NUCLEAR FUEL SERVICES, INC.
a BWX Technologies, Inc. company

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21G-16-0191
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ACF-16-0304
November 21, 2016

Director
Office of Nuclear Material Safety and Safeguards
U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Subject: Submittal of License Condition S-2 Authorized Changes

Reference: Docket No. 70-143: SNM License 124

Pursuant to License Condition (LC) S-2, Nuclear Fuel Services, Inc. (NFS) hereby submits changes to the following chapters of License SNM-124:

Chapter 1 - General Information
Chapter 2 - Organization and Administration
Chapter 4 - Radiation Protection
Chapter 5 - Nuclear Criticality Safety
Chapter 9 - Environmental Protection
Chapter 11 - Management Measures

The revised chapters are attached in their entirety, in addition to a revised Chapter Index. Changes are denoted by vertical lines in the right margin of affected pages. In accordance with the criteria of LC S-2, NFS has determined that the changes reflected in the above chapters do not reduce safety or the level of effectiveness of the License Application and do not require prior NRC approval.

If you or your staff have any questions, require additional information, or wish to discuss this matter further, please contact me at (423) 743-1705, or Mr. Randy Shackelford, Nuclear Safety and Licensing Manager, at (423) 743-2504. Please reference our unique document identification number (21G-16-0191) in any correspondence concerning this letter.

Sincerely,

NUCLEAR FUEL SERVICES, INC.

Richard J. Freudenberger

Richard J. Freudenberger, Director
Safety and Safeguards

NMSS20

Attachment 1: NFS License Condition S-2 Authorized Changes Description
Attachment 2: SNM-124 Chapter Index
Attachment 3: SNM-124, Chapter 1
Attachment 4: SNM-124, Chapter 2
Attachment 5: SNM-124, Chapter 4
Attachment 6: SNM-124, Chapter 5
Attachment 7: SNM-124, Chapter 9
Attachment 8: SNM-124, Chapter 11

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Attachment 1

NFS License Condition S-2 Authorized Changes Description

(4 pages to follow)

NFS License Condition S-2 Authorized Changes Description

The majority of the changes described below are administrative in nature, as determined by reviewing the examples provided in NRC Regulatory Guide 3.74, "Guidance for Fuel Cycle Facility Change Processes." For such changes, a detailed evaluation to demonstrate that the criteria in License Condition (LC) S-2 are met is unnecessary. However, for the changes that are more than just administrative in nature, an evaluation against the LC S-2 criteria is provided.

Chapter 1

The administrative changes to Chapter 1 are as follows:

- Page 1-4, Relocated the respirator fit-test facility from Building 104 to Building 350 in order to conduct these tests in the Medical Facility
- Page 1-17, Corrected the distance between the NFS site boundary and the Nolichucky River stated in the "Demographics" section so that it does not conflict with the more accurate distances stated on page 1-18 in the "Hydrology" section

Please see Attachment 3 for a copy of Chapter 1 with the changes noted.

Chapter 2

The administrative changes to Chapter 2 are as follows:

- Page 2-4, Revised the listing of Safety "functional areas" in Section 2.3.5 to remove two programs that are not stand-alone functional areas and that are included in the Safety functions described in Subsections 2.3.5.1 through 2.3.5.5
- Page 2-4, Clarified that the emergency preparedness program is not a stand-alone functional area and that it is supported by the other Safety functions as needed
- Pages 2-6 and 2-7, Deleted the words "or equivalent experience" from four places for consistency between the Safety function subsections and because the definition of equivalent experience in Chapter 1 applies throughout the license
- Page 2-7, Removed responsibility for the emergency response team from the Industrial Safety function because it is now under a Safety Discipline manager and, like the emergency preparedness program, is supported by other functional areas as needed
- Page 2-8, Clarified the wording on experience requirements for the manager of the ISA function with regard to fire protection
- Page 2-12, Updated the Functional Organization Chart to reflect that the Analytical Services function, and its association with laboratory quality activities, are no longer part of the Safety & Safeguards organization; the Analytical Services function is in the same organization as the Quality Assurance (QA) function but it does not report to the QA function manager

Please see Attachment 4 for a copy of Chapter 2 with the changes noted.

Chapter 4

The administrative changes to Chapter 4 are as follows:

- Page 4-4, Clarified that the safety review committee does not review "all" proposed facility modifications but rather only the "major" modifications; this is consistent with the Change Control requirements in Chapter 11 since the safety review committee also functions as the Change Control Board
- Page 4-35, Deleted references to the plant security fence and changed the terminology to match that used in Chapter 1 regarding the posting and labeling exemption
- Page 4-35, Clarified that access to the Restricted Area is controlled, and removed redundant wording

Regarding the change to Page 4-12, Section 4.6.2.2, please see the following evaluation per LC S-2 criteria:

Does the change decrease the level of effectiveness of the design basis as described in the License Application?
No. The design basis for the Laboratory Area Containment Enclosures and the associated ventilation system is not being changed. The level of effectiveness for such equipment is determined by how well worker exposures are controlled. Though the total air flow for each enclosure will not be determined, the proper average face velocity will still be established for each enclosure such that worker exposures will continue to be controlled well below any applicable limits. Routine exposures in the Laboratory areas are currently less than 1% of the applicable DAC values, and this change will not affect the exposures.

Does the change result in a departure from the methods of evaluation described in the License Application used in establishing the design basis?
No. The method of evaluation used in establishing the design basis will continue to be the ANSI/AIHA 29.5-2003 recommendations. The standard states in Section 2.1.1 that "The containment and capture of a laboratory hood shall be considered adequate if, in combination with prudent practice, laboratory worker chemical exposures are maintained below acceptable in-house exposure limits." While NFS is not referring to chemical exposures for this issue, the same statement applies to radioactive materials. Though the total air flow for each enclosure will not be determined, the proper average face velocity will still be established for each enclosure such that worker exposures will continue to be controlled well below any applicable limits.

Does the change result in a degradation of safety?
No. This change will result in having the proper average face velocity being determined for the enclosure, and the worker exposures will be maintained below applicable limits.

Does the change affect compliance with applicable regulatory requirements?
No. NUREG-1520, Revision 2, Section 4.4.6.1 identifies 10CFR20 Subpart H and 10CFR70.22(a)(7) as the applicable regulatory requirements; and, Section 4.4.6.3 of the NUREG lists the acceptance criteria for meeting these requirements. This change will not modify the operating criteria for the ventilation and containment systems, nor will it change the means to measure the performance of these systems.

Does the change conflict with an existing license condition?
No. There are no Conditions specific to the ventilation and containment systems.

Please see Attachment 5 for a copy of Chapter 4 with the changes noted.

Chapter 5

The administrative change to Chapter 5 is as follows:

- Page 5-7, Deleted the "storm-watch mode" due to modifications to the Criticality Accident Alarm System (CAAS) that now allow the CAAS to remain operational during storm conditions without implementing additional compensatory measures such as manual monitoring of the alarm panel

Please see Attachment 6 for a copy of Chapter 5 with the change noted.

Chapter 9

Regarding the changes to Pages 9-10 and 9-13, and the respective Sections 9.2 and 9.2.6, the changes represent a more conservative approach to monitoring the radioactivity in groundwater. Rather than performing gross alpha and gross beta analyses, then comparing the results to specified action levels, and then performing isotopic analyses **if** the action levels are exceeded, NFS is having isotopic uranium and plutonium analyses performed on the quarterly samples regardless of their initial activity. Also, isotopic thorium analyses are not needed because thorium has not been detected in groundwater samples above any action levels, even where there is known thorium contamination in the soil. This is due to the high distribution coefficient of thorium. NFS also performs analysis for Technetium-99 in the quarterly samples regardless of the initial activity.

These changes do not decrease the level of effectiveness of the environmental monitoring program for groundwater radioactivity nor do they degrade safety on or off the plant site. Compliance with applicable regulatory requirements and license conditions is still maintained.

Please see Attachment 7 for a copy of Chapter 9 with the changes noted.

Chapter 11

The administrative changes to Chapter 11 are as follows:

- Page 11-5, Deleted the statement that the design requirements will ultimately reside in a single electronic database because NFS has to maintain classified information on a network separate from the network containing the unclassified information
- Page 11-6, Deleted the reference to "non-CM" changes because the Change Control Process described in Chapter 11 is applied in a graded manner to changes which are categorized as administrative, minor, or major changes and does not have to be applied to changes outside the CM program
- Page 11-6, Specified that administrative changes are approved by the Engineering discipline manager/designee rather than the less-defined, previously-stated approval by the "CM function"; this is a more structured approach and is consistent with the approval of minor changes
- Page 11-7, Changed the word "prepared" to "implemented" to clarify the intent of ensuring that a Change Control Package (CCP) is implemented based on the approved change request (CR); preparation of the CCP can occur at any point in time, but it cannot be implemented until the associated CR is approved
- Pages 11-16 and 11-20, Replaced references to "vice president" with "director" in three locations in order to be consistent with the management positions described in Chapter 2
- Page 11-20, Clarified that the assessments apply to the Safety Discipline functional areas and the Configuration Management function described in Chapter 2; the previous wording referenced "programs" which was more vague and subject to misinterpretation
- Page 11-20, Included "emergency preparedness" in the list of assessments that can be conducted at the discretion of management because the Emergency Preparedness Program is audited by the Quality Assurance discipline on an annual basis (as stated in Section 11.5.1)

Please see Attachment 8 for a copy of Chapter 11 with the changes noted.

Attachment 2

SNM-124 Chapter Index

(1 page to follow)

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124

CHAPTER INDEX

Attachment 3

SNM-124, Chapter 1

(25 pages to follow)

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 1

GENERAL INFORMATION
Table of Contents

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 1

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 1

GENERAL INFORMATION

1.1 Facility and Process Information

1.1.1 General Facility Description

The Nuclear Fuel Services, Inc. (NFS) site is located at 1205 Banner Hill Road, within the limits of the City of Erwin. The Protected Area of approximately 18 acres is located within approximately 70 acres of NFS-owned land, the remainder of which is either devoted to vehicle parking areas, is undeveloped, or is undergoing decommissioning. Additional information describing the NFS facility, including its location with respect to geographic features, roadways, population centers, industrial facilities, and public facilities, is provided in Section 1.3, "Site Description."

1.1.2 Facility Buildings and Structures

The facilities within the NFS site consist of numerous buildings, the majority of which are located within the Protected Area. The buildings and structures include the major SNM-processing production facilities, SNM-handling support facilities (storage, waste treatment, etc.), and a large number of non-SNM-handling support facilities (materials warehouses, maintenance shops, office buildings, etc.).

Buildings within the plant have been designated with numbers and names as shown in Figure 1-1. The major site features and descriptions of their current primary function(s) are provided below for informational purposes and are not intended to be restrictive of future potential activities in those facilities.

High Enriched Uranium (HEU) Fuel Production Facilities
(Bldgs. 302, 303, 304, 306, & 307)

Unit operations which produce a classified product containing high enriched uranium, as well as uranium recovery operations. Receipt, handling, and shipment of feed and product materials.

Blended Low Enriched Uranium (BLEU) Production Facilities

1. **Uranyl Nitrate Building (UNB) (Bldg. 510)**
   Receipt, handling, and storage of liquid uranyl nitrate.

Figure 1-1: Plant Layout and Property Boundaries

This drawing is "Official Use Only" and has been moved to the "Sensitive Information" ADDENDUM.

2. **Commercial Development Line (CDL) Facility (Bldg. 301)**
   Conversion of HEU materials to uranium oxides or to uranyl nitrate solution for subsequent purification and downblending in the adjacent BLEU Preparation Facility.

3. **BLEU Preparation Facility (BPF) (Bldg. 333)**
   Conversion of HEU materials to pure HE uranyl nitrate solution, preparation of blendstock (N uranyl nitrate solution), subsequent mixture of the HE uranyl nitrate and blendstock solution to form a LE uranyl nitrate solution (product), and uranium recovery operations.

4. **Oxide Conversion Building (OCB) (Bldg. 520)**
   Conversion of low enriched uranyl nitrate liquids into uranium oxides. Loading of powder for shipment.

5. **Effluent Processing Building (EPB) (Bldg. 530)**
   Treatment of process waste streams generated at the OCB (Bldg. 520) prior to discharge and/or disposal.

6. **LEU Dilution and Loading Facility (Bldg. 440)**
   Dilution of LEU produced by the BLEU Preparation Facility (Bldg. 333) to customer specifications. Loading of diluted LEU for shipment.

Laboratories

1. Building 220 – analytical laboratory.
2. Building 100 – NDA laboratory.
3. Research and Development (R&D) Laboratories (Buildings 105, 110, & 131)
   Facilities for conducting engineering studies and R&D of chemical and radioactive material processing, manufacturing, and treatment technologies in support of ongoing production efforts or new business development.
4. Central Analytical Laboratory (Building 105, Building 110, and the northwest portion of Building 303)
   Receipt and handling of 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.

Waste Water Treatment Facility (WWTF) (Buildings 330 and 335)

Treatment and discharge of liquid effluents generated by the process facilities, R&D laboratories, laundry, decommissioning activities, and analytical laboratory.

Other Support Facilities

Warehousing

Warehouse and material storage facilities include the Industrial Park Facility (IPF) Warehouse, Buildings 250, 300, 310, 311, southeast portion of 304, south and east sections of 306, 135, 136, 133, 132, and the UNB (Bldg. 510). Non-nuclear supply storage; nuclear materials storage in sealed containers while awaiting processing, treatment, or shipment off-site; rail siding and intermodal container transfer area.

Maintenance

The maintenance facilities reside in Buildings 110B, 120, 121, 300B, and the east section of 306. The plant's primary maintenance facility is located in Buildings 120 and 121.

Respirator Facility (Building 104)

Respirator laundry; and an inspection, testing, and quality assurance area.

Medical Facility (Building 350)

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 a respirator fit-test facility.

Building 111

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.

Administration Buildings

Buildings 105, 130 (east annex), 120 (north end), 305, 320, and 345 house offices and computer facilities.

Plant Utilities (Bldg. 130)

Non-radioactive plant utility services (compressed air, deionized water, and steam). This building contained uranium processes in the past, and covered fixed radioactive contamination exists.

Emergency Electrical Power

Emergency electrical power is provided for the Criticality Accident Alarm System and other surveillance systems from uninterruptible power supply (UPS) systems. Automatic transfer switches detect loss of off-site power, send a start signal to diesel engine generators, transfer the load to the generators when an appropriate output voltage has been reached, and transfer back to utility power after off-site power has been restored for a predetermined time. The automatic transfer switches then allow the generators to operate for a predetermined cool-down period prior to shutdown. This automatic switchover with UPS provides for continuous criticality detection and other surveillance functions during the absence of off-site power. Emergency power generators, transfer switches, and UPS systems are periodically functionally tested.

1.1.3 General Process Description

There are two primary operations at the NFS site involving licensed material: 1) the manufacture of a classified product containing high enriched uranium and 2) the downblending of surplus DOE high enriched uranium (HEU) to low enriched uranium (LEU).

High Enriched Uranium Fuel Production Facilities

Uranium is received in various forms and then processed to make a classified product. The product is tested to verify that it meets the customer specifications and then grouped into lots. The lots are packaged and then shipped to a fabricator for manufacture into reactor fuel components. Product that does not meet customer specifications is returned to the uranium recovery area of the facility for further processing.

Blended Low Enriched Uranium (BLEU) Production Facilities

Uranyl nitrate solution is produced at the BPF by downblending HEU to LEU. The HEU consists of, but is not limited to, feed materials such as uranium oxide, uranium-metal buttons, uranium-aluminum ingots, reactor elements, and UF_6. Incoming uranium feed materials to CDL or BPF may be converted into uranium oxide or processed as received for subsequent dissolution into uranyl nitrate solution. The HEU solutions are processed in CDL or BPF and downblended

with natural uranium in the BPF. The LE uranyl nitrate solution is transferred to the UNB (Bldg. 510), to the LEU Dilution and Loading Facility (Bldg. 440), or loaded directly into a shipping package at the BPF after verification that the solution meets the product specifications. Uranyl nitrate solution transferred to Bldg. 440 is diluted to meet customer specifications, loaded into shipping containers, and shipped to a fabricator for further manufacturing. Product that does not meet customer specifications is returned to the uranium recovery area of the facility for further processing.

Uranyl nitrate solution is received at the UNB from an off-site supplier via shipping containers or via pipeline from the BPF. The solution is transferred to the OCB for conversion into uranium oxide powder. The uranium oxide powder is loaded into shipping containers and shipped to a fabricator for manufacture into commercial reactor fuel bundles for ultimate transport to utility customers.

1.1.4 Raw Materials, Products, By-Products and Wastes

Various forms of uranium are used as feed materials for the classified process in the HEU Fuel Production facilities. The feed materials for the BLEU Production facilities include uranium oxide, uranium-metal buttons, uranium-aluminum ingots, reactor elements, and UF_6. The production, support, and waste processing activities are supported by a number of non-radiological chemical materials, such as bulk quantities of ammonium hydroxide, hydrogen, nitric acid, sodium hydroxide, sodium hydrosulfide, and sulfuric acid. A significant number of chemicals are used on site in lesser quantities.

Finished products containing licensed material include a classified product, uranyl nitrate solution, and uranium oxide powder.

There are no by-products produced or recovered at the NFS site that are sold for commercial use.

Liquid process wastes are collected in tanks in or near the various process buildings. Prior to pumping these wastes to the Waste Water Treatment Facility (WWTF), they are analyzed and must show levels below internal action guide limits. Waste water is treated in the WWTF on a batch basis, and the average discharge is approximately 15,000 gallons. Treatment typically involves adjustment of pH, and precipitation and removal of fluoride ions and uranium. The precipitate is de-watered, and the solids are packaged for land burial. The solutions may undergo ammonia removal by use of a stripping tower or by break-point chlorination prior to neutralization for discharge. The treated water is discharged directly to the Nolichucky River. A sample from each batch is collected and analyzed prior to discharge to assure compliance with 10 CFR 20 and applicable State of Tennessee regulations.

Plant sanitary wastes are discharged through piping which goes to the City of Erwin publicly owned treatment works (POTW). The inputs for the sanitary sewer system from the NFS site include bathrooms, showers, and water from the Groundwater Treatment Facility, where groundwater is collected and treated as part of ongoing site decommissioning and remediation activities. Non-contact cooling water, treated process waste water, and sanitary sewage from the BLEU Complex (Bldgs. 510, 520, 530) facilities are also discharged to the POTW.

The NFS site produces a variety of regulated solid wastes (obsolete equipment, used ventilation filters and personal protective equipment, waste treatment residues/filter cakes, demolition debris, miscellaneous combustible wastes, etc.). Solid waste materials could be radiologically contaminated, non-contaminated, hazardous, or mixed (hazardous and radioactive). These wastes are typically containerized for shipment to a licensed disposal facility.

The site facilities discharge airborne effluents to the atmosphere via a number of process stacks. HEPA filtration and scrubber systems (i.e., venturi, demisting, packed-bed) are used as needed to remove radioactive particulates and chemicals from airborne effluents to assure compliance with 10 CFR 20 and applicable State of Tennessee regulations prior to discharge to the atmosphere.

1.2 Institutional Information

1.2.1 Corporate Identity

The full name and address of the applicant and the facility are as follows:

Nuclear Fuel Services, Inc.
1205 Banner Hill Road
Erwin, Tennessee 37650-9718

The U.S. Nuclear Regulatory Commission (NRC) license number for this facility is SNM-124 (Docket Number 70-143).

The Nuclear Fuel Services, Inc., (NFS), facilities are located within the City of Erwin, in Unicoi County, Tennessee. At this site, NFS maintains buildings for administrative, laboratory, manufacturing, and support activities. The activities described in Section 1.2.4 are performed at 1205 Banner Hill Road, 1080 S. Industrial Drive, and 200 Oxide Lane. These locations are in Erwin, Tennessee.

The applicant, Nuclear Fuel Services, Inc., is incorporated in the State of Delaware, with its Corporate Offices located at 1205 Banner Hill Road, Erwin,

Tennessee 37650-9718. NFS is a subsidiary of NFS Holdings, Inc., which is a subsidiary of NOG-Erwin Holdings, Inc., which is a wholly-owned subsidiary of BWXT Nuclear Operations Group, Inc., incorporated in Delaware. A summary listing of NFS affiliates is provided in Appendix 1A, along with a figure (Figure 1A-1) showing the reporting relationships.

1.2.2 Financial Qualifications

As a result of the indirect transfer of control in 2008 of Nuclear Fuel Services, Inc., from NFS Services, LLC, to NOG-Erwin Holdings, Inc., NFS was required to provide details to the NRC 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 10.

1.2.3 Type, Quantity, and Form of Licensed Material

1.2.3.1 Uranium Enriched in the ^{235}U Isotope

Maximum quantity on site – **This information is "Official Use Only" and has been moved to the "Sensitive Information" ADDENDUM.**

Isotopic content – any, up to maximum enrichment and up to an average of 10^{-6} grams of plutonium per gram of uranium, 0.25 millicuries of fission products per gram of uranium, and 1.5×10^{-5} grams of transuranic materials (including plutonium) per gram of uranium, as contaminants;

Chemical and physical forms – as described in Appendix 1B.

1.2.3.2 Uranium Enriched in the ^{233}U Isotope

1. Maximum quantity on site – **This information is "Official Use Only" and has been moved to the "Sensitive Information" ADDENDUM.**
   Isotopic content – any, up to maximum enrichment;
   Chemical and physical forms – any form, but limited to residual contamination from past operational activities.

2. Maximum quantity on site – **This information is "Official Use Only" and has been moved to the "Sensitive Information" ADDENDUM.**
   Isotopic content – any, up to maximum enrichment;
   Chemical and physical forms – any form, as received for analysis and/or for input into development studies.

1.2.3.3 Plutonium

1. Counting and Calibration Standards
   Maximum quantity on site – 10 millicuries as counting and calibration standards;

2. Residual Contamination and Mixed Oxide Process Holdup
   a. Buildings 110 & 234
      The possession limits, including quantity, isotopic content and chemical and physical forms, for plutonium residual contamination and mixed oxide holdups for Buildings 110 and 234 were previously described in letters dated October 17, 1988; and January 21, 1994. **This information is "Official Use Only" and has been moved to the "Sensitive Information" ADDENDUM.**

   b. Site-Wide Decommissioning
      NFS is authorized to possess residual plutonium contamination, as-is from former plutonium operations, in in-situ soil and debris, as well as waste and waste holdups that is generated during NFS plant site decommissioning activities, including Building 234.

3. Materials Input to R&D Studies
   Maximum quantity on site – **This information is "Official Use Only" and has been moved to the "Sensitive Information" ADDENDUM.**
   Chemical and physical forms – any form, received for analysis and/or for input into development studies.

4. Materials Received for Decontamination and Volume Reduction
   Maximum quantity on site – **This information is "Official Use Only" and has been moved to the "Sensitive Information" ADDENDUM.**
   Chemical and physical forms – any form, as contamination on equipment and materials received for decontamination and volume reduction.

1.2.3.4 Transuranic Isotopes

Maximum quantity on site – **This information is "Official Use Only" and has been moved to the "Sensitive Information" ADDENDUM.**
Chemical and physical forms – as waste resulting from processing enriched uranium.

1.2.3.5 Fission Products

Maximum quantity on site – **This information is "Official Use Only" and has been moved to the "Sensitive Information" ADDENDUM.**
Chemical and physical forms – as waste resulting from processing enriched uranium.

1.2.4 Authorized Uses

This application authorizes the use of special nuclear material (SNM) for operations involving enriched uranium pursuant to 10 CFR Part 70 as listed below. Typical support activities related to the production of these products include, but are not limited to, the receipt and storage of raw materials; the storage of finished products; the preparation and transport of these products off-site; SNM recycling/recovery operations; the processing/disposal of SNM-bearing waste materials, excluding on-site burial; process and product development activities; laboratory operations; and maintenance/repair of contaminated equipment and facilities.

1.2.4.1 Product Processing Operations

1. UF_6 Conversion
   Conversion of high enriched uranium hexafluoride to other uranium compounds.

2. Fuel Manufacturing
   Production of fuel containing high enriched uranium.

3. Uranium Recovery
   Recovery and purification of LEU and HEU from process scrap materials, either internally generated or generated at other facilities.

4. Enrichment Blending and Conversion
   Enrichment blending of high enriched liquid UNH to produce a low enriched UNH solution, and conversion of downblended UNH solution to uranium oxide (U_xO_x).

1.2.4.2 Laboratory Operations

Laboratories are equipped to perform wet chemical and instrumental analyses and a wide variety of physical tests on material consisting of and/or containing special nuclear materials.

1.2.4.3 General Services Operations

1. Storage of special nuclear material compounds and mixtures in areas with containers arranged specifically for maintenance of radiological and nuclear safety.
2. Maintenance and repair of special nuclear materials processing equipment and auxiliary systems.
3. Decontamination of equipment and materials, including personnel protective clothing and respiratory devices.

1.2.4.4 Research and Development Operations

Research and development work is performed on natural, source, and special nuclear material compounds and mixtures in areas with containers arranged specifically for maintenance of radiological and nuclear safety.

1.2.4.5 Waste Treatment and Disposal

1. Decontamination of materials and equipment.
2. Volume reduction, treatment, packaging and storage of both liquid and solid wastes contaminated with or containing non-recoverable uranium and plutonium.
3. Shipment of radioactive waste to licensed facilities or to licensed burial sites for disposal.
4. Treatment, packaging, and storage of hazardous or mixed waste for off-site disposal.

1.2.4.6 Period of License

The period of License No. SNM-124 is twenty-five (25) years with an expiration date of August 31, 2037.

1.2.5 Special Exemptions and Special Authorizations

1.2.5.1 Criticality Monitoring

Special Nuclear Material stored in authorized shipping containers complying with the requirements of the Code of Federal Regulations, Title 10, Part 71, and which are in isolated arrays or on a transport vehicle and which are no more reactive than that approved for transport are exempt from criticality monitoring requirements of 10 CFR 70.24.

1.2.5.2 Posting and Labeling

Pursuant to the requirements of 10 CFR 20.1904(a), each entrance into a Restricted Area will be posted "Caution, Radioactive Materials, Every container or vessel within this area may contain Radioactive Materials." This is in lieu of the requirement to have a "Caution, Radioactive Material," or "Danger, Radioactive Material," label affixed to each container of licensed material. See Chapter 4 for additional details.

1.2.5.3 Contamination-Free Articles

NFS is authorized to use the limits specified in "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material," U.S. Nuclear Regulatory Commission, April, 1993, (See Chapter 4) for determining contamination levels on facilities released to uncontrolled areas, and on equipment released for unrestricted use.

1.2.5.4 Decommissioning Funding Plan

NFS is exempt from the requirements in 10 CFR 70.25(e) specifying that one of the listed methods in 10 CFR 70.25(f) must be used for financial assurance. The financial arrangements to assure that decommissioning funds will be available are set forth in Chapter 10. This exemption is limited to the use of a statement of intent (or an equivalent contract clause) from a government agency, as outlined below.

1. The exemption stated above is applicable to the decommissioning activities for which the U.S. Government has assumed liability per Appendix 10A of Chapter 10. The NFS/USDOE Contract language in said

Appendix 10A also makes it necessary for NFS to establish a cost estimate and a financial assurance plan for those decommissioning activities not covered by the Government.

1.2.5.5 Decommissioning-Related Activities Performed Prior to the End of Plant Life

Facilities or grounds may be remediated/decontaminated on a project-by-project basis prior to the end of plant life. These projects will address portions of the facility no longer in use or in need of decontamination to protect the environment. The portions of the NFS plant subject to these operations may be used for future licensed activities, require clean-up to protect the environment, or be conducted as a precursor to decommissioning an area under a NRC approved final status survey and release plan. Decommissioning-related activities, including associated procedures, are reviewed against the criteria in 10 CFR 70.38(g)(1) to determine if a decommissioning plan is required and the results of the review are documented. If required, the plan must be submitted to NRC for review and approval prior to starting the activities. Such operations are described further in Chapter 10.

1.2.5.6 Transportation of SNM

NFS is authorized to ship SNM up to and exceeding a formula quantity using physical protection measures for SNM of low strategic significance under 10 CFR 73.67(g) when certain conditions are met. The conditions are contained in the NFS Category III Physical Protection Plan. This exemption is limited to material in transit; fixed site security requirements remain unchanged.

1.2.5.7 Use of ICRP 68 DAC and ALI Values

Notwithstanding the requirements, the derived air concentration (DAC) values and the annual limit on intake (ALI) values listed in Appendix B of 10 CFR Part 20, NFS may use adjusted DAC values and adjusted ALI values specified in Publication 68 of the International Commission on Radiation Protection (ICRP-68). Additional information can be found in Section 4.7.9.1 of this application.

1.2.6 Security of Classified Information

NFS has been issued a facility security clearance in accordance with 10 CFR 95.

1.2.7 Terminology/Definitions

Definitions for terms specific to a particular safety function may be given in the corresponding chapter on that function. The following definitions apply to terms used in this license:

Term	Definition
^{235}U Enrichments	"Low enriched uranium" is defined as any compound of uranium in which the enrichment in the isotope uranium-235 is less than 20 percent by weight. "High enriched uranium" or "highly enriched uranium" is defined as any compound of uranium in which the enrichment in the isotope uranium-235 is equal to or greater than 20 percent by weight.
Nuclear Safety	Nuclear criticality safety
Will, Shall	A requirement.
Should	A recommendation.
May	Permission (optional), neither a requirement nor a recommendation.
Are	An existing practice for which there is a requirement to continue.
Monthly	An interval not to exceed 35 days.
Quarterly	An interval not to exceed 4 months.
Semi-Annually	An interval not to exceed 7 months.
Annually	An interval not to exceed 14 months.
Biennially	An interval not to exceed 28 months.
Triennially	An interval not to exceed 42 months.
Criticality Control	The administrative and technical requirements established to minimize the probability of achieving inadvertent criticality in the environment analyzed.
Work Area Air Samplers	Stationary air samplers demonstrated to be representative of workers breathing air. If stationary air samplers have not been demonstrated to be representative, the results of lapel air samplers will constitute work area air samples.

