see Figure IV-3). The MPC is in the vertical position. The outside concrete shielding wall with an initial thickness of 152.4 cm was positioned 10'11.74" (334.62cm) (see Figure IV-3) from the MPC. The dose points (designated by a star symbol) were established at an operator position -17'0" (-518.16cm) and at 5'10.9" (180.09cm).

Note: The highest dose to the operator occurs during the time the MPC is being transferred from the MPC cask to the DC. That is why the operator is positioned at -17'0" (-518.16cm).

The geometry and source information (AFR) as described above were used as input to the QAD shielding program. Reference 5.11, disk #1, files: MPCA FR.WPD, contains the input and output listings for the QAD run as described above. The dose rate was determined at 1 foot intervals of concrete wall thickness. The dose rate results from the initial QAD fuel region run are listed in Table IV-1 column 3.

Next, the contribution of the dose rate from the smeared LTP was determined by positioning the LTP

Canister Handling Area Shield Wall Analysis

Figure IV-3 MPC Geometry for QAD Input

at the point of origin and updating the source information for the LTP in the QAD program. Reference 5.11, disk #1, files: MPCLOW.WPD contains the input and output listings for the LTP QAD computer runs as described above. The dose rate results are listed in Table IV-1, column 4.

Next, the contribution of the dose rate from the smeared TTP was determined by positioning the TTP at the point of origin and updating the source information for the TTP in the QAD

Canister Handling Area Shield Wall Analysis

program. Reference 5.11, disk #1, files: MPCUPPER.WPD contains the input and output listings for the TTP QAD run as described above. The dose rate was determined at 1 foot intervals of concrete thickness. The dose rate results from the QAD TTP run are listed in Table IV-1 column 5.

Next, the contribution of the dose rate from the smeared TFGP was determined by positioning the TFGP at the point of origin and updating the source information for the TFGP in the QAD program. Reference 5.11, disk #1, files: MPCPLEN.WPD contains the input and output listings for the TFGP QAD run as described above. The dose rate was determined at 1 foot intervals of concrete thickness. The dose rate results from the QAD TFGP run are listed in Table IV-1 column 6.

Table IV-1. QAD-vCGGP-A Results for Canister Handling Area						
Dose Point Coordinates: X, Y, Z (cm)	Thickness (cm)	Dose Rate (mrem / h) AFR	Dose Rate (mrem / h) LTP ^a	DoseRate (mrem / h) TTP ^b	DoseRate (mrem / h) TFGP	Total Dose Rate (mrem/h)
365.1, 0, 180.09	30.48	2.25e+04	1.05e+03	7.87e+02	3.19e+02	2.47e+04
395.58, 0, 180.09	60.96	3.86e+02	2.35e+01	1.43e+01	6.51e+00	4.30e+02
426.06, 0, 180.09	91.44	7.73e+00	4.86e-01	2.55e-01	1.27e-01	8.60e+00
456.54, 0, 180.09	121.92	1.66e-01	9.94e-03	4.67e-03	2.15e-03	1.83e-01
Dose Point from base of MPC.						
365.1, 0, -518.16	30.48	5.96e+00	6.70e+02	1.07e-02	4.03e-03	6.76e+02
395.58, 0, -518.16	60.96	1.09e-02	3.02e+00	1.28e-06	5.40e-07	3.03e+00
426.06, 0, -518.16	91.44	4.48e-05	1.72e-02	4.82e-10	2.21e-10	1.72e-02
456.54, 0, -518.16	121.92	4.47e-07	1.27e-04	4.48e-13	2.19e-13	1.27e-04

- a - Coordinates for X&Y are the same. Z was changed from 180.09 to -183.90 and changed from -518.16 to -514.35 to reflect movement of the origin to LTP.
- b - Coordinates for X&Y are the same. Z was unchanged in the top 4 rows of this column and was changed from -518.16 to -878.34 for the bottom 4 rows of this column.

The total dose rate verses concrete wall thickness in cm is plotted in Figure IV-4 below.

The recommended shield wall thickness is 4'0" (121.92 cm) based on the mid-plane dose rate.

Canister Handling Area Shield Wall Analysis

Figure IV-4. CHA - MPC Removal

Canister Handling Area Shield Wall Analysis

Appendix IV.1

IV.1 Calculation of Homogenized Material Data for SS316L

1. Obtain the elemental composition in weight percent (W_i) for SS316L from Reference 5.9 page 44.
2. Obtain the total density (D_T) of SS316L from Reference 5.8 page IV-1. The total density for SS316L = 7.9497 g/cc.
3. Calculate the partial density, D_i , of element I in the region:

$$D_i = (W_i * D_T) / 100$$

Table IV-2. Partial Density for SS316L		
Element	Weight Percent W_i	Partial Density D_i
C	0.03	0.0024
N	0.1	0.0079
Si	0.753	0.0599
P	0.045	0.0036
S	0.03	0.0024
Cr	17	1.3514
Mn	2	0.159
Fe	65.545	5.2106
Ni	12	0.954
Mo	2.5	0.1987
	100.003	7.9497

The partial densities calculated above will be used as input to QAD-vCGGP-A.

DC Staging Cell Shield Wall Analysis

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DC Staging Cell Shield Wall Analysis**1. Introduction**

The objective of this analysis is to estimate the shield wall thickness for the DC Staging Cell (DCSC). This shielding is required to reduce, to within acceptable limits, the worker radiological exposure to radiation originating from the DCSC operations.

Note: Due to the unqualified/unconfirmed input data used in this analysis the output/conclusions from this analysis can not be used as input into documents supporting procurement, fabrication, or construction unless they are controlled and tracked as TBV or TBD in accordance with NLP-3-15.

2. Summary

The objective of this analysis was to estimate the shield wall thickness for the DCSC. The graph of the data (see page 6 of this analysis) indicates that 2'8.67" (83.00 cm) of concrete would provide adequate shielding in the DCSC to meet the design dose rate criteria of 0.25 mrem per hour (see Assumption 4.3.1.1 of the main report) for a single DC using the assumptions (see Assumptions 4.3.1 and 4.3.4) and the conceptual design concepts presented in this attachment. At the current phase of the design the recommendation is that the shielding wall thickness for the DCSC be 3'0" (91.44 cm) of concrete.

3. Analysis Method

Radiation source terms for the analyses are established. Radiation source terms are an essential and necessary element of shielding and design analysis.

Analytical models are developed to simulate the DC staging operation. The analytical models describe the sources, configurations, and materials required for use in the analyses using QAD.

A figure showing a plan and elevation view of the source-shield configuration being assessed was then established. The figure shows the DC regions surrounded by the liner dimensions and the distance from a concrete wall. The figure shows the materials used in each region.

Generic QAD input listings are then generated based on the source-shield configuration and materials used. Dose points in mrem per hour are then established at 1'0" (30.40cm), 2'0" (61.96cm), and 3'0" (91.44cm) intervals in the concrete wall at the package mid-plane and at two lateral distances of 4' (121.92 cm) and 8'0" (243.84 cm). Iterative calculations for each of the three region of the source were then made to determine the shield thickness requirements. The dose rate results from the spent fuel assemble (SFA) active fuel region (AFR), top tie plate (TTP), top fission gas plenum (TFGP) and lower tie plate (LTP) regions were then tabulated.

Plots were generated for each iterative calculation. These plots are useful to assist the design team as equipment is sized and located in the facility. They are also useful in determining facility exposures.

DC Staging Cell Shield Wall Analysis**4. Analysis**

The calculation performed in this section determined the dose rate in mrem per hour from a single DC positioned as shown in Figure V-1

Refer to Attachment VIII, pg VIII-1 Waste Handling Building (functional layout) for orientation of the areas listed in Table V-1 with respect to the rest of the WHB. For orientation purposes, the Carrier Bay is considered north and the DC Transporter Airlock is considered south. The DCSC is located east of the DC Handling Cell.

