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AIRBORNE EXPRESS

21G-04-0045
GOV-01-55-04
ACF-04-0074

March 19, 2004

Director
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

References:
1) Docket No. 70-143; SNM License 124
2) Letter from B.M. Moore to NRC, License Amendment Request for the Oxide Conversion Building and Effluent Processing Building at the BLEU Complex, dated October 23, 2003 (21G-03-0277)
3) NRC Licensing Review to Support License Amendment Request for the Oxide Conversion Building and Effluent Processing Building, conducted on February 10-11, 2004

Subject: Commitment Letter to Address NRC Licensing Review Questions Pertaining to Radiation Safety and Emergency Preparedness for the OCB and EPB

Dear Sir:

Nuclear Fuel Services, Inc. (NFS) hereby submits responses to questions raised during the licensing review that was conducted in Rockville, Maryland (Reference 3). These responses reflect the discussions with your staff during the licensing review that was conducted in the referenced meeting.

As noted in the attached responses, safety basis documents supporting this licensing review for the Oxide Conversion Building (OCB) and Effluent Processing Building (EPB) will be updated as necessary.

Public
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If you or your staff have any questions, require additional information, or wish to discuss this, please contact me, or Mr. Rik Droke, Licensing and Compliance Director at (423) 743-1741. Please reference our unique document identification number (21G-04-0045) in any correspondence concerning this letter.

Sincerely,

NUCLEAR FUEL SERVICES, INC.

B. M Moore

B. Marie Moore
Vice President
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JSK/lsn
Attachment

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Attachment

NRC Licensing Review Questions Pertaining to Radiation Safety and Emergency Preparedness for the OCB and EPB

NRC Question (1): In accordance with 10CFR Part 70.64(a)(6), "Emergency capability" describe how the design for the OCB provides for emergency capability to maintain control of licensed material and hazardous chemicals produced from licensed material, evacuation of on-site personnel, and onsite emergency facilities and services that facilitate the use of available offsite services.

NFS Response: Information contained in Section 6.2, *Baseline Design Criteria for Credited IROFS,* in the ISA Summary for the Oxide Conversion and Effluent Processing Buildings was incorporated in the design of these two facilities. As noted (see pages 344-347) in the referenced ISA Summary, these design basis requirements ensure adequate emergency capabilities are instituted within the two facilities in accordance with each of the provisions specified in 10 CFR 70.64(a)(6).

NRC Question (2): In accordance with 10 CFR Part 20.1406, "minimization of contamination" describe how the OCB was designed and procedures for operations will minimize contamination of the facility and the environment and minimize the generation of radioactive waste.

NFS Response: NFS has incorporated into the design basis of the OCB and EPB features that serve to minimize contamination as specified in 10 CFR 20.1406. The waste/contamination features and commitments have been integrated into the NFS ALARA Program as specified in Section 3.1.1 *ALARA Policy* of SNM-124.

NRC Question (3): So that we may audit your implementation of the radiation program requirements in 10CFR Part 20.110, please provide:

> **(A) source term and dose rate profiles for the OCB for both normal and off normal environment for both direct radiation and airborne radioactivity**
>
> **(B) ALARA review for the OCB.**

NFS Response: To address these requests, the following documents are provided in Enclosures I and II, respectively.

- ISA Source Term Data for the Oxide Conversion Building (OCB) Project (Revision 0)
- ALARA Design Guidance and Dose Rate Profile, Oxide Conversion Building

NRC Question (4): Describe any changes in NFS Radiation Safety Program due to the addition of the OCB.

NFS Response: No changes to NFS' existing Radiation Safety Program have been necessary to support operations at the OCB/EPB.

Enclosure I

ISA Source Term Data for the Oxide Conversion Building (OCB) Project, Revision 0

Nuclear Fuel Services, Inc.

ISA Source Term Data for the Oxide Conversion Building (OCB) Project

Revision 0

February 6, 2004

Originator: _____
K.A. Roberts, CHP
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Originator: _____
S.L. Sanders
Health Physicist
Nuclear Fuel Services, Inc.

Reviewer: _____
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RECORD OF REVISIONS

Page/Section	Description of Change
Entire Document	This is a new document.

TABLE OF CONTENTS

1. INTRODUCTION 5
2. IDENTIFICATION OF RADIOACTIVE MATERIALS AT RISK 5
2.1 Uranyl Nitrate Building (UNB) Feed to the OCB (Precipitation) 5
2.1.1 NFS Feed to the UNB 5
2.1.2 Savannah River Site (SRS) Feed to the UNB 5
2.2 Product (Dryer/Calciner) 7
2.3 Uranium Recovery 7
2.4 Ammonia Recovery System 7
2.5 Liquid Waste Treatment 8
2.6 IX Elution 8
2.7 Scrap Recovery Dissolution 8
2.8 Natural Uranium Dissolution 8
3. SUMMARY 9
ATTACHMENT *OXIDE CONVERSION BUILDING SOURCE TERM* 10

LIST OF TABLES

Table 2-1 Source Terms for the Processes of the OCB 6
Table 2-2 Uranium Isotopic Abundance 9

ACRONYMS

ADU	Ammonium diuranate
AR	Ammonium Recovery
ISA	Integrated safety analysis
NFS	Nuclear Fuel Services, Inc.
OCB	Oxide Conversion Building
SRS	Savannah River Site
U_3O_8	Uranium oxide
UN	Uranyl nitrate
UO_2	Uranium dioxide

1. INTRODUCTION

This document provides the radiological source terms in support of the Oxide Conversion Building (OCB) Integrated Safety Analysis (ISA) at the Nuclear Fuel Services, Inc. (NFS) plant site at Erwin, Tennessee. Ammonium diuranate (ADU) is precipitated from uranyl nitrate solution (UN) by neutralizing the acidic solution with ammonium hydroxide. The slurry is fed into a centrifuge for separation of the ADU solids. The ADU solids are dried and then calcined to uranium dioxide (UO_2). Most of the powder from the calciner is routed to a blender, but a portion of the calciner product is diverted to a calcined powder oxidizer, where batches of powder are oxidized to uranium oxide (U_3O_8) and then transferred to the blender. A homogeneous mix of UO_2 and U_3O_8 is blended and loaded into shipping pails.

2. IDENTIFICATION OF RADIOACTIVE MATERIALS AT RISK

The objective of this evaluation is to identify the radioactive materials (radionuclides, concentrations and physical forms) that could be present in OCBs 520 and 530 based on proposed input sources and process descriptions. The input sources used to estimate the source term are given in the *Oxide Conversion Building Source Term* provided as an Attachment.

2.1 Uranyl Nitrate Building (UNB) Feed to the OCB (Precipitation)

Transfer of UN from the UNB to the OCB is accomplished by moving the UN to the last tank in the bank being used and then using the transfer system to pump the UN into the UN Receipt Tank in the OCB. The nominal UN concentration is projected to be [redacted] with a density of 1.14 g/cm^3 as documented in the *Oxide Conversion Building Source Term*. The UNB Source Term is shown in Table 2-1. The UNB Source Term is assumed as the sum of the NFS and Savannah River Site (SRS) activity decayed for one year. These two feeds are discussed in the following sections.