Term	Definition
Equivalent Experience	For the purpose of meeting educational requirements described throughout the license, two (2) years experience is considered to be equivalent to one (1) year of post-secondary education. For example, two (2) years of post-secondary education (associate degree) in a relevant field and four (4) years experience will satisfy the requirement for a B.S. degree (4 years of post-secondary education).
U-233 Action Levels	The action levels used for U-233 shall be those used for highly enriched uranium (HEU).
Protected Area	A site area bounded by a security barrier and outer fence, separated by an exclusion zone, designed to provide physical security. The area contains radioactive material processing, storage, and laboratory areas, as well as support functions.
Restricted Area	A site area in which individuals may be exposed to radiation or radioactive material at levels or concentrations in excess of that allowed for the general public (see definition in 10 CFR 20.1003). This could include any location at the NFS Erwin facility, depending upon activities conducted and the exposure potential as evaluated by the safety function.
Radiologically Controlled Area	A site area where uncontained radioactive material is present, such that contamination levels are likely to be encountered in excess of acceptable levels for unrestricted use. This type of area, designated for contamination control purposes, requires various levels of protective clothing and other personnel protective actions. It could include any location within the Restricted Area, either on a permanent or temporary basis.
Uncontrolled Area	A site area where radioactive materials may be handled in the form of sealed sources, in packages or closed containers, in small amounts (air samples, bioassay samples, etc.), or not at all. This type of area is designated for contamination control purposes and is not likely to have contamination at levels in excess of those acceptable for unrestricted use.

Term	Definition
Conditions Adverse to Safety	As used in Sections 2.2, 2.5.1, and 11.6, events that could have the potential to impact the safety of licensed activities, including equipment failures, malfunctions, or deficiencies; procedure problems, errors, or omissions; improper installations; non-conformances with regulatory requirements or commitments; quality-related issues; or a significant condition, such that if uncorrected, could have a serious effect on safety.

1.3 Site Description

1.3.1 Site Geography

The NFS site is located at 1205 Banner Hill Road, inside the city limits of Erwin, in Unicoi County, Tennessee. The Protected Area of approximately 18 acres lies within approximately 70 acres of land owned by NFS. The property is situated at approximately latitude 36°07'47"N and longitude 82°25'57"W.

The facility is bounded on the north by Martin Creek; on the south by residential properties; on the east by Banner Hill Road, an asphalt roadway providing access to the NFS site; and on the west by CSX Railroad. Interstate 26 is located just west of the NFS property, within one (1) mile of the site boundary.

There are four (4) bodies of surface water adjacent to or in the immediate vicinity of the plant. The site contains a natural spring (Banner Spring), which originates on the NFS property. Banner Spring forms Banner Spring Branch, which is routed through an underground pipe across the site and empties into Martin Creek at the site boundary. Martin Creek empties into North Indian Creek approximately 3,500 feet north of the NFS site, and North Indian Creek empties into the Nolichucky River approximately one (1) mile from the site boundary.

The site is located in a southwest-to-northeast oriented valley, bounded on both sides by the Blue Ridge Mountains of the Appalachian Mountain chain. The surrounding mountains have a maximum elevation of approximately 2,480 feet above sea level. The topography of the NFS property is relatively level, with site elevations ranging from approximately 1,640 to 1,680 feet above sea level.

1.3.2 Demographics

The NFS site is located inside the Erwin city limits. The city of Erwin, the seat of Unicoi County, has a population of approximately 6,100 people, and the population of Unicoi County is approximately 17,700 people. Approximately 2,800 people live within one (1) mile of the NFS site.

Erwin Health Care Center, a nursing home, is the only public facility within one (1) mile of the NFS site. Four other schools, Love Chapel Elementary School, Unicoi County Intermediate School, Unicoi County Middle School, and Unicoi County High School, are approximately 1.3 miles northeast of the NFS site. The nearest hospital, Unicoi County Memorial Hospital, and an adjacent nursing home, Center for Aging and Health, are approximately 1.2 miles northeast of the NFS site.

Land use within one (1) mile of the NFS site is a mixture of residential and agricultural activities, as well as several industrial facilities. The industrial facilities, including Erwin Resin Solutions, Inc., a low-level radioactive waste processing facility, are located adjacent to the southern NFS site boundary. A railroad yard owned by CSX Transportation is located adjacent to the western NFS site boundary.

The Nolichucky River, located near the site boundary, is used primarily for recreational purposes (white water rafting, canoeing, fishing, etc.) and serves as irrigation water for agricultural activities. The Nolichucky River also serves as a source of drinking water for the Town of Jonesborough, and the water treatment plant intake is located approximately 8 miles downstream of the NFS site.

1.3.3 Meteorology

Prevailing winds at the NFS site tend to be from the southwest following the orientation of the valley, southwest to northeast. The 30-year average wind speed is 6.9 mph.

The East Tennessee region has a climate with warm, humid summers and relatively mild winters. The average total annual rainfall in the Erwin area is 37.3 inches, and the average total annual snowfall is 25 inches. The average annual temperature is 55°F, with a monthly average minimum temperature in January of 25°F and a monthly average maximum temperature in July of 87°F.

Severe storm conditions are infrequent in the Erwin area, due to the fact that the area is east of the center of tornado activity, south of most blizzard conditions, and too far inland to be affected by hurricane activity. According to National

Oceanic and Atmospheric Administration (NOAA) data since 1950 for Unicoi County, the maximum high wind recorded was 69 miles per hour.

There have been 2 tornadoes recorded in Unicoi County since 1950 which occurred in 1980 and 2011. Due to the low frequency of tornadoes in this region, no specific design criteria relative to tornadoes are required in the International Building Code.

Lightning risk at the NFS site has been addressed by evaluating facility operations and the potential for damage due to lightning strikes. See Chapter 7 for additional details.

1.3.4 Hydrology

There are four (4) bodies of surface water adjacent to or in the immediate vicinity of the plant. The site contains a natural spring (Banner Spring), which originates on the NFS property. Banner Spring forms Banner Spring Branch, which is routed through an underground pipe across the site and empties into Martin Creek at the site boundary. Martin Creek empties into North Indian Creek approximately 3,500 feet north of the NFS site, and North Indian Creek empties into the Nolichucky River approximately one (1) mile from the site boundary.

Based on the 2008 National Flood Insurance Map published by FEMA for the Erwin area, the NFS site is located outside of the 100-year floodplain of the Nolichucky River. However, the northern portion of the NFS site is located within the 100-year floodplain of Martin Creek. The culvert that allows Martin Creek to pass under the CSX Railroad was enlarged in 1990, and NFS has constructed a berm along the northern site boundary, both of which effectively lower the potential for flooding of the NFS site due to Martin Creek. The floodplain elevation mapping has not been updated to take these factors into account. Potential impacts due to flooding in facilities located in the northern portions of the NFS site are further minimized by early warning and associated mitigative efforts (removal/relocation of materials and equipment susceptible to water damage, sandbagging, etc.) during potential flooding conditions.

Depth to the water measurements taken at wells in the vicinity of the NFS site range from 5 to 19 feet below land surface, with an average of 11 feet. Groundwater elevation measurements and modeling indicate that groundwater generally flows in a northwest direction toward the Nolichucky River, which is a major discharge zone for the groundwater flowing under the NFS site, at an average rate of 0.5 to 114 feet/day, with an average of 22 feet/day. There are no known household, public, or industrial users of groundwater downgradient of the NFS site. A potentiometric surface map for the groundwater under the NFS site is included in Chapter 9.

The uppermost aquifer at the NFS site is the alluvial aquifer. This alluvial aquifer is limited in areal extent and is found mainly in the lowland areas. The alluvial aquifer pinches out just north and south of the site due to the presence of shallow bedrock. Alluvial deposits are generally very heterogeneous in sediment size, composition, and depositional pattern, causing varying degrees of anisotropy throughout these deposits. The presence of large amounts of clay in suspended and mixed-load stream deposits commonly causes the vertical hydraulic conductivity to be orders of magnitude less than in a horizontal direction.

1.3.5 Geology

The NFS site lies in the Valley and Ridge physiographic province of northeastern Tennessee. The area topography consists of a series of alternating valleys and ridges that have a northeast-southeast trend, with the NFS site located in a valley. The present topography of the valley is the result of stream erosion of softer shales and limestones. The bedrock strata at the NFS site are consolidated, providing firm foundations for buildings that lie directly on the strata or that are supported by footings. Foundations for buildings that house licensed activities are supported by soil which meets the bearing capacities required by the building design.

Although common in the mountainous terrain surrounding the NFS site, slope failures are not common on the former flood plain where slopes are flat. Structures are set back sufficiently from the Nolichucky River and Martin Creek to avoid destabilization due to erosion or slope failures along the waterway banks.

The NFS site is located in the moderately active Appalachian Tectonic Belt, Seismic Zone 2, indicating that moderate damage could occur as the result of earthquakes. There is no evidence of capable faults as defined by 10 CFR 100 in the immediate vicinity of the NFS site. A seismic analysis of the NFS site conducted in 2001 determined that the horizontal component of ground motion for a safe shutdown earthquake with a 1000-year return period has a peak ground acceleration of 0.06 gravity, and the vertical acceleration is two-thirds of the horizontal, or 0.04 gravity.

APPENDIX 1A

NUCLEAR FUEL SERVICES, INC.
AFFILIATES

1. **BWX Technologies, Inc.** is a corporation that owns 100% of the stock of BWXT Investment Company.

2. **BWXT Investment Company** is a corporation that owns 100% of the stock of BWXT Government Group, Inc.

3. **BWXT Government Group, Inc.**, is a corporation that owns 100% of the stock of BWXT Nuclear Operations Group, Inc.

4. **BWXT Nuclear Operations Group, Inc.**, is a corporation that owns 100% of the stock of NOG-Erwin Holdings, Inc.

5. **NOG-Erwin Holdings, Inc.**, is a corporation which owns 100% of the stock of NFS Holdings, Inc.

6. **NFS Holdings, Inc.**, is a corporation which owns 100% of the stock of Nuclear Fuel Services, Inc.

7. **Nuclear Fuel Services, Inc. (NFS)**, is a manufacturer and processor of specialty nuclear fuels which is also engaged in decontamination, decommissioning, and remediation services. These services are performed at NFS' Erwin, Tennessee, location.

NOTE: This listing does not include certain affiliate companies that are not relevant to licensed activities.

Figure 1A-1
NFS Corporate Structure

NOTE: This chart is a simplified organization chart and does not include certain affiliate companies that are not relevant to licensed activities.

APPENDIX 1B

LISTING OF CHEMICAL AND PHYSICAL FORMS OF URANIUM AUTHORIZED

The physical forms of uranium which may be used in licensed operations are:

Solid Forms,
Liquid Forms, and
Gaseous Forms

The following listing contains the chemical compounds of uranium which may be present in licensed operations. Other compounds may be present as transitory compounds. This listing does not include materials in which uranium may be present as a mixture:

LISTING OF URANIUM COMPOUNDS	
Compound Name	**Compound Formula**
Acid deficient uranyl nitrate	$UO_2 (NO_3)_x$ where x is less than 2
Ammonium diuranate	$(NH_4)_2U_2O_7$
Ammonium uranyl carbonate	$(NH_4)_4UO_2(CO_3)_3$
di-Ammonium uranylcarbonate	$2(NH_4)_2CO_3UO_2CO_32H_2O$
Ammonium pentauranylfluoride	$(NH_4)_3UO_2F_5$
Potassium metauranate	K_2UO_4
Potassium uranyl acetate	$KUO_2(C_2H_3O_2)_3H_2O$
Potassium uranyl carbonate	$2K_2CO_3UO_2CO_3$
Potassium uranyl sulfate	$K_2SO_4UO_2SO_42H_2O$
Sodium metauranate	Na_2UO_4
Sodium uranyl acetate	$NaUO_2(C_2H_3O_2)_3$
Sodium uranyl carbonate	$2Na_2CO_3UO_2CO_3$
Uranium (metal)	U
Uranium diboride	UB_2
Uranium tetrabormide	UBr_4
Uranium tribromide	UBr_3
Uranium dicarbide	UC_2
Uranium carbide	UC_x, where x is less than 2
Uranium pentachloride	UCl_5
Uranyl hydroxide	$UO_2(OH)_2$
Uranium tetrachloride	UCl_4
Uranium trichloride	UCl_3
Uranium hexafluoride	UF_6
Uranium tetrafluoride	UF_4

LISTING OF URANIUM COMPOUNDS	
Compound Name	**Compound Formula**
Uranium trifluoride	UF_3
Uranium hydride	UH_3
Uranium tetraiodide	UI_4
Uranium mononitride	UN
Uranium dioxide	UO_2
Uranium peroxide	$UO_4 2H_2O$
Uranium trioxide	UO_3
triUranium octoxide	U_3O_8
Uranium sulfate	$U(SO_4)_2 4H_2O$
Uranium sulfate	$U(SO_4)_2 8H_2O$
Uranium sulfate	$U(SO_4)_2 9H_2O$
Uranium disulfide	US_2
Uranium monosulfide	US
Uranium sesquisulfide	U_2S_3
Uranyl acetate	$UO_2(C_2H_3O_2)_2 2H_2O$
Uranyl benzoate	$UO_2(C_7H_5O_2)_2$
Uranyl bromide	UO_2Br_2
Uranyl carbonate	UO_2CO_3
Uranyl perchlorate	$UO_2(ClO_4)_2 6H_2O$
Uranyl chloride	UO_2Cl_2
Uranyl fluoride	UO_2F_2
Uranyl formate	$UO_2(CHO_2)_2H_2O$
Uranyl iodate	$UO_2(IO_3)_2$
Uranyl iodate	$UO_2(IO_3)_2H_2O$
Uranyl iodide	UO_2I_2
Uranyl nitrate hexahydrate	$UO_2(NO_3)_2 6H_2O$
Uranyl nitrate	$UO_2(NO_3)_2$
Uranyl nitrate hydrate	$UO_2(NO_3)_2XH_2O$, where X is less than 6
Uranyl oxalate	$UO_2(C_2O_4)_2 3H_2O$
Uranyl mono-H phosphate	$UO_2HPO_4 4H_2O$
Uranyl potassium carbonate	$UO_2CO_3 2K_2CO_3$
Uranyl sodium carbonate	$UO_2CO_3 2Na_2CO_3$
Uranyl sulfate	$UO_2SO_4 3H_2O$
Uranyl sulfate	$2(UO_2SO_4) 7H_2O$
Uranyl sulfide	UO_2S
Uranyl sulfite	$UO_2SO_3 4H_2O$

Attachment 4

SNM-124, Chapter 2

(13 pages to follow)

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 2

ORGANIZATION AND ADMINISTRATION
Table of Contents

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 2

ORGANIZATION AND ADMINISTRATION

2.1 General Safety Policy and Responsibilities

It is NFS' policy that radiation exposures to employees and the general public be kept as low as reasonably achievable (ALARA). Responsibility for safety in the various production lines, processes, and services is delegated to the lowest practical level of supervision. Safety is the responsibility of each supervisor within his own area. Through training and periodic retraining, each individual, regardless of position, is made aware that safety in his work area is ultimately his responsibility.

2.2 Site Organization

The NFS corporate organization provides the management, administrative, and technical capabilities for ensuring that NFS site operations utilizing SNM are conducted in a manner that is protective of its workers, the public, and the surrounding environment, and remain in compliance with applicable Federal, State, and local regulations, licenses, and permits. This responsibility is implemented through the functional disciplines of production, decommissioning, engineering, safety, material control and accountability, security, and quality assurance, as described in the sections below, all of which have safety-related responsibilities. Figure 2-1 shows the current NFS functional organization.

The management positions for each discipline together have the delegated responsibility for plant safety and for compliance with conditions of SNM licenses and with federal, state, and local regulations and laws governing operation of a nuclear facility in order to maintain a safe work place for all employees. Each discipline management team is responsible for

- ensuring that all activities in their area are performed in a safe and effective manner;
- managing and directing operations within their discipline;
- ensuring that all operations under its guidance comply with safety and license conditions, requirements for quality-related safety activities, and safety-related configuration management requirements;
- being knowledgeable of the safety procedures and programs as they relate to their area of responsibility;
- developing, approving, and implementing procedures that incorporate safety and quality controls and limits commensurate with the particular operation involved; and

- ensuring that conditions adverse to safety are reported and investigated promptly, and that corrective actions are tracked to completion and, as applicable, monitored for effectiveness.

2.3 Organizational Responsibilities, Authority, and Qualifications

This section describes the functional responsibilities, education, and experience of key positions required by this license.

Key personnel are those individuals who are responsible for safety and for safe operation of the site and include the president and the senior managers of the disciplines described in this section. Senior managers include discipline directors and managers that meet the qualifications specified 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.

The positions described in this section are intended to be generic in nature and do not reflect specific organizational titles or jobs. The responsibilities of the positions described may be fulfilled by one or more different organizational positions as long as the minimum position qualifications specified in this chapter are met.

2.3.1 President

The president, or the director authorized to be his alternate, has the overall responsibility for the safety, security, quality, and operational aspects of all activities conducted at the NFS site. Daily responsibility for licensed activities may be delegated in writing to one or more of the director positions specified in Sections 2.3.2 and 2.3.5.

The minimum qualifications for the president are a BS/BA degree in science or engineering and ten years of experience in industry or nuclear reactor operations – five of which have been in a supervisor position in the nuclear industry or reactor operations.

2.3.2 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 discipline includes activities associated with product

research and development, process engineering, enriched uranium processing, transportation and waste management, and related equipment installation, start-up, and maintenance. This discipline 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.

The qualifications for a production discipline director are a BS/BA degree in science or engineering and ten years of experience in industry or nuclear reactor operations – five of which have been in a supervisory position in the nuclear industry or reactor operations.

The minimum qualifications for a production discipline manager are a BS/BA degree and at least five years experience in production-related activities, two years of which have been in the nuclear fuel cycle.

2.3.3 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.

The minimum qualifications for a decommissioning discipline manager are a BS/BA degree and at least five years experience in decommissioning-related activities, two years of which have been in the nuclear fuel cycle.

2.3.4 Engineering

The Engineering Discipline designs and installs new and modified facilities and equipment; supplies maintenance and process engineering support; conducts activities associated with product research and development; and assures that all equipment and facilities have appropriate safety controls and have been evaluated within the spirit and intent of ALARA.

The minimum qualifications for an engineering discipline manager are a BS/BA degree in science or engineering and at least five years experience in engineering-related activities, two years of which have been in the nuclear fuel cycle.

2.3.4.1 Configuration Management

The Configuration Management function is responsible for establishing consistency among design and regulatory requirements, physical configuration, and facility configuration information; and for maintaining this consistency throughout the life of the facilities and activities until the point that CM is no longer needed.

The minimum qualifications for a CM function manager are a BS/BA degree and at least five years experience understanding CM requirements (understanding document control, equipment identification, change control, etc.), two years of which have been in the nuclear fuel cycle.

2.3.5 Safety

The Safety Discipline provides programs, procedures, and reviews to assure worker health and safety; environmental protection; and compliance with licenses and permits. These activities are conducted with the ALARA principle in mind. Functional areas include nuclear criticality safety, radiation protection, industrial safety, environmental protection, licensing, and integrated safety analysis. Emergency preparedness and response programs are supported by each functional area as needed. The Safety Discipline monitors operations to ensure they are conducted in compliance with federal, state, and local regulations, and is authorized to suspend operations, approve re-start of operations, and/or require additional safety precautions when such measures are necessary in the interest of plant safety. The Safety Discipline is administratively independent of the Production Discipline, but both disciplines may report to a common management position. The Safety Discipline is responsible for overseeing the safety review committee, and as such provides a manager to fill the role of the chairman. The ISA process is supported by each functional area providing ISA Team members as needed.

The qualifications for a safety discipline director are a BS/BA degree in science or engineering and ten years of experience in industry or nuclear reactor operations – five of which have been in a supervisory position in the nuclear industry or reactor operations.

The qualifications for a safety discipline manager are a BS/BA degree in science or engineering with at least eight years experience in applied health physics and/or nuclear safety. An MS degree in radiological physics or nuclear engineering may be substituted for two years of the experience.

2.3.5.1 Nuclear Criticality Safety Function

The nuclear criticality safety function has responsibility for the development and implementation of a comprehensive nuclear criticality safety program, as defined in Chapter 5. Key responsibilities include the performance of nuclear criticality evaluations of applicable SNM operations and proposed changes to those operations; establishing limits and controls based on those evaluations; assuring the proper incorporation of limits and controls into applicable procedures and work instructions; and monitoring plant compliance with the nuclear criticality safety requirements through inspections and audits.

The qualifications for the manager of the nuclear criticality safety function are a BS/BA degree in science or engineering and at least three years experience in nuclear criticality safety.

The qualifications for a nuclear criticality safety senior member are a BS/BA degree in science or engineering and at least three years experience in criticality safety work. The qualifications for a criticality safety junior member are a BS/BA degree in science or engineering and at least one year of experience in criticality safety work.

2.3.5.2 Radiation Protection Function

The radiation protection function has responsibility for establishing and maintaining the radiation safety program necessary to ensure the protection of employees at NFS and the community, as defined in Chapter 4. Key responsibilities include management of the ALARA, dosimetry, and radiation monitoring and surveillance programs; analysis and approval of operations involving radiological safety and proposed changes to those operations; establishing radiation protection criteria, procedures, and training programs to control contamination and exposure to individuals and the environment; and monitoring plant compliance with the radiological protection criteria through inspections and audits. Radiation monitoring includes measurement of airborne radionuclide concentration, contamination level, and external radiation levels; evaluation of the operational integrity and reliability of radiation detection instruments; and maintenance of records related to the radiation monitoring program. These tasks are accomplished through the use of radiation technicians, radiation technician supervisors, health physicists, the radiation monitoring manager, and the manager of the radiation protection function.

The manager of the radiation protection function is responsible for administering the activities associated with radiological safety. He or she is responsible directly to the NFS president (or equivalent) in vital matters of radiological safety. This includes monitoring and control of areas of airborne radioactivity, surface

contamination, containment, ventilation, internal and external dosimetry, and bioassay services. To assist the manager, health physicists have been charged with developing and implementing radiological control programs to meet program goals and objectives.

The radiation monitoring manager administers the safety monitoring program to comply with license conditions and government regulations. The radiation technician supervisors coordinate and assign daily radiation monitoring tasks supporting health physics. The radiation technicians perform the monitoring tasks as assigned by the supervisors.

The qualifications for the manager of the radiation protection function are a BS/BA degree in science or engineering and at least three years of experience in applied health physics in a program dealing with radiation safety problems similar to the one managed.

The qualifications for a health physicist are a BS/BA degree in science or engineering, and at least one year of experience in health physics. A Master's degree in health physics or related discipline may be substituted for one year of experience.

The qualifications for the radiation monitoring manager and the radiation technician supervisor are at least two years of college, and one year of experience in applied health physics.

Training and qualifications for radiation protection personnel are based on guidance from NRC Regulatory Guide 1.8 (2000).

2.3.5.3 Environmental Protection Function

The environmental protection function has responsibility for establishing and maintaining the environmental protection program necessary to ensure the protection of the public and the environment, as defined in Chapter 9. Key responsibilities include identification of environmental requirements of federal, state, and local regulations governing NFS' operations; assurance of proper federal and state permits, licenses, and registrations for radiological and non-radiological discharges from the facility; analysis and approval of operations involving potential environmental releases and proposed changes to those operations; establishing environmental protection criteria, procedures, and training programs to monitor gaseous and liquid effluents; and monitoring plant compliance with the environmental protection criteria through inspections and audits.

The qualifications for the manager of the environmental protection function are a BS/BA degree in science, and at least three years experience in applied health physics or environmental protection.

The qualifications for an environmental protection analyst are a BS/BA degree in science, and at least one year of applied health physics or environmental protection experience.

2.3.5.4 Industrial Safety Function

The industrial safety function has responsibility for industrial hygiene or chemical safety; industrial safety; and respiratory protection. Key responsibilities include analysis and approval of operations involving industrial safety and proposed changes to those operations; establishing industrial safety criteria, procedures, and training programs to protect the workers from industrial hazards; and monitoring plant compliance with the industrial safety/hygiene program through inspections and audits.

The qualifications for the manager of the industrial safety function are a BS/BA degree in industrial hygiene, or safety, or other appropriate field, and at least three years industrial experience in fire protection, respiratory protection, industrial hygiene, or other closely related areas.

The requirements for advanced industrial safety specialist positions are a BS/BA degree with specialized training in environmental health, fire protection, industrial safety/hygiene, or closely related field, and at least three years of industrial safety experience. Lower level positions require at least two years of industrial safety or equivalent plant experience. Only personnel knowledgeable in hazards evaluation and control methods for chemical process safety will perform chemical process safety reviews.

2.3.5.5 Licensing and Integrated Safety Analysis Function

The licensing function has overall responsibility for acquiring and maintaining safety-related licenses as required to operate facilities at the NFS site, as well as the broad responsibility for interface with regulatory agencies.

The integrated safety analysis function has the overall responsibility for the ISA program, as defined in Chapter 3. Key responsibilities include the performance of chemical, radiological, and fire evaluations of applicable SNM operations and proposed changes to those operations; establishing IROFS based on those evaluations; assuring the proper incorporation of IROFS into applicable procedures and work instructions; coordinating updates to the ISA; and

monitoring plant compliance with ISA requirements through inspections and audits. The ISA function also has responsibility for managing the Safety Related Equipment program for functionally testing IROFS on a periodic basis, as defined in Chapter 11, as well as the fire protection program defined in Chapter 7.

The qualifications for the manager of the licensing and/or ISA functions are a BS/BA degree in science or engineering and at least three years experience in licensing, regulatory compliance, safety, or safety analysis in the nuclear or another highly regulated industry. In addition, the manager of the ISA function must have at least three years experience in fire protection.

2.3.6 Material Control and Accountability

The Material Control and Accountability (MC&A) 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.

The minimum qualifications for an MC&A discipline manager are a BS/BA degree and at least five years experience in MC&A-related activities, two years of which have been in the nuclear fuel cycle.

2.3.7 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.

The minimum qualifications for a security discipline manager are a BS/BA degree and at least five years experience in security-related activities, two years of which have been in the nuclear fuel cycle.

2.3.8 Quality Assurance

The Quality Assurance Discipline assesses 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 performing audits, surveillances, and assessments of quality-related safety activities. The quality assurance program is based on, but is not limited to, applicable requirements and guidance such as ASME NQA-1,

MIL-Q-9858A, or other similar guidance. The quality assurance discipline is administratively independent of operations, and has no other duties or responsibilities unrelated to quality assurance that would interfere with carrying out the duties of this discipline.

The qualifications for a quality assurance discipline director are a BS/BA degree in science or engineering and ten years of experience in industry or nuclear reactor operations – five of which have been in a supervisory position in the nuclear industry or reactor operations.

The qualifications for a quality assurance discipline manager are a BS/BA degree and at least five years experience in quality assurance-related activities, two years of which have been in the nuclear fuel cycle.

2.4 Safety Review Committee

The safety review committee membership includes senior managers of the following disciplines:

- Production;
- Engineering;
- Safety;
- Material Control and Accountability; and,
- Security.

The chairman, other committee members, and their alternates, are appointed by the president, or the director authorized to be his alternate. At a minimum, the chairman is required to have the qualifications specified for a safety function manager, and the other committee members are required to have the qualifications specified for a discipline manager. Members of the safety review committee, as identified above, have completed training in incident investigation methods, and the completion of the training is documented.

The committee is responsible to the president, or the director authorized to be his alternate, who retains overall authority for the approval or disapproval of committee actions.

The authority and responsibilities of the full safety review committee include the following:

- Reviewing proposed license changes affecting safety, physical security, and/or material control and accountability before the associated license amendment applications are submitted to the NRC.

- Reviewing the ALARA program for at least the following:
  - ❖ Trends in air activity,
  - ❖ Cumulative exposure,
  - ❖ Engineering design and personnel work practices.
- Working with the safety discipline to implement the ALARA program.
- Reviewing results of safety inspections, audits, and investigations which the license requires be conducted.
- Reviewing all violations of regulations or license conditions having safety significance.

The committee will meet at the following frequencies:

- to discuss topics such as proposed license changes – as needed;
- to discuss ALARA considerations – at least semiannually;
- to review license-required safety inspections, audits, investigations, and violations of regulations or license conditions – at least quarterly.

Its proceedings, findings, and recommendations will be documented in writing and made available to the president, discipline directors, and discipline managers. Such reports will be retained for at least five years.

The chairman of the safety review committee, with concurrence of the remaining committee members, is authorized to select individual committee members to review and approve new or revised operating and general safety procedures. However, the review and approval of such procedures, as described herein, include at a minimum the initiating discipline manager, the safety discipline manager, and the appropriate safety review committee members, as selected by the safety review committee chairman. If an active procedure has not been revised within a three-year period, the chairman may select individual committee members to review the procedure to ensure it remains current and relevant. Records of procedural changes will be maintained for a minimum of five years.

Committee review of matters other than the bulleted items above may be conducted by either individual review or collectively at a meeting; however, individual members of the committee have the authority to request a meeting of the entire committee on any given matter.

2.5 Administration

2.5.1 Reporting of Potentially Unsafe Conditions or Activities

A problem identification system is available for any person at the NFS site to report potentially unsafe conditions or activities to the Safety Discipline. Prompt reporting is expected so that conditions adverse to safety can be corrected as soon as practicable. The concern is entered in the system, and processed through a screening committee with Safety Discipline representation. The screening committee assigns the issue to an owner and defines follow-up investigation/evaluation requirements. Corrective actions are assigned and tracked to completion. The Corrective Action Program is discussed further in Chapter 11.