Figure V-1 depicts the geometric model for the QAD DCSC calculations. The lower portion of Figure V-1 is an elevation view of the DCSC, showing a single DC with its dimensions and materials. The top portion of Figure V-1 is a plan view of half the DCHC, showing multiple DC's and their relative location. The spacing was obtained from Reference 5.9, pg II-12.

As shown in Figure V-1, the fuel assembly region is divided into five axial zones for the limiting B&W 15x15 Mark B fuel assembly type, including the TTP, TFGP, AFR, LFGP and LTP.

For this analysis, the QAD code was run for a single DC, using the model in Figure V-1. The dose rates were determined at 1'0" (30.96cm) intervals inside the 5'0" (152.4 cm) thick concrete shield wall midplane of the DC. The dose rates from all the QAD runs were then tabulated and dose rates summed.

4.1 Active Fuel Gamma Source Terms

The first step in the calculation was to determine the source spectrum for the DC-AFR. The DC-Source(s) used in the QAD are the same as the source used in the Attachment I.

The next step in the design process was to establish the geometry as shown in Figure V-1 for the AFR. The dimensions used in this calculation were obtained from Section 4.1 - Input Parameters of the main report. Using the source-shield geometry shown in Figure V-1, a QAD input was established with the DC positioned as shown in Figure V-1.

The dose rates were determined at 1'0" (30.96cm) intervals inside the 5'0" (152.4 cm) thick concrete shield wall midplane of the DC, at each DC staging location as shown in Figure V-1. The coordinates for the dose points are shown in column 1 of Table V-1. The dose rates from all the QAD runs were tabulated in Table V-1, column 2. The QAD input/output file for the AFR are located in Reference 5.11, disk #1, files: STAGEAFR.WPD.

Next, the contribution of the dose rate from the smeared TFGP was determined by positioning the TFGP at the point of origin and updating the source information for the TFG in the QAD program. Reference 5.11, disk #1, files: STAGPLEN.WPD contains the input/ output listings for the TFGP QAD run as described above. The dose rate was determined at 1 foot intervals of concrete thickness. The dose rate results from the QAD TFGP run are listed in Table IV-1 column 3.

DC Staging Cell Shield Wall Analysis

Figure V-1. QAD Model of DC Staging Cell

DC Staging Cell Shield Wall Analysis

Next, the contribution of the dose rate from the smeared TTP was determined by positioning the TTP at the point of origin and updating the source information for the TTP in the QAD program. Reference 5.11, disk #2, files: STAGEUP.WPD contains the input/output listings for the TTP QAD run as described above. The dose rate was determined at 1 foot intervals of concrete thickness. The dose rate results from the QAD TTP run are listed in Table V-1 column 4.

Next, the contribution of the dose rate from the smeared LTP was determined by positioning the LTP at the point of origin and updating the source information for the LTP in the QAD program. Reference 5.11, disk #2, files: STAGELOW.WPD contains the input/output listings for the LTP QAD computer runs as described above. The dose rate results are listed in Table V-1, column 5.

The dose rates were then summed and the results are shown in Table V-1, column 6.

The dose rate data verses shield wall thickness from Table V-1 was then plotted. A line at the DDRL of 0.25 was also included. The resultant plot is shown in Figure V-2.

Table V-1. Dose Rate in DC Staging Cell Operating Gallery					
Dose Point Coordinates (X, Y, Z= Midplane of DC)	Dose Rate (mrem per hour)				
	AFR	TGFP	TTP	LTP	Total
-350.52, 0	1.63e+02	1.38e+00	1.72e+00	4.83e+00	1.71e+02
-350.52, 365.76	1.81e+01	5.80e-01	5.93e-01	1.47e+00	2.08e+01
-350.52, 731.52	3.45e-01	3.10e-02	2.18e-02	5.43e-02	4.53e-01
-350.52, 1097.28	5.69e-03	8.31e-04	3.85e-04	1.12e-03	8.03e-03
-350.52, 1463.04	1.05e-04	1.78e-05	6.21e-06	1.93e-05	1.48e-04
-350.52, 1828.8	2.30e-06	3.62e-07	1.06e-07	3.26e-07	3.10e-06
-350.52, 2194.56	6.40e-08	7.38e-09	1.93e-09	5.55e-09	7.89e-08
Total from 7 DC's	1.81e+02	1.99e+00	2.34e+00	6.36e+00	1.92e+02
-381.00, 0	3.26e+00	2.55e-02	2.94e-02	9.30e-02	3.41e+00
-381.00, 365.76	1.26e-01	3.38e-03	3.21e-03	8.93e-03	1.41e-01
-381.00, 731.52	2.58e-04	1.76e-05	1.18e-05	3.19e-05	3.19e-04
-381.00, 1097.28	4.23e-07	3.05e-08	1.34e-08	4.21e-08	5.09e-07
-381.00, 1463.04	1.41e-09	3.70e-11	1.25e-11	4.02e-11	1.50e-09
-381.00, 1828.8	1.44e-11	3.03e-14	1.17e-14	3.59e-14	1.45e-11
-381.00, 2194.56	3.06e-13	4.29e-17	1.15e-17	3.24e-17	3.06e-13
Total from 7 DC's	3.39e+00	2.89e-02	3.26e-02	1.02e-01	3.55e+00
-411.48, 0	6.97e-02	4.97e-04	5.38e-04	1.84e-03	7.26e-02
-411.48, 365.76	1.29e-03	2.70e-05	2.43e-05	7.36e-05	1.41e-03
-411.48, 731.52	6.23e-07	2.18e-08	1.40e-08	4.07e-08	6.99e-07
-411.48, 1097.28	4.70e-10	3.95e-12	1.66e-12	5.52e-12	4.81e-10
-411.48, 1463.04	1.72e-12	4.25e-16	1.41e-16	4.67e-16	1.72e-12
-411.48, 1828.8	1.34e-14	3.87e-20	1.13e-20	3.46e-20	1.34e-14
-411.48, 2194.56	1.37e-16	3.08e-24	1.13e-24	2.62e-24	1.37e-16
Total from 7 DC's	7.10e-02	5.24e-04	5.62e-04	1.91e-03	7.40e-02
-350.52, 762.00	2.45e-01	2.33e-02	1.58e-02	3.96e-02	3.24e-01
-381.00, 762.00	1.50e-04	1.07e-05	6.82e-06	1.87e-05	1.86e-04
-411.48, 762.00	3.25e-07	1.10e-08	6.76e-09	2.00e-08	3.63e-07

DC Staging Cell Shield Wall Analysis

Figure V-2: DC Staging Cell - Dose Rate vs Shield Wall Thickness

Drying Vessels/Transfer Canal Shield Wall Analysis

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Drying Vessels/Transfer Canal Shield Wall Analysis**1. Introduction**

The objective of this analysis is to estimate the shield wall thickness between the SNF Drying Vessels and the Incline Transfer Canal. The shielding is required to meet the design dose rate criteria of 0.25 mrem per hour.

Note: Due to the unqualified/unconfirmed input data used in this analysis the output/conclusions from this analysis can not be used as input into documents supporting procurement, fabrication, or construction unless they are controlled and tracked as TBV or TBD in accordance with NLP-3-15.

2. Summary

The objective of this analysis was to estimate the shield wall thickness required between the SNF Drying Vessels and the Incline Transfer Canal. The data from Table VI-1 indicates that 5'0" (152.4 cm) of concrete would provide adequate shielding in the SNF Drying Vessels and the Incline Transfer Canal to meet the design dose rate of 0.25 mrem per hour (see Section 4.3.1.) using the assumptions (see Assumptions 4.3.1 and 4.3.3) and conceptual design concepts presented in this attachment. At this phase of the design the recommendation is that the shielding wall thickness between the SNF Drying Vessels and the Incline Transfer Canal be 5'0" (152.4 cm) of concrete.