2.1.1 NFS Feed to the UNB

UN is transferred to the UNB from the NFS downblending facility via a pipeline into a [redacted] tank. The nominal UN concentration is projected to be [redacted] If the UN concentration exceeds [redacted] as indicated by the recirculation density/flowmeter, the transfer is stopped. The average radionuclide concentrations are assigned as specified in the *Oxide Conversion Building Source Term*. It is assumed that the NFS feed contributes 53% to the UNB product, and the NFS Feed Source Term is provided in the Attachment.

2.1.2 Savannah River Site (SRS) Feed to the UNB

UN is transported from the SRS for storage in the UNB in specially designed shipping containers. The UN is transferred from the shipping containers into a [redacted] tank. The nominal UN concentration is projected to be [redacted] per the *Oxide Conversion Building Source Term*. If the UN concentration exceeds [redacted] as indicated by the recirculation density/flowmeter, the transfer is stopped. The average radionuclide concentrations are provided in the source term document. It is assumed that there is a 15.1 HEU to LEU ratio, that solvent extraction is 100% efficient and that the SRS feed contributes 47% to the UNB product. The SRS Feed Source Term is provided in the *Oxide Conversion Building Source Term*.

Table 2-1 Source Terms for the Processes of the OCB

Constituent	UNB Product (Ci/gU)	Dryer (Product) (Ci/g U)	Uranium Recovery (Ci/g U)	Ammonia Recovery (Ci/g U)	Liquid Waste Stream (Ci/g U)	IX Effluent (Ci/gU)
U-232	3.26E-08	3.26E-08	3.26E-13	3.26E-13	1.86E-12	5.56E-09
U-233	2.43E-08	2.43E-08	2.43E-13	2.43E-13	1.39E-12	4.15E-09
U-234	6.79E-06	6.79E-06	6.79E-11	6.79E-11	3.88E-10	1.16E-06
U-235	1.21E-07	1.21E-07	1.21E-12	1.21E-12	6.92E-12	2.06E-08
U-236	1.39E-06	1.39E-06	1.39E-11	1.39E-11	7.94E-11	2.37E-07
U-237	1.12E-13	1.12E-13	1.12E-18	1.12E-18	6.40E-18	1.91E-14
U-238	3.53E-07	3.53E-07	3.53E-12	3.53E-12	2.02E-11	6.01E-08
Tl-207	4.13E-12	8.26E-13	3.30E-12	1.83E-12	1.04E-11	1.65E-12
Tl-208	9.66E-09	1.93E-09	7.73E-09	4.27E-09	2.44E-08	3.86E-09
Tl-209	4.47E-14	8.94E-15	3.58E-14	1.98E-14	1.13E-13	1.79E-14
Th-227	4.11E-12	4.11E-12	4.11E-17	4.11E-17	2.35E-16	2.06E-17
Th-228	2.70E-08	2.70E-08	2.70E-13	2.70E-13	1.54E-12	1.35E-13
Th-229	2.30E-12	2.30E-12	2.30E-17	2.30E-17	1.31E-16	1.15E-17
Th-230	6.36E-10	6.36E-10	6.36E-15	6.36E-15	3.63E-14	3.18E-15
Th-231	1.21E-07	1.21E-07	1.21E-12	1.21E-12	6.92E-12	6.05E-13
Th-232	2.45E-11	2.45E-11	2.45E-16	2.45E-16	1.40E-15	1.23E-16
Th-234	3.53E-07	3.53E-07	3.53E-12	3.53E-12	2.02E-11	1.76E-12
Tc-99	4.59E-12	0.00E+00*	4.59E-12	2.54E-12	1.45E-11	2.30E-12
Sr-90	8.80E-12	4.40E-12	4.40E-12	2.43E-12	1.39E-11	2.20E-12
Sm-147	3.02E-24	3.02E-24	3.02E-29	3.02E-29	1.72E-28	1.51E-29
Rn-219	4.15E-12	2.07E-12	2.07E-12	1.15E-12	6.55E-12	1.04E-12
Rn-220	2.70E-08	1.35E-08	1.35E-08	7.46E-09	4.26E-08	6.74E-09
Rn-222	1.86E-12	9.30E-13	9.30E-13	5.14E-13	2.94E-12	4.65E-13
Ra-223	4.15E-12	2.07E-12	2.07E-12	1.15E-12	6.55E-12	1.04E-12
Ra-224	2.70E-08	1.35E-08	1.35E-08	7.46E-09	4.26E-08	6.74E-09
Ra-225	2.16E-12	1.08E-12	1.08E-12	5.97E-13	3.41E-12	5.40E-13
Ra-226	1.86E-12	9.30E-13	9.30E-13	5.14E-13	2.94E-12	4.65E-13
Ra-228	1.97E-11	9.83E-12	9.83E-12	5.43E-12	3.11E-11	4.91E-12
Pu-238	8.54E-10	8.54E-10	8.54E-15	8.54E-15	4.88E-14	4.27E-15
Pu-239	9.12E-12	9.12E-12	9.12E-17	9.12E-17	5.21E-16	4.56E-17
Pu-240	9.12E-12	9.12E-12	9.12E-17	9.12E-17	5.21E-16	4.56E-17
Pu-241	4.56E-09	4.56E-09	4.56E-14	4.56E-14	2.61E-13	2.28E-14
Po-210	2.66E-14	1.33E-14	1.33E-14	7.37E-15	4.21E-14	6.66E-15
Po-211	1.13E-14	5.65E-15	5.65E-15	3.13E-15	1.79E-14	2.83E-15
Po-212	1.73E-08	8.65E-09	8.65E-09	4.78E-09	2.73E-08	4.33E-09
Po-213	2.02E-12	1.01E-12	1.01E-12	5.60E-13	3.20E-12	5.06E-13
Po-214	1.86E-12	9.30E-13	9.30E-13	5.14E-13	2.94E-12	4.65E-13
Po-215	4.15E-12	2.07E-12	2.07E-12	1.15E-12	6.55E-12	1.04E-12
Po-216	2.70E-08	1.35E-08	1.35E-08	7.46E-09	4.26E-08	6.74E-09
Po-218	1.86E-12	9.30E-13	9.30E-13	5.14E-13	2.94E-12	4.65E-13
Pm-147	4.04E-13	4.04E-13	4.04E-18	4.04E-18	2.31E-17	2.02E-18
Pb-209	2.07E-12	2.03E-12	4.14E-14	2.29E-14	1.31E-13	2.07E-14
Pb-210	5.21E-14	5.11E-14	1.04E-15	5.76E-16	3.29E-15	5.21E-16
Pb-211	4.15E-12	4.06E-12	8.29E-14	4.59E-14	2.62E-13	4.15E-14
Pb-212	2.70E-08	2.64E-08	5.39E-10	2.98E-10	1.70E-09	2.70E-10
Pb-214	1.86E-12	1.82E-12	3.72E-14	2.06E-14	1.18E-13	1.86E-14
Pa-231	2.27E-11	2.27E-11	2.27E-16	2.27E-16	1.30E-15	1.13E-16
Pa-233	3.09E-09	3.09E-09	3.09E-14	3.09E-14	1.77E-13	1.54E-14