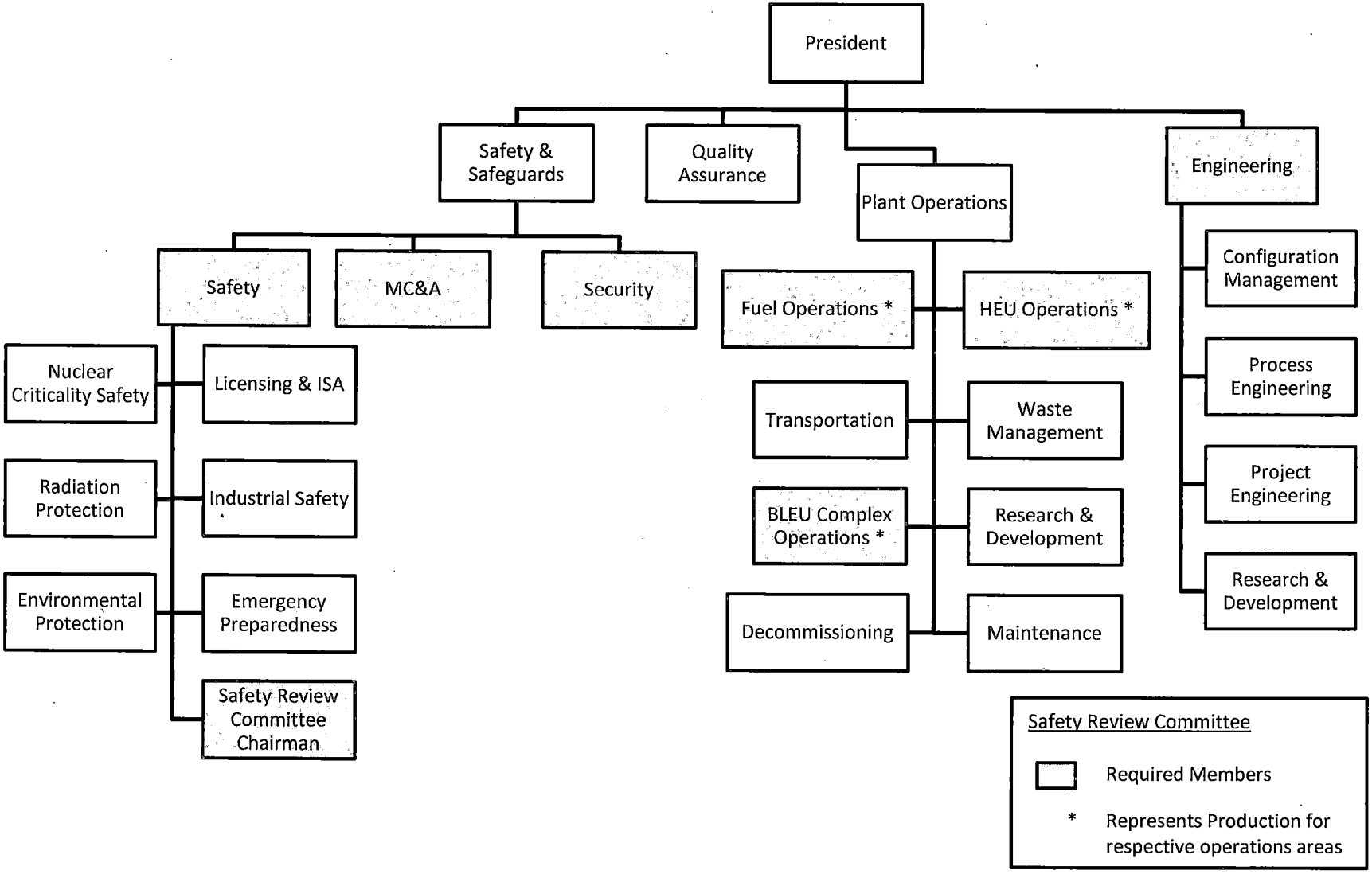
2.5.2 Management Measures

Management measures that ensure the reliability and availability of IROFS are established as described in Chapter 11.

2.5.3 Off-Site Emergency Response Resources

Written agreements with off-site emergency response organizations are described in Chapter 8.

Figure 2-1: Functional Organization Chart



Attachment 5

SNM-124, Chapter 4

(57 pages to follow)

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

RADIATION PROTECTION
Table of Contents

RADIATION PROTECTION

4.1 Radiation Protection Program

NFS will establish, maintain, and implement a Radiation Protection Program (RPP) commensurate with the scope and extent of licensed activities and sufficient to ensure compliance with the provisions of 10 CFR 20.1101. This will include:

- Use of Engineered and Administrative Controls to maintain radiation exposure as low as reasonably achievable (ALARA).
- Development of procedures for implementation of the RPP.
- Implementation of a self assessment program to periodically (at least annually) review the RPP.
- A staff of suitably trained radiation protection personnel, with sufficient resources to implement the RPP independent from facility operations.

The RPP will be structured to include a specific program for:

- ALARA
- Contamination Control
- Internal and External Dosimetry
- Dose Registry
- Training
- Safety (Radiation) Work Permits
- Airborne Radioactivity Monitoring
- Sealed Source Control

Key program personnel with program ownership and responsibility, as defined in Section 2.3.5.2 of this license, will be established.

The NFS program for implementation of radiation protection, including those used to monitor personnel and public exposures, facilitate contamination control and those ensuring that exposures are maintained 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).
- Safety Work Permits (SWPs).
- NFS ALARA Program Document.

4.2 ALARA Program

It is the policy of NFS to maintain a comprehensive RPP whose objective is to keep the radiation doses to workers and the off-site releases of radioactivity not only below regulatory limits, but also as low as reasonably achievable; i.e. "ALARA." In implementing this policy, the following guidelines are adhered to:

- Each person working within a Restricted Area receives sufficient radiation safety training to understand the reasons for radiation safety and the principles of ALARA.
- NFS' safety review committee serves as the ALARA Committee and assures that operating procedures incorporate controls to ensure that exposure to radiation and the release of radioactivity are maintained as far below regulatory limits as is reasonably achievable.
- The Erwin Plant is operated and maintained in a manner which minimizes to the extent practical radiation exposures, the spread of contamination, contamination of facilities in support of eventual decommissioning, the generation of radioactive wastes, and the release of radioactivity to unrestricted areas. Each discipline manager is responsible for assuring that appropriate radiation protection controls are incorporated into all activities under their supervision. Each person working within a Restricted Area accepts the responsibility for maintaining his/her exposure ALARA by complying with approved procedures.

- Modifications or changes to the Erwin Plant are designed and constructed giving full consideration to the ALARA concept. These modifications and changes to the Erwin Plant shall incorporate the ALARA concepts specified in 10 CFR 20.

NFS' management is committed to and will make appropriate assignments to implement an ALARA program.

An ALARA Report will be issued to NFS management on a quarterly basis to review employee exposure and effluent release data. In addition to this report, performance metrics are maintained and/or periodic reports are made to the safety review committee to:

- Determine if there are any upward trends developing in personnel exposures for identifiable categories of workers or types of operations or effluent releases.

- Determine if exposures and effluents might be lowered under the concept of as low as reasonably achievable.

- Determine if equipment for effluent and exposure control is being properly used, maintained, and inspected.

- Review other required audits and inspections performed during the period of the report.

- Review the data from employee exposures, dosimetry results, effluent releases, in-plant airborne radioactivity, and environmental monitoring.

- Report the results of airborne concentrations of radioactivity and surface contamination at work stations and areas.

4.2.1 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 4.2.2.

4.2.2 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, that 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; ALARA goals will incorporate, when appropriate, new approaches, technologies, operating procedures, or changes that could reduce potential radiation exposures at a reasonable cost,

  (f) Establishing an ALARA technical review committee composed of the safety review committee to review all proposed major 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, and

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

- NFS has appointed a health physicist, within the radiation protection function, with responsibility for overseeing and coordinating the ALARA Program.

4.3 Organization and Personnel Qualification

An organization has been established and will be maintained to implement the RPP independent of facility operations. Positions, qualifications of the program manager and staff, responsibility and authority are detailed in Chapter 2. The Radiation Safety Officer responsibilities are fulfilled by the radiation protection function manager.

4.4 Safety Procedures

Activities performed for the Radiation Protection Program are in accordance with approved written procedures. These procedures, which instruct in duties such as radiological surveillance and monitoring, and collecting and analyzing samples, are made available to personnel working in the safety function. Training and other means to assure that the procedures are understood and followed are conducted.

4.4.1 "A" Procedures

"A" Procedures are primarily for supervisory or technical 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 occupational 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

4.4.2 "B" Procedures

"B" Procedures are primarily for hourly 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

4.4.3 "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

4.4.4 "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
- Administering safety work permits
- Collection of bioassay samples
- Reporting radiation exposure summaries
- External radiation monitoring
- Radiological posting and labeling
- Radiation protection training

4.4.5 Other Procedures

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

4.4.6 Safety Work Permit (SWP) Program

As specified in Section 11.4, routine and repetitive work performed in Restricted Areas is administered by the use of operating procedures, letters of authorization, or special work instructions. Non-routine activities in these areas, which are not normally covered by documented procedures, are administered by the work request system. This includes facility construction, modification, repair, equipment maintenance, and service work.

SWPs are required within the work request system for non-routine activities involving significant hazards. SWPs include Radiation Work Permits (RWPs) and Industrial Safety Permits (ISPs). The health physicist will evaluate the need for a RWP based on the work scope, the radiological hazards, and the sufficiency of radiological controls provided by other means (job coverage, HP oversight, training or other work control documentation).

RWPs are used to delineate radiological controls, special monitoring & surveillance, and safety precautions that must be taken to maintain exposure ALARA. RWP controls and job site/work evolution are reviewed prior to beginning work. This review normally includes a visual inspection of the work site to determine the appropriateness of proposed controls and includes a pre-job briefing for workers. RWPs are approved by a health physicist or a radiation technician supervisor.

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 non-routine work is responsible for obtaining an RWP. The individual requesting the RWP is also responsible for assuring the RWP is approved and 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 maintained in a conspicuous location throughout the duration of the activity; and the work is monitored by a member of the radiation protection function.

Upon completion of the work under the RWP, the individual responsible for the work and the radiation 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 protection function for filing. RWPs are kept for a minimum of two years.

The SWP Program may also be used to administer permits for non-radiological hazards (ISPs) and prescribe appropriate controls, monitoring, and personal protective equipment. Responsibilities and elements of the SWP system are documented in written procedures.

4.5 Training

A Radiation Protection Training Program has been implemented sufficient to:

- Demonstrate compliance with the requirements of 10 CFR Parts 19 and 20
- Provide training, to all personnel and visitors entering restricted areas, commensurate with the health risk to which they may be exposed, or to provide trained escorts who have received training
- Provide a level of training based on the potential radiological health risks associated with that employee's work responsibilities
- Incorporate, in the Radiation Protection Training Program, the provisions in 10 CFR 19.12 and topics such as:
    - ❖ Correct handling of radioactive materials
    - ❖ Minimization of exposures to radiation and/or radioactive materials
    - ❖ Access and egress controls and escort procedures
    - ❖ Radiation safety principles, policies, and procedures
    - ❖ Monitoring for internal and external exposures
    - ❖ Monitoring instruments
    - ❖ Contamination control, including protective clothing and equipment
    - ❖ ALARA and exposure limits
    - ❖ Radiation hazards and health risks
    - ❖ Emergency response

The radiation protection function will review the Radiation Protection Training Program at least every 3 years, including an evaluation of the effectiveness and adequacy of the training program curriculum and instructors. Refresher training will be conducted at least every 3 years, to address changes in policies, procedures, requirements, and the facility ISA.

The following Regulatory guidance will be used to develop the Radiation Protection Training Program:

- Regulatory Guide 8.10, Rev. 1-R, May 1977: "Operating Philosophy For Maintaining Occupational Radiation Exposures As Low As Reasonably Achievable"
- Regulatory Guide 8.13, Rev. 3, June, 1999: "Instructions Concerning Prenatal Radiation Exposure"
- Regulatory Guide 8.29, February, 1996: "Instructions Concerning Risks From Occupational Radiation Exposure"
- ASTM E1168-95, 2008: "Standard Guide for Radiological Protection Training for Nuclear Facility Workers"

Further information on training is found in Chapter 11.

4.6 Ventilation and Respiratory Protection Program

4.6.1 Occupied Area Ventilation

In buildings where special nuclear materials are handled:

- Air flow shall be designed to have flow from areas of low contamination potential to areas of increasing relative potential for radioactive contamination when uncontained radioactive material is present. Face velocity measurements at the openings between occupied areas will be performed at least monthly to ensure compliance with this requirement.
- Ventilation for occupied areas shall be designed and installed to maintain average work station concentrations of airborne radioactive materials, during normal conditions, below the occupational Derived Air Concentration (DAC) values specified in 10 CFR Part 20, Appendix B.
- Ventilation for occupied areas shall be designed and installed to meet the intent of the company's ALARA (As Low As Reasonably Achievable) Program.

- In special circumstances where personnel occupation is limited, or during maintenance, decommissioning, equipment modification, facility shutdown, etc., where installation of such engineering controls is impracticable and/or infeasible, alternatives such as the use of portable containment, respiratory protection devices, or enhanced monitoring, shall be used to control exposure to radioactive materials.

4.6.2 Process Enclosure and Exhaust Ventilation

Process containment, enclosure, and/or exhaust ventilation designed to maintain average concentrations of airborne radioactive materials, under normal conditions, below the DAC are provided. Should failure or degradation of process ventilation occur whereby average concentrations greater than the DAC are experienced for seven days or more, investigation and corrective actions are initiated.

4.6.2.1 Process Area Containment Enclosures

The design criteria for inward air flow through the open face of a containment enclosure in a process area, used to handle radioactive material which has a propensity to suspend in air, shall be at least 125 (+/-25) linear feet per minute (LFM). For operations, the inward air flow through the open face of containment enclosures, used to process radioactive material which has a propensity to suspend in air, shall be at least 100 (+/- 20) LFM, except for the following.

- Openings used to transfer containerized material or equipment.
- Enclosures designed to facilitate surface contamination control rather than provide airborne radioactivity containment.
- Hoods and dryboxes where low radiotoxicity materials (radioactive material with a specific activity <2.4 uCi/g) are handled.
- Open face enclosures where excessive air flow interferes with sensitive analytical equipment or process operations.

The minimum rate of flow into these hoods shall be established by internal procedures.

Air flow measurement checks are performed at least monthly on containment enclosures to ensure compliance with these requirements. In addition, air flow measurements will be performed after significant modifications or changes to the ventilation system to ensure compliance.

Devices are provided to measure the differential pressure within a containment enclosure with respect to the outside atmosphere, except in containment enclosures where the nature of an operation makes this requirement impractical for processing purposes.

Minimum differential pressure control levels are 0.5 inches water negative for high-enriched uranium, and 0.25 inches water negative for low-enriched uranium systems. These differential pressures are checked when used to ensure compliance with these requirements.

Inert atmosphere or positive pressure boxes are maintained at pressures not to exceed 1.0 inch of water positive. These enclosures are also provided with over pressurization protection. Process air (air inside a containment enclosure) that is routinely discharged to the room air is HEPA filtered and sampled via the airborne radioactivity monitoring program.

4.6.2.2 Laboratory Area Containment Enclosures

The design criteria for inward air flow through the open face of a containment enclosure in laboratory areas, used to handle radioactive material which has a propensity to suspend in air, shall be in accordance with ANSI/AIHA Z9.5-2003 recommendations. NFS will determine the proper average face velocity for the containment enclosure.

Any ventilated containment with an open door or port through which uncontainerized radioactive material is routinely handled is subject to these requirements (however, the intermittent opening of a door, glove port, etc. for the sole purpose of adding or removing containerized material or equipment does not constitute handling radioactive material with a propensity to suspend in air). In addition, any ventilated containment with an opening to the room which is high efficiency particulate air (HEPA) filtered for exhaust or over-pressurization protection is excluded from inward air flow requirements.

4.6.3 Filtration System Specifications

Exhaust systems where dry material is processed with potentially contaminated airborne effluents are either equipped with HEPA filter media (selected to maintain integrity when subjected to chemicals and solvents in the processes) or scrubber/demister. These systems will meet the effluent requirements of SNM-124, Chapter 9, and the nuclear criticality safety requirements of SNM-124,

Chapter 5. The HEPA filters are rated at least 99.97% efficient for removal of 0.3 micron particles and have a fire resistant rating of UL 586. All HEPA filters (both primary and secondary) in the exhaust system are equipped with a device for measuring differential pressure.

HEPA filter integrity is evaluated when the differential pressure across the filter exceeds four inches of water. A HEPA filter is replaced following evidence of the inability of the filter or the exhaust system to perform its function properly. In no case will filters continue to be operated at differential pressure values which exceed the manufacturer's rating for the filter. These pressures are checked by personnel prior to each use.

4.6.4 Respiratory Protection Program

The NFS Respiratory Protection Program was developed utilizing the regulatory requirements provided in 10 CFR 20, Subpart H, "Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas." This program was started on July 7, 1987.

The program's primary objective is to prevent or mitigate the hazardous condition at the source, where feasible, through engineered controls such that respiratory protection is not necessary. The program specifically delineates responsibility, use conditions, and guidelines for limitations on work periods.

Typical respiratory protection equipment used at NFS for protection from internal exposure is summarized in Table 4-1.

4.6.4.1 User Qualification

Prior to initial use, and on an annual basis, potential respirator users are qualified. Qualification includes:

- Medical Evaluation – The plant's medical staff reviews the medical status of each individual to determine if he/she is physically able to perform the work and use respiratory protective equipment.

- Initial or Requalification Training – All potential respirator users are given detailed training on aspects of the respiratory protection program commensurate with their respirator use potential. Testing is used to assure the effectiveness of this training.

- Fit Testing – Individuals must successfully qualify on each type of respirator mask he/she may potentially use. The Respirator Facility has dedicated equipment for qualitative and/or quantitative mask fit testing.

Table 4-1
Respiratory Protection Equipment at NFS-Erwin[1]

Type	Model	Mode	Available Cartridges /Canisters	Protection Factor[2]	Comments
Air Purifying					
Full face mask	MSA Ultravue	NP	Magenta	100	Radioactive particulates
		NP	Olive		Ammonia, chlorine, acid gases, organic vapors
	MSA PAPR	PP	Magenta	1,000	Radioactive particulates
Fully Encapsulated Suit	BLU Suit	PP	Magenta	2,000	Radioactive Particulates
Half face mask	MSA 200LS	NP	Magenta	10	Radioactive Particulates
		NP	Olive		Ammonia, chlorine, acid gases, organic vapors
Supplied Air (Combination Respirator)					
Full face mask	MSA Constant-Flo	PP/CF	N/A	1,000	Low pressure air line respirators
	MSA Dual-flow	PP /NP	N/A /Magenta	1,000 /100	Low pressure air line respirators
Self-Contained Breathing Apparatus					
SCBA	MSA/Scott	PD	N/A	10,000	

CF – Continuous Flow
NP – Negative Pressure
PD – Positive Pressure, Pressure Demand
N/A – Not Applicable
PP – Positive Pressure

1 While this listing is representative, it is not all inclusive. Also, upon industry development, these devices may be upgraded or replaced with other equipment having comparable or superior operating characteristics.

2 Applicable for those respirator wearers with current qualifications.

4.6.4.2 Testing and Cleaning of Equipment

Respirators may be reused by the same individual multiple times during a single wear period (work shift).

Used respirators are deposited in designated receptacles after the final use. Each respirator is processed for cleaning, inspection, and replacement of parts as necessary. Air-purifying cartridges and canisters are challenge-atmosphere and pressure-differential tested according to internal procedures if reused beyond a wear period.

Self-contained breathing devices are inspected for operational capability and are cleaned and reinspected after each use. Oxygen or breathing air cylinders are refilled by an outside service contractor.

4.6.4.3 Respiratory Protection Procedures

Written operational and administrative procedures give program details on the following subjects:

- Responsibilities
- Proper selection and issuing of respiratory equipment
- Use of respiratory equipment
- Cleaning and sanitizing respiratory equipment
- Contamination checks, inspection, maintenance, recertification, and storage
- Medical qualification
- Fit testing
- Records of the Respiratory Protection Program (including training for respirator use and maintenance)
- Respiratory Protection Program audit

All respiratory protection equipment procedures will be reviewed and revised, as necessary, to address processing, facility, or equipment changes.

4.7 Radiological Surveys and Monitoring

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

4.7.1 Monitoring of the Work Place

4.7.1.1 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. 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 radiologically controlled areas.
- Routine exposure rate surveys performed at specified locations and frequencies.
- Continuous work station air sampling at fixed locations.

4.7.1.2 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. The operational work place monitoring program includes, where appropriate:

- Operational contamination surveys required to adequately assess conditions during a special or non-routine 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.

4.7.1.3 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 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. 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.

4.7.2 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. NFS will sum external and internal exposures consistent with the requirements of 10 CFR 20.1202 and through procedures consistent with Regulatory Guides 8.7 (2005) or 8.34 (1992).

4.7.2.1 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. 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.

4.7.2.2 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. The operational monitoring program for individuals includes:

- Utilizing breathing zone air samplers to assess intake potential for individuals working on non-routine 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 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.

4.7.2.3 Special Monitoring

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

- Diagnostic 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.

4.7.3 Environmental Monitoring

Environmental monitoring is addressed in Chapter 9.

4.7.4 Radiation Exposure Control

4.7.4.1 Administrative Action Levels

Administrative action levels are established to assure that the occupational exposure of NFS employees is kept as low as reasonably achievable (ALARA) and within the limits established in 10 CFR 20.1201. These levels are established by NFS management and maintained by the radiation protection function and are documented in accordance with procedures covering the specific type of analysis or monitoring system. General guidelines are given in this section, while more detailed information may be found in the appropriate written procedures.

4.7.4.2 Personnel Exposure Guidelines

The philosophical basis and technical approach to ensure radiation worker exposures are ALARA is provided by internal procedures and manuals. Specific actions implemented at NFS to evaluate the significance of an exposure to radiation or radioactive materials and provide appropriate follow-up to prevent recurrence are given in Table 4-2. Action levels for internal exposure, and external dose equivalent, including whole body, skin, and extremities are also given.

Table 4-2
Administrative Action Levels – Personnel Exposure Control

External Dose Equivalent		Action
Whole body	0.5 rem/qtr.	Investigate cause and recommend corrective actions to prevent recurrence
	1.0 rem/qtr.	Restriction pending result of investigation and action to prevent recurrence
Lens of Eye	1.5 rem/qtr.	Investigate cause and recommend corrective actions to prevent recurrence
	3.0 rem/qtr.	Restriction pending result of investigation and action to prevent recurrence
External Dose Equivalent		**Action**
Extremities	5.0 rem/yr.	Investigate cause and recommend corrective actions
	12.5 rem/yr.	Restriction pending result of investigation and action to prevent recurrence
Skin	5.0 rem/yr.	Investigate cause and recommend corrective actions
	12.5 rem/yr.	Restriction pending result of investigation and action to prevent recurrence
Declared Pregnant Worker	0.050 rem/ month	Investigate cause and recommend corrective action
Visitor/Member of Public*	0.010 rem/ year	Investigate cause and recommend corrective action
Internal Exposure		**Action**
Airborne – Any result which shows potential exposure > 40 DAC-hrs.		Initiate confirmatory bioassay; determine individuals potentially exposed and evaluate work history for total intake; and investigate as to cause and recommend corrective actions.
Airborne – Any result which shows potential exposure > 200 DAC-hrs.		Take action indicated above. Establish work restriction pending intake assessment; perform detailed exposure evaluation utilizing bioassay methodology which provides the greatest measurement sensitivity interpreted with applicable metabolic models.
URINALYSIS and/or FECAL ANALYSIS – Any positive result which shows potential exposure > 40 DAC-hrs.		Confirm result where possible; determine if other workers were involved; initiate follow-up bioassay and evaluate work history for total intake; review air sampling data for representativeness; investigate as to cause and recommend corrective actions.

URINALYSIS and/or FECAL ANALYSIS – Any positive result which shows potential exposure > 200 DAC-hrs.	Take action indicated above. Establish work restriction pending intake assessment; perform detailed exposure evaluation utilizing bioassay methodology which provides the greatest measurement sensitivity interpreted with applicable metabolic models.
Internal Exposure	**Action**
IN-VIVO LUNG COUNT – Any positive result > 40 DAC-hrs. above previously evaluated result (known lung burden)	Confirm result where possible; determine if other workers were involved; initiate follow-up bioassay and evaluate work history for total intake; review air sampling data for representativeness; investigate as to cause and recommend corrective actions.
IN-VIVO LUNG COUNT – Any positive result > 200 DAC-hrs. above previously evaluated result (known lung burden)	Take action indicated above. Establish work restriction pending intake assessment; perform detailed exposure evaluation utilizing bioassay methodology which provides the greatest measurement sensitivity interpreted with applicable metabolic models.
NOTE: 40 DAC-hrs. = 0.1 rem exposure, and 200 DAC-hrs. = 0.5 rem exposure * With respect to visitors, this action guide applies only to those individuals who have not received formal training in accordance with 10 CFR Part 19.12, "Instructions to Workers."	

4.7.5 Internal Radiation Exposure - Personnel Monitoring Program

4.7.5.1 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, committed effective dose equivalent [CEDE]). Worker participation in the NFS internal radiation monitoring program follows the guidelines as set forth in Regulatory Guidance document 8.9 (1993) and 8.34 (1992). These requirements are also listed in section 4.7.5.3 of the license application.

To accomplish this objective, monitoring of both the working environment and workers is required. Breathing zone air samplers and/or 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 measurement 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.

NFS' routine bioassay program includes urinalysis and in vivo counting. The special and/or diagnostic bioassay program includes, in addition, fecal analysis, nasal smears, sputum samples, etc., as appropriate for the exposure conditions under investigation. Worker participation in the program is primarily dependent on their potential for exposure, and does not differentiate between employees and others. Bioassay frequencies at a minimum will be established in accordance with Table 1 of Regulatory Guide 8.34 and guidance given in Regulatory Guide 8.9.

The routine frequency for the collection and analysis of urine samples to measure intakes of uranium by individuals who could be exposed to highly soluble compounds of uranium with specific activity less than or equal to 2.4 µCi/gU is at least twice a month, with a maximum interval between sampling not to exceed 20 days. In addition, the action level for investigation, intake assessment, and follow-up sampling is 20 micrograms uranium per liter of urine, or less.

Actions based on results will be, at a minimum, those specified in Regulatory Guide 8.9.

A quality assurance program for in vitro and in vivo measurements performed by a vendor and by NFS is in place.

4.7.5.2 Capabilities

On-site capability exists in dedicated facilities for the analysis of urine samples, and nasal smears. An on-site in vivo chest counter was installed and operational in 1987. Contract laboratories are currently 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.

Natural dietary intake of uranium for the NFS population has been determined to result in an excretion rate as specified in internal NFS documents. Any result in excess of this value is considered a positive result. NFS will periodically evaluate this baseline excretion rate of a representative population as determined using methods for bioassay analysis available on plant site.

Typical minimum detectable amounts are listed in Table 4-3:

Table 4-3
Typical Bioassay Minimum Detectable Amounts

	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.

(3) Dosimetry based on the more sensitive urine or fecal analysis.

4.7.5.3 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 4.7.2.2. Special bioassays are collected or in vivo measurements made to adequately assess intakes as outlined in Sections 4.7.2.3, 4.7.5.1, and 4.7.9.1.

4.7.5.4 Uranium Chemical Toxicity

When individuals may have been exposed to soluble compounds (Class D/F) of uranium with specific activity less than 2.4μCi/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 4-4 apply:

Table 4-4: Exposure Action Levels

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)

4.7.5.5 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 for workers whose annual intakes must be monitored under 10 CFR 20.1502(b) because intakes are likely to exceed 10% of an annual limit on intake (ALI) and whose dose of record will be based primarily on air sampling are periodically used to verify the validity of the work place air monitoring program and the effectiveness of the respiratory protection program. The ratio of the sum of the intakes calculated from air sampling divided by the sum of the intakes calculated from bioassay measurements should exceed 0.7 when averaged for all workers included in the comparison. The ratio for each individual worker should exceed 0.5 for each individual worker, as specified in Regulatory Guide 8.25. Respirator use protection factors are applied as appropriate. This program is separate from the other validity checks on the air sampling program discussed in this chapter.

4.7.6 External Radiation Exposure - Personnel Monitoring Program

Dosimetry devices, provided and processed by a NVLAP 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 (SRDs) may be used in specific areas as an ALARA tool. Worker participation in the NFS external radiation monitoring program follows the guidelines as set forth in Regulatory Guidance document 8.34 (1992).

Individual dose monitoring is provided based upon the radiation protection function evaluation of the individual's potential for exposure. Beta/gamma-sensitive dosimetry is provided for individual monitoring and is exchanged at specified frequencies. The range of these monitoring devices is typically 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.

4.7.7 Work-Area Air Sampling

4.7.7.1 Airborne Radioactivity in Work Areas

The control of radioactive materials in Restricted Areas is affected 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 DACs 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 ALARA concept.

4.7.7.2 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 site to monitor work area exposures/ concentrations and to detect unsafe concentrations.

Air monitoring systems are calibrated in accordance with manufacturer's recommendations using the guidance found in NRC Regulatory Guides 8.21 and 8.24, dated 1979.

4.7.7.3 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.

Stationary air sample collection frequencies are established in written procedures. Each air sampler consists of a particulate filter and a rotometer so that the volume of air sampled can be determined. These rotometers are calibrated or replaced annually.

Stationary air samples in areas where annual intakes are likely to exceed 10% of the annual limit on intake (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.

Guidelines are given in Table 4-5 for response levels and actions for the various air monitoring systems used at NFS.

Table 4-5
Air Sampling System/Response Levels and Actions

Sample Type	Action Level	Action Taken
Stationary Air Samplers*	Individual samples greater than DAC during any shift (i.e., >1 DAC-hr. for each hr. of the shift)	Investigate cause. Consider additional evaluation of personnel exposure.
Breathing Zone Air* Samplers (Lapels)	Individual samples equal to or greater than 8 DAC-hrs.	Investigate cause. Consider additional evaluation of personnel exposure.
High-Volume Samplers*	≥ Derived Air Concentration (DAC)	Notify area supervision and require respirator use.
Continuous Air Monitors (CAMs)*	Alarm at a maximum of 40 DAC-hrs. in a day	Investigate cause. Consider additional evaluation of personnel exposure, or changes to personal protective equipment (PPE).

*May be corrected for decay and respiratory protection.

The stationary air sampling analytical system must have a detection limit of at least 0.3 DAC.

The routine survey data and individual personnel exposure assignments are monitored to evaluate the effectiveness of the radiological controls.

Stationary air sampling of work areas for airborne alpha radioactivity is based on guidance provided in Regulatory Guide 8.25, dated June, 1992. If it is likely that a worker intake could exceed 0.1 times the ALI (Annual Limit of Intake) value and stationary air sampling is used as the primary means to assign the intake of record, then such sampling must be shown to be representative.

