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21G-08-0145
GOV-01-55-04
ACF-08-0265
September 12, 2008

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) NRC Letter from Kevin M. Ramsey to B. Marie Moore, dated March 13, 2008, (TAC L32667), Nuclear Fuel Services, Inc., Acknowledgement of Receipt of Annual Update to Part II of SNM-124
3) NFS letter from B. Marie Moore to Director, Office of Nuclear Material Safety and Safeguards, dated January 30, 2008, Annual Update to Part II of SNM-124 Reflecting Changes Made During Calendar Year 2007 (21G-08-0026)

Subject: Redacted Version of Annual Update to Part II of SNM-124 Reflecting Changes Made During Calendar Year 2007

Dear Sir:

Per a request from your staff (Reference 2), Nuclear Fuel Services, Inc. (NFS) hereby submits a redacted version of the Annual Update to Part II of SNM-124 (Reference 3) suitable for public release.

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-08-0145) in any correspondence concerning this letter.

Sincerely,

NUCLEAR FUEL SERVICES, INC.

BMMoore

B. Marie Moore
Vice President, Safety and Regulatory

DMG/pdj
Attachments

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ATTACHMENT 1

Changes to Part II of SNM-124
Reflecting Changes Made During Calendar Year 2007

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PART II

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CHAPTER 9

GENERAL INFORMATION

A. General Information

9.1 Corporate Information

Nuclear Fuel Services, Inc., (NFS) has its principal offices in Erwin, Tennessee. Chapter 11 discusses the relationship and responsibilities of the Corporate organization as it relates to safety.

9.2 Financial Qualification

As a result of the transfer of ownership in 1987 of Nuclear Fuel Services, Inc., to NFS Services, LLC, from Texaco, NFS Services, LLC, was required to provide details to the Nuclear Regulatory Commission which demonstrate its financial capability to operate and decommission the Erwin facility. The financial arrangements to assure that decommissioning funds will be available are set forth in Chapter 7 and are reflected in the Contract language provided in Appendices A, B, and C of this Chapter.

9.3 Summary of Operating Objective and Process

Reference Sections 1.4 and 1.5 of Part I, which provide a summary of special nuclear material possession limits and authorized activities.

9.4 Site Description

Reference Chapter 3 of the Environmental Report (December 1996), which was approved by the NRC concurrently with the license renewal dated July 2, 1999.

Reference Chapter 3 of the Supplemental Environmental Report (November 2001) for a description of the plant expansion associated with construction and operation of BLEU Complex, which is comprised of the Uranyl Nitrate Building (UNB), the BLEU Prep. Facility (BPF), the Oxide Conversion Building (OCB), and Effluent Processing Building (EPB). These expanded operations were approved by the NRC in License Amendments #39 (Federal Register Vol. 56, No. 131, p. 45555, dated July 9, 2002), #47 (Federal Register Vol. 68, No. 207, p. 61235, dated October 27, 2003), and #51 (Federal Register Vol. 69, No. 117, p. 34198, dated June 18, 2004), respectively.

9.5 Location of Buildings on Site

Locations of buildings on the NFS plant site are shown and discussed in Chapter 10, "Facility Description."

9.6 Maps and Plot Plans

Figure 9.1 shows the location of the NFS plant site in relation to the state, the county, and the general environs. Figure 9.2 shows the plant layout.

9.7 License History

The license history is shown in Table 9.1.

Figure 9.1
Location Map of the Nuclear Fuel Services Plant, Erwin, Tennessee

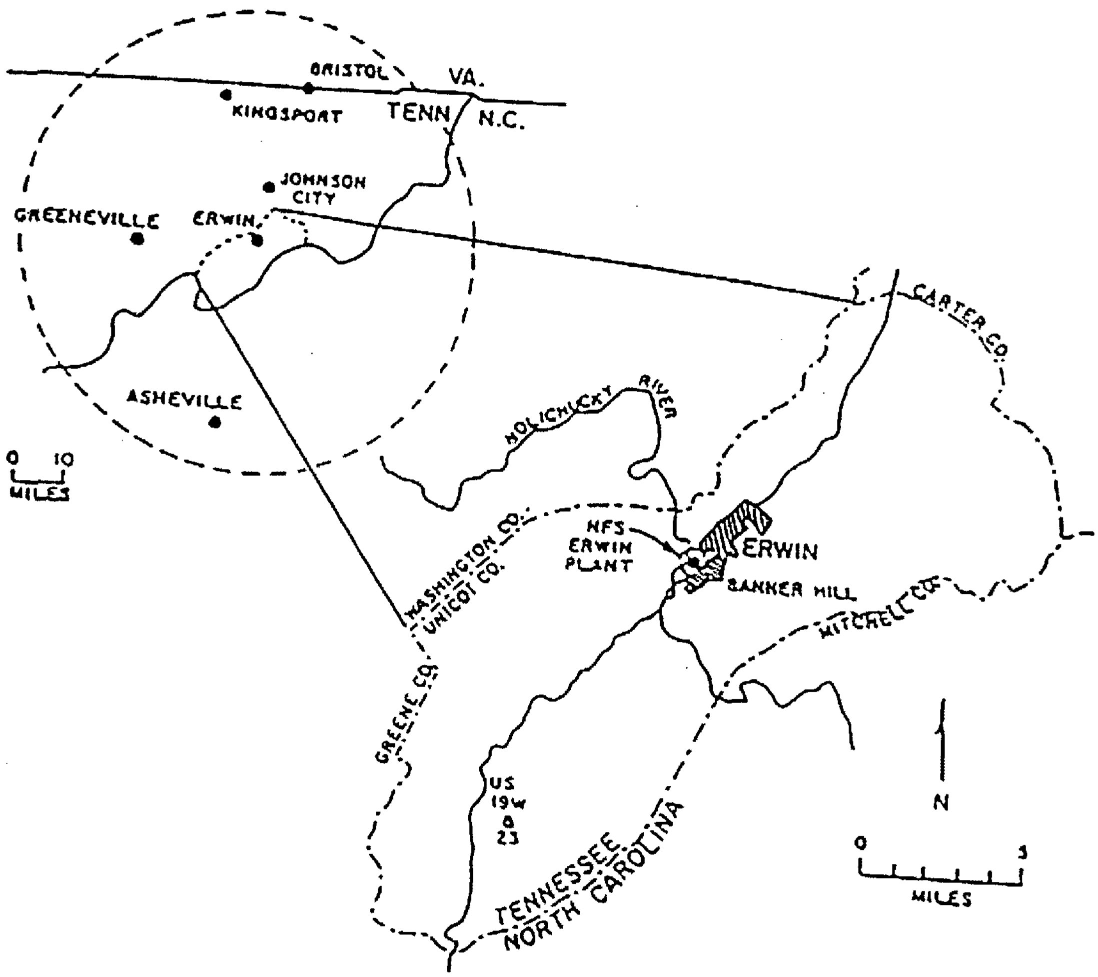


Figure 9.2
Plant Layout and Property Boundaries

Table 9.1
License History

SNM-124 was most recently renewed by the NRC on July 2, 1999. The following amendments have been issued subsequent to that renewal.

Amendment Number	Subject	Effective Date
1	Authorization to Operate KAST Fuel Process Areas 100-900, A-C, and Auxiliary Systems	08/03/1999
2	Authorization to Allow Use of QC Vault and to Delete License Conditions S-6 and S-7	02/04/2000
3	Authorization to Delay Conducting Physical Inventory Pursuant to 10 CFR 70.34	04/03/2000
4	Authorization to Delete License Condition S-13	04/03/2000
5	Authorization to Operate KAST Uranium Recovery Areas D-J	05/05/2000
6	Revisions to the Fundamental Nuclear Material Control Plan	05/16/2000
7	Authorization to Delay Conducting Physical Inventory Pursuant to 10 CFR 70.34	06/02/2000
8	Clarification of Possession Limits for Pu Residual Contamination, Special Air Sampling, and Internal Exposure Assessments	06/13/2000
9	Bulk Chemical Storage Tanks Analysis	07/03/2000
10	Authorization to Adjust Annual Limit on Intake (ALI) and Derived Air Concentration (DAC)	08/11/2000
11	Addition of Industrial Park Facility	09/13/2000
12	Authorization to Adjust Liquid Effluent Discharge Limits and NRC Correction of Previous Amendments	10/27/2000
13	Revision to Fundamental Nuclear Material Control Plan and Change to Safeguard Condition SG-4.16	11/30/2000
14	Revision of License Conditions S-39 and S-41	12/13/2000
15	Approval of NFS Site Security Training Plan, Revision 15, Safeguards Contingency Response Plan, Revision 0, and Emergency Plan, Revision 4	12/22/2000
16	Approval of Request for Time Extension to Conduct a Physical Inventory	01/15/2001
17	Revision of License Condition SG-6.1	01/30/2001
18	Revision of License Condition S-28	01/30/2001

Amendment Number	Subject	Effective Date
19	Revision of License Condition S-25	02/28/2001
20	Amendment to License Condition S-1	03/01/2001
21	Approval of Request for Time Extension to Submit the Physical Inventory Summary Report	03/26/2001
22	Deletion of License Conditions S-43 and S-44	03/26/2001
23	Authorization to Amend License Condition S-41 for Extension of Compensatory Measures from April 30, 2001 to June 30, 2001	04/24/2001
24	Deletion of License Condition S-20 and Review of 04/27/2001 Revised Safety Demonstration (S-27)	04/27/2001
25	Amend License Conditions for Safety Related Equipment	06/04/2001
26	Revision of License Condition S-22	06/04/2001
27	Approval of North Site Decommissioning Plan	06/19/2001
28	Revisions to HEU FNMC Plan, License Condition SG-5.1	06/27/2001
29	Authorization to Extend Safety Condition S-41 to July 31, 2001	06/29/2001
30	Authorization to Extend Deadline for Safety Conditions S-28, S-29, S-31, S-32, S-33, S-34, S-36, and S-37 to November 1, 2001	07/18/2001
31	Approval of ISA Plan and Deletion of License Conditions S-28 through S-38	10/30/2001
32	Deletion of License Conditions S-41 and S-45	02/22/2002
33	Revisions to HEU FNMC Plan, License Condition SG-5.1	03/29/2002
34	Approval of Emergency Plan, Revision 5	05/03/2002
35	Time Extension to Submit the Physical Inventory Summary Report	07/19/2002
36	Revised Fundamental Nuclear Material Control Plan	08/30/2002
37	Revised Appendix A to Chapter 5 of North Site Decommissioning Plan	03/31/2003
38	Authorization to Reduce Source Term at the Site Through Soil Removal	05/07/2003
39	Authorize Use of UNB and Increased Possession Limit	07/07/2003
40	Authorize Use of ICRP 68 Values	08/21/2003
41	Approve Time Extension to Perform Receipt Measurements	08/29/2003
42	Approve Time Extension to perform Receipt Measurements	09/15/2003
43	Approve Revision 4 to NFS Physical Protection Plan	10/10/2003
44	Approve Time Extension to Perform Independent Assessment of MC&A Program	10/24/2003

Amendment Number	Subject	Effective Date
45	Approve Exemption from Decommissioning Financial Assurance Requirements for Specific Equipment	11/13/2003
46	Approve Time Extension to Perform Receipt Measurements	12/31/2003
47	Authorize Use of BLEU Prep. Facility	01/13/2004
48	Approve Revisions to FNMC Plan	02/19/2004
49	Approve Organizational Changes to Chapter 2	03/13/2004
50	Approve Revisions to HEU FNMC Plan	05/25/2004
51	Approve Operation of the BLEU OCB/EPB	07/30/2004
52	Remove Sampling Requirements for Banner Spring Branch	09/13/2004
53	Approve Time Extension to Perform Receipt Measurements	10/15/2004
54	Administrative Change – Revision of Physical Protection Plan, Safeguards Contingency Plan, and T&Q Plan	10/29/2004
55	Approve Modification of Certain Material Inventory Measurements	11/05/2004
56	Approve Revision to FNMC Plan	12/08/2004
57	Approve Time Extension to Perform Receipt Measurements	01/10/2005
58	Approve Administrative Changes to Air Sampling and Bioasssay Programs	01/13/2005
59	Approve Deletion of License Conditions S-2, S-4, and S-5	01/28/2005
60	Approve Updated Schedule for North Site Decommissioning	02/29/2005
61	Approve Revised Date for Annual Update of Safety Demonstration Section	06/17/2005
62	Approve Possession Limit Increase	06/28/2005
63	Approve Revision 1 of the Physical Protection Plan	08/11/2005
64	Approve Changes to Certain Administrative Programs	08/24/2005
65	Approve Revisions to FNMC Plan	11/16/2005
66	Approve Changes to the Physical Protection Plan	11/28/2005
67	Approve Changes to Procedure Reviews by SSRC	12/12/2005
68	Approve Changes to FNMC Plan, and Replacement of Table 5.1	12/21/2005
69	Approve Final Status Survey Method for Subsurface Soils	02/15/2006
70	Approve Extension of Safeguards Condition SG-4.34	04/13/2006
71	Approve One-Time Exemption From Physical Inventory Deadline	06/06/2006

Amendment Number	**Subject**	**Effective Date**
72	Approve Change to Required Experience of Discipline Vice-President	07/03/2006
73	Approve Exemption of Low-Level Waste Shipments From Certain Physical Security Requirements	07/17/2006
74	Authorize Use of Shipper's Quantities to Resolve Shipper-Receiver Difference	08/08/2006
75	Incorporate Changes to Chapter 3	01/05/2007
76	Approve Extension of Safeguards Condition SG-4.34	04/11/2007
77	Approve Administrative Changes to Part I of SNM-124	05/09/2007
78	Partial Approval of Changes to Physical Protection Plan for Category I, High-Enriched Uranium	10/18/2007
79	Approve Increase in Possession Limit	11/23/2007

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124, Chapter 9
Appendix B
Revision 3
January 30, 2008

EXCERPTS FROM DOE PRIME CONTRACT

No. DE-SA09-01SR18976/TVA No. P-01N8A-249655-001

EXCERPTS FROM ARTICLE V FROM THE INTERAGENCY AGREEMENT BETWEEN THE DOE AND TVA FOR THE OFF-SPECIFICATION FUEL PROGRAM (DE-SA09-01SR18976/TVA NO. P-01N8A-249655-001)

Clause Article V from the DOE/TVA Contract DE-SA09-01SR18976/TVA No. P-01N8A-249655/001

Article V – DOE Delivery of Highly Enriched Uranium to TVA

(c) The Department of Energy (DOE) shall reimburse the Tennessee Valley Authority (TVA) for the Actual Cost(s) incurred to process and blend down the 7,400 Kg U of HEU in the form of U-Al alloy ingots and 5,900 Kg U of HEU in various chemical forms. Actual Costs shall include DOE's share of decontamination and decommissioning costs associated with the processing, recovery, and blend down of U-Al alloy ingots and the HEU in various other chemical forms. DOE's share of decontamination and decommissioning costs shall be limited to equipment and facilities used solely for processing and blending down of the 13,300 Kg U of HEU. TVA shall not finalize its contract with its contractor for this service until after TVA has submitted the relevant portion of the contract and price justification to DOE for review and has received DOE's written concurrence. TVA shall invoice DOE for the Actual Cost(s) incurred for such HEU processing, recovery, and downblending. DOE shall promptly pay such invoices within 30 days of receipt.

FACILITIES DESCRIPTION

A. Site Layout

10.1 Site Layout

The Nuclear Fuel Services (NFS) site is located in Erwin, Tennessee. The facilities within the site consist of numerous small buildings, the majority of which are within the Protected Area fencing. The administration buildings and guardhouses are glass/concrete structures and structures made of local brick; the process and process support buildings are predominantly constructed of precast concrete panels and white painted cement block. Metal "Butler-style" buildings are also used to house process support facilities such as respirator cleaning and testing, offices, decommissioning activities, and equipment and supplies storage.

Buildings within the plant have been designated with numbers and names as shown in Figure 9.2. Descriptions of the principal activities for each building are provided in the following sections. Buildings layouts and more detailed process descriptions are included in separate safety documentation submitted to the NRC.

10.1.1 Warehouse Facilities

Warehouse and material storage facilities include the Industrial Park Facility (IPF) Warehouse (see Appendix 10.1-A to this Section), Buildings 300, 310, 311, southeast portion of 304, south and east sections of 306, 135, 133, 132, and the UNB (Bldg. 510) located within the BLEU Complex. No unstable or radioactively contaminated chemicals are stored in these buildings such that release to the environs is probable.

10.1.2 Maintenance Facilities

The maintenance facilities reside in Buildings 120, 121, and the east section of 306. The plant's main maintenance facility is in Buildings 120 and 121.

10.1.3 Materials Staging and Medical Facility

Building 350 is a multi-function facility which includes medical facilities (e.g., medical records, examining rooms, Fitness-for-Duty testing facility, and emergency decontamination), the in vivo counting facility, and the shipping/receiving staging area.

10.1.4 Building 220

Building 220 contains offices and an analytical laboratory.

10.1.5 Service Building and NDA Laboratory

Building 100 contains change rooms, lunch rooms, First-Aid Station, office area, vending food storage, laundry facility, and NDA Laboratory.

10.1.6 Research and Development Laboratories

The Research and Development (R&D) Laboratories are located in Buildings 105, 110, and 131. These buildings contain facilities for conducting engineering studies and process evaluations for environmental remediation projects and for new business opportunities for NFS. These studies and evaluations may involve separating hazardous material and radioactive material, recovering resources from hazardous and/or radioactive waste, and treating hazardous and/or radioactive waste material. They may also involve R&D of chemical and radioactive material processing and manufacturing technology in support of new business development. These laboratories also conduct laboratory analyses in support of the engineering studies, as well as for customer and NFS process or waste materials. These activities are also licensed by the State of Tennessee.

10.1.7 Respirator Facility

Building 104 houses the respirator facility which includes a respirator laundry; an inspection, testing, and quality assurance area; a fit-test facility; and offices for individuals involved in these activities.

10.1.8 Building 111

Building 111 is used for the storage and staging of decommissioning materials in support of ongoing decontamination and decommissioning activities. The facility may also be used for the receipt, storage, and handling of materials separately licensed by the State of Tennessee.

10.1.9 Administration Buildings

Buildings 105, 130 (east annex), 120 (north end), 305, 320, and 345 house offices and computer facilities. Buildings 325 and 340 are the main security check points for vehicles and persons entering and leaving the site's Protected Area.

10.1.10 Central Analytical Laboratory

Building 105 and the northwest portion of Building 303 contain the plant's Central Analytical Laboratory. This laboratory handles samples from all plant processing facilities (HEU, LEU, natural U, and depleted U), scrap recovery facilities, waste water treatment facilities, and select environmental monitoring programs.

10.1.11 Plant Utilities

Building 130 houses non-radioactive plant utility services. This building contained uranium processes in the past, and covered fixed radioactive contamination exists. The utilities located in this building are compressed air, deionized water, and steam.

10.1.12 Production Facility

10.1.13 Waste Water Treatment Facility

Waste water treatment is conducted in Buildings 330 and 335. Building 330 contains the process for treating liquid effluents generated by the process facilities, R&D laboratories, laundry, decommissioning activities, and analytical laboratory. The effluents are treated to meet the requirements of a National Pollutant Discharge Elimination System permit issued to NFS by the State of Tennessee and 10 CFR Part 20. Treated effluent is discharged to the Nolichucky River. Building 335 processes groundwater from the site. The groundwater is treated to meet the requirements of a pretreatment permit issued to NFS by the Erwin Publicly Owned Treatment Works (POTW) and 10 CFR Part 20. Treated groundwater is discharged to the Erwin POTW.

10.1.14 Motor Pool (Bldg. 420)

Building 420 is for storage and maintenance of large earth-moving equipment.

10.1.15 Uranyl Nitrate Building (Bldg. 510)

Uranyl Nitrate liquid is stored in Building 510. This building contains 25 storage tanks, 1 receipt tank, and 1 uranyl nitrate tank for a capacity of approximately 270,000 gallons. Support facilities for unloading the Uranyl Nitrate liquid containers are part of this building. Liquid is stored until required by the Oxide Conversion Building (OCB) (Bldg. 520) process.

10.1.16 Guard/Security Building (Bldg. 540)

10.1.17 BLEU Preparation Facility (Bldg. 333)

The BLEU Preparation Facility (BPF) houses the processes to convert HEU materials to pure HE uranyl nitrate solution, a process to prepare blendstock (N uranyl nitrate solution), and a process to mix the HE uranyl nitrate and blendstock solution to form a LE uranyl nitrate solution (product). Building 333 also houses a uranium recovery system. The LE uranyl nitrate solution is transferred to the Uranyl Nitrate Building or to the LEU Dilution and Loading Facility after verification that the solution meets the product specifications.

10.1.18 Oxide Conversion Building (Bldg. 520)

Low-enriched uranyl nitrate liquids stored at the Building 510 are transferred to and converted into uranium oxides at Building 520. In addition, natural uranium oxide is dissolved in Building 520 and transferred to Building 510 for storage and shipping.

10.1.19 Effluent Processing Building (Bldg. 530)

Process waste streams generated at the OCB are treated at the Building 530 prior to discharge and/or disposal in accordance with applicable regulator requirements.

10.1.20 Security Ready Room (Bldg. 107)

10.1.21 CD Line Facility (Bldg. 301)

10.1.22 LEU Dilution and Loading Facility (Bldg. 440)

This facility will take LEU produced by the BLEU Preparation Facility and dilute it to customer specifications. It will also house an indoor loading bay where the diluted LEU will be loaded into container(s) for shipment to the customer.

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 10

FACILITIES DESCRIPTION

B. Utilities

10.2 Utilities

10.2.1 Electrical

10.2.1.1 Emergency Electrical Power

10.2.1.2 Re-establishment of Normal Electrical Service

10.2.2 Compressed Air

Two air compressors are centrally located within the main plant area, providing a total capacity of 1475 cfm of compressed air at 100 psig. One unit serves as a backup to the other in order to provide a constant supply. Two air compressors are located in the BLEU Complex (one in Building 510 and one in Building 520). The air compressor in Building 510 is rated at 70 cfm at 100 psig. The air compressor in Building 520 is rated at 70 cfm at 100 psig. Air exiting the compressors passes through receiving tanks, prefilters, desiccant air dryers, and after-filters prior to distribution.

10.2.3 Water Supply

Water is supplied to the NFS site from the Erwin Utilities system. Erwin Utilities is municipally owned and operated. The Erwin Utilities system consists of seven storage tanks having a combined storage capacity of 2,113,000 gallons. Three wells and one spring serve the system with a combined low flow generation capacity of 2,300,000 gallons per day. Another spring is available but not currently in use. The line pressure for the portion of the system serving the main plant area has been measured at 100 psig static. At 1400 gallons per minute flowing to the main plant area the residual line pressure is 50 psig. The line pressure for the portion of the system serving the BLEU Complex has been measured at 98 psig static. At 1220 gallons per minute flowing to the BLEU Complex the residual line pressure is 64 psig.

The Erwin Utilities system supplies the NFS site needs for process water, potable water, and fire protection water. The fire protection water systems at the site are described in Section 10.5 of this chapter. The emergency backup water supply for these systems is the Erwin Utilities system.

Figure 10.2 is a schematic of the water supply system for the NFS plant. The nonfire-protection connections to the Erwin Utilities system consist of three four-inch lines and one eight-inch line off the main at Carolina Avenue.

One of the four-inch lines serves Buildings 350, 325, and 320. One of the four-inch lines serves the remainder of the plant. One of the four-inch lines, entering the north end of the plant, serves the north site D&D office and change trailers. The eight-inch line serves the BLEU Complex.

At its entrance to each building, in the plant and to the BLEU Complex, the water supply is separated into two systems (potable water and process water). Potable water is restricted to drinking, sanitary needs, safety showers, and eye washes. Process water supplies process and other facility needs. Reduced pressure back-flow preventers preclude back flow of process water into the potable water system under any condition. The backflow preventers are tested upon installation and yearly thereafter.

Under normal conditions, the average daily water use is 43,000 gallons. Abnormally high usage could occur during activation of the fire protection systems.

Figure 10.2
City Water Lines

FACILITIES DESCRIPTION

I. Fire Protection

10.5 Fire Protection

10.5.1 Fire Protection Program

The NFS Fire Protection Program is designed, implemented, and assessed by meet the following objectives:

- Minimize the potential for the occurrence of a fire or related perils.

- Ensure that fire does not cause an unacceptable on-site or off-site release of hazardous material that will threaten the public health and safety or the environment.

- Establish requirements consistent with the National Fire Protection Association (NFPA) 101, *Life Safety Code*, that will provide an acceptable degree of life safety to NFS, Nuclear Regulatory Commission (NRC), and contractor personnel.

- Ensure that process control and safety systems are not damaged by fire or related perils.

- Ensure that essential programs or projects will not suffer unacceptable delays as defined by the senior program official as a result of fire and related perils.

- Ensure that property damage from fire and related perils does not exceed NFS established levels.

The Fire Protection Program responds to the following documents and criteria, where they are applicable: Southern Building Code Congress International, Inc., (SBCCI); NFPA Codes and Standards; and, Code of Federal Regulations (CFR), Title 29, Parts 1910 and 1926 (29 CFR 1910 and 1926).

The facility and processes have been insured for nuclear liability and property coverage by American Nuclear Insurers (ANI) and ACE, respectively. The Nuclear Liability Insurance Endorsement is maintained on file at NFS.

10.5.2 Administrative Controls

10.5.2.1 Fire Protection Plan

Details of the NFS fire protection plan, including organization, management responsibility, training, inspection, equipment testing, etc., are included in approved program documents and procedures.

10.5.2.2 Facility Audits

Facility audits and inspections are performed at two levels, as follows:

1) Routine audits and inspections are performed that include fire protection, combustible loading, and housekeeping status.

2) ANI and ACE are responsible for performing independent fire prevention, protection, and fire brigade inspections of the facility.

Corrective action plans are developed to address findings from these inspections and are tracked in the plant's Problem Identification, Resolution, and Correction System (PIRCS).

10.5.3 Building Design

10.5.3.1 Fire Areas

Fire areas subdivide specified processes or materials involving significant fire hazards, to confine the spread of fire to the area of origin. In particular, the building areas are considered and fire barriers provided as recommended by a fire hazard analysis.

10.5.3.2 Means of Egress

To enable rapid egress from the facilities in the event of a fire, NFS utilizes the applicable portions of guidance from NFPA 101, *Life Safety Code.*

10.5.3.3 Exposure Fire Risk

To minimize exposure fire risk, NFS employs the applicable portions of guidance from NFPA 80A, *Recommended Practice for Protection of Buildings from Exterior Fire Exposures.*

10.5.3.4 Nuclear Processing Facilities

Design and construction criteria for facilities for processing uranium include an evaluation to determine the proper methods to prevent, detect, extinguish, limit, and control fires and explosions. Fire-resistive and non-combustible materials are used as appropriate.

10.5.3.5 Flammable and Combustible Liquids

When the use of flammable or combustible liquids cannot be avoided, the reduction of risk and hazard from fires involving these materials is accomplished by the use of minimum volumes of liquids; by isolation with fire barriers; by isolated location; by provision for automatic fire suppression; and by installation of catch basins designed to minimize the spread of fluid leakage, as appropriate, for the fire hazard.

NFPA 30, *Flammable and Combustible Liquids Code*, is used as a guideline for the handling, use, and storage of flammable or combustible liquids.

10.5.3.6 Combustible and Flammable Gases

Systems using combustible and flammable gases are installed in accordance with the applicable NFPA standards. Ventilation systems exhausting combustible and flammable vapors at concentrations between 25 percent (25%) or greater of their lower flammable limit (LFL) and the upper flammable limit (UFL) are constructed of non-combustible materials in accordance with NFPA standards.

10.5.4 Water Supply

Plant fire protection water is provided through a loop system with two 8-inch connections to the City of Erwin Water Distribution System. Additionally, there are two dead-ended feeds. One is a 6-inch supply and the other is an 8-inch supply. Both of these are also connected to the City of Erwin Water Distribution System. All connections are arranged in an above-ground, heated vault with a double backflow preventer and double isolation valves.

Figure 10.13 gives the details of this system. The Erwin system details are discussed briefly in Section 10.2.3. Fire protection connections to the City system are independent of those providing water to the plant for other purposes. Post indicator sectional valves are provided, as appropriate, to isolate portions of the system should the need arise. The NFS system contains several fire department connections which allow tankers to pressurize all or certain portions of the system should the need arise.

10.5.5 Hydrants and Hose Houses

Numerous hydrants and hose houses are provided throughout the fire protection loop (see Figure 10.11). Locations are such that they allow ready access for quick use when needed to assist in fire fighting. In addition, several hose houses are provided on major process building roofs. Hydrants and hose houses are installed and equipped in accordance with the applicable NFPA standard. Two monitor nozzles are included on the fire protection loop to be utilized in the event of a need in the bulk chemical facility. Hose houses are not provided in the area of Bldgs. 510, 520, 530, and 540.

10.5.6 Fixed Fire Protection Systems

Fixed fire protection systems, including automatic sprinkler, carbon dioxide, and Halon systems, are utilized throughout the plant. Fire hazard analyses are performed to confirm the adequacy of the fire suppression system to respond to the maximum credible fire scenario. These systems are installed and periodically checked as required by applicable standards. Automatic systems are located in various facilities as presented in Table 10.1.

10.5.7 Fixed Fire Detection Systems

Several areas within the plant are equipped with fixed fire detection systems. Fire hazard analyses are performed to confirm the adequacy of the fixed fire detection system to respond to the maximum credible fire scenario. These systems include both smoke detectors and heat sensors. Heat sensors operate on both a fixed temperature and rate compensated basis. Detection systems are all equipped with alarming circuitry. NFPA 72, *National Fire Alarm Code*, is utilized as a basis for the design, installation, and testing of the fire detection and alarm systems. Fire detection systems located in various facilities are presented in Table 10.1.

10.5.8 Portable Fire Extinguishers

Portable fire extinguishers are located throughout the plant, as required by applicable standards.

10.5.9 Plant General Alarm

In addition to the various fixed protection and detection systems, the plant has a plant-wide manual pull-box fire alarm capability. All plantwide fire protection, detection, and alarm systems are integrated into the plant's central alarm panel which also serves the criticality monitoring system. As such, this system is equipped with an uninterrupted power supply.

The fire alarm tone provided by this system is different from the criticality evacuation tone. The alarm is sounded bi-monthly to keep all personnel aware of the different tones. When activated, the alarm can be heard throughout the plant. Fire brigade members are trained to respond to the fire alarm system.

10.5.10 Fire Brigade

NFS maintains an organized group of employees who are knowledgeable, trained, and skilled in basic fire-fighting operations. Training and education are provided for fire brigade members commensurate with those duties and functions that fire brigade members are expected to perform. Fire Brigade personnel do not respond to Bldgs. 510, 520, 530, and 540. Fire response to the BLEU Complex is solely provided by the City of Erwin Fire Dept.

Fire-fighting equipment is provided for the fire brigade and maintained in a plant fire brigade ready room to assure the safe operational condition of the equipment.

10.5.11 Pre-Fire Plans

NFS maintains pre-fire plans that provide information needed by fire fighting personnel responding to the emergency. The pre-fire plans are located for ready access for the plant fire brigade and local fire departments which may respond to an emergency at NFS.

10.5.12 Fire Hazard Analyses

Fire Hazard Analyses (FHAs) are developed to address the requirement for facility-related FHAs as outlined in NFPA 801, "*Standard for Fire Protection for Facilities Handling Radioactive Materials*." These documents, which are prepared for NFS facilities which are involved in the processing or storage of nuclear materials, evaluate the current levels of fire protection and life safety features present in the building. FHAs are also prepared for non-nuclear facilities which may present an unusual fire risk. FHAs, which are developed in conjunction with an outside fire protection engineering consultant, generate recommendations which are prioritized and tracked until resolved.

Additionally, these documents are revised in response to facility modifications which may impact fire risk.

Table 10.1
Fixed Fire Suppression and Detection System Location Summary

Table 10.1
Fixed Fire Suppression and Detection System Location Summary

Figure 10.13
Fire Protection Waterlines

ORGANIZATION AND PERSONNEL

A. Functional Organization

11.1 Organizational Responsibilities

Nuclear Fuel Services, Inc. (NFS), owner and operator of the nuclear fuel manufacturing facility located in Erwin, Tennessee, is a subsidiary of NFS Services, LLC. The NFS corporate organization provides the management, administrative, and technical capabilities to direct the development and operational aspects of the Erwin plant. The site organization has responsibility for operating the plant in a safe and efficient manner. This responsibility is implemented through the functional disciplines of production, decommissioning, engineering, safety, material control and accountability, security, and quality assurance, which are described in the sections which follow. Table 11.1 identifies the NFS senior managerial positions and individuals that have been assigned the responsibilities of President, Discipline Vice President/Director, and Discipline Manager, as described in Chapter 2 within these functional discipline descriptions. Figure 11.1 shows the current NFS functional organization. Table 11.1 provides a cross-reference between the functional positions described in Figure 11.1, the NFS organizational position responsible for each function, and the individuals currently assigned responsibility for each function. The qualifications for the individuals provided in Table 11.1 are included in Appendix A of this chapter.

11.1.1 Production

The Production Discipline is responsible for production related activities involving the handling and processing of special nuclear material, including developing operating procedures and maintaining facilities and equipment in a safe operating condition. This function includes activities associated with product research and development, research and development laboratory operations, analytical laboratory operations, process engineering, enriched uranium processing, transportation and waste management, and nuclear fuel production equipment installation and start-up. This function manages a majority of the hourly work force, and has line management responsibility for implementation of the safety programs and systems for conducting an active ALARA Program.

11.1.2 Decommissioning

The Decommissioning Discipline develops plans for the decommissioning of facilities and equipment, writes and obtains approval of procedures to conduct decommissioning, obtains any special equipment and/or facilities needed for decommissioning, and assures that decommissioning activities are conducted in accordance with approved documents and in the spirit and intent of ALARA.

11.1.3 Engineering

The Engineering Discipline designs and installs new and modified facilities and equipment; supplies maintenance and process engineering support, and assures that all equipment and facilities have appropriate safety controls and have been evaluated within the spirit and intent of ALARA. The Engineering Discipline is responsible for configuration management.

11.1.4 Safety

The Safety Discipline provides programs, procedures, and reviews to oversee and assure site safety in the areas of nuclear criticality safety, radiation safety and protection, industrial safety, chemical safety, fire safety, environmental protection, and ALARA, and to monitor operations to ensure they are conducted in compliance with federal, state, and local regulations. The Safety Discipline is responsible for the Safety and Safeguards Review Council (SSRC).