Table 2-1 Source Terms for the Processes of the OCB

Constituent	UNB Product (Ci/gU)	Dryer (Product) (Ci/g U)	Uranium Recovery (Ci/g U)	Ammonia Recovery (Ci/g U)	Liquid Waste Stream (Ci/g U)	IX Effluent (Ci/gU)
Pa-234	5.65E-10	5.65E-10	5.65E-15	5.65E-15	3.23E-14	2.82E-15
Pa-234m	3.53E-07	3.53E-07	3.53E-12	3.53E-12	2.02E-11	1.76E-12
Np-237	3.09E-09	3.09E-09	3.09E-14	3.09E-14	1.77E-13	1.54E-14
Fr-221	2.07E-12	4.14E-13	1.66E-12	9.17E-13	5.24E-12	8.29E-13
Fr-223	5.82E-14	1.16E-14	4.65E-14	2.57E-14	1.47E-13	2.33E-14
Eu-154	3.70E-13	3.70E-13	3.70E-18	3.70E-18	2.12E-17	1.85E-18
Cs-134	3.54E-13	3.36E-13	1.77E-14	9.78E-15	5.59E-14	8.84E-15
Cs-137	1.24E-11	1.18E-11	6.20E-13	3.43E-13	1.96E-12	3.10E-13
Bi-210	5.10E-14	2.55E-15	4.84E-14	2.68E-14	1.53E-13	2.42E-14
Bi-211	4.15E-12	2.07E-13	3.94E-12	2.18E-12	1.24E-11	1.97E-12
Bi-212	2.70E-08	1.35E-09	2.56E-08	1.42E-08	8.09E-08	1.28E-08
Bi-213	2.07E-12	1.04E-13	1.97E-12	1.09E-12	6.22E-12	9.84E-13
Bi-214	1.86E-12	9.30E-14	1.77E-12	9.77E-13	5.58E-12	8.83E-13
Ba-137m	1.18E-11	1.12E-11	5.88E-13	3.25E-13	1.86E-12	2.94E-13
At-217	2.07E-12	4.14E-13	1.66E-12	9.17E-13	5.24E-12	8.29E-13
Am-241	1.46E-10	1.46E-10	1.46E-15	1.46E-15	8.37E-15	7.32E-16
Ac-225	2.07E-12	4.14E-13	1.66E-12	9.17E-13	5.24E-12	8.29E-13
Ac-227	4.21E-12	8.42E-13	3.37E-12	1.86E-12	1.06E-11	1.68E-12
Ac-228	1.96E-11	3.92E-12	1.57E-11	8.67E-12	4.96E-11	7.84E-12
*It is assumed that 100% of the Tc-99 is removed from the powder in the centrifuge						

2.2 Product (Dryer/Calciner)

ADU is precipitated from UN by neutralizing the acidic solution with ammonium hydroxide. The slurry is fed into a centrifuge for separation of the ADU solids. The ADU solids are dried and then calcined to UO_2. The densities of the solids are 3.0 g/cm^3 in the dryer and decreases to 2.0 g/cm^3 in the calciner based on the *Oxide Conversion Building Source Term*. Most of the powder from the calciner is routed to a blender. A portion of the calciner product is diverted to a calcined powder oxidizer, where batches of powder are oxidized to U_3O_8 and then transferred to the blender. A homogeneous mix of UO_2 and U_3O_8 is blended and loaded into shipping pails. The Attachment identifies the percent of the constituent that stays with the powder and the resultant activity of the product (powder). The source term for the dryer is shown in Table 2-1.

2.3 Uranium Recovery

Liquid discharged from the centrifuge is passed through a cross-flow filter to recover insoluble uranium solids and an ion exchange system to collect soluble uranium for return to the process. The *Oxide Conversion Building Source Term* projects the liquid decanted from the centrifuge at [redacted], the cross flow return at [redacted], the cross flow filtrate and ion exchange feed at [redacted], the ion exchange effluent at [redacted], the ion exchange eluate at [redacted] and the acid wash to the recycle tanks at [redacted] The percentage of the constituents that does not stay with the powder is removed in the uranium recovery stream. The source term for the uranium recovery process is provided in Table 2-1.

2.4 Ammonia Recovery System

The ammonia recovery (AR) system is located in the Effluent Processing Building. Due to the potential for the escape of ammonia vapors from the process equipment, the AR process is

partitioned from the other processes in the building. The AR process is designed to reclaim ammonia from the uranyl nitrate conversion waste stream for reuse. Sodium hydroxide will be added to the OCB effluent followed by steam stripping to remove ammonia. The ammonia will be recovered and returned to the oxide conversion process. The bottoms from the ammonia recovery operation is primarily a sodium nitrate solution, but also contains radionuclides such as Tc-99 that did not precipitate out in the conversion process. The radionuclides that are 100% precipitated into the product (see the Attachment) are assumed to be associated with a uranium concentration of [redacted]. The concentration of the nuclides that were not precipitated 100% is decreased by the ratio of 7.3/13.2 per the *Oxide Conversion Building Source Term*. This reduction is due to the addition of the sodium hydroxide. The AR source term is provided in Table 2-1.

2.5 Liquid Waste Treatment

The liquid bottoms from the ammonia stripper column contain approximately [redacted] sodium nitrate per the source term document. This solution is pumped to intermediate storage before being concentrated in the evaporator. The evaporator overhead is free of sodium nitrate and radioactive contaminants and is stored in the overhead storage tanks prior to discharge to the sewer. The evaporator concentrates the bottoms to a concentration of approximately [redacted] sodium nitrate per the *Oxide Conversion Building Source Term*. This concentrated solution is solidified in concrete and disposed as radioactive waste. The bottoms from the ammonia recovery operation will be evaporated (5.7 [400/70] concentration factor) and stabilized in a solid matrix for disposal. The source term for liquid waste treatment is provided in Table 2-1.

2.6 IX Elution

The IX Elution source term is a combination of uranium in the resin and the uranium recovery source term. A maximum quantity of [redacted] of uranium can be in the resin, and the remaining volume is filled with the uranium recovery solution. A conservative assumption for three resin exchange columns is for one column to be fully loaded with resin, another to be filled with the filtrate, while the third is considered empty. The volume of one column is [redacted] and the maximum uranium concentration in the product feed is assumed to be [redacted]. The source term is shown in Table 2-1.

2.7 Scrap Recovery Dissolution

Enriched scrap uranium oxide is dissolved in separate dissolvers using a nitric acid solution. The dissolved scrap uranium is recycled to the ADU precipitation process. The *Oxide Conversion Source Building Term* projects the UN from the dissolver at [redacted] and the UN product at [redacted]. The concentration of the dryer given in Table 2-1 was used as the source term for the scrap recovery dissolution.

2.8 Natural Uranium Dissolution

The natural uranium dissolution is not part of the oxide conversion process. A natural uranium dissolver system is provided to produce [redacted] uranyl nitrate solution for shipment to SRS per the Attachment. Natural uranium (UO_3) is dissolved in nitric acid solution and then sent to storage in the UNB prior to offsite truck shipment. This operation is kept completely separate from enriched uranium processing systems to prevent accidental mixing of enriched and natural uranium.

The isotopic abundances of natural uranium isotopes are shown in Table 2-2. Dose calculations will be limited to the dose due to these isotopes of uranium as they account for over 90% of the dose associated with radioactivity in natural uranium.

Table 2-2 Uranium Isotopic Abundance

Isotope	Weight Percent – Natural (g/g U)
U-234	0.0054%
U-235	0.7111%
U-238	99.2836%

3. SUMMARY

This evaluation has been performed to identify radioactive materials that could be present in Oxide Conversion Buildings 520 and 530 based on proposed input sources and process descriptions.