3. Analysis Method

Radiation source terms for the analyses are established. Radiation source terms are an essential and necessary element of shielding and design analysis.

Analytical models are developed to simulate the MPC transfer operation. The analytical models describe the sources, configurations, and materials required for use in the analyses using QAD.

A figure showing a plan and elevation view of the source-shield configuration being assessed was then established. The figure shows the SFA Drying Vessel/Incline Transfer Canal. The figure shows the materials used in each region.

Generic QAD input listings are then generated based on the source-shield configuration and materials used. Dose points in mrem per hour are then established at 1'0" (30.40cm), 2'0" (61.96cm), and 3'0" (91.44cm) intervals in the concrete wall at the package mid-plane and at two lateral distances of 4' (121.92cm) and 8'0" (243.84cm). Iterative calculations for each of the three region of the source were then made to determine the shield thickness requirements. The dose rate results from the spent fuel assemble (SFA) active fuel region (AFR), top tie plate (TTP), top fission gas plenum (TFGP) and lower tie plate (LTP) regions were then tabulated.

Plots were generated for each iterative calculation. These plots are useful to assist the design team as equipment is sized and located in the facility. They are also useful in determining facility exposures.

Drying Vessels/Transfer Canal Shield Wall Analysis**4. Analysis**

The following calculations were performed to determine dose rate in mrem per hour at contact (1 cm) with the walls of the Incline Transfer Canal located in the Upper Assembly Handling Area. Refer to Attachment VIII, pg VIII-1 Waste Handling Building (functional layout) for orientation of the SNF Drying Vessels with the rest of the WHB. The SFP Drying Vessel are located in the Upper Assembly Handling which is south of the spent fuel pools and north of the Assembly DC Load and Decon Cells.

The AFR, TTP, TFGP, and LTP of the SFA Drying Vessel are taken to be homogeneous by region. Figure VI-1 contains a plan view of the SNF Drying Vessels and Incline Transfer Canal.

4.1 Dose Rate Contribution From the AFR, TTP, TFGP, and LTP Regions

The first step in the design process was to establish the geometry as shown in Figure VI-1. The dimensions used in this calculation were obtained from Section 4.1 - Input Parameters of the main report. An AFR cylindrical source 4'0.03" (122cm) o.d. and 11'9.73" (360 cm) in height, containing 21 PWR spent fuel assemblies was centered inside the SNF Drying Vessel 8'6" (259.08 cm) in diameter. Two SNF Drying Vessels are located 5'0" (152.4 cm) from the Incline Transfer Canal as shown in Figure VI-1. A dose point (designated by a star symbol) was established in the Incline Transfer Canal at the SNF Drying Vessel mid-plane and another dose point (designated by a star symbol) was established halfway between the two SNF Drying Vessels as shown in Figure VI-1.

The geometry and source information (AFR) as described above were used as input to the QAD shielding program. Reference 5.11, disk #2, files: VESAFR.WPD, contains the input/output listings for the QAD AFR run as described above. The dose rate was determined at 1 foot intervals of concrete wall thickness. The dose rate results from the initial QAD AFR computer run are listed in Table VI-1 column 2.

Next, the contribution of the dose rate from the smeared LTP was determined by positioning the LTP at the point of origin and updating the source information for the LTP in the QAD program. Reference 5.11, disk #2, files: VESLTP.WPD contains the input/output listing for the QAD LTP computer run as described above. The dose rate results are listed in Table VI-1, column 3.

Next, the contribution of the dose rate from the smeared TTP was determined by positioning the TTP at the point of origin and updating the source information for the TTP in the QAD program. Reference 5.11, disk #2, files: VESTTP.WPD contains the input/output listing for the QAD TTP computer run as described above. The dose rate was determined at 1 foot intervals of concrete thickness. The dose rate results from the QAD TTP run are listed in Table VI-1 column 4.

Next, the contribution of the dose rate from the smeared TFGP was determined by positioning the TFGP at the point of origin and updating the source information for the TFGP in the QAD program. Reference 5.11, disk #2, files: VESPLEN.WPD contains the input/output listing for the QAD TFGP computer run as described above. The dose rate was determined at 1 foot intervals of concrete thickness. The dose rate results from the QAD TFGP run are listed in Table VI-1 column 5.

Drying Vessels/Transfer Canal Shield Wall Analysis

Figure VI-1 SNF Drying Vessel Geometry

The summary of the QAD results for the AFR, TTP, TFGP and LTP are shown in Table VI-1, column 6. The dose point halfway between the two SNF Drying Vessels was multiplied by 2 to account for the symmetry of the geometry and the results are shown in Table VI-1, column 7.

Drying Vessels/Transfer Canal Shield Wall Analysis

Table VI-1. QAD Results for SNF Drying Vessel						
Dose Point Coordinates: X, Y, Z (cm)	Dose Rate (mrem / hour)					
	AFR	LTP	TTP	TFGP	Total Dose	Symmetry Dose
284.21, 0, 180.09	1.05e-02	9.89e-06	8.24e-12	1.19e-06	1.05e-02	N/A
Dose Point Halfway Between SNF Drying Vessels						
284.21, 207.01, 180.09	3.64e-06	3.60e-09	8.44e-15	5.05e-10	3.64e-06	7.29e-06

In conclusion, 5'0" (152.4 cm) of concrete between the SFA Drying Vessels and the Incline Transfer Canal is more than enough to satisfy the DDRL of 0.25 mrem per hour.

Disposal Container Handling Cell Shield Wall Analysis

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Disposal Container Handling Cell Shield Wall Analysis**1. Introduction**

The objective of this analysis is to estimate the shield wall thicknesses in the Disposal Container Handling Cell (DCHC) Welding Operating Gallery floor and walls. This shielding is required to reduce, to within acceptable limits, the worker radiological exposure to radiation originating from the DCHC operations.

Note: Due to the unqualified/unconfirmed input data used in this analysis the output/conclusions from this analysis can not be used as input into documents supporting procurement, fabrication, or construction unless they are controlled and tracked as TBV or TBD in accordance with NLP-3-15.

2. Summary

The objective of this analysis was to estimate the shield wall thicknesses at various locations in the Disposal Container Handling Cell (DCHC) Welding Operating Gallery floor and walls. The graph of the data (see page 7 of this analysis) indicates that 4'0" (121.92 cm) of concrete on the floor and side of the Welding Operating Gallery would provide adequate shielding in the DCHC to meet the design dose rate criteria of 0.25 mrem per hour (see Assumption 4.3.1.1 of the main report) for a single DC using the assumptions (see Assumption 4.3.1 and 4.3.8 of the main report) and conceptual design concepts presented in this attachment. At this phase of the design the recommendation is that the shielding wall thickness for the DCHC be 4'0" (121.92 cm) of concrete on the floor and walls.

3. Analysis Method

Radiation source terms for the analyses are established. Radiation source terms are an essential and necessary element of shielding and design analysis.

Analytical models are developed to simulate the DCHC operation. The analytical models describe the sources, configurations, and materials required for use in the analyses using QAD.

A figure showing a plan and elevation view of the source-shield configuration being assessed was then established. The figure shows the DC regions surrounded by the liner dimensions and the distance from a concrete wall. The figure shows the materials used in each region.

Generic QAD input listings are then generated based on the source-shield configuration and materials used. Dose points in mrem per hour are then established at 1'0" (30.40cm), 2'0" (61.96cm), and 3'0" (91.44cm) intervals in the concrete wall at the package mid-plane and overhead. Iterative calculations for each of the three region of the source were then made to determine the shield thickness requirements. The dose rate results from the spent fuel assemble (SFA) active fuel region (AFR), top tie plate (TTP), top fission gas plenum (TFGP) and lower tie plate (LTP) regions were then tabulated.