11.1.5 Material Control and Accountability

The Material Control and Accountability Discipline maintains programs to assure that SNM is received, processed, stored, and transferred in accordance with federal regulations, and implements these functions through the areas of SNM safeguards, SNM accountability, shipping, receiving, and warehousing.

11.1.6 Security

The Security Discipline provides on-site security forces which control access to protected and material access areas; administers facility and personnel security clearance programs and protects against material and equipment theft and unauthorized personnel entry.

11.1.7 Quality Assurance

The Quality Assurance Discipline approves systematic programs for indoctrination and training of personnel performing quality-related safety activities, for specifying during the design phase the extent of quality assurance or confidence necessary for quality-related safety structures, systems, and components, and for validation of computer software used to calculate or develop data or process control of safety-related items.

11.2 Functions of Key Personnel

Key personnel are those individuals who are responsible for safety and for safe operation of the site. Key personnel include the president, the senior managers of the disciplines described in Section 11.1, and the individuals responsible for the safety functions described below. Company policy requires written delegation of authority when senior managers are unavailable to perform their duties. The emergency plan delineates responsible management personnel and reporting relationships for handling site emergency situations.

11.2.1 President and/or CEO

The president and/or CEO have overall responsibility for the safety, security, quality, and operational aspects of all activities conducted at the NFS site.

11.2.2 Discipline Vice President/Director

The discipline vice president/director functions have the delegated responsibility for plant safety and for compliance with conditions of this license and NRC regulations in order to maintain a safe work place for all employees.

The safety discipline vice president is responsible for ensuring that plant operations comply with all regulatory requirements of governmental agencies and good practices in the areas of safety, and material control and accountability.

The production discipline vice presidents/directors are responsible for the safe and efficient implementation of activities affecting site operations, including quality assurance, security, engineering, research and development, uranium operations, maintenance, decommissioning, transportation, waste management, training, and laboratory operations. The production discipline vice president has the overall responsibility and authority for the configuration management program.

11.2.3 Discipline Manager

The discipline manager function is responsible for the safe operation, control, and quality of activities in their designated areas, and for the safety of the environs as influenced by the activities conducted therein. The discipline manager function establishes written operating procedures, incorporating safety and quality controls and limits commensurate with the particular operations involved. Discipline managers are assigned to each of the disciplines described in Section 11.1.

11.2.4 Safety Functions

The following sections describe the responsibilities of key personnel within the safety disciplines, specifically those individuals with responsibilities in the areas of licensing and compliance, nuclear safety, radiation safety and protection, environmental safety, industrial safety, chemical safety, and fire protection.

11.2.4.1 Safety Discipline Manager

The Safety Discipline Manager function has senior safety management responsibilities in the areas of radiation safety and protection, industrial safety, chemical safety, fire safety, environmental protection, nuclear criticality safety, and emergency preparedness, and for performance of safety and hazards analyses associated with new and re-started equipment, systems, and operations. The Safety Discipline Manager is responsible to ensure programs and procedures are developed and implemented to assure effective implementation of a safety program that is protective of the workers and the general public, that is in keeping with the ALARA principle, and which complies with the license and applicable federal, state, and local regulations. The safety discipline manager function is also a member of the Safety and Safeguards Review Council.

11.2.4.2 Nuclear Safety

The Nuclear Safety manager function oversees the performance of criticality safety analyses to establish safe batches, geometries, concentrations, and spacing of special nuclear materials and equipment. The Nuclear Safety manager function provides authoritative, professional advice and counsel to discipline managers on matters of control against accidental criticality, and develops and establishes a criticality safety control program and measures the effectiveness of the program by reviewing the application of methods and data to actual plant situations derived through audits and inspections. The Nuclear Safety manager function approves new and changed equipment and facilities with criticality safety implications during the design phase and the installation thereof prior to operation, and has responsibility for conducting educational programs in criticality control matters. Within this function, separate and independent dual criticality safety reviews of new and modified operations, equipment, and facilities are performed. The Nuclear Safety manager function has authority to immediately suspend any operation which involves nuclear safety practices believed to threaten the health and safety of employees or the public. The Nuclear Safety manager function reviews criticality safety analyses performed by junior and senior members of the function, and reports to the Safety Discipline Manager function.

11.2.4.3 Nuclear Safety Senior Member

The nuclear safety senior member is administratively free of production responsibilities and performs criticality safety analyses to establish safe batches, geometries, concentrations, and spacing of special nuclear materials and equipment. The nuclear safety senior member provides authoritative, professional advice and counsel to discipline managers on matters of control against accidental criticality, and performs inspections and audits of operations to determine compliance with operating procedures and the criticality safety limits and controls imposed therein. The nuclear safety senior member reviews criticality safety analyses performed by junior members of the function, instructs other members of the safety function in methods used in criticality safety audits and inspections and may serve as an ISA team member. The nuclear safety senior member function reports to the Nuclear Safety manager function.

11.2.4.4 Nuclear Safety Junior Member

The nuclear safety junior member is administratively free of production responsibilities and performs criticality safety analyses to establish safe batches, geometries, concentrations, and spacing of special nuclear materials and equipment. The criticality safety reviews performed by the nuclear safety junior member are reviewed by either the Manager or the Senior Member of the function. The nuclear safety junior member performs inspections of operations to determine compliance with operating procedures and the criticality safety limits and controls imposed therein and may serve as an ISA team member. The nuclear safety junior member reports to the Nuclear Safety manager function.

11.2.4.5 Industrial Safety

The Industrial Safety manager function is responsible for establishing safety programs for non-radiological hazards work place monitoring, including industrial safety, industrial hygiene, chemical safety, fire prevention and protection, and respiratory protection. The Industrial Safety manager function performs analyses of new, changed, or existing systems, equipment, and operations as they related to non-radiological hazards, and provides assistance in employee training in general safety and non-radiological material hazards matters. The Industrial Safety manager function reports to the Safety Discipline manager function.

An Industrial Safety specialist function supervises the conduct of the respiratory protection, fire protection, chemical safety, and industrial safety surveys, audits, inspections, and measurements. The Industrial Safety specialist function provides advice and counsel to discipline managers and their staffs on matters of industrial safety and control. The Industrial Safety specialist function performs inspections and audits of operations to determine compliance with the limitations and controls established for industrial safety purposes, and may serve as an ISA team member, providing fire protection and chemical safety expertise. The Industrial Safety specialist function reports to the Industrial Safety manager function.

11.2.4.6 Health Physics

The Health Physics manager function establishes a radiological safety program which includes formulation of radiation protection criteria; analyses of new, changed, or existing systems, equipment, and operations; and recommendations for improvements. Health Physics manager function is responsible for the ALARA program. The Health Physics manager function designs programs to gather timely information on the adequacy of material containment as it relates to contamination and airborne radiological hazards, and audits plant conformance with radiological safety criteria to assure compliance with the license and applicable state and federal regulations. The Health Physics manager function reports to the Safety Discipline manager function.

11.2.4.7 Health Physicist

The Health Physicist function performs radiation safety analyses and evaluations of processes and equipment within the facility. The Health Physicist function daily reviews radiation monitoring data and consults with the appropriate discipline managers' staffs on problem areas and the corrective action required, and provides authoritative, professional advice and counsel to discipline managers on matters of the control of radiation exposure. The Health Physicist function performs inspections and audits of operations to determine compliance with operating procedures and the controls established for radiation safety purposes, and serves as an ISA team member. The Health Physicist function reports to the Health Physics manager function.

11.2.4.8 Radiation Monitoring

The Radiation Monitoring manager function manages the radiation monitoring function and is responsible for implementing and overseeing the conduct of the radiation monitoring and surveillance program. The Radiation Monitoring manager function is responsible for ensuring that: qualified personnel are assigned to perform radiation surveys and that radiation monitoring and surveillance is conducted in accordance with approved procedures; radiation survey meters and counting instruments used in the radiation monitoring program are properly maintained and calibrated; and radiation monitoring records are properly maintained. The Radiation Monitoring manager function reports to the Health Physics manager function.

A Radiation Technician supervisor function is responsible for the assignment of technicians on each shift who conduct radiation measurements and surveys and perform scheduled safety inspections. The radiation technician supervisor function is responsible for the maintenance and calibration of radiation survey meters and counting instruments, and reports to the Radiation Monitoring manager function.

11.2.4.9 Environmental Safety

The Environmental Safety manager function establishes an environmental protection program which includes formulation of environmental protection criteria and a measurement system; analyzes new, changed, or existing systems, equipment, and operations as they related to gaseous or liquid effluents; and, provides and implements a liquid and gaseous effluent monitoring program. The Environmental Safety manager function audits conformance with environmental protection criteria; interprets and determines applicability of federal, state, and local regulations to plant operations with respect to environmental protection concerns; provides advice and counsel to discipline managers on environmental matters; and provides assistance in employee training in environmental matters. The Environmental Safety manager function reports to the Safety Discipline manager function.

An environmental scientist function is administratively independent of production responsibilities and performs analyses and evaluations of proposed changes in facility, process, or equipment changes as they relate to environmental protection. The environmental scientist function provides authoritative professional advice and counsel to discipline managers and their staffs on matters of environmental protection. The environmental scientist function performs reviews of environmental monitoring data to determine compliance with the limitations and controls established for regulatory compliance purposes, and may serve as an ISA team member. The environmental scientist function reports to the Environmental Safety manager function.

11.2.5 Quality Assurance

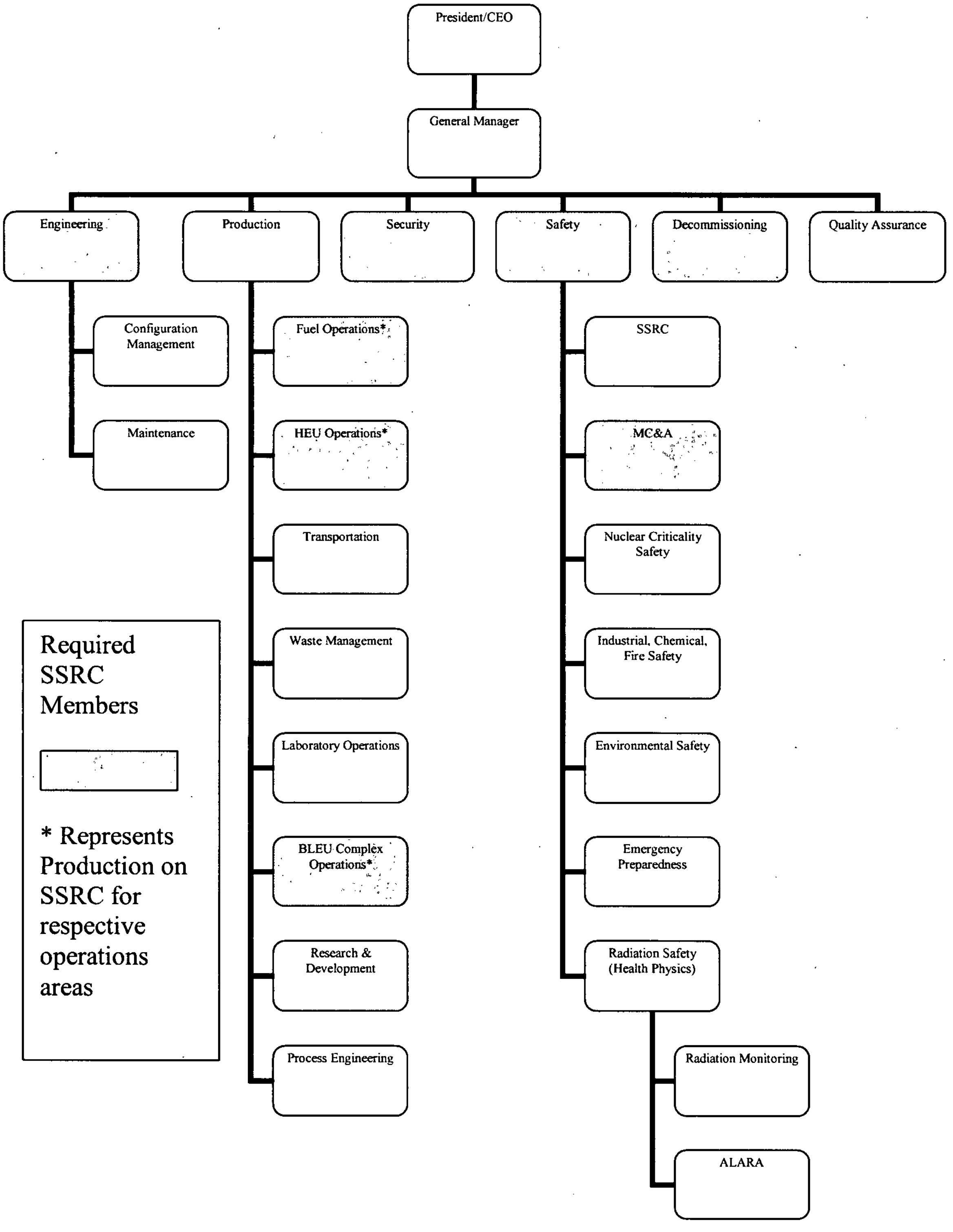
The Quality Assurance manager function is responsible for the management and implementation of the quality assurance program. The quality assurance program provides for the planning and accomplishment of quality-related safety activities under suitable controlled conditions, including the use of appropriate equipment, suitable environmental conditions for accomplishing the activity, and assurance that prerequisites for a given activity have been satisfied. The quality assurance organization oversees the quality assurance program to provide additional assurance that specified quality-related safety activity requirements have been met, accomplished through audits, surveillance, and assessment of quality-related safety activities. The Quality Assurance manager function has direct access to the President/Chief Executive Officer.

11.3 Responsibilities, Education, and Experience of Key Personnel

Table 11.1
Cross Reference of Functional Positions to NFS Organization and Individuals

* Some NFS positions and/or individuals may be responsible for more than one functional position.

Figure 11.1: Functional Organization Chart



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APPENDIX A
CHAPTER 11

RESUMES

RADIATION PROTECTION

A. Administrative Program Requirements

12.1 Program

The NFS programs for conducting radiation surveys, including those used to monitor personnel exposures, contamination control and those ensuring that exposures are ALARA, are described in various written procedures. Implementation of the following program documentation assures that program objectives are met:

- Safety Procedures (including "A," "B," "E," and "GH" procedures)
- Support Group Procedures (including laboratory and training procedures)
- Standard Operating Procedures (SOPs)
- Letters of Authorization (LOAs)
- Radiation Work Permits (RWPs)
- NFS ALARA Program Document

12.1.1 Safety Procedures

12.1.1.1 "A" Procedures

"A" Procedures are primarily for safety supervisory personnel and deal with administrative and technical aspects of the safety monitoring programs. Examples of the subjects addressed in "A" procedures are:

- The bioassay program, including investigating results above plant action limits
- Instrument calibration, including laboratory and portable radiation measuring systems
- Radiation Technician training and qualification
- Ventilation system performance testing
- Data reduction techniques for both in-plant and environmental samples

- Conduct of safety audits and inspections

- Safety document standards and control

- Inspection of emergency equipment and supplies

- Off-site dose calculation

- Respiratory protection

- Work injury classification

12.1.1.2 "B" Procedures

"B" Procedures are primarily for hourly Safety personnel and deal with the inspection of safety systems, collection and analysis of samples, and conduct of surveys to support the various Safety programs. Examples of subjects addressed in "B" procedures are:

- Radiological surveillance and monitoring

- Radiological posting

- Sample collection and analysis for the in-plant effluent and environmental monitoring programs

- Inspection of radiological safety equipment

- Industrial safety/hygiene monitoring of the workplace

- Instrument repairs

12.1.1.3 "E" Procedures

"E" Procedures are emergency plan implementing instructions. They detail the duties and responsibilities of various plant personnel in the event of an emergency. Examples of subjects addressed in "E" procedures are:

- Plant emergency evacuation

- Emergency radiological monitoring both on- and off-site

- Emergency communications

- Fire fighting
- Hazardous material spill cleanup and containment
- Emergency off-site dose estimates
- Emergency contamination control
- Specific instructions to individuals with emergency responsibilities

12.1.1.4 "GH" Procedures

"GH" Procedures establish general policy and expectations for the safety programs which are applicable plant-wide or to several disciplines. Examples of subjects addressed in "GH" procedures are:

- Plant-wide contamination control
- Protective clothing, including the use of respiratory protection
- Treating and reporting work injuries
- Radiation Work Permits
- Collection of bioassay samples
- Reporting radiation exposure summaries
- External radiation monitoring

12.1.2 Other Procedures

Support group procedures, Standard Operating Procedures, and Letters of Authorization are addressed in Chapter 11.

12.1.3 Radiation Work Permits

RWPs are used to delineate radiological controls and special safety precautions that must be taken when work is to be done of a non-routine nature where exposure to radiological hazards is considered credible. RWP controls and job site/work evolution are reviewed with the job supervisor prior to beginning work. This review normally includes a visual inspection of the work site to determine the appropriateness of proposed controls. RWPs are approved by a Health Physicist or a Radiation Technician Supervisor.

RWPs are issued for all work on or involving entry into a system where radioactive material is present and there exists a potential for generation of significant airborne radioactivity, surface contamination, or personnel exposure, unless the operation or maintenance work is covered by an effective operating procedure. The RWP specifies the nature and location of the work, and the necessary safety controls, as appropriate, including personnel monitoring devices, protective clothing, respiratory protective equipment, special air sampling, and additional precautionary measures to be taken.

The individual responsible for the nonroutine work is responsible for obtaining an RWP. The individual requesting the RWP is also responsible for signing the RWP and assuring that only personnel who have completed required safety training are assigned to perform work under the RWP.

A copy of the RWP, listing any specific radiation safety precautions, is posted in a conspicuous location throughout the duration of the activity; and the work shall be monitored by a member of the radiation safety and protection function.

Upon completion of the work under the RWP, the individual responsible for the work and radiation safety and protection function are responsible for assuring that the RWP is properly terminated to allow the work area to be returned to normal conditions. The completed RWP is sent to the radiation safety and protection function for filing. RWPs are kept for a minimum of two years.

12.1.4 NFS ALARA Program Document

The NFS ALARA Program Document provides specific guidance for ALARA philosophy implementation. The Program Document was developed, utilizing the guidance provided in Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Reasonably Achievable." The measures to implement the NFS ALARA Program are discussed in detail in Section 12.11.

12.2 Posting and Labeling

NFS is granted an exemption from the requirements of 10 CFR 20.1904(a). Instead, each entrance into the plant security fence shall be posted:

CAUTION
RADIOACTIVE MATERIALS
EVERY CONTAINER OR VESSEL
WITHIN THIS AREA MAY CONTAIN
RADIOACTIVE MATERIALS

This posting at the entrance to the plant security fence also satisfies the posting requirements of 10 CFR 20.1902(e) for the entire plant area.

Areas are classified and posted for other radiological hazards as appropriate.

Determination of the area postings is made by the radiation safety and protection function. The radiation safety and protection functions routinely inspect for proper postings.

12.3 External Radiation – Personnel Monitoring

Dosimetry devices provided, and processed by a NAVLAP accredited vendor, are utilized at NFS for monitoring individual external radiation exposure. These devices (typically thermoluminescent dosimeters[TLDs]) provide the dose of record. Self reading dosimeters may be used in specific areas as an ALARA tool.

Individual dose monitoring is provided based upon the radiation safety and protection function evaluation of the individual's potential for exposure. Beta/gamma-sensitive dosimetry is provided for individual monitoring and exchange at specified frequencies. The range of these monitoring devices is 10 millirem to approximately 1,000 rem.

Where appropriate, as determined by evaluation of the specific operations, dosimetry may be used for monitoring extremity exposure.

Exposure results are monitored and evaluated by the radiation safety and protection function. Appropriate investigative action is taken if the exposure exceeds predetermined action guides. The circumstances are determined; and corrective actions are taken, where necessary, to minimize, to the extent reasonable, further exposures above action guides.

12.4 Radiological Surveys

Survey procedures have been developed, implemented, periodically reviewed, and, as needed, amended to reflect changing circumstances. These surveys include three categories of monitoring, each of which is further subdivided into three distinct types.

12.4.1 Monitoring of the Work Place

12.4.1.1 Routine Monitoring

Routine monitoring is intended to show that the working environment is satisfactory for continued operations and that no change has taken place calling for reassessment of operating procedures. It is largely of a confirmatory nature. At NFS the routine work place monitoring program includes, where appropriate:

- Surface contamination surveys performed on a specified frequency at various locations throughout active and inactive process areas or other controlled areas.

- Routine exposure rate surveys performed at specified locations and frequencies.

- Continuous work station air sampling at fixed locations.

12.4.1.2 Operational Monitoring

Operational monitoring is intended to provide a check on a particular operation and to give, if necessary, a basis for immediate or future decisions on the conduct of the operation. At NFS, the operational work place monitoring program includes, where appropriate:

- Operational contamination surveys required to adequately assess conditions during a special or nonroutine operation.

- Continuous alarming type air monitors.

- Operational monitoring of individuals through the use of breathing zone air samplers.

- Special exposure rate surveys to evaluate area radiation levels.

12.4.1.3 Special Monitoring

Special monitoring may cover either a situation in the working environment where insufficient information is available to achieve adequate control or an operation which is being performed under abnormal circumstances that could include accident potential. Special monitoring is intended to provide more detailed information to identify the problems and to define future procedures. Special monitoring, therefore, has limited duration, clear-cut objectives, and is terminated in favor of appropriate routine or operational monitoring once the objectives have been achieved. At NFS, the special work place monitoring program includes:

- The sampling of airborne materials through the use of special, short-duration high-volume air samplers.

- The collection and analysis of samples from the fixed air sampling system at other than the normally scheduled time.

- External exposure surveys performed at appropriate locations to characterize the extent of a problem.

- Special contamination monitoring at sufficient locations to adequately characterize an area.

- Special collection of process ventilation duct samples, where provided and if applicable to the circumstances.

- Readings from the criticality monitoring or area radiation monitoring systems.

12.4.2 Individual Monitoring

Individual monitoring includes the making of measurements by equipment carried on the person of workers and/or measurements of quantities of radioactive materials on or in their bodies or excreta, and the interpretation of those measurements.

12.4.2.1 Routine Monitoring

Routine individual monitoring consists of regularly repeated or continuous measurements made on an individual worker. In cases where routine individual monitoring techniques or instrumentation are not capable of facilitating the estimates of dose equivalent or intakes for individuals with the necessary confidence, programs of monitoring of the work place may be used to provide estimates of the relevant values. At NFS, the routine individual monitoring program includes:

- Utilizing, where indicated, bioassay analyses and interpretation, including urine, and in vivo conducted at regular intervals.

- Utilizing dosimeters, where indicated, worn by individuals to provide an estimate of external radiation levels.

- Routine monitoring for contamination on the skin and/or clothing.

12.4.2.2 Operational Monitoring

Operational monitoring of an individual is similar to work place operational monitoring in that it is intended to provide a check on a particular operation or to give additional information which is used for future planning. The focus is, however, on the individual. At NFS, the operational monitoring program for individuals includes:

- Utilizing breathing zone air samplers to assess intake potential for individuals working on nonroutine operations or cases where the work place stationary air samplers are not considered representative of the work environment.

- Nasal, saliva, urine, and/or fecal samples collected from individuals, as well as in vivo chest counts, when action limits are exceeded or whenever deemed necessary by the Radiation Safety and Protection function.

- Lung solubility and particle size studies conducted to provide information on these parameters, which is in turn used in the interpretation of bioassay results.

12.4.2.3 Special Monitoring

Special individual monitoring may be conducted during actual or suspected abnormal conditions, including accidents, and may include the following:

- Special bioassay samples collected during the period following a known or suspected upset condition at a frequency that allows assessment of individual intake.

- In vivo counting as close as possible in time to the event.

- Evaluation of dosimeters as soon as practicable.

- Evaluation of indium foils and/or induced radioactivity in the body or personal items in the event of a criticality accident.

12.4.3 Environmental Monitoring

Environmental monitoring is addressed in Chapter 13, Section 13.3.

12.4.4 Survey Data

Survey data are examined for significant trends by Radiation Safety and Protection personnel. From these analyses, individual aspects, as well as the overall safety program, may be evaluated for their effectiveness and appropriateness.

12.5 Records and Reports

Records appropriate to criticality and radiation safety activities, occupational exposure of personnel to radiation, releases of radioactive materials to the environment, and other pertinent activities are maintained in such a manner as to demonstrate compliance with commission license conditions and regulations.

Records associated with ALARA findings, employee training, personnel radiation exposures, and environmental activities are generated and retained in such a manner as to comply with the relevant requirements of 10 CFR 20. See Table 12.1 for a more comprehensive listing.

Records related to safety results discussed in this chapter are periodically validated and microfilmed for permanent storage, when required.

Reports are made in accordance with internally established requirements and procedures. Formal reports are issued in accordance with the requirements of 10 CFR 20 and other applicable regulations.

Table 12.1
Records and Their Minimum Retention Time

Type Record	Minimum Retention Period
Individual radiation exposure	Until disposal is authorized by the NRC
Surface contamination surveys	Three years
Radiological safety training	Period of employment plus 3 years
Instrument calibration	Three years
Environmental surveys	Until disposal is authorized
External radiation surveys	Three years
Process changes and additions	Five years
Radiation work permit	Two years
Radiological and environmental safety analysis	Life of project plus 6 months (2-year minimum)
Accident investigations (involving releases or exposure)	Until disposal is authorized
Audits and inspection reports	Two years
Radiological exposure trends (including ALARA findings)	Two years
Safety review committee meetings	Five years
Equipment and material release surveys	Three years

RADIATION PROTECTION

B. Instrumentation

12.6 Instrumentation

12.6.1 General

An adequate number of radiation detection instruments is available to ensure that proper radiation surveys can be performed. Selection criteria for portable and laboratory counting equipment are based on the types of radiation detected, maintenance requirements, ruggedness, interchangeability, and the upper and lower limits of detection. The radiation safety and protection function reviews the types of instruments being used for each monitoring function and makes appropriate recommendations based upon regular input and ongoing evaluation.

12.6.2 Instrument Types

Table 12.2 summarized the radiation detection instruments employed at NFS. It must be noted that while representative, the list is not all inclusive.

Furthermore, upon industry development, the instruments may be upgraded or replaced with other equipment having comparable or superior operating characteristics.

12.6.3 Equipment Storage, Maintenance, and Calibration

Radiation detection equipment is stored and made available for routine use at various plant locations, such as the radiation monitoring laboratories, controlled zone exits, change rooms, and other designated locations. Additional emergency equipment is stored and made available in designated site emergency locations as specified in the Emergency Plan and the implementing procedures developed in support of the plan.

Maintenance and calibration are provided at specified frequencies in several dedicated facilities including electronics engineering, maintenance function, and safety function. These services may also be provided by offsite vendor contracts.

Monitoring instruments utilized for routine radiation safety purposes are calibrated before initial use, after major maintenance, and on a routine basis in accordance with manufacturer's recommendation following the last calibration.

Table 12.2
Typical Radiation Detection Instruments/Systems Used at NFS

Fixed Installation Equipment	**Model**
Criticality Warning System (GM type)	Eberline RMS-3
Area Radiation Monitor	Eberline RMS-3, Eberline ECX-4, Ludlum Model 375
Continuous Air Monitor	Canberra Alpha Sentry
In vivo lung counter (Canberra Industries)	Canberra Industries Custom System
Kinetic Phosphorescence Analyzer for Uranium Urinalysis	Chemchek Instruments KPA-11A
Portable Contamination Instrumentation	**Model**
Alpha Contamination Surveys	Ludlum Model 3, 4, 2221, or 2224 with either 43-5 or 43-90 probes
Personnel monitoring (scintillation, GM, and gas-flow proportional type instruments)	Eberline RM-14, RM-18, RM-19, RM-20, RM-25, or Ludlum Model 177 with Ludlum 43-5 probes (alpha monitoring) or GM pancake probe (Beta-Gamma Monitoring), Aptec Alpha-7 Hand and Foot Monitor
Beta-Gamma Contamination Surveys	Ludlum Model 3, 4, 2221, or 2224 with GM pancake probe.
Portable Exposure Rate Instrumentation	**Model**
Neutron	Eberline E600 or ASP2e with BF3 sphere, Ludlum 12-4 with He3 sphere
Beta/Gamma (GM type)	Eberline E-520 with HP-270 probe, Ludlum Model 78, Ludlum Model 3 with 44-38 probe
Beta/Gamma (Ion Chamber type)	Eberline PIC-6A/6B, Eberline RO-2, RO-2A, RO-20, Ludlum Model 9, Victoreen 451P-RYR
Gamma (scintillation)	Ludlum 2350 with NI detector
Gamma (self reading dosimeter)	Rados RAD-60R
Laboratory Instrumentation	**Model**
Automatic low background alpha/beta proportional counting system	Tennelec LB 5100, Tennelec LB-4100, Protean WPC-9550
Automatic alpha/beta dual phosphor system	Protean ASC-DP
Manual alpha/beta counting system	Eberline SAC-4, Ludlum Model 2929, Ludlum Model 3030

NOTE: Eberline is now Thermo Fisher Scientific.

The accuracy of calibration sources should be, as a minimum, ±5 percent of the stated value and traceable to the National Institute of Standards and Technology. Calibrations of each instrument will include, where applicable, two points separated by at least 50 percent of each linear scale, or with a calibration at one point near the midpoint of each decade or logarithmic scales. A survey instrument may be considered properly calibrated when the instrument readings are within ±10 percent of the calculated or known values for each point checked. Readings within ±25 percent are considered acceptable if a calibration chart or graph is prepared and attached to the instrument.

Background and source checks are performed daily for laboratory counting instruments during periods when the equipment is in use except for environmental sample counting that employ long counting times. Efficiency is determined using radioactive sources of known activity.

12.6.4 Criticality Detection System

The NFS criticality detection system is consistent with the requirements of 10 CFR 70.24. Monitoring is performed with GM detector systems.

The criticality detection system consists of two essential parts: the readout module and the detector. The detector collects a charge caused by incident radiation. This charge is then conditioned and transmitted via multiconductor cable and displayed on the readout meter.

A calibration check is performed on a semi-annual basis. In addition, at least one detector pair will be response tested monthly on a rotating basis. Periodically, the alarm is sounded for familiarity, training, or drills.

To meet regulatory requirements in 10 CFR 70.24 and to assure a limited number of false alarms, the system is set up with two detectors at each detector location. Alarm actuation is caused by both detectors at a location exceeding their alarm trip point, or by a single detector failure coupled with the second detector in alarm, which results in a plant-wide evacuation and worker accountability. Detector or other electronic component failure will result in a warning signal. This signal will initiate contingency measures which may include evacuation of personnel, suspension of operations, deployment of auxiliary monitoring equipment, and/or immediate system repair.

Detector locations and system configuration are subject to modification as necessary to maintain adequacy of coverage. This determination is made by the Safety Department.

The alarm controls have limited access. Manual initiation of the alarm is provided for testing. A warning signal is generated at the central control unit in the event of a system malfunction. Provisions are incorporated into the alarm system to allow appropriate testing and remote readouts are present at manned posts that will alert personnel in the event of component failure.

The system is demonstrated to respond to a minimum criticality accident of concern. A criticality accident producing an absorbed dose in air of 20 rads at 2 meters within one minute is the limiting accident considered for the demonstration of the system response.

The compliance of the system is demonstrated by accounting for shielding from plant materials between a postulated accident and the detectors, as well as distance. The accident is evaluated from a number of locations to demonstrate the possible effects of attenuation. Common modeling codes are used to perform the evaluations such as Microshield and MCNP. Compliance is demonstrated if modeling results indicate that the postulated minimum accident of concern will result in an exposure rate exceeding the alarm set-point at a detector location.

Figure 12.1

DELETED

RADIATION PROTECTION

D. Exposure Monitoring

12.10 Occupational Exposure Analysis

As an appendix to this chapter, NFS provides an analysis of occupational exposures (external and internal). The analysis includes data from at least the past two years of plant operation for each plant area and type of operation performed. It has been developed utilizing the guidance provided in Regulatory Guide 3.52 (Rev. 1, November 1986), "Standard Format and Content for the Health and Safety Sections of License Renewal Applications for Uranium Processing and Fuel Fabrication."

12.11 Measures Taken to Implement ALARA

Nuclear Fuel Services, Inc., is committed to the philosophy of ALARA. That commitment is manifested in:

- A published Radiation Safety policy, signed by the President of NFS, declares, to all employees, the policy and intent of NFS to maintain exposure as low as reasonably achievable.

- NFS has developed a formal written ALARA Program Document, approved by senior level managers, which implement the NFS policy by:

  (a) Requiring training in ALARA philosophy for all radiation workers

  (b) Requiring the development, approval, and implementation of specific ALARA goals for selected operating units and the designation of an ALARA Coordinator, as appropriate, for each group to review the progress toward the attainment of specific ALARA goals

  (c) Requiring the measurement and monitoring of progress toward goal achievement and the issuance of regular progress reports to management and supervision

  (d) Requiring the performance of specific ALARA reviews during the design phase of engineering projects for new facilities or facility and/or equipment modification

  (e) Defining, as appropriate, specific long-term ALARA goals

(f) Establishing an ALARA technical review committee composed of the SSRC (Safety and Safeguards Review Council) to review all proposed facility modifications and their ALARA evaluations, operating procedures, and ALARA reports

(g) Requiring a periodic report of radiation and other safety-related monitoring and audits to appropriate levels of management together with recommendations on methods for lowering exposures, both occupational and environmental

(h) Requiring the analysis of monitoring data for trends which might indicate an increase in radiation exposures

(i) Conducting a periodic audit of the ALARA program implementation

(j) Requiring routine inspections of operating areas focused on implementation of radiological controls

- NFS has appointed an ALARA Health Physicist, within the Radiation Protection organization, with responsibility for overseeing and coordinating the ALARA Program

12.12 Internal Exposure Monitoring

12.12.1 General

The primary objective of the internal radiation monitoring program is to assure that significant internal radiation exposures are detected, properly evaluated, and recorded. The internal radiation monitoring program, including bioassay procedures, is designed to ultimately express measurements in terms of estimated dose (e.g., DAC-hrs, CEDE).

To accomplish this objective, monitoring of both the working environment and workers is required. Breathing zone air samplers and representative fixed air sampling are used as the primary means of determining intakes for workers. Bioassay measurements, when they possess the necessary sensitivity, may be used as an overcheck of the air sampling program and may be used to make adjustments or additions to an individual worker's dose record.

The sensitivity of a particular bioassay procedure is a function of body metabolism of the radionuclide, its route of entry into the body, and the exposing conditions (i.e., acute versus chronic exposure). Directly related factors are lung solubility of the material, particle size, the overall sensitivity of the laboratory used to analyze bioassay samples, and the time(s) after exposure the bioassay sample is collected and analyzed.