Attachment ***Oxide Conversion Building Source Term***

FRAMATOME ANP

JLZ:03:001

Date: December 15, 2003

Distribution

To: C. J. Yeager

File/LB

From: J. L. Zito

Subject: **Oxide Conversion Building Source Term**

Scope

This document outlines the source term assumptions used in the analysis of Oxide Conversion Building (OCB) radiological hazards.

General Description

The source term assumptions for the OCB were derived using information related to process operations performed on feed material from the Uranyl Nitrate Building (UNB). The UNB source term was derived using information on the handling and storage of feed materials from Savannah River (SRS) and Nuclear Fuel Services (NFS).

Construction Details/Information

Two feed materials are received at the BLEU complex. The first feed material is uranyl nitrate solution from SRS. This material is received by truck and off-loaded into UNB [redacted] The second feed material is uranyl nitrate solution from NFS. The NFS material is produced on the NFS site and transferred to the UNB [redacted] by pipeline. These two materials make up the basis for the UNB source term. SRS feed source term details are given in Table 1. NFS source term details are given in Table 2.

Information about feed material handling, storage and mixing were used to derive the UNB source term. Details on the UNB source term are shown in Table 3. The UNB source term was then used as the basis for source term information for the various OCB processes. The UNB source term was modified based on known parameters related to the handling and processing of UNB feed material in the OCB. Details on the assumptions used for determining OCB source term information are contained in Tables 4 and 5.

Framatome ANP

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Table 1 SRS Feed Source Term

Constituent	[1]SRS Feed Sample Data	[2]SRS Feed w/o SX (pCi/gLEU)	[3]SRS Feed w/o SX (pCi/gU)	[4]SRS Feed w/o SX (pCi)	[5]SRS Feed with SX (Ci)
U-232	0.001ug/g U		2.14E+04	[redacted]	[redacted]
U-233	0.00E+00				
U-234	800 ug/g U		5.00E+06	[redacted]	[redacted]
U-235	4.95%		1.07E+05	[redacted]	[redacted]
U-236	13000 ug/g U		8.41E+05	[redacted]	[redacted]
U-238	95%		3.19E+05	[redacted]	[redacted]
Th-232	1.0 ug/gLEU	1.10E-01	1.39E+00	[redacted]	[redacted]
Pu-238	5 Bq/gLEU	1.35E+02	1.71E+03	[redacted]	[redacted]
Pu-239/240	negligible				
Pu-241	negligible				
Np-237	25 Bq/gLEU	6.76E+02	8.54E+03	[redacted]	[redacted]
Thorium Series					
Ra-224	1.9E4 MeV/s/kgHEU	3.22E+04	4.07E+05	[redacted]	
Pb-212	1.1E6 MeV/s/kgHEU	1.85E+06	2.33E+07	[redacted]	
Bi-212	4.3E5 MeV/s/kgHEU	2.41E+05	3.05E+06	[redacted]	
Tl-208	8.0E6 MeV/s/kgHEU	2.09E+06	2.64E+07	[redacted]	
Fission Products					
Cs-137	3.5E6 MeV/s/kgHEU	2.16E+06	2.73E+07	[redacted]	
Additional SRS					
Pa-233	1.2E5 MeV/s/kgHEU	1.57E+05	1.98E+06	[redacted]	
Nb-94	4.9E4 MeV/s/kgHEU	2.54E+04	3.21E+05	[redacted]	
Sb-125	3.2E4 MeV/s/kgHEU	2.55E+04	3.22E+05	[redacted]	

[1]SRS Feed Sample Data as provided by Richard Frushour through Charlie Holman
[2] ug/gLEU conversion based on specific activity. MeV/s/kgHEU conversion assumes a 15.1 HEU to LEU ratio as provided by RichardFrushour through Charlie Holman and primary gamma energies and probability per decay as specified by RadDecay Version 5 software
[3]Uranium isotope conversion based on specific activity. LEU conversion assumes [redacted] and [redacted] (as per Charlie Holman)
[4]Assumed contributes 47% to UNB Product a[redacted]

SRS Feed (gU/L)	[redacted]
% Contribution	0.47

[5]Assumed Solvent Extraction 100% efficient for sample data nuclides provided in 'per kgHEU' terms

Table 2 NFS Feed Source Term

Constituent	[1]NFS Feed (pCi/g U)	[2]NFS Feed (Ci)
U-232	3.27E+04	[redacted]
U-233	3.11E+04	[redacted]
U-234	6.50E+06	[redacted]
U-235	1.08E+05	[redacted]
U-236	1.40E+06	[redacted]
U-238	3.12E+05	[redacted]
Th-228	3.12E+04	[redacted]
Th-230	7.33E+02	[redacted]
Th-231	9.33E+04	[redacted]
Th-232	3.07E+01	[redacted]
Th-234	1.51E+03	[redacted]
Pu-238	3.61E+02	[redacted]
Pu-239/240	2.32E+01	[redacted]
Pu-241	6.10E+03	[redacted]
Am-241	1.77E+02	[redacted]
Np-237	2.56E+02	[redacted]
Thorium Series		
Ra-228	2.43E+01	[redacted]
Ac-228	2.43E+01	[redacted]
Ra-224	3.12E+04	[redacted]
Pb-212	3.09E+04	[redacted]
Bi-212	1.76E+04	[redacted]
Tl-208	9.96E+03	[redacted]
Uranium Series		
Pa-234	1.11E+01	[redacted]
Pa-234m	4.19E+03	[redacted]
Ra-226	2.04E+00	[redacted]
Actinium Series		
Pa-231	2.57E+01	[redacted]
Ac-227	4.66E+00	[redacted]
Th-227	4.55E+00	[redacted]
Ra-223	4.58E+00	[redacted]
Fission Products		
Sr-90	1.15E+01	[redacted]
Tc-99	5.85E+00	[redacted]
Cs-134	6.31E-01	[redacted]
Cs-137	1.62E+01	[redacted]
Pm-147	6.70E-01	[redacted]
Eu-154	5.11E-01	[redacted]

[1]Average radionuclide concentrations assigned as specified in the *ISA Source Term Data and Radioactive Effluent Estimates* for the TVA Project (HEA-21, 21T-02-0300, BPG-02-011)

