GE Nuclear Energy

LICENSE RENEWAL APPLICATION

June 5, 1997

USNRC MATERIALS LICENSE SNM-1097

Docket 70-1113

CONTAINS

ORIGINAL LETTER

9706180048 970605
PDR ADOCK 07001113
B PDR

The wording in Section 1.2.2, 1) has been changed to delete the reference to metal. Section 4.10 has been modified to add the EHS function as an acceptable organizational assignment for the qualified reviewer. Also, the last sentence of Section 4.10 has been deleted, because it is not necessary as licensed information. All changes made in this consolidation are minor and consistent with prior NRC approvals and do not decrease the effectiveness of our licensed safety program. This application replaces all previously related submittals in their entirety.

All pages have been dated 6/5/97 and identified as Revision 0. Appendix A has been added, and contains copies of letters referenced in various chapters of the application, excluding the license page changes. Also, other typographical errors have been corrected.

Ten copies of this submittal are being provided for your use.

Please contact Charlie Vaughan on (910) 675-5656 or me on (910) 675-5889, if you have any questions or would like to discuss this matter further.

Sincerely,

GE NUCLEAR ENERGY

Ralph J. Reda 6/5/97

Ralph J. Reda, Manager
Fuels & Facility Licensing

/zb
Enclosure

cc: RJR-97-075
G. L. Troup, NRC-Atlanta
M. Fry, State of NC

TABLE OF CONTENTS

Section	Title	Page
	REVISIONS BY CHAPTER	1
	CHAPTER 1.0 GENERAL INFORMATION	
1.1	Facility and Process Description	1.1
1.2	Institutional Information	1.7
1.3	Special Authorizations and Exemptions	1.10
	CHAPTER 2.0 ORGANIZATION AND ADMINISTRATION	
2.1	Policy	2.1
2.2	Organizational Responsibilities and Authority	2.1
2.3	Safety Committees	2.10
	CHAPTER 3.0 CONDUCT OF OPERATIONS	
3.1	Configuration Management (CM)	3.1
3.2	Maintenance	3.2
3.3	Quality Assurance (QA)	3.4
3.4	Training and Qualification	3.6
3.5	Human Factors	3.7
3.6	Audits and Assessments	3.7
3.7	Incident Investigations	3.9
3.8	Records Management	3.10
3.9	Procedures	3.11
	CHAPTER 4.0 INTEGRATED SAFETY ANALYSIS	
4.1	Integrated Safety Analysis	4.1
4.2	Site Description	4.1
4.3	Facility Description	4.1
4.4	Process Description	4.2

TABLE OF CONTENTS

Section	Title	Page
4.5	Process Safety Information	4.2
4.6	Training and Qualifications of the ISA Team	4.2
4.7	ISA Methods	4.2
4.8	Results of the ISA	4.3
4.9	Controls for Prevention and Mitigation of Accidents	4.4
4.10	Administrative Control of the ISA	4.7

CHAPTER 5.0
RADIATION SAFETY

Section	Title	Page
5.1	ALARA (As Low As is Reasonably Achievable) Policy	5.1
5.2	Radiation Safety Procedures and Radiation Work Permits (RWPS)	5.1
5.3	Ventilation Requirements	5.2
5.4	Air Sampling Program	5.3
5.5	Contamination Control	5.5
5.6	External Exposure	5.7
5.7	Internal Exposure	5.7
5.8	Summing Internal and External Exposure	5.9
5.9	Action Levels for Radiation Exposures	5.9
5.10	Respiratory Protection Program	5.9
5.11	Instrumentation	5.10

CHAPTER 6.0
NUCLEAR CRITICALITY SAFETY

Section	Title	Page
6.1	Program Administration	6.1
6.2	Technical Practices	6.5
6.3	Control Documents	6.29
6.4	Criticality Accident Alarm System	6.36

CHAPTER 7.0
CHEMICAL SAFETY

Section	Title	Page
7.1	Chemical Safety Program	7.1
7.2	Contents of Chemical Safety Program	7.1

TABLE OF CONTENTS

Section	Title	Page
	CHAPTER 8.0 FIRE SAFETY	
8.1	Fire Protection Program Responsibility	8.1
8.2	Fire Protection Program	8.1
8.3	Administrative Controls	8.2
8.4	Building Construction	8.2
8.5	Ventilation Systems	8.3
8.6	Process Fire Safety	8.3
8.7	Fire Detection and Alarm Systems	8.3
8.8	Fire Suppression Equipment	8.4
8.9	Fire Protection Water System	8.4
8.10	Radiological Contingency and Emergency Plan (RC&EP)	8.5
8.11	Emergency Response Team	8.5
	CHAPTER 9.0 RADIOLOGICAL CONTINGENCY AND EMERGENCY PLAN	9.1
	CHAPTER 10.0 ENVIRONMENTAL PROTECTION	
10.1	Air Effluent Controls and Monitoring	10.1
10.2	Liquid Treatment Facilities	10.1
10.3	Solid Waste Management Facilities	10.2
10.4	Program Documentation	10.2
10.5	Evaluations	10.3
10.6	Off-site Dose	10.3
10.7	ALARA	10.4
	CHAPTER 11.0 DECOMMISSIONING	11.1
	APPENDIX	A.1

REVISIONS BY CHAPTER

Page	Application Date	Page	Application Date
REVISIONS BY CHAPTER		**CHAPTER 6.0**	
1	06/05/97	1 through 36	06/05/97
TABLE OF CONTENTS		**CHAPTER 7.0**	
1 through 5	06/05/97	1 through 3	06/05/97
CHAPTER 1.0		**CHAPTER 8.0**	
1 through 22	06/05/97	1 through 5	06/05/97
CHAPTER 2.0		**CHAPTER 9.0**	
1 through 11	06/05/97	1	06/05/97
CHAPTER 3.0		**CHAPTER 10.0**	
1 through 12	06/05/97	1 through 16	06/05/97
CHAPTER 4.0		**CHAPTER 11.0**	
1 through 8	06/05/97	1	06/05/97
CHAPTER 5.0		**APPENDIX**	
1 through 13	06/05/97	1	06/05/97

CHAPTER 1.0

GENERAL INFORMATION

1.1 FACILITY AND PROCESS DESCRIPTION

The primary purpose of the GE-Nuclear Energy Production facility in Wilmington, North Carolina (identified in this document as GE-Wilmington) is the manufacture of fuel assemblies for commercial nuclear reactors. Nuclear materials enriched to less than or equal to 5 weight percent U-235 are utilized in the product manufacturing operations authorized by this license. The safety, environmental, quality assurance and emergency preparedness aspects of the manufacturing operations are managed and controlled as described in this license.

1.1.1 SITE DESCRIPTION AND LOCATION

GE-Wilmington is situated on a 1,664-acre tract of land, located on U.S. Highway 117 and approximately six miles north of the City of Wilmington, North Carolina in New Hanover County (refer to Figures 1.1 and 1.2). New Hanover County is situated in the southern coastal plains section of southeastern North Carolina, with the Atlantic Ocean on the east and the Cape Fear River on the west. The Atlantic Ocean lies approximately 10 miles east and 26.4 miles south of GE-Wilmington. The surrounding terrain is low-lying, with an average elevation of less than 40 feet above mean sea level.

Castle Hayne, the nearest community, is approximately three miles north of GE-Wilmington. The region around the site is lightly settled with large areas of heavily timbered tracts of land. Farms, single-family dwellings and light commercial activities are located along U.S. 117. The Wilmington airport is located approximately 3.5 miles southeast of the site.

1.1.2 FACILITY DESCRIPTION

The location and arrangement of buildings at the GE-Wilmington site, and their relative distance from the site boundary are shown in Figure 1.3. Located on the GE-Wilmington site are the following major facilities: (1) the GE Aircraft Engine (AE)