These variables disallow the establishment of internal action guides for exposure control based on bioassay results per se. Rather, action guides are based on an interpretation of each bioassay result.

The NFS internal exposure monitoring program currently utilizes the IMBA Expert Computer Program developed by the UK National Radiological Protection Board and ACS and Associates, Inc. However, NFS reserves the right to modify these programs or adopt alternate programs that have equivalent or superior capabilities upon industry development.

The computer program relies on International Commission on Radiological Protection (ICRP) models which estimate intakes from the interpretation of bioassay results. The estimated intake can then be compared to internal action levels and to the ALI. Also, the dose to the worker is estimated.

The model structure is based upon Reference Man models summarized in ICRP Publications. Intake pathways considered include inhalation, ingestion, instantaneous uptake, and delayed uptake through a wound.

Intake retention functions based in ICRP Publication 68 dose models are used in the design and operation of the NFS bioassay program including:

- the identification of those bioassay procedures that have sufficient sensitivity and accuracy for the detection of appropriate internal action levels,
- the determination of derived investigation levels (DILs).
- the determination of the frequency of monitoring required, to ensure the detection of an internal action level, and
- in cases involving accidents, the determination of special bioassay procedures that can be used to confirm or make better estimates of the intake and other dose estimates over time intervals appropriate to the specific case.

12.12.2 Capabilities

On-site capability exists in dedicated facilities for the analysis of urine samples, nasal smears, and work place particle size determinations. An on-site in vivo chest counter was installed and operational in 1987. Contract laboratories are utilized, where appropriate, for urine and fecal isotopic analysis, lung solubility determinations on samples from the NFS work place, and quality assurance sample checks on the NFS urinalysis laboratory.

Typical minimum detectable amounts are listed:

	U-233	U-235	Plutonium(1)
Urinalysis	0.04 μg/l	0.04 μg/l(2)	0.5 DPM/l
Fecal Analysis	0.1 DPM/g	0.5 DPM/g	0.1 DPM/g
In Vivo Lung Count	N/A(3)	0.2 nCi	0.5 nCi

(1) MDA is specific to the radionuclide in the mixture, or as in the case of lung counting, the daughter Am-241.
(2) Based on Kinetic phosphorescence analysis of total uranium analysis performed on-site. (0.3 μg/l based on natural dietary intake.)
(3) Dosimetry based on the more sensitive urine or fecal analysis.

12.12.3 Bioassay Frequencies

Routine bioassay frequencies are determined as outlined in Table 1 of Regulatory Guide 8.34 and guidance given in Regulatory Guide 8.9. When measurement capability is a limiting factor, frequencies are increased. Participants and types of bioassays are determined by the radiation safety and protection function based on work assignments and review of exposure history.

Urinalysis is the preferred technique for soluble (Class D/F) radioactive material work areas, while in vivo and fecal analyses are relied upon more heavily for insoluble (Classes W/M and Y/S) radioactive material work areas. Lung solubility determinations at work stations are based on either actual measurement or the classification in Appendix B to 10 CFR 20. These classifications are based on the theoretical reaction products at a particular work station and are used for planning purposes in the routine bioassay program. For significant exposure evaluations, solubility is determined from a series of bioassay measurements, when feasible.

Operational bioassay measurements are required as outlined in Section 12.4.2.2. Special bioassays are collected or in vivo measurements made to adequately assess intakes as outlined in Sections 12.4.2.3, 12.8.1, and 12.8.2.

12.12.4 Quality Control of Other Programs

A secondary objective of the bioassay program is to provide a quality control check to assure adequate protection of workers from internal radiation exposure. As such, bioassay results are periodically used to verify the validity of the work place air monitoring program and the effectiveness of the respiratory protection program. When bioassay-based exposure estimates indicate exposures are approximately equal to or less than those generated from the air monitoring program, then the air monitoring program is considered adequate. Respirator use protection factors are applied as appropriate. This program is separate from the other validity checks on the air sampling program discussed later in this chapter.

12.12.5 Work Restrictions

When significant exposures occur or are suspected, in addition to other actions required by this license and NRC regulations, work restrictions are imposed:

- Diagnostic restriction means a reassignment of an individual to a position or work area to minimize the potential for additional exposure which would complicate the exposure evaluation process. Once the radiation safety and protection function has adequate samples/information to assign an estimate of the exposure to an individual, he/she may be allowed to return to a normal work assignment.

- Regulatory restriction means a reassignment of an individual to a position or work area with significantly lower exposure potential for the remainder of the reporting period in which the exposure occurred. This type of restriction usually follows a diagnostic restriction and is provided to allow adequate control of individual exposures below the NRC reporting limit in 10 CFR 20.1201.

An indication from any of the safety monitoring programs that an exposure above 200 DAC-hrs may have occurred after applying decay and respiratory protection factors, if applicable, is cause for diagnostic restriction. An assigned exposure greater than or equal to the limits set forth in 10 CFR 20.1201 results in a regulatory restriction. In the event a measurement indicates an intake of an individual is equal to or exceeds 10 milligrams of Class D/F uranium ($\leq$2.4 μCi/gU specific activity) in a week, a medical restriction is imposed.

Internal exposures are assigned to the calendar year in which the exposure event occurred.

12.13 Air Sampling

12.13.1 Airborne Radioactivity in Work Areas

The control of radioactive materials in restricted areas is effected by means of equipment design, containment, and associated ventilation.

Processing of radioactive materials in which significant potential for release of airborne contaminants exists is conducted in a drybox or hood with sufficient ventilation to minimize the release of radioactivity. When a system fails to perform in such a way as to maintain applicable specifications, prompt corrective action is instituted to minimize exposure of personnel to the lowest practicable levels.

In general, the limits set forth in 10 CFR 20 will be reached or exceeded only under abnormal circumstances. Design objectives, corrective actions, management responses, etc., are made within the framework of the "as low as reasonably achievable" concept.

12.13.2 Air Monitoring Systems

To verify the effectiveness of the containment capabilities, surface smear and airborne radioactivity surveys are conducted on a routine basis, the frequency of which is dependent on the potential for radioactivity release. A number of air monitoring systems exist at the NFS Erwin facility to monitor work area exposures/concentrations and to detect unsafe concentrations.

12.13.2.1 Stationary Air Samplers (SAS)

Continuous air sampling of process work areas for airborne alpha and/or beta radioactivity is performed by drawing air through a particulate filtering or collection media with a known collection efficiency and measured periodically by counting the filter media with a low background gross alpha/beta counter. Each air sampler is coupled to an air-flow-rate meter so that the volume of air sampled can be accurately determined.

Stationary air samples in areas where annual intakes are likely to exceed 10% of the ALI are collected every operating work shift. Other active air samples are collected at a lower frequency in accordance with written procedures, based on the potential for exposure to occur.

12.13.2.2 Continuous Air Monitors (CAMs)

A number of airborne alpha and/or beta monitors may be positioned in various plant areas to identify airborne problems as they occur, if deemed necessary by the radiation safety and protection function. These instruments are equipped with a particulate filter and solid state detector. Whenever airborne activity could result in the exposure of an individual to greater than 40 DAC-hrs in a day, the monitor activates an alarm.

When such an alarm occurs, workers in the area are required to evacuate or wear respiratory protection equipment until the high level alarm is investigated and resolved.

12.13.2.3 Lapel Samplers (BZAs)

The primary purpose of lapel air sampling (BZA) is to monitor personnel exposure to airborne radioactivity. They are also used to determine the representativeness of the SASs. When used to monitor personnel exposure, the sample filters are routinely counted prior to the individual's next scheduled work shift.

All wearers are instructed in the proper use of lapel samplers. Depending upon the analytical results of the lapel sampler filter, the wearer may be required to complete a questionnaire, submit a bioassay sample, and in some cases have nasal smears performed (see Section 12.8 for action limits).

12.13.2.4 High-Volume Sampling

Immediate assessment of airborne radioactivity levels are made with high-volume air samplers using filter media or impactor heads. The samples are promptly counted for gross alpha activity. The resulting information is important for recommending respiratory protection or other necessary measures.

12.13.3 Quality Assurance/Quality Control (QA/QC) Considerations

In the event stationary air samplers are used for assigning exposure, the following QA/QC steps will be taken to verify the representativeness of work area air sampling. This is accomplished by comparing data generated from the SASs to data generated by lapel samplers worn by operators performing work in the area under consideration.

If the lapel or stationary sampler result does not exceed the value excluded by Table 1 of Regulatory Guide 8.25, dated June 1992, no further test is performed and the stationary air samplers are ruled representative. If this excluded level is exceeded by the lapel or stationary sampler, the ratio of the stationary air sample result to the lapel sample result must exceed the value of 0.5 for the stationary air sample(s) to be ruled representative. The results from more than one shift may be averaged to make this determination.

Other QA/QC methods are used, including periodic equipment calibrations, daily source and background checks, to assure proper operating characteristics. These practices are documented and audited to assure that all duties are performed according to procedures.

12.13.4 Action Guidelines

Action points for various air sampling systems are provided in Section 12.8. Reports are also part of the actions initiated by elevated sampling station data.

Because airborne radioactivity can be a significant source of radiation exposure at the NFS Erwin facility, a summary report of all individual plant air samples which exceeds the DAC, corrected for decay, is prepared and circulated to plant management at least monthly. Problem areas are identified and, if known, the cause of increased airborne radioactivity is given.

Metrics on airborne radioactivity performance are maintained and reported to management on a regular basis. Problem areas are identified and recommendations for reduction of airborne radioactivity levels are made as necessary.

The design objective of process equipment and confinement is to maintain the average airborne radioactivity concentrations at less than 25% of the appropriate DAC value. Whenever airborne concentrations at any work station exceed 25% of the appropriate DAC value as averaged over a work week, and no cause has been identified, the work station is investigated, including the equipment in use, operator work habits, ventilation effectiveness, etc. Such investigations and the corrective action taken, or initiated, are documented.

Confirmation that any work station average airborne radioactivity concentration as averaged over a work shift (or over the sampling period, in areas where less frequent samples are collected), is in excess of the DAC initiates the following actions:

- Confirmation of the continued existence of airborne radioactivity in the area through short-term high-volume air sampling. Determine the number and identify of personnel who may have been exposed.

- If confirmed, posting of the room, area, or building with signs indicating the need for respiratory equipment (including type and instructions to all occupants to use it).

- Investigate to determine the sources of airborne radioactivity.

- Initiate appropriate corrective action to control further releases of radioactivity.

- Suspend routine operations if the airborne radioactivity concentration exceeds 100 times the DAC. Corrective action is initiated and documented for routine operations. Non-routine operations performed under an RWP requiring respiratory protection could continue if adequate measures are in place and approved by the areas health physicist on a case-by-case basis.

12.13.5 Particle Size Adjustment

For inhalation exposures, NFS may elect to adjust Derived Air Concentrations (DACs), Annual Limits of Intake (ALIs), and Committed Dose Equivalents (CDEs to specific organs) which are based on 1-micron AMAD (Activity Median Aerodynamic Diameter) particles using the results of particle activity-size distribution measurements. The method of obtaining such measurements, analyzing data, and applying results is described below.

Particle (activity) size distribution measurements will be taken using an Anderson Marple Model 296 Personal Cascade Impactor (or equivalent). Typically this impactor is operated at a flow rate of 2 liters/minute and consists of 6 impactor stages and a back-up filter. The typical particle size range collected by each stage in micrometers (um) is as follows: >9.8 um; 6.0 to 9.8 um; 3.5 to 6.0 um; 1.55 to 3.5 um; 0.93 to 1.55 um; and, 0.52 to 0.93 um. The size range for the back-up filter is 0. to 0.52 um.

If NFS chooses to adjust DACs, ALIs, and CDEs, at least three particle size samples will be collected and analyzed for radioactivity for each grouping of locations. The locations for which particle sizing data analysis will be performed will be limited to Inhalation Class W/M or Class Y/S process areas. The analytical results will be averaged for each impactor stage and for the back-up filter to determine a single average measured activity-size distribution to use in subsequent analyses described below. Particle size analysis will be performed at least semi-annually in each group of locations for which particle size credit is taken. After one year, the frequency may be relaxed to annually if data for a group of locations does not differ significantly from previous measurements. Particle size will be reassessed following significant process changes deemed likely to change the particle size distribution. Using the results of particle size measurements and knowledge of the process, a Health Physicist will decide whether specific operations or specific locations can be grouped together for characterization purposes.

The method of analysis used to determine the particle size consists first of specifying the parameters (i.e., a weight, geometric mean, and geometric standard deviation) for up to four underlying lognormal sub-distributions. Then by changing parameters of each underlying lognormal distribution (i.e., weight, geometric mean, and geometric standard deviation), a numerical iteration technique is employed to minimize the sum of the squared differences between the predicted and measured activity on each stage. In this numerical technique, the geometric mean will be constrained to values < 20 μm; and the geometric standard deviation (GSD) will be constrained to values between $1.1<GSD<2.5$. As an indication of the goodness-of-fit for the final predicted (hypothesized) distribution as compared to the measured distribution, the chi-square statistic is used. The chi-square test statistic is calculated assuming n-2 degrees of freedom, where n is the number of stages including the back-up filter (i.e., $n = 7$). If a statistically good fit exists (as indicated by the p-value), the fractional activities represented by each underlying distribution and their associated AMADs (equivalent to the geometric mean for a lognormal distribution) are used to determine an appropriate adjustment factor to apply to the associated 1-micron based DAC, ALI, and CDE values in accordance with equations in Appendix B to this chapter. The level of confidence required to demonstrate goodness-of-fit will be "p" greater than or equal to 0.8, where "p" is the probability of obtaining a value equal to or less than the calculated chi-square statistic when the hypothesized distribution is true. If this level cannot be achieved, the data will be discarded and additional data taken. However, if additional data cannot be fitted (i.e., "p" is also less than 0.8), then a conservative analysis will be used to determine a particle size distribution and the analysis will be documented.

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SNM-124, Chapter 12
Appendix A
Revision 15
January 30, 2008

Analysis of Exposures

Internal and external doses are reported for four (4) categories of workers: NFS employees, Murray Guards (security force), contractors, and visitors. The regulatory limit is five (5) rem per year total effective dose equivalent (TEDE) for occupational workers, and 0.1 rem for nonoccupational workers (visitors). The sum of their external [deep dose equivalent (DDE)] and internal [committed effective dose equivalent (CEDE)] equals the TEDE. These results are summarized in Tables 12A-1 through 12A-6 for the five year period of 2002 to 2006.

External Radiation Exposures

Table 12A-1 shows the DDE values for all monitored personnel, which are used in computing the total individual exposures. There were four (4) individuals for the five year period whose DDE exceeded five hundred (500) mrem, which is ten percent (10%) of the total exposure limit for a calendar year; none of these doses exceed 1 rem. Total (collective) external exposures are lower for 2006 when compared to 2005 and higher than the period of 2002-2004. The increase in collective exposure is attributed to an increase in the amount of recycled high enriched uranium (which has a higher external dose component) stored and processed at the plant site in support of the BLEU Project, (which involves material with a higher external dose component compared to past materials). The maximum dose values for the reporting period show a measurable decrease over 2005 experience. Average dose values show a decrease from 2005. Table 12A-2 shows DDE results by individual work groups. The number of employees slightly increased compared to the previous reporting period; their total (collective) dose decreased; their maximum value decreased; and their average values decreased. The number of security guards and contractors decreased slightly. External radiation exposure for these groups generally decreased over previous experience. Visitor dose was insignificant.

Internal Radiation Exposures

The internal radiation exposure, due to an intake of radioactive material, is monitored as committed dose equivalent (CDE) and committed effective dose equivalent (CEDE). An intake is determined by a combination of lapel air sampling data, stationary air sampling data, time in area, and respiratory protection factors. Adjustments to doses are allowed by SNM-124 (as of CY 2000) to account for particle size effects in certain process area handling Class W and Class Y enriched Uranium. In addition, new dose conversion factors from ICRP 68 were implemented during the fourth quarter of 2003. Exposure from an intake is confirmed by monitoring individuals in the bioassay program, which consists of in vivo lung counting and in vitro urinalysis methods.

Table 12A-3 shows the CEDE values for all monitored individuals, which are used in computing the total individual exposure. The collective dose for 2006 was significantly lower than 2005. Maximum and average internal doses for 2006 showed a decrease. The maximum CEDE for 2006 was 158 mrem with no individuals over 500 mrem. Table 12A-4 shows CEDE by individual work groups. Collective CEDE, maximum CEDE, and average CEDE for 2006 were generally lower than 2005 values for the individual work groups. Visitor dose was insignificant for the period.

Total Radiation Exposures Summary

Regulations limit the annual radiation dose of occupational workers to 5,000 mrem from the total of internal and external exposure. The total effective dose equivalent (TEDE) is used to measure compliance with this limit and is calculated as the sum of the DDE and the CEDE for each individual.

The TEDE results calculated for 2002 through 2006 are shown in Table 12A-5 and Table 12A-6. These results indicate a significant decrease in collective TEDE for 2006, and still much lower than previous years (2002-2004). It is notable that both maximum TEDE and average TEDE decreased for 2006. The maximum TEDE for 2006 was 586 mrem and the average TEDE was 29 mrem. The decrease in exposure was primarily the result of a temporary shutdown of SNM operations due to a labor dispute and temporary shutdown of BPF operations due to a regulatory compliance issue. An aggressive ALARA program was also implemented due to the storage and processing of TVA material which caused increased radiation levels during processing and storage.

Bioassay Program

NFS' bioassay program is divided into routine and special programs. The routine monitoring program is employed to verify air sampling results – the data normally used to assign worker's dose. Routine sampling frequencies vary and individuals may be on a weekly, bi-weekly, monthly, quarterly, or other schedule. The special bioassay program is used when air sampling indicates a potentially significant intake (typically greater than 8 DAC-hrs) may have occurred. Results of special bioassay measurements are used to evaluate air sample data and determine dose to the impacted individuals.

The bioassay program provides for urinalysis, fecal analysis, and lung counting. Of these, urinalysis and lung counting are normally performed on site using equipment purchased or contracted for that purpose. Urinalysis is performed using kinetic phosphorescence techniques and lung counting is performed using gamma spectrometry techniques. Fecal analyses are performed infrequently with the samples being analyzed by an off site contracted laboratory using alpha spectroscopy techniques.

Translation of bioassay results into dose is a process determined on a case-by-case basis and depends on the exposure scenario and the chemical and physical characteristics of the material to which the individual was exposed. The health physicist evaluates each situation in which a bioassay result is both greater than the MDA and greater than the individual's previous result. Some individuals working at NFS have historically high body burdens and only when there is an increase above the previously detected levels, signifying a recent intake, does the health physicist become involved. The IMBA internal dose assessment program (which utilizes models for reference man as summarized in ICRP publications) is used to determine the individual's intake based on bioassay results. Each investigation resulting from bioassay measurements is documented and placed in the employee's permanent exposure file maintained at NFS.

Urinalysis

Urinalysis are performed using the Chemchek Kinetic Phosphorescence Analyzer (KPA, model KPA-11). The KPA works by using laser pulses to create measurable characteristic phosphorescence from the uranium atoms in a sample. This technique has a typical detection sensitivity of 0.03 μg uranium/L which equates to approximately 0.2 dpm/L and 6 dpm/L for 5% enriched uranium and 95% enriched uranium, respectively. Background uranium concentrations in urine have been measured and a decision level at the 95% confidence level has been set at approximately 0.3 μg uranium/L. When in use, the KPA is calibrated on a weekly basis and typical QA/QC samples (blanks and spikes) are analyzed with every batch of samples.

In Vivo Lung Counting

Lung counting is performed using the Canberra lung counter (LC) system which employs four low-energy, high-purity, germanium detectors. The system has a large library and can provide quantification of gamma emitters in the 10 keV to 400 keV energy range. To be counted, a subject enters the shielded steel room of the LC which is constructed of a minimum 6 inches of steel with a lead/copper liner. The detectors are placed on the chest wall of the subject (two closely spaced detectors over each lung) and the subject is typically counted for 30 minutes. The LC has typical minimum detectable activities (MDAs) for U-235, Th-232, Am-241, and Pu-239 of approximately 0.12 nCi, 0.16 nCi, 0.15 nCi, and 90 nCi, respectively, for a nominal 22mm chest wall thickness. The LC is calibrated annually by Canberra using the humanoid systems phantom. When in use, a quality control check is performed daily using special radioactive sources. Background checks are performed weekly and all data generated by the LC are reviewed by Canberra at the end of each monthly reporting period. Any increase (above the MDA) in the detected amount of radionuclides in the lungs initiates actions by the LC operator and Health Physicist.

Table 12A-1
External Radiation Monitoring Deep Dose Equivalent (DDE)
Overall Site Summary

Group	2006	2005	2004	2003	2002
0.00	656	744	505	450	392
0.00-<0.10	471	343	697	930	720
0.10-<0.25	66	118	95	42	49
0.25-<0.50	8	37	3	2	2
0.50-<1.00	0	4	0	0	0
1.00-<5.00	0	0	0	0	0
>5.00	0	0	0	0	0
Total individuals	1,201	1,246	1,300	1,424	1,163
Total dose (rem)	29.532	45.193	28.877	24.458	19.207
Maximum dose (rem)	0.449	0.928	0.288	0.383	0.278
Average dose (rem)	0.025	0.036	0.022	0.017	0.017

NOTE: Rows 1-7 provide the number of individuals within the specified DDE group/range.

Table 12A-2
External Radiation Monitoring Deep Dose Equivalent (DDE)
Individual Group Summary

Group	2006	2005	2004	2003	2002
Employees (#)	**708**	**702**	**715**	**767**	**750**
Total dose (rem)	25.850	41.269	25.357	21.372	16.841
Maximum dose (rem)	0.449	0.928	0.288	0.383	0.278
Average dose (rem)	0.037	0.059	0.035	0.028	0.022
Murray Guards (#)	**166**	**174**	**189**	**164**	**109**
Total dose (rem)	1.662	1.594	1.603	1.489	0.909
Maximum dose (rem)	0.055	0.054	0.112	0.093	0.131
Average dose (rem)	0.010	0.009	0.008	0.009	0.008
Contractors (#)	**248**	**248**	**273**	**322**	**182**
Total dose (rem)	2.020	2.296	1.917	1.584	1.457
Maximum dose (rem)	0.158	0.203	0.138	0.119	0.162
Average dose (rem)	0.008	0.009	0.007	0.005	0.008
Visitors (#)	**79**	**122**	**124**	**167**	**117**
Total dose (rem)	0.000	0.034	0.000	0.013	0.0
Maximum dose (rem)	0.000	0.034	0.000	0.012	0.0
Average dose (rem)	0.000	0.000	0.000	0.000	0.0

Table 12A-3
Internal Radiation Monitoring Committed Effective Dose Equivalent (CEDE)
Overall Site Summary

Group	2006	2005	2004	2003	2002
0.00	533	582	558	548	332
0.00-<0.10	665	657	718	701	610
0.10-<0.25	3	7	23	131	113
0.25-<0.50	0	0	1	32	69
0.50-<1.00	0	0	0	9	25
1.00-<5.00	0	0	0	3	14
>5.00	0	0	0	0	0
Total individuals	1,201	1,246	1,300	1,424	1,163
Total dose (rem)	5.588	11.450	13.227	56.840	97.11
Maximum dose (rem)	0.158	0.195	0.301	1.126	2.297
Average dose (rem)	0.005	0.009	0.010	0.040	0.084

NOTE: Rows 1-7 provide the number of individuals within the specified CEDE group/range.

Table 12A-4
Internal Radiation Monitoring Committed Effective Dose Equivalent (CEDE)
Individual Group Summary

Group	2006	2005	2004	2003	2002
Employees (#)	**708**	**702**	**715**	**767**	**750**
Total dose (rem)	4.604	10.173	9.292	39.898	58.216
Maximum dose (rem)	0.158	0.195	0.221	1.126	1.183
Average dose (rem)	0.007	0.014	0.013	0.052	0.078
Murray Guards (#)	**166**	**174**	**189**	**164**	**109**
Total dose (rem)	0.364	0.773	0.543	2.648	7.847
Maximum dose (rem)	0.011	0.028	0.012	0.072	0.353
Average dose (rem)	0.002	0.004	0.003	0.016	0.072
Contractors (#)	**248**	**248**	**272**	**322**	**182**
Total dose (rem)	0.620	0.495	3.389	14.234	30.831
Maximum dose (rem)	0.071	0.059	0.301	1.055	2.297
Average dose (rem)	0.003	0.002	0.012	0.044	0.169
Visitors (#)	**79**	**122**	**124**	**167**	**117**
Total dose (rem)	0.000	0.009	0.003	0.060	0.216
Maximum dose (rem)	0.000	0.006	0.002	0.009	0.087
Average dose (rem)	0.000	0.000	0.000	0.000	0.002

Table 12A-5
Total Dose Monitoring Total Effective Dose Equivalent (TEDE)
Overall Site Summary

Group	2006	2005	2004	2003	2002
0.00	477	540	411	285	190
0.00-<0.10	638	509	739	901	679
0.10-<0.25	77	143	135	160	160
0.25-<0.50	8	44	15	63	92
0.50-<1.00	1	10	0	12	28
1.00-<5.00	0	0	0	3	14
>5.00	0	0	0	0	0
Total individuals	1,201	1,246	1,300	1,424	1,163
Total dose (rem)	35.120	56.643	42.104	81.298	116.317
Maximum dose (rem)	0.586	0.953	0.429	1.213	2.331
Average dose (rem)	0.029	0.045	0.032	0.057	0.100

NOTE: Rows 1-7 provide the number of individuals within the specified TEDE group/range.

Table 12A-6
Total Dose Monitoring Total Effective Dose Equivalent (TEDE)
Individual Group Summary

Group	2006	2005	2004	2003	2002
Employees (#)	**708**	**702**	**715**	**767**	**750**
Total dose (rem)	30.454	51.442	34.649	61.270	75.057
Maximum dose (rem)	0.586	0.953	0.351	1.213	1.275
Average dose (rem)	0.043	0.073	0.048	0.080	0.100
Murray Guards (#)	**166**	**174**	**189**	**164**	**109**
Total dose (rem)	2.026	2.367	2.146	4.137	8.759
Maximum dose (rem)	0.057	0.072	0.117	0.112	0.359
Average dose (rem)	0.012	0.014	0.011	0.025	0.080
Contractors (#)	**248**	**248**	**272**	**322**	**182**
Total dose (rem)	2.640	2.791	5.306	15.818	32.288
Maximum dose (rem)	0.183	0.262	0.429	1.060	2.331
Average dose (rem)	0.011	0.011	0.020	0.049	0.177
Visitors (#)	**79**	**122**	**124**	**167**	**117**
Total dose (rem)	0.000	0.043	0.003	0.073	0.216
Maximum dose (rem)	0.000	0.034	0.002	0.012	0.087
Average dose (rem)	0.000	0.000	0.000	0.000	0.002

RADIATION PROTECTION

E. Exposure Control

12.14 Surface Contamination Control Program

12.14.1 General

The NFS surface contamination control program requires that administrative action guidelines be established to assure that contamination levels and employee exposures are kept as low as reasonably achievable (ALARA) and within the limits established by internal action guides.

To comply with these limits, NFS has a protective clothing program and a program for monitoring area contamination levels and personnel contamination.

12.14.2 Area Classification

Classification of the areas and the limits applied to areas within the plant protected area is based on the use to which the specific area is committed and the potential hazard presented by the presence of surface contamination, particularly with regard to inhalation and resuspension propensity. The area designations are "uncontrolled" and "controlled," and are defined in Chapter 1. Controlled areas may be further subdivided into contamination areas, special controlled areas, RWP areas, buffer zones, step off pads, etc., where appropriate. Typical areas within buildings where "controlled areas" are frequently established are presented in Figure 12.2.

12.14.3 Action Guide Levels

Action guides are established to ensure appropriate corrective actions are taken for contamination control. The guideline levels are designed to be conservative in nature and are not to be regarded as the borderline between "safe" and "unsafe."

If contamination in excess of the action guide levels occurs, the necessary remedial action (decontamination, stabilization, excavation, disposal, etc.) is based upon the particular circumstances and the behavior of the material involved.

Action levels are given in Table 12.6. Response is based on the need to avoid transfer of contamination to uncontrolled areas and to maintain exposures as low as is reasonably achievable. Timeliness of the response is based on the above considerations and is set by internal guidelines.

Figure 12.2
Radiologically Controlled Areas

All areas are required to be surveyed for removable alpha and/or beta contamination (as appropriate for the radioactive material processed/stored) on routine frequencies. Areas in which the potential for surface contamination is high, or the probability for human intake is high, are surveyed more frequently.

Table 12.6
Contamination Survey Action Points

Location	Transferable Alpha Contamination (dpm/100 sq cm)	Transferable Beta Contamination (dpm/100 sq cm)
Uncontrolled Area	200	1,000
Uranium Controlled Area	5,000	50,000
Plutonium Controlled Area	1,000	N/A

Surface contamination on offsite shipments of radioactive materials complies with Department of Transportation (or other regulatory agency) requirements.

12.14.4 Survey Practices

Removable radioactive contamination is determined by taking a smear from a known surface area (normally 100 cm^2) by applying moderate pressure and assessing the amount of radioactive material on the smear with an appropriate instrument of known efficiency. Wet smears may be taken as necessary and dried appropriately for analysis. In determining removable contamination on objects of lesser surface area, the pertinent levels are reduced proportionally; and the entire surface is wiped. Large area wipes may also be used for onsite release or gross indicators of contamination on an object or in an area.

Measurements of total alpha/beta contamination may be made as a part of the contamination control program. Actions are taken based on the results of the transferable contamination levels.

The interior surfaces of containment systems such as ventilated hoods, gloveboxes, cells, etc., are excluded from the limits for removable contamination in controlled areas and, therefore, are not routinely surveyed. Special diked areas, drip pans, and the like, although open to room air, are limited to traffic access and, therefore, create less potential for transfer or resuspension.

These areas are surveyed routinely for removable contamination with acceptable levels, decontamination actions, and survey frequencies set by internal guidelines.

Only alpha contamination surveys are performed routinely. Beta contamination surveys are performed only under special circumstances when the conditions warrant such surveys. Contamination surveys are performed on the basis of process operations and the contamination trends. Decontamination is performed in accordance with the action points designated in Table 12.6. Measurements are recorded in units of dpm per area of surface surveyed or dpm per wipe for large area wipes.

12.14.5 Control Practices

The contamination buildup within controlled areas is primarily controlled by physical containment of materials in station enclosures. Frequent mopping of floors and wiping down of equipment, ducts, pipes, etc., are used as an additional control measure.

During or at the conclusion of each contamination survey, supervision or management is advised by the surveyor of all areas which exceed the action guides. The responsible party then initiates action to assure timely decontamination. Such action is documented on the survey form.

Each day (Monday through Friday, except holidays) a qualified member of the radiation safety and protection function reviews the contamination surveys for trends, problem areas, timely decontamination, etc. He/she identifies to the area manager those locations considered to be a problem.

A monthly summary of surface contamination results is prepared, and distributed to plant management.

12.14.6 Personnel Contamination Control Guidance

To prevent the spread of contamination from the controlled areas and to minimize exposure to employees, the following requirements are enforced:

- All personnel wear anti-contamination clothing as directed by internal procedures while in controlled areas. This may include coveralls, gloves, hoods, shoe covers, or booties, as appropriate.

- All personnel remove protective clothing at the designated boundary and deposit them in the dirty laundry or disposal receptacles.

- All personnel survey for contamination at designated locations after exiting the controlled areas. If the levels in Table 12.7 are exceeded, decontamination is

Table 12.7
Personnel Survey Actions/Limits

Range/Limit* **(dpm/100 cm^2)**	**Skin**	**Personal Clothing**	**Personal Shoes**	**Protective Clothing**
0-MDA	No action	No action	No action	No action
≩ MDA – 2500	Decontaminate and resurvey. Notify foreman if decontamination is not successful.	Decontaminate and resurvey. Notify Safety Department if decontamination is not successful and change into clean clothing.	Decontaminate and resurvey. Notify Safety Department if decontamination is not successful and change into clean shoes.	Deposit in dirty laundry container.
> 2500	Notify area foreman. Decontaminate and resurvey. Notify Safety Department if decontamination is not successful.	Decontaminate and resurvey. Notify Safety Department if decontamination is not successful and change into clean clothing.	Notify Safety Department. Decontaminate and resurvey. Notify Safety Department if decontamination is not successful and change into clean shoes.	Notify Safety Department. Deposit in dirty laundry container.
* Corrected for background. This measurement is for total alpha or beta contamination as appropriate. A correction will be made for active surface area of the probe used. MDA is defined in Part 1, Chapter 3, Table 3.1.				

performed. If protective clothing is suspected of being contaminated, the affected areas are also monitored. Actions specified in Table 12.7 are taken.

- Hands and feet are surveyed at a minimum. Additional body or clothing locations are surveyed based on initiating actions (e.g., area contacted liquid or contaminated equipment). Guidance for determining initiating actions and necessary survey(s) are specified in Health and Safety Procedures.

- Periodic overcheck surveys are performed at various locations and documented to assure that, upon leaving the plant protected area, contamination of personnel does not exceed instrument detection levels.

12.14.7 Contamination Control for Release of Material or Equipment and for Shipping

Surface contamination surveys are conducted for contamination prior to release of potentially contaminated packages, equipment, scrap, or waste from controlled to uncontrolled areas or for unrestricted release. No equipment or package brought from a controlled area is removed from the NFS Erwin facility unless radioactivity contamination levels are at or below the guidelines given in Chapter 1, Appendix A.

Shipments of radioactive materials meet Department of Transportation regulations regarding radiation and contamination levels.

If contamination is detected or is known to have been covered, a reasonable effort is made to eliminate the contamination; (i.e., decontamination procedures are repeated until additional effort does not significantly reduce the contamination levels). If the value of the item does not justify this level of effort, it is disposed of as radioactive waste or limited to use within the controlled area. If the value of the item or the need to remove the item from the controlled area is very great, then a conditional release is granted under very strict control conditions designed to prevent the spread of contamination or the exposure of personnel. These conditions are set by internal guidelines.

12.15 Uranium Chemical Toxicity

When individuals may have been exposed to soluble compounds (Class D/F) of uranium with specific activity less than 2.4 μCi/gU, the chemical toxicity limit of 10 milligrams inhaled in a week may be more restrictive than the radiological limit. If this type of exposure is possible, the action levels in Table 12.3 are modified as follows:

Internal Exposure	Action
Airborne – Any result which shows a potential exposure > 0.2 mg U/m^3 averaged over a calendar week	Initiate confirmatory bioassay; determine individuals potentially exposed and evaluate work history for total intake; and investigate as to cause and recommend corrective actions. Establish work restriction pending intake assessment; perform detailed exposure evaluation utilizing urinalysis.
URINALYSIS – Any result which shows a potential exposure > 10 mg U in a calendar week	Initiate confirmatory bioassay; determine individuals potentially exposed and evaluate work history for total intake; and investigate as to cause and recommend corrective actions. Establish work restriction pending intake assessment; perform detailed exposure evaluation.