[2]Assumed contributes 53% to UNB Product at 185 gU/L

Table 3 UNB Source Term Derivation

Constituent	SRS Contribution to UNB LEUN (pCi/g U)	Activity (SRS) After 1 Year (Ci)	NFS Contribution to UNB LEUN (pCi/gU)	Activity (NFS) After 1 Year (Ci)	UNB Product at 1 Year (pCi)	UNB Product at 1 Year (pCi/gU)
U-232	2.12E+04	[redacted]	3.24E+04	[redacted]	[redacted]	3.26E+04
U-233	3.31E-02	[redacted]	3.10E+04	[redacted]	[redacted]	2.43E+04
U-234	4.99E+06	[redacted]	6.51E+06	[redacted]	[redacted]	6.79E+06
U-235	1.07E+05	[redacted]	1.08E+05	[redacted]	[redacted]	1.21E+05
U-236	8.42E+05	[redacted]	1.41E+06	[redacted]	[redacted]	1.39E+06
U-237			1.43E-01	[redacted]	[redacted]	1.12E-01
U-238	3.19E+05	[redacted]	3.12E+05	[redacted]	[redacted]	3.53E+05
Tl-207	2.77E-02	[redacted]	5.25E+00	[redacted]	[redacted]	4.13E+00
Tl-208	2.30E+03	[redacted]	1.13E+04	[redacted]	[redacted]	9.66E+03
Tl-209	2.46E-08	[redacted]	5.70E-02	[redacted]	[redacted]	4.47E-02
Th-227	3.00E-02	[redacted]	5.23E+00	[redacted]	[redacted]	4.11E+00
Th-228	6.48E+03	[redacted]	3.16E+04	[redacted]	[redacted]	2.70E+04
Th-229	1.42E-06	[redacted]	2.93E+00	[redacted]	[redacted]	2.30E+00
Th-230	4.49E+01	[redacted]	7.91E+02	[redacted]	[redacted]	6.36E+02
Th-231	1.07E+05	[redacted]	1.08E+05	[redacted]	[redacted]	1.21E+05
Th-232	1.39E+00	[redacted]	3.07E+01	[redacted]	[redacted]	2.45E+01
Th-234	3.19E+05	[redacted]	3.12E+05	[redacted]	[redacted]	3.53E+05
Tc-99			5.85E+00	[redacted]	[redacted]	4.59E+00
Sr-90			1.12E+01	[redacted]	[redacted]	8.80E+00
Sm-147			3.84E-12	[redacted]	[redacted]	3.02E-12
Rn-219	2.77E-02	[redacted]	5.27E+00	[redacted]	[redacted]	4.15E+00
Rn-220	6.41E+03	[redacted]	3.16E+04	[redacted]	[redacted]	2.70E+04
Rn-222	9.43E-03	[redacted]	2.37E+00	[redacted]	[redacted]	1.86E+00
Ra-223	2.77E-02	[redacted]	5.27E+00	[redacted]	[redacted]	4.15E+00
Ra-224	6.41E+03	[redacted]	3.16E+04	[redacted]	[redacted]	2.70E+04
Ra-225	1.25E-06	[redacted]	2.75E+00	[redacted]	[redacted]	2.16E+00
Ra-226	9.72E-03	[redacted]	2.37E+00	[redacted]	[redacted]	1.86E+00
Ra-228	1.57E-01	[redacted]	2.50E+01	[redacted]	[redacted]	1.97E+01
Pu-238	1.69E+03	[redacted]	3.58E+02	[redacted]	[redacted]	8.54E+02
Pu-239/240			2.33E+01	[redacted]	[redacted]	1.82E+01
Pu-241			5.81E+03	[redacted]	[redacted]	4.56E+03

Table 3 UNB Source Term Derivation *(Cont.)*

[1]Constituent	[2]SRS Contribution to UNB LEUN (pCi/g U)	[3]Activity (SRS) After 1 Year (Ci)	[4]NFS Contribution to UNB LEUN (pCi/gU)	[5]Activity (NFS) After 1 Year (Ci)	[6]UNB Product at 1 Year (pCi)	UNB Product at 1 Year (pCi/gU)
Po-210	2.91E-05	[redacted]	3.40E-02	[redacted]	[redacted]	2.66E-02
Po-211	7.57E-05	[redacted]	1.44E-02	[redacted]	[redacted]	1.13E-02
Po-212	4.09E+03	[redacted]	2.03E+04	[redacted]	[redacted]	1.73E+04
Po-213	1.11E-06	[redacted]	2.58E+00	[redacted]	[redacted]	2.02E+00
Po-214	9.41E-03	[redacted]	2.37E+00	[redacted]	[redacted]	1.86E+00
Po-215	2.77E-02	[redacted]	5.27E+00	[redacted]	[redacted]	4.15E+00
Po-216	6.41E+03	[redacted]	3.16E+04	[redacted]	[redacted]	2.70E+04
Po-218	9.43E-03	[redacted]	2.37E+00	[redacted]	[redacted]	1.86E+00
Pm-147			5.15E-01	[redacted]	[redacted]	4.04E-01
Pb-209	1.13E-06	[redacted]	2.64E+00	[redacted]	[redacted]	2.07E+00
Pb-210	9.55E-05	[redacted]	6.64E-02	[redacted]	[redacted]	5.21E-02
Pb-211	2.77E-02	[redacted]	5.27E+00	[redacted]	[redacted]	4.15E+00
Pb-212	6.38E+03	[redacted]	3.16E+04	[redacted]	[redacted]	2.70E+04
Pb-214	9.41E-03	[redacted]	2.37E+00	[redacted]	[redacted]	1.86E+00
Pa-231	2.26E+00	[redacted]	2.79E+01	[redacted]	[redacted]	2.27E+01
Pa-233	8.53E+03	[redacted]	2.56E+02	[redacted]	[redacted]	3.09E+03
Pa-234	5.11E+02	[redacted]	5.00E+02	[redacted]	[redacted]	5.65E+02
Pa-234m	3.19E+05	[redacted]	3.12E+05	[redacted]	[redacted]	3.53E+05
Np-237	8.53E+03	[redacted]	2.56E+02	[redacted]	[redacted]	3.09E+03
Fr-221	1.14E-06	[redacted]	2.64E+00	[redacted]	[redacted]	2.07E+00
Fr-223	4.87E-04	[redacted]	7.39E-02	[redacted]	[redacted]	5.82E-02
Eu-154			4.72E-01	[redacted]	[redacted]	3.70E-01
Cs-134			4.51E-01	[redacted]	[redacted]	3.54E-01
Cs-137			1.58E+01	[redacted]	[redacted]	1.24E+01
Bi-210	9.01E-05	[redacted]	6.50E-02	[redacted]	[redacted]	5.10E-02
Bi-211	2.77E-02	[redacted]	5.27E+00	[redacted]	[redacted]	4.15E+00
Bi-212	6.38E+03	[redacted]	3.16E+04	[redacted]	[redacted]	2.70E+04
Bi-213	1.14E-06	[redacted]	2.64E+00	[redacted]	[redacted]	2.07E+00
Bi-214	9.41E-03	[redacted]	2.37E+00	[redacted]	[redacted]	1.86E+00
Ba-137m			1.50E+01	[redacted]	[redacted]	1.18E+01
At-217	1.14E-06	[redacted]	2.64E+00	[redacted]	[redacted]	2.07E+00
Am-241			1.87E+02	[redacted]	[redacted]	1.46E+02

Table 3 UNB Source Term Derivation *(Cont.)*

[1]Constituent	[2]SRS Contribution to UNB LEUN (pCi/g U)	[3]Activity (SRS) After 1 Year (Ci)	[4]NFS Contribution to UNB LEUN (pCi/gU)	[5]Activity (NFS) After 1 Year (Ci)	[6]UNB Product at 1 Year (pCi)	[7]UNB Product at 1 Year (pCi/gU)
Ac-225	1.14E-06	[redacted]	2.64E+00	[redacted]	[redacted]	2.07E+00
Ac-227	3.52E-02	[redacted]	5.35E+00	[redacted]	[redacted]	4.21E+00
Ac-228			2.50E+01	[redacted]	[redacted]	1.96E+01

[1]Projected nuclides after 1 year using RadDecay Version 5 assuming original constituents as per SRS/NFS LEU feed data. Italicized entries are NFS specific.