Demonstration that stationary samples are representative is performed in accordance with written procedures that are based on Regulatory Guide 8.25

(June 1992). When, for various reasons, the stationary air samplers cannot be made representative, other appropriate forms of work-area monitoring must be provided.

The airborne concentration of radioactivity at each sampling location is estimated in a timely manner after each sampling period in order to detect an unexpected release of radioactive materials.

4.7.7.4 Breathing Zone Air Sampling

Breathing Zone Air Samplers (BZA), sometimes called lapel samplers, are used in the verification program of the stationary air samplers and to monitor personnel exposure to airborne radioactivity. BZAs are worn by operators while working at a station. The results are then used to assure adequate representation is provided by the stationary air samplers.

BZAs may also be used to augment the stationary air sampling program or for personnel monitoring purposes. When BZAs are used to monitor personal internal exposure to airborne radioactivity, the filters of the BZAs are collected each shift and analyzed for radioactivity.

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, or submit to diagnostic bioassay as appropriate for the exposure conditions under investigation. (see Section 4.7.7.3 for action levels).

4.7.7.5 Continuous Air Monitors (CAMs)

Continuous Air Monitors may be positioned in various plant areas, as deemed necessary by the radiation protection function, to identify airborne problems as they occur. These instruments are equipped with a particulate filter and solid-state detector. The instruments are also equipped with a local and/or remote alarm. When in use, the alarm is set to sound in situations where there is a potential for accidents to cause intakes exceeding 40 DAC-hours in a day. Air sample filter media is replaced as needed. 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. Written procedures are provided for proper response to these alarms.

4.7.7.6 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 used for recommending respiratory protection, evacuation or other necessary protective measures.

4.7.7.7 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.

4.7.7.8 Action Levels

Action levels for various air sampling systems are provided in Section 4.7.7.3. Reports are also part of the actions initiated by elevated sampling data.

Because airborne radioactivity can be a significant source of radiation exposure at the NFS site, 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 documented.

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.

An indication 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. Determination of the number and identity of personnel who may have been exposed.
- Posting of the room, area, or building with signs indicating the need for respiratory protection equipment, as appropriate.
- Investigation to determine the sources of airborne radioactivity.
- Initiation of appropriate corrective action to control further releases of radioactivity.

Routine operations are suspended if the airborne radioactivity concentration at the work station exceeds 100 times the DAC. Corrective action is initiated and documented for routine operations. Corrective actions will be implemented through the NFS formal corrective action program and subject to tracking and auditing. Non-routine operations performed under an SWP requiring respiratory protection could continue if adequate measures are in place and approved by the area health physicist on a case-by-case basis.

4.7.8 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. Two types of restrictions are utilized:

- 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 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 (≤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.

4.7.9 Radiation Exposure Assessment

4.7.9.1 Internal Exposure Assessment

Procedures have been established which address internal exposure monitoring assessments, investigation, action, recording, and other reference levels with respect to NRC exposure requirements. Internal radiation exposure control methods are selected to ensure that significant exposures are prevented. Assessment methods are designed to ensure exposures are detected, properly investigated, and recorded. This requires monitoring of both the working environment and the workers. NFS recognizes that neither bioassay nor air sampling and analysis are mutually exclusive; both may be required for an accurate assessment of internal radiation exposures and doses. When bioassay procedures do not have the sensitivity that is required for detecting a particular reference or control level of interest, then other measurement systems (e.g., Stationary or Breathing Zone Air Samplers) will be utilized to estimate intakes and internal radiation doses of workers.

The NFS internal exposure assessment program for bioassay data 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.

Bioassay result interpretation and internal dose assessments are conducted in accordance with written procedures. The methods employed are consistent with requirements in 10 CFR Part 20.1204 and NRC Regulatory Guides 8.9 and 8.34.

The concentrations of airborne radioactivity may be assessed, for the purpose of assigning effective doses to workers, using DAC/ALI values for an aerosol particle size of 5 microns specified in ICRP 68 in lieu of those contained in 10 CFR 20, Appendix B.

As allowed by 10 CFR 20.1204[c], when specific information on the physical and biochemical properties of the radionuclides taken into the body or the behavior of the material in an individual is known (e.g., lung solubility classifications or aerosol particle size distribution), this information may be used instead of the methods cited above to adjust the DAC and ALI, determine intakes and subsequently committed effective dose equivalent. If individual or material-

specific information is used, that information is documented in the individual's record. Additionally, if NFS chooses to assess intakes of Class Y/S material, NFS may delay the recording and reporting of the assessments for a period up to 7 months.

4.7.9.2 External Exposure Assessment

Where required by 10 CFR Part 20, all personnel, including employees, contractors and visitors wear a personnel monitoring device and a badge containing a strip of indium foil when entering plant Restricted Areas. Personnel exposure will be analyzed and evaluated on a periodicity commensurate with exposure potential, using the monitoring device. The indium foil will be evaluated in the event of an emergency. With regard to accident dosimetry, fixed accident dosimeters are provided in selected areas throughout the facilities.

Exposure results are monitored and evaluated by the radiation 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.

On a periodic basis, each routinely occupied work station within any facility handling, processing, or storing significant quantities of licensed material is surveyed for radiation levels. Minimum survey frequencies are given in Table 4-6. Normally, this survey is performed for gamma radiation. However, where significant beta radiation may be present, the radiation levels of beta activity are measured. The results of surveys are documented. Significant differences in exposure potential as measured by the personnel monitoring device of record (TLD) and calculated from radiological surveillance data are investigated by the safety function; appropriate corrective actions are taken based on the results of the investigation. Where available, self reading dosimeters (SRDs) can be used for this purpose.

4.7.9.3 Declared Pregnant Worker and Dose to Embryo Fetus

Procedures have been established to address a declared pregnancy and dose management to the embryo fetus. These procedures include exposure limitations to maintain dose ALARA, counseling by a member of the radiation protection function, and opportunities for work re-assignment. Radiation dose to the embryo fetus will be controlled and calculated in accordance with the guidance in Regulatory Guide 8.36 "Radiation Dose to the Embryo Fetus."

4.7.10 Posting and Labeling

NFS is granted an exemption from the radioactive material labeling requirements of 10 CFR 20.1904(a). Instead, each entrance into a Restricted Area shall be posted:

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

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

Areas are posted for specific radiological hazards including Radiation, High Radiation, Very High Radiation, and Airborne Radioactivity, and other hazards as appropriate.

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

4.7.11 Contamination Control Program

Access to the Restricted Area at NFS is controlled. The Restricted Area includes the Northsite Remediation Project and plant Protected Area which encompasses manufacturing operations as well as radioactive material storage.

Within the Restricted Area are clean (uncontrolled) areas and potentially contaminated Radiologically Controlled Areas (RCAs). Contamination control is implemented through classification of areas, use of barriers, radiological postings, routine surveillance and monitoring, protective clothing, and training.

4.7.11.1 Area Classification

Classification of areas within the plant Restricted Area and the internal action guidelines applied 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 "radiologically controlled," and are defined in Chapter 1. RCAs may be further subdivided into special controlled areas,

contamination areas, buffer zones, step off pads, etc., where appropriate. Typical areas where RCAs are frequently established are presented in Figure 4-1.

Radiological postings inform workers of radiological conditions and requirements for entry/exit. Training and qualification of workers, including site orientation, general employee, radiation worker, radiation safety technician, and specialty training, is provided commensurate with the hazard and planned activities. Routine contamination surveys are conducted on a scheduled basis in accordance with Table 4-6, to establish trends and identify off-normal conditions.

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 regulatory limits.

Acceptable levels and decontamination actions are established by approved procedures. To comply with these action guidelines, NFS has a protective clothing (anti-contamination clothing) program and a program for monitoring area contamination levels and personnel contamination.

Protective clothing requirements for a specific area or operation are determined by the radiation protection function. Available clothing includes items such as caps, hoods, laboratory coats and coveralls, safety shoes, shoe covers, gloves, sleeve protectors, safety glasses and goggles, and respiratory protection equipment, as appropriate.

Where practical, change rooms provide an area to change from "street clothing" into protective clothing before working in a RCA. Change rooms are used to accommodate the protective clothing and street clothing storage.

Used protective clothing is doffed or surveyed at RCA boundaries to prevent the spread of contamination. Laundered protective clothing is periodically surveyed to verify the effectiveness of laundering practices.

Located at or near the entrance/exit of RCAs are monitoring devices for personnel contamination detection. Upon leaving a RCA, all persons shall survey for contamination. Procedures state various levels of acceptable contamination and the associated response actions.

Table 4-6
MINIMUM SURVEY FREQUENCIES*

AREA	SURFACE CONTAMINATION		RADIATION (GAMMA)
	(REMOVABLE)	(FIXED)	
Uranium RCAs	Weekly	As needed**	Semi-annually
Plutonium RCAs	Daily	As needed**	Quarterly
Shipping, Receiving, Warehousing	Monthly	As needed**	Semi-annually
Chemical Metallurgical Lab	Weekly	As needed**	Annually
Lunchroom/Break Areas	Monthly	Annually	n/a
Administrative (Process Support)	Monthly	Annually	n/a
Administrative (Other)	Semi-annually	n/a	n/a
Outside Areas (Process Support)	Weekly	Semi-annually	Semi-annually
Outside Areas (Other)	Semi-annually	n/a	n/a
Non-nuclear Miscellaneous Facilities	Annually	n/a	n/a
RCA Personnel Exits	Daily	Quarterly	Semi-annually

* These frequencies may be reduced for buildings which are not in operation.

** Fixed surface contamination surveys are performed in RCAs as investigative only.

Figure 4-1: Radiologically Controlled Areas

This drawing is "Official Use Only" and has been moved to the "Sensitive Information" ADDENDUM.

4.7.11.2 Surface Contamination Monitoring

Routine surface contamination monitoring is performed for process and manufacturing areas, warehousing, and support facilities. Uncontrolled areas inside the plant are also surveyed periodically to ensure that radioactive materials are adequately confined in the RCAs. Removable contamination surveys are utilized primarily to assess contamination levels and the potential for transfer to uncontrolled areas; however, fixed contamination measurements may also be made for information purposes.

The frequency and type of routine surveys depends on the nature of the work being conducted, the quantities and physical characteristics of material being processed, and the specific facilities, equipment, and procedures used to protect the worker from intake. Minimum survey frequencies are as shown in Table 4-6. The radiation protection function will determine the need for a greater frequency of surveys from review of contamination trends. Survey frequencies may be increased or decreased based on contamination levels detected in accordance with criteria established by the licensee.

Survey results are compared to action guidelines as specified in internal procedures.

4.7.11.3 Action Guidelines

Action guidelines 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."

General guidelines for surface contamination are outlined in the following Table 4-7. Decontamination or access restriction is the action typically taken when the values in this table are exceeded.

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

Response is based on the need to avoid transfer of contamination to uncontrolled areas and to maintain exposures ALARA. Timeliness of the response is based on the above considerations and is set by internal procedures.

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 from resuspension is high, are surveyed more frequently.

Table 4-7
Surface Contamination Action Guidelines

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

4.7.11.4 Contamination 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 as a gross indicator of contamination on an object or in an area.

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. Measurements are recorded in units of dpm per area of surface surveyed or dpm per wipe for large area wipes.

Measurements of total (fixed) 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 RCAs and, therefore, are not routinely surveyed. Special controlled areas, diked areas, drip pans, and other containment devices open to room air, are limited to traffic access and create less potential for transfer or resuspension; therefore, less restrictive surface contamination action guidelines may be established for these areas. These areas are surveyed periodically for

removable contamination with acceptable levels, decontamination actions, and survey frequencies set by internal procedures.

4.7.11.5 Area Contamination Control Practices

The contamination buildup within RCAs 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 guidelines. The responsible party then initiates action to assure timely decontamination. Such action is documented on the survey form.

Periodically a qualified member of the radiation protection function reviews the contamination surveys for trends, problem areas, timely decontamination, etc. He/she identifies to area management those locations considered to be a problem.

A monthly summary of surface contamination results is prepared, reviewed by the manager of the radiation protection function, and distributed to plant management.

4.7.11.6 Personnel Contamination Control Guidance

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

- All personnel wear protective clothing, as appropriate (anti-contamination clothing), as directed by internal procedures while in RCAs. This may include coveralls, laboratory coats, gloves, hoods, shoe covers, or booties, as appropriate.

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

- All personnel survey for contamination at designated locations when exiting RCAs. If the levels in Table 4-8 are exceeded, decontamination is performed. If protective clothing is suspected of being contaminated, the affected areas are also monitored. Corrective actions will be implemented through the NFS formal corrective action program and subject to tracking and auditing. Additional actions are specified in Table 4-8.

- 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 internal procedures.

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

Table 4-8
Personnel Survey Action Levels

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 Safety Department 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	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 detector used.

4.7.11.7 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, vehicles, scrap, or waste from RCAs to uncontrolled areas or for unrestricted release.

Unrestricted release of potentially contaminated equipment and material from the plant site or to uncontrolled areas shall be in accordance with the "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material," April, 1993, (included as Appendix 4A).

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 may be disposed of as radioactive waste or limited to use within the RCAs. If the value of the item or the need to remove the item from the RCA is very great, then a conditional release may be granted under very strict control conditions designed to prevent the spread of contamination or the exposure of personnel. These conditions are set by internal procedures.

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

4.7.12 Radioactivity Measurement Instruments

4.7.12.1 Equipment Description

An adequate number of radiation detection instruments are 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 protection function reviews the types of instruments being used for each monitoring purpose and makes appropriate recommendations based upon regular input and ongoing evaluation.

4.7.12.2 Instrument Types

Table 4-9 summarizes the typical radiation detection instruments employed at NFS. Table 4-10 provides typical types & instrument uses. 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.

4.7.12.3 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, RCA 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 protection purposes are calibrated before initial use, after major maintenance, and on a routine basis in accordance with manufacturer's recommendation following the last calibration using the guidance in NRC Regulatory Guides 8.21 and 8.24 dated 1979, as well as American National Standards Institute recommendations found in ANSI N323A-1997, N323B-2003,.and N323D-2002.

Table 4-9
Typical Radiation Detection Instruments/Systems Used at NFS

Fixed Installation Equipment	Model
Criticality warning system (GM type)	Eberline RMS
Continuous Air Monitor	Canberra Alpha Sentry
Area Radiation Monitor	Eberline RMS-3, Eberline ECX-4 Ludlum Model 375
Fixed Installation Equipment	**Model**
Kinetic Phosphorescence Analyzer for Uranium Urinalysis	Chemchek Instruments KPA-11A
In vivo lung counter (Canberra Industries)	Canberra Industries Custom System
Hand and Foot Monitor	Aptec Personal Monitor, Alpha 7
Personal Contamination Monitors	Eberline PCM-2 Personal Monitor
Portable Contamination Instrumentation	**Model**
Alpha survey/contamination meter	Ludlum 3, 4, 2221, or 2224 with either 43-5 or 43-90 probes
Beta - Gamma Contamination Survey Meter	Eberline RM-14, 19, or 25 with GM pancake probes
Personnel monitoring (scintillation, gas-flow proportional, GM type instruments) Friskers	Eberline RM-19, Eberline RM-20, or Eberline RM-25, or Ludlum Model 177 with 43-5 probe (Alpha Monitoring) or GM pancake probe (Beta-Gamma Monitoring)
Portable Exposure Rate Instrumentation	**Model**
Beta/Gamma (GM-type) Meter	Eberline E-520 with HP-270 probe, Ludlum 78, Ludlum 3 with 44-38 probe
Beta/Gamma (ionization chamber type) meter	Eberline RO-2, RO-2A, or RO-20, Ludlum Model 9, Victoreen Model 451P-RYR
Gamma (pressurized ion chamber)	Eberline PIC-6A or Pic-6B
Neutron Counter	Eberline E600 or ASP2e with BF_3 sphere, Ludlum 12-4 with He-3 sphere
Gamma Self-Reading Dosimeter	Rados RAD-60R
Gamma (scintillation type) meter	Ludlum 2350 with NI detector
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

Table 4-10
Types and Uses of Available Instruments (Typical)

TYPE	TYPICAL RANGE	USE
Dose Rate Meters		
GM Low Range	0.01 mR/hr-200 mR/hr	Area Exposure Rate Survey (Beta/Gamma)
GM High Range	0.05 mR/hr-2000 mR/hr	Emergency Monitoring (Beta/Gamma)
Ion Chamber	1 mR/hr-5000 mR/hr	Emergency Monitoring or Area Dose Rate Survey
Ion Chamber	1 mR/hr-1000 R/hr	Emergency Surveying
Alpha Survey Meters	50 cpm-5x10^5 cpm	Direct Personnel & Equipment Surveys
Beta Survey Meters	0.1 mR/hr-5000 mR/hr	Direct Personnel & Equipment Surveys
Laboratory Instrumentation		
Automatic air sample counter	N/A	Lab Analysis
Windowless gas-flow proportional counter manual operation	N/A	Lab Analysis
Window gas-flow proportional counter automatic operation	N/A	Lab Analysis
In Vivo Lung Counter	N/A	Lung Deposition Measurements

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 analog instruments 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. Digital instruments require a one point calibration. 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.

Instrument calibration details are defined further in approved written procedures.

4.7.12.4 Criticality Detection System

The NFS criticality detection system is consistent with the requirements of 10 CFR 70.24. Monitoring is performed with ionization chamber and/or GM detector systems. The criticality alarm system meets the guidance established in ANSI/ANS 8.3 "Criticality Accident Alarm Systems," (with the exceptions cited in NRC Regulatory Guide 3.71).

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 for all units in service on a semi-annual basis. Detector pairs are also response tested in accordance with internal procedures to ensure continued operability. 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 discipline.

The placement of criticality detectors is such that all areas of the plant where monitoring is required will be covered. Typically, the alarm trip point is set at 20 mR/hr. Higher alarm set points may be necessary due to ambient radiation levels. This trip point allows for minimization of an alarm from sources other than criticality. When the alarm trip point has been reached or exceeded, the system will produce an alarm throughout the plant which will continue regardless of the radiation level until manually reset. 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. Alarm system testing is performed in accordance with approved procedures.

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/or 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.

4.8 Additional Program Commitments

4.8.1 Survey and Monitoring Data

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

4.8.2 Records and Reports

Records appropriate to radiation protection 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 4-11 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.

Any incident, in which the resulting dose exceeds either 10 CFR 20.2202 dose limits or reporting requirements per 10 CFR 70.74, will be referred to the corrective action program.

An annual report of the results of individual monitoring, consistent with the requirements of 10 CFR 20.2206(b), will also be submitted to the NRC.

Table 4-11
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
Safety 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

Additional information on records is found in Chapter 11.

4.8.3 Sealed Sources

Sealed sources authorized by this license are subject to leak testing and other actions specified in this section. A physical inventory is conducted every six (6)

months to account for all sealed sources and devices received and possessed. Records of the inventories are maintained for inspection for a minimum of 2 years. All sealed sources will be disposed of by transfer to an authorized recipient or disposal site.

Each sealed source is tested for leakage at intervals not to exceed 6 months. In the absence of a certificate from a transferor indicating that a test has been made within 6 months prior to the transfer, the sealed source will not be put into use until tested. The completed document is maintained for inspection and kept in units of microcuries.

The test must be capable of detecting the presence of 0.005 microcurie of removable contamination on the test sample. The test sample must be taken from the source or from appropriate accessible surfaces of the device in which the sealed source is permanently or semi-permanently mounted or stored.

If the test reveals the presence of 0.005 microcurie or more of removable contamination, the sealed source will immediately be withdrawn from use, decontaminated and repaired by a person appropriately licensed to make such repairs or to be disposed of in accordance with the current regulations. Within 5 days after determining that any source has leaked, a report will be filed with the U.S. Nuclear Regulatory Commission, Washington, DC 20555, describing the source, test results, extent of contamination, apparent or suspected cause of source failure, and corrective action taken. A copy of the report will be sent to the Administrator of the nearest NRC Regional Office listed in Appendix D of Title 10, Code of Federal Regulations, Part 20.

It is not required that a sealed source be surveyed if it contains 100 microcuries or less of beta gamma emitting material or 10 microcuries or less of alpha emitting material. Sources that have been removed from service are not required to be leak tested, but will be leak tested prior to being returned to service if the source has been in storage for more than six months.

APPENDIX 4A

GUIDELINES FOR DECONTAMINATION OF FACILITIES AND EQUIPMENT PRIOR TO RELEASE FOR UNRESTRICTED USE OR TERMINATION OF LICENSES FOR BYPRODUCT, SOURCE, OR SPECIAL NUCLEAR MATERIAL

U.S. Nuclear Regulatory Commission
Division of Fuel Cycle Safety and Safeguards
Washington, DC 20555

April 1993

APPENDIX 4A

GUIDELINES FOR DECONTAMINATION OF FACILITIES AND EQUIPMENT PRIOR TO RELEASE FOR UNRESTRICTED USE OR TERMINATION OF LICENSES FOR BYPRODUCT, SOURCE, OR SPECIAL NUCLEAR MATERIAL

The instructions in this guide, in conjunction with Table 4A-1, specify the radionuclides and radiation exposure rate limits which should be used in decontamination and survey of surfaces or premises and equipment prior to abandonment or release for unrestricted use. The limits in Table 4A-1 do not apply to premises, equipment, or scrap containing induced radioactivity for which the radiological considerations pertinent to their use may be different. The release of such facilities or items from regulatory control is considered on a case-by-case basis.

1. The licensee shall make a reasonable effort to eliminate residual contamination.

2. Radioactivity on equipment or surfaces shall not be covered by paint, plating, or other covering material unless contamination levels, as determined by a survey and documented, are below the limits specified in Table 4A-1 prior to the application of the covering. A reasonable effort must be made to minimize the contamination prior to use of any covering.

3. The radioactivity on the interior surfaces of pipes, drain lines, or ductwork shall be determined by making measurements at all traps, and other appropriate access points, provided that contamination at these locations is likely to be representative of contamination on the interior of the pipes, drain lines, or ductwork. Surfaces of premises, equipment, or scrap which are likely to be contaminated but are of such size, construction, or location as to make the surface inaccessible for purposes of measurement shall be presumed to be contaminated in excess of the limits.

4. Upon request, the Commission may authorize a licensee to relinquish possession or control of premises, equipment, or scrap having surfaces contaminated with materials in excess of the limits specified. This may include, but would not be limited to, special circumstances such as razing or buildings, transfer of premises to another organization continuing work with radioactive materials, or conversion of facilities to a long-term storage or standby status Such requests must:

a. Provide detailed, specific information describing the premises, equipment or scrap, radioactive contaminants, and the nature, extent, and degree of residual surface contamination.

b. Provide a detailed health and safety analysis which reflects that the residual amounts of materials on surface areas, together with other considerations such as prospective use of the premises, equipment, or scrap, are unlikely to result in an unreasonable risk to the health and safety of the public.

5. Prior to release of premises for unrestricted use, the licensee shall make a comprehensive radiation survey which establishes that contamination is within the limits specified in Table 4A-1. A copy of the survey report shall be filed with the U.S. Nuclear Regulatory Commission, Washington, DC 20555, and also the Administrator of the NRC Regional Office having jurisdiction. The report should be filed at least 30 days prior to the planned date of abandonment. The survey report shall:

a. Identify the premises.

b. Show that reasonable effort has been made to eliminate residual contamination.

c. Describe the scope of the survey and general procedures followed.

d. State the findings of the survey in units specified in the instruction.

Following review of the report, the NRC will consider visiting the facilities to confirm the survey.

Table 4A-1: ACCEPTABLE SURFACE CONTAMINATION LEVELS			
NUCLIDES[a]	AVERAGE[bcf]	MAXIMUM[bdf]	REMOVABLE[bcf]
U-nat, U-235, U-238, and associated decay products	5,000 dpm α/100 cm^2	15,000dpm α/100 cm^2	1,000 dpm α/100 cm^2
Transuranics, Ra-226, RA-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1,000 dpm/100 cm^2	3,000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5,000 dpm βγ/100 cm^2	15,000 dpm βγ /100 cm^2	1,000 dpm βγ /100 cm^2

[a] Where surface contamination by both alpha-and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

[b] As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c] Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[d] The maximum contamination level applies to an area of not more than 100 cm^2.

[e] The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

[f] The average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/hr at 1 cm and 1.0 mrad/hr at 1 cm, respectively, measured through not more than 7 milligrams per square centimeter of total absorber.

Attachment 6

SNM-124, Chapter 5

(30 pages to follow)

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 5

NUCLEAR CRITICALITY SAFETY
Table of Contents

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 5

NUCLEAR CRITICALITY SAFETY

5.1 Nuclear Criticality Safety Program Management

5.1.1 Nuclear Criticality Safety Program and Philosophy

NFS provides an adequate and effective nuclear criticality safety (NCS) program, including methodologies and technical practices, to support safe operation of the facilities. Controls and barriers that are designated as Items Relied on for Safety (IROFS) to prevent an inadvertent nuclear criticality are documented in NCS Evaluations and the Integrated Safety Analysis (ISA) Summary as appropriate.

NFS provides for the appropriate management of the NCS program. The responsibilities and authorities of individuals that develop and implement the NCS program are also provided. In addition, facility management measures are provided that support implementation and maintenance of the NCS program.

Subcriticality is maintained for all normal and credible abnormal conditions. To support this overarching requirement, process designs incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible. Sufficient redundancy and diversity should be implemented on changes in one process condition such that at least two unlikely, independent, and concurrent errors, accidents, or equipment malfunctions must occur before a criticality accident is possible. The focus should be on understanding each credible change in process conditions and implementing the best overall controls to maintain subcriticality such that no single credible event or failure will result in a criticality accident. When considering NCS accident sequences, guidance from ANSI/ANS-8.1-1998, Appendix A is used.

NFS relies on passive, active, enhanced administrative, and simple administrative controls to maintain subcriticality. Where practicable, reliance is placed on equipment design in which "favorable" geometry is used rather than on administrative controls.

5.1.2 Management of the Nuclear Criticality Safety Program

NFS provides effective management of the NCS program as well as sufficient resources to implement an effective NCS program. The NFS NCS program includes the following commitments:

1) NFS shall develop, implement, and maintain an NCS program that meets the regulatory requirements of 10 CFR 70.

2) NFS shall establish NCS program objectives (refer to the NCS program objectives below).

3) NFS shall establish and maintain NCS safety limits, controls, and procedures.

4) NFS shall outline an NCS program structure and define the responsibilities and authorities of key program personnel.

5) NFS shall keep the plant configuration current and consistent with the NCS-established safety limits and IROFS by means of the configuration management function.

6) NFS shall use the NCS program to establish and maintain NCS safety limits and NCS operating limits for IROFS in fissile material processes and NFS shall maintain adequate management measures to ensure the availability and reliability of the IROFS.

7) NFS shall prepare NCS postings, provide NCS training, and provide NCS emergency procedure training.

8) NFS shall adhere to the NCS baseline design criteria requirements in 10 CFR 70.64(a).

9) NFS shall use the NCS program to evaluate modifications to operations, to recommend process changes necessary to maintain the safe operation of the facility, and to select appropriate IROFS and management measures.

The objectives of the NFS NCS program include the following:

1) Preventing an inadvertent nuclear criticality;

2) Protecting against the occurrence of an identified accident sequence in the ISA Summary that could lead to an inadvertent nuclear criticality;

3) Complying with the NCS performance requirements of 10 CFR 70.61;

4) Establishing and maintaining NCS safety limits, controls, and procedures;

5) Establishing and maintaining NCS safety limits and NCS operating limits for IROFS;

6) Conducting NCS evaluations to assure that under normal and credible abnormal conditions, all nuclear processes remain subcritical, and maintaining an approved margin of subcriticality for safety;

7) Establishing and maintaining NCS IROFS based on current NCS determinations;

8) Providing training in emergency procedures in response to an inadvertent nuclear criticality;

9) Complying with the NCS baseline design criteria requirements in 10 CFR 70.64(a);

10) Complying with the NCS ISA Summary requirements in 10 CFR 70.65(b);

11) Complying with the NCS ISA Summary change process requirements in 10 CFR 70.72; and

12) Complying with the reporting requirements in 10 CFR 70, Appendix A.

5.2 Organization and Administration

Information regarding general organization and administration is described in Chapter 2. Chapter 2 also includes the organizational positions, functional responsibilities, experience, educational requirements, and authorities of NCS management and staff who develop, organize, implement, and administer the NCS program.

The NCS organization and administration includes the following commitments:

1) NFS shall comply with the requirements of ANSI/ANS-8.1-1998 and ANSI/ANS-8.19-2005 as they relate to organization and administration.

2) NFS shall use personnel, skilled in the interpretation of data pertinent to NCS and familiar with the operation of the facility, as a resource in NCS management decisions. These specialists should be independent of operations supervision (Refer to Section 4.1.1 of ANSI/ANS-8.1-1998).

3) NFS shall provide NCS postings in areas, operations, work stations, and storage locations, as appropriate.

4) NFS personnel shall report defective NCS conditions to the NCS function and perform response/corrective actions only in accordance with written,

approved procedures. Unless a specific procedure deals with the situation, personnel shall report defective NCS conditions to the NCS function and take no action until the NCS function has evaluated the situation and provided recovery directions.

5) NFS shall describe organizational positions, experience and qualifications of personnel, functional responsibilities, and organizational relations among the individual positions (Refer also to Chapter 2).

6) NFS shall designate an NCS function manager who will be responsible for implementation of the NCS program.

7) NFS shall adequately staff the NCS program with suitably-trained personnel and provide sufficient resources for its operation.

5.3 Management Measures

5.3.1 General Management Measures

Information regarding management measures programs is described in Chapter 11. These programs include the management measures identified in 10 CFR 70.62 and are used to implement and maintain the NCS program.

5.3.2 Employee Training

NFS complies with the requirements of ANSI/ANS-8.19-2005 and ANSI/ANS-8.20-1991 as they relate to training. NFS also provides training to all personnel to recognize the Criticality Accident Alarm system (CAAS) signal and to evacuate promptly to a safe area. In addition, NFS employees receive instruction training regarding the NCS Policy.