NOTE:

0.2 mg U/m^3 (Class D/F)	=	14% DAC (0.36 μCi/gU)
	=	27% DAC (0.677 μCi/gU)
	=	40% DAC (1.0 μCi/gU)
	=	80% DAC (2.0 μCi/gU)
	=	95% DAC (2.4 μCi/gU)

10 mg U (Class D/F)	=	6 DAC-hr (0.36 μCi/gU)
	=	11 DAC-hr (0.677 μCi/gU)
	=	17 DAC-hr (1.0 μCi/gU)
	=	33 DAC-hr (2.0 μCi/gU)
	=	40 DAC-hr (2.4 μCi/gU)

ENVIRONMENTAL PROTECTION

A. ENVIRONMENTAL PROTECTION

13.1 Summary of Environmental Data and Impacts

Complete radiological and non-radiological environmental summaries for the NFS Erwin facility are included in the NFS Environmental Report, dated December 1996 (submitted to the NRC May 2, 1997). A summary of the NRC's Environmental Assessment and its Finding of No Significant Impact (FONSI) to the 1999 renewal of SNM-124 is included in a 1999 notice in the Federal Register (Vol. 64, No. 23, pp. 5681-5683).

A Supplemental Environmental Report provides a summary of radiological and non-radiological impacts attributable to the Blended Low-Enriched Uranium (BLEU) Project. This Supplemental Environmental Report specified the environmental impacts associated with constructing and operating the Uranyl Nitrate Building, BLEU Prep. Facility (BPF), Oxide Conversion Building, and Effluent Processing Building. The NRC noticed an EA/FONSI in the Federal Register (Vol. 67, No. 131, P. 45555-45559) concerning the license amendment needed to construct and operate the Uranyl Nitrate Building on July 9, 2002. Additionally, this EA/FONSI served to address the impacts associated with increasing the possession limit from 7,000 kilograms to 14,000 kilograms of ^{235}U and those attributable to the entire BLEU Project. The NRC noticed an EA/FONSI in the Federal Register (Vol. 68, No. 207, pp. 61235-61238) concerning the license amendment needed to operate the BPF on October 27, 2003. The NRC noticed an EA/FONSI in the Federal Register (Vol. 69, No. 117, pp. 34198-34202) concerning the license amendment needed to operate the Oxide Conversion Building, and Effluent Processing Building on June 18, 2004.

13.2 Off-site Doses

NFS routinely calculates dose to members of the general public due to air effluents from operating stacks. These calculations utilize data from the air effluent monitoring program, along with site-specific meteorological data and physical characteristics of stacks. A summary of the results of these calculations for the maximally exposed individual for the period beginning the first quarter of 1995 through the fourth quarter of 2006 are given in Table 13.1.

13.3 Effluent Control and Environmental Monitoring

Effluent monitoring and environmental monitoring are implemented through compliance with a number of Safety Department procedures. These procedures outline: sampling technique, sample processing and analysis methodologies, quality assurance, and other necessary information for maintaining a viable program. Current sampling locations are identified in Table 13.2 and Table 13.3. These tables may also include sampling locations that are not required by license SNM-124 and applicable regulations. The minimum environmental protection sampling locations are identified in Chapter 5 of Part I. Action levels associated with effluent monitoring and environmental monitoring are provided in Table 5.1 of Chapter 5, Part I, of this license.

13.3.1 Air Sampling

Airborne effluents from process ventilation stacks and vents are sampled continuously for radioactivity, during the processing of radioactive materials. Samples, representative of the total discharge, are routinely collected at frequencies specified in Safety Department procedures. All airborne effluent samples are analyzed for gross alpha and gross beta radioactivity.

13.3.2 Liquid Sampling

The Wastewater Treatment Facility (WWTF) treats and discharges process wastewater on a batch basis. Prior to discharge, each batch is sampled and analyzed for gross alpha and gross beta radioactivity as well as selected chemical parameters. The WWTF batches are discharged when they meet regulatory compliance parameters. The batch samples collected from the WWTF are composited and analyzed for specific radionuclides.

The NFS and BLEU Complex Sanitary Sewers continuously discharge to the City of Erwin – Publicly Owned Treatment Works (POTW), and each sanitary sewer's discharge point is sampled continuously using a proportional sampler. All sanitary sewer samples are collected and analyzed in accordance with the specifications of Table 5.1 in Chapter 5.

Radioactive material may be discharged under 10 CFR 20.2003 regulations to the NFS Sanitary Sewer from the Groundwater Treatment Facility (GWTF) and to the BLEU Complex Sanitary Sewer from the Effluent Processing Building (EPB). When operating, grab samples of the GWTF and EPB liquid effluents are collected. The grab sample s are composited monthly, for each discharge stream, and analyzed for insoluble radioactivity. The grab samples from the EPB's batch discharges are also analyzed for gross alpha and beta radioactivity prior to discharging each batch into the BLEU Complex Sanitary Sewer. When gross alpha or gross beta radioactivity results from a grab sample of the

EPB's effluent batch exceed an action level, the batch will not be discharged until the elevated concentrations are lowered to acceptable levels.

Storm water is continually discharged to Martin Creek. Storm water is sampled annually for chemical and radiological attributes.

NFS routinely collects grab samples from locations that are upstream and downstream of the NFS facility along Martin Creek and the Nolichucky River. Individual grab samples are analyzed for specific radionuclides, and the grab samples from the downstream locations are composited prior to analyzing them for specific radionuclides.

Site groundwater monitoring wells requiring sampling are shown in Figure 13.1. Other site wells include those located inside of the protected area and those located outside of the protected area to the north and the west. A number of wells are sampled both monthly and quarterly and analyzed for both chemical and radiological parameters. The routine radiological parameters are gross alpha and gross beta activity. If the action points of 15 pCi/liter alpha or 50 pCi/liter beta activity are exceeded, isotopic analysis will be performed.

13.3.3 Soil, Sediment, and Vegetation

Soil, sediment, and vegetation grab samples are routinely collected from the locations listed in Table 13.2. Soil sampling is typically restricted to the surface layer, which is indicative of the recent deposition of airborne radioactivity at a given location. Sediment sampling emphasizes shallow sediments, which is indicative of recent deposition from liquid effluents. Vegetation sampling relies primarily on annual plant growth, which is indicative of the radioactivity taken up through the roots and deposited on plant surfaces. In addition, sludge samples are routinely collected from the City of Erwin POTW and analyzed for uranium isotopes.

13.3.4 Environmental Dosimeters

Environmental dosimeters are at onsite and offsite locations to monitor ambient external doses and to assist with the assessment of potential accidents. Environmental dosimeter data are used to monitor external dose rates in unrestricted areas, determine doses to members of the public, and demonstrate compliance with regulatory dose limits. Doses to members of the public will be calculated per 10 CFR 20.1302(b)(1), and may include considerations for the amount of time a member of the public is actually present at or the amount of time a member of the public may be present at a given location.

Table 13.1
Results of Quarterly Dose Assessments for the Maximally Exposed Off-site Individual

Period	Maximum TEDE (mrem)	Maximun CDE (mrem)	Maximally Exposed Organ	Location of Maximum Exposure	
				Sector	Distance (m)
1st Qtr. 1995		0.05	Child-Lung	ESE	300
2nd Qtr. 1995		0.02	Child-Lung	ESE	300
3rd Qtr. 1995		0.02	Child-Lung	ESE	300
4th Qtr. 1995		0.02	Child-Lung	ESE	300
1st Qtr. 1996		0.05	Child-Lung	SE	215
2nd Qtr. 1996		0.03	Child-Lung	ESE	300
3rd Qtr. 1996		0.04	Child-Lung	SE	215
4th Qtr. 1996		0.05	Child-Lung	SE	215
1st Qtr. 1997		0.07	Child-Lung	SE	215
2nd Qtr. 1997		0.03	Child-Lung	SE	215
3rd Qtr. 1997		0.05	Child-Lung	NNE	210
4th Qtr. 1997		0.04	Child-Lung	NNE	210
1st Qtr. 1998	0.004	0.03	Lung	NNE	250
2nd Qtr. 1998	0.073	0.43	Lung	NNE	650
3rd Qtr. 1998	0.013	0.10	Bone Surfaces	NNE	200
4th Qtr. 1998	0.011	0.07	Lung	NNE	550
1st Qtr. 1999	0.009	0.06	Bone Surfaces	NNE	500
2nd Qtr. 1999	0.012	0.13	Bone Surfaces	NNE	300
3rd Qtr. 1999	0.013	0.14	Bone Surfaces	NNE	300
4th Qtr. 1999	0.020	0.07	Bone Surfaces	NE	100
1st Qtr. 2000	0.014	0.09	Bone Surfaces	SSE	900
2nd Qtr. 2000	0.013	0.13	Bone Surfaces	NNE	300
3rd Qtr. 2000	0.006	0.06	Bone Surfaces	NNE	300
4th Qtr. 2000	0.005	0.04	Bone Surfaces	NNE	300
1st Qtr. 2001	0.005	0.05	Bone Surfaces	NNE	300

Period	Maximum TEDE (mrem)	Maximun CDE (mrem)	Maximally Exposed Organ	Location of Maximum Exposure	
				Sector	Distance (m)
2nd Qtr. 2001	0.010	0.06	Lung	NNE	250
3rd Qtr. 2001	0.007	0.05	Lung	NNE	300
4th Qtr. 2001	0.009	0.06	Lung	NNE	250
1st Qtr. 2002	0.012	0.11	Bone Surfaces	NNE	250
2nd Qtr. 2002	0.007	0.05	Lung	NNE	250
3rd Qtr. 2002	0.009	0.07	Bone Surfaces	NNE	250
4th Qtr. 2002	0.005	0.04	Bone Surfaces	NNE	250
1st Qtr. 2003	0.005	0.04	Bone Surfaces	NNE	300
2nd Qtr. 2003	0.009	0.06	Lung	NNE	350
3rd Qtr. 2003	0.003	0.02	Lung	NNE	500
4th Qtr. 2003	0.003	0.02	Lung	NNE	500
1st Qtr. 2004	0.003	0.09	Lung	NNE	450
2nd Qtr. 2004	0.002	0.07	Lung	NNE	500
3rd Qtr. 2004	0.006	0.05	Lung	NNE	650
4th Qtr. 2004	0.006	0.03	Spleen	NNE	300
1st Qtr. 2005	0.004	0.02	Spleen	NNE	350
2nd Qtr. 2005	0.001	0.001	Spleen	NNE	400
3rd Qtr. 2005	0.001	0.002	Spleen	NNE	450
4th Qtr. 2005	0.001	0.002	Spleen	NNE	450
1st Qtr. 2006	0.001	0.002	Spleen	NNE	400
2nd Qtr. 2006	0.002	0.003	Spleen	NNE	400
3rd Qtr. 2006	0.001	0.001	Spleen	NNE	500
4th Qtr. 2006	0.001	0.001	Spleen	NNE	450

NOTES: TEDE is the total effective dose equivalent from air emissions for period (NFS began reporting TEDE the 1st Qtr 1998).
CDE is the committed dose equivalent to the maximally exposed organ from air emissions during period.
All dose predictions represent 50-year dose commitments for internal exposure pathways.
As of the 1st quarter of 1998 assessment techniques changed and doses are now computed using the CAP88-PC computer code.

Table 13.2
Routine Environmental Monitoring Sampling Locations

Surface Water	Ground Water	Ambient Air	Vegetation	Soil/Sediment
Martin Creek Upstream Sampling Point		NW Perimeter Air Sampling Sta #170	Asheville Highway Sampling Point	Asheville Highway Sampling Point (soil)
Martin Creek Downstream Sampling Point		W Perimeter Air Sampling Sta #171	Banner Hill Rd Sampling Point	Banner Hill Rd Sampling Point (soil)
		S Perimeter Air Sampling Sta #172	Little Mountain Sampling Point	Little Mountain Sampling Point (soil)
			Burial Ground Sampling Point	First St Sampling Point (soil)
	Well 52 Background Well		First St Sampling Point	Burial Ground Sampling Point (soil)
Nolichucky River Upstream Sampling Point	Wells 98A, 99A, 100A, 100B, 101A, 102A, 103A, 104A, 105A, 106A, Quarterly Downgradient Wells	E Perimeter Air #1 Sampling Sta #174		Nolichucky River Upstream Sampling Point (sediment)
Nolichucky River Downstream Sampling Point		E Perimeter Air #2 Sampling Sta #218		Nolichucky River Downstream Sampling Point (sediment)
				Martin Creek Upstream Sampling Point (sediment)
	Other(1)	Near Ponds 1 & 2 Air Sampling Sta #555		Martin Creek Downstream Sampling Point #1 (sediment)

Surface Water	Ground Water	Ambient Air	Vegetation	Soil/Sediment
		S. BLEU Facility Air Sampling Sta #556		
		W. BLEU Facility Air Sampling Sta #557		Martin Creek Downstream Sampling Point #2 (sediment)
		W Perimeter Air Sampling Sta #668		
		Banner Hill Rd Air Sampling Sta #323		
		Prk. Lot/Ent Air Sampling Sta #372		
		B. Hill/Stalling Air Sampling Sta #381		
		Stalling Ln SE Air Sampling Sta #382		
		Highland/1st St Air Sampling Sta #383		
		Spar Mill Rd Air Sampling Sta #384		
		Security Fence SE Air Sampling Sta #385		
		Asheville Hwy Air Sampling Sta #324		
		NW of North NFS Burial Ground Sampling Sta #553		

Surface Water	Ground Water	Ambient Air	Vegetation	Soil/Sediment
		Images West Sampling Sta #581		
		NFS Training Center Sampling Sta #582		
		Little Mountain Air Sampling Sta #322		
		E of North NFS Burial Ground Sampling Station #677		
		NE of North NFS Burial Ground Sampling Station #678		
		N of North NFS Burial Ground Sampling Station #679		

NOTES:

(1) In accordance with established procedures, a number of well samples are collected on a routine basis. At a minimum, these are analyzed for gross alpha and beta radioactivity (see Figure 13.1).

Table 13.3
Effluent Sampling Locations[1]

Gaseous Effluents (Stacks/Vents)	Liquid Effluents (Streams)
Stack 416, Main Stack	Wastewater Treatment Facility (WWTF)
Stack 185, Building 131	
Stack 234, Building 234	NFS Sanitary Sewer
Stack 327, Building 330	BLEU Complex Sanitary Sewer
Stack 376, Building 301	EPOTW Sludge
Stack 421, Building 100	
Stack 424, Building 100	
Stack 503, Building 530	
Stack 573, Building 306-W	
Stack 600, Building 110	
Stack 615, Building 306-W	
Stack 646, Building 110	
Stack 649, Building 335	
Stack 701, Building 307	
Stack 702, Building 307	
Stack 703, Building 333	
Stack 704, Building 333	
Stack 501, Building 510	
Stack 502, Building 520	
Stack 773, Building 440	
Stack 774, Building 301	

Note:
1 - Normally, samples are only collected from these locations when discharges occur (i.e., stack/vent operates or facility discharges liquid effluents).

Figure 13.1
Site Groundwater Monitoring Well Locations

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APPENDIX A
CHAPTER 13
ENVIRONMENTAL MONITORING

13A.1 Gaseous Effluents

Stacks 416, 600 and 421 accounted for 90% of the radioactivity released from NFS stacks during this reporting period. Average radionuclide concentrations and total activities released in effluent air during this reporting period are provided in **Table 13A-1**, for each of the seventeen (17) stacks that operated.

The last column of **Table 13A-1** provides the ratios of radionuclide concentrations to the effluent concentration values (ECV). The ECVs in **Table 13A-1** are equivalent to the concentrations listed in 10 CFR Part 20, Appendix B, Table 2, Column 1. The fraction of the ECV is useful for determining the significance of a specific stack or radionuclide when compared to the rest of NFS' gaseous effluents. The sum of fractions (SOF) also provides a useful means for examining trends associated with stack releases. Note that the SOF should not be used to estimate offsite doses associated with gaseous effluents.

13A.2 Liquid Effluents

Radionuclides discharged from NFS' liquid effluent streams (i.e. BLEU Complex Sanitary Sewer, NFS Sanitary Sewer, and Waste Water Treatment Facility) during this reporting period are provided in **Table 13A-2**. The last column in **Table 13A-2** provides the ratio of the effluent radioactivity concentration to the ECV as a fraction. The ECVs for the Waste Water Treatment Facility (WWTF) are equivalent to the concentrations presented in 10 CFR Part 20, Appendix B, Table 2, Column 2, and the ECVs for the BLEU Complex and NFS Sanitary Sewers are equivalent to the Table 3 values. The goal is to maintain annual average liquid effluent concentrations below the concentrations specified in 10 CFR Part 20, Appendix B, Table 2 Column 2 and Table 3. Maintaining the SOF below a value of one demonstrates that the constraints of 10 CFR Part 20, Subpart D were not exceeded. In the event that a liquid effluent stream exceeds a SOF value of one, a dose assessment may be used to demonstrate compliance with the dose limit specified in 10 CFR Part 20, Subpart D.

13A.2.1 Sanitary Sewer

Proportional sewer samples are collected daily (normal operating days). The NFS Environmental Laboratory performs gross alpha and beta analyses on the daily samples. These results are compared to predetermined action levels to determine if action may be needed to reduce radioactive effluents by this pathway. Aliquots from daily samples are composited to generate a representative sample for each month. Radionuclide specific analyses are performed on the monthly composite sample to determine the average concentration released. Monthly composite sample results are used, along with the discharge volume, to determine the volume-weighted average concentrations reported in **Table 13A-2**.

13A.2.1.1 BLEU Complex Sanitary Sewer

Discharges to the BLEU Complex Sanitary Sewer began in August 2004. The total activity released to the BLEU Sewer was estimated at 1.9E-04 Ci for the year 2006. This value is well below the annual limit of 1 Ci, specified by NRC regulations and State of Tennessee regulations.

When operating, representative samples are obtained from each Effluent Processing Building (EPB) batch, prior to discharge to the sanitary sewer. The samples are composited monthly and analyzed for insoluble radioactivity. The insoluble radioactivity data from the EPB is used to demonstrate compliance with 10 CFR 20.2003 requirements applicable to the BLEU Complex Sanitary Sewer discharges. Compliance with insoluble radioactivity requirements is demonstrated monthly and documented quarterly.

13A.2.1.2 NFS Sanitary Sewer

The total activity released to the sewer was estimated as 2.5E-03 Ci for the year 2006. This value is well below the annual limit of 1 Ci, specified by NRC regulations and State of Tennessee regulations.

The Groundwater Treatment Facility treats groundwater and discharges it to the NFS Sanitary Sewer. When operating, daily grab samples are obtained from the Groundwater Treatment Facility. The grab samples are composited monthly and analyzed for insoluble radioactivity. The Groundwater Treatment Facility did not operate during 2006.

13A.2.2 Waste Water Treatment Facility

The WWTF receives and treats NFS process water prior to discharging the wastewater. Samples are collected from each batch processed in the WWTF and screened onsite for gross alpha and beta activity prior to permitted discharge to the Nolichucky River via Outfall 001. Sample results are compared to predetermined action levels to determine if the batch can be released without special approval. Aliquots from daily samples are composited to generate a representative sample for each month. Radionuclide specific analyses are performed on each monthly composite sample to determine the average concentrations released. Monthly composite sample results are used, along with the discharge volume, to determine the volume-weighted average concentrations reported in **Table 13A-2**.

In addition to radiological characteristics, the NPDES permit for Outfall 001 issued by the State of Tennessee requires that NFS monitor several chemical parameters in WWTF discharges. Results of these chemical analyses are summarized in **Table 13A-3**.

13A.3 Storm Water Runoff

13A.3.1 NFS Storm Water Runoff

Quarterly Storm Water Quality Inspections were conducted.

Annual storm water sampling was conducted on March 20, 2006 for Outfall A and Outfall B. All attributes sampled were within NPDES permit limits with the following exceptions on both outfalls: Nitrate/Nitrite as Nitrogen, Total Recoverable Magnesium, and Total Recoverable Aluminum. The nitrate/nitrite as nitrogen and magnesium attributes have exceeded the permit limits since 1998. These exceedances have been identified to the State of Tennessee Division of Water Pollution Control. Research has demonstrated that the nitrate/nitrite as nitrogen and the magnesium are due to naturally occurring background levels in surface water and groundwater in the vicinity of the NFS site. This was identified to the State of Tennessee in 2003 and a request was made by NFS to modify the permit limits. The State of Tennessee agreed with the research, but has not modified the permit limits. The origin of the aluminum attribute still remains unidentified as it has since 2003.

13A.3.2 BLEU Complex Storm Water Runoff

Quarterly Storm Water Quality Inspections were conducted. No problems were identified.

Annual storm water sampling was conducted on April 17, 2006. All attributes were within NPDES Permit limits with the exception of Total Recoverable Magnesium, Total Recoverable Aluminum, and Nitrate/Nitrite as Nitrogen. This was identified to the State of Tennessee with the same explanation as in 13A.3.1.

13A.4 Surface Water

The routine sampling program for radioactivity in area surface water involves collection and analysis of water samples from Martin Creek and the Nolichucky River. These surface water systems are sampled and analyzed for gross alpha and gross beta levels to establish radioactivity concentrations at upstream and downstream locations from the site. A monthly composite is made from Martin Creek downstream weekly grab samples. A quarterly grab sample is collected from Martin Creek upstream. Nolichucky River upstream and downstream samples are collected monthly. The Martin Creek and Nolichucky River samples are also analyzed for isotopic uranium (i.e. $^{233/234}U$, $^{235/236}U$, and ^{238}U), isotopic thorium (i.e. ^{228}Th, ^{230}Th, and ^{232}Th), and isotopic plutonium (i.e. ^{238}Pu and $^{239/240}Pu$). Downstream surface water monitoring data for the year 2006 are summarized in **Table 13A-4** for Martin Creek and **Table 13A-5** for the Nolichucky River.

13A.5 Groundwater

NFS routinely monitors groundwater to determine if plant activity has impacted water quality downgradient of the plant site and to observe the progress of remediation. This program compares groundwater quality on site to background water quality. Further, the program monitors groundwater offsite to evaluate potential releases. Also, the Groundwater Monitoring Program examines groundwater quality in a former Tc-99 release area. Lastly, the program tracks the status of ongoing remediation activities in effected areas both on and offsite.

During 2006, NFS routinely sampled monitoring wells on the NFS plant site in conjunction with groundwater remediation near the maintenance shop area. Samples were also collected from routine wells monthly and quarterly. Additionally, samples were collected semi-annually from wells located in the adjacent industrial park.

Discussion of Groundwater Areas

Routine Groundwater

Background - Well 52 is monitored to determine background groundwater quality. Samples are collected monthly and analyzed for gross alpha and beta activity. Data on the background well appears in **Table 13A-6**.

Building 234 – Wells 234-2 and 234-3 are monitored to determine the water quality downgradient of the former Plutonium processing area. The Groundwater Monitoring Program has continued to collect samples after the building's demolition to determine the effect of source removal on the groundwater quality. Wells are sampled quarterly for gross alpha and beta activity, and if triggered, for isotopic uranium, plutonium, Tc-99, and thorium. **Table 13A-7** contains the results.

Perimeter Wells – Samples collected from wells along the property boundary identify potential releases to off site properties. Wells 98A, 99A, 100A, 100B, 101A, 102A, 103A, 104A, 105A, and 106A are located along the northeast side of the plant's perimeter. Wells are sampled monthly or quarterly for gross alpha and beta activity, total uranium, and Tc-99. **Table 13A-8** presents the data for the perimeter wells.

Off-Site Wells - Wells 116A, 116B, 117A, 117B, 118A, 118B, 119A, 120A, and 120B monitor groundwater quality downgradient of the NFS facility in the adjacent industrial properties owned by NFS (NFS Industrial Park Facility), CSX Railroad, and Impact Plastics, Inc. Offsite well data for gross alpha, gross beta, total uranium, and Tc-99 analysis appears in **Table 13A-9.**

Tc-99 Release Area

In order to monitor the impact of a Tc-99 release in 1999, the wells located in close proximity to the Tc-99 release sites (230 dike area and wastewater treatment facility) were monitored. Data from wells 234-2, 234-3, 38R, and the boundary wells downgradient of the Pond 4 building provide insight to the extent of the Tc-99 contamination area.

The NFS internal action level for gross beta activity is 50 pCi/L, which if exceeded, will trigger the analysis of Tc-99. The MCL of 900 pCi/L represents the maximum amount of contaminant in the water that can be delivered to any user of a public water system. Tc-99 data for upgradient and downgradient wells in relation to where the release occurred are discussed below.

Upgradient - Tc-99 sample results in Wells 234-2 and 234-3 do not exceed the MCL during 2006 **(Table 13A-7).** The Tc-99 concentration for Well 38R is below the MCL as shown in **Table 13A-10**.

Downgradient - All of the Tc-99 concentrations in wells downgradient of the release area were well below the MCL of 900 pCi/L (**Table 13A-8 and 13A-9**).

Groundwater Remediation

Maintenance Shop – In 2006, the in-situ reactive zone (IRZ) technology continued reducing the concentration of the total dissolved uranium in groundwater. The total and dissolved uranium plume has been reduced to approximately 0.15 acres from approximately 0.75 acres at baseline. Results for wells in which uranium has been detected are shown in **Table 13A-11**.

13A.6 Environmental Air

The routine environmental air monitoring program consists of sampling both onsite and offsite ambient air. Individual samples are analyzed for gross alpha and beta radioactivity. Composite samples are analyzed for isotopic uranium quarterly for off-site samplers and semi-annually for perimeter and onsite samplers. Additionally, offsite composite samples are required to be analyzed annually for isotopic thorium and isotopic plutonium.

A summary of the results for the gross alpha and beta radioactivity concentrations is provided in **Table 13A-12.**

A summary of the radionuclide specific results from the composite sample analyses is provided in **Table 13A-13**. Results are also presented in terms of a sum of fractions and effective dose equivalent for each sampling station, relative to the concentrations provided in 10 CFR Part 20, Appendix B, Table 2, Column 1. The effective dose equivalent values presented in **Table 13A-13** are very conservative and **are presented for informational purposes only**. The doses are computed by assuming that a person is present at the location of the sampler 24 hours/day and 365 days/year (a very conservative assumption) and that the chemical form of the material is the most conservative inhalation class.

13A.7 Sediment, Soil, and Vegetation

The environmental monitoring program for sediment, soil, and vegetation consists of collecting quarterly samples at the locations listed in **Table 13A-14**. These samples are analyzed for gross alpha and beta radioactivity, isotopic uranium, isotopic thorium, and isotopic plutonium. **Table 13A-14** provides a summary of the average results for samples collected during this period.

13A.8 Ambient Offsite Radiation Levels

Ambient offsite radiation levels attributable to NFS operations are measured by positioning environmental dosimeters along NFS' property boundaries and at offsite locations. Radioactive materials onsite are stored and/or transferred in accordance with facility needs and disposal activities, therefore the overall location/quantity of material may change regularly. Due to the transient nature of the material present in the facility, a quarterly property boundary survey is used as a tool to verify proper dosimeter placement and to demonstrate compliance with the dose rate limit of 2 mrem/hr as per 10 CFR 20.1301. The environmental dosimeter results remain as the confirmation of regulatory compliance.

Environmental dosimeter results have been evaluated to demonstrate compliance with NFS' 80 mrem/yr ALARA constraint. NFS' ALARA constraint is for external dose at its property boundaries using an occupancy factor of one (1). The results from the environmental dosimeters that are used to determine the ambient radiation levels at or beyond NFS' property boundaries are provided in **Table 13A-15**.

A dose assessment is performed to demonstrate compliance with the dose limits in 10 CFR 20.1301, TDEC Rule 1200-2-5-.61, and 40 CFR 190. This assessment is also used to demonstrate compliance with NFS' 10 mrem/yr ALARA goal.

13A.9 Meteorology

Wind velocity and direction are continually measured by sensors located at an onsite tower for use in emergency response. In development of the site 1996 Environmental Report, a tabulation of the average wind speed and wind frequency data was completed using site-specific meteorological data collected during the 5-year period of 1991 through 1995. This 5-year dataset was tabulated for use as reference wind speed and wind frequency data, for estimating offsite doses associated with airborne releases. The dataset indicates that the primary wind direction is from the south-southwest to the north-northeast (18.1%) at an average speed of 7.50 mph.

Precipitation totals for Erwin are recorded at the Erwin Publicly Owned Treatment Works (EPOTW). Individual monthly totals and the cumulative total for the reporting period are provided in **Table 13A-16.** Approximately 48.81 inches of precipitation fell during 2006.

13A.10 Off-site Dose Estimates

As required by Chapter 5 of license SNM-124, estimates of offsite dose to the maximally exposed individual (MEI) from gaseous and liquid effluents are calculated each calendar quarter. **Figures 13A.1 and 13A.2** display the trends of quarterly offsite doses attributable to gaseous and liquid effluents respectively. Doses are quantified in terms of the total effective dose equivalent (TEDE), which are expressed in units of mrem per quarter. The TEDE for the offsite MEI continues to be well below the applicable regulatory limits (i.e. the EPA's 25 mrem/yr limit from 40 CFR 190 and the NRC's limit of 100 mrem/yr from 10 CFR 20.1301), regulatory constraints (i.e. NRC's ALARA constraint for gaseous effluents is 10 mrem/yr from 10 CFR 20.1101), and NFS' ALARA goal (i.e. NFS' ALARA goal for gaseous and liquid effluents is 2 mrem/yr).

The deep-dose equivalent to the MEI is assessed using the environmental dosimeter results each calendar quarter. The offsite doses to the MEI are calculated using an appropriate occupancy factor for the receptor location to demonstrate compliance with the dose limits in 10 CFR 20.1301, TDEC Rule 1200-2-5-.61, and 40 CFR 190. This assessment is also used to demonstrate compliance with NFS' 10 mrem/yr ALARA goal. NFS' ALARA goal is for external dose to offsite members of the public using appropriate adjustments for occupancy. **Figure 13A.3** displays the trend of quarterly offsite doses attributable to the ambient radiation emanating from NFS facilities. The results are well below the applicable regulatory limits (i.e. the EPA's 25 mrem/yr limit from 40 CFR 190 and the NRC's limit of 100 mrem/yr from 10 CFR 20.1301) and NFS' ALARA goal of 10 mrem/yr for doses to offsite members of the public from external radiation (ambient radiation).

An estimate of the offsite dose to the MEI from all effluent pathways is calculated for each calendar quarter. **Figure 13A.4** displays the trend of quarterly offsite doses attributable to all effluent pathways. Note that this calculated dose is very conservative because it is the sum of the MEI doses attributable to gaseous effluents, liquid effluents, and ambient radiation regardless of location. The MEI locations for each of these effluent pathways are normally very different from each other, such that a single individual would never receive the entire estimated dose. The TEDE for the offsite MEI continues to be well below the applicable regulatory limits (i.e. the EPA's 25 mrem/yr limit from 40 CFR 190 and the NRC's limit of 100 mrem/yr from 10 CFR 20.1301).

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Location	Total Volume (m^3)	Activity Concentration (µCi/ml)	Error Estimate (µCi/ml)	LLD (µCi/ml)	Quantity Released (Ci)	Quantity Released (g)	Fraction of ECV[1]
Main Stack 416		**1022.80 m^3/min**	**17.05 m^3/sec**				
Ra-226	278,366,506	3.95E-14	6.33E-15	7.18E-15	1.10E-05	1.11E-05	4.39E-02
Tc-99	530,103,502	1.46E-13	3.40E-14	4.67E-14	7.76E-05	4.59E-03	1.63E-04
Th-228	530,103,502	2.38E-16	4.82E-17	5.29E-17	1.26E-07	1.54E-10	1.19E-02
Th-230	530,103,502	1.49E-16	3.05E-17	3.37E-17	7.91E-08	3.91E-06	7.46E-03
Th-232	530,103,502	9.50E-17	1.94E-17	2.15E-17	5.04E-08	4.62E-01	2.37E-02
U-234	530,103,502	7.29E-14	1.48E-14	1.62E-14	3.86E-05	6.19E-03	1.46E+00
U-235	530,103,502	4.80E-15	9.73E-16	1.07E-15	2.54E-06	1.18E+00	8.00E-02
U-238	530,103,502	7.02E-16	1.44E-16	1.59E-16	3.72E-07	1.11E+00	1.17E-02
						Total:	**1.64E+00**
Stack 185 Bldg. 131		**101.79 m^3/min**	**1.70 m^3/sec**				
Tc-99	53,794,086	6.58E-15	2.93E-14	5.40E-14	3.54E-07	2.09E-05	7.31E-06
Th-230	53,794,086	5.81E-20	5.97E-19	1.26E-18	3.12E-12	1.55E-10	2.90E-06
Th-231	53,794,086	8.40E-17	3.74E-16	6.89E-16	4.52E-09	8.49E-15	9.33E-09
U-234	53,794,086	1.30E-15	1.33E-14	2.81E-14	6.98E-08	1.12E-05	2.60E-02
U-235	53,794,086	2.18E-17	2.24E-16	4.72E-16	1.17E-09	5.42E-04	3.63E-04
U-238	53,794,086	2.51E-20	2.58E-19	5.43E-19	1.35E-12	4.03E-06	4.18E-07
						Total:	**2.63E-02**
Stack 327 Bldg. 330		**667.16 m^3/min**	**11.12 m^3/sec**				
Tc-99	328,459,991	4.75E-14	2.64E-14	4.13E-14	1.56E-05	9.24E-04	5.28E-05
Th-230	328,459,991	2.75E-19	5.17E-19	9.04E-19	9.04E-11	4.48E-09	1.38E-05
Th-231	328,459,991	6.07E-16	3.38E-16	5.27E-16	1.99E-07	3.75E-13	6.74E-08
U-234	328,459,991	6.15E-15	1.15E-14	2.02E-14	2.02E-06	3.24E-04	1.23E-01
U-235	328,459,991	1.03E-16	1.94E-16	3.39E-16	3.39E-08	1.57E-02	1.72E-03
U-238	328,459,991	1.19E-19	2.23E-19	3.90E-19	3.91E-11	1.17E-04	1.98E-06
						Total:	**1.25E-01**
Stack 376 Bldg. 301		**168.75 m^3/min**	**2.81 m^3/sec**				
Tc-99	88,721,444	8.58E-15	2.19E-14	4.07E-14	7.61E-07	4.50E-05	9.53E-06
Th-230	88,721,444	3.63E-20	4.23E-19	9.22E-19	3.22E-12	1.60E-10	1.82E-06
Th-231	88,721,444	1.10E-16	2.80E-16	5.19E-16	9.72E-09	1.83E-14	1.22E-08
U-234	88,721,444	8.12E-16	9.44E-15	2.06E-14	7.20E-08	1.15E-05	1.62E-02
U-235	88,721,444	1.36E-17	1.58E-16	3.46E-16	1.21E-09	5.60E-04	2.27E-04
U-238	88,721,444	1.57E-20	1.82E-19	3.98E-19	1.39E-12	4.16E-06	2.62E-07
						Total:	**1.65E-02**
Stack 421 Bldg. 100		**20.98 m^3/min**	**0.35 m^3/sec**				
Tc-99	11,085,435	5.78E-14	5.89E-14	9.66E-14	6.40E-07	3.79E-05	6.42E-05
Th-230	11,085,435	7.47E-19	1.25E-18	2.23E-18	8.28E-12	4.10E-10	3.73E-05
Th-231	11,085,435	7.37E-16	7.52E-16	1.23E-15	8.17E-09	1.54E-14	8.19E-08
U-234	11,085,435	1.67E-14	2.79E-14	4.97E-14	1.85E-07	2.96E-05	3.34E-01
U-235	11,085,435	2.80E-16	4.68E-16	8.35E-16	3.10E-09	1.44E-03	4.67E-03

[1] ECV: Effluent Concentration Value from 10-CFR-20, Appendix B. Fraction of ECV at the stack is provided for reference only. Concentrations at off-site locations are significantly less than those reported here (at stack) due to the atmospheric dispersion that occurs before the effluent exits the site.