Plots were generated for each iterative calculation. These plots are useful to assist the design team as equipment is sized and located in the facility. They are also useful in determining facility exposures.

Disposal Container Handling Cell Shield Wall Analysis**4. Analysis****4.1 Single Disposal Container Model at Weld Station**

The calculation performed in this section determined the dose rate in mrem per hour from a single DC positioned in the weld station area as shown in Figure VII-1.

Refer to Attachment VIII, section 3.4 Waste Handling Building (functional layout) for orientation of the DCHC with respect to the rest of the WHB. For orientation purposes, the Carrier Bay is considered north and the DC Transporter Airlock is considered south. The DC Transporter Load and DC Transporter Airlock are located south of the DCHC. The WP Remediation Cell is south and west of the DCHC. The Electrical Distribution Room, el 100+ is located south of the DCHC. The Canister DC Load/Decon Cells are located north of the DCHC. The DC Tilting Station is located on the west side of the DCHC, north of the DC Transporter Load Area and east of the DC Staging Area.

4.2 Geometric Model

Figure VII-1 depicts the geometric model for the QAD calculations. The DC was initially positioned 1 cm from the concrete shield wall. For modeling purposes it will be assumed that the shielding walls contain no internal shielding materials such as the steel lattice support structure (see Assumption 4.3.1.2).

As per Assumption 4.3.8.1, it is assumed that the top outer lid of the DC has been removed to facilitate the welding on the inner barrier lid, thus reducing the available shielding between the DC sources and the Weld Operating Gallery.

4.3 Analysis

The next step in the design process was to establish the geometry as shown in Figure VII-1 for the AFR. The dimensions used in this calculation were obtained from Section 4.1 - Input Parameters of the main report. Using the source-shield geometry shown in Figure VII-1, a QAD input a vertical shield wall was placed on the south side of the DC Handling Cell at a centerline distance of 10'0" (304.8 cm) from the DC and a horizontal shield wall was placed 24'0" (731.52 cm) over the DC.

Dose points were placed at 1'0" (30.48 cm) intervals in the 5'0" (152.4 cm) operating gallery floor from 10'0" (304.8 cm) to -1'0" (30.48 cm) from DC centerline.

The geometry and source information (AFR) as described above were used as input to the QAD shielding program. Reference 5.11, disk #2, files: WELDAFR.WPD, contains the input/output listing for the QAD run as described above. The dose rate was determined at 1 foot intervals of concrete wall thickness. The dose rate results from the initial QAD AFR run are listed in Table VII-1, column 2 and column 6.

Disposal Container Handling Cell Shield Wall Analysis

Figure VII-1. DC Handling Cell - QAD Geometric Model

Disposal Container Handling Cell Shield Wall Analysis

Next, the contribution of the dose rate from the smeared TTP was determined by positioning the TTP at the point of origin and updating the source information for the TTP in the QAD program. Reference 5.11, disk #3, files: WELDUP.WPD contains the input/output listing for the QAD TTP run as described above. The dose rate was determined at 1 foot intervals of concrete thickness. The dose rate results from the QAD TTP run are listed in Table VII-1, column 3 and column 7.

Next, the contribution of the dose rate from the smeared TFGP was determined by positioning the TFGP at the point of origin and updating the source information for the TFGP in the QAD program. Reference 5.11, disk #3, files: WELDPLEN.WPD contains the input/output listing for the TFGP QAD run as described above. The dose rate was determined at 1 foot intervals of concrete thickness. The dose rate results from the QAD TFGP run are listed in Table VII-1, column 4 and column 8.

The dose rates from all the QAD runs were tabulated in Table VII-1. The first column shows the horizontal distance in centi-meters (cm) from the DC. Column 2 and column 6 are the dose rate in milli-rem-per-hour (mr/hr) from the AFR at 1'0" (30.48 cm) shield wall/floor intervals. Column 3 and column 7 are the dose rate in milli-rem-per-hour (mr/hr) from the TTP region at 1'0" (30.48 cm) shield wall/floor intervals. Column 4 and column 8 are the dose rate in milli-rem-per-hour (mr/hr) from the TFGP region at 1'0" (30.48 cm) shield wall/floor intervals. Column 5 and column 9 contains the summation of the dose rates from the AFR, TTP and TFGP regions at 1'0" (30.48 cm) shield wall/floor intervals.

From Table VII-1, the DC Handling Cell South wall that is 9'0" (274.32 cm) from the DC centerline will require 2'8.7" (83.00 cm) of concrete to reduce the dose rate in the Electrical Distribution Room to less than 0.25 mrem per hour on the cold side of the south DC Handling Cell wall.

From Figure VII-3.3, 4'0" (121.92 cm) of concrete in the floor and sides of the Welding Operator Gallery would be adequate to ensure operational exposure are ALARA and less than 0.25 mrem per hour and is judged ALARA for this stage of the design and will be used as input to a more detailed cost benefit analysis and will be performed later in accordance with CDA Key 090.

Disposal Container Handling Cell Shield Wall Analysis

Table VII-1 Dose Rate in Welding Operating Gallery

Dose Rate (mrem/hr) in Welding Operating Gallery Floor								
Distance (cm) from DC	0'0" Shield Wall Thickness				1'0" Shield Wall Thickness			
	AFR	TTP	TFGP	Total	AFR	TTP	TFGP	Total
-304.8	8.43E+01	1.77E+04	1.46E+03	1.92E+04	1.70E+00	2.64E+02	2.00E+01	2.86E+02
-274.32	2.78E+02	2.38E+04	2.31E+03	2.64E+04	4.28E+00	4.42E+02	3.88E+01	4.85E+02
-243.84	5.42E+02	3.17E+04	3.63E+03	3.59E+04	8.70E+00	6.96E+02	7.07E+01	7.76E+02
-213.36	1.04E+03	4.09E+04	5.44E+03	4.74E+04	1.92E+01	1.05E+03	1.22E+02	1.20E+03
-182.88	1.96E+03	5.08E+04	7.79E+03	6.06E+04	3.90E+01	1.54E+03	1.96E+02	1.77E+03
-152.4	3.42E+03	6.27E+04	1.03E+04	7.65E+04	6.99E+01	2.11E+03	2.90E+02	2.47E+03
-121.92	5.21E+03	7.50E+04	1.33E+04	9.35E+04	1.10E+02	2.86E+03	4.07E+02	3.38E+03
-91.44	7.37E+03	8.59E+04	1.65E+04	1.10E+05	1.69E+02	3.50E+03	5.43E+02	4.21E+03
-60.96	9.26E+03	9.63E+04	1.89E+04	1.25E+05	2.10E+02	4.15E+03	6.54E+02	5.01E+03
Distance (cm) from DC	2'0" Shield Wall Thickness				3'0" Shield Wall Thickness			
	AFR	TTP	TFGP	Total	AFR	TTP	TFGP	Total
-304.8	2.96E-02	3.47E+00	2.84E-01	3.78E+00	5.04E-04	4.91E-02	4.41E-03	5.40E-02
-274.32	6.50E-02	6.58E+00	6.17E-01	7.26E+00	1.32E-03	1.05E-01	1.04E-02	1.16E-01
-243.84	1.67E-01	1.20E+01	1.25E+00	1.35E+01	3.42E-03	2.07E-01	2.28E-02	2.34E-01
-213.36	3.82E-01	2.02E+01	2.39E+00	2.30E+01	8.39E-03	3.86E-01	4.61E-02	4.40E-01
-182.88	8.21E-01	3.34E+01	4.15E+00	3.84E+01	1.72E-02	6.75E-01	8.42E-02	7.77E-01
-152.4	1.45E+00	4.95E+01	6.65E+00	5.76E+01	3.19E-02	1.07E+00	1.43E-01	1.25E+00
-121.92	2.41E+00	7.33E+01	9.79E+00	8.55E+01	5.41E-02	1.66E+00	2.18E-01	1.93E+00
-91.44	3.69E+00	9.52E+01	1.37E+01	1.13E+02	8.15E-02	2.26E+00	3.14E-01	2.66E+00
-60.96	4.69E+00	1.16E+02	1.72E+01	1.38E+02	1.05E-01	2.80E+00	4.02E-01	3.31E+00
Distance (cm) from DC	4'0" Shield Wall Thickness				5'0" Shield Wall Thickness			
	AFR	TTP	TFGP	Total	AFR	TTP	TFGP	Total
-304.8	1.13E-05	7.71E-04	7.51E-05	8.57E-04	3.52E-07	1.35E-05	1.37E-06	1.52E-05
-274.32	3.20E-05	1.78E-03	1.86E-04	2.00E-03	9.24E-07	3.20E-05	3.54E-06	3.65E-05
-243.84	8.32E-05	3.69E-03	4.30E-04	4.21E-03	2.29E-06	6.98E-05	8.37E-06	8.05E-05
-213.36	1.86E-04	7.47E-03	9.05E-04	8.56E-03	5.22E-06	1.46E-04	1.80E-05	1.70E-04
-182.88	4.02E-04	1.34E-02	1.71E-03	1.55E-02	1.07E-05	2.68E-04	3.49E-05	3.14E-04
-152.4	7.46E-04	2.25E-02	3.02E-03	2.62E-02	1.92E-05	4.66E-04	6.29E-05	5.48E-04
-121.92	1.25E-03	3.55E-02	4.73E-03	4.14E-02	3.18E-05	7.42E-04	1.01E-04	8.75E-04
-91.44	1.88E-03	5.02E-02	6.82E-03	5.89E-02	4.77E-05	1.08E-03	1.45E-04	1.27E-03
-60.96	2.44E-03	6.84E-02	9.08E-03	8.00E-02	8.21E-05	4.82E-03	4.10E-04	5.31E-03