FIGURE 1.1
PLANT SITE - STATE AND COUNTY LOCATIONS

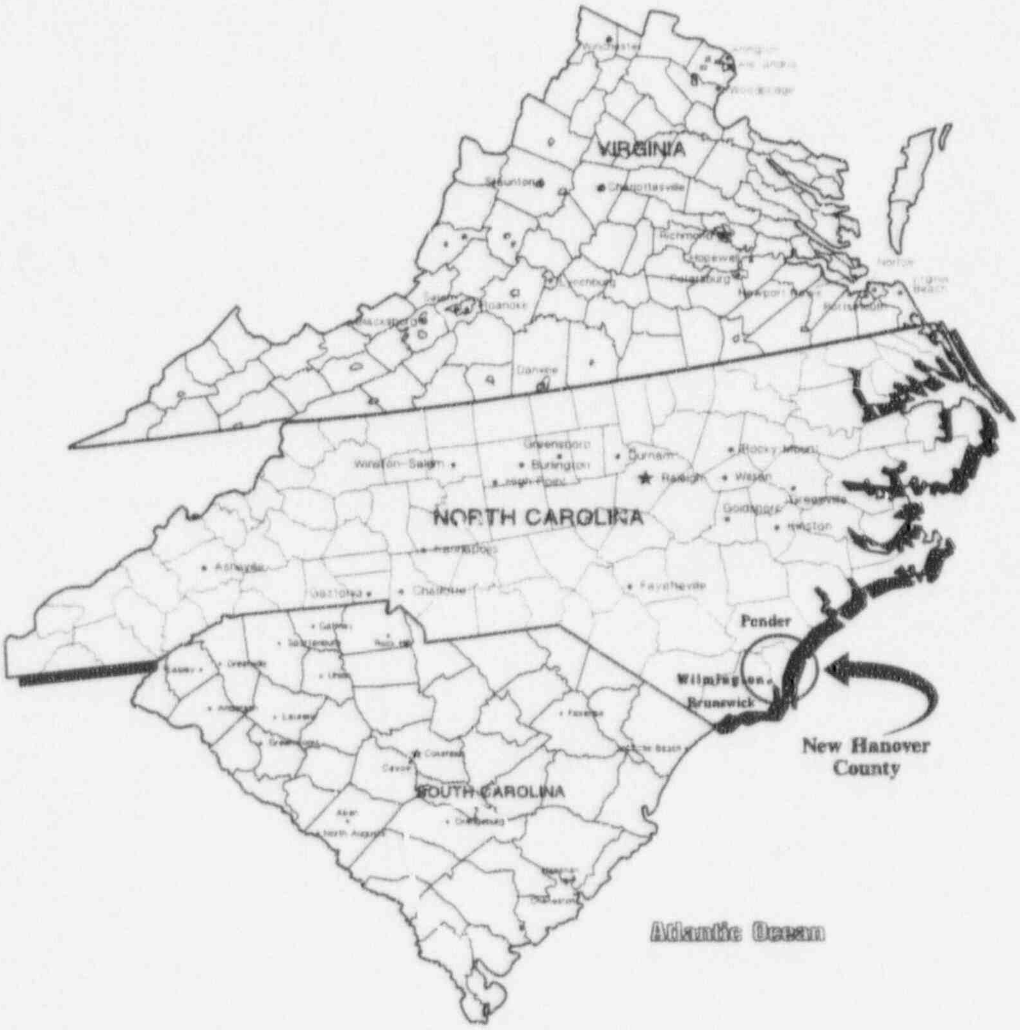


FIGURE 1.2
NEW HANOVER COUNTY AND ADJACENT COUNTIES

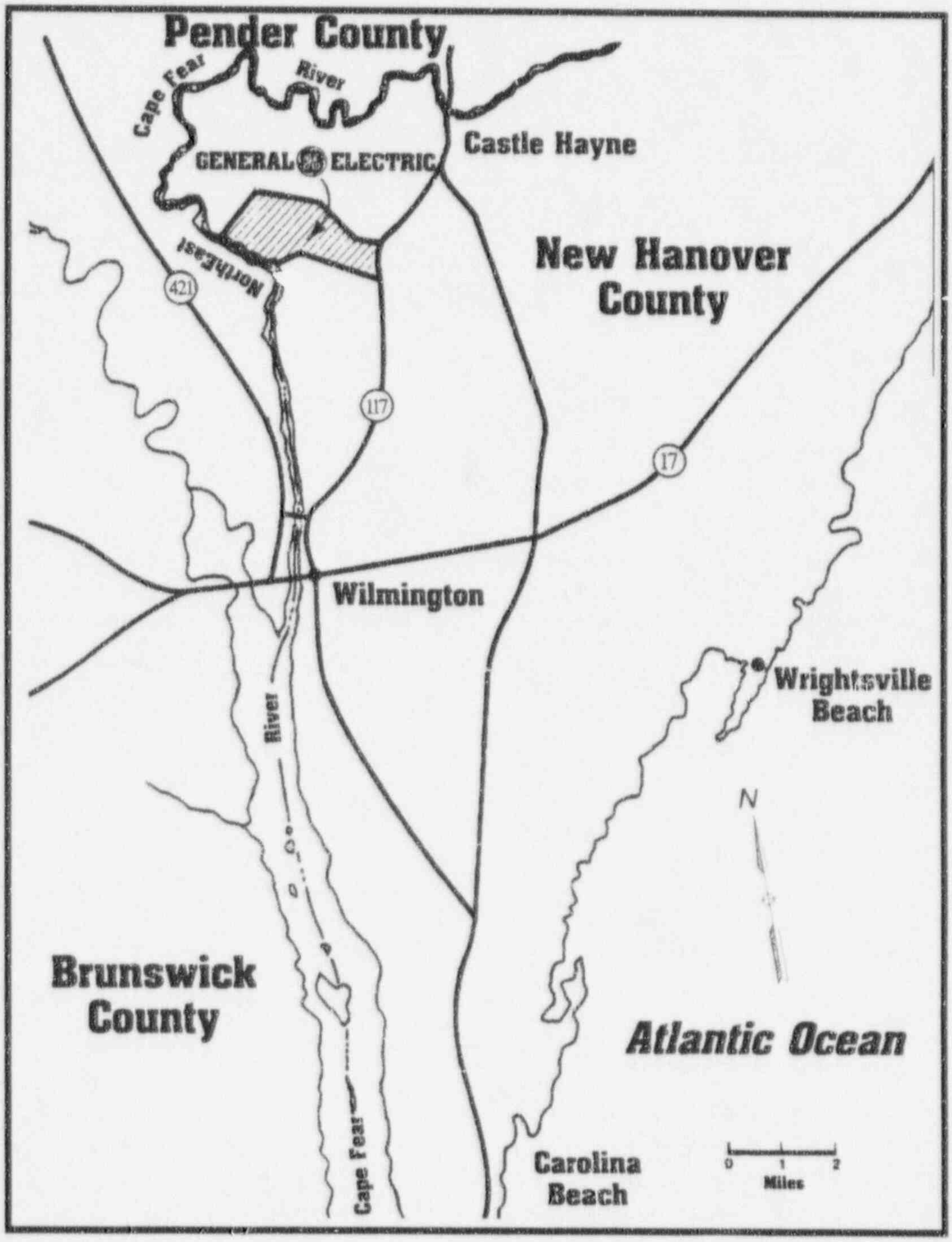


FIGURE 1.3
GE-WILMINGTON SITE PLAN

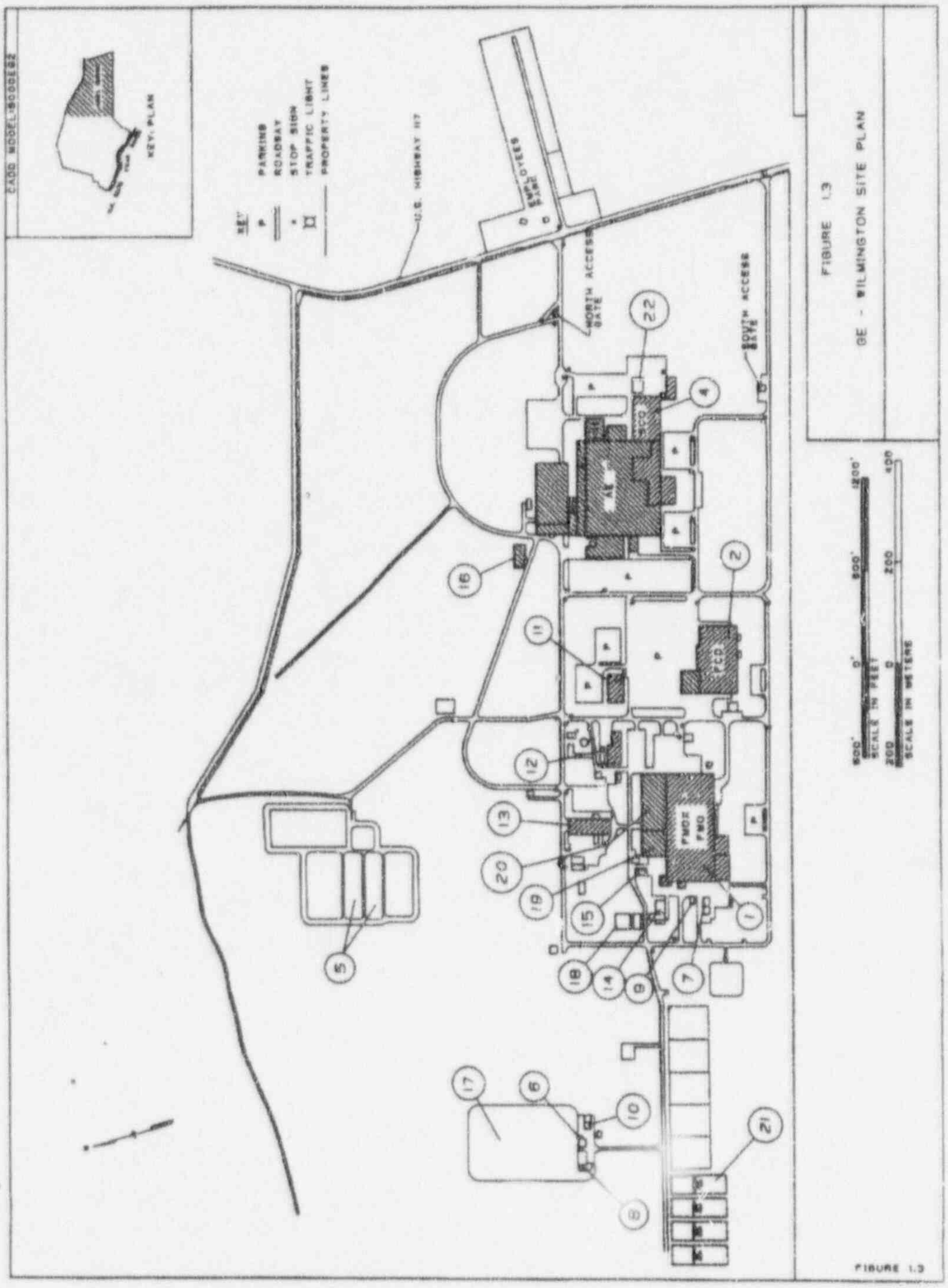


FIGURE 1.3 (Continued)

GE-WILMINGTON SITE PLAN LEGEND

1	:	Fuel Manufacturing Operation (FMO)
2	:	Fuel Components Operation (FCO)
3	:	Aircraft Engine Operation (AE)
4	:	Services Components Operation (SCO)
5	:	Final Process Basins
6	:	Waste Treatment Facility
7	:	Incinerator Building
8	:	Filter Facility
9	:	DA Building
10	:	Boiler / URLS
11	:	Office Building
12	:	Site Maintenance
13	:	Site Warehouse
14	:	FMO Storage Building
15	:	FMO Maintenance Building
16	:	AE Maintenance Building
17	:	Waste Treatment Basins
18	:	Fuel Examination Technology
19	:	Dry Conversion Process Building (DCP)
20	:	Warehouse
21	:	CaF_2 Storage Warehouse

facility which is not involved in the nuclear fuel manufacturing operation, (2) The Services Components Operation (SCO) facility where non-radioactive reactor components are manufactured, (3) the Fuel Components Operation (FCO) facility where non-radioactive components for reactor fuel assemblies are manufactured, and (4) the fuels complex containing the fuel manufacturing facility. The fuels complex, which includes the Fuel Manufacturing Operation and Dry Conversion Process (FMO/FMOX & DCP) buildings and supporting facilities, is enclosed by a fence with restricted access. This complex is called the Controlled Access Area (CAA).

1.1.3 FACILITY RESISTANCE TO ENVIRONMENTAL EVENTS

In the coastal area of North Carolina, where GE-Wilmington is located, severe weather conditions may result from hurricanes, tornadoes, ice storms, and snow storms. The greatest severe weather threat in this area is due to high winds from hurricanes and possible tornadoes. Facility construction meets or exceeds local codes for strength and in the case of hurricanes, advance notice provides an opportunity for further mitigating actions. Since high winds could impact electrical power, key safety systems are protected with adequate back-up power supplies or fail safe features. Earthquakes are not considered a major threat because this section of the southern Atlantic Seaboard is an area of relatively low seismic activity.

The Fuel Manufacturing Operation building in which radioactive materials are processed and stored, is designed to provide for containment of material under adverse environmental conditions such as fire, wind, flooding or earthquake to the limits of the building code. The roof construction meets Factory Mutual requirements for fire hazard and wind resistance.

Detailed information regarding the facility resistance to the effects of potential credible accident events is contained in Chapters 2 and 5 of the Radiological Contingency and Emergency Plan for GE-Wilmington, which is described in Chapter 9.0 of this license, and in Chapters 2 and 6 of the Environmental Report for GE-Wilmington which is described in Chapter 10.0 of this license.

1.1.4 PROCESS DESCRIPTION

The product manufacturing operations authorized by this license consist of receiving low-enriched, less than or equal to 5.0 weight percent U-235, uranium hexafluoride; converting the uranium hexafluoride to uranium dioxide powder; and processing the

uranium dioxide through pelletizing steps, fuel rod loading and sealing, and fuel assembly fabrication.

Two types of processes are used to convert uranium hexafluoride to uranium dioxide powder -- the Ammonium Diuranate (ADU) process, and Dry Conversion Process (DCP). The manufacturing operations are served by support systems such as scrap recovery, waste disposal, laboratory, and manufacturing technology development, which are also described in this license.

1.2 INSTITUTIONAL INFORMATION

The GE-Wilmington NRC license number is SNM-1097 (Docket #70-1113).

1.2.1 IDENTITY AND ADDRESS

This application for license renewal is filed by the GE-Nuclear Energy Production facility of the General Electric Company, a major corporation with corporate headquarters in Fairfield, Connecticut. General Electric's nuclear energy business, known as GE Nuclear Energy, is headquartered in San Jose, California, with the principal manufacturing facility located in Wilmington, North Carolina.

The full address is as follows: GE Nuclear Energy Production, (name of person and mail code), P.O. Box 780, Wilmington, NC 28402.

1.2.2 TYPE, QUANTITY, AND FORM OF LICENSED MATERIAL

Uranium normally will be used at GE-Wilmington in the Controlled Access Area (CAA) only. Conversion and fabrication is conducted within the fuel manufacturing building (FMO/FMOX & DCP). Small quantities (i.e., less than one safe batch of uranium in a non-dispersible form) may be temporarily moved to other buildings or site locations outside of the CAA for special tests under special authorizations and controls.

The following types, maximum quantities, and forms of special nuclear materials are authorized:

1) 50,000 kilograms of U-235 contained in uranium enriched to a maximum enrichment of less than or equal to 5%, in any chemical or physical form;

2) 500 kilograms of U-235 at enrichments from 5% to <10% contained in uranium compounds for use in laboratory and process development operations;

3) 9.649 kilograms of U-235 at enrichments from 10% to <15% contained in uranium compounds for use in laboratory and process development operations;

4) 350 grams of U-235 in any form contained in uranium at any enrichment, for use in measurements, detection, research or development activities;

5) Plutonium - 1 milligram in samples for analytical purposes, 1 milligram as standards for checking the alpha radiation response of radiation detection instrumentation, 20 grams as sealed neutron sources, and in nuclear fuel rods at a level of less than 1×10^{-6} gram of plutonium per gram of U^{235}; and

6) 50 milligrams U-233 for analytical purposes.

1.2.3 ACTIVITY

GE-Wilmington complies with applicable parts of Title 10, Code of Federal Regulations, unless specifically amended or exempted by NRC staff.