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Location	Total Volume (m^3)	Activity Concentration (µCi/ml)	Error Estimate (µCi/ml)	LLD (µCi/ml)	Quantity Released (Ci)	Quantity Released (g)	Fraction of ECV[1]
Stack 421 Bldg. 100		**20.98 m^3/min**	**0.35 m^3/sec**				
U-238	11,085,435	3.22E-19	5.39E-19	9.61E-19	3.57E-12	1.07E-05	5.37E-06
						Total:	**3.38E-01**
Stack 424 Bldg. 100		**27.77 m^3/min**	**0.46 m^3/sec**				
Tc-99	14,677,008	3.35E-14	3.52E-14	5.80E-14	4.92E-07	2.91E-05	3.72E-05
Th-230	14,677,008	1.55E-19	6.66E-19	1.34E-18	2.27E-12	1.13E-10	7.74E-06
Th-231	14,677,008	4.28E-16	4.50E-16	7.40E-16	6.28E-09	1.18E-14	4.75E-08
U-234	14,677,008	3.46E-15	1.49E-14	3.00E-14	5.08E-08	8.14E-06	6.92E-02
U-235	14,677,008	5.81E-17	2.50E-16	5.04E-16	8.52E-10	3.95E-04	9.68E-04
U-238	14,677,008	6.69E-20	2.88E-19	5.81E-19	9.81E-13	2.93E-06	1.11E-06
						Total:	**7.02E-02**
Stack 501 Bldg. 510		**59.66 m^3/min**	**0.99 m^3/sec**				
Tc-99	31,615,061	1.81E-14	1.68E-14	2.65E-14	5.73E-07	3.39E-05	2.01E-05
Th-228	31,615,061	2.48E-16	9.21E-16	1.86E-15	7.83E-09	9.56E-12	1.24E-02
Th-230	31,615,061	2.78E-16	1.04E-15	2.09E-15	8.80E-09	4.36E-07	1.39E-02
Th-232	31,615,061	2.60E-16	9.67E-16	1.95E-15	8.22E-09	7.54E-02	6.50E-02
U-234	31,615,061	8.46E-16	3.15E-15	6.34E-15	2.67E-08	4.28E-06	1.69E-02
U-235	31,615,061	1.58E-16	5.88E-16	1.18E-15	5.00E-09	2.31E-03	2.63E-03
U-238	31,615,061	2.74E-16	1.02E-15	2.06E-15	8.67E-09	2.59E-02	4.57E-03
						Total:	**1.15E-01**
Stack 502 OCB		**203.53 m^3/min**	**3.39 m^3/sec**				
Tc-99	108,147,745	1.95E-14	1.79E-14	2.83E-14	2.10E-06	1.25E-04	2.16E-05
Th-228	108,147,745	7.95E-17	8.99E-16	1.95E-15	8.60E-09	1.05E-11	3.97E-03
Th-230	108,147,745	8.94E-17	1.01E-15	2.20E-15	9.67E-09	4.79E-07	4.47E-03
Th-232	108,147,745	8.35E-17	9.44E-16	2.05E-15	9.03E-09	8.28E-02	2.09E-02
U-234	108,147,745	2.72E-16	3.07E-15	6.67E-15	2.94E-08	4.71E-06	5.43E-03
U-235	108,147,745	5.07E-17	5.74E-16	1.25E-15	5.49E-09	2.54E-03	8.46E-04
U-238	108,147,745	8.81E-17	9.96E-16	2.16E-15	9.53E-09	2.84E-02	1.47E-03
						Total:	**3.71E-02**
Stack 503 EPB		**6.28 m^3/min**	**0.10 m^3/sec**				
Tc-99	3,390,139	3.17E-14	1.83E-14	2.62E-14	1.07E-07	6.35E-06	3.52E-05
Th-228	3,390,139	2.76E-16	9.22E-16	1.83E-15	9.36E-10	1.14E-12	1.38E-02
Th-230	3,390,139	3.11E-16	1.04E-15	2.06E-15	1.05E-09	5.21E-08	1.55E-02
Th-232	3,390,139	2.90E-16	9.68E-16	1.92E-15	9.83E-10	9.02E-03	7.25E-02
U-234	3,390,139	9.43E-16	3.15E-15	6.25E-15	3.20E-09	5.12E-07	1.89E-02
U-235	3,390,139	1.76E-16	5.88E-16	1.17E-15	5.97E-10	2.77E-04	2.94E-03
U-238	3,390,139	3.06E-16	1.02E-15	2.03E-15	1.04E-09	3.10E-03	5.10E-03
						Total:	**1.29E-01**
Stack 573 Bldg 306-W		**77.95 m^3/min**	**1.30 m^3/sec**				
Tc-99	40,080,662	1.05E-14	2.40E-14	4.46E-14	4.20E-07	2.49E-05	1.17E-05

[1] ECV: Effluent Concentration Value from 10-CFR-20, Appendix B. Fraction of ECV at the stack is provided for reference only. Concentrations at off-site locations are significantly less than those reported here (at stack) due to the atmospheric dispersion that occurs before the effluent exits the site.

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Location	Total Volume (m^3)	Activity Concentration (µCi/ml)	Error Estimate (µCi/ml)	LLD (µCi/ml)	Quantity Released (Ci)	Quantity Released (g)	Fraction of ECV[1]
Stack 573 Bldg 306-W		**77.95 m^3/min**	**1.30 m^3/sec**				
Th-230	40,080,662	5.45E-20	4.54E-19	1.00E-18	2.19E-12	1.08E-10	2.73E-06
Th-231	40,080,662	1.34E-16	3.07E-16	5.69E-16	5.37E-09	1.01E-14	1.49E-08
U-234	40,080,662	1.22E-15	1.02E-14	2.23E-14	4.88E-08	7.83E-06	2.44E-02
U-235	40,080,662	2.05E-17	1.70E-16	3.75E-16	8.20E-10	3.80E-04	3.41E-04
U-238	40,080,662	2.36E-20	1.96E-19	4.32E-19	9.44E-13	2.82E-06	3.93E-07
						Total:	**2.47E-02**
Stack 600 Bldg. 110		**361.84 m^3/min**	**6.03 m^3/sec**				
Tc-99	191,215,764	5.28E-13	3.66E-14	3.73E-14	1.01E-04	5.97E-03	5.86E-04
Th-230	191,215,764	1.12E-17	9.71E-19	8.26E-19	2.14E-09	1.06E-07	5.59E-04
Th-231	191,215,764	6.74E-15	4.67E-16	4.76E-16	1.29E-06	2.42E-12	7.49E-07
U-234	191,215,764	2.50E-13	2.17E-14	1.84E-14	4.78E-05	7.66E-03	5.00E+00
U-235	191,215,764	4.19E-15	3.64E-16	3.10E-16	8.02E-07	3.71E-01	6.99E-02
U-238	191,215,764	4.83E-18	4.19E-19	3.57E-19	9.24E-10	2.76E-03	8.05E-05
						Total:	**5.07E+00**
Stack 615 Bldg. 306-W		**42.83 m^3/min**	**0.71 m^3/sec**				
Tc-99	21,694,848	9.89E-15	2.34E-14	4.33E-14	2.14E-07	1.27E-05	1.10E-05
Th-230	21,694,848	8.34E-20	4.55E-19	9.76E-19	1.81E-12	8.96E-11	4.17E-06
Th-231	21,694,848	1.26E-16	2.99E-16	5.53E-16	2.74E-09	5.15E-15	1.40E-08
U-234	21,694,848	1.86E-15	1.02E-14	2.18E-14	4.04E-08	6.48E-06	3.73E-02
U-235	21,694,848	3.13E-17	1.71E-16	3.66E-16	6.79E-10	3.14E-04	5.21E-04
U-238	21,694,848	3.60E-20	1.96E-19	4.21E-19	7.82E-13	2.33E-06	6.00E-07
						Total:	**3.78E-02**
Stack 646 Bldg. 110		**58.04 m^3/min**	**0.97 m^3/sec**				
Tc-99	30,671,253	9.49E-15	3.64E-14	6.68E-14	2.91E-07	1.72E-05	1.05E-05
Th-230	30,671,253	-2.68E-20	7.08E-19	1.55E-18	-8.21E-13	-4.07E-11	-1.34E-06
Th-231	30,671,253	1.21E-16	4.65E-16	8.53E-16	3.72E-09	6.99E-15	1.35E-08
U-234	30,671,253	-5.98E-16	1.58E-14	3.46E-14	-1.84E-08	-2.94E-06	-1.20E-02
U-235	30,671,253	-1.00E-17	2.66E-16	5.81E-16	-3.08E-10	-1.43E-04	-1.67E-04
U-238	30,671,253	-1.16E-20	3.06E-19	6.69E-19	-3.55E-13	-1.06E-06	-1.93E-07
						Total:	**-1.21E-02**
Stack 701 Bldg. 307		**172.22 m^3/min**	**2.87 m^3/sec**				
Tc-99	91,016,454	1.38E-14	2.89E-14	5.11E-14	1.26E-06	7.46E-05	1.54E-05
Th-230	91,016,454	3.47E-19	6.24E-19	1.19E-18	3.16E-11	1.56E-09	1.74E-05
Th-231	91,016,454	1.77E-16	3.68E-16	6.53E-16	1.61E-08	3.02E-14	1.96E-08
U-234	91,016,454	7.76E-15	1.39E-14	2.65E-14	7.06E-07	1.13E-04	1.55E-01
U-235	91,016,454	1.30E-16	2.34E-16	4.45E-16	1.19E-08	5.49E-03	2.17E-03
U-238	91,016,454	1.50E-19	2.69E-19	5.13E-19	1.37E-11	4.07E-05	2.50E-06
						Total:	**1.57E-01**

[1] ECV: Effluent Concentration Value from 10-CFR-20, Appendix B. Fraction of ECV at the stack is provided for reference only. Concentrations at off-site locations are significantly less than those reported here (at stack) due to the atmospheric dispersion that occurs before the effluent exits the site.

Radioactivity in Effluent Air
January 1, 2006 to December 31, 2006

Location	Total Volume (m^3)	Activity Concentration (µCi/ml)	Error Estimate (µCi/ml)	LLD (µCi/ml)	Quantity Released (Ci)	Quantity Released (g)	Fraction of ECV[1]
Stack 702 Bldg. 307		**156.25 m^3/min**	**2.60 m^3/sec**				
Tc-99	82,576,257	6.02E-15	3.10E-14	5.76E-14	4.97E-07	2.94E-05	6.69E-06
Th-230	82,576,257	7.34E-20	6.39E-19	1.34E-18	6.06E-12	3.00E-10	3.67E-06
Th-231	82,576,257	7.69E-17	3.96E-16	7.35E-16	6.35E-09	1.19E-14	8.54E-09
U-234	82,576,257	1.64E-15	1.43E-14	2.99E-14	1.35E-07	2.17E-05	3.28E-02
U-235	82,576,257	2.75E-17	2.40E-16	5.01E-16	2.27E-09	1.05E-03	4.58E-04
U-238	82,576,257	3.17E-20	2.76E-19	5.77E-19	2.62E-12	7.81E-06	5.28E-07
						Total:	**3.32E-02**
Stack 703 Exhaust Room Air		**710.13 m^3/min**	**11.84 m^3/sec**				
Tc-99	375,287,608	4.09E-15	2.67E-14	4.99E-14	1.53E-06	9.08E-05	4.54E-06
Th-228	194,151,533	6.14E-17	8.19E-16	1.76E-15	1.19E-08	1.46E-11	3.07E-03
Th-230	375,287,608	2.70E-17	3.60E-16	7.72E-16	1.01E-08	5.02E-07	1.35E-03
Th-231	181,136,075	5.78E-17	3.29E-16	6.11E-16	1.05E-08	1.97E-14	6.42E-09
Th-232	194,151,533	5.81E-17	7.75E-16	1.66E-15	1.13E-08	1.03E-01	1.45E-02
U-234	375,287,608	7.62E-16	1.03E-14	2.17E-14	2.86E-07	4.58E-05	1.52E-02
U-235	375,287,608	5.18E-17	6.93E-16	1.48E-15	1.94E-08	8.99E-03	8.63E-04
U-238	375,287,608	1.72E-17	2.29E-16	4.91E-16	6.45E-09	1.93E-02	2.86E-04
						Total:	**3.53E-02**
Stack 704 Process Exhaust (H2)		**61.41 m^3/min**	**1.02 m^3/sec**				
Tc-99	32,451,504	5.22E-15	2.89E-14	5.37E-14	1.69E-07	1.00E-05	5.80E-06
Th-228	12,473,051	7.22E-18	8.76E-16	1.91E-15	9.00E-11	1.10E-13	3.61E-04
Th-230	32,451,504	2.44E-18	2.86E-16	6.26E-16	7.93E-11	3.92E-09	1.22E-04
Th-231	19,978,453	7.01E-17	3.58E-16	6.62E-16	1.40E-09	2.63E-15	7.79E-09
Th-232	12,473,051	6.83E-18	8.29E-16	1.81E-15	8.52E-11	7.82E-04	1.71E-03
U-234	32,451,504	1.95E-15	1.21E-14	2.48E-14	6.33E-08	1.01E-05	3.90E-02
U-235	32,451,504	3.61E-17	6.17E-16	1.32E-15	1.17E-09	5.43E-04	6.02E-04
U-238	32,451,504	1.54E-18	1.82E-16	3.98E-16	4.99E-11	1.49E-04	2.56E-05
						Total:	**4.18E-02**

[1] ECV: Effluent Concentration Value from 10-CFR-20, Appendix B. Fraction of ECV at the stack is provided for reference only. Concentrations at off-site locations are significantly less than those reported here (at stack) due to the atmospheric dispersion that occurs before the effluent exits the site.

Table 13A-2

Radioactivity in Effluent Liquid
January 1, 2006 to December 31, 2006

Location	Total Volume (l)	Activity Concentration (µCi/ml)	Error Estimate (µCi/ml)	LLD (µCi/ml)	Quantity Released (Ci)	Quantity Released (g)	Fraction of ECV[1]
BLEU Sewer							
Pu-238	11,387,339	6.67E-13	1.22E-10	3.14E-10	7.60E-09	4.44E-10	3.34E-05
Pu-239/240	11,387,339	-1.97E-11	1.05E-10	3.18E-10	-2.24E-07	-3.61E-06	-9.85E-04
Tc-99	11,387,339	1.53E-08	3.70E-08	6.29E-08	1.75E-04	1.03E-02	2.56E-04
Th-228	11,387,339	1.14E-10	2.01E-10	3.92E-10	1.30E-06	1.58E-09	5.69E-04
Th-230	11,387,339	3.38E-10	2.12E-10	3.10E-10	3.85E-06	1.90E-04	3.38E-03
Th-232	11,387,339	1.26E-11	1.13E-10	2.80E-10	1.44E-07	1.32E+00	4.22E-04
U-233/234	7,132,705	5.54E-10	3.58E-10	3.81E-10	3.95E-06	6.33E-04	1.85E-03
U-234	4,254,634	5.15E-10	2.98E-10	3.54E-10	2.19E-06	3.51E-04	1.72E-03
U-235/236	11,387,339	1.02E-10	1.61E-10	2.90E-10	1.16E-06	5.36E-01	3.39E-04
U-238	11,387,339	2.42E-10	2.33E-10	3.28E-10	2.75E-06	8.22E+00	8.06E-04
						Total:	**8.38E-03**
Sewer							
Pu-238	56,460,076	1.15E-11	8.31E-11	2.06E-10	6.52E-07	3.81E-08	5.77E-05
Pu-239/240	56,460,076	-8.56E-13	8.64E-11	2.56E-10	-4.83E-08	-7.77E-07	-4.28E-06
Tc-99	56,460,076	2.01E-08	3.61E-08	6.13E-08	1.13E-03	6.71E-02	3.35E-05
Th-228	56,460,076	2.48E-11	1.51E-10	3.74E-10	1.40E-06	1.71E-09	1.24E-05
Th-230	56,460,076	8.12E-11	1.24E-10	2.53E-10	4.58E-06	2.27E-04	8.12E-05
Th-232	56,460,076	-3.06E-12	8.43E-11	2.62E-10	-1.72E-07	-1.58E+00	-1.02E-05
U-233/234	27,128,091	9.95E-09	1.30E-09	4.78E-10	2.70E-04	4.33E-02	3.32E-03
U-234	29,331,985	2.71E-08	1.92E-09	3.47E-10	7.95E-04	1.27E-01	9.04E-03
U-235/236	56,460,076	1.01E-09	4.25E-10	3.53E-10	5.72E-05	2.65E+01	3.37E-04
U-238	56,460,076	3.45E-09	7.07E-10	3.72E-10	1.95E-04	5.81E+02	1.15E-03
						Total:	**1.40E-02**
WWTF							
Ac-227	6,470,762	1.69E-12	4.07E-13	2.10E-13	1.10E-08	1.52E-10	3.39E-04
Am-241	6,470,762	5.90E-12	1.44E-10	4.34E-10	3.82E-08	1.11E-08	2.95E-04
Cs-137	6,470,762	1.84E-09	1.88E-09	2.04E-09	1.19E-05	1.37E-07	1.84E-03
Na-22	6,470,762	3.51E-11	1.19E-09	1.88E-09	2.27E-07	3.64E-11	5.85E-06
Np-237	6,470,762	-4.25E-11	1.55E-10	5.40E-10	-2.75E-07	-3.91E-04	-2.13E-03
Pa-231	6,470,762	3.40E-12	8.17E-13	4.22E-13	2.20E-08	4.66E-07	5.67E-04
Pb-212	6,470,762	1.14E-09	3.98E-09	4.22E-09	7.39E-06	5.35E-12	5.71E-04
Pu-238	6,470,762	3.56E-11	6.92E-11	1.86E-10	2.30E-07	1.35E-08	1.78E-03
Pu-239/240	6,470,762	-6.11E-12	5.65E-11	2.06E-10	-3.95E-08	-6.35E-07	-3.05E-04
Pu-241	6,470,762	1.08E-09	9.53E-09	1.65E-08	7.02E-06	6.81E-08	1.08E-03
Ra-224	6,470,762	1.82E-08	4.58E-09	8.94E-09	1.18E-04	7.41E-10	9.11E-02
Tc-99	6,470,762	2.23E-08	3.42E-08	5.78E-08	1.45E-04	8.56E-03	3.72E-04
Th-228	6,470,762	1.50E-10	2.33E-10	4.57E-10	9.69E-07	1.18E-09	7.49E-04
Th-230	6,470,762	1.99E-10	2.16E-10	3.56E-10	1.29E-06	6.37E-05	1.99E-03
Th-231	6,470,762	2.01E-08	6.18E+02	5.48E-08	1.30E-04	2.44E-10	4.01E-04
Th-232	6,470,762	4.65E-11	1.30E-10	3.02E-10	3.01E-07	2.76E+00	1.55E-03

[1] ECV: Effluent Concentration Value from 10-CFR-20, Appendix B.

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Table 13A-2 cont.

Radioactivity in Effluent Liquid
January 1, 2006 to December 31, 2006

Location	Total Volume (l)	Activity Concentration (µCi/ml)	Error Estimate (µCi/ml)	LLD (µCi/ml)	Quantity Released (Ci)	Quantity Released (g)	Fraction of ECV[1]
WWTF							
U-232	6,470,762	1.09E-11	2.10E-10	5.28E-10	7.08E-08	3.31E-09	1.82E-04
U-233/234	6,470,762	5.52E-08	3.06E-09	3.93E-10	3.57E-04	5.72E-02	1.84E-01
U-235/236	6,470,762	3.21E-09	7.71E-10	4.02E-10	2.08E-05	9.61E+00	1.07E-02
U-238	6,470,762	9.03E-10	4.16E-10	3.51E-10	5.85E-06	1.75E+01	3.01E-03
						Total:	**2.98E-01**

[1] ECV: Effluent Concentration Value from 10-CFR-20, Appendix B.

Table 13A-3
Summary of WWTF Chemical Characteristics 2006

Characteristic	NPDES Permit Limits				Six Month Maximum for the WWTF Outfall Located at Mile 94.6 on the Nolichucky River			
	Monthly		Daily		Monthly		Daily	
	Average Conc. (mg/l)	Average Amt. (lb/day)	Maximum (mg/l)	Maximum (lb/day)	Avg Conc (mg/l)	Avg Amt (lb/day)	Max Conc (mg/l)	Max Amt (lb/day)
Flow (MGD)	Report		Report		0.014566		0.016288	
COD	-	-	370	-	-	-	359	-
Total Suspended Solids	30	-	40	-	5.81	-	29	-
Nitrogen, Ammonia Total	20	-	30	-	18.50	-	30	-
Nitrite plus Nitrate (as N)	-	-	-	558	-	-	-	196.07
Fluoride, Soluble	15	-	20	-	<10	-	<10	-
Arsenic, Total	-	-	Report	-	-	-	<0.006	-
Cadmium, Total	-	-	0.01	-	-	-	0.0067	-
Uranium, Natural, Total	2.0	-	4.0	-	0.22	-	0.93	-
Chromium, Total	-	-	Report	-	-	-	0.00346	-
Copper, Total	-	-	1.0	-	-	-	0.0258	-
Lead, Total	-	-	0.1	-	-	-	0.0046	-
Mercury, Total	0.00037	-	0.05	-	0.00045	-	0.003842	-
Nickel, Total	-	-	Report	-	-	-	0.0151	-
Zinc, Total	-	-	Report	-	-	-	0.118	-
Silver, Total	-	-	0.05	-	-	-	0.004	-
Solids, Settleable	-	-	0.5 ml/l	-	-	-	0.5	-
Tetrachloroethylene	-	-	Report	-	-	-	<0.00025	-
Chlorine, Total Res.	-	-	2.0	-	-	-	N/A	-
pH (maximum)	-	9.0	-	-	-	8.90	-	-
pH (minimum)	-	6.0	-	-	-	6.20	-	-

N/A (Mercury)- Not applicable because monthly average only applies if Mercury testing occurs four or more days a week.

N/A (Chlorine)- Not applicable because Chlorine was not used in process during period.

Table 13A-3 cont.

Characteristic	NPDES Permit Limits				Six Month Maximum for the WWTF Outfall Located at Mile 94.6 on the Nolichucky River			
	Monthly		Daily		Monthly		Daily	
	Average Conc. (mg/l)	Average Amt. (lb/day)	Maximum (mg/l)	Maximum (lb/day)	Avg Conc (mg/l)	Avg Amt (lb/day)	Max Conc (mg/l)	Max Amt (lb/day)
Flow (MGD)	Report		Report		0.014322		0.015262	
COD	-	-	370	-	-	-	363	-
Total Suspended Solids	30	-	40	-	5.22	-	18	-
Nitrogen, Ammonia Total	20	-	30	-	20.25	-	26	-
Nitrite plus Nitrate (as N)	-	-	-	420	-	-	-	99.44
Fluoride, Soluble	15	-	20	-	<10	-	<10	-
Arsenic, Total	-	-	Report	-	-	-	0.0138	-
Cadmium, Total	-	-	0.01	-	-	-	0.0046	-
Uranium, Natural, Total	2.0	-	4.0	-	0.1	-	0.17	-
Chromium, Total	-	-	Report	-	-	-	0.00184	-
Copper, Total	-	-	1.0	-	-	-	0.160	-
Lead, Total	-	-	0.1	-	-	-	0.00646	-
Mercury, Total	0.00037	-	0.05	-	0.000778	-	0.00228	-
Nickel, Total	-	-	Report	-	-	-	<0.010	-
Zinc, Total	-	-	Report	-	-	-	0.0684	-
Silver, Total	-	-	0.05	-	-	-	0.03176	-
Solids, Settleable	-	-	0.5 ml/l	-	-	-	0.1	-
Tetrachloroethylene	-	-	Report	-	-	-	<0.00025	-
Chlorine, Total Res.	-	-	2.0	-	-	-	N/A	-
pH (maximum)	-	9.0	-	-	-	8.89	-	-
pH (minimum)	-	6.0	-	-	-	6.50	-	-

N/A (Mercury)- Not applicable because monthly average only applies if Mercury testing occurs four or more days a week.

N/A (Chlorine)- Not applicable because Chlorine was not used in process during period.

Table 13A-4

Environmental Monitoring Data
Martin Creek Downstream

2006

Sampling Period		Gross Alpha pCi/l	Error +/-		Gross Beta pCi/l	Error +/-		Total U pCi/l	Error +/-		Total Pu pCi/l	Error +/-		Total Th pCi/l	Error +/-
Jan-2006	U	0.54	0.64		2.67	1.23	U	0.84	0.42	U	-0.00	0.06	U	0.08	0.24
Feb-2006		3.57	1.64		9.33	2.40	U	6.98	0.84	U	-0.06	0.13	U	0.56	0.61
Mar-2006	U	0.29	0.66		2.82	1.42	U	0.89	0.62	U	0.05	0.11	U	0.37	0.73
Apr-2006		4.31	1.51		1.62	0.77	U	3.79	1.12	U	0.01	0.20	U	0.11	0.39
May-2006		5.03	1.76	U	1.29	0.95		5.61	1.34	U	0.09	0.26	U	0.14	0.33
Jun-2006		3.67	1.25		6.28	1.40	U	4.24	1.91	U	0.03	0.27	U	0.76	0.73
Averages	**U**	**2.90**	**1.24**	**U**	**4.00**	**1.36**	**U**	**3.72**	**1.04**	**U**	**0.02**	**0.17**	**U**	**0.34**	**0.51**

Sampling Period		Gross Alpha pCi/l	Error +/-		Gross Beta pCi/l	Error +/-		Total U pCi/l	Error +/-		Total Pu pCi/l	Error +/-		Total Th pCi/l	Error +/-
Jul-2006		4.20	1.85		3.68	1.92	U	2.67	0.58	U	-0.11	0.18	U	0.09	0.17
Aug-2006		2.14	1.31		4.30	1.06	U	3.28	1.36	U	0.04	0.31	U	0.37	0.60
Sep-2006		4.82	1.99	U	2.70	1.80		4.17	1.01	U	0.09	0.19	U	-0.23	0.29
Oct-2006		3.65	0.71	U	0.89	0.59	U	2.26	0.86	U	-0.01	0.22	U	0.19	0.33
Nov-2006		2.80	0.58		1.53	0.76	U	3.62	1.13	U	0.01	0.19	U	0.29	0.38
Dec-2006		6.26	1.47		3.91	1.30		5.48	1.20	U	-0.12	0.13	U	0.31	0.41
Averages		**3.98**	**1.32**	**U**	**2.83**	**1.24**	**U**	**3.58**	**1.02**	**U**	**-0.02**	**0.20**	**U**	**0.17**	**0.36**

U = Below Lab Detection Limits
Total U = U-233/234 + U-235/236 + U-238
Total Pu = Pu-238 + Pu-239/240
Total Th = Th-228 + Th-230 + Th-232
Includes only monthly samples.

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Table 13A-5

Environmental Monitoring Data
Nolichucky River Downstream
2006

Sampling Period		Gross Alpha pCi/l	Error +/-		Gross Beta pCi/l	Error +/-		Total U pCi/l	Error +/-		Total Pu pCi/l	Error +/-		Total Th pCi/l	Error +/-
Jan-2006		6.14	1.73		7.98	1.61	U	0.66	0.38	U	-0.02	0.07		0.66	0.36
Feb-2006	U	0.19	0.50	U	1.31	1.05	U	0.33	0.51	U	-0.24	0.19	U	0.54	0.52
Mar-2006	U	0.13	0.72	U	1.11	1.29	U	0.22	0.56	U	-0.05	0.06	U	0.17	0.48
Apr-2006	U	-0.06	0.70	U	-0.23	0.55	U	0.06	0.25	U	-0.07	0.14	U	0.20	0.25
May-2006	U	0.08	0.60	U	0.80	0.70	U	0.59	0.89	U	0.03	0.18	U	0.07	0.25
Jun-2006	U	0.20	0.50	U	0.97	1.39	U	0.20	0.40	U	0.03	0.13	U	0.32	0.42
Averages	U	**1.11**	**0.79**	U	**1.99**	**1.10**	U	**0.34**	**0.50**	U	**-0.05**	**0.13**	U	**0.33**	**0.38**

Sampling Period		Gross Alpha pCi/l	Error +/-		Gross Beta pCi/l	Error +/-		Total U pCi/l	Error +/-		Total Pu pCi/l	Error +/-		Total Th pCi/l	Error +/-
Jul-2006	U	0.37	0.74	U	0.82	0.99	U	0.02	0.13	U	-0.05	0.09	U	-0.03	0.14
Aug-2006	U	1.01	1.02	U	1.15	0.97	U	0.17	0.56	U	-0.02	0.19	U	0.29	0.38
Sep-2006	U	0.17	0.69		3.00	1.67	U	0.20	0.28	U	-0.02	0.13	U	0.00	0.26
Oct-2006	U	0.36	0.55	U	0.77	0.70	U	0.11	0.29	U	-0.06	0.10	U	0.10	0.41
Nov-2006	U	-0.07	0.29	U	-0.13	0.70	U	0.32	0.43	U	0.11	0.20	U	0.19	0.39
Dec-2006	U	0.39	0.66	U	1.36	1.12	U	0.14	0.27	U	0.00	0.12	U	0.37	0.40
Averages	U	**0.37**	**0.66**	U	**1.16**	**1.02**	U	**0.16**	**0.33**	U	**-0.01**	**0.14**	U	**0.15**	**0.33**

U = Below Lab Detection Limits
Total U = U-233/234 + U-235/236 + U-238
Total Pu = Pu-238 + Pu-239/240
Total Th = Th-228 + Th-230 + Th-232
Includes only monthly samples.