Disposal Container Handling Cell Shield Wall Analysis**DC Handling Operating Gallery -
Dose Rate @ Various Shield Wall Floor Thicknesses**

Figure VII-2. Dose Rate vs Distance for Various DC Operating Gallery Floor Shield Wall Thicknesses

Title: Waste Handling Building Shield Wall Analysis

Waste Handling Building - Drawing

LWR Radiological Database Outputs**Waste Handling Building Shield Wall Analysis - LWR Radiological Database Outputs:**

Physical Description Report For B&W 15 x 15 B&W Mark B pg IX-1 - IX-11
 Photons Report For PWR 48,086, 4.2% Enrichment and 10 Years Decay pg IX-12
 DHLW Photon Spectrum Per Canister vs. Decay Time pg IX-13

(1) **PHYSICAL DESCRIPTION REPORT:**
FOR B&W 15 X 15 B&W Mark B
OVERALL ASSEMBLY CHARACTERISTICS

EIA Assembly Code B1515B
 CDB Assembly Class B&W 15 X 15
 Initial Year of Use 1974
 Latest Discharge Year (to Dec. 31, 1989) 1989
 Total Number Discharged (to Dec. 31, 1989) 151
 Assembly Width (inches) 8.536
 Assembly Length (inches) 165.625
 with Control Rod Inserted..... (This was blank in the computer printout)
 including Holddown Device, etc..... (This was blank in the computer printout)
 Rod Pitch (inches) 0.568
 Total Assembly Weight (lbs) 1515.0
 Average Kilograms of Uranium (discharged fuel) 463.2
 Enrichment Range (% U235) 2.01-2.99
 Average Discharge Burnup (MWd/MTIHM) 27837
 Maximum Discharge Burnup (MWd/MTIHM) 35458
 Linear Heat Rating (KW/foot) 6.30

NOTE: Data for all blank fields are not available.

Comments:

Zircaloy clad fuel manufactured by Babcock & Wilcox for B&W 15 x 15 Class reactors. Mark B is a generic designation, encompassing several early B&W 15 x 15 Class fuels (Mark B2, Mark B3, and Mark B4) and is used when the specific assembly type is unknown. Additionally, there is likely fuel which should be designated Mark B1.

PHYSICAL DESCRIPTION REPORT:
FOR B&W 15 X 15 B&W Mark B

FUEL ASSEMBLY HARDWARE PARTS AND MATERIALS

Part Name	Parts/ Assembly	Weight(kg) / Assembly	Zone	Material Name	Material Fraction
SPACER-BOTTOM	1	1.3	BOTTOM	Inconel 718	1.00000
BOTTOM NOZZLE	1	8.16	BOTTOM	St.Steel CF3	1.00000
GRID SUPPORTS	7	0.64	IN CORE	Zircaloy-4	1.00000
GUIDE TUBES	16	8	IN CORE	Zircaloy-4	1.00000
HOLDDOWN SPRING	1	1.8	TOP	Inconel 718	1.00000
INSTRUMENT TUBE	1	0.64	IN CORE	Zircaloy-4	1.00000
LOWER NUT	16	0.15	BOTTOM	St.Steel CF3	1.00000
SPACER-PLENUM	1	1.04	GAS PLENUM	Inconel 718	1.00000
SPRING RETAINER	1	0.91	TOP	St.Steel CF3	1.00000
SPACER-INCORE	6	4.9	IN CORE	Inconel 718	1.00000
TOP NOZZLE	1	7.48	TOP	St.Steel CF3	1.00000

Title: Waste Handling Building Shield Wall Analysis**LWR Radiological Database Outputs**

UPPER NUT	16	0.51	TOP	St.Steel CF304L	1.00000
UPPER END PLUG	2	0.06	TOP	St.Steel CF304	1.00000

Drawing Numbers Associated With Assembly:

02-32958F
11-55248F

Comments:

The SFD hardware (and weights) listed for this assembly type is an estimate based on the SFD hardware for other similar assembly types.

PHYSICAL DESCRIPTION REPORT:
FOR B&W 15 X 15 B&W Mark B

FUEL ROD DESCRIPTION TABLE

Fuel Rod Positions per Assembly	225
Typical Number of Fueled Rods per Assembly	208
Rod Diameter (inches)	0.430
Rod Length (inches)	153.68
Active Fuel Length (inches)	141.8-144.0
Weight per Rod (lbs)	7.00
Clad Material	Zircaloy-4
Clad Thickness (inches)	0.0265
Clad Final Conditioning	SRA
Fuel-Clad Gap (inches)	0.0042
Fill Gas Used.	He
Initial Gas Pressure (psig)	415
Nitrogen Content of Fill Gas (percent)	3.0

PHYSICAL DESCRIPTION REPORT:
FOR B&W 15 X 15 B&W Mark B

FUEL ROD DESCRIPTION TABLE (continued)

Fuel Pellet Material	Uranium Oxide
Fuel Pellet Shape	Dished, Chamfered
Fuel Pellet Diameter (inches)	0.37
Fuel Pellet Length (inches)	0.70
Fuel Pellet Weight per Rod (lbs)	5.58
Open Porosity (percent)	< 1%
Grain Size (microns)	10-14
Fuel Density (% theoretical)	95
O/U Ratio	2-2.02:1
Smear Density (gr/cm3)	9.75
Spacer Pellet Material	Zircaloy-4
Spacer Pellet Length (inches).....(This was blank in the computer printout)	

LWR Radiological Database Outputs

Plenum Spring Material	St.Steel 302
Plenum Spring Weight per Assembly (lbs)	0.042
Plenum Length (inches)	11.720
Plenum Volume (cubic inches)	1.308