Authorized activities at GE-Wilmington include:

1.2.3.1 Product Processing Operations

- UF_6 Conversion - Conversion of uranium hexafluoride to uranium oxides by the ADU process, and the Dry Conversion Process.
- Fuel Manufacture - Fabrication of nuclear reactor fuel (powder, pellets, or assemblies) containing uranium.
- Scrap Recovery - Reprocessing of unirradiated material from GE-Wilmington and from other sources with nuclear safety characteristics not significantly different from GE-Wilmington in-process materials.
- Waste Recovery - Recovery of uranium from wet and dry material stored in on-site pits and basins. The recovered uranium will be returned to the fuel processing facility.

1.2.3.2 Process Technology Operations

- Development and fabrication of reactor fuel, fuel elements and fuel assemblies of advanced design in small amounts.
- Development of scrap recovery processes.
- Determination of interaction between fuel additives and fuel materials.
- Chemical analysis and material testing, including physical and chemical testing and analysis, metallurgical examination and radiography of uranium compounds, alloys and mixtures.
- Instrument research and calibratio: , including development, calibration and functional testing of nuclear inst: umentation and measuring devices.
- A conversion of UF_6 to UO_2 and other intermediate compounds using chemical and dry process...
- Other process tec'.nology development activities related to, but not limited by, the above.

1.2.3.3 Laboratory Operations

Chemical, physical or metallurgical analysis and testing of uranium compounds and mixtures, including but not limited to, preparation of laboratory standards.

1.2.3.4 General Services Operations

- Storage of unirradiated fuel assemblies, uranium compounds and mixtures in areas arranged specifically for maintenance of criticality and radiological safety.
- Design, fabrication and testing of uranium prototype processing equipment.
- Maintenance and repair of uranium processing equipment and auxiliary systems.
- Storage and nondestructive testing of fuel rods containing trace amounts of plutonium as authorized in the license.

1.2.3.5 Waste Treatment and Disposal

- Treatment, storage and disposal and/or shipment of liquid and solid wastes whose discharges are regulated.
- Decontamination of non-combustible contaminated wastes to reduce uranium contamination levels, and subsequent shipment of such low-level radioactive wastes to licensed burial sites for disposal or as authorized by the NRC.
- Treatment or disposal of combustible waste and scrap material by incineration pursuant to 10 CFR 20.2002 and 10 CFR 20.2004.

1.2.3.6 Off-site Activities

Testing, demonstration, non-destructive modification and storage of materials and devices containing unirradiated uranium, provided that such materials and devices shall be under GE control at all times.

1.3 SPECIAL AUTHORIZATIONS AND EXEMPTIONS

1.3.1 AUTHORIZATIONS TO MAKE CHANGES TO LICENSE COMMITMENTS

1.3.1.1 Changes Requiring Prior NRC Approval

GE will not make changes to the licensed safety program that decrease the effectiveness of commitments without prior NRC approval. For these changes GE will submit to the NRC for review and approval an application to amend the license. The change will not be implemented until approval is granted. This includes changes to single parameter controlled processes or equipment as specified in Section 6.2.3 and Table 6 of the license.

1.3.1.2 Changes Not Requiring Prior NRC Approval

GE is authorized to make changes to the commitments in the licensed safety program without prior approval, provided that the changes do not decrease the effectiveness of the approved commitments. This authorization will allow GE to make changes, conduct tests or modify activities in a facility or process upon documented

completion of an ISA (Integrated Safety Analysis, Chapter 4) for that facility or process subject to the following conditions:

- There is no degradation in the safety commitment in the license
- The change, test or activity does not impair the licensee's ability to meet all Federal regulations
- The change, test or activity does not conflict with any requirements specifically stated in the license
- The change relates to Section 6.2.3 and Table 6 of the license where changes from one parameter to another parameter for process subareas or equipment in which multiple (at least two) parameters are controlled are made in accordance with established change control measures.

Records of such changes, tests or activities will be maintained, including technical justification and management approval, and available on site for inspection. A report containing a description of each change, test or activity and, where necessary, revised pages to the License Application will be submitted to the NRC within 3 months of implementing the change.

1.3.2 AUTHORIZED GUIDELINES FOR CONTAMINATION-FREE ARTICLES

Authorization to use the guidelines, contamination and exposure rate limits specified at the end of this Section, "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material," US NRC, April 1993 for decontamination and survey of surfaces or premises and equipment prior to abandonment or release for unrestricted use.

1.3.3 AUTHORIZED TRANSFER OF CONTAMINATION-FREE LIQUIDS

1.3.3.1 Transfer of Hydrofluoric Acid (HF) for Testing

Authorization to transfer test quantities of HF to potential buyers/customers or laboratories for the purpose of analyzing, examination or evaluation, without continuing NRC controls as described in GE-Wilmington's letter to the NRC dated February 27, 1996.

Test quantities may not contain more than 3 PPM uranium with an enrichment not to exceed 6% U-235.

The recipients will be advised that this material is not a nuclear hazard, but will be advised that the material should be handled carefully and in such a manner so as not to be consumed by humans nor used in products used on or in the body or in the food chain.

1.3.3.2 Transfer of Hydrogen Fluoride Solutions as Product

Authorization, pursuant to 10 CFR 70.42(b)(3), to transfer liquid hydrofluoric acid to any commercial chemical company/supplier without either company possessing an NRC or Agreement State license for special nuclear material, provided that the concentration of uranium does not exceed three parts per million by weight of the liquid and the enrichment is less than or equal to 6 weight percent U-235.

The hydrofluoric acid is transferred and used in such a manner that the minute quantity of uranium does not enter into any food, beverage, cosmetic, drug or other commodity designated for ingestion or inhalation by, or application to, a human being such that the uranium concentration in these items would exceed that which naturally exists. Additionally, the acid is used in a process which will not release the low levels of radioactivity to the atmosphere as airborne material and whose residues will remain in a wastewater or other treatment system.

Prior to shipment, each transfer is sampled and measured to assure that the concentration does not exceed three parts per million of uranium.

GE-Wilmington shall maintain records under this condition of license including, as a minimum, the date, uranium concentration and quantity of hydrofluoric acid transferred.

1.3.3.3 Transfer of Nitrate-Bearing Liquids

Authorization to transfer nitrate-bearing liquids, provided that the uranium concentration does not exceed a 30-day average of 5 parts per million by weight of the liquids and the enrichment is less than or equal to 6 weight percent U-235 by transport to an off-site liquid treatment system located at International Paper, Riegelwood, North Carolina (or similar commercial paper operation), in which decomposition of the nitrates will occur and from which the denitrified liquids will be discharged in the effluent from the system.

Environmental samples will be taken periodically to monitor effluent releases.

1.3.4 AUTHORIZATION TO TRANSFER TEST QUANTITIES OF CALCIUM FLUORIDE

Authorization to transfer test quantities of calcium fluoride (CaF_2) to potential buyers for the purpose of their examination and evaluation as described in GE-Wilmington's letter to the NRC dated September 24, 1992.

Test quantities may not contain more than 30 pCi per gram on a dry weight basis and are limited to 1 gram U-235 at each off-site location.

Test activities and end uses of the material will be limited to those that do not allow chemical separation of the uranium or entry of the product into the food chain.

1.3.5 AUTHORIZATION TO TRANSFER OF CALCIUM FLUORIDE (CAF_2) TO VENDORS FOR BENEFICIAL REUSE

Authorization to transfer quantities of industrial waste treatment products (primarily CaF_2) to commercial firms, for the purpose of briquette manufacturing and use as a steel flux forming material in the production of steel as described in GE-Wilmington's letter to the NRC dated December 20, 1989.

Measurements are made using a sample plan to provide at a 95% confidence level that the population mean for each shipment is less than 30pCi of uranium per gram of material on a dry weight basis.

Activities and end use of the material will be limited to those that do not allow chemical separation of the uranium or entry of the product into the food chain.

1.3.6 AUTHORIZATION TO DISPOSE OF INDUSTRIAL WASTE TREATMENT PRODUCTS

Notwithstanding any requirements for state or local government agency disposal permits, GE-Wilmington is authorized to dispose of industrial waste treatment products without continuing NRC controls provided that either of the two following conditions are met:

1.3.6.1 Free-standing liquid shall be removed prior to shipment.

The uranium concentration in the material shipped for disposal shall not exceed 30 pCi per gram after free-standing liquid has been removed.

The licensee shall possess authorization from appropriate state officials prior to disposing of the waste material. The authorization shall be available for inspection at the GE-Wilmington facility.

1.3.6.2 The uranium concentration in the material shipped for disposal only at approved facilities such as Pinewood, South Carolina (licensed by the State of South Carolina), shall not exceed 250 pCi per gram of uranium activity, of which no more than 100 pCi per gram shall be soluble.

1.3.7 AUTHORIZATION TO STORE SANITARY SLUDGE PENDING FINAL DISPOSAL

Dried sanitary sludge is collected and disposed of at approved offsite facilities in accordance with Section 1.3.6. Authorization to store treated sanitary sludge containing trace amounts of uranium in the sanitary sludge land application area pending final disposal.

1.3.8 AUTHORIZATION FOR THE USE OF MATERIALS AT OFF-SITE LOCATIONS

1.3.8.1 Authorization to use up to 15 grams of U-235 at other sites within the limits of the United States and at temporary job sites of the licensee anywhere in the United States where the Nuclear Regulatory Commission maintains jurisdiction for regulating the use of licensed material.

The manager of the radiation safety function shall establish the safety criteria for material being used at off-site locations and shall designate the individual who will be responsible for carrying out these criteria.

1.3.8.2 Authorization to store at nuclear reactor sites, uranium fully packaged for transport in any NRC approved package, in accordance with the conditions of a license authorizing delivery of such containers to a carrier for NRC approved transport, at

locations in the United States providing such locations minimize the severity of potential accident conditions to be no greater than those in the design bases for the containers during transportation.

Provisions for compliance with applicable 10 CFR 73 requirements are described in the NRC-approved GE-Wilmington Physical Security Plan as currently revised in accordance with regulatory provisions.

Storage at nuclear reactor sites is subject to the financial protection and indemnity provision of 10 CFR 140.

1.3.8.3 Authorization to store at nuclear reactor sites, arrays of finished reactor fuel rods and/or assemblies in any of the inner metal containers of the RA-series shipping package described in NRC Certificate of Compliance Number 4986 at locations in the United States providing such locations minimize the severity of potential accident conditions to be no greater than those in the design bases for the containers during transportation.

Arrays may be constructed without limit to the number of containers so stored, except that each array shall be stacked to the smaller of 4 containers high or the height demonstrated to comply with the criticality safety requirements of this license. Each container must be separated by nominal 2-inch wooden studs, with the width and length for each array and separation between arrays determined only by container handling requirements.

Provisions for compliance with applicable 10 CFR 73 requirements are described in the NRC-approved GE Wilmington Physical Security Plan as currently revised in accordance with regulatory provisions.

Storage at nuclear reactor sites is subject to the financial protection and indemnity provision of 10 CFR 140.

1.3.8.4 Authorization to transfer, possess, use and store unirradiated reactor fuel of GE-Wilmington manufacture or procured to GE specification at nuclear reactor sites, for purposes of inspection, fuel bundle disassembly and assembly, including fuel rod replacement, provided that the following conditions are met:

- A valid NRC license has been issued to the reactor licensee, which authorizes receipt, possession and storage of the fuel at the reactor site. GE Wilmington possesses the fuel only within the indemnified location.
- For dry fuel reconstitution, not more than 99 (9x9 lattices or greater) or 88 (8x8 lattices) unassembled fuel rods may be possessed by GE-Wilmington at any one reactor site at any one time, except when the fuel has been packaged for transport or as described in Section 1.3.8.3. The fuel rods must be of the types described in NRC Certificate of Compliance Number 4986.
- For underwater fuel reconstitution, not more than one fuel assembly plus unassembled fuel rods so that the total number of rods, including the assembly, possessed by GE-Wilmington at any one reactor site at any one time does not exceed 99 (9x9 lattices or greater) or 88 (8x8 lattices), except when the fuel has been packaged for transport or as described in Section 1.3.8.3. The fuel rods must be of the types described in NRC Certificate of Compliance Number 4986.
- Operations involving the fuel are conducted by or under the direct supervision of a member of the GE-Wilmington staff who shall be responsible for work on the fuel element assembly. The person shall comply with applicable reactor license and procedure requirements as directed by reactor site representatives, including appropriate actions that are to be taken in the event of emergencies at the site.
- Loose rods are stored in RA-series inner metal containers.
- Fuel is handled in accordance with pertinent provisions of the reactor license, and also in accordance with applicable GE-Wilmington procedures which are jointly verified for completion by GE-Wilmington and the reactor licensee.
- Records of the operation, including the GE-Wilmington procedures used, are maintained at the GE-Wilmington facility.