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Table 13A-6
Groundwater Monitoring Data-Background Well

Groundwater Monitoring Data

Well 52 - Background Well						
Sampling Period		Gross Alpha pCi/l	Error +/-		Gross Beta pCi/l	Error +/-
Jan-2006	U	1.00	1.00	U	5.00	5.00
Feb-2006	U	-1.00	1.00	U	2.00	4.00
Mar-2006	U	1.00	2.00	U	5.00	5.00
Apr-2006	J	3.00	2.00	U	5.00	5.00
May-2006	U	2.00	2.00		12.00	6.00
Jun-2006	U	0.00	2.00	U	7.00	6.00
Jul-2006	U	1.00	2.00	U	3.00	4.00
Aug-2006	U	0.00	2.00	U	5.00	5.00
Sep-2006	U	0.00	1.00		10.00	5.00
Oct-2006	U	1.00	2.00	J	6.00	5.00
Nov-2006	U	1.00	2.00	U	4.00	5.00
Dec-2006	U	1.00	2.00	U	3.00	5.00
Mean		**0.83**			**5.58**	
Std Dev		**1.03**			**2.91**	

Only Dissolved (Filtered) Results Are Included
J = Estimated Result
U = Below Lab Detection Limits

Table 13A-7
Groundwater Monitoring Data-Bldg. 234 Wells

Groundwater Monitoring Data

Well 234-2

Sampling Period	Gross Alpha pCi/l	Gross Alpha Error +/-	Gross Beta pCi/l	Gross Beta Error +/-	U-234 pCi/l	U-234 Error +/-	U-235 pCi/l	U-235 Error +/-	U-238 pCi/l	U-238 Error +/-	Tc-99 pCi/l	Tc-99 Error +/-
1st Qtr-2006	1124.00	29.00	361.00	20.00	1252.00	83.90	54.62	5.39	9.18	1.61	15.15	1.94
2nd Qtr-2006	170.00	11.00	60.00	8.00	292.00	23.63	11.02	2.32	1.19	0.64	20.83	2.39
3rd Qtr-2006	343.00	17.00	232.00	17.00	843.00	57.04	49.74	5.02	4.25	1.04	49.81	2.45
4th Qtr-2006	264.00	14.00	135.00	12.00	430.40	29.17	25.06	3.05	2.27	0.72	36.07	2.23
Mean	**475.25**		**197.00**		**704.35**		**35.11**		**4.22**		**30.46**	
Std Dev	**438.24**		**130.04**		**433.68**		**20.62**		**3.54**		**15.63**	

Well 234-3

Sampling Period	Gross Alpha pCi/l	Gross Alpha Error +/-	Gross Beta pCi/l	Gross Beta Error +/-	U-234 pCi/l	U-234 Error +/-	U-235 pCi/l	U-235 Error +/-		U-238 pCi/l	U-238 Error +/-	Tc-99 pCi/l	Tc-99 Error +/-
1st Qtr-2006	61.00	7.00	72.00	9.00	31.37	3.43	1.11	0.65		0.66	0.40	33.69	2.20
2nd Qtr-2006	89.00	8.00	86.00	10.00	70.82	5.98	2.66	0.88		0.43	0.34	31.86	1.96
3rd Qtr-2006	59.00	7.00	96.00	11.00	30.18	3.28	1.06	0.62	J	0.17	0.20	39.29	2.33
4th Qtr-2006	43.00	6.00	65.00	9.00	19.87	2.64	1.11	0.60		0.70	0.43	32.41	2.19
Mean	**63.00**		**79.75**		**38.06**		**1.48**			**0.49**		**34.31**	
Std Dev	**19.11**		**13.91**		**22.44**		**0.79**			**0.24**		**3.41**	

Well 234-2

Sampling Period		Pu-238 pCi/l	Pu-238 Error +/-	Pu-239 pCi/l	Pu-239 Error +/-		Th-228 pCi/l	Th-228 Error +/-		Th-230 pCi/l	Th-230 Error +/-		Th-232 pCi/l	Th-232 Error +/-
1st Qtr-2006		ND	ND	ND	ND	U	0.27	0.25		0.36	0.21	U	0.06	0.12
2nd Qtr-2006	U	0.02	0.21	5.36	0.72	U	0.04	0.12		1.46	0.48	U	0.04	0.12
3rd Qtr-2006	U	-0.17	0.42	2.71	0.83	U	0.48	0.84	U	0.61	0.65	U	0.00	0.00
4th Qtr-2006	U	-0.23	0.46	2.14	0.71	U	-0.14	0.43	J	0.27	0.28	U	0.00	0.00
Mean		**-0.13**		**3.41**			**0.16**			**0.67**			**0.02**	
Std Dev		**0.13**		**1.72**			**0.27**			**0.54**			**0.03**	

Well 234-3

Sampling Period		Pu-238 pCi/l	Pu-238 Error +/-	Pu-239 pCi/l	Pu-239 Error +/-		Th-228 pCi/l	Th-228 Error +/-		Th-230 pCi/l	Th-230 Error +/-		Th-232 pCi/l	Th-232 Error +/-
1st Qtr-2006	U	-0.02	0.19	3.10	0.52	B	0.59	0.34		0.28	0.21	J	0.11	0.11
2nd Qtr-2006		0.55	0.29	14.72	1.12	U	0.18	0.15	U	0.13	0.13	U	0.00	0.00
3rd Qtr-2006	U	-0.06	0.34	3.41	0.88	U	0.00	0.19	U	0.33	0.28	U	0.05	0.09
4th Qtr-2006	U	0.06	0.56	4.11	0.96	U	-0.08	0.35		0.85	0.52	U	0.15	0.22
Mean		**0.13**		**6.33**			**0.17**			**0.40**			**0.08**	
Std Dev		**0.28**		**5.61**			**0.30**			**0.31**			**0.07**	

Only Dissolved (Filtered) Results Are Included
B = Sample Result < 5 times activity detected in the blank
J = Estimated Result
U = Below Lab Detection Limits
ND = No Data

Table 13A-8
Groundwater Monitoring Data-Plant Perimeter Wells

Well 98A

Sampling Period		Gross Alpha pCi/l	Gross Alpha Error +/-		Gross Beta pCi/l	Gross Beta Error +/-	Total U pCi/l	Tc-99 pCi/l	Tc-99 Error +/-
1st Qtr-2006	J	10.00	3.00	J	17.00	6.00	ND	ND	ND
2nd Qtr-2006		9.00	4.00	J	22.00	7.00	16.70	ND	ND
3rd Qtr-2006		10.00	3.00		22.00	7.00	ND	ND	ND
4th Qtr-2006		7.00	3.00		16.00	6.00	ND	ND	ND
Mean		**9.00**			**19.25**		**16.70**		
Std Dev		**1.41**			**3.20**		**0.00**		

Well 99A

Sampling Period		Gross Alpha pCi/l	Gross Alpha Error +/-		Gross Beta pCi/l	Gross Beta Error +/-	Total U pCi/l	Tc-99 pCi/l	Tc-99 Error +/-
1st Qtr-2006	U	2.00	2.00		13.00	6.00	ND	ND	ND
2nd Qtr-2006	U	0.00	4.00	U	10.00	11.00	ND	ND	ND
3rd Qtr-2006	U	2.00	2.00	U	6.00	6.00	ND	ND	ND
4th Qtr-2006	U	1.00	2.00		11.00	6.00	ND	ND	ND
Mean		**1.25**			**10.00**				
Std Dev		**0.96**			**2.94**				

Well 100A

Sampling Period		Gross Alpha pCi/l	Gross Alpha Error +/-		Gross Beta pCi/l	Gross Beta Error +/-	Total U pCi/l	Tc-99 pCi/l	Tc-99 Error +/-
1st Qtr-2006	U	2.00	2.00	J	13.00	6.00	ND	ND	ND
2nd Qtr-2006		4.00	3.00		15.00	6.00	ND	ND	ND
3rd Qtr-2006		4.00	2.00		15.00	6.00	ND	ND	ND
4th Qtr-2006		4.00	2.00		15.00	6.00	ND	ND	ND
Mean		**3.50**			**14.50**				
Std Dev		**1.00**			**1.00**				

Well 100B

Sampling Period		Gross Alpha pCi/l	Gross Alpha Error +/-		Gross Beta pCi/l	Gross Beta Error +/-	Total U pCi/l	Tc-99 pCi/l	Tc-99 Error +/-
1st Qtr-2006	U	-1.00	2.00	J	13.00	6.00	ND	ND	ND
2nd Qtr-2006	U	2.00	3.00		16.00	7.00	ND	ND	ND
3rd Qtr-2006	U	0.00	2.00		20.00	7.00	ND	ND	ND
4th Qtr-2006	U	0.00	2.00		18.00	7.00	ND	ND	ND
Mean		**0.25**			**16.75**				
Std Dev		**1.26**			**2.99**				

Well 101A

Sampling Period		Gross Alpha pCi/l	Gross Alpha Error +/-		Gross Beta pCi/l	Gross Beta Error +/-	Total U pCi/l		Tc-99 pCi/l	Tc-99 Error +/-
1st Qtr-2006	U	2.00	2.00	J	35.00	8.00	ND		ND	ND
2nd Qtr-2006		5.00	4.00		31.00	10.00	ND		ND	ND
3rd Qtr-2006	U	2.00	3.00		63.00	13.00	ND	U	-1.92	1.77
4th Qtr-2006	U	3.00	3.00		50.00	11.00	ND	U	-5.03	1.74
Mean		**3.00**			**44.75**				**-3.48**	
Std Dev		**1.41**			**14.66**				**2.20**	

Well 102A

Sampling Period		Gross Alpha pCi/l	Gross Alpha Error +/-		Gross Beta pCi/l	Gross Beta Error +/-	Total U pCi/l		Tc-99 pCi/l	Tc-99 Error +/-
1st Qtr-2006	U	2.00	2.00	J	17.00	7.00	ND		ND	ND
04-01-06		ND	ND		ND	ND	1.43		ND	ND
2nd Qtr-2006		3.00	2.00		14.00	6.00	ND		ND	ND
3rd Qtr-2006	U	1.00	1.00		20.00	6.00	ND		ND	ND
4th Qtr-2006	U	1.00	2.00		19.00	7.00	ND		ND	ND
Mean		**1.75**			**17.50**		**1.43**			
Std Dev		**0.96**			**2.65**		**0.00**			

Well 103A

Sampling Period		Gross Alpha pCi/l	Gross Alpha Error +/-		Gross Beta pCi/l	Gross Beta Error +/-	Total U pCi/l		Tc-99 pCi/l	Tc-99 Error +/-
Jan-2006	J	12.00	5.00	J	131.00	17.00	0.94	U	-2.00	1.84
Feb-2006	J	8.00	4.00	J	108.00	14.00	0.89	U	-1.03	3.49
Mar-2006		13.00	5.00		95.00	15.00	1.51	U	0.49	3.68
Apr-2006	J	3.00	2.00	J	26.00	7.00	2.12	U	-9.03	4.50
04-01-06		ND	ND		ND	ND	1.01		ND	ND
May-2006	U	5.00	4.00		27.00	10.00	1.93	U	-8.70	3.43
Jun-2006		5.00	3.00		14.00	6.00	2.83	U	-0.31	1.50
Jul-2006	U	3.00	4.00		57.00	12.00	0.21	U	0.15	1.62
Aug-2006		18.00	5.00		66.00	11.00	0.35	U	-5.88	2.01
Sep-2006		44.00	8.00		158.00	18.00	0.28	U	-2.68	1.70
Oct-2006	J	38.00	8.00	J	166.00	18.00	0.25	U	-3.40	1.64
Nov-2006		17.00	4.00		71.00	11.00	4.79	U	-7.38	1.78
Dec-2006	U	1.00	2.00		32.00	8.00	1.03	U	-7.30	1.71
Mean		**13.92**			**79.25**		**1.40**		**-3.92**	
Std Dev		**13.87**			**52.46**		**1.30**		**3.56**	

Table 13A-8 cont.

Well 104A									
Sampling Period		**Gross Alpha**			**Gross Beta**		**Total U**	**Tc-99**	
		pCi/l	**Error +/-**		**pCi/l**	**Error +/-**	**pCi/l**	**pCi/l**	**Error +/-**
1st Qtr-2006	U	0.00	2.00	U	6.00	5.00	ND	ND	ND
2nd Qtr-2006	U	2.00	2.00	U	6.00	5.00	ND	ND	ND
3rd Qtr-2006	U	0.00	1.00		11.00	5.00	ND	ND	ND
4th Qtr-2006	U	1.00	2.00		10.00	6.00	ND	ND	ND
Mean		**0.75**			**8.25**				
Std Dev		**0.96**			**2.63**				

Well 105A									
Sampling Period		**Gross Alpha**			**Gross Beta**		**Total U**	**Tc-99**	
		pCi/l	**Error +/-**		**pCi/l**	**Error +/-**	**pCi/l**	**pCi/l**	**Error +/-**
1st Qtr-2006	U	1.00	2.00	U	3.00	5.00	ND	ND	ND
2nd Qtr-2006	U	0.00	1.00		7.00	5.00	ND	ND	ND
3rd Qtr-2006	U	1.00	1.00	U	3.00	5.00	ND	ND	ND
4th Qtr-2006	U	0.00	2.00	U	5.00	5.00	ND	ND	ND
Mean		**0.50**			**4.50**				
Std Dev		**0.58**			**1.91**				

Well 106A									
Sampling Period		**Gross Alpha**			**Gross Beta**		**Total U**	**Tc-99**	
		pCi/l	**Error +/-**		**pCi/l**	**Error +/-**	**pCi/l**	**pCi/l**	**Error +/-**
1st Qtr-2006	U	-1.00	1.00	U	4.00	4.00	ND	ND	ND
2nd Qtr-2006	U	1.00	1.00	U	0.00	3.00	ND	ND	ND
3rd Qtr-2006	U	1.00	1.00	U	3.00	4.00	ND	ND	ND
4th Qtr-2006	U	0.00	2.00	U	3.00	4.00	ND	ND	ND
Mean		**0.25**			**2.50**				
Std Dev		**0.96**			**1.73**				

Table 13A-9
Groundwater Monitoring Data-Off-site Wells

Well 116A

Sampling Period		Gross Alpha pCi/l	Gross Alpha Error +/-	Gross Beta pCi/l	Gross Beta Error +/-	Total U pCi/l		Tc-99 pCi/l	Tc-99 Error +/-
04-01-06		ND	ND	ND	ND	0.39		ND	ND
Mean						**0.39**			
Std Dev						**0.00**			

Well 116B

Sampling Period		Gross Alpha pCi/l	Gross Alpha Error +/-	Gross Beta pCi/l	Gross Beta Error +/-	Total U pCi/l		Tc-99 pCi/l	Tc-99 Error +/-
04-01-06	U	0.06	0.77	43.40	7.60	0.37		68.00	12.00
4th Qtr-2006	U	1.14	0.90	44.50	7.50	0.25		74.00	13.00
Mean		**0.60**		**43.95**		**0.31**		**71.00**	
Std Dev		**0.76**		**0.78**		**0.08**		**4.24**	

Well 117A

Sampling Period		Gross Alpha pCi/l	Gross Alpha Error +/-	Gross Beta pCi/l	Gross Beta Error +/-	Total U pCi/l		Tc-99 pCi/l	Tc-99 Error +/-
1st Qtr-2006	U	2.00	2.00	34.00	7.00	ND		ND	ND
2nd Qtr-2006		17.00	4.00	91.00	12.00	ND		45.83	2.13
3rd Qtr-2006		7.00	3.00	64.00	11.00	ND		18.74	2.07
4th Qtr-2006		6.00	3.00	53.00	9.00	ND		15.22	2.06
Mean		**8.00**		**60.50**				**26.60**	
Std Dev		**6.38**		**23.81**				**16.75**	

Well 117B

Sampling Period		Gross Alpha pCi/l	Gross Alpha Error +/-	Gross Beta pCi/l	Gross Beta Error +/-	Total U pCi/l		Tc-99 pCi/l	Tc-99 Error +/-
1st Qtr-2006		11.00	3.00	71.00	10.00	ND		38.94	2.27
2nd Qtr-2006		10.00	3.00	51.00	9.00	ND		15.69	2.32
3rd Qtr-2006		17.00	5.00	105.00	14.00	1.60		45.08	2.39
4th Qtr-2006		10.00	4.00	86.00	12.00	0.75		44.24	2.45
Mean		**12.00**		**78.25**		**1.17**		**35.99**	
Std Dev		**3.37**		**22.88**		**0.60**		**13.80**	

Well 118A

Sampling Period		Gross Alpha pCi/l	Gross Alpha Error +/-	Gross Beta pCi/l	Gross Beta Error +/-	Total U pCi/l		Tc-99 pCi/l	Tc-99 Error +/-
04-01-06	U	1.20	1.10	6.50	2.00	3.41	U	1.90	3.70
4th Qtr-2006		3.30	1.40	10.40	2.30	4.39	U	2.00	3.40
Mean		**2.25**		**8.45**		**3.90**		**1.95**	
Std Dev		**1.48**		**2.76**		**0.69**		**0.07**	

Table 13A-9 cont.

Well 118B										
Sampling Period		Gross Alpha pCi/l	Gross Alpha Error +/-		Gross Beta pCi/l	Gross Beta Error +/-	Total U pCi/l		Tc-99 pCi/l	Tc-99 Error +/-
04-01-06	U	0.86	0.95		7.30	2.10	0.94	U	0.80	3.70
4th Qtr-2006		2.00	1.20		7.60	1.90	1.06	U	1.00	3.40
Mean		**1.43**			**7.45**		**1.00**		**0.90**	
Std Dev		**0.81**			**0.21**		**0.09**		**0.14**	

Well 119A										
Sampling Period		Gross Alpha pCi/l	Gross Alpha Error +/-		Gross Beta pCi/l	Gross Beta Error +/-	Total U pCi/l		Tc-99 pCi/l	Tc-99 Error +/-
04-01-06	U	0.29	0.79	U	1.60	1.40	0.20	U	0.10	3.80
4th Qtr-2006	U	0.21	0.67		11.00	2.40	0.35		17.70	5.10
Mean		**0.25**			**6.30**		**0.28**		**8.90**	
Std Dev		**0.06**			**6.65**		**0.11**		**12.45**	

Well 120A								
Sampling Period		Gross Alpha pCi/l	Gross Alpha Error +/-	Gross Beta pCi/l	Gross Beta Error +/-	Total U pCi/l	Tc-99 pCi/l	Tc-99 Error +/-
04-01-06	U	0.43	0.73	30.50	5.50	0.46	47.40	9.30
10-01-06	U	0.24	0.55	15.40	3.00	0.23	18.60	5.20
Mean		**0.34**		**22.95**		**0.35**	**33.00**	
Std Dev		**0.13**		**10.68**		**0.16**	**20.36**	

Well 120B								
Sampling Period		Gross Alpha pCi/l	Gross Alpha Error +/-	Gross Beta pCi/l	Gross Beta Error +/-	Total U pCi/l	Tc-99 pCi/l	Tc-99 Error +/-
04-01-06	U	2.00	1.40	28.90	5.30	0.32	41.90	8.40
10-01-06	U	0.56	0.63	24.10	4.30	0.40	30.40	6.80
Mean		**1.28**		**26.50**		**0.36**	**36.15**	
Std Dev		**1.02**		**3.39**		**0.06**	**8.13**	

Table 13A-10
Tc-99 Release Area Wells

Well 38R												
Sampling Period	**Gross Alpha**		**Gross Beta**		**U-234**		**U-235**		**U-238**		**Tc-99**	
	pCi/l	**Error +/-**	**pCi/l**	**Error +/-**	**pCi/l**	**Error +/-**	**pCi/l**	**Error +/-**	**pCi/l**	**Error +/-**	**pCi/l**	**Error +/-**
1st Qtr-2006	359.00	25.00	817.00	41.00	1210.00	81.12	70.01	6.45	30.95	3.38	418.39	5.25
2nd Qtr-2006	953.00	43.00	J 1061.00	51.00	801.70	54.62	33.31	3.86	18.57	2.44	271.05	3.95
3rd Qtr-2006	1008.00	43.00	1327.00	55.00	925.60	62.50	44.40	4.64	22.27	2.73	409.47	4.99
4th Qtr-2006	1314.00	51.00	1702.00	63.00	J 1256.00	82.69	J 85.70	7.36	28.85	3.15	587.42	5.86
Mean	**908.50**		**1226.75**		**1048.33**		**58.36**		**25.16**		**421.58**	
Std Dev	**399.27**		**379.16**		**219.96**		**23.84**		**5.74**		**129.51**	

Only Dissolved (Filtered) Results Are Included
J = Estimated Result
ND = No Data

Produced by: ______________________

Table 13A-11
Groundwater Monitoring Data-Groundwater Remediation Wells

Well 108A

Sampling Period	Gross Alpha pCi/l	Gross Alpha Error +/-	Gross Beta pCi/l	Gross Beta Error +/-	U-234 pCi/l	U-234 Error +/-	U-235 pCi/l	U-235 Error +/-	U-238 pCi/l	U-238 Error +/-	Tc-99 pCi/l	Tc-99 Error +/-
04-01-06	ND	ND	ND	ND	7.25	1.49	1.03	0.62	2.68	0.85	ND	ND
06-01-06	6.00	4.00 J	62.00	11.00	ND	ND	ND	ND	ND	ND	ND	ND
10-01-06	ND	ND	ND	ND	25.94	2.95	2.73	0.97	11.45	1.78	ND	ND
Mean	**6.00**		**62.00**		**16.60**		**1.88**		**7.07**			
Std Dev	***0.00***		***0.00***		**13.22**		**1.20**		**6.20**			

Well 109A

Sampling Period	Gross Alpha pCi/l	Gross Alpha Error +/-	Gross Beta pCi/l	Gross Beta Error +/-	U-234 pCi/l	U-234 Error +/-	U-235 pCi/l	U-235 Error +/-	U-238 pCi/l	U-238 Error +/-	Tc-99 pCi/l	Tc-99 Error +/-
04-01-06	ND	ND	ND	ND	21.86	2.71	1.25	0.65	17.68	2.35	ND	ND
10-01-06	ND	ND	ND	ND	19.33	2.47	1.65	0.73	20.58	2.56	ND	ND
Mean					**20.60**		**1.45**		**19.13**			
Std Dev					**1.79**		**0.28**		**2.05**			

Well 110A

Sampling Period	Gross Alpha pCi/l	Gross Alpha Error +/-	Gross Beta pCi/l	Gross Beta Error +/-	U-234 pCi/l	U-234 Error +/-	U-235 pCi/l	U-235 Error +/-	U-238 pCi/l	U-238 Error +/-	Tc-99 pCi/l	Tc-99 Error +/-
04-01-06	ND	ND	ND	ND	2.03	0.73	0.07	0.26	0.48	0.34	ND	ND
10-01-06	ND	ND	ND	ND	1.37	0.63	0.42	0.40	0.17	0.20	ND	ND
Mean					**1.70**		**0.25**		**0.32**			
Std Dev					**0.47**		**0.25**		**0.22**			

Well 111A

Sampling Period	Gross Alpha pCi/l	Gross Alpha Error +/-	Gross Beta pCi/l	Gross Beta Error +/-	U-234 pCi/l	U-234 Error +/-	U-235 pCi/l	U-235 Error +/-	U-238 pCi/l	U-238 Error +/-	Tc-99 pCi/l	Tc-99 Error +/-
04-01-06	ND	ND	ND	ND	121.71	9.69 U	7.99	1.62 U	12.00	1.61	ND	ND
10-01-06	ND	ND	ND	ND	38.78	3.85	3.00	0.96	4.57	1.04	ND	ND
Mean					**80.24**		**5.50**		**8.28**			
Std Dev					**58.64**		**3.53**		**5.26**			

Well 93

Sampling Period	Gross Alpha pCi/l	Gross Alpha Error +/-	Gross Beta pCi/l	Gross Beta Error +/-	U-234 pCi/l	U-234 Error +/-	U-235 pCi/l	U-235 Error +/-	U-238 pCi/l	U-238 Error +/-	Tc-99 pCi/l	Tc-99 Error +/-
04-01-06	ND	ND	ND	ND	58.23	5.93	4.95	1.42	8.69	1.76	ND	ND
10-01-06	ND	ND	ND	ND	37.85	3.88	1.89	0.86	2.88	0.86	ND	ND
Mean					**48.04**		**3.42**		**5.78**			
Std Dev					**14.41**		**2.17**		**4.11**			

OW-1

Sampling Period	Gross Alpha pCi/l	Gross Alpha Error +/-	Gross Beta pCi/l	Gross Beta Error +/-	U-234 pCi/l	U-234 Error +/-	U-235 pCi/l	U-235 Error +/-	U-238 pCi/l	U-238 Error +/-	Tc-99 pCi/l	Tc-99 Error +/-
04-01-06	ND	ND	ND	ND	28.70	3.15	2.92	0.98	10.42	1.67	ND	ND
10-01-06	ND	ND	ND	ND	27.32	3.06	2.94	0.95	11.80	1.80	ND	ND
Mean					**28.01**		**2.93**		**11.11**			
Std Dev					**0.98**		**0.01**		**0.98**			

Only Dissolved (Filtered) Results Are Included
B = Sample Result < 5 times activity detected in the blank
J = Estimated Result
U = Below Lab Detection Limits
ND = No Data

Table 13A-12

SUMMARY OF ENVIRONMENTAL AIR SAMPLING AVERAGE GROSS RADIOACTIVITY (uCi/ml)

Parameter Location	1st Half 2006	2nd Half 2006
Gross Alpha		
170-Perimeter NW	9.25E-16	1.45E-15
171-Perimeter W	1.00E-15	1.82E-15
172-Perimeter S	8.80E-16	1.66E-15
174-Perimeter E #1	6.67E-16	1.01E-15
218-Perimeter E #2	5.99E-16	1.53E-15
322-Little Mountain	1.55E-15	9.23E-16
323-Banner Hill Road	9.55E-16	1.37E-15
324-Asheville Hwy.	1.35E-15	1.46E-15
372-Parking Lot Entrance	8.00E-16	1.56E-15
381-BH Road/Stalling	7.45E-16	1.44E-15
382-Stalling Lane SE	1.56E-15	1.42E-15
383-Highland Ave./1st St.	1.58E-15	1.55E-15
384-Spar Mill Road	1.53E-15	1.37E-15
385-Security Fence SE	1.09E-15	1.64E-15
553-N. NFS Mound	1.27E-15	2.09E-15
555-Perimeter ENE	7.95E-16	8.23E-16
556-South of BLEU Facility	1.47E-15	1.41E-15
557-West of BLEU Facility	1.85E-15	9.92E-16
581-Images West	1.47E-15	1.68E-15
582-NFS Training Center	1.79E-15	1.53E-15
668-Perimeter W	7.70E-16	1.60E-15
677-East Burial Ground	8.13E-16	1.38E-15
678-NorthEast Burial Ground	9.64E-16	1.65E-15
679-North Burial Ground	9.03E-16	1.51E-15

a - Result not Verified
b - Result not Validated

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SUMMARY OF ENVIRONMENTAL AIR SAMPLING
AVERAGE GROSS RADIOACTIVITY (uCi/ml)

Parameter *Location*	**1st Half** *2006*	**2nd Half** *2006*
Gross Beta		
170-Perimeter NW	3.58E-14	4.28E-14
171-Perimeter W	4.01E-14	4.71E-14
172-Perimeter S	2.99E-14	4.62E-14
174-Perimeter E #1	1.38E-14	2.82E-14
218-Perimeter E #2	2.25E-14	3.58E-14
322-Little Mountain	3.61E-14	2.82E-14
323-Banner Hill Road	3.20E-14	4.20E-14
324-Asheville Hwy.	3.47E-14	4.20E-14
372-Parking Lot Entrance	3.14E-14	4.19E-14
381-BH Road/Stalling	3.16E-14	4.15E-14
382-Stalling Lane SE	3.66E-14	4.09E-14
383-Highland Ave./1st St.	3.62E-14	4.26E-14
384-Spar Mill Road	3.60E-14	3.95E-14
385-Security Fence SE	3.30E-14	4.40E-14
553-N. NFS Mound	3.51E-14	4.19E-14
555-Perimeter ENE	1.28E-14	1.52E-14
556-South of BLEU Facility	3.44E-14	4.14E-14
557-West of BLEU Facility	3.72E-14	3.79E-14
581-Images West	3.82E-14	4.50E-14
582-NFS Training Center	3.98E-14	4.50E-14
668-Perimeter W	2.95E-14	3.86E-14
677-East Burial Ground	2.99E-14	3.95E-14
678-NorthEast Burial Ground	3.19E-14	4.47E-14
679-North Burial Ground	3.18E-14	4.25E-14

a - Result not Verified
b - Result not Validated

Environmental Air Sampler Data
January through December 2006

Environmental Air Sampling Station	Average Concentration (µCi/ml)[1]									Uncorrected		Background Corrected[4]	
	Th-228	Th-230	Th-232	U-232	U-233/234	U-235/236	U-238	Pu-238	Pu-239/240	SOF[2]	Dose[3] (mrem)	SOF[2]	Dose[3] (mrem)
Background Sampler													
# 324 (Asheville Highway, A-5)	4.0E-17	6.5E-17	5.1E-17	-2.3E-18	8.8E-17	7.5E-18	9.0E-17	5.9E-18	6.3E-18	0.0217	1.09		
Group 1: Offsite Samplers (Located Beyond NFS Facility Property Boundary)													
# 322 (Little Mountain, A-8)	5.6E-17	5.3E-17	3.7E-17	9.2E-18	9.8E-17	2.1E-17	4.6E-17	4.6E-18	4.4E-18	0.0192	0.96	-0.0025	-0.12
# 382 (Stalling Lane SE, A-3)	2.4E-17	5.7E-17	4.3E-17	2.0E-17	1.1E-16	1.2E-17	7.0E-17	-1.9E-18	-6.0E-19	0.0201	1.01	-0.0016	-0.08
# 383 (First St, A-4)	2.3E-17	5.1E-17	2.2E-17	-7.0E-18	1.2E-16	4.7E-17	5.5E-17	2.1E-18	5.1E-18	0.0130	0.65	-0.0087	-0.44
# 384 (Spar Mill Rd, A-6)	3.5E-17	5.9E-17	1.0E-17	-8.7E-18	9.7E-17	-1.1E-17	2.3E-17	9.4E-18	9.7E-18	0.0095	0.48	-0.0122	-0.61
# 556 (BLEU Facility S)	4.1E-17	4.9E-17	2.1E-17	1.3E-17	9.4E-17	1.1E-17	5.9E-17	9.3E-18	1.1E-18	0.0146	0.73	-0.0071	-0.36
# 557 (BLEU Facility W)	2.3E-17	1.1E-16	2.1E-17	1.7E-18	1.2E-16	-1.4E-17	4.1E-17	1.8E-18	-3.6E-18	0.0151	0.75	-0.0066	-0.33
# 581 (Industrial Park at Images, A-7)	2.3E-17	5.3E-17	4.4E-17	1.2E-17	1.2E-16	1.4E-17	5.3E-17	-1.5E-18	9.1E-18	0.0200	1.00	-0.0018	-0.09
# 582 (NFS Training Center)	3.3E-17	2.5E-17	2.0E-17	1.2E-17	5.1E-17	2.6E-17	4.5E-17	-3.1E-18	9.7E-18	0.0116	0.58	-0.0101	-0.51
Group 2: Boundary Samplers (Located Onsite Near NFS Facility Property Boundary)													
# 323 (Banner Hill Rd, A-2)	4.6E-17	7.7E-17	4.2E-17	3.1E-17	1.0E-16	9.4E-18	5.1E-17	1.1E-17	-4.7E-18	0.0232	1.16	0.0015	0.07
# 372 (Banner Hill Rd at NFS Entrance)	2.1E-17	4.8E-17	2.6E-17	3.6E-18	8.8E-17	2.0E-17	5.0E-17	1.9E-18	7.3E-18	0.0136	0.68	-0.0081	-0.41
# 381 (Banner Hill Rd at Stalling Lane)	2.9E-17	1.6E-17	1.7E-17	-9.6E-18	1.2E-16	1.9E-17	5.3E-17	-3.1E-18	-1.6E-18	0.0090	0.45	-0.0127	-0.63
# 385 (Security Fence SE)	4.2E-17	4.3E-17	1.8E-17	4.1E-18	1.0E-16	8.3E-18	5.8E-17	8.8E-19	-2.0E-18	0.0121	0.60	-0.0096	-0.48
# 553 (NW of Burial Ground)	1.3E-17	2.6E-18	2.0E-17	-4.9E-18	1.7E-16	1.3E-17	4.1E-17	1.8E-18	1.2E-17	0.0102	0.51	-0.0115	-0.58
# 677 (East of Burial Ground)	3.8E-17	1.6E-17	4.2E-17	-2.0E-18	1.4E-16	1.5E-17	7.6E-17	-5.3E-18	1.8E-17	0.0180	0.90	-0.0037	-0.19
# 678 (NE of Burial Ground, A-1)	2.9E-17	6.0E-17	2.2E-17	2.1E-17	1.1E-16	5.3E-18	6.4E-17	2.9E-18	-7.2E-19	0.0154	0.77	-0.0063	-0.31
# 679 (North of Burial Ground)	3.3E-17	6.6E-17	4.5E-17	2.1E-19	2.5E-16	1.3E-17	5.1E-17	7.8E-18	1.0E-17	0.0232	1.16	0.0014	0.07

Environmental Air Sampling Station	Th-228	Th-230	Th-232	U-232	U-233/234	U-235/236	U-238	Pu-238	Pu-239/240	Uncorrected SOF[2]	Uncorrected Dose[3] (mrem)	Background Corrected[4] SOF[2]	Background Corrected[4] Dose[3] (mrem)
Environmental Air Sampler Data January through December 2006													
	Average Concentration (µCi/ml)[1]												
Group 3: Perimeter Samplers (Located Onsite Near NFS Facility Protected Area Boundary)													
# 170 (Perimeter NW)	2.1E-17	5.5E-17	2.1E-17	1.0E-17	2.2E-16	1.6E-16	4.5E-17	-6.8E-19	1.5E-18	0.0180	0.90	-0.0037	-0.19
# 171 (Perimeter W)	4.6E-17	2.8E-17	1.2E-17	-1.2E-17	1.9E-16	3.5E-17	3.2E-17	4.7E-18	1.5E-19	0.0106	0.53	-0.0111	-0.56
# 172 (Perimeter S)	3.3E-17	3.1E-17	3.1E-17	1.5E-17	6.7E-17	5.4E-18	7.0E-17	-2.4E-18	-7.0E-19	0.0148	0.74	-0.0069	-0.35
# 174 (Perimeter E)	4.0E-17	2.1E-17	2.1E-17	3.5E-18	1.6E-16	9.0E-18	3.0E-17	1.6E-18	-3.1E-18	0.0125	0.62	-0.0093	-0.46
# 218 (Duplicate of #174)	3.5E-17	3.3E-17	3.0E-17	6.1E-18	1.3E-16	2.5E-17	4.1E-17	-2.3E-18	-1.2E-18	0.0150	0.75	-0.0068	-0.34
# 555 (Perimeter N)	3.8E-17	2.6E-17	2.7E-17	-1.6E-17	5.8E-16	1.1E-16	7.6E-17	1.7E-18	-1.3E-18	0.0230	1.15	0.0013	0.06
# 668 (Perimeter W)	3.1E-17	2.5E-17	2.0E-17	-7.0E-18	1.8E-16	2.7E-17	1.0E-16	1.5E-18	-5.2E-18	0.0128	0.64	-0.0089	-0.45

Notes:

1 - NM = not measured; Negative values occur when the radioactivity of the sample is less than the background radioactivity measured by the detector.

2 - Sum of fractions (SOF) is the sum of the ratios of the radionuclide concentration at the sampler to the values listed in 10 CFR Part 20, Appendix B, Table II, Column 1.

3 - The effective dose equivalent (TEDE) is estimated by multiplying the SOF value by 50, because a SOF of 1.0 corresponds to a TEDE of 50 mrem.