5.3.3 Training and Qualifications of NCS Staff

A formal training and qualification program is developed and maintained for NCS staff. Elements of the program include the following: on-the-job training, off-site NCS-related training courses, and mentoring by senior NCS engineers.

5.3.4 Auditing, Assessing, and Upgrading the NCS Program

NFS complies with the requirements of ANSI/ANS-8.19-2005 as it relates to audits and assessments. NCS inspections of selected site operations involving special nuclear material are performed weekly by NCS Engineers to determine if activities are being conducted in accordance with nuclear criticality safety requirements and limits.

Quarterly NCS audits of selected plant activities are conducted such that SNM processing or storage facilities are audited biennially. The purpose of the audits is to determine that: (a) site operations are conducted in compliance with license conditions, operating procedures, and posted limits; (b) administrative controls and postings are consistent with NCSEs; (c) equipment and operations comply with NCSEs; and, (d) corrective actions relative to findings of NCS inspections are adequate.

An independent assessment of the nuclear criticality safety program is conducted every three (3) years.

Findings and observations from NCS audits, inspections, and assessments are entered into the corrective action program and tracked until closure. Refer to Chapter 11 for a discussion of the corrective action program.

5.3.5 Procedures

NFS commits to the requirements of ANSI/ANS-8.19-2005 as it relates to procedures and to the policy that no single, inadvertent departure from a procedure could cause an inadvertent nuclear criticality.

Operating procedures are provided for activities involving special nuclear material; and, the procedures incorporate safety limits and controls as appropriate. These procedures are reviewed and approved by the nuclear criticality safety function. During the review and approval process, the NCS staff may recommend or require modifications (to the procedures) to reduce the likelihood of occurrence of an inadvertent nuclear criticality.

5.3.6 NCS Reviews of New or Modified Equipment

Each proposed addition of new equipment or change to existing equipment used in the processing or storage of SNM, and any procedure changes resulting therefrom, are reviewed and approved by the nuclear criticality safety function. During the review and approval process, the NCS staff may recommend or

require modifications (to the design and/or to the procedures) to reduce the likelihood of occurrence of an inadvertent nuclear criticality.

5.3.7 Posting of Nuclear Criticality Safety Limits

Nuclear criticality safety requirements issued by the nuclear criticality safety function for each process system are available at each work area in the form of operating procedures. Clear, visible signs or notices may be posted at work stations, as appropriate, to supplement the procedures by emphasizing specific limits and controls.

Posted nuclear criticality safety requirements are defined by the nuclear criticality safety function and include, as appropriate:

- Limits on material types and forms;
- Allowable quantities by mass or number of items/containers;
- Allowable enrichments;
- Limits on reflecting materials;
- Required spacing between units;
- Control limits (when applicable) on quantities such as moderation, concentration/density and the presence of additives.

5.3.8 Integrated Safety Analysis (ISA) Summary Revisions

Refer to Chapter 3 for a discussion of ISA Summary revisions.

5.3.9 Corrective Action Program

A corrective action program is implemented to document and manage NCS-related problems, observations, findings, investigations, corrective actions, and any unacceptable NCS-related performance deficiencies. Refer to Chapter 11 for a discussion of the corrective action program.

5.3.10 Records Retention

Records of the NCS program are retained in accordance with the regulatory retention program. These records include NCS analyses and documentation of corrective actions taken.

5.4 Criticality Accident Alarm System (CAAS) and Emergency Management

5.4.1 Criticality Accident Alarm System (CAAS)

A criticality accident alarm system (CAAS) is designed and installed to provide prompt detection and annunciation of an inadvertent nuclear criticality. The system satisfies the requirements of 10 CFR 70.24. A documented evaluation will be maintained that demonstrates that the CAAS meets the requirements of 10 CFR 70.24. The system is also designed to remain operational during credible events or the system will alarm during credible failure modes. Refer to Chapter 4 for a discussion of the CAAS.

Exemptions from the CAAS monitoring requirements include the following: 1) materials and/or containers that satisfy the fissile material exceptions in 49 CFR; or 2) materials packaged in authorized shipping containers which are in isolated arrays or on a transport vehicle and which are no more reactive than that approved for transport.

Whenever the criticality alarm system is out of service, or being tested or repaired, compensatory measures are established (e.g., stop movement and/or monitoring of the criticality alarm panel). Periods when the criticality alarm system is out of service are minimized to the extent practical.

Emergency power is provided for the CAAS (e.g., uninterruptible power supply).

5.4.2 Emergency Management

With regard to emergency management, refer to Chapter 8 for a discussion of the emergency management program and emergency plan. With regard to accident dosimetry, refer to Chapter 4. Guidance from ANSI/ANS-8.23-2007 is also used for nuclear criticality accident emergency planning and response.

5.5 Methodologies and Technical Practices

5.5.1 Means of Control

The relative effectiveness and reliability of controls are considered during the nuclear criticality safety analysis process. Engineered controls or design features are preferred over administrative controls. Passive engineered controls or design features are preferred over all other system controls and are utilized when practicable and appropriate. Active engineered controls are the next preferred method of control. Administrative controls are the least preferred method of control; however, when administrative controls are deemed necessary, enhanced administrative controls are preferred over simple administrative controls.

1) Passive engineered controls (most preferred) use fixed design features or devices to maintain safe process conditions. No human intervention or action is required. Assurance is maintained through initial verification prior to operation and/or periodic inspections as appropriate. Assurance is also maintained through the configuration management program.

2) Active engineered controls use add-on, active hardware (e.g., electrical, mechanical) or moving parts to maintain safe process conditions. No human intervention or action is required during operation. Assurance is maintained through initial and periodic inspection, functional testing, and/or calibration as appropriate. Active engineered controls detect an undesirable change in process conditions and automatically secure the system to a safe condition. Active engineered controls are designed to be "fail safe," meaning they are designed to place the system in a safe state due to signal loss or power failure.

3) Enhanced administrative controls rely on human judgment, training, and personal responsibility for implementation and are augmented by warning devices (visual or audible) which requires human action according to procedure. A visual or audible alarm alerts the operator to an undesirable change in process conditions, which requires human action or intervention in accordance with approved procedures to maintain or return the process to a safe condition. Alarm integrity and reliability is ensured by initial and periodic inspection or functional testing as appropriate.

4) Simple administrative controls (least preferred) rely on human judgment, training, and personal responsibility for implementation when the control function is needed. The control is a procedural human action that is required to maintain safe process conditions. Assurance is maintained through periodic verification, audit, or training.

5.5.2 Methods of Control

The following recognized control methods are also referred to as parameters which may be controlled for nuclear criticality safety purposes (i.e., controlled parameters). When evaluating an SNM-bearing system for criticality safety, each of these parameters will be assumed to be at its optimum credible condition (i.e., most reactive credible condition) unless acceptable controls are specified and implemented to limit the parameters to certain values. When computer codes are used to determine the safety of a system, the values meet the k_{eff} limits of this chapter. Criticality safety may also be based on data provided in handbooks, reference documents, experimental data or the values listed in Tables 5A-1 thru 5A-6 of this chapter. The safety factors as presented for Tables 5A-1 thru 5A-6 must be applied to critical values; or, maximum subcritical values may be used as provided in handbooks or standards (e.g., maximum subcritical values provided in ANSI/ANS-8.1).

1. **Geometry** – Geometry control is achieved by increasing neutron leakage by limiting the dimensions of defined geometrical shapes. Equipment relying upon favorable geometry for control include adequate factors of safety to ensure reliability under credible accident conditions. Before beginning an operation, all dimensions relied upon for geometry control are verified. The facility configuration management program is used to maintain these dimensions. Periodic inspections are performed on those systems where credible changes in equipment dimensions may occur that could result in the inability to meet established nuclear criticality safety limits. Standard buckling equations may also be used to determine the geometric limits for finite units.

2. **Spacing (or Unit Interaction)** – Spacing (or Unit Interaction) control is a method of limiting the introduction of neutrons leaked from one SNM unit into a neighboring SNM unit by controlling the separation distance between units. Where spacing control is required, a passive engineered device (e.g., a spacer or bumper) is the preferred method of control and is used where practicable. The structural integrity of any spacers/racks should be sufficient for normal and credible abnormal conditions. If not practicable, administrative controls may be utilized and should include such items as procedural instructions, postings, and visual indicators, as appropriate.

   Equipment, facilities, and individually subcritical units may be considered to be effectively non-interacting or neutronically isolated when their surfaces are separated by any of the following:

- A 12-inch-thick layer of water, or by the distance which is equivalent in isolational ability to a 12-inch-thick layer of water, or
- 12 feet of air, or
- Sub-arrays separated by not less than the smallest dimension of the facing surfaces of the sub-arrays, or
- The greatest distance across an orthographic projection of the largest of the fissile accumulations on a plane perpendicular to the line joining their centers, or
- 12 inches of solid concrete (block or poured) of density greater than or equal to 140 pounds per cubic foot.

The design conditions for interaction between multiple units or between arrays that experience neutron interaction will be based on values that can be demonstrated safe by one of the following methods:

- Unit Storage Criteria
- Solid Angle Method
- Surface Density
- Areal Density – When criticality safety is contingent only upon maintenance of a limited areal density of fissile material, controls will be implemented to ensure that the limit is not exceeded. The controls will limit the areal density to a safe value, which is defined to be no more than 45 percent of the minimum critical areal density.
- Monte-Carlo Calculations (Each application of Monte-Carlo calculations must comply with the requirements of this chapter).
- American National Standard, ANSI/ANS-8.7-1998, "Nuclear Criticality Safety in the Storage of Fissile Materials"
- NRC and/or DOT packaging or transportation regulations (e.g., staging of packages in accordance with the Criticality Safety Index)

With regard to the storage of SNM, NFS complies with the requirements of ANSI/ANS-8.7-1998.

3. **Volume** – Volume control is a method of limiting the volume of SNM to an acceptable value. Equipment relied upon for volume control includes adequate factors of safety to ensure that a safe volume is maintained under credible accident conditions. Prior to the equipment being released for use, the volume of the equipment is verified. The facility configuration management program is used to maintain the volume. Periodic inspections are performed on those systems where credible changes in equipment volume may occur that could result in the inability to meet established nuclear criticality safety limits. When the solution volume is measured (i.e., quantity of solution), appropriate instrumentation is used.

4. **Neutron Absorber (Fixed/Soluble)** – Neutron absorber control is a method of reducing the number of neutrons in a fissile material system available to cause a fission event, by introducing a parasitic neutron absorber (i.e., poison) into the system. This method of control includes use of fixed or soluble neutron absorbers. When evaluating absorber effectiveness, neutron spectra are considered (e.g., cadmium is an effective absorber for thermal neutrons, but ineffective for fast neutrons).

   Fixed neutron absorber control is a method of increasing neutron absorption in material by placing a solid absorber (i.e., poison) in the system that may include the use of "poison fixtures" as well as taking credit for the neutron absorption properties of structural materials. For fixed neutron absorbers, the thickness of the absorber is measured and documented prior to first use. The composition of the absorber will be verified unless the chemical properties of the materials consist of standard structural materials (e.g., stainless steel, carbon steel, etc.). Controls, as necessary, are exercised to maintain the continued presence and the intended distribution and contribution of the absorber. NFS complies with ANSI/ANS-8.21-1995 as it relates to fixed neutron absorbers.

   Borosilicate-glass Raschig rings are used according to the requirements of the American National Standard ANS/ANS-8.5-1996 with the following exceptions:

   - Rough-cut rings may be used. Accumulation of glass fines in the bottom of such vessels is inconsequential, and determining the vessel volume and loss of glass volume due to breakage and settling is unnecessary. Precautionary inspections are performed on an annual frequency to ensure that settling does not result in an accumulation of solution in a ring-free region near the top of the vessel.

- Vessels with open tops may be constructed so as to overflow at 100 percent of free volume, provided the vessel is filled with rings and inspected on an annual frequency.

- Analysis for boron content in a representative sample of rings is performed once every 2 years.

- The glass volume fraction of a vessel need not be determined, since such a value is only used as a basis for limiting the maximum solution concentration of fissile material.

- The drop test need not be performed. Breakage of Raschig rings is accounted for by periodic inspections of the level in the tank.

- The limits of 5% enrichment are used for up to 6% enriched uranium.

Soluble neutron absorber control is a method of increasing neutron absorption in material by placing a soluble neutron absorber (i.e., poison) in a liquid system. Soluble neutron absorbers are only used as secondary NCS control. When soluble neutron absorbers are used, appropriate measurements are taken to ensure their initial presence and their continuous presence at the correct concentration.

5. **Piece Count** – Piece count is a method of limiting fissile material mass and/or geometry by limiting the number of containers or components with known amounts of SNM and/or fixed geometries.

6. **Mass** – Mass control is a method of limiting the amount of SNM at a given location to an acceptable value. Mass control may be used on its own or in combination with other control methods. When a given mass of material has been determined, a percentage factor is used to determine the mass percentage of SNM in that material. When fixed geometric devices are used to limit the mass of SNM, a conservative process density is used. When the mass is measured, instrumentation is used (e.g., scales, non-destructive assay equipment, etc.).

7. **Moderation** – Moderation control is a method of limiting or excluding either interstitial (i.e., within the SNM) or interspersed (i.e., between SNM units) moderating materials or both. NFS complies with the requirements of ANSI/ANS-8.22-1997 as it relates to limiting and controlling moderators. The most common moderating materials contain hydrogen; however, moderating materials may also include materials such as carbon and beryllium. Nuclear criticality safety based on control of moderation

requires that sources of moderation be identified and controlled. When designing physical structures for moderation control, the design should preclude the ingress of moderation. When developing firefighting procedures for use in a moderation controlled area, restrictions should be placed on the use of moderator material. After evaluating all credible sources of moderation for the potential for intrusion into a moderation-controlled area/workstation, the ingress of moderation is precluded or controlled. When moderation is measured, the measurement is obtained by using instrumentation, calculation, or by visual inspection as appropriate. In addition, when the NCS organization requires samples to be taken and analyzed by the NFS laboratory (i.e., wt% H_2O) to determine compliance with moderation limits, dual independent sampling methods will be employed. Shipper information may also be used as a basis for moderation content.

8. **Concentration** – Concentration control is a method of measuring and controlling the concentration of SNM in hydrogenous liquids to an acceptable value. When concentration control is utilized, the concentration is determined by appropriate sampling and analysis techniques (e.g., dual independent sampling) or by instrumentation which has been properly maintained and calibrated (e.g., in-line monitor). The analysis will consider the solubility limits of the SNM composition and possible concentrating mechanisms (e.g., precipitation, evaporation, settling, chemical phase change) and controls are established, as necessary, to prevent such mechanisms. When a tank containing concentration-controlled solution is used, the tank is normally closed.

9. **Material Composition** – Material composition (e.g., material type, density, heterogeneity, etc.) control is based on consideration of the physical, chemical, and/or nuclear properties of a material such that the ^{235}U density and neutron absorption of other materials within the compound are identified and understood (e.g., metal versus oxide versus nitrate, etc.). Manufacturing variability and measurement uncertainty are considered when using material specification as a method of control. Possible misidentification is considered for feed materials when using the feed material specification as control. With regard to heterogeneity, heterogeneous effects are particularly relevant for low-enriched uranium processes, where, all other parameters being equal, heterogeneous systems are more reactive than homogeneous systems. With regard to density, when the density is measured, the measurement is obtained by the use of instrumentation.

10. **Enrichment** – Enrichment control utilizes the inherent differences in critical attributes (critical dimensions, mass, etc.) of uranium at different enrichments of ^{235}U. A method of segregating enrichments is used to

ensure differing enrichments will not be interchanged, or else the most limiting enrichment is applied to all material. When the enrichment needs to be measured, the measurement is obtained by using instrumentation (e.g., lab analysis, non-destructive assay equipment, etc.).

11. **Reflection** – Reflection control is a method of control which limits neutron return back into an SNM-bearing system. Refer to the following reflection requirements:

General Reflection Requirements

Conservative reflection conditions are established when evaluating the criticality safety of individual units or arrays. If reflection conditions are uncontrolled, the maximum credible amount of water reflection is considered when calculating system subcriticality. Under certain conditions, however, materials such as concrete, beryllium, carbon, and polyethylene may be more effective reflectors than water. The thickness and location of these types of reflectors are considered in the model. If it is credible for reflection conditions to exceed those used in the analysis of system subcriticality, then reflection controls are implemented to maintain conditions to within the bounds of the analysis. Where positive barriers are used to maintain reflection control, the barriers are maintained through the configuration management and maintenance programs.

Single or Individual Units

A single unit (e.g., vessel or container) is shown to be subcritical when reflected by at least 30 cm of close-fitting water unless:

1) reflector(s) more effective than water are within 30 cm of the unit, or
2) where 30 cm of close-fitting water reflection is not credible.

When reflectors more effective than water are within 30 cm of a single unit, the thickness and location of these reflectors are considered in the model. When a reflector is offset from a single unit, subcriticality will be demonstrated for the thickness and material of the reflector at no more than the offset distance. The efficacy of the controls implemented to maintain the offset spacing is considered in the analysis of unit subcriticality. Subcriticality for single unit reflection may be demonstrated with calculations or by reference to documented subcritical values (e.g., maximum subcritical values provided in ANSI/ANS-8.1).

Multiple Portable Containers

Unless controls are implemented to limit reflection, evaluation of multiple portable containers must be shown to be subcritical with at least a 10 cm thick, 180 cm tall, and 60 cm wide close-fitting water reflector on one side of the containers and at least 2.5 cm of tangential water reflection on the remaining sides of the containers. These reflectors are modeled as slabs or a box that is tangential to the group of containers, or as a tangential cylinder around the group of containers. Figures 5-1 and 5-2 show these configurations.

Figure 5-1
Tangential Cylinder and Slab Around Containers

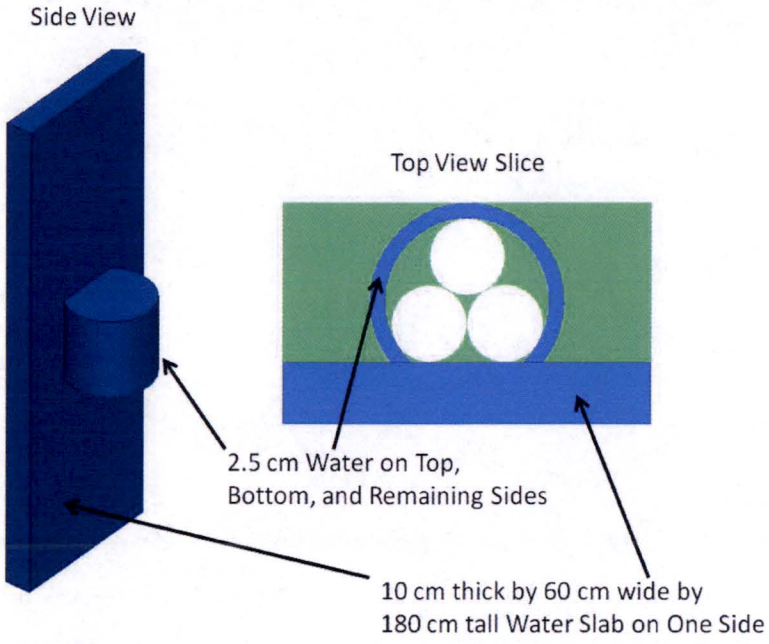
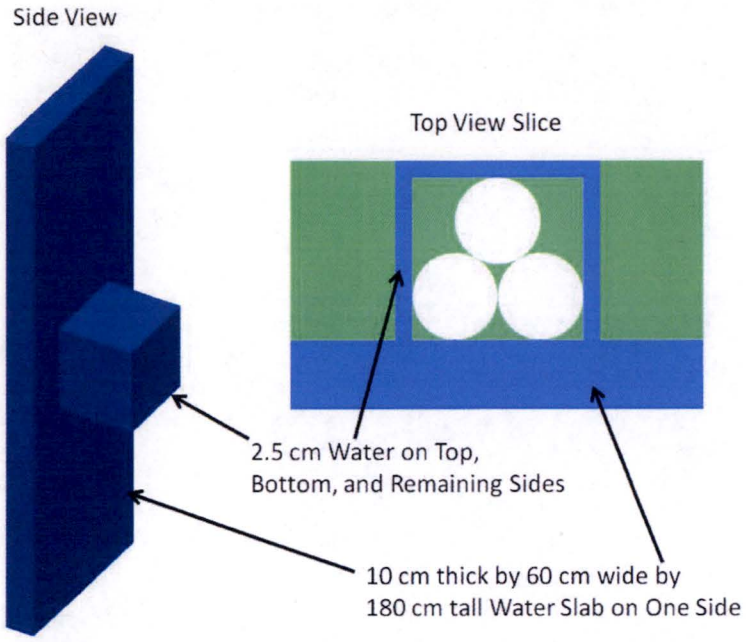


Figure 5-2
Tangential Box and Slab Around Containers



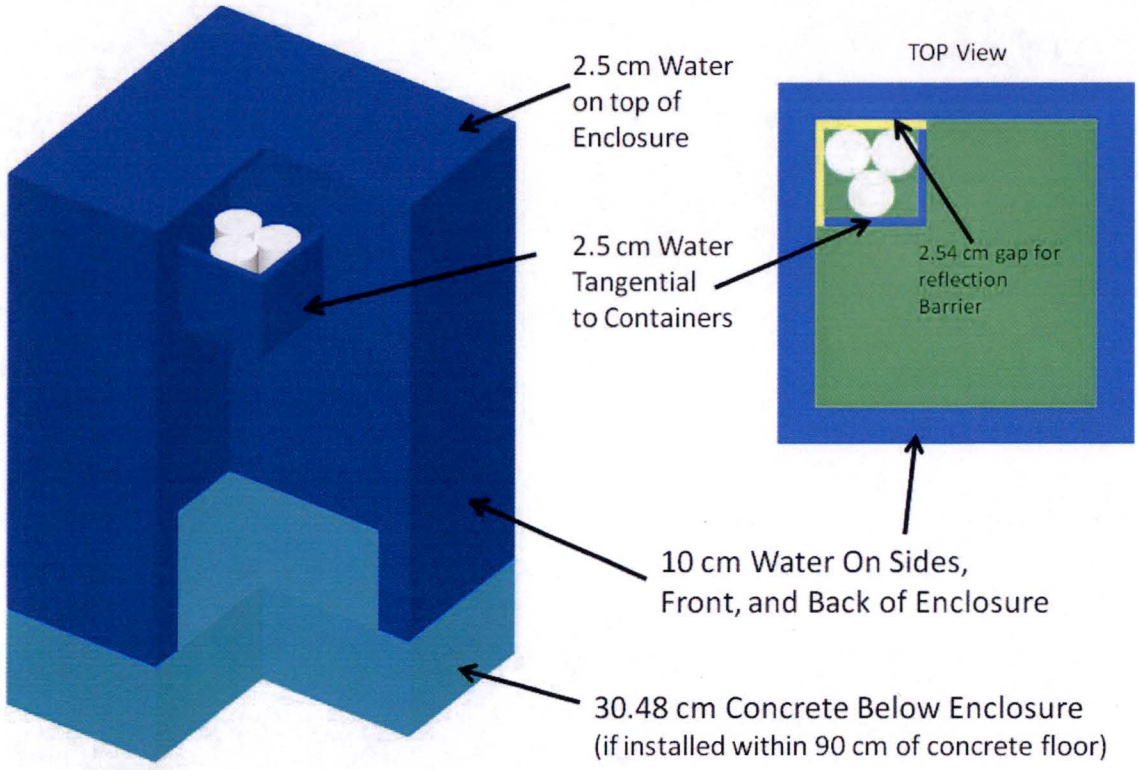
Enclosures/Gloveboxes

Subcriticality of enclosures is demonstrated with the accessible outer front, back, and sides of the enclosures reflected by 10 cm of close-fitting water and the outer top of the enclosures reflected by at least 2.5 cm of close-fitting water. The thickness and location of reflectors that may be more effective than water (e.g., concrete, beryllium, carbon, and polyethylene) must be considered in the model if they are located within 60 cm from the front, back, sides, and top of the enclosure or within 90 cm from the bottom of the enclosure.

Evaluation of overall enclosure reactivity is calculated with portable containers inside the enclosure positioned as close as possible to the outer water reflectors, taking into account any fixed spacing controls that may be present in the enclosure. At least 2.5 cm of water reflection is modeled on the remaining sides of the portable containers. This reflection is modeled as slabs or a box that is tangential to the group of containers, or as a tangential cylinder around the container or group of containers. Note that a larger calculated effective neutron multiplication might result when the portable containers are spaced farther away from the outer reflectors, but nearer to other fixed fissile units that are inside the

enclosure. Figure 5-3 shows the configuration of the enclosure model containing a box that is tangential to the containers, whereas if a cylinder were used it would be similar to that shown in Figure 5-1.

Figure 5-3
Enclosure Reflection Boundary Conditions



Arrays

Ordered arrays (more than two units evenly spaced in one or more dimension) of columns or storage racks are demonstrated to be subcritical using a 10 cm thick and 180 cm tall water boundary around the array (e.g., boundary framework). The thickness and location of reflectors that may be more effective than water (e.g., concrete, beryllium, carbon, and polyethylene) are also considered.

The density of water interspersed between units within the array is varied from 0.0 to 0.1 grams per cubic centimeter, to bound conditions that may exist during fire sprinkler activation or up to full density water if full flooding is credible. Where personnel may physically enter the array (e.g., storage vault), a 10 cm thick and 180 cm tall water slab must be modeled in each

aisle, unless the array remains subcritical for any amount of interspersed water.

5.5.3 Transfers from Favorable to Unfavorable Geometry

Transfers from favorable geometry (e.g., column) to unfavorable geometry (e.g., tank) are controlled by one (1) of the following three (3) general provisions:

1) Multiple engineered hardware controls (e.g., in-line monitors) capable of preventing an unsafe transfer; or

2) At least one (1) engineered hardware control (e.g., in-line monitor) capable of preventing an unsafe transfer plus a determination of safe conditions (e.g., sampling) and actuation of transfer by an individual; or

3) A design requiring independent actions by two (2) individuals before transfer is possible, each action supported by independent measurements of material to be transferred, and a determination of safe conditions. In this case, physical impediments should be included in the system design which will prohibit either individual from performing both of the actions intended to be performed independently.

5.5.4 Computer Codes and Associated Safety Limits

Computer Codes

Computer codes may be used to calculate system reactivity (i.e., k_{eff}). NFS complies with ANSI/ANS-8.1-1998 as it relates to computer codes.

The calculational margin is determined for the computer code. As one acceptable method, the margin may be based on a validation against applicable benchmark experiments using a one-sided 95% tolerance limit at a 95% confidence level less an additional 0.015 Δk_{eff}.

Computer codes are validated to ensure that they calculate within acceptable ranges and that the assumptions are appropriate. The validation reports are incorporated into the configuration management program. NFS commits to the intent of the validation report statement in NRC Regulatory Guide 3.71, August 1998, which states that the following should be demonstrated: (1) the adequacy of the margin of safety for subcriticality by assuring that the margin is large compared to the uncertainty in the calculated value of k_{eff}; (2) that the calculation of k_{eff} is based on a set of variables whose values lie in a range for which the

methodology used to determine k_{eff} has been validated; and (3) that trends in the bias support the extension of the methodology to areas outside the area(s) of applicability.

The validation report should have:

(a) A description of the theory of the methodology that is sufficiently detailed and clear to allow understanding of the methodology and independent duplication of results.

(b) A description of the area(s) of applicability that identifies the range of values for which valid results have been obtained. In accordance with the provisions in ANSI/ANS-8.1-1998, any extrapolation beyond the area(s) of applicability should be supported by an established mathematical methodology.

(c) A description of the use of pertinent computer codes, assumptions, and techniques in the methodology.

(d) A description of the proper functioning of the mathematical operations in the methodology (e.g., a description of mathematical testing).

(e) A description of the data used in the methodology, showing that the data were based on reliable experimental measurements.

(f) A description of the plant-specific benchmark experiments and the data derived therefrom that were used for validating the methodology.

(g) A description of the bias, uncertainty in the bias, uncertainty in the methodology, uncertainty in the data, uncertainty in the benchmark experiments, and margin of subcriticality for safety, as well as the basis for these items, as they are used in the methodology. If the bias is determined to be advantageous, a bias of 0.0 is used (e.g., in a critical experiment where the k_{eff} is known to be 1.00 and the code calculates 1.02, a bias of 0.02 cannot be used to allow calculations to be made above the value of 1.00).

(h) A description of the software and hardware that will use the methodology.

(i) A description of the verification process and acceptable results.

When modifications are made to the computer code system, the impact of the change is assessed to determine if the system needs to be re-

validated. If there are changes to the computational platform, then the computer code system will be verified. As a minimum, verification is performed upon installation of a code package.

Safety Limits

When determining subcriticality based on computer code calculations, the following k_{eff} safety limits are not to be exceeded:

System	Safety Limit
High-enriched systems (uranium enriched in ^{235}U greater than 10 wt%)	$k_{eff} + 2\sigma \leq 0.95$
Low-enriched systems (uranium enriched in ^{235}U less than or equal to 10 wt%)	$k_{eff} + 2\sigma \leq 0.97$

The k_{eff} values of 0.95 and 0.97 above are exact limit values, and do not imply that compliance need only be shown to 2 significant figures. Compliance with these values allows for purely calculational inaccuracies, such as Monte Carlo variance, by meeting the limit with a margin in the conservative direction of at least two standard deviations. Any rounding is in the conservative direction.

Appendix 5A

Note: Criticality safety may be based on data provided in handbooks, reference documents, experimental data or the values listed in Tables 5A-1 thru 5A-6 of this chapter. The safety factors as presented for Tables 5A-1 thru 5A-6 must be applied to critical values; or, maximum subcritical values may be used as provided in handbooks or standards (e.g., maximum subcritical values provided in ANSI/ANS-8.1).