1.3.9 AUTHORIZATION TO USE A DILUTION FACTOR FOR AIRBORNE EFFLUENTS

Pursuant to 10CFR20.1302, GE is authorized to utilize a dilution factor of 100 to the measured stack discharges for the purpose of evaluating the airborne radioactivity at the closest site boundary.

This conservative dilution factor is derived using standard diffusion models and conservative assumptions regarding physical and atmospheric characteristics of the site. Records of the derivation of this factor are maintained on site for inspection.

1.3.10 AUTHORIZATION FOR WORKPLACE AIR SAMPLING ADJUSTMENTS

Authorization to adjust Derived Air Concentration (DAC) limits and Annual Limit of Intake (ALI) values in process areas to reflect chemical and physical characteristics of the airborne uranium .

1.3.11 EXEMPTION TO CRITICALITY MONITORING SYSTEM REQUIREMENTS

Authorization that it is not necessary to maintain the criticality accident monitoring system requirements of 10 CFR 70.24 when it is demonstrated that a credible criticality risk does not exist for each area in which there is not more than:

1.3.11.1 A quantity of finished reactor fuel rods equal to or less than 45% of a minimum critical number under conditions in which double batching is credible, or equal to or less than 75% of a minimum critical number under conditions in which double batching is not credible, or

1.3.11.2 The quantity of uranium authorized for delivery to a carrier when fully packaged as for transport according to a valid NRC authorization for such packages without limit on the number of such packages, provided storage locations preclude mechanical damage and flooding, or

1.3.11.3 Arrays of finished reactor fuel rods and/or assemblies in any of the inner metal containers of the RA-series shipping package described in NRC Certificate of Compliance Number 4986, under storage conditions described in Section 1.3.8.3, or

1.3.11.4 Unassembled fuel rods under the restrictions and transfer, possession, use and storage conditions in Section 1.3.8.4.

1.3.12 EXEMPTION TO POSTING REQUIREMENTS

Authorization to post areas within the Controlled Access Area in which radioactive materials are processed, used, or stored, with a sign stating "Every container in this area may contain radioactive material" in lieu of the labeling requirements of 10 CFR 20.1904.

1.3.13 EXEMPTION TO EXTREMITY DOSE DETERMINATION REQUIREMENTS

Authorization to use a skin thickness of 38 milligrams/cm^2 in the assessment of worker fingertip doses from uranium and for determining compliance to NRC extremity dose limits.

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

The instructions in this guide, in conjunction with Table 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 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 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 of buildings, transfer of premises to another organization continuing work with radioactive materials, or conversion of facilities to a long-term storage or standby status. Such requests must:

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

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

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

   a. Identify the premises.
   b. Show that reasonable effort has been made to eliminate residual contamination.
   c. Describe the scope of the survey and general procedures followed.
   d. State the findings of the survey in units specified in the instruction.

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

TABLE 1

ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDES[a]	AVERAGE[bcf]	MAXIMUM[bdf]	REMOVABLE[bef]
U-nat, U-235, U-238, and associated decay products	5,000 dpm α/100 cm^2	15,000 dpm 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	1000 dpm/100 cm^2	3000 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 $\beta\gamma$/100 cm^2	15,000 dpm $\beta\gamma$/ 100 cm^2	1,000 dpm $\beta\gamma$/100 cm^2

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

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

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

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

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

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

CHAPTER 2.0

ORGANIZATION AND ADMINISTRATION

2.1 POLICY

The GE-Wilmington policy is to maintain a safe work place for its employees, to protect the environment, and to assure operational compliance within the terms and conditions of special nuclear material licenses and applicable NRC regulations.

2.2 ORGANIZATIONAL RESPONSIBILITIES AND AUTHORITY

2.2.1 KEY POSITIONS WITH RESPONSIBILITIES IMPORTANT TO SAFETY (FIGURE 2.1)

Responsibilities, authorities, and interrelationships among the GE-Wilmington organizational functions with responsibilities important to safety are specified in approved position descriptions and in documented and approved practices.

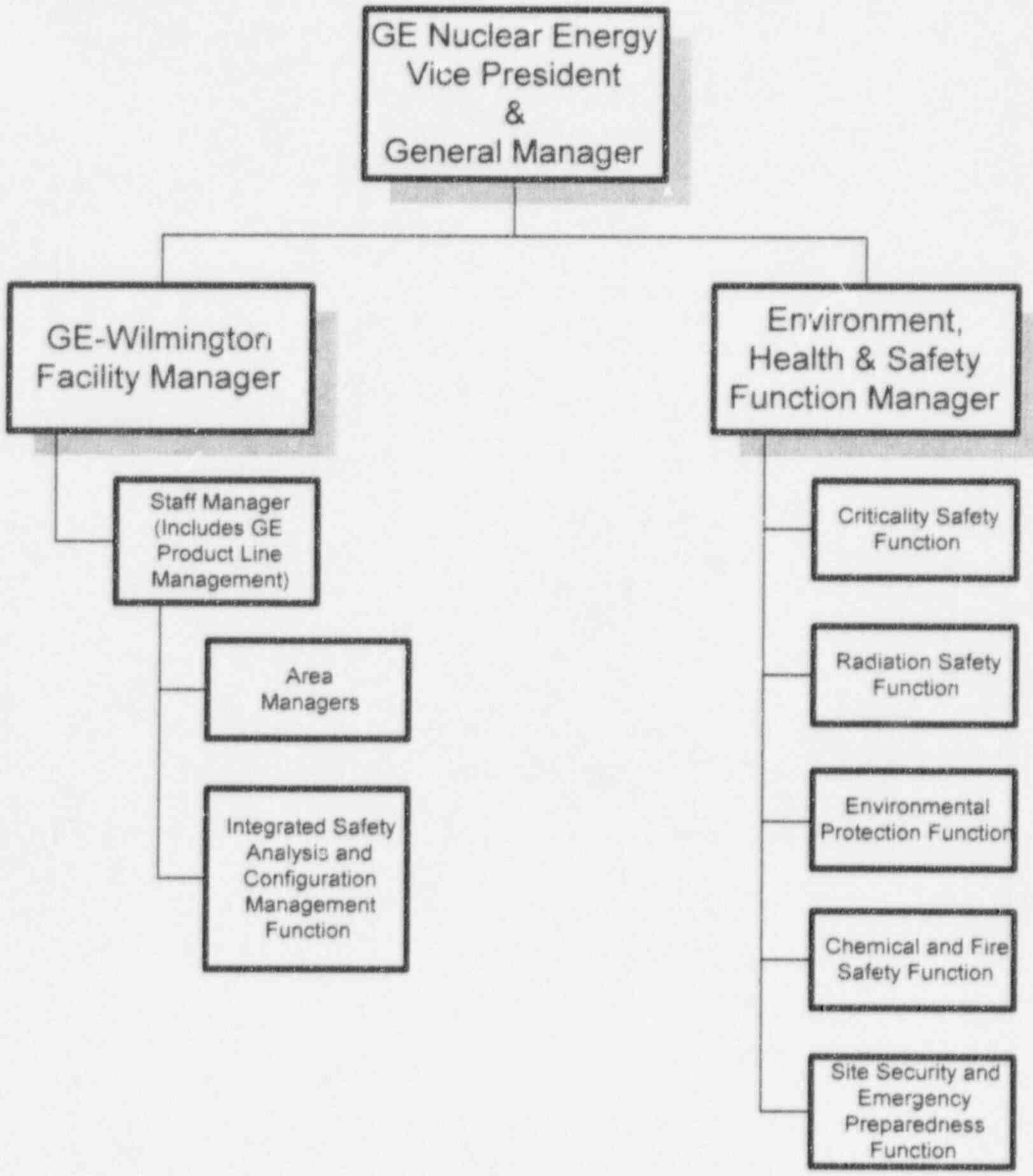
2.2.1.1 GE-Wilmington Facility Manager

The GE-Wilmington facility manager is the individual who has overall responsibility for safety and activities conducted at the GE-Wilmington facility. The GE-Wilmington facility manager directs operations by procedure, or through other management personnel. The activities of the GE-Wilmington facility manager are performed in accordance with GE policies, procedures, and management directives. The GE-Wilmington facility manager provides for safety and control of operations and protection of the environment by delegating and assigning responsibility to qualified Area Managers.

The GE-Wilmington facility manager is knowledgeable of the safety program concepts as they apply to the overall safety of a nuclear facility, and has the authority to enforce the shutdown of any process or facility.

Figure 2.1

GE-Wilmington Organization



2.2.1.2 Area Manager

The Area Manager is the designated individual who is responsible for ensuring that activities necessary for safe operations and protection of the environment are conducted properly within their designated area of the facility in which uranium materials are processed, handled or stored. Designated Area Manager responsibilities include:

- Assure safe operation, maintenance and control of activities
- Assure safety of the environs as influenced by operations
- Assure performance of integrated safety analyses for the assigned facility area, as required
- Assure application of assurance elements to safety controls, as appropriate
- Assure configuration control for safety controls for the assigned facility area, as required
- Use approved written operating procedures which incorporate safety controls and limits
- Provide adequate operator training

The minimum qualifications of an Area Manager is a BS or BA degree in a technical field, and two years of experience in manufacturing operations, one of which is in nuclear fuel manufacturing; or a high school diploma with five years of manufacturing experience, two of which are in nuclear fuel manufacturing.

Area Managers shall be knowledgeable of the safety program procedures (including chemical, radiological, criticality, fire, environmental and industrial safety) and shall have experience in the application of the program controls and requirements, as they relate to their areas of responsibility. The assignment of individuals to the position of Area Manager is approved by the GE-Wilmington facility manager, and the listing of Area Managers by area of responsibility is maintained current at the facility.

2.2.1.3 Integrated Safety Analysis and Configuration Management Function

The integrated safety analysis and configuration management function is administratively part of the fuel production operations at GE-Wilmington. Designated responsibilities include:

- Establish and maintain the integrated safety analysis program
- Establish and maintain the assurance program for safety controls
- Provide advice and counsel to Area Managers on matters of the integrated safety analysis program
- Establish and maintain the configuration control system for fuel manufacturing equipment and safety controls, and related record retention
- Establish and maintain the operating procedure systems

Minimum qualification requirements for the manager of the integrated safety analysis and configuration management function are a BS or BA degree in science or engineering and two years experience in related manufacturing assignments; or a high school diploma with eight years of manufacturing experience. The manager of the integrated safety analysis and configuration management function shall have experience in the understanding and management of the assigned programs.

2.2.1.4 Criticality Safety Function

The criticality safety function is administratively independent of production responsibilities and has the authority to shutdown potentially unsafe operations. Designated responsibilities include:

- Establish the criticality safety program including design criteria, procedures and training
- Provide criticality safety support for integrated safety analyses and configuration control
- Assess normal and credible abnormal conditions
- Determine criticality safety limits for controlled parameters

- Perform methods development and validation to support criticality safety analyses
- Perform neutronics calculations, write criticality safety analyses and approve proposed changes in process conditions or equipment involving fissionable material
- Specify criticality safety control requirements and functionality
- Provide advice and counsel to Area Managers on criticality safety control measures, including review and approval of operating procedures
- Support emergency response planning and events
- Assess the effectiveness of the criticality safety program through audit programs

The criticality safety function manager shall hold a BS or BA degree in science or engineering, have at least four years experience in assignments involving regulatory activities, and have experience in the understanding, application and direction of nuclear criticality safety programs.

Minimum qualifications for a senior engineer within the criticality safety function are a BS or BA degree in science or engineering with at least three years of nuclear industry experience in criticality safety. A senior engineer shall have experience in the assigned safety function, and has authority and responsibility to conduct activities assigned to the criticality safety function.

Minimum qualifications for an engineer within the criticality safety function are a BS/BA degree in science or engineering. An engineer shall have experience in the assigned safety function, and has authority and responsibility to conduct activities assigned to the criticality safety function, with the exception of independent verification of criticality safety analyses.