[2]Assumed contributes 47% to UNB Product at [redacted]
SRS Feed (gU/L) [redacted]
% Contribution 0.47

[3]Projected activity after 1 year using RadDecay Version 5 assuming original activity as per SRS LEU feed sample data

[4]Assumed contributes 53% to UNB Product at [redacted]
NFS Feed (gU/L) [redacted]
% Contribution 0.53

[5]Projected activity after 1 year using RadDecay Version 5 assuming original activity as per NFS LEU feed data

[6]Assumed UNB total activity as sum of NFS and SRS activity contributions at 1 year

[7]Assumed UNB Product at [redacted]

Table 4 Process Design Data Assumptions

Description	Value	Units
Density of Uranyl Nitrate Feed	1.14	g/cm^3
Uranium Concentration (UN Feed)	[redacted]	g U/l
ADU to Dryer Density	2	g/cm^3
ADU to Calciner Density	3	g/cm^3
Liquid decanted from centrifuge	[redacted]	g U/l
Cross flow return	[redacted]	g U/l
Ion Exchange feed	[redacted]	U ppm
Ion Exchange Effluent	[redacted]	U ppm
Ion Exchange Eluate	[redacted]	U ppm
Acid Wash to Recycle Tank	[redacted]	U ppm
Uranyl Nitrate from Dissolver	[redacted]	g U/l

Table 5 Additional Assumptions

Description	Value	Units
NFS feed contribution	53%	
NFS U concentration	[redacted]	g U/l
SRS feed contribution	47%	
SRS U concentration	[redacted]	g U/l
SRS LEU concentration	[redacted]	g LEU/ml
HEU to LEU ratio	15.1	
OCB Solvent extraction efficiency	100%	
Ammonia Recovery System Reduction	7.3/13.2	
Liquid Waste Concentration Ratio	400/70	

Table 6 Other Source Term Derivation

Constituent	UNB Product (Ci/gU)[1]	Percentage of Activity Precipitated with Solids[2]	Dryer (Product) (Ci/g U)[3]	Uranium Recovery (Ci/g U)[4]	Ammonia Recovery (Ci/g U)[5]	Liquid Waste Stream (Ci/g U)[6]
U-232	3.26E-08	100%	3.26E-08	3.26E-13	3.26E-13	1.86E-12
U-233	2.43E-08	100%	2.43E-08	2.43E-13	2.43E-13	1.39E-12
U-234	6.79E-06	100%	6.79E-06	6.79E-11	6.79E-11	3.88E-10
U-235	1.21E-07	100%	1.21E-07	1.21E-12	1.21E-12	6.92E-12
U-236	1.39E-06	100%	1.39E-06	1.39E-11	1.39E-11	7.94E-11
U-237	1.12E-13	100%	1.12E-13	1.12E-18	1.12E-18	6.40E-18
U-238	3.53E-07	100%	3.53E-07	3.53E-12	3.53E-12	2.02E-11
Tl-207	4.13E-12	20%	8.26E-13	3.30E-12	1.83E-12	1.04E-11
Tl-208	9.66E-09	20%	1.93E-09	7.73E-09	4.27E-09	2.44E-08
Tl-209	4.47E-14	20%	8.94E-15	3.58E-14	1.98E-14	1.13E-13
Th-227	4.11E-12	100%	4.11E-12	4.11E-17	4.11E-17	2.35E-16
Th-228	2.70E-08	100%	2.70E-08	2.70E-13	2.70E-13	1.54E-12
Th-229	2.30E-12	100%	2.30E-12	2.30E-17	2.30E-17	1.31E-16
Th-230	6.36E-10	100%	6.36E-10	6.36E-15	6.36E-15	3.63E-14
Th-231	1.21E-07	100%	1.21E-07	1.21E-12	1.21E-12	6.92E-12
Th-232	2.45E-11	100%	2.45E-11	2.45E-16	2.45E-16	1.40E-15
Th-234	3.53E-07	100%	3.53E-07	3.53E-12	3.53E-12	2.02E-11
Tc-99	4.59E-12			4.59E-12	2.54E-12	1.45E-11
Sr-90	8.80E-12	50%	4.40E-12	4.40E-12	2.43E-12	1.39E-11
Sm-147	3.02E-24	100%	3.02E-24	3.02E-29	3.02E-29	1.72E-28
Rn-219	4.15E-12	50%	2.07E-12	2.07E-12	1.15E-12	6.55E-12
Rn-220	2.70E-08	50%	1.35E-08	1.35E-08	7.46E-09	4.26E-08
Rn-222	1.86E-12	50%	9.30E-13	9.30E-13	5.14E-13	2.94E-12
Ra-223	4.15E-12	50%	2.07E-12	2.07E-12	1.15E-12	6.55E-12
Ra-224	2.70E-08	50%	1.35E-08	1.35E-08	7.46E-09	4.26E-08
Ra-225	2.16E-12	50%	1.08E-12	1.08E-12	5.97E-13	3.41E-12
Ra-226	1.86E-12	50%	9.30E-13	9.30E-13	5.14E-13	2.94E-12
Ra-228	1.97E-11	50%	9.83E-12	9.83E-12	5.43E-12	3.11E-11
Pu-238	8.54E-10	100%	8.54E-10	8.54E-15	8.54E-15	4.88E-14
Pu-239	9.12E-12	100%	9.12E-12	9.12E-17	9.12E-17	5.21E-16
Pu-240	9.12E-12	100%	9.12E-12	9.12E-17	9.12E-17	5.21E-16
Pu-241	4.56E-09	100%	4.56E-09	4.56E-14	4.56E-14	2.61E-13