Radiological Report

Page 1 of original reference

for B&W 15 X 15 B&W Mark B

Pressurized Water Reactor

High Burnup (60000 Mwd) - 10 Years After Discharge

Isotope	Curies
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HOLDDOWN SPRING (Top Zone)

Weight: 1.80 Kg; Material(s): Inconel X-750(100.0%)

H 3	5.747e-07
BE 10	4.072e-10
C 14	6.653e-03
P 32	1.810e-11
S 35	5.671e-16
CL 36	8.282e-10
CA 45	2.148e-10
SC 46	1.114e-14
V 50	4.616e-17
CR 51	3.250e-38
MN 54	3.137e-04
FE 55	1.048e+00
FE 59	3.714e-25
CO 58	3.290e-14
CO 60	6.892e+01
NI 59	7.380e-02
NI 63	1.204e+01
ZN 65	7.080e-08
Y 90	3.259e-10
Y 91	1.562e-26
ZR 93	3.063e-08
ZR 95	2.279e-24
NB 93M	1.361e-08
NB 94	5.202e-03
NB 95	5.060e-24
TC 98	2.461e-20
TC 99	1.625e-13
Totals	8.209e+01

Totals
8.209E+01

SPRING RETAINER (Top Zone)

Weight: 0.91 Kg; Material(s): St.Steel CF3M(100.0%)*

C 14	3.363e-03
MN 54	1.629e-03
FE 55	2.674e+00

*For St.Steel CF3M, radiological characteristics are based on St.Steel 304.2

LWR Radiological Database Outputs

Radiological Report (continued) Page 2 of original reference
 for B&W 15 X 15 B&W Mark B
 Pressurized Water Reactor
 High Burnup (60000 MWd) - 10 Years After Discharge

Isotope	Curies
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SPRING RETAINER (Top Zone)

Weight: 0.91 Kg; Material(s): St.Steel CF3M(100.0%)*

CO 60	4.326e+00
NI 59	4.611e-03
NI 63	7.524e-01
NB 94	2.929e-05
TC 99	9.149e-16

Totals	7.763E+00
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TOP NOZZLE (Top Zone)

Weight: 7.48 Kg; Material(s): St.Steel CF3M(100.0%)*

C 14	2.765e-02
MN 54	1.339e-02
FE 55	2.198e+01
CO 60	3.556e+01
NI 59	3.790e-02
NI 63	6.185e+00
NB 94	2.407e-04
TC 99	7.520e-15

Totals	6.381E+01
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UPPER NUT (Top Zone)

Weight: 0.51 Kg; Material(s): St.Steel 304L(100.0%)*

C 14	1.885e-03
MN 54	9.132e-04
FE 55	1.499e+00
CO 60	2.425e+00
NI 59	2.584e-03
NI 63	4.217e-01
NB 94	1.641e-05
TC 99	5.127e-16
Tot als	4.350e+00

UPPER END PLUG (Top Zone)

Weight: 0.06 Kg; Material(s): St.Steel 304(100.0%)

For St.Steel CF3M, radiological characteristics are based on St.Steel 304.

Radiological Report (continued) Page 3 of original reference

for B&W 15 X 15 B&W Mark B

Pressurized Water Reactor

High Burnup (60000 MWd) - 10 Years After Discharge

Isotope	Curies
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UPPER END PLUG (Top Zone)

Weight: 0.06 Kg; Material(s): St.Steel 304(100.0%)

LWR Radiological Database Outputs

C 14	2.218e-04
MN 54	1.074e-04
FE 55	1.763e-01
CO 60	2.853e-01
NI 59	3.040e-04
NI 63	4.961e-02
NB 94	1.931e-06
TC 99	6.032e-17
Tot als	5.118e-01

SPACER-PLENUM (Gas Plenum Zone)

Weight: 1.04 Kg; Material(s): Inconel 718(100.0%)

C 14	7.688e-03
FE 55	2.005e+00
CO 60	5.764e+01
NI 59	6.139e-02
NI 63	1.002e+01
NB 94	3.711e-02
TC 99	1.552e-04
Tot als	6.977e+01

GRID SUPPORTS (In Core Zone)

Weight: 0.64 Kg; Material(s): Zircaloy-4(100.0%)

C 14	1.457e-03
FE 55	5.996e-02
CO 60	3.715e-01
NI 59	7.269e-06
NI 63	1.231e-03
SR 90	1.468e-05
ZR 93	1.321e-03
NB 93M	5.871e-04
NB 94	2.471e-04
TC 99	9.739e-08

*For St.Steel CF3M, radiological characteristics are based on St.Steel 304.

Radiological Report (continued)

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for B&W 15 X 15 B&W Mark B

Pressurized Water Reactor

High Burnup (60000 MWD) - 10 Years After Discharge

Isotope	Curies
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GRID SUPPORTS (In Core Zone)

Weight: 0.64 Kg; Material(s): Zircaloy-4(100.0%)

SN 119M	6.328e-04
SN 121M	2.360e-03
SB 125	5.393e-01
TE 125M	1.316e-01
I 129	1.046e-15
Totals	1.110e+00

GUIDE TUBES (In Core Zone)

LWR Radiological Database Outputs

Weight: 8.00 Kg; Material(s): Zircaloy-4(100.0%)

C 14	1.821e-02
FE 55	7.495e-01
CO 60	4.644e+00
NI 59	9.087e-05
NI 63	1.539e-02
SR 90	1.835e-04
ZR 93	1.651e-02
NB 93M	7.338e-03
NB 94	3.089e-03
TC 99	1.217e-06
SN 119M	7.910e-03
SN 121M	2.950e-02
SB 125	6.741e+00
TE 125M	1.645e+00
I 129	1.307e-14
Tot als	1.388e+01

INSTRUMENT TUBE (In Core Zone)

Weight: 0.64 Kg; Material(s): Zircaloy-4(100.0%)

C 14	1.457e-03
FE 55	5.996e-02
CO 60	3.715e-01
NI 59	7.269e-06
NI 63	1.231e-03

*For St.Steel CF3M, radiological characteristics are based on St.Steel 304.

Radiological Report (continued)

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for B&W 15 X 15 B&W Mark B

Pressurized Water Reactor

High Burnup (60000 Mwd) - 10 Years After Discharge

Isotope Curies

INSTRUMENT TUBE (In Core Zone)

Weight: 0.64 Kg; Material(s): Zircaloy-4(100.0%)

SR 90	1.468e-05
ZR 93	1.321e-03
NB 93M	5.871e-04
NB 94	2.471e-04
TC 99	9.739e-08
SN 119M	6.328e-04
SN 121M	2.360e-03
SB 125	5.393e-01
TE 125M	1.316e-01
I 129	1.046e-15
Totals	1.110e+00

SPACER-INCORE (In Core Zone)

Weight: 4.90 Kg; Material(s): Inconel 718(100.0%)

C 14	1.811e-01
FE 55	4.723e+01
CO 60	1.358e+03

LWR Radiological Database Outputs

NI 59	1.446e+00
NI 63	2.360e+02
NB 94	8.743e-01
TC 99	3.657e-03

Totals	1.644E+03
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SPACER-BOTTOM (Bottom Zone)

Weight: 1.30 Kg; Material(s): Inconel 718(100.0%)

C 14	9.610e-03
FE 55	2.506e+00
CO 60	7.205e+01
NI 59	7.674e-02
NI 63	1.252e+01
NB 94	4.639e-02
TC 99	1.940e-04

Totals	8.722e+01
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*For St.Steel CF3M, radiological characteristics are based on St.Steel 304.