2.2.1.5 Radiation Safety Function

The radiation safety function is administratively independent of production responsibilities and has the authority to shutdown potentially unsafe operations. Designated responsibilities include:

- Establish the radiation protection and radiation monitoring programs
- Establish the radiation protection design criteria, procedures and training programs to control contamination and exposure to individuals
- Evaluate radiation exposures of employees and visitors, and ensure the maintenance of related records
- Conduct radiation and contamination monitoring and control programs
- Evaluate the integrity and reliability of radiation detection instruments
- Provide radiation safety support for integrated safety analyses and configuration control
- Provide analysis and approval of proposed changes in process conditions and process equipment involving radiological safety
- Provide advice and counsel to Area Managers on matters of radiation safety
- Support emergency response planning and events
- Assess the effectiveness of the radiation safety program through audit programs

The radiation safety function manager shall hold a BS or BA degree in science or engineering, have at least two years experience in assignments that include responsibility for radiation safety, and have experience in the understanding, application and direction of radiation safety programs.

Minimum qualifications for a senior member of the radiation safety function are a BS or BA degree in science or engineering with at least two years of nuclear industry experience in the assigned function. Alternate minimum experience qualification for a senior member of the radiation safety function is professional certification in health physics. A senior member shall have experience in the assigned safety function, and has authority and responsibility to conduct activities assigned to the radiation safety function.

2.2.1.6 Environmental Protection Function

The environmental protection function is administratively independent of production responsibilities and has the authority to shutdown operations with potentially uncontrolled environmental conditions. Designated responsibilities include:

- Identify environmental protection requirements from federal, state and local regulations which govern the GE-Wilmington operation
- Establish systems and methods to measure and document adherence to regulatory environmental protection requirements and license conditions
- Provide advice and counsel to Area Managers
- Evaluate and approve new, existing or revised equipment, processes and procedures involving environmental protection activities
- Provide environmental protection support for integrated safety analyses and configuration control
- Assure proper federal and state permits, licenses and registrations for non-radiological discharges from the facilities

Minimum qualifications for the manager of the environmental protection function are a BS or BA degree in science or engineering and two years of experience in assignments involving regulatory activities or equivalent.

2.2.1.7 Chemical and Fire Safety Function

The chemical and fire safety function is administratively independent of the production responsibilities and has the authority to shutdown operations with potentially hazardous health and safety conditions. Designated responsibilities include:

- Identify fire protection requirements from federal, state, and local regulations which govern GE-Wilmington operations
- Develop practices regarding non-radiological chemical safety affecting nuclear activities
- Provide advice and counsel to Area Managers on matters of chemical and fire safety

- Provide consultation and review of new, existing or revised equipment, processes and procedures regarding chemical safety and fire protection
- Provide chemical and fire safety support for integrated safety analyses and configuration control

Minimum qualifications of the manager of the chemical and fire safety function are a BS or BA degree in science or engineering and two years of experience in related assignments.

2.2.1.8 Site Security and Emergency Preparedness Function

The site security and emergency preparedness function is administratively independent of the production responsibilities. Designated responsibilities include:

- Provide physical security for the GE-Wilmington facility
- Establish and maintain the emergency preparedness program, including training and program evaluations
- Provide advice and counsel to Area Managers on matters of physical security and emergency preparedness
- Maintain agreements and preparedness with off-site emergency support groups

Minimum qualifications are a BS or BA degree in science or engineering, one year of experience in related assignments, or a high school diploma with eight years of experience in related assignments.

2.2.1.9 Environment, Health & Safety (EHS) Function

The EHS function is administratively independent of production responsibilities but has the authority to enforce the shutdown of any process or facility in the event that controls for any aspect of safety are not assured. This function has designated overall responsibility to establish the radiation safety, criticality safety, environmental protection, chemical safety, fire protection and emergency preparedness programs to ensure compliance with federal, state and local regulations and laws governing operation of a nuclear manufacturing facility. These programs are designed to ensure

the health and safety of employees and the public as well as protection of the environment. The managers of the criticality safety, radiation safety, environmental protection, chemical and fire safety, and site security and emergency preparedness functions report to the EHS function manager.

The manager of the EHS function must hold a BS or BA degree in science or engineering and have five years of management experience in assignments involving regulatory activities. The manager of the EHS function must have appropriate understanding of health physics, nuclear criticality safety, environmental protection, and chemical and fire safety programs.

2.2.2 MANAGEMENT CONTROLS

Management controls for the conduct and maintenance of the GE-Wilmington health, safety and environment protection programs are contained in documented plant practices described in Section 3.9.1, which are approved by cognizant management. Such practices are part of a controlled document system, and appropriately span the organizational structure and major plant activities to control interrelationships, and to specify program objectives, responsibilities and requirements. Personnel are appropriately trained to the requirements of these management controls, and compliance is monitored through internal and independent audits and evaluations.

Management controls documented in practices address requirements including:

- Configuration Management
- Integrated Safety Analysis
- Radiation Safety
- Criticality Safety
- Environmental Protection
- Chemical Safety
- Fire & Explosion Safety
- Emergency Preparedness
- Quality Assurance
- Training

- Procedures
- Maintenance
- Audits
- Incident Investigation & Reporting
- Fissile Material Accountability and Control

2.3 SAFETY COMMITTEES

2.3.1 WILMINGTON SAFETY REVIEW COMMITTEE

The functions of the Wilmington Safety Review Committee include responsibility for the following:

- An annual ALARA review which considers:
  - Programs and projects undertaken by the radiation safety function and the Radiation Safety Committee
  - Performance including, but not limited to, trends in airborne concentrations of radioactivity, personnel exposures, and environmental monitoring results
  - Programs for improving the effectiveness of equipment used for effluent and exposure control
- Review of major changes in authorized plant activities which may affect nuclear or non-nuclear safety practices
- Professional advice and counsel on environmental protection, and criticality, radiation, chemical and fire safety issues affecting the nuclear activities.

The committee is responsible to the GE-Wilmington facility manager. Its proceedings, findings and recommendations are reported in writing to the GE-Wilmington facility manager and to appropriate staff level managers responsible for operations which have been reviewed by the committee. Such reports shall be retained for at least three years.

The committee holds at least three meetings each calendar year with a maximum interval of 180 days between any two consecutive meetings.

2.3.2 RADIATION SAFETY COMMITTEE

The objective of the Radiation Safety Committee is to maintain occupational radiation exposures as low as reasonably achievable (ALARA) through improvements in fuel manufacturing operations.

The committee meets monthly to maintain a continual awareness of the status of projects, performance measurement and trends, and the current radiation safety conditions of shop activities. The maximum interval between meetings does not exceed 60 days.

A written report of each Radiation Safety Committee meeting is forwarded to cognizant Area Managers and the manager of the EHS function. Records of the committee proceedings are maintained for three years.

The committee consists of managers or representatives from key manufacturing functions with activities affecting radiation safety.

CHAPTER 3.0

CONDUCT OF OPERATIONS

3.1 CONFIGURATION MANAGEMENT (CM)

3.1.1 CONFIGURATION MANAGEMENT PROGRAM

A formal configuration management process, governed by written, approved practices, ensures that plant design changes do not adversely impact on safety, health, or environmental protection programs at GE-Wilmington. The configuration management program ensures that the information used to operate and maintain safety controls is kept current. Safety controls are systems, structures, components and procedures which prevent and/or mitigate the risk of accidents. The use of current plant information is an integral part of the integrated safety analysis program described in Chapter 4.0.

The CM program includes the following activities:

- Maintenance of the design information for the plant
- Control of information used to operate and maintain the plant
- Documentation of changes
- Assurance of adequate safety reviews for changes
- Periodic comparison assessment of the conformance of specific safety controls to the documentation of plant design bases

3.1.2 PLANT DESIGN RFQUIREMENTS

Written plant practices define the development, application, and maintenance of the design specifications and requirements. Plant design specifications and requirements are maintained as controlled information. The specific content of the information depends on the age of the design and the requirements in place at the time of design. As a minimum the information required for safe operation of the facility is available.

3.1.3 CHANGE CONTROL

Written plant practices describe the configuration management program for change management, including approval to install and operate facility changes. Facility changes are assessed by a trained and approved safety reviewer to determine if the applicable ISA is impacted, and if further review and approval is required by an ISA team as described in Chapter 4.0.

The written plant practices also prescribe controls and define the distinction between types of changes, ranging from replacement with identical designs which are authorized as part of normal maintenance, to new or different designs which require specified review and approval.

3.1.4 DOCUMENT CONTROL

Documented plant practices define the control system, including creation, revision, storage, tracking, distribution and retrieval of applicable information including :

- Operating procedures
- Drawings
- Technical specifications and requirements
- Software for safety controls
- Calibration instructions
- Functional test instructions

The documented plant practices describe the responsibilities and activities which maintain consistency between the facility design, the physical facility, and the documentation. They also describe how the latest approved revisions are made available for operations.

3.2 MAINTENANCE

The purpose of planned and scheduled maintenance for safety controls is to assure that systems are kept in a condition of readiness to perform the planned and designed functions when required. Area Managers are responsible to assure the operational readiness of safety controls in their assigned facility areas. For this reason the maintenance function is administratively part of or closely coupled to fuel production

operations. The maintenance function utilizes a systems-based program to plan, schedule, track and maintain records for maintenance activities. Maintenance instructions are an integral part of the maintenance system for maintenance activities. Discrimination between specified safety controls and other systems based on integrated safety analyses is maintained in the database. Key maintenance requirements for safety controls such as calibration, functional testing, and replacement of specified components are derived from integrated safety analyses described in Chapter 4.0, and the application of the graded approach to assurance elements.

Maintenance activities generally fall into the categories described below:

3.2.1 SCHEDULED PREVENTIVE MAINTENANCE

Examples of safety controls included for scheduled preventive maintenance are :

- Radiation Measurement Instruments
- Criticality Detection Devices
- Effluent Measurement & Control Devices
- Emergency Power Generators
- Fire Detection and Control Systems
- Pressure Relief Valves
- Air Compressors
- Steam Boilers

3.2.2 PERIODIC FUNCTIONAL TESTING

Examples of safety controls included for periodic functional testing include :

- Criticality Warning System
- Fire Alarm System
- Specified Active Engineered Controls on Process Equipment

Frequencies and requirements for functional testing of various safety controls are derived through quality and reliability activities using a graded approach to assurance

as described in Section 3.3. The integrated safety analysis is the basis for this implementation.

3.2.3 REPAIR OF SAFETY CONTROLS

The maintenance planning and control system provides documentation and records of systems and components which have been repaired or replaced.

When a component of specified safety control is repaired or replaced, the component is functionally verified to assure that it has the capability to perform its planned and designed function when called upon to do so.

If the performance of a repaired or replaced safety control could be different from that or the original component, the change to the safety control is specifically approved under the configuration management program and tested to assure it is likely to perform its desired function when called upon to do so.

3.3 QUALITY ASSURANCE (QA)

The application of assurance measures to safety controls at GE-Wilmington focuses on assuring that these controls are designed, installed, operated and maintained such that their planned function is not compromised.

3.3.1 ASSURANCE ELEMENTS

The following assurance elements are applied to safety controls at GE-Wilmington:

- Organization
- Program
- Equipment/System Design Control
- Procurement Documentation Control
- Instructions, Procedures, and Drawings
- Document Control
- Control of Purchased Materials, Equipment, and Services
- Identification and Control of Materials, Parts, and Components

- Control of Special Processes
- Internal Inspections
- Test Control
- Control of Measuring and Test Equipment
- Handling, Storage and Shipping Controls
- Inspection, Test, and Operating Status
- Control of Nonconforming Materials, Parts, or Components
- Corrective Action
- Records
- Audits

3.3.2 ASSURANCE ELEMENT APPLICATION TO SAFETY CONTROLS

In accordance with documented internal practices, the assurance elements are applied to safety controls in proportion to their importance to safety, and as an integral part of the Integrated Safety Assessment program described in Chapter 4.0. This graded approach segregates safety controls and activities into three categories in applying the assurance elements:

- For safety controls intended to prevent or mitigate the consequences of the highest risk category, each of the assurance elements are specifically evaluated and applied to the control, and their application requirements documented as part of the ISA. Justification for each assurance element not applicable to a control in this category is also documented.
- For safety controls intended to prevent or mitigate the consequences of the mid-level risk category, each of the assurance elements is thoroughly evaluated and applicable assurance elements and their requirements are applied and documented.
- Safety Controls in the low risk category are operated and maintained as part of routine and prudent industry practice, and are controlled by means of normal, established manufacturing assurance systems. No extraordinary assurance element requirements are documented.

