



NUCLEAR FUEL SERVICES, INC.
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ACF-11-0161
May 13, 2011

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

- Reference:
- 1) Docket No. 70-143; SNM License 124
 - 2) Letter from B. Marie Moore to the NRC, dated June 30, 2009, (21G-09-0104), Renewal of Special Nuclear Material (SNM) License 124
 - 3) Letter from NRC to Mark P. Elliott, dated June 15, 2010, (TAC No. L32830), Request for Additional Information Concerning License Renewal
 - 4) Letter from Mark P. Elliott to NRC, dated August 16, 2010, (21G-10-0163), Response to the Request for Additional Information Concerning License Renewal for SNM-124
 - 5) Letter from Mark P. Elliott to NRC, dated December 6, 2010, (21G-10-0225), Submittal of Revised License Pages for License Amendment to Chapter 3 Regarding Radiation Protection
 - 6) Letter from NRC to Mark P. Elliott, dated December 21, 2010, (TAC No. L33008), Nuclear Fuel Services, Inc., Amendment 92, Approval of Changes to Chapter 3 Regarding Ventilation Requirements

Subject: Revised Chapter 4 for Renewal of SNM License 124

Nuclear Fuel Services, Inc. (NFS) hereby submits the revised Chapter 4, Radiation Protection, for the renewal of SNM License 124. The Attachment contains proposed changes to incorporate the responses to the Request for Additional Information (Reference 4); additional information discussed with your staff during a conference call held on November 18, 2010, for topics related to Chapter 4; and recently approved license requirements related to ventilation systems (Reference 6).

If you or your staff have any questions, require additional information, or wish to discuss this, please contact me, or Ms. Jennifer Wheeler, Licensing & ISA Manager, at (423) 735-5429. Please reference our unique document identification number (21G-11-0100) in any correspondence concerning this letter.

Sincerely,

NUCLEAR FUEL SERVICES, INC.

Mark P. Elliott, Director
Quality, Safety, and Safeguards

nuclear fuel services, inc., a subsidiary of The Babcock & Wilcox Company

21G-11-0100

DML/pj

Attachment: SNM-124, Chapter 4, Revision 1, dated May 13, 2011

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Attachment

**SNM-124, Chapter 4
Revision 1**

dated May 13, 2011

(58 pages to follow)

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
Docket 70-143

Chapter 4

RADIATION PROTECTION

**SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4**

**RADIATION PROTECTION
Table of Contents**

SECTION	TITLE	STARTS ON PAGE
4.1	Radiation Protection Program	4-1
4.2	ALARA Program	4-2
4.2.1	NFS ALARA Program Document	
4.2.2	Measures Taken to Implement ALARA	
4.3	Organization and Personnel Qualification	4-5
4.4	Safety Procedures	4-5
4.4.1	"A" Procedures	
4.4.2	"B" Procedures	
4.4.3	"E" Procedures	
4.4.4	"GH" Procedures	
4.4.5	Other Procedures	
4.4.6	Safety Work Permit (SWP) Program	
4.5	Training	4-9
4.6	Ventilation and Respiratory Protection Program	4-10
4.6.1	Occupied Area Ventilation	
4.6.2	Process Enclosure and Exhaust Ventilation	
4.6.2.1	Process Area Containment Enclosures	
4.6.2.2	Laboratory Area Containment Enclosures	
4.6.3	Filtration System Specifications	
4.6.4	Respiratory Protection Program	
4.6.4.1	User Qualification	
4.6.4.2	Testing and Cleaning of Equipment	
4.6.4.3	Respiratory Protection Procedures	
4.7	Radiological Surveys and Monitoring	4-17
4.7.1	Monitoring of the Work Place	
4.7.1.1	Routine Monitoring	
4.7.1.2	Operational Monitoring	
4.7.1.3	Special Monitoring	
4.7.2	Individual Monitoring	
4.7.2.1	Routine Monitoring	
4.7.2.2	Operational Monitoring	
4.7.2.3	Special Monitoring	
4.7.3	Environmental Monitoring	
4.7.4	Radiation Exposure Control	
4.7.4.1	Administrative Action Levels	
4.7.4.2	Personnel Exposure Guidelines	

**SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4**

SECTION	TITLE	STARTS ON PAGE
4.7.5	Internal Radiation Exposure - Personnel Monitoring Program	
4.7.5.1	General	
4.7.5.2	Capabilities	
4.7.5.3	Bioassay Frequencies	
4.7.5.4	Uranium Chemical Toxicity	
4.7.5.5	Quality Control of Other Programs	
4.7.6	External Radiation Exposure - Personnel Monitoring Program	
4.7.7	Work-Area Air Sampling	
4.7.7.1	Airborne Radioactivity in Work Areas	
4.7.7.2	Air Monitoring Systems	
4.7.7.3	Stationary Air Samplers (SAS)	
4.7.7.4	Breathing Zone Air Sampling	
4.7.7.5	Continuous Air Monitors (CAMs)	
4.7.7.6	High-Volume Sampling	
4.7.7.7	Quality Assurance/Quality Control (QA/QC) Considerations	
4.7.7.8	Action Levels	
4.7.8	Work Restrictions	
4.7.9	Radiation Exposure Assessment	
4.7.9.1	Internal Exposure Assessment	
4.7.9.2	External Exposure Assessment	
4.7.9.3	Declared Pregnant Worker and Dose to Embryo Fetus	
4.7.10	Posting and Labeling	
4.7.11	Contamination Control Program	
4.7.11.1	Area Classification	
4.7.11.2	Surface Contamination Monitoring	
4.7.11.3	Action Guidelines	
4.7.11.4	Contamination Survey Practices	
4.7.11.5	Area Contamination Control Practices	
4.7.11.6	Personnel Contamination Control Guidance	
4.7.11.7	Contamination Control for Release of Material or Equipment and for Shipping	
4.7.12	Radioactivity Measurement Instruments	
4.7.12.1	Equipment Description	
4.7.12.2	Instrument Types	
4.7.12.3	Equipment Storage, Maintenance, and Calibration	
4.7.12.4	Criticality Detection System	
4.8	Additional Program Commitments	4-48
4.8.1	Survey and Monitoring Data	
4.8.2	Records and Reports	
4.8.3	Sealed Sources	

**SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4**

NUMBER	FIGURE	STARTS ON PAGE
4-1	Radiologically Controlled Areas	4-38

NUMBER	TABLES	STARTS ON PAGE
4-1	Respiratory Protection Equipment at NFS-Erwin	4-15
4-2	Administrative Action Levels – Personnel Exposure Control	4-21
4-3	Typical Bioassay Minimum Detectable Amounts	4-24
4-4	Exposure Action Levels	4-25
4-5	Air Sampling System/Response Levels And Actions	4-28
4-6	Minimum Survey Frequencies	4-37
4-7	Surface Contamination Action Guidelines	4-40
4-8	Personnel Survey Action Levels	4-42
4-9	Typical Radiation Detection Instruments/Systems Used at NFS	4-45
4-10	Types and Uses of Available Instruments (Typical)	4-46
4-11	Records and Their Minimum Retention Time	4-49
4A-1	Acceptable Surface Contamination Levels	4-54

**SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4**

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. |

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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.

SPECIAL NUCLEAR MATERIAL LICENSE

SNM-124

CHAPTER 4

- 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.

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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 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,

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

- (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

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

- 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

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

- 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

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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.

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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.

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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.

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

- 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.

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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 total air flow for each type of containment enclosure to ensure proper installation and function. The total flow will then be correlated to a 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,

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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.

SPECIAL NUCLEAR MATERIAL LICENSE

SNM-124

CHAPTER 4

- 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.

**SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4**

**Table 4-1
Respiratory Protection Equipment at NFS-Erwin¹**

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.

SPECIAL NUCLEAR MATERIAL LICENSE

SNM-124

CHAPTER 4

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.

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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.

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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).

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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:

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

- 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.

**SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4**

**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.

**SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4**

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

SPECIAL NUCLEAR MATERIAL LICENSE

SNM-124

CHAPTER 4

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 $\mu\text{Ci/gU}$ 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

**SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4**

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

**SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4**

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 2.4 μ Ci/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/m³ (Class D/F) = 14% DAC (0.36 μ Ci/gU)
 = 27% DAC (0.677 μ Ci/gU)
 = 40% DAC (1.0 μ Ci/gU)
 = 80% DAC (2.0 μ Ci/gU)
 = 95% DAC (2.4 μ Ci/gU)

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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.

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

(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.

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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 ($\leq 2.4 \mu\text{Ci/gU}$ 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

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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-

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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."

SPECIAL NUCLEAR MATERIAL LICENSE

SNM-124

CHAPTER 4

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 the plant security fence shall be posted:



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



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

Areas are 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

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

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,

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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.

**SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4**

**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
<p>* These frequencies may be reduced for buildings which are not in operation.</p> <p>** Fixed surface contamination surveys are performed in RCAs as investigative only.</p>			

**SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4**

Figure 4-1: Radiologically Controlled Areas

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

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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 cm²) 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

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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.

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

- 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 cm²)	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.				

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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.

SPECIAL NUCLEAR MATERIAL LICENSE

SNM-124

CHAPTER 4

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.

**SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4**

**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

**SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4**

**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		
Alpha Survey Meters	50 cpm-5x10 ⁵ cpm	Direct Personnel & Equipment Surveys
Beta Survey Meters		
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

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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.

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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.

**SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4**

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)

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

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.

**SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4**

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

**SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4**

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:

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

- 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.

SPECIAL NUCLEAR MATERIAL LICENSE
SNM-124
CHAPTER 4

Table 4A-1: ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDES ^a	AVERAGE ^{b,c,f}	MAXIMUM ^{b,d,f}	REMOVABLE ^{b,c,f}
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
<p>^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.</p> <p>^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.</p> <p>^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.</p> <p>^d The maximum contamination level applies to an area of not more than 100 cm².</p> <p>^e The amount of removable radioactive material per 100 cm² 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.</p> <p>^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.</p>			