Table 5A-1

Limits for Fully Reflected Units of Homogeneous Low Enriched Materials

ENR WT% ^{235}U	MASS KG ^{235}U		CYL DIA INCHES	SLAB THICKNESS INCHES				VOLUME LITERS
	(A)	(B)	(C)	(D)	(E)	(F)	(G)	(H)
10.0	.570	.950	7.9	1.7	2.9	3.3	3.7	14.8
9.0	.600	1.000	8.2	1.8	3.0	3.4	3.8	15.8
8.0	.625	1.040	8.4	1.9	3.2	3.6	4.0	17.3
7.0	.660	1.100	8.7	2.0	3.4	3.8	4.2	19.3
6.0	.720	1.200	9.0	2.1	3.5	4.0	4.5	22.0
5.5	.755	1.255	9.5	2.2	3.7	4.2	4.7	23.4
5.0	.800	1.340	10.1	2.3	3.8	4.4	4.9	25.3
4.5	.855	1.425	10.3	2.4	4.1	4.6	5.1	28.8
4.0	.935	1.560	10.5	2.6	4.4	5.0	5.6	33.2
3.5	1.040	1.735	11.3	2.9	4.8	5.4	6.0	38.1
3.0	1.210	2.015	12.4	3.2	5.4	6.1	6.8	46.4
2.5	1.510	2.515	13.9	3.8	6.4	7.2	7.8	61.5
2.0	2.165	3.605	16.5	4.6	7.7	8.7	9.2	98.6
1.5	4.500	7.500	22.0	6.6	10.9	12.4	13.9	210.7
1.0	6.840	11.400	--	--	--	--	--	--

Table 5A-2

Limits for Fully Reflected Units of Heterogeneous Low Enriched Materials

ENR WT% ^{235}U	MASS KG ^{235}U		CYL DIA INCHES	SLAB THICKNESS INCHES				VOLUME LITERS
	(A)	(B)	(C)	(D)	(E)	(F)	(G)	(H)
10.0	.530	.885	7.1	1.5	2.5	2.8	3.0	6.0
9.0	.550	.915	7.2	1.6	2.6	3.0	3.1	6.6
8.0	.565	.945	7.3	1.6	2.7	3.1	3.3	7.5
7.0	.595	.995	7.5	1.7	2.9	3.2	3.5	8.5
6.0	.635	1.055	7.8	1.8	3.0	3.4	3.6	10.4
5.5	.660	1.100	7.9	1.8	3.1	3.5	3.7	10.8
5.0	.700	1.165	8.1	1.9	3.2	3.6	3.9	11.7
4.5	.735	1.230	8.3	2.0	3.3	3.7	4.1	13.0
4.0	.785	1.310	8.5	2.1	3.4	3.9	4.3	14.9
3.5	.850	1.415	8.9	2.2	3.6	4.1	4.6	16.8
3.0	.960	1.600	9.3	2.3	3.9	4.4	5.0	20.3
2.5	1.120	1.870	10.0	2.5	4.2	4.8	5.5	24.5
2.0	1.430	2.380	11.3	2.9	4.9	5.5	6.0	32.2
1.5	2.090	3.485	13.8	3.8	6.4	7.2	8.1	52.6
1.0	6.615	11.025	23.1	6.0	10.0	11.4	12.0	203.0

Table 5A-3

Limits for Fully Reflected Units of Homogeneous 100.0 WT% ^{235}U Compounds and Water

MAX. DENSITY (gU/cc)	MASS KG ^{235}U		CYL DIA INCHES	SLAB THICKNESS INCHES				VOLUME LITERS
	(A)	(B)	(C)	(D)	(E)	(F)	(G)	(H)
4.0	0.365	0.610	4.7	0.7	1.2	1.3	1.4	4.2
5.8	0.365	0.610	4.5	0.6	1.1	1.2	1.3	3.7
8.0	0.365	0.610	4.3	0.6	1.0	1.1	1.2	3.3

Table 5A-4

Limits for Fully Reflected Units of Uranium Metal at 100.0 WT% ^{235}U for all Values of H/X

MASS KG ^{235}U		CYL DIA INCHES	SLAB THICKNESS INCHES				VOLUME LITERS
(A)	(B)	(C)	(D)	(E)	(F)	(G)	(H)
0.365	0.610	2.5	0.2	0.4	0.4	0.5	0.935

(A) Limit is ≤45% of minimum critical mass – double batching is credible
(B) Limit is ≤75% of minimum critical mass – double batching not credible
(C) Limit is ≤90% of minimum critical cylinder diameter
(D) Limit is ≤45% of minimum critical slab thickness
(E) Limit is ≤75% of minimum critical slab thickness
(F) Limit is ≤85% of minimum critical slab thickness
(G) Limit is ≤90% of minimum critical slab thickness
(H) Limit is ≤75% of minimum critical spherical volume

Table 5A-5

Critical and Safe Concentrations in Aqueous Solutions

ENRICHMENT (Wt% ^{235}U)	CRITICAL CONCENTRATION* (g ^{235}U/liter)	SAFE CONCENTRATION LIMIT (g ^{235}U/liter)
>5%	11.8	5.0
≤5%	14.1	6.1
≤4%	14.6	6.3
≤3%	15.6	6.7
≤2.5%	16.2	7.0
≤2%	17.6	7.6
* Reference: ARH 600, Volume II; Table III.B-2; Figures III.B.10(5)-2, III.B.10(4)-2, III.B.10(3)-2, III.B.10(2.5)-2, and III.B.10(2)-2 at K_∞=0.99.		

Dry ^{235}U Limit:

Safe dry mass = 10 kgs ^{235}U (hydrogen moderated only)
5.25 kgs ^{235}U (hydrogen and carbon moderated)

"Dry" mass limits meet the following criteria:

- H/^{235}U ≤ 4.0 for uranium compounds at greater than 10% to 100% enriched (hydrogen moderated only).
- H/^{235}U ≤ 10.0 for uranium compounds at 0.72% to 10% enriched (hydrogen moderated only).
- H/U ≤ 2.0 and C/U ≤ 900 for uranium compounds at 93.15% or less enrichment.

^{235}U Area Density Limits:

- 0.19 grams $^{235}U/cm^2$ for uranium compounds at ^{235}U enrichments greater than 5 wt% to 100 wt%.
- 0.25 grams $^{235}U/cm^2$ for uranium compounds at ^{235}U enrichments ≤ 5 wt%.

^{233}U Limit:

Because the form of the materials containing the ^{233}U isotope is restricted to residual contamination from past operations and small quantities for analysis and development studies, the nuclear safety limits are based primarily on the safe wet mass value of 250 grams ^{233}U. However, limits for geometry controls on individual units containing ^{233}U may be established and implemented.

Transuranic Limits:

Plutonium is generally present in small quantities, such as residual radioactivity from prior operations, samples received for laboratory analysis, materials received for development studies, and processing of materials containing trace amounts of plutonium as a contaminant. The maximum subcritical mass limit, as stated in ANSI/ANS-8.1-1998, will be applied for these limited operations.

If, based on analytical results or an engineering evaluation, multiple fissile isotopes are determined to be present at greater than 15 ppm (uranium basis) in scrap material received for storage or processing, the effect of the multiple fissile isotopes on nuclear criticality safety will be evaluated. A ^{235}U fissile gram equivalent (FGE) for each fissile nuclide present in the material will be determined using the maximum subcritical mass limits as stated in ANSI/ANS-8.1-1998 and ANSI/ANS-8.15-1981.

Enrichment Blending System

A system for performing the blending of a high-enriched uranyl nitrate (UN) solution with a natural, depleted, or slightly enriched UN solution to produce a low-enriched UN solution may be operated. The blended product may be discharged into a large geometry vessel. The system will implement at least one (1) in-line measurement of the ^{235}U concentration of the blended solution prior to discharge into the large geometry vessel (e.g., in-line monitor).

Engineering and procedural controls are utilized to prevent the solution in the tank from exceeding the criticality control limit for g^{235}U/liter. The controls meet the double contingency principle.

Limiting conditions are placed on certain parameters for the blending operation. These operational parameters are grams ^{235}U per liter for the high enriched feed solutions and volume of the two (2) feed solutions.

The limiting conditions of operation for the operational parameters are set based on Curve C of Figure 5A-1. Curve C defines the limiting condition of operation for the blend tank. It is derived by taking 85% of Curve B and limited to a maximum of 6% enrichment. Curve B depicts the conditions where $k_{eff} + 2\sigma = 0.95$. For a final enrichment greater than 6%, 85% of the concentration which has a k_∞ value of 0.95 will be applied per Table 5A-6.

UN Tanks

Uranyl nitrate (UN) solutions produced by an NFS process or received in authorized shipments may be transferred into large geometry vessels for storage or final blend adjustments. For any enrichment no greater than 20 wt% ^{235}U, limiting conditions corresponding to 85% of the concentration which has a k_∞ value of 0.95 will be applied. Concentration limits may be found in Table 5A-6.

The only authorized activities involving the UN solutions in these tanks (e.g., sampling, blending, and dilution) will not increase the ^{235}U concentrations above the limits in Table 5A-6.

Figure 5A-1
Curves for Defining Limiting Conditions of Operation for the Enrichment Blend

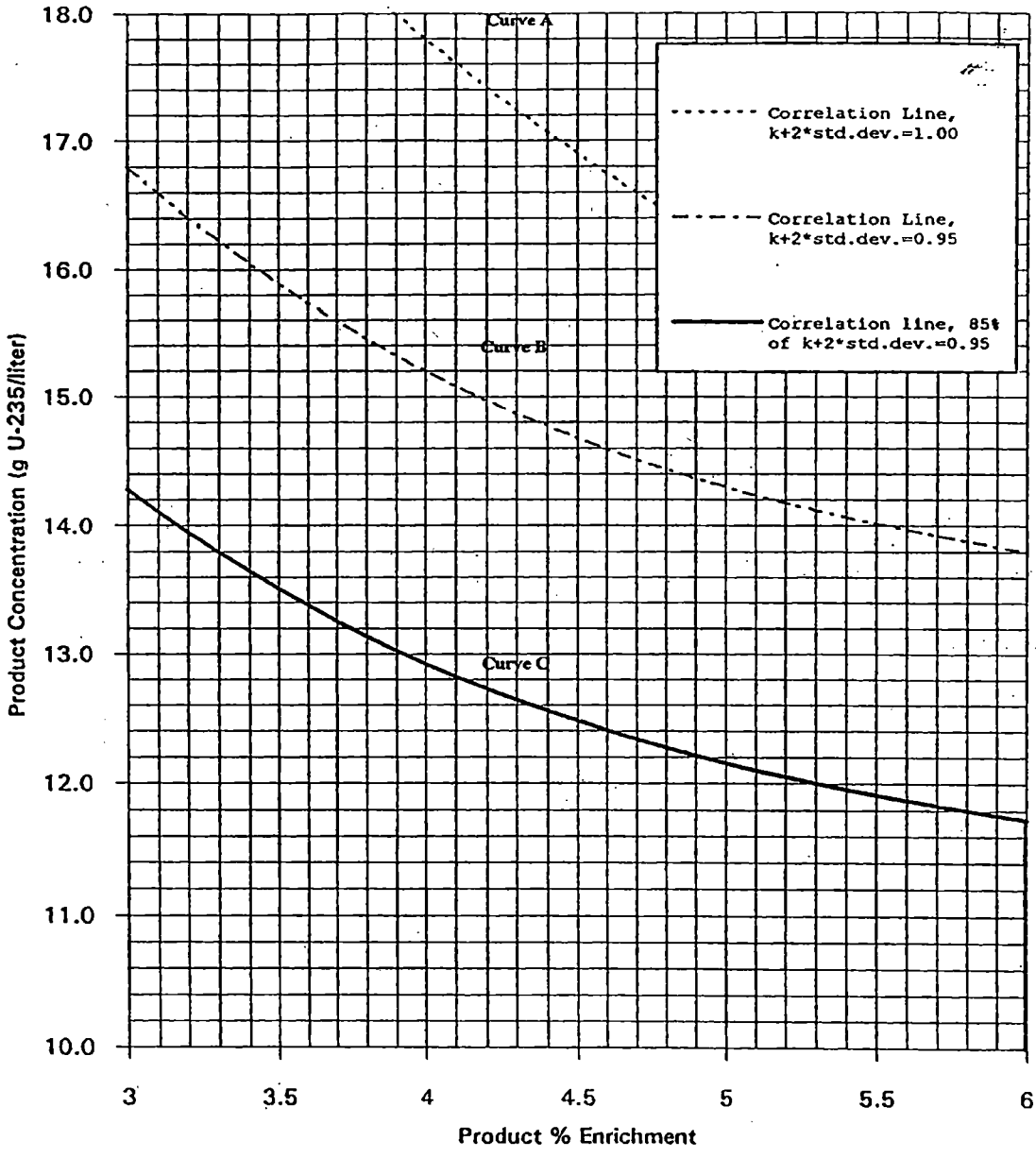


Table 5A-6
Limiting Conditions of Operations for UN Tanks

Enrichment (Wt% ^{235}U)	^{235}U Concentration at k_∞=1.0 (g/l)	^{235}U Concentration at k_∞=0.95 (g/l)	^{235}U Concentration at 85% of k_∞=0.95 Value (g/l)
1.96	There are no restrictions for UN solutions with enrichments ≤1.96 wt% ^{235}U (Reference Table 2, ANSI/ANS-8.1-1998).		
3	19.07	16.20	13.77
4	16.83	14.65	12.45
5	15.76	13.86	11.78
6	15.11	13.37	11.37
7	14.69	13.04	11.08
8	14.37	12.79	10.87
9	14.13	12.61	10.72
10	13.94	12.46	10.59
11	13.78	12.33	10.48
12	13.65	12.23	10.40
13	13.54	12.14	10.32
14	13.45	12.07	10.26
15	13.36	12.00	10.20
16	13.29	11.94	10.15
17	13.23	11.89	10.11
18	13.17	11.84	10.07
19	13.12	11.80	10.03
20	13.07	11.77	10.00

Attachment 7

SNM-124, Chapter 9

(19 pages to follow)

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 9

ENVIRONMENTAL PROTECTION
Table of Contents

ENVIRONMENTAL PROTECTION

Introduction

Effluent releases from the NFS site can occur via two pathways: airborne or liquid. The control systems for each of these pathways are addressed in the following sections. In addition, periodic reviews of the radiation protection program, including effluent control and environmental monitoring, are described in Sections 4.1 and 4.2.

The action levels and the minimum detectable concentrations are documented in procedural guidance and technical basis documents. All documents are maintained current and are available for review at NFS.

Environmental Report - Summary of Environmental Data and Impacts

A complete revision of the NFS Environmental Report was submitted concurrently with the 2009 application for license renewal. The report includes radiological and non-radiological environmental summaries for the NFS site.

Effluent Control and Environmental Monitoring

Effluent controls and environmental monitoring are implemented through compliance with procedural guidance controlled by the Safety discipline. These procedures outline sampling techniques, sample processing and analysis methodologies, quality assurance, and other necessary information for maintaining a viable program. In addition, offsite samples are collected and analyzed routinely to verify the effectiveness of controls and provide data in the event of an emergency situation. Typical sampling locations are provided in Figure 9-1.

9.1 Effluent Control Systems

The objective of the effluent control program is to ensure that radioactive air and liquid effluents are as low as reasonably achievable, and thus protective of the public and environment. This objective is supported by performing routine measurements and calculations, comparing results to action levels, and reporting results to plant management and the NRC, as appropriate. Internal action levels are implemented by procedural guidance to provide early identification of potential problems and prevent exceedance of guidelines set forth in 10 CFR 20.1301. For air effluents, action levels are maintained below the ALARA constraint set forth in 10 CFR 20.1101. For liquid effluents, action levels are

established to limit the total effective dose equivalent to less than 10% of the limit established in 10 CFR 20.1301.

Unless otherwise noted, if an action level is exceeded for airborne or liquid effluents, the following actions will occur: (1) the environmental protection function manager and the responsible process engineering control personnel will be notified, (2) an investigation will be undertaken to identify the cause of the exceedance, and (3) appropriate corrective action(s) will be initiated to reduce observed levels that are above the action levels and to minimize the likelihood of a recurrence. Corrective actions will be documented. If necessary, the environmental protection function manager may order processing activities in an area to be halted until appropriate corrective actions are implemented.

9.1.1 Airborne Effluents

Flow rates on all process ventilation stacks are checked annually and whenever any process changes occur that have the potential to significantly alter the flow rate. Each individual effluent discharge point is evaluated for isotopic distribution based upon process knowledge and historical characterization data. Any significant change to the materials processed will be re-evaluated using isotopic analysis to verify accuracy of characterization data.

Screening levels are based on the gross alpha and gross beta analyses performed onsite from the routinely collected gaseous and liquid effluents, as well as historical knowledge of the process and material. Characterization data is used to determine if there is a significant change in the primary dose contributors. An offsite isotopic analysis will re-evaluate the characterization data. When radionuclides are detected above the action levels, the following will occur: notification of the environmental protection function manager and the responsible process engineering personnel, an investigation will be undertaken to identify the cause of the exceedance, and appropriate corrective action(s) will be initiated to reduce observed levels that are above the action levels and to minimize the likelihood of a recurrence.

9.1.1.1 Source-Point Sampling of Airborne Effluents

Effluent sampling is representative of the total discharge. All process stacks and vents with the potential to release airborne radioactivity at concentrations greater than or equal to 10% of the values in 10 CFR 20, Appendix B, Table 2, Column 1, are sampled continuously, with the exception of equipment malfunctions, during processing of radioactive materials. Samples are collected daily from active processing areas and at least weekly from decommissioning areas and inactive processing areas.

To minimize effects of radon progeny on measured activity, air sample results may be decay-corrected for radon progeny or a waiting period may be used to eliminate the interference. Samples are routinely analyzed for gross alpha/beta activity and compared against action points for early detection and investigation of potential problems.

9.1.1.2 Action Level for Airborne Effluents

Gross alpha and beta activity data from the stack sampling program are compared to the action levels established by procedural guidance. Action levels within procedural guidance are specified for combined releases from stacks with similar physical and radiological release characteristics and include values for both the cumulative activity released in a 12-month period and the monthly average activity concentration.

The cumulative alpha and beta activities released in the previous 12-month period are calculated monthly for each indicated stack grouping and compared to action levels. The monthly average alpha and beta activity concentrations are determined for each group of stacks by dividing the total alpha and total beta activities released by the group by the total volume of air released by the group. The stack action levels as defined within procedural guidance were derived using a dose-based approach with the intent of preventing the maximally exposed off-site receptor from receiving an annual total effective dose equivalent from air effluents, greater than the ALARA dose constraint cited in 10 CFR 20.1101(d). Dose calculations are performed using ICRP 66 and ICRP 68 methodology, assuming an Activity Median Aerodynamic Diameter (AMAD) of one micrometer.

9.1.1.3 Reporting Method

Alpha and beta activity releases in airborne effluents are summarized in monthly and quarterly reports that are maintained as internal documents. These reports include information on both alpha and beta activity emissions for each individual stack and for the site as a whole.

Activity release data are accumulated and reported on a semiannual basis to the NRC as required by 10 CFR 70.59. To meet the semiannual reporting condition in 10 CFR 70.59, a preliminary assessment may be performed if any sample results are pending. A format similar to that presented in Regulatory Guide 4.16 is followed for this report. If semiannual average activity concentrations in stack effluents exceed concentrations listed in Appendix B, Table 2, Column 1, to 10 CFR Part 20, results of an assessment of the maximum concentration at the site boundary and of the total effective dose equivalent to the maximally exposed off-

site receptor from air effluents will be included in this semiannual report to the NRC. The methods used to perform this assessment are discussed in Section 9.1.1.4.

9.1.1.4 Routine Assessment of Concentrations at Site Boundary and Off-site Dose

Each calendar quarter, stack sampling data is compiled and used to calculate the maximum concentration at the site boundary and the total effective dose equivalent (TEDE) to the potential maximally exposed off-site receptor due to air emissions. If the maximum concentration at the site boundary exceeds values in Appendix B, Table 2, Column 1, to 10 CFR Part 20, or if the resulting TEDE exceeds 25% of the annual ALARA constraint for air emissions cited in 10 CFR 20.1101, appropriate corrective actions will be identified and implemented to reduce future dose levels. Each calendar year, the annual dose to the maximally exposed off-site receptor is calculated. If the annual dose exceeds the ALARA constraint as listed in 10 CFR 20.1101, appropriate reports will be submitted to the NRC in accordance with 10 CFR 20.2203.

Assessment of the maximum concentration at the site boundary and maximum off-site dose is performed using the Comply Code (U.S. Environmental Protection Agency [EPA]), the CAP88-PC Computer Code 3.0 or higher (U.S. Department of Energy [DOE]), or an equivalent methodology. Site specific meteorological data is used in the assessment when available. Otherwise, conservative values are used for meteorological parameters. Air samples may be analyzed for uranium lung solubility class and enrichment in order to characterize the material released. Otherwise, conservative values are used for solubility class and enrichment. NFS follows procedural documents to perform the calculations. Parameter values used in modeling are based on data collected during the assessment period, previous monitoring history, or the professional judgment of an environmental scientist or health physicist.

9.1.2 Liquid Effluents

Typically, process waste water is collected in tanks in or near the various process buildings. Based upon the origin of the liquid waste, samples may be required to be collected and analyzed prior to transfer to the Waste Water Treatment Facility (WWTF) for treatment. Internal action limits are established to control concentrations of radionuclides transferred to the WWTF. The WWTF is operated in accordance with a State of Tennessee issued NPDES permit. Waste water is treated, analyzed, and released on a batch basis. Authorization to release the treated water to the Nolichucky River is procedurally controlled.

Discharges to the sanitary sewer system include: groundwater treatment facility effluents, BLEU Complex treated process wastewater, all plant bathrooms, and plant showers. Sanitary sewer discharges to the City of Erwin Publicly Owned Treatment Works (POTW), are conducted in accordance with a locally-issued pretreatment permit.

The NFS plant site has two storm water drainage pathways which empty into Martin Creek. Storm water from NFS and NFS' BLEU Complex is collected in a series of drains and directed into an open ditch which parallels the northwest plant boundary and empties into Martin Creek. The second storm water pathway directs drainage from the eastern portion of NFS property into an underground pipe which also empties into Martin Creek. The underground pipe was designed to carry the flow of Banner Spring around the North Site. The storm water permit is issued by the State of Tennessee and sampling is performed in accordance with the permit. In addition, storm water runoff is monitored by the weekly collection of grab samples from Martin Creek and the quarterly collection of samples of the two storm water pathways. Martin Creek empties into the Nolichucky River which is routinely sampled on a quarterly basis.

9.1.2.1 Source-Point Sampling of Liquid Effluents

The WWTF treats and discharges process waste water on a batch basis. Prior to discharge, each batch is sampled and analyzed for gross alpha and gross beta radioactivity. A monthly composite sample is collected and analyzed for isotopes of uranium. The monthly composite is analyzed for other radionuclides if materials in addition to uranium are suspected to be present in process waste water at levels exceeding 10% of the concentration values in Appendix B, Table 2, Column 2, 10 CFR Part 20. The chemical parameters prescribed in the State of Tennessee NPDES permit are also analyzed at least on the frequency specified in the permit. Samples of the treated waste water are collected from the final neutralization or storage tank prior to discharge.

Sanitary sewer wastes are discharged through two main streams (one for the BLEU Complex and one for the remainder of the main NFS plant site), to the Erwin-POTW. When process water containing radioactive materials is disposed of by release into the sanitary sewerage, in accordance with 10 CFR 20.2003 requirements, samples representative of the total discharge from the applicable sanitary sewer discharge point are collected and analyzed as identified in Table 9-1. The monthly composite samples are analyzed for additional radionuclides, when the concentrations of those radionuclides exceed 10% of the concentration values in 10 CFR 20, Appendix B, Table 2, Column 2. Solubility is determined in accordance with 10 CFR 20.2003 and serves as the guidance for the insoluble radioactivity analyses.

The sewage sludge at the Erwin-POTW is sampled quarterly, provided a blow-down sample is available. The sewage sludge samples are analyzed in accordance with Table 9-1.

Martin Creek downstream samples are collected and analyzed for gross alpha and gross beta radioactivity and the action levels are implemented in procedures.

Table 9-1
Summary Table of Environmental Radiological Monitoring Program

Sampling Point	Sample Type/ Collection Frequency	Parameters Analyzed
Liquid Effluents		
Surface Water		
Martin Creek Upstream	Grab/Quarterly	Gross Alpha Gross Beta
Nolichucky River Upstream	Grab/Quarterly	Gross Alpha Gross Beta
Martin Creek Downstream	Grab/Weekly	Gross Alpha Gross Beta
Nolichucky River Downstream	Grab/Quarterly	Gross Alpha Gross Beta
Process Waste Water		
Waste Water Treatment Facility	Grab/each batch	Gross Alpha Gross Beta
	Composite/Monthly	Isotopic U
NFS Sanitary Sewer[2]	Continuous/Daily[1]	Gross Alpha Gross Beta
	Composite/Monthly	Isotopic U
	Composite/Monthly[3]	Insoluble Radioactivity
BLEU Complex Sanitary Sewer[2]	Continuous/Daily[1]	Gross Alpha Gross Beta
	Composite/Monthly	Isotopic U
	Composite/Monthly[4]	Insoluble Radioactivity
Environmental Media		
Sludge (Erwin POTW)	Grab/Quarterly	Isotopic U
Storm Water Pathway		
Banner Spring Branch	Grab/Quarterly	Gross Alpha Gross Beta Isotopic U
Perimeter North West Ditch	Grab/Quarterly	Gross Alpha Gross Beta Isotopic U

NOTES:

1 Daily means normal operating days, Monday-Friday, excluding holidays and weekends. On holidays and weekends, samplers will continue to accumulate a sample; however, the sample will not be collected until the next normal operating day.

2 Sampling is only required for disposal of process water containing licensed materials into the sanitary sewerage in accordance with 10 CFR 20.2003.

3 The compliance sampling location for insoluble radioactivity on this discharge point is the Ground Water Treatment Facility (GWTF), because this is the only stream that discharges radioactive material into the NFS sanitary sewer. Insoluble radioactivity sampling is not required on this discharge point when the GWTF is not operational.

4 The compliance sampling location for insoluble radioactivity on this discharge point is the Effluent Processing Building (EPB), because this is the only stream that discharges radioactive material into the BLEU Complex sanitary sewer. Insoluble radioactivity sampling is not required on this discharge point when the EPB is not operational.

9.1.2.2 Action Levels for Liquid Effluents

Prior to final discharge from the WWTF, a gross alpha and beta radioactivity analysis is performed to determine the acceptability for discharge. The batch concentrations allowed to be released, without prior approval of the environmental protection function, are the action levels stated in procedural guidance. These action levels are at or below concentrations listed in 10 CFR 20 Appendix B, Table 2 Column 2.

Waste solutions in which the alpha or beta concentration exceeds one of these action levels is discharged only after approval by the environmental protection function manager or designated individual. If it is found that any discharges over a 12-month period caused the dose to members of the public (from WWTF effluents) to exceed 10% of the dose limit specified in 10 CFR 20.1301, the NRC will be notified of the event in writing within 30 days.

The results of the insoluble radioactivity measurements performed on the sanitary sewer samples are compared to the amount of insoluble radioactivity present in similarly processed background water samples. If insoluble radioactive material is detected in sanitary sewer discharges at concentrations that are statistically greater than the concentrations measured in background samples, discharges of radioactive material to the appropriate sanitary sewer stream will be stopped until appropriate corrective actions are implemented.

Action levels for sewer discharges, and other surface water effluents are at or below concentrations listed in 10 CFR 20, Appendix B, Table 2, Column 2 and are monitored as indicated in Table 9-1.

9.1.2.3 Reporting Methods

Radioactivity in liquid effluents is summarized in a quarterly liquid effluent report that is maintained as an internal document. This report includes information on both the gross alpha and gross beta radioactivity in each liquid effluent stream (i.e., WWTF, NFS sanitary sewer, and BLEU Complex sanitary sewer).

Activity release data are accumulated and reported on a semiannual basis to the NRC as required by 10 CFR 70.59. To meet the semiannual reporting condition in 10 CFR 70.59, a preliminary assessment may be performed if any sample results are pending. A format similar to that presented in Regulatory Guide 4.16 is followed for this report. If the semiannual average activity concentration exceeds the SOF[1] of one for WWTF effluents, results of an assessment of the effective dose equivalent to the maximally exposed off-site receptor from these effluents will be included in this semiannual report to the NRC. The methods used to perform this assessment and additional action levels are discussed in Section 9.1.2.4.

1. SOF = Sum of Fractions for the mixture of radionuclides. The SOF is determined by computing the sum of the ratios of various nuclides divided by their applicable effluent concentration value in Appendix B, Table 2, Column 2 to 10 CFR Part 20. If the SOF for WWTF exceeds 1.0, results of a dose assessment to the maximally exposed off-site receptor will be reported as indicated above.

9.1.2.4 Routine Assessment of Maximum Concentration and Off-Site Dose from WWTF Effluents to the Maximally Exposed Off-Site Receptor

Each calendar quarter, WWTF liquid effluent data is compiled and used to calculate the concentration of radioactive materials at the location of the maximally exposed off-site receptor and the dose (TEDE) to the maximally exposed off-site receptor. Each calendar quarter, the dose for the four previous (consecutive) quarters is calculated. If the calculated dose for this annualized period exceeds 10% of 10 CFR 20.1301, corrective actions will be implemented and the NRC will be notified in writing within 30 days.

Assessment of the maximum concentration and TEDE to the maximally exposed off-site receptor is performed using: (1) National Council on Radiation Protection and Measurements (NCRP) Report No. 123, "Screening Models for Releases of Radionuclides to Atmosphere, Surface Water, and Ground," or (2) pathway analysis models that consider all exposure pathways and accurately reflect site conditions. Site-specific characteristics of the surface waters receiving liquid effluents are assessed. NFS follows written procedures to perform these calculations. Parameter values are based on information contained in NCRP Report No. 123, data collected during the assessment period, publicly available information (e.g., stream flow data compiled by the U.S. Geological Survey (USGS)), previous monitoring history, or the professional judgment of the environmental protection function manager.

9.2 Environmental Surveillance Program

In addition to the effluent monitoring and reporting requirements of this chapter, NFS maintains an Environmental Surveillance Program. The program is established to provide:

1. Additional validation of effluent monitoring systems.
2. Early detection and response to a negative trend in environmental data.
3. Support data in the event of a release of radioactive material.

The monitoring program is detailed in written procedures. The site environmental monitoring program is dynamic, and changes are made as dictated by changes in operations and/or the emergence of new-found information. Typical sampling locations are provided in Figure 9-1.