Assurance element requirements and application decisions are based on sound engineering practices and judgment.

Assurance element descriptions and application, are included in documented practices as part of the GE-Wilmington management system. These practices also specify the requirements for related record retention.

3.4 TRAINING AND QUALIFICATION

Training is provided for each individual at GE-Wilmington, commensurate with assigned duties. Training and qualification requirements are met prior to personnel fully assuming the duties of safety-significant positions, and before assigned tasks are independently performed. Formal training relative to safety includes radiation and radioactive materials, risks involved in receiving low level radiation exposure in accordance with 10CFR19.12, basic criteria and practices for radiation protection, nuclear criticality safety principles not verbatim, but in general conformance with ANSI/ANS 8.20 guidance, chemical and fire safety, maintaining radiation exposures and radioactivity in effluents As Low As Reasonably Achievable (ALARA), and emergency response.

The system established for maintaining records of training and retraining is described in Section 3.8.

3.4.1 NUCLEAR SAFETY TRAINING

Training policy requires that employees complete formal nuclear safety training prior to unescorted access in the airborne radioactivity controlled area. Methods for evaluating the understanding and effectiveness of the training includes passing an initial examination covering formal training contents and observations of operational activities during scheduled audits and inspections.

Such training is performed by trained instructors approved by the manager of the criticality safety function and the manager of the radiation safety function. Training program contents are reviewed on a scheduled basis by the manager of the criticality safety and radiation safety functions to ensure that training program contents are current and adequate.

Previously trained employees who are allowed unescorted access to the airborne radioactivity controlled area are retrained at least every two years. The effectiveness

of the training program is evaluated by an initial training exam. Visitors are trained commensurate with the scope of their visit and/or escorted by trained employees.

3.4.2 OPERATOR TRAINING

Operator training is performance based, and incorporates the structured elements of analysis, design, development, implementation, and evaluation. Job-specific training includes applicable procedures and safety provisions, and requirements. Emphasis is placed on safety requirements where human actions are important to safety. Operator training and qualification requirements are met prior to process safety-related tasks being independently performed or before startup following significant changes to safety controls.

3.5 HUMAN FACTORS

Human factors are an integral part of the management and operational safety philosophy at GE-Wilmington. The consideration of human factors in relation to operational safety is included in integrated safety analyses.

Human factors concepts are also considered in:

- Equipment design
- Safety control design
- Operator training
- Maintenance
- Audits and assessments
- Incident investigations

3.6 AUDITS AND ASSESSMENTS

Planned and scheduled internal and independent audits are performed to evaluate the application and effectiveness of management controls and implementation of programs related to activities significant to plant safety. Written operating procedures are based on GE-Wilmington practices, applicable regulations and license conditions. Audits are performed to assure that operations are conducted in

accordance with the operating procedures, and to assure that safety programs reflected in the operating procedures are maintained.

3.6.1 CRITICALITY, RADIATION, CHEMICAL AND FIRE SAFETY AUDITS

Representatives of the criticality safety function, the radiological safety function, and the chemical and fire safety function conduct formal, scheduled safety audits of fuel manufacturing and support areas in accordance with documented, approved practices. These audits are performed to determine that operations conform to criticality, radiation, and chemical and fire safety requirements.

Criticality and radiological audits are performed quarterly (at intervals not to exceed 110 days) under the direction of the manager of the criticality safety function and the manager of the radiation safety function. Chemical and fire safety audits are performed under the direction of the chemical and fire safety function manager. Personnel performing audits do not report to the production organization and have no direct responsibility for the function and area being audited.

Audit results are communicated in writing to the cognizant Area Manager and to the manager of the environment, health & safety function. Required corrective actions are documented and approved by the Area Manager, reported to the GE-Wilmington facility manager, and tracked to completion by the environment, health & safety function.

Radiation protection personnel within the radiation safety function conduct weekly nuclear safety inspections of fuel manufacturing and support areas in accordance with documented procedures. Inspection findings are documented and sent to the affected Area Manager for resolution.

Records of the audit or inspection, instructions and procedures, persons conducting the audits or inspections, audit or inspection results, and corrective actions for identified violations of license conditions are maintained in accordance with procedural requirements for a minimum period of three years.

3.6.2 ENVIRONMENTAL PROTECTION AUDITS

An audit schedule of the environmental protection program is developed by the environmental protection function on an annual basis. Audits are conducted in

accordance with documented practices to ensure that operational activities conform to documented environmental requirements.

Personnel under the direction of the manager of the environmental protection function perform the environmental protection audits. Personnel performing the audits do not report to the production organization and have no direct responsibility for the function and area being audited.

Audit findings are communicated to the cognizant Area Manager, who is responsible for nonconformance corrective action commitments in accordance with documented practices. The manager of the environmental protection function or delegate is responsible for resolution follow-up for identified nonconformances. Audit results in the form of corrective action items are reported to the GE-Wilmington facility manager and staff for monitoring of closure status.

3.6.3 INDEPENDENT AUDITS

The GE-Wilmington safety program elements (radiation, criticality, chemical, fire protection, industrial safety and environmental protection) are audited biennially by appropriately trained and experienced individuals who have a degree of independence of the GE-Wilmington organization, and are not involved in the routine performance of the work or program being audited. The scope of independent audits covers the adequacy of the safety program as well as compliance to requirements.

Audit results are reported in writing to the GE-Wilmington facility manager, the Area Managers, the manager of the radiation safety function, and the manager of the criticality safety function, as appropriate. The safety function and/or Area Managers, as appropriate, take necessary response actions in accordance with documented corrective action commitments.

Audit results in the form of corrective action items are reported to the GE-Wilmington facility manager and staff for tracking until closure.

3.7 INCIDENT INVESTIGATIONS

Unusual events which potentially threaten or lessen the effectiveness of health, safety or environmental protection are reviewed by the Area Manager and reported to the environment, health & safety function in accordance with documented practices and

methods. Each event is considered in terms of reporting requirements in accordance with applicable regulatory requirements. The depth of investigation relates to the severity or potential severity of the event in judgment of such factors as levels of uranium released and/or the degree of potential for exposure of workers, the public or the environment.

Documented incident investigation practices provide for:

- Formal and systematic analyses for determination of root cause(s)
- Investigations by independent, qualified teams when warranted
- Documented identification and tracking of corrective actions
- Documentation and record retention for purposes of application of "lessons learned"

The environment, health and safety function is responsible for maintaining a list of agencies to be notified, determining if a report to an agency is required, and for notifying the agency when required. This function has the responsibility for continuing communications with government agencies.

3.8 RECORDS MANAGEMENT

Records appropriate to integrated safety analyses and the application of appropriate assurance elements to resulting controls, criticality and radiation safety activities, training/retraining, occupational exposure of personnel to radiation, releases of radioactive materials to the environment, and other pertinent safety activities are maintained in such a manner as to demonstrate compliance with license conditions and regulations.

Records of integrated safety analyses and results are retained during the conduct of the activities analyzed and for six months following cessation of such activities to which they apply or for a minimum of three years.

Records of criticality safety analyses are maintained in sufficient detail and form to permit independent review and audit of the method of calculation and results. Such records are retained during the conduct of the activities and for six months following cessation of such activities to which they apply or for a minimum of three years.

Records associated with personnel radiation exposures are generated and retained in such a manner as to comply with the relevant requirements of 10 CFR 20. The

following additional radiation protection records will be maintained for at least three years:

- Records of the safety review committee meetings
- Surveys of equipment for release to unrestricted areas
- Instrument calibrations
- Safety audits
- Personnel training and retraining
- Radiation work permits
- Surface contamination surveys
- Concentrations of airborne radioactive material in the facility
- Radiological safety analyses

Records associated with the environmental protection activities described in Chapter 10 are generated and retained in such a manner as to comply with the relevant requirements of 10 CFR 20 and this license.

3.9 PROCEDURES

Licensed material processing or activities will be conducted in accordance with properly issued and approved practices and procedures (P&P), plant practices or operating procedures.

3.9.1 PLANT PRACTICES

Licensed material activities are conducted in accordance with management control programs described in administrative and general plant practices approved and issued by cognizant management at a level appropriate to the scope of the practice. These documented practices direct and control activities across the manufacturing functions, and assign functional responsibilities and requirements for these activities. Management controls described in Chapter 2.0 are included in these practices. These practices are reviewed for updating at least every two years.

3.9.2 OPERATING PROCEDURES

Area Managers are responsible to assure preparation of written, approved and issued operating procedures incorporating control and limitation requirements established by the criticality safety function, the radiation safety function, the environmental protection function and the chemical and fire safety function. Integrated safety analysis results as described in Chapter 4.0 are used to identify procedures necessary for human actions important to safety. Operating procedures are initiated and controlled within the guidelines of the configuration management system described in Section 3.1. Area Managers assure that operating procedures are made readily available in the work area and that operators are trained to the requirements of the procedures and that conformance is mandatory. Operators are trained to report inadequate procedures, and/or the inability to follow procedures.

Nuclear safety control procedure requirements for workers in uranium processing areas are incorporated into the appropriate operating, maintenance and test procedures in place for uranium processing operations .

The safety program design requires the establishment and maintenance of documented procedures for environmental, health and safety limitations and requirements to govern the safety aspects of operations. Requirements for procedure control and approval authorities are documented. Procedure review for updating frequencies are as follows:

Document	Review Frequency	Reviewing & Approving Functional Manager
Operating Procedures (OPs) {Note: Nuclear Safety Release/Requirement (NSR/R) limitations and requirements are incorporated into OPs}	When changed (1)	Area Manager and Affected EHS Discipline (Radiation, Criticality, Environmental, Industrial (4), or MC&A)
Operating Procedures (OPs)	Every 3 Years (3)	Area Manager and Affected EHS Discipline (Radiation, Criticality, Environmental, Industrial (4), or MC&A)
Nuclear Safety Instructions (NSIs)	Every 2 Years (2)	Radiation & Criticality Safety
Environmental Protection Instructions (EPIs)	Every 2 Years (2)	Environmental Protection

1) The safety awareness portions of these OPs are reviewed and updated by the appropriate environment, health, and safety (EHS) discipline when warranted based on process related facility change requests.
2) Every 2 years means a maximum interval of 26 months.
3) Every 3 years means a maximum interval of 39 months
4) EHS Discipline - Industrial means normal worker safety, chemical safety, and fire and explosion protection.

Nuclear safety control procedure requirements for workers in uranium processing areas are incorporated into the appropriate operating, maintenance and test procedures in place for uranium processing operations.

CHAPTER 4.0

INTEGRATED SAFETY ANALYSIS

4.1 INTEGRATED SAFETY ANALYSIS

Integrated Safety Analysis (ISA) is the focal point for safety at GE-Wilmington. ISA is a process in which multifunctional teams analyze the hazards at the site to determine accident scenarios and risk, and ensure that controls are in place to prevent and/or mitigate accidents. The risk associated with an accident scenario is used to judge the level of ongoing assurance that is applied to controls which are in place to prevent the accident. The broad scope of the team's analysis includes criticality safety, radiological safety, environmental protection and industrial safety including chemical safety and fire protection. The accident scenarios identified in the ISA are reviewed by the appropriate safety functions to ensure that the plant continues to comply with site safety policy and regulatory limits.

GE commits to establish and maintain the controls identified in the ISA and to provide an appropriate level of assurance to ensure their reliability. The ISA will be maintained current through the configuration management process (Section 4.10).

This program applies to the Dry Conversion Process (DCP) and other process areas as they become baselined using the ISA process.

4.2 SITE DESCRIPTION

A general description of the site is included in Chapter 1.0. More detailed site information is included in the Environmental Report described in Chapter 10.0. The credible external events which are considered by the ISA teams are defined in an established written practice.

4.3 FACILITY DESCRIPTION

Safety-significant information describing the facility, including arrangement of buildings on the site, location with respect to the site boundary, and the facility's ability to withstand credible external events, is included in drawings and reports maintained under configuration management.