4 - Negative values occur when the concentrations at the sampler location are less than the concentrations at the background sampler

Table 13A-14
Average Radioactivity in Stream, Sediment, Soil, and Vegetation 2006

Sampling Period		Gross Alpha pCi/g	Error +/-		Gross Beta pCi/g	Error +/-		Total U pCi/g	Error +/-		Total Pu pCi/g	Error +/-		Total Th pCi/g	Error +/-	Tc-99 pCi/g	Error +/-
Sediment																	
Martin Creek Down @ Linear Pa		11.90	3.66		18.70	4.06		2.06	0.42	U	0.01	0.05		2.34	0.55		
Martin Creek Down @ RR Trestl	U	2.89	1.79	U	3.72	2.78	U	2.82	0.55	U	0.22	0.10		1.99	0.49		
Martin Creek Up		4.40	2.25	U	6.57	3.08	U	0.89	0.29	U	0.00	0.03		2.27	0.57		
Nolichucky River Downstream	U	6.87	3.18		18.05	4.15	U	0.73	0.26	U	-0.01	0.03		1.07	0.37		
Nolichucky River Upstream		11.33	3.68		20.35	4.33		1.90	0.43	U	0.01	0.05		2.73	0.65		
Soil																	
Asheville Highway		17.75	5.60		34.35	6.49	U	2.01	0.45	U	0.04	0.07		3.72	0.76		
Banner Hill Road		20.45	4.52		18.25	3.91		3.67	0.67	U	0.00	0.04		4.68	0.69		
Burial Ground		26.30	5.84		27.35	4.85		4.49	0.77	U	0.00	0.04		6.89	1.18		
First Street		18.65	4.44		16.35	4.14		2.77	0.51	U	0.02	0.04		4.43	0.69		
Little Mountain		9.03	2.95		11.15	3.58		2.87	0.54	U	0.00	0.03		4.31	0.66		
Vegetation																	
Asheville Highway	U	0.71	1.76		14.98	3.97	U	0.09	0.11	U	-0.01	0.04	U	0.08	0.15		
Banner Hill Road	U	3.08	2.29		18.95	4.26		0.89	0.37	U	0.00	0.04	U	1.10	0.53		
Burial Ground	U	3.61	2.61		20.35	4.34	U	0.56	0.22	U	-0.00	0.06	U	0.76	0.34		
First Street	U	4.43	2.57		16.70	4.15	U	0.73	0.39	U	-0.01	0.02		0.95	0.36		
Little Mountain	U	0.07	1.44		20.85	4.10	U	0.91	0.29	U	0.02	0.05	U	1.13	0.45		

U = Below Lab Detection Limits
Total U = U-233/234 + U-235/236 + U-238
Total Pu = Pu-238 + Pu-239/240
Total Th = Th-228 + Th-230 + Th-232

Sampling Period		Gross Alpha pCi/g	Error +/-		Gross Beta pCi/g	Error +/-		Total U pCi/g	Error +/-		Total Pu pCi/g	Error +/-		Total Th pCi/g	Error +/-	Tc-99 pCi/g	Error +/-
Sediment																	
Martin Creek Down @ Linear Pa		10.45	2.37		13.95	2.05	U	1.49	0.52	U	0.06	0.07		2.00	0.68		
Martin Creek Down @ RR Trestl		5.93	1.79		7.37	1.65	U	0.56	0.31	U	0.01	0.06		1.01	0.53		
Martin Creek Up		7.10	2.09		7.51	1.75	U	0.55	0.28	U	0.01	0.05		1.16	0.54		
Nolichucky River Downstream	U	8.73	2.19		20.90	2.59	U	1.08	0.40	U	0.00	0.05		1.58	0.62		
Nolichucky River Upstream		13.20	3.03		18.40	2.30	U	0.41	0.26	U	0.00	0.05		1.04	0.52		
Soil																	
Asheville Highway		18.50	3.38		35.70	3.20	U	1.55	0.48	U	0.02	0.07		3.70	0.92		
Banner Hill Road		12.10	2.57		26.80	2.84	U	2.87	0.70	U	0.00	0.05		4.00	0.91		
Burial Ground		19.65	3.25		28.70	2.66	U	4.29	0.98	U	0.00	0.05		6.40	1.26		
First Street		11.90	2.53		13.10	2.38	U	2.50	0.68	U	-0.00	0.04		4.25	0.99		
Little Mountain		11.25	2.36		9.04	1.79	U	3.65	0.90	U	0.02	0.04		3.78	0.87		
Vegetation																	
Asheville Highway	U	2.60	1.32		17.65	2.34	U	0.26	0.24	U	0.00	0.06	U	0.38	0.31		
Banner Hill Road	U	1.10	1.17		27.40	2.78	U	0.14	0.16	U	0.00	0.05	U	0.25	0.28		
Burial Ground	U	1.09	1.17		18.45	2.16	U	0.19	0.19	U	-0.02	0.05	U	0.22	0.27		
First Street	U	1.04	1.42		20.40	2.67	U	0.09	0.14	U	-0.00	0.04	U	0.14	0.22		
Little Mountain	U	2.34	1.42		17.25	2.25	U	0.56	0.28	U	-0.01	0.04	U	0.71	0.36		

U = Below Lab Detection Limits
Total U = U-233/234 + U-235/236 + U-238
Total Pu = Pu-238 + Pu-239/240
Total Th = Th-228 + Th-230 + Th-232

Table 13A-15

Summary of Environmental Dosimeter Data Assuming Occupancy Factor of One (1)						
Dosimeter Location		2006 1st Qtr (mrem)	2nd Qtr (mrem)	3rd Qtr (mrem)	4th Qtr (mrem)	YTD (mrem)
D001 - Background Dosimeter Asheville Highway at Env Air Sample Station 324	Gross	13	12	17	11	53
D002 - Carolina Ave at Env Air Sample Station 323	Gross	1	2	7	3	
	Net	0	0	0	0	0
D003 - Little Mountain at Env Air Sample Station 322	Gross	7	9	10	5	
	Net	0	0	0	0	0
D005 - North property boundary near Env Air Sample Station 553	Gross	6	6	10	9	
	Net	0	0	0	0	0
D013 - West of Building 131 on Protected Area fence	Gross	7	4	8	4	
	Net	0	0	0	0	0
D015 - West of Building 410 on Protected Area fence	Gross	9	7	8	9	
	Net	0	0	0	0	0
D016 - Located west of Building 310 on Protected Area fence	Gross	17	20	25	16	
	Net	4	8	8	5	25
D018 - East property boundary fence near midpoint of main parking lot	Gross	5	7	6	6	
	Net	0	0	0	0	0
D019 - North property boundary at Env Air Sample Station 677	Gross	9	6	9	5	
	Net	0	0	0	0	0
D020 - North property boundary at Env Air Sample Station 678	Gross	8	7	10	5	
	Net	0	0	0	0	0
D021 - North property boundary at Env Air Sample Station 679	Gross	7	5	8	2	
	Net	0	0	0	0	0
D022 - South property boundary south of Building 350	Gross	11	11	20	12	
	Net	0	0	3	1	4
D029 - West property boundary for the BLEU Complex	Gross	14	10	13	9	
	Net	1	0	0	0	1
D030 - Due south of BLEU Complex vehicle gate along property boundary	Gross	14	13	12	10	
	Net	1	1	0	0	2
D031 - Boundary fence between Bldgs. 300B and 310	Gross	9	7	11	7	
	Net	0	0	0	0	0
D032 - Southwest of 310 Building on Protected Area fence	Gross	15	11	15	8	
	Net	2	0	0	0	2
D033 - On start of fence bordering gravel road to upper parking lot	Gross	11	11	17	7	
	Net	0	0	0	0	0

TOTAL MONTHLY PRECIPITATION
January 1, 2006 through December 31, 2006
As Measured in Erwin, TN

MONTH	PRECIPITATION AS MEASURED IN ERWIN
January 2006	3.73 in
February 2006	2.49 in
March 2006	3.18 in
April 2006	7.00 in
May 2006	6.50 in
June 2006	5.81 in
July 2006	4.24 in
August 2006	3.76 in
September 2006	5.43 in
October 2006	3.09 in
November 2006	2.89 in
December 2006	0.69 in
TOTAL:	**48.81 in**

Figure 13A.1

Quarterly Offsite Dose - Air

Value/Goal Supported

H,S & Env. Compliance
Enhance Image with NRC

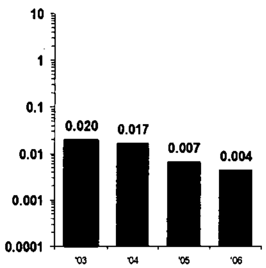
Metric Formula

The maximum dose (total effective dose equivalent in units of mrem) to an offsite receptor in a calendar quarter from plant gaseous effluents (monitored stacks and vents).

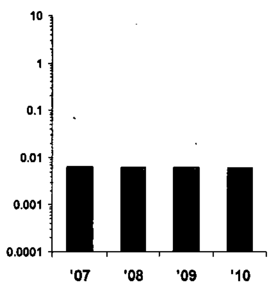
Notes

Based on methods described in procedure NFS-HS-A-27.

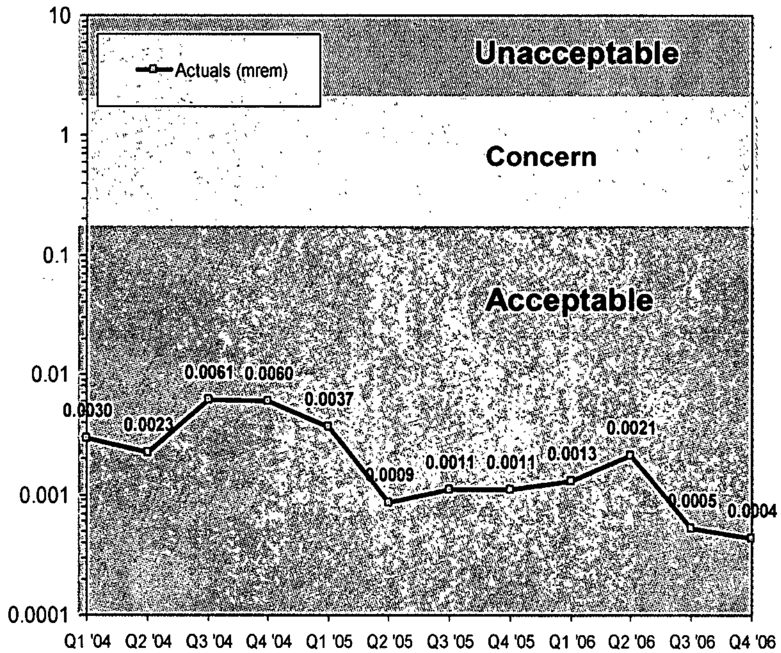
Historical Annual Total (mrem)



Annual Stretch Goals (mrem)



Current Performance



Company Confidential

Enhancement Plans

None identified at this time

Benchmarks

None identified at this time

Figure 13A.2

Quarterly Offsite Dose - WWTF

Value/Goal Supported

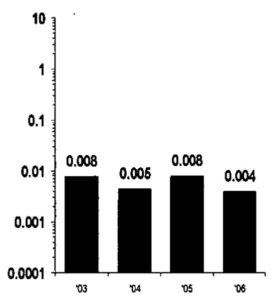
H,S & Env. Compliance
Enhance Image with NRC

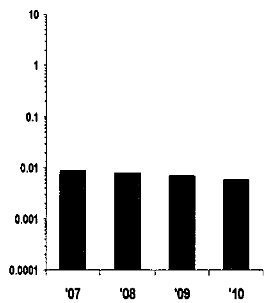
Metric Formula

The maximum dose (total effective dose equivalent in units of mrem) at the City of Jonesborough Water Intake in a calendar quarter.

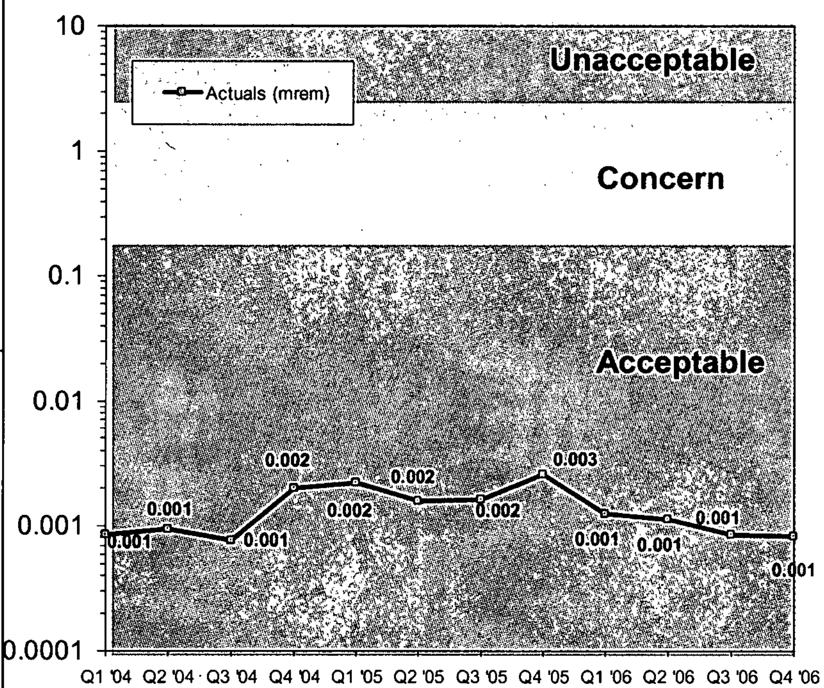
Notes

Based on methods described in NCRP 123, Parts I and II & NFS-HS-A-66.





Current Performance



Company Confidential

Enhancement Plans

None identified at this time

Benchmarks

None identified at this time

Figure 13A.3

SMS Quarterly Offsite Dose - External

Value/Goal Supported

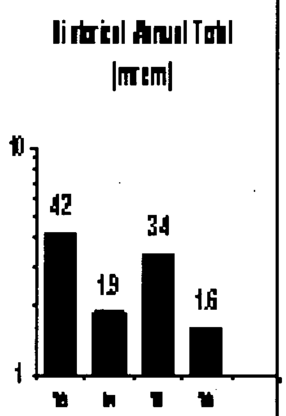
H,S & Env. Compliance
Enhance Image with NRC

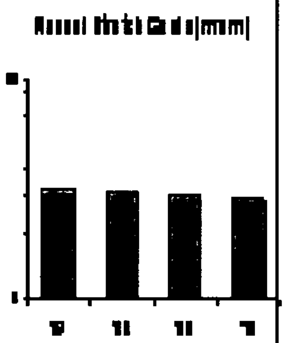
Metric Formula

Maximum net external dose (deep-dose equivalent in units of mrem) based on the highest dosimeter measurement at or beyond NFS' property boundaries. These values have been adjusted to account for occupancy, per the specifications of procedure NFS-HS-A-82.

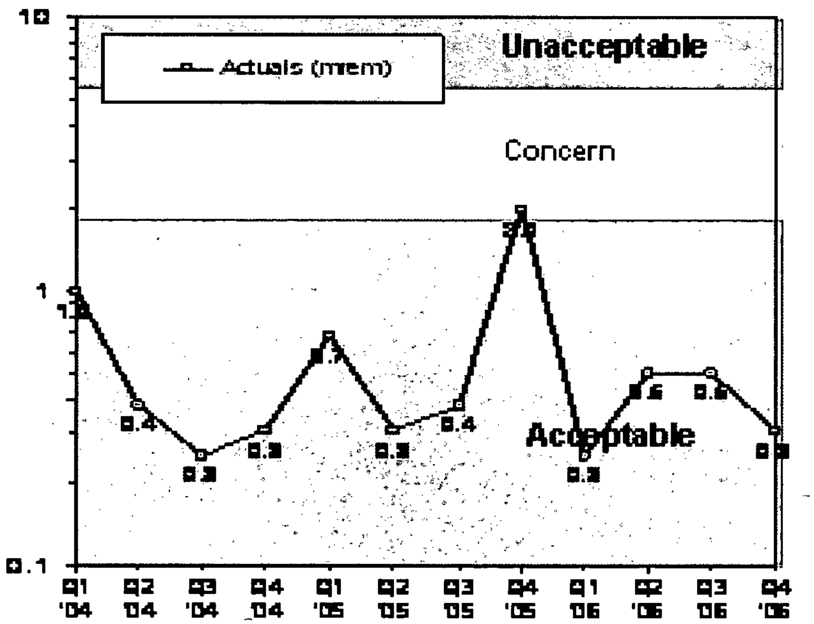
Notes

Based on methods described in procedure NFS-HS-A-82.





Current Performance



Company Confidential

Enhancement Plans

Additional dosimeters placed 03/06 as diagnostic tools around locations of interest

Benchmarks

None identified at this time

Figure 13A.4

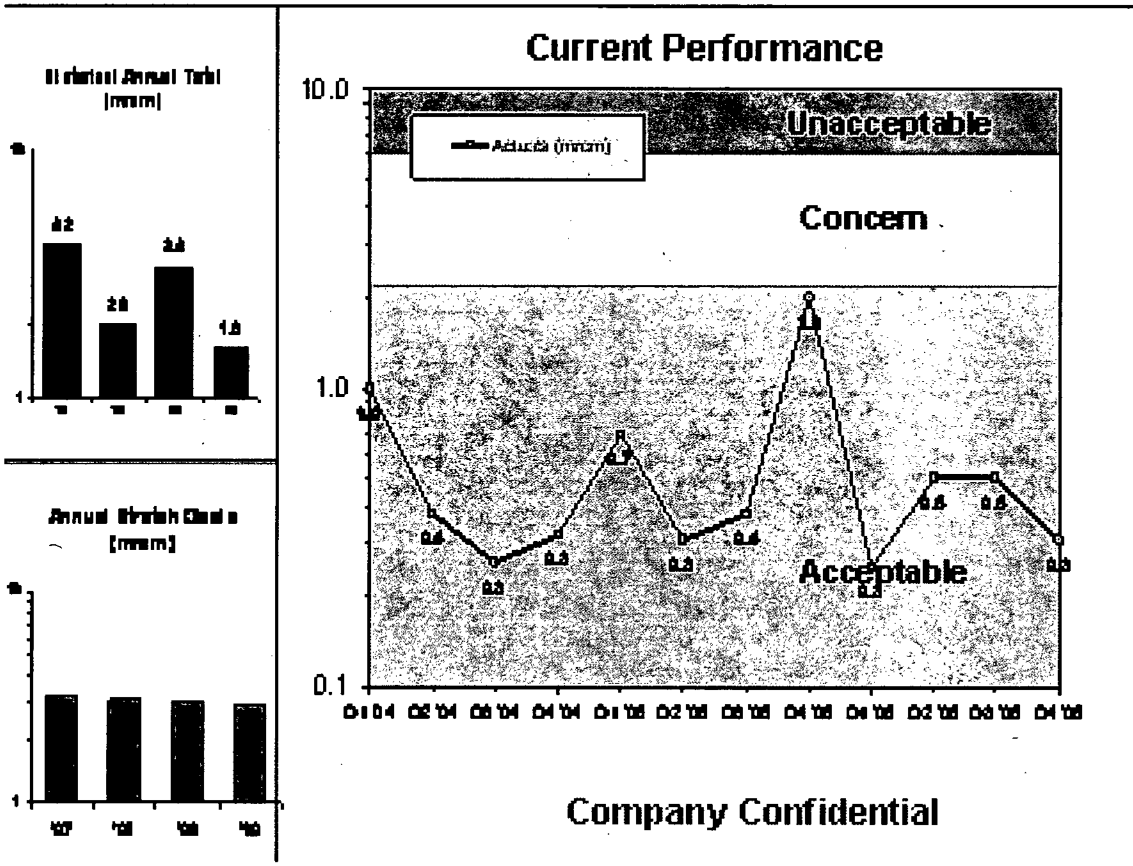
Offsite Total Effective Dose Equivalent (TEDE)

Value/Goal Supported

H, S & Env. Compliance
Enhance Image with NRC

Metric Formula

The TEDE to the maximally exposed individual (MEI) from all effluent pathways (i.e. gaseous, liquid and ambient radiation) regardless of location. This calculated dose is very conservative because the MEI locations for each of these effluent pathways are normally very different from each other, such that a single individual would never receive the entire estimated dose.



Enhancement Plans

None identified at this time

Benchmarks

None identified at this time

NUCLEAR CRITICALITY SAFETY

A. Administrative Requirements

This chapter provides additional information regarding the administrative and technical practices, design approaches, nuclear criticality safety (NCS) methodology, and analytical methods which are used to ensure NCS in the processing, handling, and storage of fissile material.

14.1 Administrative and Technical Practices

The administrative and technical practices described below include details in the areas of procedures, training, and NCS responsibilities.

14.1.1 Operating Procedures

All activities involving SNM are described in procedures which incorporate NCS limits and controls. These limits and controls are described in Criticality and Licensing (CL) procedures and are included in standard operating procedures as appropriate. These procedures include specifications regarding posting of limits, storage and transfer of SNM, and use of unfavorable geometry equipment, including sampling and inspection thereof.

They may also include forms for documenting the performance of NCS-related activities. The CL procedures are reviewed and approved by the Safety and Safeguards Review Council (SSRC).

One other form of instruction is the Letter of Authorization (LOA) as described in Chapter 11. The LOA allows deviation from an operating procedure. To assure safety, all LOAs are reviewed and approved by the SSRC.

14.1.2 Safety Procedures

The major activities of the NCS function that are required to be performed by the license are described in safety procedures. These activities include, but are not limited to, conducting audits and inspections and performing NCS analyses.

14.1.3 Training

All personnel who are authorized to have unescorted access into the restricted areas of the plant receive training in NCS commensurate with their function before entry. This training is given either by a member of the NCS function or by the Training Department. NCS training is augmented by refresher training on an annual basis.

14.1.4 Nuclear Criticality Safety Responsibilities

An effective NCS program includes cooperation among management, supervision, and the NCS function and relies upon conformance with operating procedures by all employees. Management has overall responsibility for safety of operations. Management assigns responsibility and delegates commensurate authority to implement the NCS program.

Management provides personnel familiar with the physics of nuclear criticality and with associated safety practices to furnish authoritative professional advice and counsel to discipline managers and their representatives on NCS matters. The NCS function provides technical guidance for the design of equipment and processes and for the development of operating procedures. The NCS function also analyzes proposed changes in equipment and processes involving SNM. To measure the effectiveness of the NCS program, the NCS function conducts audits and inspections of NCS practices and compliance with procedures.

Each supervisor is responsible for the safety of operations under his authority and is knowledgeable concerning relevant aspects of NCS. Each supervisor assures that the personnel under his supervision have an understanding of procedures and safety considerations such that they are able to perform their functions without undue risk.

14.2 Preferred Approach to Design

The preferred approach to the design of process equipment is to incorporate geometry as the primary NCS control. Geometry control is implemented in designs by restriction of key dimensions.

Whenever NCS control is directly dependent on the integrity of a structure used to retain the specified geometric limitations or the spacing within a storage array, the structure is designed with an adequate strength factor to ensure against failure under credible loads or accident conditions. Materials of construction should also be fire resistant. In addition, the degree to which any corrosive environment might affect the key dimensions of structures relied upon for NCS should be considered, and corrosive resistant materials or coatings applied as necessary.

Equipment and processes are designed to minimize the accumulation of fissile material in unplanned locations. Periodic inspections (e.g., visual or NDA measurements) during operations verify the effectiveness of the designs.

If geometry control alone is not used for ensuring NCS, other controls as described in Chapter 4 are utilized. When used together, combinations of these controls can ensure acceptable NCS after a single failure. An example is arrays of storage rockets, where the mass of fissile material inside each rocket, as well as the spacing between rockets, are used together for safety.

Whenever it is necessary to use an unfavorable geometry container, an analysis is conducted to justify its use. The analysis will identify potential contributing causes of criticality accidents, demonstrate how such causes will be subject to controls, and demonstrate how the double contingency principle is applied. Appendix B of Regulatory Guide 3.52, Appendix A of ANS-8.1, and other NCS reference materials are used for guidance in identifying potential contributing causes of criticality accidents.

14.3 Controls and Safety Margins

This section described controls on moderation, reflection, mass, and concentration/density which are implemented to maintain these parameters within acceptable limits. Limits have been established for mass, concentration, geometry, and areal density by the application of various safety margins to the applicable critical parameters. These margins and the sources for critical parameters are documented herein.

14.3.1 Moderation

When relied upon for NCS, moderation is controlled by one or more of the following techniques:

- Moderation is removed by an engineered system, and the resulting fissile material is inspected and/or sampled to verify proper functioning of the equipment.

- Moderation is added procedurally and controlled by limiting the mass and/or volume of the moderator and fissile material.

- Moderation of the fissile material is determined by analysis of samples of the material prior to using moderation as an NCS control.

The proper degree of moderation is maintained by one or more of the following methods:

- Fissile material is kept in a tightly closed container.

- Fissile material is in a glovebox or other processing equipment that contains no uncontrolled source of moderation and has no source of moderation connected directly to it.

14.3.2 Reflection

When relied upon for NCS, neutron-reflecting materials are controlled by one or more of the following techniques:

- A physical barrier is installed around each unit (or group of units) to control the close approach of reflecting media (e.g., personnel, walls, etc.). An example of such barriers includes support structures around columns.

- Postings will clearly warn personnel of those locations where reflection due to personnel and equipment must be restricted.

- Procedural controls require that the unit be emptied prior to any activities that require personnel to use ladders or scaffolding adjacent to the unit.

- The glovebox itself provides spacing between fissile units inside the glovebox and reflectors provided by the floor and ceiling. Fixed positions within a glovebox (e.g., storage rings, stationary filter housings, etc.) provide spacing between fissile units inside the glovebox and reflectors around the outside of the glovebox (e.g., personnel, etc.). Gloveboxes are designed to either preclude the introduction of liquid reflectors or are equipped with drains to minimize the accumulation of liquid reflectors.

14.3.3 Mass

When relied upon for NCS, the mass of fissile material is determined by one or more of the following techniques:

- The mass of fissile material is determined by qualified counting methods (e.g., lab analysis, scales, or NDA).

- The total mass or change in mass is determined as the product of the volume and the most reactive fissile material concentration/density assumed to be credible in the unit.

- The total mass remaining inside a unit is determined by subtracting the measured unit material outputs from the measured unit material inputs (i.e., mass balance sheet).

14.3.4 Concentration/Density

When relied upon for NCS, fissile material concentration or density is determined by one or more of the following qualified techniques:

- Dividing the mass of fissile material, as measured by qualified counting methods, by the volume occupied by the fissile material.

- On-line measurement of concentration or density (if the worst credible concentration is assumed).

The maximum uranium density used for evaluating any process or stage within a process will be the theoretical density of the most reactive credible chemical form unless the nature of the material or the process limits the density to a lesser value that will not be exceeded by any credible process upset.

14.3.5 Safety Margins

The following safety margins are used:

Parameter	Fraction of Critical
Mass	0.45/0.75
Concentration	0.45*
Cylinder Diameter	0.90
Slab Thickness/Depth	0.45/0.75/0.85
Volume	0.75
Areal density	0.45

* When precipitation, settling, or layering into regions of unsafe concentrations is not prevented, the concentration is limited to either that which limits the mass in the unit to $\leq 0.75 \times M_c$ (M_c = the minimum critical mass) or to $\leq 0.45 \times C_c$ (C_c = the minimum critical concentration), whichever is smaller.

The safety margins for mass, cylinder diameter, and volume limits are those given in Appendix B of the Regulatory Guide 3.52. The safety margins for slab thickness are equal to or greater than those given in the above Regulatory Guide. The safety margin for areal density is the same as for a mass limit when double batching is credible.

Limits for fully water reflected units in Tables 4.3 and 4.4 were developed from the information provided in References 3 and 5 for systems of UO_2 and water at an enrichment of 100 wt.% ^{235}U. Only Reference 5 was used to determine the limits for Tables 4.3 and 4.4 at enrichments of 10.0 wt.% ^{235}U or less.

In order to end up with consistent parameters, the data in the references were converted into identical units. The parameters were then plotted versus uranium density and a curve fit was performed. The curve fit equation was then used to determine the parameter value corresponding to the density of interest. It should be noted that curve fitting the mass data was not required since the minimum critical values were obtained from References 3 and 5 for all densities. Furthermore, the data in Reference 3 did not need to be curve fitted since it was presented graphically.

The data in References 3 and 5 correspond to maximum enrichments of 93.0 and 93.3 wt.% ^{235}U, respectively. Therefore, the data for each parameter was extrapolated to a 100.0 wt.% ^{235}U by using the data from the critical uranium metal-water values in Reference 1 at 93.5% and 100% (except the mass limit from Reference 5 was determined according to the method described as follows for uranium metal systems). In order to compare representative critical parameters at 93.0 and 100.0 wt.% ^{235}U, the critical dimensions for volume, slab thickness, and cylinder diameter were obtained from the maximum uranium density in Reference 1 at 1 gU/cc. This is conservative since the data indicates that the difference between the dimensions for each of these parameters between 93% and 100% decrease as the density increases. Therefore, by using 1 gU/cc the critical dimensions near 8 gU/cc are bounded. However, a density of 0.06 gU/cc was used for mass since the minimum critical mass occurs at a low density at which the difference between the 93% and 100% is minimal. The formula used for the extrapolation is provided below; (CP = Critical Parameter):

$$\mathrm{CP}(100\%) = \mathrm{CP}(\%\mathrm{Enriched}) - \frac{(\%\mathrm{enriched} - 100) \times (\mathrm{CP}(93.5\%) - \mathrm{CP}(100\%))}{(93.5 - 100)}$$

The uranium metal parameter limits for mass and volume were obtained by extending the heterogeneous curves in Reference 5, Figures 22-23 from 93.3 to 100.0 wt.% ^{235}U. Because Figures 12 and 13 had more conservative critical values for the slab thickness and cylinder diameter than Figures 24 and 25, these values were extrapolated to 100.0 wt.% ^{235}U by applying the same difference in slab thickness and cylinder diameter that was obtained from extending the curves in Figures 24 and 25, respectively.

After comparing the data generated from References 3 and 5, it was determined that Reference 5 provided the most conservative values. Therefore, the data from Reference 5 (in conjunction with the extrapolation for Reference 1) was used as the safety basis for the limits in Tables 4.3 and 4.4.

14.3.6 Soluble Poisons(1)(2)(3)(4)

Soluble boron may be used as a secondary NCS control with the following considerations and limitations:

a. The use of soluble boron in an SNM processing area shall be approved by the SSRC.

b. By secondary control, it is meant that the soluble boron will be contained in a closed system that under normal conditions allows no direct contact with SNM. If a leak occurs, the soluble boron will provide the necessary additional neutron absorption to maintain a safe system under credible accident conditions.

c. The poison-bearing material utilized will be boron of natural isotopic composition. The boron must be adequately soluble and chemically stable in the working solutions. Supporting lab analysis demonstrating compliance must be documented prior to system start-up.

d. Poison concentrations must be equal to or greater than the acceptable concentrations established by the most currently available experimental data and validated computations. Data from References 3 and 4 on boron-poisoned U(100)-H_2O solutions will be utilized unless more current validated data is available.

e. The poison-bearing solution must be prepared, sampled, and analyzed according to approved procedures. These procedures must also contain provisions for the Nuclear Criticality Safety Manager to affirm that the poison concentration and ^{10}B isotopic content are acceptable before SNM processing is authorized. Reanalysis of the poison concentration will be performed annually and also anytime that the closed system is breached or opened for any reason; results of such reanalysis must be given to the Nuclear Criticality Safety Manager so he can ensure that the poison concentration remains above the specified limit prior to restart of the process.

f. Unpoisoned streams must be physically separated from the poisoned streams or vessels to prevent accidental dilution of the poison concentration. All poisoned streams will be a closed loop system with no connections to other streams by any means.

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1 Trans. Am. Nucl. Soc., Vol. 15, pp. 86 (1972)

[2] Trans. Am. Nucl. Soc., Vol. 19, pp. 184-185 (1974)
[3] Trans. Am. Nucl. Soc., Vol. 30, pp. 256-268 (1978)
[4] Proposed ANS-8.14, Draft 7, Part 1 (1981)

14.4 Fixed Poisons

All uses of borosilicate-grass Raschig rings follow the conditions specified in Section 4.2.4. Inspections and tests of the rings are performed and documented as required by operating and/or NCS procedures.

14.5 Structural Integrity Policy and Review Program

Changes and additions made to equipment, systems, or facilities are reviewed for safety by the SSRC. In these reviews, the structural integrity is reviewed along with other safety considerations.

NUCLEAR CRITICALITY SAFETY

B. Analytical Methods

14.6 Analytical Methods

The analytical methods used by the nuclear criticality safety (NCS) function in NCS analyses are standard throughout the industry. These methods include three types. First is the use of experimental or published data along with hand calculations utilizing well-known formulas or relationships. Second are the empirical interaction techniques which are used to evaluate the NCS of complex systems or entire facilities. Third is the use of validated computer codes and cross-sections to model the system and calculate its effective neutron multiplication (k_{eff}).

14.6.1 Individual Unit Analysis

A typical analytical method used with experimental or published data is the calculation of effective neutron multiplication factors using the reactivity formula. The reactivity formula is the well-known relationship:

$$k_{eff} = \frac{k\infty}{1 + B^2 M^2}$$

Where

k_∞	is the published infinite neutron multiplication factor,
M^2	is the published migration area, and
B^2	is the geometric buckling (itself a function of the material extrapolation distance which is also obtained from published data)

Functional relationships of buckling factors on geometric dimensions are well known for simple geometries such as cylinders, slabs, and spheres. Single parameter limits for fully reflected infinite units may be modified to apply to finite units by application of these buckling relationships.

14.6.2 Interaction Analysis

The interaction analysis techniques include solid angle calculations, surface density calculations, unit storage criteria, Monte Carlo calculations (KENO and MCNP), and ANSI Standard N16.5-1975/ANS8.7, "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials." These analyses are accomplished with the use of hand calculators, computers, and tables/graphs from acceptable NCS reference documents.

14.6.2.1 Solid Angle Calculations

The solid angle calculational method is a widely recognized method for specifying safe parameters of an array of interacting units. This method assumes the array multiplication is a function of the effective neutron multiplication factor of individual components, k_{eff}, and the probability of neutron interaction between components of the array. Before using this method to demonstrate acceptable interaction for any major process modification involving equipment that is not individually safe when fully water reflected, the method will be shown to be conservative. The technique to be followed is based on the following:

- The effective multiplication factor, k_{eff}, of an individual unreflected unit is determined according to the method in Section 14.6.1.
- The solid angle method is applicable only to arrays whose units each have an optimum bare k_{eff} of not more than 0.8.
- Individual units must be separated by at least 8 inches for a two-unit array or 12 inches for an array of more than two units.
- Individual units should be subcritical under all credible conditions of moderation, geometry, and reflection.

The maximum allowed solid angle (Ω) limit (steradians) subtended at a unit of an array is given by the relation:

Ω limit = 9-(10*k-effective) (when k_{eff} is 0.3 to 0.8)

Ω limit = 6 steradians (when k_{eff} is less than 0.3)

Units or parts of units shielded from the component of interest by (a) units of the array whose interaction is accounted for, or (b) isolation by shielding, where the component of interest obeys the requirements specified in Chapter 4 for shielding, may be neglected in computing the total solid angle subtended.

In accounting for the total neutron interaction, the contribution by the following components is considered insignificant and does not have to be accounted for:

- Single components that subtend no more than 0.005 steradians at the point of interest.

- Single pipes no more than 0.5 inches in diameter.

- Components which contain homogeneous solutions of safe fissile concentrations.

- Slabs with thickness (T) no more than 15% of T(MR)(critical) and perpendicular to the longitudinal center line of the component of interest.

The total interaction solid angle subtended at a unit of an array by all the other units is the sum of the individual contributions from all other units.

Formulas used in computing solid angles of interaction are based upon the point-to-plane model.

If the array being analyzed consists of identical units, only the most centrally located unit need be considered as the center point for solid angle calculations. In other cases, it is usually necessary to choose more than one "center point" unit to assure that the k_{eff} – solid angle relationship is satisfied for all units. A component which is limited to a total fissile mass no greater than a safe fissile mass may be represented as a sphere at its most reactive conditions.