Table 6 Other Source Term Derivation *(Cont.)*

Constituent	UNB Product (Ci/gU)[1]	Percentage of Activity Precipitated with Solids[2]	Dryer (Product) (Ci/g U)[3]	Uranium Recovery (Ci/g U)[4]	Ammonia Recovery (Ci/g U)[5]	Liquid Waste Stream (Ci/g U)[6]
Po-210	2.66E-14	50%	1.33E-14	1.33E-14	7.37E-15	4.21E-14
Po-211	1.13E-14	50%	5.65E-15	5.65E-15	3.13E-15	1.79E-14
Po-212	1.73E-08	50%	8.65E-09	8.65E-09	4.78E-09	2.73E-08
Po-213	2.02E-12	50%	1.01E-12	1.01E-12	5.60E-13	3.20E-12
Po-214	1.86E-12	50%	9.30E-13	9.30E-13	5.14E-13	2.94E-12
Po-215	4.15E-12	50%	2.07E-12	2.07E-12	1.15E-12	6.55E-12
Po-216	2.70E-08	50%	1.35E-08	1.35E-08	7.46E-09	4.26E-08
Po-218	1.86E-12	50%	9.30E-13	9.30E-13	5.14E-13	2.94E-12
Pm-147	4.04E-13	100%	4.04E-13	4.04E-18	4.04E-18	2.31E-17
Pb-209	2.07E-12	98%	2.03E-12	4.14E-14	2.29E-14	1.31E-13
Pb-210	5.21E-14	98%	5.11E-14	1.04E-15	5.76E-16	3.29E-15
Pb-211	4.15E-12	98%	4.06E-12	8.29E-14	4.59E-14	2.62E-13
Pb-212	2.70E-08	98%	2.64E-08	5.39E-10	2.98E-10	1.70E-09
Pb-214	1.86E-12	98%	1.82E-12	3.72E-14	2.06E-14	1.18E-13
Pa-231	2.27E-11	100%	2.27E-11	2.27E-16	2.27E-16	1.30E-15
Pa-233	3.09E-09	100%	3.09E-09	3.09E-14	3.09E-14	1.77E-13
Pa-234	5.65E-10	100%	5.65E-10	5.65E-15	5.65E-15	3.23E-14
Pa-234m	3.53E-07	100%	3.53E-07	3.53E-12	3.53E-12	2.02E-11
Np-237	3.09E-09	100%	3.09E-09	3.09E-14	3.09E-14	1.77E-13
Fr-221	2.07E-12	20%	4.14E-13	1.66E-12	9.17E-13	5.24E-12
Fr-223	5.82E-14	20%	1.16E-14	4.65E-14	2.57E-14	1.47E-13
Eu-154	3.70E-13	100%	3.70E-13	3.70E-18	3.70E-18	2.12E-17
Cs-134	3.54E-13	95%	3.36E-13	1.77E-14	9.78E-15	5.59E-14
Cs-137	1.24E-11	95%	1.18E-11	6.20E-13	3.43E-13	1.96E-12
Bi-210	5.10E-14	5%	2.55E-15	4.84E-14	2.68E-14	1.53E-13
Bi-211	4.15E-12	5%	2.07E-13	3.94E-12	2.18E-12	1.24E-11
Bi-212	2.70E-08	5%	1.35E-09	2.56E-08	1.42E-08	8.09E-08
Bi-213	2.07E-12	5%	1.04E-13	1.97E-12	1.09E-12	6.22E-12
Bi-214	1.86E-12	5%	9.30E-14	1.77E-12	9.77E-13	5.58E-12
Ba-137m	1.18E-11	95%	1.12E-11	5.88E-13	3.25E-13	1.86E-12
At-217	2.07E-12	20%	4.14E-13	1.66E-12	9.17E-13	5.24E-12
Am-241	1.46E-10	100%	1.46E-10	1.46E-15	1.46E-15	8.37E-15
Ac-225	2.07E-12	20%	4.14E-13	1.66E-12	9.17E-13	5.24E-12
Ac-227	4.21E-12	20%	8.42E-13	3.37E-12	1.86E-12	1.06E-11
Ac-228	1.96E-11	20%	3.92E-12	1.57E-11	8.67E-12	4.96E-11

[1]From UNB Source term with values converted form pCi/g U to Ci/g U
[2]Obtained from e-mail from C. Holman to S. Sanders, September 14, 2003
[3]The UNB Product times the Percentage of Activity Precipitated with Solids
[4]The UNB Product times 1- the Percentage of Activity Precipitated with Solids
[5]For Percentage of Activity = 1, the UNB Product times [redacted]), all others it is the Uranium Recovery times 7.3/13.2. [redacted] is the nominal UN concentration for the process.
[6]The Ammonia Recovery times (400/70)

Enclosure II

ALARA Design Guidance and Dose Rate Profile, Oxide Conversion Building

21T-04-0255
HEA-21

Nuclear Fuel Services, Inc.

ALARA Design Guidance and Dose Rate Profile

Oxide Conversion Building

Nuclear Fuel Services, Inc.
Erwin, TN

February 4, 2004

INTRODUCTION

It is the policy of NFS to maintain a comprehensive radiological protection program as defined in Part 1, Chapter 3 of Special Nuclear Material License SNM-124 to ensure compliance with the requirements specified in Part 20 to Title 10 of the Code of Federal Regulations. The objective of NFS' Radiological Protection Program is to keep the radiation doses to workers not only below regulatory limits, but also as low as reasonably achievable (ALARA). As in other areas of the facility, NFS maintains the radiological safety program in accordance with the ALARA program. NFS' ALARA Program applies to all areas of NFS' radiation safety program, including internal and external exposure control, contamination control, facility design, facility operations, procedure development, training and work practices. NFS' radiation protection program provides sampling, monitoring, and surveillance for radioactive materials as needed in the Oxide Conversion Building (OCB).

ALARA DESIGN GUIDANCE

The code of federal regulations in 10 CFR 20, Section 1101(b), specifies that NRC licensees shall use to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA. In keeping with that requirement the ALARA program design philosophy and goals for OCB project activities located in Building 520 include the following:

- Design facilities and processes such that the maximum internal exposure is less than 500 mrem Total Effective Dose Equivalent (TEDE) per year to each individual employee. The design objective for containment of airborne radioactive materials should be capable of reducing internal exposure potential to less than 10 percent of the applicable Derived Air Concentration (DAC) in continuously occupied areas.

- Design facilities and processes such that the maximum external exposure is less than 500 mrem TEDE per year to each individual employee. Time, distance and shielding concepts should be capable of reducing external exposure potential to less than 10 percent of the annual limits in continuously occupied areas. For this project, the following assumptions will be applied for continuously occupied areas: baseline personnel time in area is 1,500 hours per year with an occupancy weighting factor of 0.6 (estimate of time in facility in radiation area). Therefore, the design target is to maintain exposure rates in continuously occupied areas to less than 0.5 mrem/hr [e.g., 500 mrem / (1,500 hr X 0.6)]. The shielding computer program MicroShield ® will be used to help develop baseline dose rate information for operations that are likely to exceed the facility design target. The ALARA design goal for individual areas or operations may be reevaluated if shielding calculations, available mitigation techniques, and cost benefit analysis indicate the selected target is not reasonably achievable.

  Note: Generally, if dose reduction can be achieved at a cost of less than $2,000 per person-rem, then it is cost beneficial and should always be considered.)

- Maintain and operate all routine process areas well within license limits. This includes but is not limited to surface contamination and building and enclosure airflow levels.

- Reduce average effluent releases of radioactive materials to less than 10 CFR, Part 20, Appendix B, Table II, values at the point of release.

Design for proposed buildings and facilities where radioactive materials will be handled need multi-discipline involvement including engineering, operations, maintenance, and health physics personnel to ensure the best possible design. ALARA considerations for minimizing exposures to radiation and radioactive materials that should be incorporated into facility design include the following basic elements:

- The general layout of the facility is reviewed, with attention placed upon relative location of traffic patterns, emergency access and egress, offices, and change rooms to processing equipment, material storage locations, and solid waste storage areas.

- The design provides for control of liquid, solid, and gaseous effluents. The design also considers the effluent sampling and monitoring systems necessary to ensure that the effluents will be properly characterized.

- The ventilation system provides protection to workers from airborne radioactive materials. Contaminated ventilation duct runs and potentially contaminated runs should be designed to ensure that the length and volume of the runs are minimized (HEPA filters should be located as close as practical to dry glovebox operations) to reduce holdup of fissile material.

- The design of hoods and gloveboxes provides for ease of decontamination, minimizes external dose rates, maximizes operations to reduce exposure, and provides adequate confinement and containment.

- The design includes required air sampling/monitoring, nuclear criticality safety systems, and nuclear accident dosimeter locations to ensure that these systems and controls are provided, identified, and meet the regulatory standards and criteria.

- The design considers and incorporates as appropriate materials and construction practices that will aid with the ease of decontamination and decommissioning of the facility and equipment at the end of its useful life.

- Design provides for the maintainability of proposed mechanical systems (e.g., pumps and valves) and considers proposed equipment designs or equipment that will be purchased to maximize reliability and ease of maintenance.

Additional ALARA items to considered during design phase for the containment of radioactive materials and external dose reduction include:

- Locate office and work stations to ensure that these locations are situated to minimize both external and internal exposures.