Radiological Report (continued) Page 6 of original reference
 for B&W 15 X 15 B&W Mark B
 Pressurized Water Reactor
 High Burnup (60000 MWD) - 10 Years After Discharge

Isotope	Curies
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BOTTOM NOZZLE (Bottom Zone)

Weight: 8.16 Kg; Material(s): St.Steel CF3M(100.0%)*

C 14	6.032e-02
MN 54	2.922e-02
FE 55	4.796e+01
CO 60	7.759e+01
NI 59	8.269e-02
NI 63	1.349e+01
NB 94	5.252e-04
TC 99	1.641e-14

Totals	1.392e+02
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LOWER NUT (Bottom Zone)

Weight: 0.15 Kg; Material(s): St.Steel 304(100.0%)

C 14	1.109e-03
MN 54	5.372e-04
FE 55	8.816e-01
CO 60	1.426e+00
NI 59	1.520e-03
NI 63	2.481e-01
NB 94	9.655e-06
TC 99	3.016e-16

Totals	2.559e+00
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*For St.Steel CF3M, radiological characteristics are based on St.Steel 304.

Spent Fuel Disassembly Hardware

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for

B&W 15 X 15 B&W Mark B

Top Zone

Units are Curies

LWR Radiological Database Outputs

Weight: 10.76 kg.

Reduced Volume: 0.0013341 cu m.

High Burnup (60000 MWD)

10 Years After Discharge

Totals by isotope for Top Zone:

Isotope	Curies	Curies/kg	Curies/cu m
H 3	6.242e-06	5.801e-07	4.679e-03
BE 10	4.472e-09	4.156e-10	3.352e-06
C 14	3.977e-02	3.696e-03	2.981e+01
P 32	6.422e-10	5.968e-11	4.814e-07
S 35	1.267e-14	1.178e-15	9.497e-12
CL 36	1.850e-08	1.719e-09	1.387e-05
CA 45	2.148e-10	1.996e-11	1.610e-07
SC 46	1.114e-14	1.035e-15	8.350e-12
V 50	3.380e-16	3.141e-17	2.534e-13
CR 51	2.383e-37	2.215e-38	1.786e-34
MN 54	1.636e-02	1.520e-03	1.226e+01
FE 55	2.738e+01	2.545e+00	2.052e+04
FE 59	8.408e-24	7.814e-25	6.302e-21
CO 58	5.313e-14	4.938e-15	3.983e-11
CO 60	1.115e+02	1.036e+01	8.358e+04
NI 59	1.192e-01	1.108e-02	8.935e+01
NI 63	1.944e+01	1.807e+00	1.457e+04
ZN 65	7.165e-08	6.659e-09	5.371e-05
Y 90	3.439e-10	3.196e-11	2.578e-07
Y 91	1.562e-26	1.452e-27	1.171e-23
ZR 93	3.233e-08	3.005e-09	2.423e-05
ZR 95	2.279e-24	2.118e-25	1.708e-21
NB 93M	1.437e-08	1.336e-09	1.077e-05
NB 94	5.490e-03	5.102e-04	4.115e+00
NB 95	5.340e-24	4.963e-25	4.003e-21
TC 98	2.598e-20	2.414e-21	1.947e-17
TC 99	1.715e-13	1.594e-14	1.286e-10

Class C Limit Calculations:

Isotope	Curies/cu m	Class C Limit	Ratio
C 14	2.981e+01	80	0.37
NI 59	8.935e+01	200	0.45
NB 94	4.115e+00	0.2	20.58
TC 99	1.286e-10	3	< 0.01
NI 63	1.457e+04	7000	2.08
Totals	2.348e+01		

Curies in Top Zone by Material:

Inconel X-750	St.Steel 304
8.210E+01	7.642E+01

Top Zone Totals:

in Curies	Curies/kg	Curies/cu m
1.585E+02	1.473E+01	1.188E+05
Spent Fuel Disassembly Hardware		Page 8 of original reference
for		

B&W 15 X 15 B&W Mark B

Gas Plenum Zone

Weight: 1.04 kg.

Reduced Volume: 0.0001270 cu m.

Units are Curies

High Burnup (60000 MWD)

10 Years After Discharge

LWR Radiological Database Outputs

Totals by isotope for Gas Plenum Zone:

Isotope	Curies	Curies/kg	Curies/cu m
C 14	7.688e-03	7.392e-03	6.054e+01
FE 55	2.005e+00	1.928e+00	1.579e+04
CO 60	5.764e+01	5.542e+01	4.539e+05
NI 59	6.139e-02	5.903e-02	4.834e+02
NI 63	1.002e+01	9.635e+00	7.891e+04
NB 94	3.711e-02	3.568e-02	2.922e+02
TC 99	1.552e-04	1.492e-04	1.222e+00

Class C Limit Calculations:

Isotope	Curies/cu m	Class C Limit	Ratio
C 14	6.054e+01	80	0.76
NI 59	4.834e+02	200	2.42
NB 94	2.922e+02	0.2	1461.21
TC 99	1.222e+00	3	0.41
NI 63	7.891e+04	7000	11.27
Totals	1.476e+03		

Curies in Gas Plenum Zone by Material:

Inconel 718
6.978E+01

Gas Plenum Zone Totals:

in Curies	Curies/kg	Curies/cu m
6.978E+01	6.710E+01	5.495E+05
Spent Fuel Disassembly Hardware		Page 9 of original reference
for		

B&W 15 X 15 B&W Mark B

In Core Zone

Weight: 14.18 kg.

Reduced Volume: 0.0020129 cu m.

Units are Curies

High Burnup (60000 MWD)

10 Years After Discharge

Totals by isotope for In Core Zone:

Isotope	Curies	Curies/kg	Curies/cu m
C 14	2.022e-01	1.426e-02	1.005e+02
FE 55	4.810e+01	3.392e+00	2.390e+04
CO 60	1.363e+03	9.612e+01	6.771e+05
NI 59	1.446e+00	1.020e-01	7.184e+02
NI 63	2.360e+02	1.664e+01	1.172e+05
SR 90	2.129e-04	1.501e-05	1.058e-01
ZR 93	1.916e-02	1.351e-03	9.518e+00
NB 93M	8.514e-03	6.004e-04	4.230e+00
NB 94	8.779e-01	6.191e-02	4.361e+02
TC 99	3.658e-03	2.580e-04	1.817e+00
SN 119M	9.176e-03	6.471e-04	4.559e+00
SN 121M	3.422e-02	2.413e-03	1.700e+01
SB 125	7.819e+00	5.514e-01	3.884e+03
TE 125M	1.908e+00	1.346e-01	9.479e+02
I 129	1.517e-14	1.070e-15	7.536e-12

Class C Limit Calculations:

Isotope	Curies/cu m	Class C Limit	Ratio
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LWR Radiological Database Outputs

C 14	1.005e+02	80	1.26
NI 59	7.184e+02	200	3.59
NB 94	4.361e+02	0.2	2180.66
TC 99	1.817e+00	3	0.61
NI 63	1.172e+05	7000	16.75
SR 90	1.058e-01	7000	< 0.01
I 129	7.536e-12	0.08	< 0.01
Totals	2.202e+03		

Curies in In Core Zone by Material:

Zircaloy-4	Inconel 718
1.608E+01	1.643E+03

In Core Zone Totals:

in Curies	Curies/kg	Curies/cu m
1.659E+03	1.170E+02	8.242E+05
Spent Fuel Disassembly Hardware		Page 10 of original reference
for		
B&W 15 X 15 B&W Mark B		

Bottom Zone

Weight: 9.61 kg.

Reduced Volume: 0.0011949 cu m.