In the event that a sample(s) specified in Table 9-2 cannot be taken, the manager of the environmental protection function will be notified. An investigation will be initiated to include an assessment of the significance of the event, the cause of the deviation from plan, and determine what corrective action is needed.

Table 9-2
Summary Table of Environmental Radiological Surveillance Program of Environmental Media

Sampling Point	Sample Type/ Collection Frequency	Parameters Analyzed
Ambient Air	Continuous/Weekly	Gross Alpha Gross Beta
	Composite/Quarterly	Isotopic U
	Composite/Annually	Isotopes of concern
Soil	Grab/Quarterly	Gross Alpha[1] Gross Beta
Silt/Sediment	Grab/Quarterly	Gross Alpha[1] Gross Beta
Vegetation	Grab/Quarterly	Gross Alpha[1] Gross Beta
Groundwater	Grab/Quarterly	Isotopic U Isotopic Pu Technetium-99

NOTE: [1] If an action level specified by procedural guidance is exceeded for this media, isotopic analysis will be performed on the sample (or a sample from the same location if the initial sample volume is insufficient).

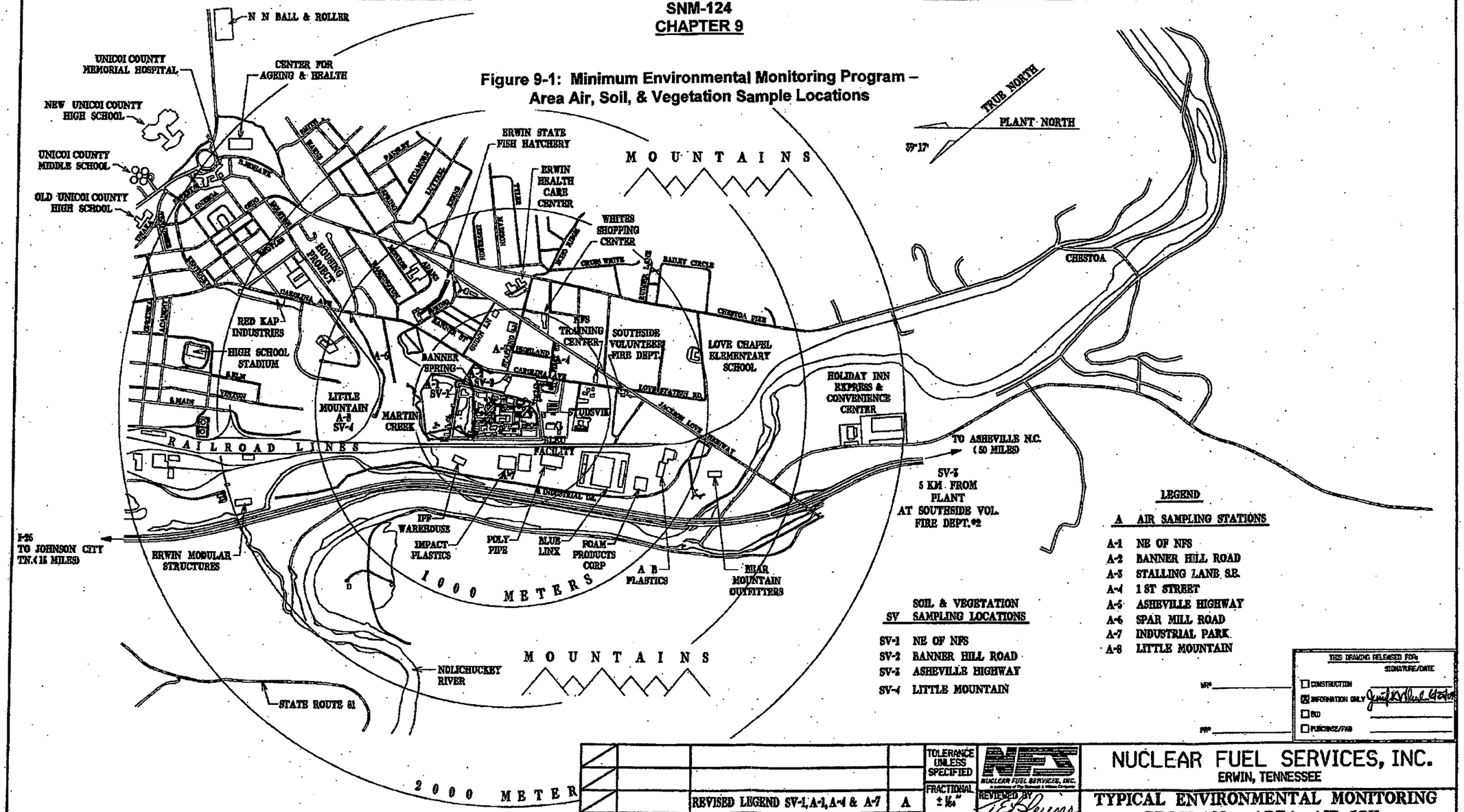
9.2.1 Air Monitoring

Air samples are collected and analyzed to monitor airborne radioactivity concentrations attributable to plant operations. The locations of these stations are concentrated along the predominant wind directions. Detailed locations are specified in appropriate written procedures.

Air samples are collected continuously, exchanged weekly, and analyzed for gross alpha and beta activity weekly. In addition, air samples are analyzed for isotopic U on a quarterly basis and additional isotopes of concern (based upon characterization data of material processed) on an annual basis for the sampling station nearest the predicted maximally exposed off-site receptor.

Ambient air sampling results are reviewed quarterly and compared to the action levels implemented by procedural guidance. If an action level is exceeded, the environmental protection function manager will be notified and an investigation will be undertaken to determine the cause of the exceedance. Depending on the severity of the event, corrective actions may be initiated to reduce air emissions from the plant.

Figure 9-1: Minimum Environmental Monitoring Program – Area Air, Soil, & Vegetation Sample Locations



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REVIEWED BY: T.E. Lewis
PROPOSED COMPLETION DATE:
DRAFTER
PROPOSED APPROVALS
REQ.

NUCLEAR FUEL SERVICES, INC.
ERWIN, TENNESSEE

TYPICAL ENVIRONMENTAL MONITORING PROGRAM - AREA AIR, SOIL & VEGETATION SAMPLE LOCATIONS

DRAFTER W.G. HENSLEY | SCALE 1"=1500' | DATE 5-18-09
AS-BUILT APPROVALS
REQ. Carolyn J. Buchanan
OWNER Carolyn J. Buchanan
DRAWING NO. 000-C0328-B
CONFIGURATION CONTROL YES ____ NO ☒

9.2.2 Soil Sampling

Soil samples are collected and analyzed quarterly for gross alpha and beta concentrations to monitor for long-term buildup of radioactivity attributable to plant operations. The locations of these samples are concentrated along the predominant wind directions. Locations are specified in appropriate written procedures. Typically, four locations are routinely monitored (Figure 9-1.) Where these analyses yield results exceeding action levels as implemented in procedural guidance, specific isotopic analysis will be performed on samples collected at the same site. The elemental isotopes for which analysis is performed will be determined by the materials and processes in which the plant is involved.

9.2.3 Vegetation Sampling

Vegetation samples are collected and analyzed quarterly for gross alpha and gross beta concentrations to determine if there is ascertainable impact from plant operations. The locations of these stations are concentrated along the predominant wind directions. Locations are specified in appropriate written procedures. Typically, four forage vegetation samples are routinely monitored (Figure 9-1.) Where these analyses yield results exceeding action levels as implemented in procedural guidance, specific isotopic analysis will be performed on samples collected at the same site. The elemental isotopes for which analysis is performed will be determined by the materials and processes in which the plant is involved.

9.2.4 Silt/Sediment Sampling

Silt/sediment samples are collected and analyzed quarterly for gross alpha and beta activity to monitor for long-term buildup of radioactivity from the deposition of liquid discharges and/or surface runoff. The locations of these samples are along streams potentially affected by plant operations. Locations are specified in appropriate written procedures. Upstream samples are collected in addition to downstream samples, where appropriate. Where these analyses yield results exceeding action levels as implemented in procedural guidance, specific isotopic analysis will be performed on samples collected at the same site. The elemental isotopes for which analysis is performed will be determined by the materials and processes in which the plant is involved.

9.2.5 Surface Water Sampling

Surface water sampling is part of the liquid effluent program as described in Section 9.1.2. The Nolichucky River is the final receiving stream for treated waste water discharged from the facility and surface water runoff from the plant drainage system. Table 9-1 lists the applicable surface water systems which are sampled and analyzed for gross alpha and gross beta levels to establish radioactivity concentrations at upstream and downstream locations from the site.

9.2.6 Ground Water Monitoring

Groundwater flow at the NFS site is to the north-northwest. Figure 9-2 depicts the water table surface for June 2008 and is representative of overall groundwater flow. To determine the impact of NFS operations on downgradient groundwater quality, one upgradient well and ten downgradient wells are monitored quarterly for isotopic uranium, isotopic plutonium, and Technetium-99. Current monitoring well locations are depicted in Figure 9-3. These monitoring well locations may be changed based upon the judgment of a qualified hydrologist/geologist employed or contracted by NFS and approval by the site environmental protection function manager or designee. Table 9-2 provides information on the sample type, collection frequency, and analysis. All groundwater analytical results are reviewed and evaluated.

9.2.7 Environmental Dosimeters

Environmental dosimeters are located both onsite and offsite 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 are 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 or potentially present at a given location.

Figure 9-2: Water Table Surface

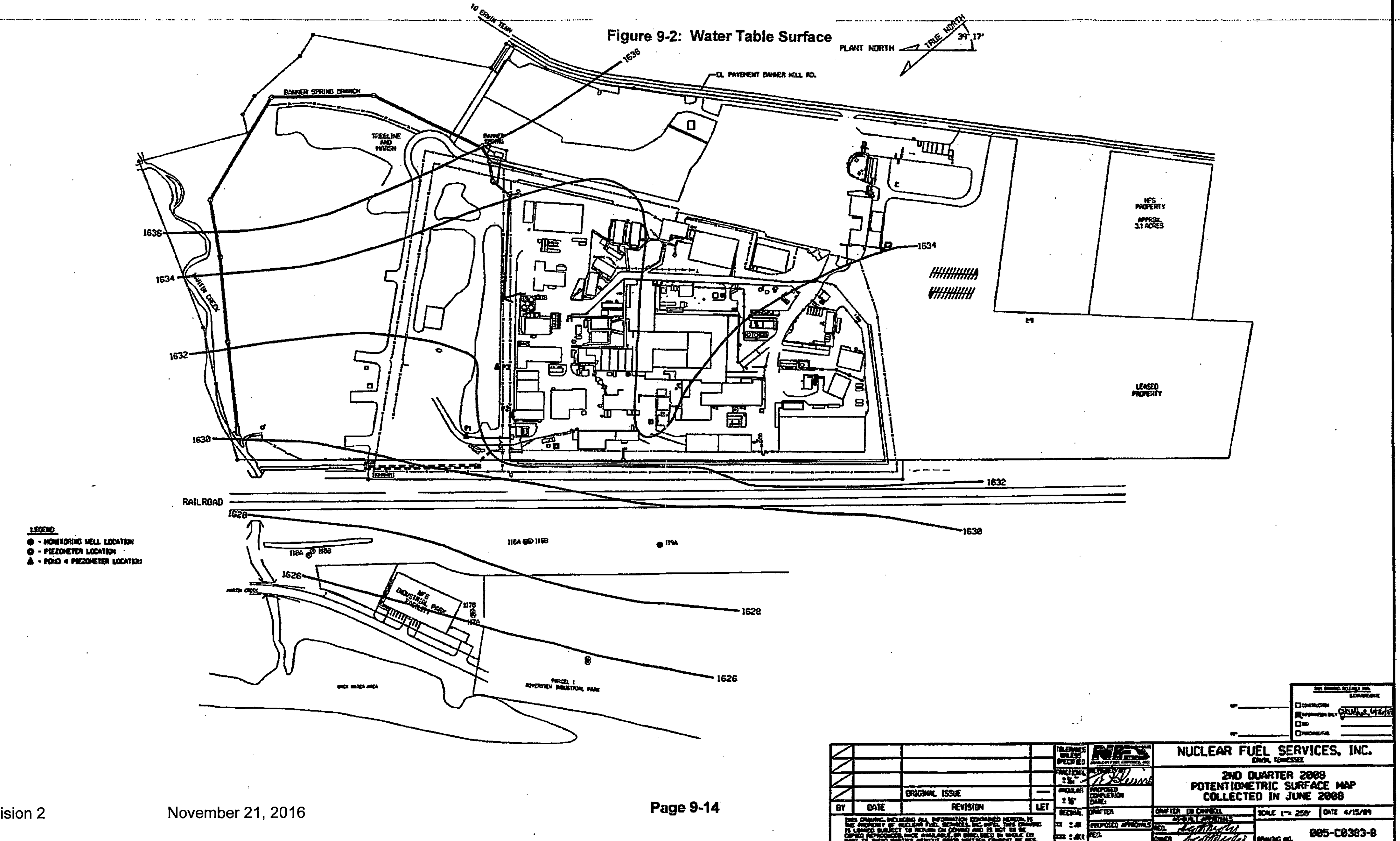

Figure 9-3
Site Groundwater Monitoring Well Locations

This drawing is "Official Use Only" and has been moved to the "Sensitive Information" ADDENDUM.

9.3 Quality Assurance of Radiation Measurements

All radiological measurements are performed with the objective of providing information that is accurate and precise, thus allowing valid assessments of plant impacts. Details of the NFS Quality Assurance Program are provided in appropriate procedures. These procedures incorporate the applicable elements of Regulatory Guide 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operations) - - Effluent Streams and the Environment," and Regulatory Guide 4.16, "Monitoring and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Processing and Fabrication Plants and Uranium Hexafluoride Production Plants." A brief summary of program elements is provided in the following sections:

9.3.1 Operating Procedures/Instructions

Procedures and/or other guidance documents are utilized to maintain the various components of a viable QA program. Procedures cover, but are not limited to, sample collection, preparation and analysis; calibration and maintenance of counting equipment and monitoring systems; reduction, evaluation, and reporting of data; quality control considerations; and general auditing concerns.

9.3.2 Records

Records are generated, updated, and retained to adequately document and ensure a reasonable QA program.

9.3.3 Quality Control in Sampling

Sampling of environmental media is undertaken in a manner to assure accuracy and representativeness of the samples. Samples are adequately labeled to guarantee proper identification. Processing and analysis of samples are conducted on a timely basis to ensure that proper sample integrity is maintained. Where storage is necessary, proper measures are taken to preserve the sample in consideration for the analyses that will later be required. Sampling systems are properly maintained and, as required, calibrated to assure operability.

9.3.4 Quality Control in the Laboratory

Laboratory instrumentation is maintained and calibrated in a manner that assures quality measurements. This includes, but is not limited to, the following: the use of NIST traceable standards (or their equivalent) which are appropriate for the

type of analysis undertaken, utilization of check sources on a reasonable basis to verify calibration and counter efficiency, background checks, usage of quality control samples (blanks, duplicates, and spiked samples) as appropriate to verify accuracy of measurements, interlaboratory analysis crosschecks as considered prudent, and a program for computational overchecks. Further, all contract laboratories must maintain adequate, verifiable QA programs.

9.3.5 Data Analysis and Review

Data from analysis of actual samples and QC measurement data are surveyed for accuracy and precision. When systems (either plant process systems or measurements analysis systems) are considered to be out of control on the basis of these data assessments, relevant investigations will be undertaken and steps taken to correct the problem(s).

9.3.6 Audits

Audits are performed to verify implementation of the quality assurance program.

9.4 Waste Minimization

It is the policy of NFS management to eliminate and/or minimize the generation of waste during planning, design, and operation of plant activities. All employees are expected to participate in waste reduction practices. Materials are recycled when judged to be reasonable and economical. The NFS Hazardous and Mixed Waste Reduction Plan is updated on an annual basis.

Attachment 8

SNM-124, Chapter 11

(28 pages to follow)

SPECIAL NUCLEAR MATERIALS LICENSE
SNM-124
CHAPTER 11

MANAGEMENT MEASURES
Table of Contents

SPECIAL NUCLEAR MATERIALS LICENSE
SNM-124
CHAPTER 11

MANAGEMENT MEASURES

Management measures are applied to activities involving the handling of SNM, generally on a continuing basis, to ensure protection of the safety and health of workers, the public, and the environment. As specified in 10 CFR 70.62(d), management measures are also applied to items relied on for safety (IROFS) to provide reasonable assurance that they remain available and able to perform their functions when needed.

The ISA Summary identifies IROFS applied to plant operating systems to assure those systems function within the performance requirements of 10 CFR 70.61. IROFS may be engineered controls (passive or active), enhanced administrative controls (active features that prompt a person to take an action), or administrative controls (actions of people). Management measures are applied to IROFS using a graded approach based on the type of control and the reduction of risk credited to that control. Methods used to select and assign management measures to IROFS are documented in written procedures.

11.1 Configuration Management

NFS maintains a Configuration Management (CM) Program to ensure the following objectives are met for selected structures, systems, and components (SSCs), processes, and activities managed by NFS:

- To establish consistency among design and regulatory requirements, physical configuration, and facility configuration information (FCI);
- To maintain this consistency throughout the life of the facilities and activities, particularly as changes are made, until the point that CM is no longer needed, and
- To help assure ongoing protection of the safety and health of workers, the public and the environment.

The NFS CM Program meets the requirements of 10 CFR 70.62(d), 10 CFR 70.64, and 10 CFR 70.72, the objectives and expectations of NUREG-1520, *Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility,* and incorporates key programmatic concepts recommended in ANSI/NIRMA CM 1.0-2007, *Configuration Management of Nuclear Facilities,* and DOE-STD-1073-2003, *Configuration Management.*

11.1.1 CM Policy

The NFS CM Policy requires establishment of an effective CM Program with clear objectives, defines the scope of the CM Program, documents NFS senior management commitment to CM, designates key NFS organizations with responsibility for implementing the CM Program, and describes the key CM Program functions. The Policy requires that, prior to implementation, changes to IROFS must be evaluated in accordance with the requirements of 10 CFR 70.72(a)(1)-(6), to determine if a license amendment is required in accordance with 10 CFR 70.72(b), and to determine if NRC approval is required in accordance with 10 CFR 70.72(c)(1)-(4). Per 10 CFR 70.72(d)(2), a brief summary of major changes that required revision of the applicable safety or environmental bases will be submitted within 30 days after the end of the calendar year during which the changes occurred.

The CM Program applies to IROFS contained in the Integrated Safety Analysis (ISA) and other structures, systems and components (SSCs) that are required to:

- Physically process, store or transfer more than 350 grams of U-235 as Special Nuclear Material (SNM) contained within the SSC at any given time. Specifically included are the active SNM processing facilities, the SNM storage vaults, the Waste Water Treatment Facility, associated Process Off-Gas Ventilation systems, and bulk chemical and gas storage and supply systems.
- Protect off-site and on-site personnel from nuclear and other hazards, as defined by the facility's ISA;
- Meet regulatory requirements for the physical protection of SNM;
- Protect the environment from significant damage or to satisfy environmental requirements or permits;
- Avoid substantial unplanned interruption of operations having significant cost or quality impact.

CM of computer programs and software applications is not within the scope of the CM Program with the exception of software contained in Programmable Logic Controllers (PLCs). CM of computer programs and software are addressed through the NFS Software Quality Assurance Program (SQAP) which is based on the sections of ASME NQA-1, *Quality Assurance Requirements for Nuclear Facility Applications,* applicable to software.

The IROFS and SSCs that are managed and controlled under the CM Program are identified as Configuration Items (CI). FCI to be managed under the CM Program includes information that reflects the design bases and requirements, performance criteria, physical characteristics, and regulatory requirements (as applicable) of IROFS and SSCs. FCI represents documents, drawings, procedures, and database information that are essential to operate, maintain, test, modify, repair and/or

replace, and substantiate the safety function(s) or operational/functional requirements of IROFS and SSCs.

The CM Program utilizes a graded approach to apply a level of resources and CM Program elements that are appropriate to the degree of risk to safety. Grading is used: (1) to help define which SSCs, facilities, processes, and activities will be subject to CM, and (2) to define the extent to which CM will be applied, e.g. the degree, rigor and extent to which applicable CM elements and requirements are applied to IROFS, SSCs, facilities, processes, and work activities. Application of the graded approach is based on:

- magnitude of any hazards involved
- magnitude of risks and consequences associated with design basis events
- relative importance of an IROFS to safety (risk and consequence reduction) and security
- importance of an SSC (or administrative control) to continued production operations
- type and technical characteristics of a facility or process
- facility or process operational status
- programmatic and technical issues
- existing programs and procedures

The CM function manager is responsible for the NFS CM Program and has direct responsibility for implementation and ongoing management of the program. All NFS employees, contractors, and organizations including Engineering, Safety, Production, Maintenance, Security, Quality Assurance, Training & Qualification, and Decommissioning are responsible for complying with the CM Program objectives and implementing Program requirements as an integral part of their respective areas of operation.

11.1.2 CM Program

The NFS CM Program includes the following seven (7) elements that are addressed in the following sections:

- Program Management
- Design Requirements
- Change Control
- Information Control
- Assessments
- Training
- Program Metrics

11.1.3 Program Management

The Program Management element identifies the NFS organizations and associated responsibilities for implementing and managing the CM Program. A CM function manager has been designated to assure the program is effectively implemented and maintained. The CM function is independent of the Production discipline.

The CM Program is applicable to all NFS organizations, including contractors, who perform construction, operation, maintenance, modification, and decommissioning activities associated with NFS facilities, SSCs and processes. Implementation of the CM Program is accomplished through procedures and instructions that delineate the responsibilities and actions of personnel to effectively implement the CM Program elements.

The Change Control Board (CCB) reviews change requests (CRs) to ensure that proper review and identification of items affected by changes have occurred. Typically the CCB serves as a "go/no go" decision maker and may approve, reject, defer or require alternative solutions for changes. The board is chaired by a senior manager and is comprised of other managers representing the Engineering, Safety, and Production disciplines. The chairman, other CCB members, and their alternates, are designated in writing. All CCB members, including the chairman, are required to have the qualifications specified for a discipline manager, as described in Chapter 2.

Other disciplines and organizations such as Material Control and Accountability (MC&A), Decommissioning, and Security are members of the CCB as ad hoc members, and are charged with review of change requests for potential impact to FCI and CI within their areas of expertise.

11.1.4 Design Requirements

The objectives of the Design Requirements element of the CM Program are to: 1) establish, document, maintain and communicate the design requirements and design bases associated with the CI managed under the CM Program, and 2) establish a design control process that effectively translates design inputs (design requirements and bases) into design outputs to implement approved changes and to control changes to design requirements and bases.

The CM Program defines design requirements as engineering or technical requirements reflected in design output information (documents and/or data) that define the form, fit and function of CI (including capabilities, capacities, physical sizes and dimensions, limits, setpoints, etc.) specified and/or approved by the Design Authority and derived from the design bases. The design bases are the set

of high-level functional requirements, interfaces and expectations of a facility or CI that are based on regulatory requirements, performance requirements and/or analyses. Each design requirement has a design basis whether documented or not.

The design requirements for CI for existing facilities are identified and documented both in electronic databases and/or in hard copy files. Where the design bases for the design requirements of existing CI may not be fully documented or readily available, the development and/or assembly (reconstitution) of the design bases is determined on a case-by-case basis in accordance with the graded approach discussed in Section 11.1.1 above.

For new facilities, processes/systems, and new CI, design requirements (and design bases) are required to be developed, reviewed, approved, and documented before input of SNM. As a minimum, the baseline design criteria (BDC) identified in 10 CFR 70.64(a) are addressed for IROFS.

Design requirements for CI are reviewed for adequacy (completeness, accuracy, and level of documentation available) when initially established or during changes when design requirements information must be developed. Design requirements (and bases) are approved by the engineering discipline manager as the Design Authority after completion of review and resolution of comments, as applicable, and concurrence by affected stakeholders, e.g., applicable Safety organizations. Review and approval of changes to design requirements (and bases) is conducted in accordance with the CM Program Change Control process (Section 11.1.5).

The level of review for design basis changes assures that all safety and technical aspects of proposed changes do not adversely affect the credited safety functions of IROFS or the operating and/or functional requirements of other CI. Where a change in design requirements does not affect the safety or design basis, the CM Program does not require a new design analysis to be performed; however, the affected design requirements are required to be updated, and the associated design bases must support any changes.

11.1.5 Change Control

The objective of the Change Control Process is to maintain consistency among design requirements, the physical configuration, and the related FCI, even as changes are made. The Change Control Process is used to ensure CI and FCI changes are properly reviewed, approved and implemented to assure that all impacts of proposed changes are identified and evaluated, design requirements (and bases) are maintained or appropriately revised, and changes are coordinated across the various NFS organizations and personnel responsible for activities and programs at NFS facilities.

The Change Control Process assures that the following items are addressed prior to implementing the change:

1. Description and reason for the change;
2. The technical basis for the change;
3. Identification of all CI and FCI impacted by the proposed change;
4. Modifications to existing operating procedures including any necessary training or retraining before operation;
5. Impact of the change on safety and health, or control of licensed material;
6. Authorization requirements for the change;
7. For temporary changes, the requested duration (e.g., expiration date) of the change;
8. The impacts or modifications to the ISA, ISA Summary, or other safety program information developed in accordance with 10 CFR 70.62; and
9. An evaluation per 10 CFR 70.72 as to whether or not a license amendment must be approved by the NRC prior to implementation of the change.

Requests for proposed changes to CI and FCI are required to be effectively documented, and this is procedurally accomplished by use of formal change requests (CRs). The Change Control Process is applied in a graded manner which categorizes changes as administrative, minor, and major changes. Administrative changes include inconsequential changes to FCI and pre-approved equivalent replacements of CI. Minor changes include initial equivalent replacements of CI, and addition, deletion, and modification of CI in existing process systems where the design requirements, safety bases, and process function(s) are not affected. Major changes include substantial modifications to existing licensed facilities and/or processes, new licensed facilities, or new processes in existing licensed facilities. Any change requiring a license amendment is also considered a major change.

Each CR is reviewed for completeness and accuracy, and a review is conducted to assure it is properly categorized. Administrative CRs are approved by the Engineering discipline manager/designee and do not require CCB review. The Engineering discipline manager/designee may approve minor CRs on behalf of the CCB, provided the change does not impact the safety or design bases. All major CRs are reviewed by the CCB.

The CCB may request review by other organizations as required, e.g., Security, QA, Decommissioning, MC&A, etc., depending on the level (minor or major) of change(s) to assure appropriate reviews are obtained. The need for safety reviews, e.g., radiological, nuclear criticality, industrial, fire, and environmental, for a proposed change is determined using a graded approach based on the type of CI(s) involved, associated risks, and type and extent of the proposed change.

Upon approval of a CR, a Change Control Package (CCP) is implemented based on the approved CR. The CCP is prepared consistent with the design control process (as applicable) and includes the approved CR, drawings, analyses, procedure changes, instructions, and/or other documents needed to properly review, implement, verify, and validate the proposed change. The CCP also defines the methods and acceptance criteria for applicable post-modification testing. CCPs are revised, updated, and supplemented as necessary during the review process, and contain a copy of all approvals. Any changes to the approved CCP that may be required are prepared, reviewed, and approved at the same level as the original CCP.

Each CCP is formally reviewed which includes a technical review, a safety review, and a management review. Other reviews may be performed as needed for such items as meeting regulatory or contractual requirements, cost/benefit, or schedule impact.

The CM function manager is responsible for ensuring that CCPs are completed and stored in an information control system. The CM function manager is also responsible for tracking the changes to associated FCI to assure they are completed and documents are updated.

Operational configuration is defined in the NFS CM Program as the "state" (e.g., on/off, open/closed, operating/not operating) of facility SSCs and processes at a particular point in time. Operational configuration information is that FCI which describes the acceptable SSC or process configurations when variable configuration conditions may exist based on operational or other needs. The CM Program requires that all variable IROFS, SSC, and process configurations (e.g., allowable "states"), together with their associated FCI, be reviewed and approved prior to use or implementation to assure they are within approved design requirements at all times. Any configuration changes, whether temporary or permanent, not covered by procedures (i.e., not pre-approved) are treated as changes which must go through the Change Control Process. Technical, independent, and safety reviews of the change, including procedures and other related FCI, specifically assure that the facility SSC or process will continue to operate safely and provide adequate protection to workers, the public, and the environment, and that IROFS are not prevented from performing their expected safety functions and/or sufficient compensatory measures are established.

11.1.6 Information Control

The objective of the Information Control element is to identify and manage FCI (both electronic and documents) related to the physical configuration and design requirements. Information control helps ensure that:

- important facility documents and information are properly stored;
- revisions to FCI are controlled, tracked, and completed in a timely manner;
- revised documents and information are formally distributed or made available to designated users; and
- information concerning pending revisions is readily available.

The most typical FCI documents and information include:

- ISAs and ISA Summaries
- Process Hazard Analyses (PHAs)
- Documents that identify or define design requirements
- Documents that demonstrate compliance with design and licensing requirements
- Design specifications and/or calculations
- Safety analyses (ACE, FHA, NCSE, etc.)
- IROFS
- Physical item databases
- Change process documentation
- Software logic and manuals for operation and maintenance of critical software (e.g., Programmable Logic Controllers or PLCs)
- Key operating and test procedures
- Key drawings
- Audit and assessment results
- Vendor technical information

An information control system is established to create, control, and track documents within the CM Program. Only the most recently approved versions of FCI are used in the process of operating, maintaining, and modifying the SSCs, facilities, and processes. As controlled information is updated to reflect changes to the requirements and/or physical installation, the CM Program ensures that updated FCI documents are uniquely identified, include a revision number and/or date, and any outdated documents and information are replaced by the latest approved versions.

FCI documents, drawings, and copies are maintained in accordance with procedures that facilitate retrievability and use, control classified information, and meet record keeping requirements. Electronic versions of classified and unclassified FCI are maintained on secure servers and made available to authorized users.

11.1.7 Assessments

The objective of the CM Program Assessment element is to detect, document, determine the cause of, and initiate correction of inconsistencies among design requirements, FCI, and physical configuration. The assessments help identify

inconsistencies between these areas, evaluate the root causes for these problems, and prescribe improvements to avoid similar inconsistencies in the future.