- Storage locations for contaminated materials and radioactive wastes should be located and ventilated to minimize dose during movement and storage. This should include minimizing

external dose rates, as appropriate, during shipping receipt, sampling, vault storage and transference to process activities.

- Personnel access to controlled areas should normally be through control points, which serve as a buffer between uncontrolled areas and the source of potential exposure. Control points can provide space for used protective equipment (hampers), and personnel survey equipment (friskers). Room should be provided for step-off pads and for personnel entering or exiting from the controlled area.

- Building ventilation systems should maintain airflow from areas of low contamination potential to areas of increasing relative potential for contamination.

- Location of overhead piping that is used to transfer contaminated or chemical solutions should be minimized in personnel access walkways. Additionally, ensure that if piping is located over walkways that only welded connections are used during installation.

- Consider the use of splashguards and drip pans around systems that used transfer/sample contaminated or chemical solutions.

The design and construction of gloveboxes and hoods requires use of quality materials and workmanship to ensure that the potential for exposure is minimized. Glovebox installation characteristics and features should include the following:

- Sufficient space around glovebox and any associated equipment to permit unrestricted entry and exit.

- Airflows designed to minimize airborne contamination in the breathing zone. Inward airflow through an open face of a hood or glovebox door shall be at least 125 linear feet per minute.

- Bag in/out ports (if used) should be large enough that most, if not all, equipment can be entered or removed without necessitating window removal.

- Sufficient gloveports of standard design should be included to permit access to all essential equipment and surfaces. Gloveport gloves should be selected to withstand chemical properties and to minimize external dose rates, when needed, of the materials present in the enclosure.

- Interior and exterior surfaces smooth and free of cracks and crevices to simplify any needed decontamination.

- Enclosures are designed to control external dose rates, as necessary, to minimize worker exposures. This should include structural capability to install shielding materials and assess arrangement of process equipment within the enclosure.

- Number of penetrations for glovebox services should be minimized. Fittings must provide a positive seal to prevent migration of contamination. Penetrations for rotating shafts should not be

permitted except where there is no other alternative.

- Glovebox exhausts should be equipped with high-efficiency particulate air (HEPA) filter that can be changed without a containment break to minimize contamination loading in the exhaust ductwork. Inlets, if provided, should also be HEPA filtered.

- HEPA filters downstream from the glovebox/hood should be readily accessible to facilitate filter changes. The use of pre-filters is recommended.

- Only equipment essential to the function of the glovebox/hood should be inside of the glovebox.

- Equipment that will be located inside of the glovebox should be designed to minimize maintenance requirements. Equipment should also be designed (or chosen) to minimize contamination traps and to facilitate decontamination.

- Equipment should be designed to minimize sharp corners, barbs or pointed parts, and pinch points that could puncture glovebox gloves and skin. All corners must be rounded and burrs removed.

ALARA initiatives should considered in the design of the facility to help minimize exposure to workers during maintenance, decontamination and decommissioning activities include:

- Potential doses to personnel who work with or service radioactive systems can be reduced by removing as much radioactive material as possible prior to beginning work. Systems that require service should be designed, to the extent practicable, to include features that permit isolation from other system components and simplified decontamination procedures. System design should also take into consideration ultimate decontamination, decommissioning and disposal of contaminated components.

- The extent of decontamination required to permit work on radioactive systems or components can be minimized by limiting the size or volume of areas or systems that contain radioactive materials. Careful design is also necessary to avoid areas where radioactive materials may inadvertently accumulate. Piping systems are particularly susceptible to such problems. Design features should include, but are not limited to, avoidance of stagnant legs, location of connections above the piping center-line, use of sloping runs, providing drains at system low points, providing clean-outs at appropriate points, including provisions for flushing, and minimizing cracks and crevices both as a function of design and construction techniques.

- Decontamination requirements can be reduced by designing into systems features that increase the probability of containment of radioactive materials, reduce the amount of any release and minimize the spread of contamination beyond the source. These measures include installation and use of auxiliary ventilation systems and drainage control systems such as curbing and sloping floors to nearby drains or sumps to limit the spread of contamination from leakage of liquid systems.

- The potential for contamination of clean services (such as sanitary water supplies, nitrogen and

argon gas supplies) and supplied air for respiratory protection, as a result of leakage from adjacent contaminated systems, can be reduced by positive separation of clean systems from contaminated systems.

- Surfaces can be decontaminated more quickly and efficiently when smooth, nonporous, and free of cracks, crevices, and sharp corners. Desirable features can be ensured by providing adequate design instructions, purchase and construction specifications, careful attention to finish work during manufacture or construction, and using appropriate protective coatings on surfaces where contamination can be anticipated. A wide range of epoxy-based coatings are available for use in severe chemical environments. These materials offer good chemical resistance, low porosity, and are readily decontaminated. For exposed metal surfaces requiring decontamination, a surface finish of 125 micro-inches (RMS) is recommended.

- The careful evaluation and selection of materials for construction or lining components that are compatible with the process can reduce corrosion or erosion and facilitate decontamination when required.

- Filters or other serviceable components that could constitute significant sources of exposure must be provided with design features that provide for removal, installation, or servicing. Design features should include provisions for containment of contamination during servicing or replacement. Standardization of components is recommended.

- Potential doses from servicing valves and similar components can be reduced by specifying and installing reliable components for the required service and by using seals and gaskets which will withstand the environment to which they will be used. Valves should be installed in a stem-up orientation to simplify maintenance and minimize particle traps. System design features must be high reliability, good performance, and rapid, simple maintenance requirements.

- If mechanical seals are used on pumps, design features that permit using flush water to clean the seals should be evaluated.

- Adequate space should be provided in and around processing equipment to facilitate personnel and to ease replacement and removal difficulties associated with worn or failed components.

- Design should include adequate lighting, ventilation, workspace, accessibility, and other work-efficiency considerations.

- If specific design features are unproven, mock-ups, tests, and other methods for evaluation of concepts and equipment should be employed to verify designs. Sufficient space should be designed into a facility to permit such work.

- Equipment should be designed to permit visual inspection without the removal of protective coverings.

DOSE RATE PROFILE

Exposure to direct radiation produced by the decay of uranium daughter products and recycled by-product isotopes will contribute to the total effective dose equivalent (TEDE). External dose calculations were performed using the Grove Engineering software MicroShield version 5.03 to conservatively estimate exposure from gamma radiation. Using the proposed source term(s), a dose rate profile was developed for the various areas of the Oxide Conversion Building; First Floor (Figure 1) and Second Floor (Figure 2).

The OCB will be a controlled area and engineering controls, as well as administrative, will be utilized. The estimated dose rate profile will be updated based on actual, extensive measurements once the facility is operating. Each operational activity involving radioactive material will be evaluated and detailed written procedures prepared. Exposure control techniques will be incorporated into operating procedures to reduce exposure to ALARA. Also, personnel training and approved procedures will be provided to allow for safe operation and quick detection of any unfavorable trend so that appropriate corrective action can be implemented. The controls afforded by NFS' Radiation Protection Program for the OCB will maintain airborne radioactivity, contamination levels, and radiation levels within specified limits to protect against radiological hazards associated with operations.

Figure 1 OCB First Floor Dose Rate Profile

Figure 2 OCB Second Floor Dose Rate Profile