Units are Curies

High Burnup (60000 MWd)

10 Years After Discharge

Totals by isotope for Bottom Zone:

Isotope	Curies	Curies/kg	Curies/cu m
C 14	7.104e-02	7.392e-03	5.945e+01
MN 54	3.096e-02	3.222e-03	2.591e+01
FE 55	5.135e+01	5.343e+00	4.297e+04
CO 60	1.510e+02	1.571e+01	1.264e+05
NI 59	1.609e-01	1.674e-02	1.347e+02
NI 63	2.626e+01	2.733e+00	2.198e+04
NB 94	4.693e-02	4.883e-03	3.928e+01
TC 99	1.940e-04	2.019e-05	1.624e-01

Class C Limit Calculations:

Isotope	Curies	Curies/kg	Curies/cu m
C 14	5.945e+01	80	0.74
NI 59	1.347e+02	200	0.67
NB 94	3.928e+01	0.2	196.38
TC 99	1.624e-01	3	0.05
NI 63	2.198e+04	7000	3.14
Totals	2.009e+02		

Curies in Bottom Zone by Material:

Inconel 718	St. Steel 304
8.722E+01	1.417E+02

Bottom Zone Totals:

in Curies	Curies/kg	Curies/cu m
2.289E+02	2.382E+01	1.916E+05
Spent Fuel Disassembly Hardware		Page 11 of original reference
for		
B&W 15 X 15 B&W Mark B		

All Zones

Units are Curies

LWR Radiological Database Outputs

Weight: 35.59 kg.

High Burnup (60000 MWD)

Reduced Volume: 0.0046689 cu m.

10 Years After Discharge

Totals by isotope for All Zones:

Isotope	Curies	Curies/kg	Curies/cu m
H 3	4.499e-05	1.264e-06	9.636e-03
BE 10	3.086e-08	8.671e-10	6.610e-06
C 14	3.207e-01	9.011e-03	6.869e+01
P 32	2.162e-09	6.075e-11	4.631e-07
S 35	6.666e-14	1.873e-15	1.428e-11
CL 36	9.732e-08	2.734e-09	2.084e-05
CA 45	2.275e-09	6.392e-11	4.873e-07
SC 46	1.180e-13	3.316e-15	2.527e-11
V 50	2.707e-15	7.606e-17	5.798e-13
CR 51	1.864e-36	5.237e-38	3.992e-34
MN 54	7.145e-02	2.008e-03	1.530e+01
FE 55	1.288e+02	3.619e+00	2.759e+04
FE 59	4.045e-23	1.137e-24	8.664e-21
CO 58	7.969e-13	2.239e-14	1.707e-10
CO 60	1.683e+03	4.729e+01	3.605e+05
NI 59	1.789e+00	5.027e-02	3.832e+02
NI 63	2.917e+02	8.196e+00	6.248e+04
ZN 65	4.392e-06	1.234e-07	9.407e-04
SR 90	2.129e-04	5.982e-06	4.560e-02
Y 90	2.130e-04	5.985e-06	4.562e-02
Y 91	7.751e-19	2.178e-20	1.660e-16
ZR 93	1.916e-02	5.384e-04	4.104e+00
ZR 95	1.438e-14	4.040e-16	3.080e-12
NB 93M	8.514e-03	2.392e-04	1.824e+00
NB 94	9.674e-01	2.718e-02	2.072e+02
NB 95	3.193e-14	8.972e-16	6.839e-12
TC 98	2.309e-09	6.488e-11	4.946e-07
TC 99	4.007e-03	1.126e-04	8.582e-01
SN 119M	9.176e-03	2.578e-04	1.965e+00
SN 121M	3.422e-02	9.615e-04	7.329e+00
SB 125	7.819e+00	2.197e-01	1.675e+03
TE 125M	1.908e+00	5.361e-02	4.087e+02
I 129	1.517e-14	4.262e-16	3.249e-12

Class C Limit Calculations:

Isotope	Curies/cu m	Class C Limit	Ratio
C 14	6.869e+01	80	0.86
NI 59	3.832e+02	200	1.92
NB 94	2.072e+02	0.2	1036.01
TC 99	8.582e-01	3	2.900e-01
NI 63	6.248e+04	7000	8.93
SR 90	4.560e-02	7000	< 0.01
I 129	3.249e-12	0.08	< 0.01
Totals	1.048e+03		

Curies in All Zones by Material:

Zircaloy-4	Inconel 718	Inconel X-750	St. Steel 304
1.608E+01	1.801E+03	8.210E+01	2.182E+02

All Zones Totals:

in Curies	Curies/kg	Curies/cu m
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LWR Radiological Database Outputs

2.117E+03

5.949E+01

4.535E+05

(2) LWR Radiological DATABASE**PHOTONS REPORT**

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 10 YEARS

The data is shown in Photons per second/MTIHM

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	2.728E+15	24.27%
2.500E-02	5.787E+14	5.15%
3.750E-02	7.125E+14	6.34%
5.750E-02	5.396E+14	4.80%
8.500E-02	3.229E+14	2.87%
1.250E-01	3.145E+14	2.80%
2.250E-01	2.640E+14	2.35%
3.750E-01	1.257E+14	1.12%
5.750E-01	4.798E+15	42.69%
8.500E-01	4.671E+14	4.16%
1.250E+00	3.849E+14	3.42%
1.750E+00	4.879E+12	0.04%
2.250E+00	7.693E+10	0.00%
2.750E+00	5.940E+09	0.00%
3.500E+00	7.264E+08	0.00%
5.000E+00	2.900E+07	0.00%
7.000E+00	3.343E+06	0.00%
9.500E+00	3.841E+05	0.00%
TOTAL	*1.124E+16	100.01%*

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/age combination you specified. Percentages have been calculated from this interpolated value and may not add up to 100 percent in all cases.

(3) PHOTON SPECTRUM PER CANISTER VS. DECAY TIME
SITE: Hanford **WASTE TYPE: Immobilized (in canisters)** **MAXIMUM RADIOACTIVITY**

ISOTOPES:	Total isotopes	UNITS: Photons/sec				
EMEAN	0YR	1YR	2YR	5YR	10YR	
1.000E-02	2.314E+15	1.692E+15	1.409E+15	1.141E+15	9.906E+14	
2.500E-02	5.234E+14	3.826E+14	3.155E+14	2.462E+14	2.070E+14	
3.750E-02	5.651E+14	4.030E+14	3.300E+14	2.629E+14	2.269E+14	
5.750E-02	4.628E+14	3.343E+14	2.763E+14	2.235E+14	1.955E+14	
8.500E-02	3.046E+14	2.095E+14	1.671E+14	1.301E+14	1.123E+14	
1.250E-01	3.157E+14	1.862E+14	1.307E+14	8.799E+13	7.428E+13	
2.250E-01	2.540E+14	1.753E+14	1.401E+14	1.089E+14	9.410E+13	
3.750E-01	1.396E+14	9.581E+13	7.470E+13	5.282E+13	4.195E+13	
5.750E-01	2.064E+15	1.939E+15	1.853E+15	1.684E+15	1.485E+15	
8.500E-01	6.904E+13	4.814E+13	3.576E+13	1.878E+13	1.046E+13	
1.250E+00	2.621E+13	1.703E+13	1.249E+13	7.699E+12	5.247E+12	
1.750E+00	2.370E+12	1.273E+12	7.714E+11	3.577E+11	2.535E+11	
2.250E+00	8.661E+12	3.580E+12	1.482E+12	1.064E+11	1.436E+09	
2.750E+00	4.721E+10	2.304E+10	1.130E+10	1.362E+09	4.211E+07	
3.500E+00	5.187E+09	2.608E+09	1.311E+09	1.667E+08	5.476E+06	
5.000E+00	7.682E+04	7.387E+04	7.112E+04	6.353E+04	5.268E+04	
7.000E+00	8.804E+03	8.463E+03	8.146E+03	7.271E+03	6.020E+03	
9.500E+00	1.008E+03	9.686E+02	9.322E+02	8.317E+02	6.879E+02	