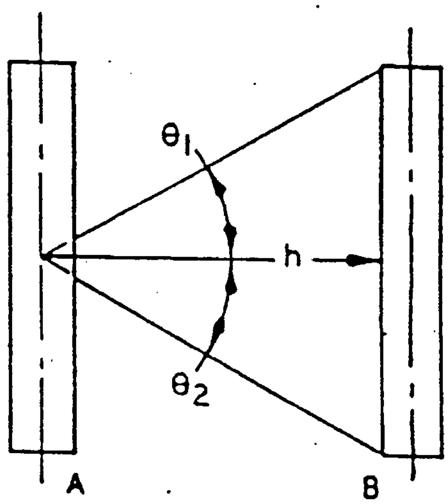
Solid angle calculations may be used to verify that the addition or relocation of process equipment will not violate the NCS criteria. Where this method is utilized to assure NCS, all equipment installed will meet the conditions required by the solid angle technique.

Examples:

Cylinders:

Calculate the solid angle subtended by cylinder B from the center of cylinder A given:

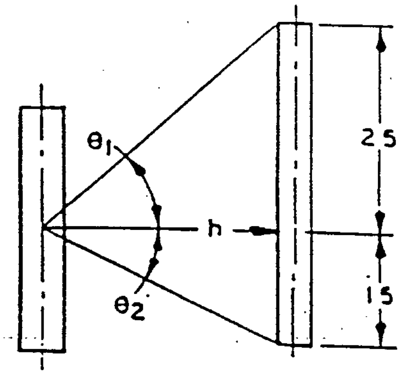
- elevations of center of cylinders A and B are both the same so that $\theta_1 = \theta_2$
- center-to-center separation = 30 inches
- diameter of cylinder B = 5 inches
- length of b = 40 inches



$\Omega = (2\ d/h)\ \sin\theta$
$h = 30\text{-}2.5 = 27.5$ inches
$\sin\theta = \sin(\tan^{-1} 20/27.5) = 0.588$
$\Omega = (2)(5)(0.588)/27.5 = 0.214$ steradians

Calculate the solid angle subtended by cylinder B from the center of cylinder A given:

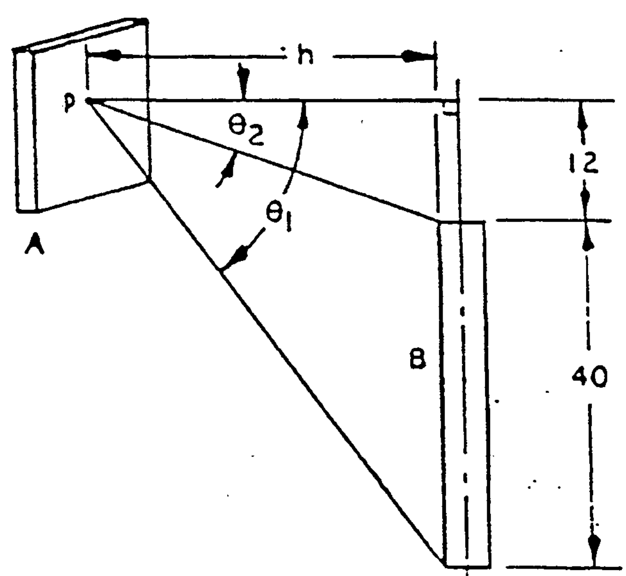
- $h = 27.5$ inches
- diameter of cylinder B = 5 inches
- $\theta_1 \neq \theta_2$



$\Omega = (d/h)(\sin\theta_1 + \sin\theta_2)$
$\sin\theta_1 = \sin(\tan^{-1} 25/27.5) = 0.673$
$\sin\theta_2 = \sin(\tan^{-1} 15/27.5) = 0.479$
$\Omega = (5/27.5)(0.673 + 0.479) = 0.209$ steradians

Calculate the solid angle subtended by cylinder B from the center of parallelepiped A given:

- h = 27.5 inches
- diam. of B = 5 inches

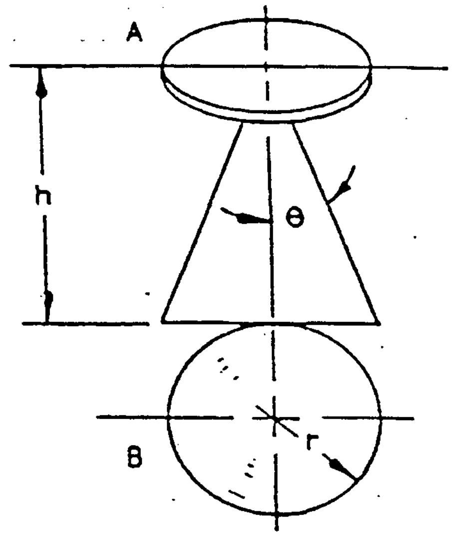


$\Omega = (d/h)(\sin\theta_1 - \sin\theta_2)$
$\sin\theta_1 = \sin(\tan^{-1} 52/27.5) = 0.884$
$\sin\theta_2 = \sin(\tan^{-1} 12/27.5) = 0.400$
$\Omega = (5/27.5)(0.884 - 0.400)$
$= 0.088$ steradians

Discs and Spheres:

Calculate the solid angle subtended by sphere B from the center of circular slab A given:

- r = 8 inches
- C-T-C separation = 50 inches

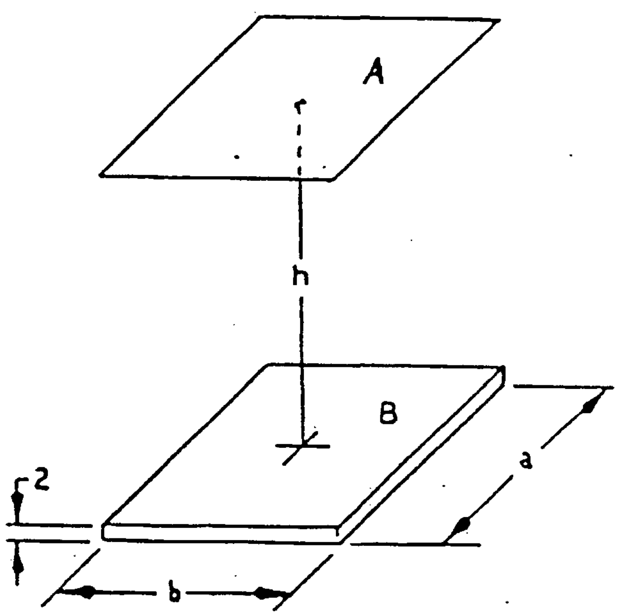


$\Omega = 2\pi(1-\cos\theta)$
H = 50-8 = 42 inches
$\cos\theta = \cos(\tan^{-1} 8/42) = 0.982$
$\Omega = 0.113$ steradians

Planes and Slabs:

Calculate the solid angle subtended by slab B from the center of slab A given:

- parallel slabs whose centers lie on a perpendicular line between the two,
- edge-to-edge separation = 31 inches
- a = 14 inches, b = 16 inches
- h = 31 inches



$$\Omega = 4\sin^{-1}\left[\frac{(a/2)(b/2)}{\sqrt{(a/2)^2+h^2}\sqrt{(b/2)^2+h^2}}\right]$$

$$\Omega = 4\sin^{-1}\left[\frac{(7)(8)}{\sqrt{(7)^2+31^2}\sqrt{8^2+31^2}}\right]$$

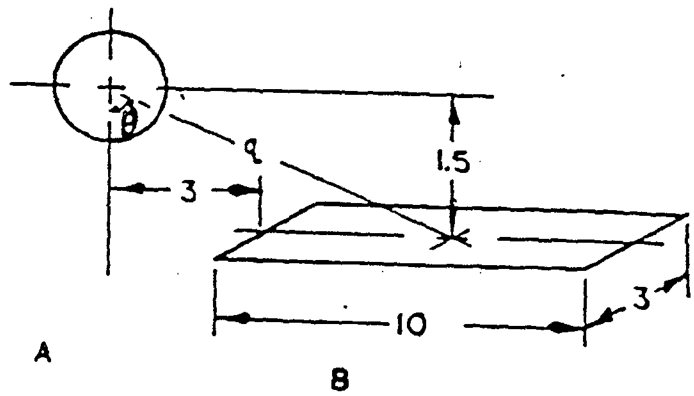
$\Omega = 4(0.0551$ radians)

$\Omega = 0.220$ steradians

Calculate the solid angle subtended by plane B from the center of sphere A with relative positions as shown:

- a = 10 inches

- b = 3 inches



$\Omega = (ab/q^2) \cos \theta$

$\cos \theta = \cos(\tan^{-1} 8/1.5) = 0.184$

$q^2 = 8^2 + 1.5^2 = 66.25$

$\Omega = (10)(3)(0.184)/(66.25) = 0.083$ steradians

14.6.2.2 Surface Density

The surface density technique for evaluating arrays is especially useful for planar distributions of fissile units, such as arrangements for storage on a floor or wall. The surface density method considers an array in terms of its fissile material density as projected onto a boundary plane. Comparison of the resulting value to the density of a water-reflected critical slab of the same composition is then the basis for establishing fissile material limits. The choice of plane is usually determined by whichever projection satisfies the "safety criterion" defined below. If just one of the planes does so, then safety is demonstrated.

In the Thomas formulation (TID-7016, Rev 2), the allowed surface density is expressed as:

$$\sigma = 0.54\ \sigma_0\ (1\text{-}1.37f),$$

where σ_0 is the surface density in grams per square centimeter of a critical water-reflected infinite slab, and 'f' is the fraction critical (i.e., the ratio of the mass of a unit in the array to the critical mass of an unreflected sphere of the same material). The fraction critical must not exceed 0.73 for this formulation to be applicable.

An unlimited planar array of units may be stored if the above conditions are not exceeded.

The surface density concept is most suitable for "squat" configurations (where the height of the array is much less than a base dimension). This method is limited and may not be used for metal-water systems.

This method is used in areas for storage of low-level materials in infinite planar storage arrays. In using this method, the main principle of limiting the surface concentration to less than an allowed value is used. Detailed calculations are not required because of the criteria. The allowed σ is determined, and all storage areas follow preset rules to assure that this requirement is not exceeded.

The safety limits may be applied to arrays reflected by concrete of any thickness by the following:

- The surface density, σ, is reduced by 40% for planar arrays,
- The average fissile material density is reduced by 50% for any shaped array, and
- The number of units is reduced by 75%.

14.6.2.3 Unit Storage Criteria

On the principle of limiting the lattice density in an array of subcritical units, the relationship of unit spacing and maximum allowable number of units in an array may be determined according to these requirements.

A maximum unit is defined in Table 14.1.

The maximum unit may consist of a group of small units in a single tightly closed container or distributed among several tightly closed containers.

Table 14.1
Maximum Sizes of Spherical Units to Which Unit Storage Criteria Limits Apply

Moderation	Fissile Content	
	^{235}U	^{233}U
$H/X \leq 2$	10.0 kg	2.8 kg
$2 < H/X \leq 3$	9.0 kg	2.5 kg
$3 < H/X \leq 5$	7.3 kg	2.2 kg
$5 < H/X \leq 10$	5.2 kg	1.8 kg
$10 < H/X \leq 20$	3.6 kg	1.3 kg
$H/X > 20$	3.6 liters	1.3 liters

Unit sizes may be based upon limited moderation when moderation is controlled according to Section 14.3.1.

The center-to-center spacing shall be at least 12 inches, but in no case may the surface-to-surface separation be less than 8 inches.

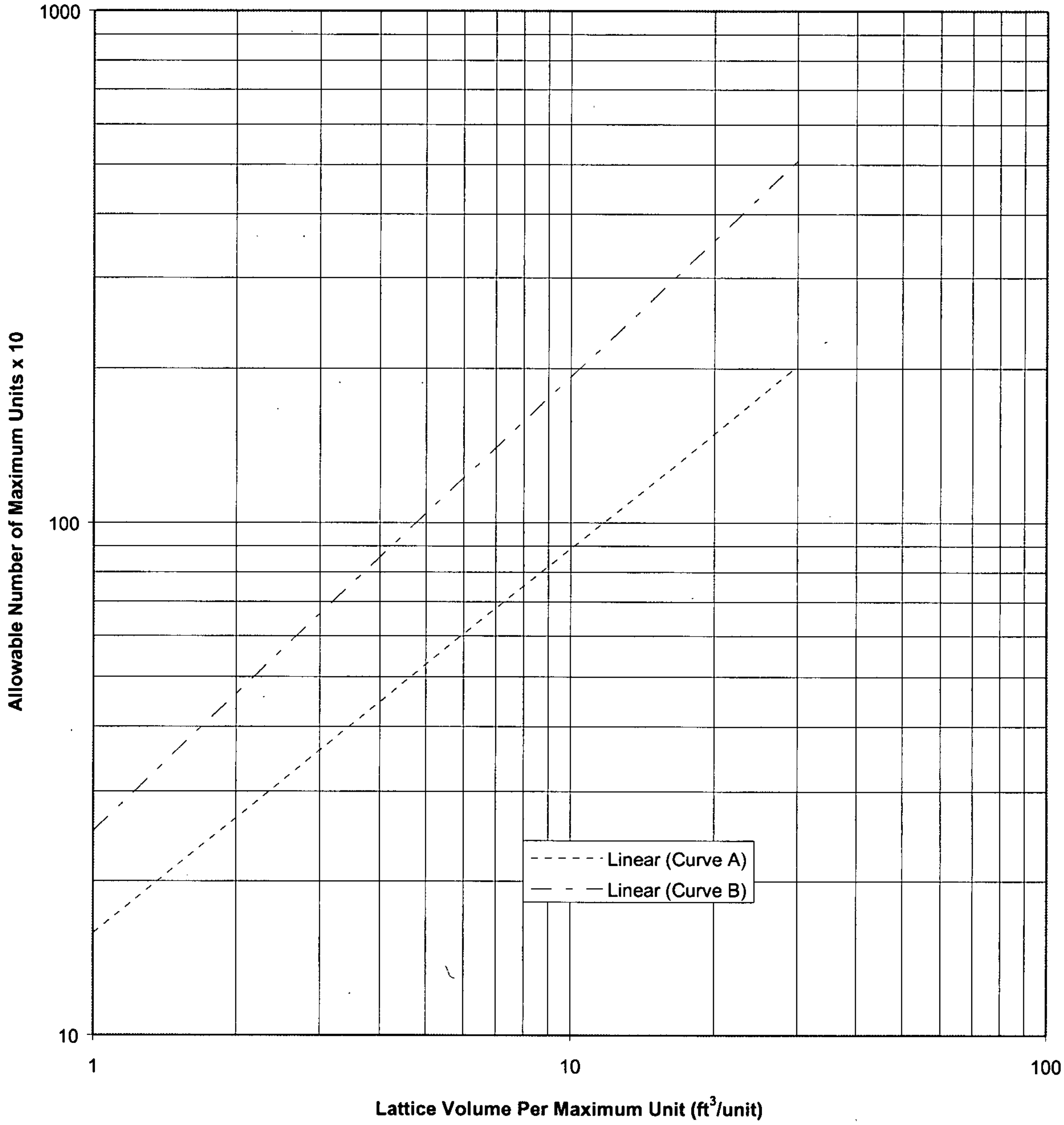
Figure 14.1 specifies the maximum allowable number of maximum size units as a function of the lattice volume per unit that may be authorized. Curve A applies where there is a thick, close-fitting reflector on more than two sides of the array, such as a thick-walled vault of concrete, metal, wood, or earth. The center-to-center (CTC) spacing in a cubic array is related to the lattice volume per unit as follows:

$$CTC(in) = [V(ft^3)]^{1/3} \times 12$$

or $$V(ft^3) = (CTC(in)/12)^3.$$

The following equations may also be used instead of reading values off the curves in Figure 14.1:

Figure 14.1
Maximum Allowable Number of Maximum Size Units from Table 14.1 in a Cubic Array



Curve A: Max # = $10^{(.749 \mathrm{x} \log V + 1.2)}$

or, $V(ft^3) = 10^{((\log(Max\#)-1.2)/.749)}$

Curve B: Max # = $10^{(.885 \mathrm{x} \log V + 1.4)}$

or, $V(ft^3) = 10^{((\log(Max\#)-1.4/.885)}$

An alternative source for determining the allowable unit size, spacing, and number of units is ANSI/ANS-8.7. This standard is a validated method for evaluating storage arrays.

14.6.3 Computer Codes

All computer codes used by outside organizations under contract to NFS are either validated by these organizations or representative calculations are rerun on a validated NFS system. Transport and Mote Carlo computational codes used at NFS are validated in accordance with ANSI/ANS-8.1, for use in evaluating systems containing uranium enriched in the ^{235}U isotope.

NUCLEAR CRITICALITY SAFETY

C. SPECIAL CONTROLS

14.7 Special Controls

This section contains additional information regarding special controls used to assure nuclear criticality safety (NCS) in operations. Various special controls are discussed in Section 4.2.4 of Chapter 4.

Diverse NCS control methods that prevent common mode failures are incorporated into all processes where the maximum credible fissile material concentration of solutions discharged to an unfavorable geometry vessel could exceed the safe level. All exceptions to this require approvals from the safety review committee and the president or designated alternate.

14.7.1 Dual Analysis

Independent laboratory analysis of split samples by two different laboratory analysts or by two different procedures is required of all solutions prior to release to vessels of unfavorable geometry and volume. Only if both analyses show the fissile material concentration to be less than the applicable release limit may the solution be released.

The laboratory ensures the reliability of its criticality control analyses as follows:

- ***Procedure Controls*** – All criticality control analysis procedures provide detailed instructions which specify scope (applicability), accuracy, precision, reagents, equipment, calibration technique, calibration frequency, standards, sample preparation, sample analysis, and calculation of results. Each procedure or revision is written by an experienced chemist to insure the method is technically correct for NFS samples. Managers who are knowledgeable in analytical methodology must then review and approve the procedure or revision before implementation. A copy of the approved procedures is retained in the Laboratory work area for initial training and for future reference.

  In addition, a separate procedure identifies all expected criticality control samples and specifies the methods of analyses. Each criticality control sample consists of two separate sample aliquots (A&B), which are analyzed by separate methods if possible. If there is not a technically acceptable second method available for the particular sample, the procedure specifies that two different individuals will analyze the two aliquots or one individual will analyze each aliquot using different dilution factors.

When only one method is used, the samples are analyzed by a gamma counting technique and a standard must be concurrently analyzed within established control limits before results are reported. This gamma analysis method is normally accurate for NFS samples, but an advantage to this system is that results will be high biased (fail-safe) if uranium solids should be present. In all cases, at least one of the aliquots of high-enriched uranium criticality control samples is analyzed by gamma counting (NFS-SA-3-8). Each shift, prior to any gamma analyses, two standards and a blank must be analyzed within established limits to verify the acceptability of the system. Numerous other standards are also normally analyzed throughout each shift, which continues to insure the accuracy of the system.

In conclusion, NFS utilizes two analytical methods and two analysts, when possible, to crosscheck each other. This approach reduces the need for a high number of standard analyses because at least one of the methods should detect any abnormal samples. When this approach is not possible, the single analysis system is well controlled with the use of accountability standards.

- ***Control of Analyst/Methods*** – All criticality control samples are analyzed in the Process Control Laboratory by Laboratory Control Technicians. Each Laboratory Control Technician initially receives four weeks of training, and must demonstrate an ability to perform the criticality control analyses by analyzing standards and/or samples. The Laboratory Control Technicians acknowledge an understanding and agree to comply with the procedures in the on-line computer qualification record system or documented sign-off records. A supervisor must verify that the analyst is qualified before the qualification is accepted by the system.

  The laboratory utilizes accountability standards to monitor and control the Control Technicians and the methods which analyze the criticality control samples. The same procedures are utilized for both accountability and criticality control samples; therefore, separate criticality control standards have not been established. The results of each analyst's standard analysis must be within established bias limits to maintain their qualification for the method. In addition, the analyst must analyze a sufficient number of standards each year to maintain qualifications. The analysis method is discontinued when standards are outside established limits and is not used until standards are analyzed within acceptable limits.

- ***Release of Results*** – After the completion of the analysis, the Control Technician calculates the results for each aliquot and enters them into the computer system. Before the results are released (available) in the system, a supervisor verifies the analyses and then releases them. The user of the result (production) can then access the reported values.

  The laboratory controls the analyses and reporting as identified above. The validation of results has detected a very small percentage of calculational errors.

  The two sample method (A&B aliquots) ensures that a single sample misidentification by a sampler or analyst will not jeopardize safety. This redundancy has been very effective and a high variance between aliquots is almost unknown. If either analysis is high, then discharge of the solution is prohibited. The laboratory normally verifies the accuracy of aliquots that show a higher than expected variance and resolves any analytical problem prior to reporting – these cases have been very rare.

  The laboratory has not utilized a high blind sample (standard) system to verify the integrity of the analysis systems; however, occasionally a criticality control samples will be "high." The laboratory has no advance notice that a sample is higher than "normal" and any high results are routinely reported.

  There is not a formal program to blind verify criticality control analyses, but our waste stream accountability analyses provide a degree of independent verification. Solutions in safe geometry tanks that are analyzed for criticality control are then discharged to a measurement tank and analyzed for accountability. This practice has served as an informal check on the criticality control analytical systems. The two analyses are not routinely reviewed for agreement, but "high" accountability discharge solution results have been identified in these instances. All investigated higher than normal discards have been attributed to a process operation, which indicates that all high criticality control samples have been correctly analyzed and re-processed. Since most tank discharges are composited prior to accountability sampling, a one-to-one verification is not possible using these accountability results.

14.7.2 Planar Arrays of Birdcages

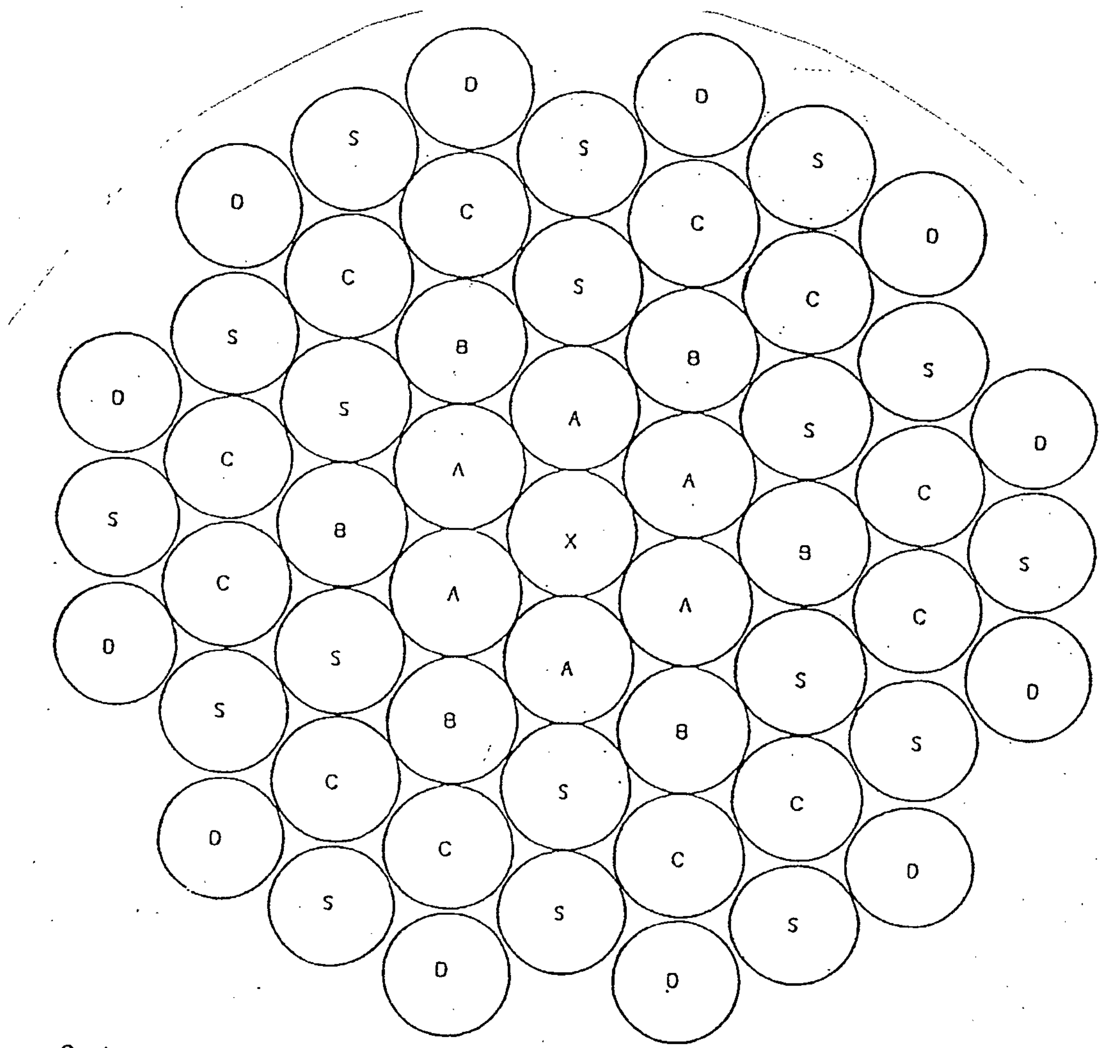
Containers authorized to contain more than a safe wet mass of fissile material are typically stored in storage racks, approved shipping packages, or storage birdcages. A storage birdcage is a favorable diameter metal sleeve with metal spacer arms located near the top and bottom of the sleeve so as to center the sleeve in an open top 55-gallon drum. Open top drums each have 3 drain holes within 1-inch of the bottom to avoid the accumulation of leaking or spilled liquids. Drums with lids that contain only solid fissile materials are not required to have drain holes at the bottom of the drum provided the drums are not left unattended in the processing areas where uranium-bearing liquids are present. Many shipping packages have birdcage-type sleeves and may have dimensions that permit storage in arrays established for in-plant storage birdcages.

Safe planar arrays of birdcages are primarily stored in storage bins. The bins have walls of approximately 12-inch thick solid concrete blocks stacked high enough to sufficiently reduce the interaction between adjacent planar arrays. Three-foot-wide openings into the bins are oriented to minimize interaction between arrays. The size of the safe planar array may be controlled by procedural limits, but the preferred method is to physically limit the size of the array by use of the block walls. Operating procedures and postings specify the size and/or number of containers allowed per safe array, as determined by the type and quantity and/or geometry of the fissile material in the containers.

The array limits are based primarily on interaction calculations by the KENO code, on comparisons to the Unit Storage Criteria, or on NRC and DOT shipping regulations. Close-packed planar arrays of birdcages are sometimes limited to a maximum width of 12 feet by the bin walls and at other times are limited by administrative controls. Figures 14.2 and 14.3 are examples of the birdcage arrangements in the two types of arrays. The close-packed arrangement is used for interaction calculations and when determining how many drums will fit in an array; however, typical storage arrangements provide enough open space to permit access to drums near the rear of the bin without having to remove a large number of drums.

Drums of material that have been shown by the techniques authorized in Chapter 4, or by the shipping regulations, to be safe in an infinite array may be conservatively substituted for one of the storage birdcages. Examples are containers limited to ^{235}U masses that result in a safe areal density for the array or shipping packages designated by the regulations as being Fissile Excepted, Fissile, or LSA.

Figure 14.2
Close-Packed Planar Array of Birdcages



Center-to-Center
Spacing to:

A 24 in.
B 41.6 in.
C 63.5 in.
D 86.5 in.
E 104.6 in.
S Shielded
X Central Unit

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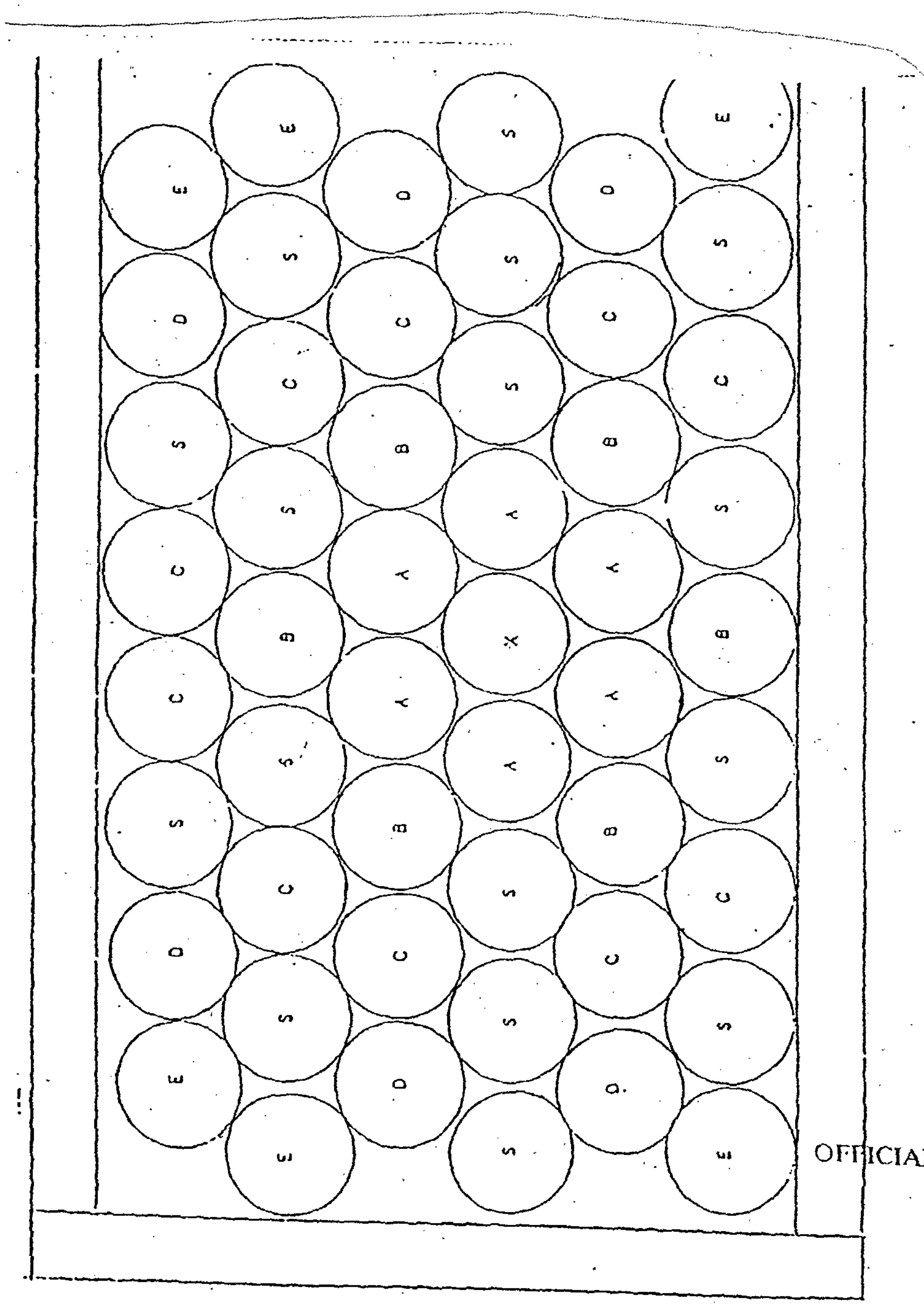
Figure 14.3
Close-Packed Planar Array of Birdcages in a 12 Feet Wide Storage Bin



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14.7.3 Shipping Regulations

In addition to considerations for the radioactive content, shippers of fissile radioactive material must also take into account certain other packaging and shipment requirements to ensure against nuclear criticality due to the fissile nature of the materials. The design of the packaging for fissile radioactive material, the criticality safety index (CSI) to be assigned, and any special procedures for packaging are prescribed in 49 CFR 173.451 through 173.459 of the DOT regulations and in 10 CFR 71 of the USNRC regulations. The packaging must be such to ensure against nuclear criticality under both normal and hypothetical accident conditions, and prevent loss of contents in transportation.

The total CSI of all packages in any single transport vehicle or storage location generally may not exceed 50. Exceeding the 50 CSI per vehicle limit is authorized only for certain specific types of shipments which are carried under the special requirements of "exclusive use" vehicles, which impose additional responsibilities on the shipper. The CSI system is designed to provide the means to assure NCS. It limits the amount of fissile materials in one location. For such fissile materials, the shipper must determine, in accordance with regulatory criteria, the appropriate CSI based on NCS. For purposes of transportation, the shipper then must assign to the package the CSI value.

Incoming or outgoing shipments of materials in approved shipping packages may be handled in arrays based on NRC and/or DOT regulations. The regulations may require that the array be limited to a maximum number of packages, in which case storage bins may be used to control the size of the arrays. The receiving areas are administratively controlled to safe planar arrays meeting the shipping regulations, or meeting the in-plant drum or birdcage storage limits if the packages comply with the drum or birdcage requirements.

Some packages are low enough in fissile material content that the regulations exempt them from any NCS-related safety controls (49 CFR 173.453). These "Fissile Excepted" packages can be stored in unlimited numbers in arrays. Low specific activity (LSA) materials, and surface contaminated objects (SCO), may also be stored in unlimited numbers in arrays. Low specific activity materials are those materials which present a relatively low hazard as a result of their limited radioactive concentration. Some of these materials are listed in the regulations, such as uranium ores and concentrates, as well as unirradiated natural or depleted uranium. Other materials must meet certain limitations related to their radioactive concentration. Objects of nonradioactive material externally contaminated with radioactive material, if the radioactive material is not readily dispersible and the surface contamination meets certain regulatory limits, may be considered as SCO or in some cases as LSA material.

14.8 Data Sources

NCS reference materials contain the results of many experiments performed expressly to determine appropriate NCS limits for specific processes and operations. These experimental values are used to benchmark nuclear analysis codes and to establish critically safe parameters for individual units and arrays of units. The most commonly used and readily available documents from which experimental parameters may be obtained are:

1. ARH-600, "Criticality Handbook," by R. D. Carter, et al.

2. DP-1014, "Critical and Safe Masses and Dimensions of Lattices of U and UO_2 Rods in Water"

3. AHSB Handbook Rev. 1, "Handbook of Criticality Data," United Kingdom Atomic Energy Authority, 1965, by J. H. Chalmers, et al.

4. TID 7028, "Critical Dimensions of Systems Containing ^{235}U, ^{239}Pu, and ^{233}U"

5. LA-10860-MS, 1986 Revision, "Critical Dimensions of Systems Containing ^{235}U, ^{239}Pu, and ^{233}U," by H. C. Paxton and N. L. Pruvost

6. TID 7016, Rev. 2, "Nuclear Safety Guide"

7. LA-3366 (Rev.), "Criticality Control in Operations With Fissile Material"

8. K-1019, Rev. 5, "Criticality Data and Nuclear Safety Guide Applicable to the Oak Ridge Gaseous Diffusion Plant"

9. LA-12808, September 1996, "Nuclear Criticality Safety Guide"

Other documents, from which data or techniques used for NCS analyses are taken, are referenced where used.