4.4 PROCESS DESCRIPTION

Processes covered by this license are summarized in Chapter 1.0. Detailed information concerning these processes is typically included in technical reports, nuclear safety analyses, operating procedures, Process & Instrumentation Drawings (P&IDs), and other detailed process information, which is maintained under configuration management.

4.5 PROCESS SAFETY INFORMATION

Process technology information is gathered and maintained for future use by ISA teams. Technical reports, which typically include process chemistry, intended inventories, and safe upper and lower limits for process variables such as temperature, pressure, flow, and composition, are maintained under configuration management.

Process equipment information is maintained in accurate condition through configuration management. Examples include P&IDs, materials of construction, electrical classification, ventilation system design, and safety systems including interlocks, detection, and suppression systems.

Hazardous material information, including toxicity, permissible exposure limits, physical data, reactivity data, corrosivity data, and thermal and chemical stability data, is available to employees and ISA teams in the form of Material Safety Data Sheets (MSDS's).

4.6 TRAINING AND QUALIFICATIONS OF THE ISA TEAM

ISAs are conducted by teams of individuals with diverse, pertinent knowledge and experience. The team members are chosen to provide operational and technical expertise in the study area, and appropriate safety expertise based on the hazards that are known to exist in the study area. The composition of the team is defined in an established plant practice.

4.7 ISA METHODS

The hazards in the facility are identified and analyzed using methodology that is widely accepted throughout the chemical industry. Examples of the methodology are

described in Guidelines for Hazard Evaluation Procedures, published by the Center for Chemical Process Safety of the American Institute of Chemical Engineers (1992). Hazards are analyzed using established methods, for example:

- Preliminary Hazards Analysis
- What If / Checklist
- Hazards and Operability Analysis
- Failure Mode and Effect
- Fault Tree
- Event Tree
- Human Reliability Analysis

Procedural guidance is provided to the ISA teams in the form of a written plant practice that outlines the special treatment these methods require when applied to processes in the nuclear industry. Examples of this special treatment includes the consideration of criticality and radiological hazards. In this procedure, the teams are instructed to consider start-up, shutdown, upsets, and maintenance, in addition to normal operating conditions. Guidance is provided concerning the external events which must be considered in ISAs.

The written plant practice also provides guidelines for ranking accident scenarios according to risk, that is, unmitigated consequence and likelihood. The team then ensures that the controls that prevent or mitigate accidents are of the appropriate quality and reliability.

4.8 RESULTS OF THE ISA

The results of the ISA team's analysis are communicated in a summary report to appropriate levels of management. This report summarizes the elements that are important to safety in the area studied. The lists of hazards and accident scenarios are compiled and maintained by the configuration management function. Guidance to the teams is provided in a written plant practice to ensure comprehensive reports.

4.9 CONTROLS FOR PREVENTION AND MITIGATION OF ACCIDENTS

Controls which are relied upon to prevent or mitigate serious accidents are maintained in a ready state through the application of a wide range of assurances. Examples of assurances typically used at GE-Wilmington include: configuration management, preventative maintenance, functional tests, quality assurance, purchasing specifications, training, procedures, audits, assessments and inspections. The level of assurance applied is consistent with the level of risk associated with the specific accident scenario. Responsible risk management requires consideration of the components of risk, specifically consequences and likelihood. Accident scenarios are rated by the ISA teams in terms of unmitigated consequences and likelihood of an initiating event according to criteria defined in written plant practices.

The general categories of consequences are defined as follows: the highest category is assigned to accidents that could result in injury to the public located outside the site boundary and to extreme on-site catastrophes. The middle level is assigned to accidents that would result in regulatory violations and/or serious on-site consequences. All other accidents are assigned to the lower level. These categories are summarized in Table 4.1.

Table 4.1
Consequence Levels

Severity Ranking	Radiological/ Criticality	Environmental/ Industrial/Chemical
3	• exposure to an individual member of the public off-site (5 rem, 30 mg intake of U) • severe exposure to an employee (400 rem internal plus external dose or 230 mg intake of U)	• fatality medical treatment for a member of the public off-site • permanent disability • off-site contamination above regulatory standards
2	• exceed regulatory limits for employee exposure (5 rem, 10 mg U internal)	• serious injury • exceed permit limits or regulatory limits • lost time injury • reportable release
1	• exceed administrative limits on daily air samples, lung counts, bioassays, contamination, TLDs • 10% of annual exposure limit	• OSHA recordable injury • first aid • exceed internal limits • spill inside containment • UIR

Accident scenarios are rated according to the likelihood of occurrence. The likelihood is categorized in qualitative terms that can easily be applied by the ISA teams. The highest category of likelihood is applied to initiating events that could occur at any time in the immediate future. The middle category is for events that are likely to occur during the life of the operation. The lowest likelihood category is used for events that are not expected to occur during the life of the facility. In order to provide consistency in ranking, quantitative levels are provided as guidelines to the teams. These levels are summarized in Table 4.2.

Table 4.2
Likelihood Levels

LEVEL	FREQUENCY	LIKELIHOOD
3	more frequent than once every two years	likely to occur in the immediate future
2	every two to fifty years	likely to occur during the life of the facility
1	less frequent than once every fifty years	not likely to occur during the life of the facility
0	incredible	likelihood is indistinguishable from zero

The levels of consequence and likelihood are combined to estimate the level of risk of initiating a particular accident. Figure 4.1 demonstrates the risk assignment matrix. This risk assignment is used by the teams to determine the level of assurance that will be applied to the controls that protect against that particular accident.

Figure 4.1
Risk Assignment Matrix

Consequence				
	3	**Mid-level Risk**	**Highest Risk**	**Highest Risk**
	2	**Low Risk**	**Mid-level Risk**	**Highest Risk**
	1	**Low Risk**	**Low Risk**	**Mid-level Risk**
		1	**2**	**3**

Likelihood

Controls that prevent or mitigate events in the highest risk category receive full evaluation and appropriate application of all assurance elements defined in Chapter 3.0. Appropriate assurance elements are applied to mid-level risk controls. Low risk controls are treated with normal, prudent attention.

4.10 ADMINISTRATIVE CONTROL OF THE ISA

The ISA is maintained current through a configuration management program that ensures that: 1) facility changes receive adequate integrated safety review, and 2) changes are adequately documented.

Proposed facility changes are reviewed by a trained and approved integrated safety reviewer to determine if the change impacts the existing ISA. If so, an ISA team is assembled, and the change is analyzed. The results of the ISA and the recommendations of the team are used in approving or rejecting the proposed change. After the change is implemented, the revised ISA becomes a part of the controlled documentation for the facility.

The trained and approved integrated safety reviewer possesses the experience, training and skills to consider criticality, radiological, environmental, chemical, and industrial impact within a predefined set of limits. The reviewer is approved by the

manager of the EHS function and reports organizationally to the manufacturing product line or the EHS function.

CHAPTER 5.0

RADIATION SAFETY

5.1 ALARA (AS LOW AS IS REASONABLY ACHIEVABLE) POLICY

The GE-Wilmington standard of care for occupationally exposed individuals is to maintain exposures below the limits established by the U.S. Nuclear Regulatory Commission. Beyond the standard of care, the GE-Wilmington professional staff has a commitment to an ALARA program which is delineated by documented plant practices. Area Managers are responsible for implementing the ALARA program via engineered controls and supervision of operations personnel. The radiation safety function ensures that occupational radiation exposures are maintained ALARA via timely exposure monitoring and interaction with production personnel.

An annual ALARA review is conducted by the Wilmington Safety Review Committee as described in Chapter 2.0. The Radiation Safety Committee, also described in Chapter 2.0, meets monthly to maintain a continual awareness of the status of projects, performance measurements and trends, and the current radiation safety conditions of shop activities.

5.2 RADIATION SAFETY PROCEDURES AND RADIATION WORK PERMITS (RWPS)

Routine work performed in radiation controlled areas is administered by the use of standard procedures described in Chapter 3.0. Non-routine activities, particularly those performed by non-GE employees, which generally are not covered by documented procedures, are administered by the RWP system. The RWP system is described in documented plant practices.

Radiation Work Permits are issued by a radiation safety technician or supervisor for non-routine operations not addressed by an operating procedure when special radiation control requirements are necessary. The RWP specifies the necessary radiation safety controls, as appropriate, including personnel monitoring devices, protective clothing, respiratory protective equipment, special air sampling, and additional precautionary measures to be taken. RWPs are reviewed by radiation safety supervision.

The RWP requirements are reviewed by each affected individual and a copy is made available to the radiation safety function throughout the duration of the activity. Work is monitored by the radiation safety function as required. RWPs have expiration dates and the status of issued RWPs is reviewed on a weekly basis by a radiation safety technician or supervisor.

5.3 VENTILATION REQUIREMENTS

5.3.1 INTER-AREA AIR FLOW DESIGN

Ventilation equipment is designed to provide air flow from areas of lesser potential contamination to areas of higher potential contamination. Direction of air flow between areas is checked monthly or after significant changes to the ventilation system. If insufficient air flow results in airborne concentrations greater than 10 DAC, then the affected processes are shut down. Specific facilities and capabilities of ventilation systems are detailed in Table 5.1.

5.3.2 ENCLOSURES AND LOCALIZED VENTILATION

Hoods and other localized ventilation designs are utilized to minimize personnel exposure to airborne uranium. Activities and process equipment that generate airborne uranium are designed with filtered enclosures, hoods, dust capturing exhaust ports and other devices which maintain air concentrations of radioactivity in work areas such that personnel exposures are below 10 CFR 20 limits under normal operating conditions.

Air flows through hood openings and localized vents are maintained in accordance with Table 5.1. Additionally, differential pressure indicators are installed across exhaust system filters to monitor system performance. The flows and differential pressures are checked monthly or after significant changes to the ventilation system. If insufficient air flow results in airborne concentrations greater than 10 DAC, then the affected processes are shut down in accordance with plant procedures.

5.3.3 EXHAUST SYSTEM

Potentially contaminated air is exhausted through high efficiency filter media which are at least 99.97% efficient for removal of 0.3 micron particles. HEPA filters in the

exhaust system are equipped with a device for measuring differential pressure. Differential pressures greater than four inches of water are investigated. In no case will filters be operated at a differential pressure which exceeds the manufacturer's ratings for the filter.

Water scrubbers or other appropriate devices are provided where necessary to treat effluents before filtration. Such scrubbers are installed so that effectiveness of filters is maintained.

5.3.4 AIR RECIRCULATION

Room air may be recirculated within the uranium processing areas after being filtered. Room air recirculated within areas where airborne concentrations are likely to exceed 0.1 DAC is filtered by HEPA filters and/or water scrubbers.

5.4 AIR SAMPLING PROGRAM

5.4.1 AIR SAMPLING

Air samples are continuously taken from each main process area where airborne concentrations are likely to exceed 0.1 DAC when averaged over 40 hours to assess the concentrations of uranium in air. The air samples are collected in such a way that the concentrations of uranium measured are representative of the air which workers breathe. Air sampling results and individual personnel exposure assignments are monitored by the radiation safety function to evaluate the effectiveness of personnel exposure controls.

Fixed filter sampling points utilized for personnel exposure assignments are evaluated for representativeness annually and as part of each radiation safety function review for licensed process or equipment changes that may affect airborne concentrations. Evaluations of air sampling representativeness are performed in accordance with the methods and acceptance criteria in Table 2 of Regulatory Guide 8.25, "Air Sampling in the Workplace".

Filters from air samplers are changed each shift during normal operating periods or at more frequent intervals following the detection of an event that may have released airborne uranium, based upon knowledge of the particular circumstances. Filters are not changed as frequently during periods when no work is in progress. The filters are processed to determine the uranium concentration in air for each area.

Each air sampler is equipped with a rotameter to indicate flow rate of air sampled. These rotameters are calibrated or replaced at least every 18 months.

Air sampling results in excess of 2.5 DAC (8 hr. sample) and not resulting from a specific known cause are investigated to determine the probable cause. Operations or equipment will be shut down, and immediate corrective action will be taken, at locations where an air sample exceeds 10 DAC without a specific known cause. Corrective actions are implemented and documented based on the frequency and magnitude of events causing releases of airborne uranium.

Routine air sampling is supplemented by portable air sample surveys as required to evaluate non-routine activities or breaches in containment. Based on these surveys, additional radiation protection requirements for the particular operation may be established.