The effectiveness of different aspects of the CM Program is assessed through physical configuration assessments, design assessments, post-implementation inspections and tests, and periodic performance assessments. Where practical, CM related reviews and assessments are combined with other periodic assessments of facilities, processes, and activities for efficiency and cost-effectiveness. All or part of the assessment of the adequacy of CM for an activity or process may be integrated into broader management and performance assessments. The results of these assessments are documented and maintained in accordance with written procedures.

NFS assures that the persons performing the assessment activities are qualified, and that any NFS personnel performing assessments have sufficient authority and freedom from line management to objectively conduct the assessments.

Assessment findings are documented as open items in the corrective action program as CM issues if they are validated to involve contradictory information among different FCI, unanswered technical questions, and/or missing, undocumented or inaccurate information.

11.1.8 Training

The objective of the CM Program Training element is to provide adequate assurance that facility personnel are aware of the CM concepts, terminology, definitions and procedures. Training will ensure that workers have an understanding of how their actions impact CM and that they are able to properly carry out their work in a way that helps NFS achieve its objective to maintain consistency between the design requirements, the FCI and the physical configuration.

11.1.9 Performance Metrics

The CM function manager tracks CI and FCI changes in progress. The status, rate of change, and backlogs are periodically reviewed to demonstrate CM Program efficiency and effectiveness.

CM-related issues entered into the corrective action program are also reviewed to determine trends and CM Program effectiveness.

11.2 Maintenance

NFS has a maintenance program designed to ensure that IROFS are maintained in a manner so as to ensure they are capable of performing their intended function when called upon. An essential element of the maintenance program requires that all maintenance activities, including functional testing of IROFS, are authorized by written procedures and/or written instructions to which appropriate personnel have been trained. Steps are included within the maintenance procedures for the notification of all affected parties before and at the completion of all maintenance activities.

The maintenance program consists of several key program elements including management systems that provide the scheduling and documentation of the following maintenance elements when applied to IROFS:

1) Surveillance and Monitoring,

2) Corrective Maintenance,

3) Preventive Maintenance, and

4) Functional Testing.

Maintenance skills training for mechanics performing maintenance activities involving IROFS is also required. Maintenance skills training is further addressed in Section 11.3. Contractors that perform work on IROFS will meet the same guidelines for IROFS training or will be under direct supervision of NFS-trained personnel that are qualified and knowledgeable of the particular IROFS involved.

11.2.1 Surveillance and Monitoring

NFS utilizes established surveillance activities to monitor the current and long-term performance of IROFS. These activities include preventive maintenance (11.2.3), functional testing (11.2.4), and follow-up to corrective maintenance (11.2.2). IROFS found to be out-of-tolerance or unable to perform their intended function are reported in a timely manner to the safety discipline through the corrective action program.

Reports of IROFS failures are entered into the corrective action program which provides a means to evaluate the failure of IROFS, identify the cause of failure, and assign appropriate corrective actions to be initiated. Records of IROFS performance issues and corrective actions are maintained within the maintenance and corrective action programs, as applicable. Records for failures of IROFS are maintained in accordance with 10 CFR 70.62(a)(3) within the corrective action program.

11.2.2 Corrective Maintenance

Corrective maintenance is performed using a planned, systematic, integrated, and controlled approach to ensure that IROFS and other systems necessary for the safe operation of the facility are properly repaired and restored to service in a manner that maintains facility safety and the function of the safety system. Maintenance activities are performed on IROFS in a manner that minimizes or eliminates the recurrence of unacceptable performance deficiencies.

Corrective maintenance is authorized, initiated, and documented through a formally established process that includes steps requiring coordination between the maintenance and operating organizations. The process also includes an evaluation to determine if IROFS have been, or may be, affected by the equipment failure/malfunction or the ensuing maintenance and whether post-modification functional testing of IROFS is required.

11.2.3 Preventive Maintenance

Preventive maintenance (PM) is performed in a preplanned and scheduled manner to refurbish or overhaul IROFS to ensure that they continue to perform their intended function. PM activities are appropriately balanced against the objective of minimizing unavailability of IROFS. After conducting PM, and before returning a safety control to service, a functional test may be required to provide reasonable assurance that the safety control performs as designed and provides the safety action expected.

A schedule for performing PM on IROFS is maintained as specified in written procedures, and frequencies are established based on operating history, manufacturer and industry guidance, feedback from surveillance and maintenance activities, and/or recommendations from the corrective action program.

11.2.4 Functional Testing

Functional testing of IROFS under the Safety Related Equipment program is performed using approved, written instructions prior to startup of facilities or process operations involving IROFS and at periodic intervals during operations to provide reasonable assurance that the safety control performs as designed and provides the desired safety action. Functional testing of IROFS will be performed prior to restart if the process operation has been inactive for more than 120 days.

Functional test instructions and frequencies are approved by, and cannot be modified without the approval of, the safety review committee, and are based on

operating history, manufacturer and industry guidance, risk assessment, feedback from surveillance and maintenance activities, and/or recommendations from the corrective action program. Minor changes to functional test instructions, as defined in a written procedure, are allowed to be approved by the safety review committee chairman on behalf of the entire committee. During process operations, compensatory measures are used as appropriate while functional testing is performed on IROFS. The results of functional testing are documented and the documentation is maintained as "records pertaining to safety" as specified in Section 11.7.

11.3 Training and Qualification

The NFS Training and Qualification Program provides workers with the knowledge and skills to safely perform their job function, recognize the importance of IROFS, effectively deal with the hazards of the workplace, and properly respond to emergency situations. The qualification aspect of this program ensures that operations and maintenance are performed only by properly trained personnel.

Requirements and methods for the training and qualification programs are approved by NFS site management, who also provide ongoing evaluation of the effectiveness of the programs. Training records, including those related to IROFS, are maintained for a minimum of two years.

This training typically falls into one of two categories:

1) General safety training not specific to a particular work station or activity; and
2) Training to assure proper performance for positions and work activities that are relied on for safety, in particular those designated as IROFS.

11.3.1 General Safety Training

The NFS Training and Qualification Program requires that all personnel who are granted unescorted access to the protected area receive formal safety orientation training. Safety orientation training covers plant safety rules, radiological, nuclear criticality, industrial, and environmental safety topics as appropriate to the job function of the individuals being trained. In addition, this training covers proper response to emergencies.

Previously trained employees receive formal refresher training in safety on an annual basis. The content of safety training is evaluated biennially by a member of the safety discipline, as appropriate for the subject of the training, to ensure it remains current and relevant. Additional details regarding periodic evaluation of the Radiation Protection Training Program is discussed in Chapter 4.

11.3.2 Training and Qualification for Activities Involving the Handling of SNM

The Training and Qualification Program includes work training for operating personnel and others who directly handle greater than laboratory sample quantities of special nuclear material. Work training typically includes classroom, on-the-job, and guided-work-experience training necessary to provide the desired knowledge and/or skill. It covers the operating procedures, alarms, emergency response actions, and radiological, nuclear criticality, industrial, and environmental safety controls and limits specific to the particular work assignment.

Work training includes appropriate reinstruction for previously qualified individuals prior to implementation of a process change or procedural modification. When changes are made relative to safety or emergency response requirements, provisions are made to assure that affected employees are appropriately informed and instructed on the changes. Previously qualified individuals are required to undergo a re-qualification process for applicable work assignments every three years (maximum interval not to exceed 42 months). Additional details about the work training program are provided in approved written procedures.

The Training and Qualification Program provides for the instruction and training of mechanics involved in maintenance activities at NFS. Maintenance skills training may include such topics as basic math, precision instrument reading, laser alignment, vibration analysis, basics of programmable logic controllers (PLC), welding, industrial electricity (basic, intermediate, and advanced), and machine tool operation, as appropriate. The type and level of training is commensurate with the job assignments.

Organization and Management of Training

The responsibility for the assurance of properly trained and qualified personnel resides with the discipline management team and pertinent line management. Support to line management for the development, implementation, and administration of the facility Training and Qualification Program is provided by the Training function. Implementation of the Training and Qualification Program is accomplished in accordance with written procedures.

All training is conducted by, or under the supervision of, individuals recognized by management as possessing the necessary knowledge and skills to conduct the training. Exemptions from training are only authorized as described in approved written procedures.

Identification of Activities Requiring Training

Positions impacting the availability/reliability of IROFS are assessed, based on a graded approach that considers the hazards and the safety responsibilities associated with each position. Input from subject matter experts, with support from the training function, is utilized as appropriate.

Position Training Requirements

Objectives and requirements for training programs are jointly agreed upon by NFS management based upon plant needs and input provided by the training function and the appropriate discipline.

Each NFS position involving personnel assigned to SNM process operations is evaluated to determine the specific requirements that apply to the defined job function. The requirements are defined in an on-line computer database. Personnel must remain current on the defined set of requirements to maintain job qualifications.

Bases for Training

The objective of training is to ensure safe and efficient operation of the facility and compliance with applicable established regulations and requirements. Learning objectives are established for those positions/activities impacting the safety of licensed material operations, and in particular the availability/reliability of designated IROFS. Objectives include, as applicable, the knowledge, skills, and abilities the trainee should demonstrate; the conditions under which required actions will take place; and the standards of performance the trainee should achieve upon completion of the training activity.

Training Materials

Lesson plans and other training guides (for both classroom and on-the-job training) developed for activities relied on for safety are based on learning objectives developed from specific job performance requirements. Information provided by various safety disciplines is included in the content of training elements with clearly defined objectives. The lesson plans also provide reasonable assurance that training is conducted in a reliable and consistent manner. The Configuration Management Program provides a means to assure that design changes and modifications to IROFS are accounted for in the training.

Evaluation of Trainee Accomplishment

Trainee understanding and command of learning objectives are evaluated. The evaluation may be accomplished through a combination of observation/skills

demonstration, written tests, or oral examinations. The results of trainee evaluations are documented.

On-the-Job Training (OJT)

OJT requirements for activities relied on for safety and listed in the ISA Summary, if applicable, are specified as part of pertinent position training requirements. Completion of OJT may be demonstrated by actual task performance (preferred) or task simulation. Completion of OJT requirements are documented.

Training Program Review

The effectiveness of the Training and Qualification Program is assessed on a periodic basis. Work assignments involving the handling of SNM are evaluated for needed re-current training and/or re-evaluation of qualification activities.

11.4 Procedure Development and Implementation

Activities involving the handling of SNM and/or IROFS are conducted in accordance with written procedures as defined in this section. NFS procedures also address the following activities: design, configuration management, procurement, construction, radiation safety, maintenance, quality assurance, training and qualification, audits and assessments, incident investigations, records management, nuclear criticality safety, fire safety, chemical process safety, and reporting requirements.

The process for the development and implementation of procedures is defined in written procedures. These procedures address how procedures are developed, reviewed, approved, distributed, revised, and deleted. The system ensures that the most current revisions of procedures are readily available to workers within their work areas (operating procedures), or in a centralized location accessible to all affected personnel (general safety and support group procedures), that any necessary training and qualification requirements are identified, and that the timeframe for which the procedure is valid is defined.

The safety review committee is responsible for reviewing and approving new and revised operating and general safety procedures, as defined below, and per Section 2.4 requirements. Support group procedures, including those developed to support management measures, are approved by the discipline manager for the originating group and by the appropriate safety function manager(s) if the procedure contains safety-related information.

Changes and/or revisions to procedures covering licensed material operations and/or IROFS are reviewed by the safety functions, as appropriate, in accordance with the requirements of the CM program, as discussed in Section 11.1, to ensure

that all associated activities and documentation (safety analyses, reviews, testing, training, etc.) are completed before procedural changes are implemented.

11.4.1 Operating Procedures

Operating procedures are documents written to authorize a) the processing of radioactive material or b) a decommissioning activity; and, within these documents, detailed instructions for operation of equipment used in the process or activity, instructions for disposition of radioactive wastes, and limits and controls established for safety purposes, including IROFS, are identified. Operating procedures may take various forms (e.g., standard operating procedures, special work instructions, etc.).

Operating procedures include provisions to place process operations in a safe condition if a step of the procedure cannot be performed as written. Work place posting of limits and controls, training, and other communication devices are used, if appropriate, to enhance comprehension and understanding of operating procedures.

During operating procedure development, the technical accuracy is verified. Changes to existing operating procedures are evaluated to determine if the scope of the change warrants a walk-down and/or an independent verification/validation. New operating procedures are validated by operations staff through walk-downs to ensure that they can be performed as written. An independent verification/validation review may also be performed to provide additional assurance that the technical information, including formulas, set points, and acceptance criteria, is all there and is correct, and may include a tabletop walkthrough or a walk-down of the procedure in the field.

11.4.2 General Safety Procedures

General safety procedures outline health and safety practices that help maintain occupational radiation exposures at levels as low as reasonably achievable (ALARA). These procedures are generally applicable on a plant-wide basis such as those governing the collection of bioassay samples, contamination control, emergency evacuation, and other similar matters. Included in this category are the Emergency Plan implementing procedures and the Criticality control procedures. General safety procedures are reviewed and approved by the safety review committee and such other discipline managers as deemed necessary by the safety discipline director.

11.4.3 Support Group Procedures

Support group procedures are documents written to authorize the conduct of activities that are not directly involved in the processing of radioactive material or a decommissioning activity, but may involve radioactive material (e.g., laboratory analytical procedures, safety monitoring procedures, material control and accountability procedures, etc.); and, within these documents, the activities are described and any special safety precautions are identified.

11.4.4 Maintenance Procedures

Significant maintenance activities are conducted under formal work packages (FWPs) which are reviewed by the safety functions, as appropriate, prior to initiation of the work. Each FWP prescribes the controls necessary to provide for safety. Items such as the release of airborne radioactivity; unusual exposure of personnel; draining, disassembling, modifying, or routing of lines to equipment that may contain special nuclear material are considered during the review.

Although FWPs are not considered to be operating procedures as defined in this section, a written and safety review committee approved procedure is in place to give instructions when an FWP is required and whose approval is required.

The following methods/practices, as applicable, are incorporated into programs, systems, or written procedures regarding maintenance of IROFS:

- Authorized maintenance instructions with identification of the IROFS;
- Parts list for IROFS;
- As-built or red-lined drawings;
- Pre-maintenance review of work to be performed on unique or complex IROFS, including procedure reviews to ensure accuracy and completeness;
- Notification before conducting repairs/maintenance or removing an IROFS from service, including notification instructions and the functional discipline(s) that will be notified;
- Safe Work Practices/Permits (e.g., lock-out/tag-out; confined space entry; nuclear, radiation, environmental, fire, and chemical safety issues);
- Requirements for replacement of like-kind parts and control of new or replacement parts;
- Compensatory measures while performing work on IROFS;
- Procedural control of removal of components from service for maintenance and for return to service;
- Ensuring safe operations during removal of IROFS from service; and,
- Notification to operations personnel that repairs have been completed.

11.4.5 Temporary Procedures

Approved temporary procedures (i.e., Letter of Authorization (LOA)) are used when permanent procedures do not exist to:

1) Direct operations during testing, maintenance, and modifications;

2) Provide guidance in unusual situations not within the scope of permanent procedures; or,

3) Provide assurance of orderly and uniform operations for periods of short duration when the plant, a system, or a component is performing in a manner not covered by existing permanent procedures, or has been modified or extended in such a manner that portions of existing procedures do not apply.

Temporary procedures are controlled, reviewed, and approved as specified by a written procedure and will not change an ISA except as authorized under 10 CFR 70.72. The review and approval process required for temporary procedures is the same as for other procedures, and a timeframe is defined for which the procedure is valid.

11.4.6 Periodic Reviews of Procedures

If an active operating or general safety procedure has not been revised within a three-year period, the procedure will be reviewed to ensure it remains current and relevant. The chairman of the safety review committee may select individual members to perform the review, rather than the entire committee. The selection process is described in Section 2.4. Any general safety procedure meeting this condition will also be reviewed by the appropriate safety function manager(s). Support group procedures are periodically reviewed in accordance with the Audits and Assessments program (see Section 11.5). Emergency procedures are reviewed per the Emergency Plan required in Chapter 8.

The corrective action program includes provisions to assess the role of procedures in adverse conditions or events evaluated within the program. Corrections of procedural deficiencies are tracked to completion within the system.

11.5 Audits and Assessments

NFS has a program for conducting audits and assessments of activities significant to facility safety and environmental protection that identifies responsibility for:

1) Determining the appropriate utilization of internal and/or external personnel for particular audit and assessment activities;
2) Assuring audit and assessment personnel have the expertise and background sufficient to successfully conduct audit and assessment activities;
3) Assuring audit and assessment personnel are sufficiently independent of the area being reviewed; and,
4) Verifying the utilization of an effective corrective action program to address findings of audits and assessments.

Written guidance and procedures used to perform the audits and assessments contain the following information:

- Activities to be reviewed;
- Frequency of reviews;
- Applicable guidance to be used in conducting the reviews;
- Responsibilities for each phase of the reviews;
- Instructions for recording the results, and recommending and approving actions to be taken; and,
- The levels of management to which results are reported.

Results, including findings and observations, are captured in the corrective action program. Corrective actions to prevent recurrence are assigned to owners, documented, and tracked to completion in accordance with the requirements specified in the corrective action program.

11.5.1 Audits

Audits are compliance-based evaluation activities with an objective of verifying compliance of operations with established regulatory requirements, license commitments, and standard industry practice. As a minimum, audits apply to the programs presented below.

Members of each safety function perform audits of activities involving the handling of SNM, including support areas, on a quarterly basis.

The Emergency Preparedness Program is audited by the Quality Assurance discipline on an annual basis. The following management measures are audited on a biennial basis by the Quality Assurance discipline:

- Maintenance;
- Procedures;
- Configuration Management;
- Training and Qualifications;
- Incident Investigations;

- Records Management; and
- Quality Assurance elements for IROFS.

Members of the Quality Assurance discipline periodically audit safety programs as directed by the president and/or a discipline director.

11.5.2 Assessments

Assessments are performance-based evaluation activities conducted to assess the effectiveness of health, safety, and environmental compliance functions in achieving their designated purpose, particularly in providing reasonable assurance of the availability and reliability of IROFS.

As a minimum, these assessments apply to the Safety discipline functional areas and the Configuration Management function described in Chapter 2. Assessments in these areas are performed on a triennial basis. The need for assessments of emergency preparedness, maintenance, procedures, training and qualifications, incident investigations, records management, and Quality Assurance elements for IROFS will be determined at the discretion of the president and/or a discipline director after considering plant activities and the results of periodic audits of these areas.

11.6 Corrective Action Program

11.6.1 Corrective Action Program

NFS implements, through written procedures, a corrective action program to investigate and document events for operations involving special nuclear materials, including those required to be reported under 10 CFR 70.50, 70.62, and 70.74. Events, including those with conditions adverse to safety, are reported, investigated, tracked, and corrective actions are assigned through a formal corrective action program. A graded, risk-based approach is used to establish the requirements for determining specific or generic root cause(s) and generic implication of events.

Events are reviewed and classified using a graded, risk-based approach that is guided by risk-tables in implementing documents. The criteria for classifying conditions adverse to safety takes into consideration safety significance and regulatory compliance, including the impact on the health and safety of the public and the environment (i.e., for a chemical spill – type of chemical, spill volume, spill location); impact on reliability or availability of equipment/facilities (i.e., IROFS failure or degradation, customer product quality); and impacts to regulatory commitments.

A multi-disciplinary committee provides further review to ensure proper classification, and based on the significance of the issue, may initiate an investigation to determine the root cause of the condition. A graded, risk-based approach is applied to the assignment of the level of investigation; and, based on severity or potential severity of the event, the investigation may be conducted by one or more individual(s). Levels of investigation, as well as reviews and approvals, are assigned for events in accordance with written procedures. Corrective actions are developed, approved, and implemented. Measures to prevent recurrence and/or to control the affected work in progress may also be taken.

11.6.2 Incident Investigations

The guidance for conducting an investigation contains the following elements:

1. A documented approach for investigating an event, separate from any required Emergency Plan. The investigation of an event should begin as soon as possible, commensurate with ensuring the safety of the investigator(s), after the event has been brought under control.

2. A description of the functions, qualification, and responsibilities of the individual who would lead the investigative team and those of the other team members; the scope of the investigator(s) authority and responsibilities; and assurance of the cooperation of management.

3. Assurance of the investigator(s)' authority to obtain all the information considered necessary and its independence from responsibility for or to the functional area involved in the incident under investigation.

4. Procedures requiring maintenance of all documentation relating to events for two years (or for the life of the operation), whichever is longer.

5. Guidance for personnel conducting the investigation on how to apply a graded, risk-based approach using reasonable, systematic, structured methods to determine, as warranted by the risk, the specific or generic root cause(s) and generic implications of the problem. The level of investigation is based on a graded approach relative to the severity of the event.

6. Requirements to make available original investigation reports to NRC on request.

7. A system for monitoring the completion of appropriate corrective actions.

8. Direction for ensuring that documented corrective actions are taken within a reasonable period to resolve findings from event investigations.

An investigation will be initiated for those events specified in 10 CFR Parts 70.50, 70.62, and 70.74 within 48 hours of discovery, or sooner, based on the safety significance of the event.

For team investigations, the team will include at least one individual knowledgeable of the area being investigated (as applicable) and at least one team member trained in root cause analysis. In addition, the investigation process and investigator(s) will be independent of the line management, and participants are assured of no retaliation for participating in investigations.

Corrective actions are documented and monitored through completion. A graded, risk-based approach is applied to prioritize completion of corrective actions so that conditions adverse to safety are corrected as soon as practicable. The process used to monitor corrective actions also includes verification of completion, and as applicable, reviews of effectiveness and management attention for those corrective actions deemed ineffective. Corrective actions generated from investigations are used to make corrections and improvements (i.e., "lessons learned") necessary to prevent or minimize single or common-mode failures. Details of the accident event sequence(s) will be compared with accident sequence(s) already considered in the ISA, and the ISA Summary will be modified to include evaluation of the risk associated with accidents of the type actually experienced.

Auditable records and documentation related to events, investigations, and root cause analysis are maintained as described in written procedures. For each event utilizing a team investigation, the incident report will include a description of the event, contributing factors, a root cause analysis, and findings and recommendations. Relevant findings are communicated to affected personnel, including appropriate levels of management. A database of events, investigations, and corrective actions is maintained for tracking, trending, and documentation purposes.

Trends involving conditions adverse to safety, including failure of IROFS, are reviewed to determine effectiveness of safety systems and to provide feedback to management for establishment of actions to minimize and/or prevent recurrence. Adverse trends are entered in the system as events and are classified using the same graded, risk-based approach used to classify events.

11.7 Records Management

A records management system, as applied to safety (i.e., ISA, radiation protection, nuclear criticality safety, chemical process safety, fire safety, emergency preparedness, and environmental protection), decommissioning, and quality assurance activities, is maintained in accordance with written procedures.

Information related to 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 license conditions and the relevant regulatory requirements of 10 CFR 20.

All records pertaining to safety will be retained for at least two years unless longer retention is required by other regulatory or license specifications. For example, records of major changes implemented under 10 CFR 70.72 will be maintained until termination of the license. Major changes are defined in 11.1.5.

Records relevant to IROFS that are maintained include the following:

- Construction specifications,
- Facility and equipment descriptions and drawings,
- Design criteria requirements,
- Records of facility changes,
- Safety analyses, reports, and assessments, including the ISA and ISA Summary,
- Procurement, including specifications for IROFS,
- Configuration Management (physical configuration of process designs, validation records for computer software, as appropriate),
- Maintenance (calibration, preventive/corrective maintenance [including schedules, test data for IROFS]),
- Training and Qualification,
- Procedures,
- Audits and Assessments/Inspections,
- Incident Investigations (investigation reports), and
- Failures of IROFS.

Records management procedures (a) assign responsibilities for records management, (b) specify the authority needed for records retention or disposal, (c) specify which records must have controlled access and provide the controls needed, (d) provide for the protection of records from loss, damage, tampering, theft, or during an emergency, and (e) specify procedures for ensuring that the records management system remains effective.

A functional organization is in place to ensure prompt detection and correction of deficiencies in the records management system or its implementation. The records management procedures shall provide the following instructions to ensure that:

- Records are prepared, verified, characterized, and maintained;
- Records are legible, identifiable, and retrievable for their designated lifetimes;
- Records are protected against tampering, theft, loss, unauthorized access, damage, or deterioration for the time they are in storage; and,

- Procedures are established and documented specifying the requirements and responsibilities for record selection, verification, protection, transmittal, distribution, retention, maintenance, and disposition.

Records are categorized by their relative importance to safety and/or regulatory compliance to identify record protection and storage needs and to designate the retention period for individual kinds of records.

For computer codes and computerized data used for activities relied on for safety, as specified in the ISA Summary, procedure(s) are established for maintaining readability and usability of older codes and data as computing technology changes. The procedures should include transfer of the older forms of information (e.g., punched cards or paper tapes) and codes for older computing equipment to contemporary computing media and equipment.

In addition, records of IROFS failures must be kept and updated in accordance with 10 CFR 70.62(a)(3). Record revisions necessitated by post-failure investigation conclusions should be made within 5 working days of the completion of the investigation.

11.8 Other QA Elements

The NFS quality system consists of the organizational structure, procedures, processes, and resources needed to implement quality management. The following elements, as appropriate, are applied on individual projects using a graded approach based on the degree of importance to safety.

1. Organization and Responsibilities
   Chapter 2 provides the commitments associated with the organizational structure, authority, and responsibilities to ensure that activities involving the handling of SNM and/or IROFS are performed safely and in compliance with license and regulatory requirements.

2. Quality Assurance Program
   NFS maintains a QA Program that is structured on ASME NQA-1 (*Quality Assurance Program Requirements for Nuclear Facilities*) under the overall responsibility of the Quality Assurance discipline. Aspects of this program may be applied to IROFS using a graded approach based on the degree of importance to safety.

3. Design Control
   Design control is an element of the Configuration Management Program as described in Section 11.1. Information control of items such as design inputs,

analyses, and other design documentation and records is discussed in Section 11.1.6.

4. Procurement Document Control
The Purchasing Program has provisions to ensure that applicable design bases and other requirements necessary to provide reasonable assurance of quality are included or referenced in documents for procurement of items or services relied on for safety. To the extent necessary, suppliers are required to have QA consistent with the quality level of the item or service to be procured.

5. Instructions, Procedures, and Drawings
Section 11.4 includes the commitment that "activities involving the handling of SNM and/or IROFS are conducted in accordance with written procedures". This section also describes the process for developing and implementing procedures. Drawings are controlled under the Configuration Management Program as described in Section 11.1.6.

6. Document Control
A process is in place for developing, implementing, and revising documents to provide reasonable assurance that the appropriate documents are in use (refer to Sections 11.1 and 11.4). Document changes are reviewed for adequacy and approved for implementation by authorized personnel.

7. Control of Purchased Items and Services
The Purchasing Program has provisions for purchase specifications that define the necessary requirements for controlling purchased material. The program provides reasonable assurance of conformance with specified requirements, and it also allows for appropriate receipt inspection, storage, and shelf life requirements for materials.

8. Identification and Control of Items
The Configuration Management and Purchasing Programs have provisions for identifying and controlling items, including IROFS, to provide reasonable assurance that incorrect or defective items are not used.

9. Control of Special Processes
Section 11.4 includes the commitment that "activities involving the handling of SNM and/or IROFS are conducted in accordance with written procedures". This commitment also applies to special processes such as welding, nondestructive testing, and chemical cleaning. Such activities are performed by qualified personnel using approved procedures and equipment.

10. Inspection
Acceptance testing and/or inspection is a part of the Configuration Management Program which ensures that IROFS meet specified requirements prior to initial use. The Surveillance and Monitoring, Preventive Maintenance, and Functional Testing functions, as described in Section 11.2, provide assurance that IROFS continue to meet specified requirements by assuring that these testing and inspection activities are scheduled and implemented.

11. Test Control
Acceptance testing and/or inspection is a part of the Configuration Management Program which ensures that IROFS meet specified requirements prior to initial use. The Surveillance and Monitoring, Preventive Maintenance, and Functional Testing elements, as described in Section 11.2, provide assurance that IROFS continue to meet specified requirements by assuring that these testing and inspection activities are scheduled and implemented.

12. Control of Measuring and Test Equipment
The Calibration Program has provisions to provide reasonable assurance that tools, gauges, instruments, and other measuring and testing devices are properly identified, controlled, calibrated, and adjusted at appropriate intervals to maintain performance within required limits.

13. Item Handling, Storage, and Shipping
The spare parts program has provisions to ensure that items are stored in such a manner as to prevent damage, loss, or deterioration caused by environmental conditions. Testing conducted prior to initial use should detect potential damage incurred during shipping, handling, or storage.

14. Inspection, Test, and Operating Status
Acceptance testing and/or inspection is a part of the Configuration Management Program which ensures that IROFS meet specified requirements prior to initial use. The Surveillance and Monitoring, Preventive Maintenance, and Functional Testing elements, as described in Section 11.2, provide assurance that IROFS continue to meet specified requirements by assuring that these testing and inspection activities are scheduled and implemented. The Configuration Management and Purchasing Programs have provisions for identifying and controlling items, including IROFS, to provide reasonable assurance that incorrect or defective items are not used.

15. Control of Nonconforming Items
The Configuration Management and Purchasing Programs have provisions for identifying, segregating, and controlling items, including IROFS, to provide reasonable assurance that incorrect or defective items are not used.

16. Corrective Action
Reports of conditions adverse to safety are promptly identified and entered into the Corrective Action Program (see Section 11.6), which provides a means to evaluate the problem, identify the cause of the problem, assign appropriate corrective actions to be initiated, and track the corrective actions to closure. Prompt identification and effective corrective actions should provide reasonable assurance that repetition of the problem will be minimized.

17. Quality Assurance Records
The Records Management Program, as described in Section 11.7, has provisions for the identification, retention, retrieval, and maintenance of records that furnish evidence of the control of quality of IROFS.

18. Audits
Section 11.5 includes the commitments for scheduling and implementing audits and assessments.

19. Updates of QA Documents
The management positions for each discipline are responsible for reviewing and updating quality program documents, as applicable, based on reorganizations, revised activities, lessons learned, changes to applicable regulations, and other quality program changes.