5.4.2 AIR SAMPLING ADJUSTMENTS

Adjustments to Derived Air Concentration (DAC) and Annual Limit of Intake (ALI) values in process areas to reflect actual physical characteristics of the airborne uranium are made in accordance with written operating procedures. GE-Wilmington site specific information on characteristics of airborne uranium is documented in internal records. For those areas in which adjusted ALI/DACs are applied, controls are established to limit soluble uranium intakes using air sampling and urinalysis.

Assigned air adjustments are not made to ALI/DACs for operations, locations or incidents where the airborne uranium physical characteristics are not documented.

Established airborne uranium limits in each area that adjusted ALI/DACs are used pursuant to the above authorization are reassessed by the Radiation Safety function at one quarter of the work locations at 6-month intervals, selecting different locations each time.

If the reassessed limit (ALI) has changed by more than 30% from the previously established limit for an area, the limit for that area is either re-established or replaced with a default value based upon 1 micron AMAD.

In addition, a reassessment is performed following process or equipment changes likely to affect the airborne particle size distribution.

5.5 CONTAMINATION CONTROL

5.5.1 SURVEYS

Routine contamination survey monitoring is performed for uranium process and manufacturing areas including non-controlled areas such as hallways and lunch rooms immediately adjacent to controlled areas. Removable contamination measurements are made based on the potential for contamination in these areas and operational experience. Survey frequencies are determined by the radiation safety function. Survey results are compared to action guide values as specified in plant procedures and appropriate responses are taken.

The minimum survey frequencies and maximum removable contamination action levels are as follows:

Area	Frequency	Action Limit (dpm α/100 cm^2)
Controlled Areas (Floors & Other Readily Accessible Surfaces)	Weekly	≥5,000
Eating Areas used primarily by Controlled Area Personnel	Weekly	≥220
Non-controlled Areas	Monthly	≥220

When contamination levels in excess of action limits are found, mitigating actions are taken within 24 hours.

Personnel contamination surveys for external contamination on clothing and the body are required by personnel when leaving controlled areas. If contamination is found in excess of background levels, the individual attempts self-decontamination at the facilities provided in the change rooms. If decontamination attempts are not successful, decontamination assistance will be provided by the radiation safety function. If skin or personal clothing is still found contaminated above background levels, the individual may not leave the area without prior approval of the radiation protection function.

5.5.2 ACCESS CONTROL

Access points to controlled areas are established through change rooms. Each change room includes a step-off area provided between the hot and cold sides. Instructions controlling entry and exit from controlled area are posted at the entry points. Personnel survey meters are provided in the step-off area of each change room for use by personnel leaving the controlled area. Posted instructions address the use of the survey meters and appropriate decontamination methods.

5.5.3 PROTECTIVE CLOTHING

Protective clothing is provided to persons who are required to enter the controlled areas where personnel contamination potential exists as determined by the radiation safety function. The amount and type of protective clothing required for a specific area or operation is determined by operational experience and the contamination potential. Available clothing includes caps, hoods, laboratory coats, coveralls, safety glasses, boots overshoes, shoe covers, rubber and cloth gloves and safety shoes.

The minimum clothing requirement for airborne controlled area entry is as follows:

Area Workers	Inspectors and Visitors Only Observing Operations
Shoe covers or work area shoes	Shoe covers
Coveralls	Laboratory coats
Head covers	Head covers
Rubber gloves	Rubber gloves (as needed)
Safety glasses	Safety glasses

The protective clothing is removed upon exit in the controlled area change rooms.

In laboratory areas where uranium is handled the minimum protective clothing requirement for entry is a laboratory coat and safety glasses.

5.5.4 LEAK TESTING OF PLUTONIUM ALPHA SOURCES

The sources when not in use shall be stored in a closed container adequately designed and constructed to contain plutonium which might otherwise be released during storage.

The sources shall be tested for loss of plutonium at intervals not to exceed 110 days, using radiation detection instrumentation capable of detecting 0.005 μCi of alpha contamination.

If any survey or measurement performed as required by the preceding paragraph discloses the loss of more than 0.005 μCi of plutonium from the source, or if a source has been damaged or broken, the source shall be deemed to be losing plutonium. The licensee shall immediately withdraw it from use, and cause the source to be decontaminated and repaired, or disposed of in accordance with the Commission regulations.

Records of test results shall be kept in units of microcuries and maintained for inspection by the Commission.

Notwithstanding the periodic test required above, any plutonium alpha source containing not more than 0.1 μCi of plutonium is exempted from the above requirements.

5.6 EXTERNAL EXPOSURE

Deep-dose equivalent and shallow-dose equivalent from external sources of radiation are determined by individually assigned dosimeters. Personnel dosimeters are exchanged quarterly and processed by a National Voluntary Laboratory Accreditation Program (NVLAP) accredited vendor. The capability exists to process dosimeters expeditiously if there is an indication of an exposure in excess of established action guides. Action guides for external exposures are established in plant procedures. Maximum radiation exposure action levels are specified in Section 5.9.

External exposures may be calculated by the radiation safety function on the basis of data obtained by investigation when the results of individual monitoring are unavailable or are invalidated by unusual exposure conditions.

5.7 INTERNAL EXPOSURE

Intakes are assigned to individuals based upon one or more types of measurements as follows: air sampling (described in Section 5.4), urinalysis and in vivo lung counting. Intakes are converted to committed dose equivalent (CDE) and committed effective dose equivalent (CEDE) for the purposes of limiting and recording occupational doses. Action levels are established in plant procedures to prevent an individual from exceeding the occupational exposure limits specified in 10 CFR 20. Maximum

radiation exposure action levels are specified in Section 5.9. Control actions include temporarily restricting the individual from working in an area containing airborne radioactivity, and actions are taken as necessary to assure against recurrence.

5.7.1 URINALYSIS PROGRAM

The urinalysis program is conducted primarily to evaluate the intake of soluble uranium to assure that the 10 CFR 20 intake limit of 10 mg is not exceeded. Individuals assigned to work in areas where soluble airborne uranium compounds are present in concentrations that are likely to result in intakes in excess of 10 percent of the applicable limits in 10 CFR 20 are monitored by urinalysis. The minimum sampling frequency for these individuals is biweekly. Urinalysis may also be used to monitor individuals involved in non-routine operations, perturbations or incidents.

Urine sampling frequencies and action levels are established in plant procedures based on the appropriate biokinetic models for the uranium compounds present. Results above the applicable action level are investigated. Urinalysis action levels are based on maximum radiation exposure action levels specified in Section 5.9. Results that exceed action levels result in a temporary work restriction for the individual to prevent additional exposure and allow a more accurate assessment of the intake.

5.7.2 IN VIVO LUNG COUNTING

Routine in vivo lung counting frequencies are established for individuals who normally work in areas where non-transportable uranium compounds are processed. Baseline and termination counts are performed when feasible. Lung counting frequencies are based upon individual airborne exposure assignments and previous counting results. The minimum count frequency is annual for individuals with an assigned intake greater than 10 percent of the Annual Limit on Intake (ALI).

Appropriate actions are taken based upon in vivo lung counting results to ensure the ALI will not be exceeded. If an individual's lung burden indicates an intake greater than the applicable action level, the individual is temporarily restricted from working in areas containing airborne uranium. In vivo lung counting action levels are based on the maximum radiation exposure action levels specified in Section 5.9.

5.8 SUMMING INTERNAL AND EXTERNAL EXPOSURE

Internal and external exposures determined as described in the preceding sections of this application are summed in accordance with the requirements of 10 CFR 20 for the purposes of limiting occupational doses and recording individual monitoring results.

5.9 ACTION LEVELS FOR RADIATION EXPOSURES

Work activity restrictions will be imposed when an individual's exposure exceeds 80% of the applicable 10 CFR 20 limit; i.e., 0.8 ALI, 1600 DAC-Hours, 4.0 rem CEDE, 4.0 rem TEDE, 4.0 rem DDE, 40 rem CDE and 40 rem SDE.

5.10 RESPIRATORY PROTECTION PROGRAM

The respiratory protection program shall be conducted in accordance with the applicable portions of 10 CFR 20. Respiratory protection equipment specifically approved by the National Institute for Occupational Safety and Health (NIOSH) is utilized.

5.10.1 QUALIFICATIONS OF RESPIRATOR USERS

Individuals designated to use respiratory protection equipment are evaluated by the medical function and periodically thereafter at a frequency specified by the medical function to determine if the individual is medically fit to use respiratory protection devices. If the individual has no restrictions, he is provided respiratory training and fitting by a qualified instructor. Additional training on the use and limitations of self-contained breathing devices is provided to designated individuals.

An adequate mask fit is determined using qualitative (irritant smoke) methods. Mask fits are re-evaluated annually.

5.10.2 RESPIRATORY PROTECTION EQUIPMENT

Half mask respirators equipped with particulate filters are utilized as a precautionary measure and to further reduce exposures during routine operations which may generate uranium dusts. A protection factor of 10 is taken for this type of respirator usage.

Full face mask respirators, equipped with an appropriate canister, are utilized as a precautionary measure and to further reduce exposures for routine and emergency actions which may require additional protection capabilities when there exists a potential for releases of soluble uranium gases. A protection factor of 50 is taken for this type of respirator usage.

NIOSH approved continuous flow airline supplied hoods and full face respirators, and self-contained breathing devices are also available for certain operations. Respiratory protection equipment of these types are operated in accordance with 10 CFR 20 and specified protection factors are utilized.

5.10.3 TESTING AND CLEANING OF EQUIPMENT

Each respirator is processed for cleaning, inspection, and replacement of parts as necessary. Air purifying cartridges and canisters are challenge atmosphere penetration and differential pressure tested against parameters according to internal procedures. The respirator and canister assembly is challenge atmosphere tested and pressure tested prior to reuse. New respirators and canisters are similarly tested on a quality control basis.

Self contained breathing devices are inspected for operational capability and are cleaned and re-inspected after each use.

5.11 INSTRUMENTATION

Appropriate radiation detection instruments are available in sufficient number to ensure adequate radiation surveillance can be accomplished. Selection criteria of portable and laboratory counting equipment is based on the types of radiation detected, maintenance requirements, ruggedness, interchangeability and upper and lower limits of detection capabilities. The radiation safety function annually reviews the appropriateness of the types of instruments being used for each monitoring function. Table 5.2 lists examples of the types and uses of available instrumentation.

5.11.1 CALIBRATION

Instrumentation is calibrated before initial use, after major maintenance, and on a routine basis at least six months following the last calibration. Calibration consists of a performance check on each range scale of the instrument with a radioactive

source of known activity traceable to a recognized standard such as the National Institute of Standards and Technology (NIST).

Prior to each use, operability checks are performed on monitoring and laboratory counting instruments. The background and efficiency of laboratory counting instruments are determined on a daily basis when in use.

TABLE 5.1
SPECIFIC FACILITIES & CAPABILITIES OF VENTILATION SYSTEMS

Facility	Alarms, Interlocks & Safety Features	Purpose
Hoods	Air flow during operation $\geq$ 80 linear feet per minute	Prevents spread of radioactive materials
	Effluent air filtered with HEPA filters	Prevents release of radioactive materials to environs
High Velocity Local Ventilation	Air flow designated to maintain an average of 200 linear feet per minute	Prevents spread of radioactive materials from work area to immediate room area
UF_6 Vaporization Chambers	Vented enclosure	Provides containment in event of cylinder rupture or abnormal leakage
Recirculating Air Systems & Exhaust Air Systems	Air filtered in potentially contaminated zones with HEPA filters or water scrubbers	Removes essentially all contaminants from room and exhaust to environs
	Pressure drop indicator set to alarm at $\geq$4" $H_2O\Delta P$ across final filter	Maintains adequate circulation for removal of dust and contaminants from the room air
	Effluent air filtered with HEPA filters	Prevents release of radioactive materials in environs

TABLE 5.2
TYPES & USES OF AVAILABLE INSTRUMENTATION (TYPICAL)

Type	Typical Range	Routine Use
DOSE RATE METERS		
GM Low Range	0.01 mR - 2000 mR	Area Dose Rate Survey, Shipment Survey
GM High Range	0.1 mR - 1000 R	Emergency Monitoring
Ion Chamber - Low Range	0.1 mR - 10 R	Area Dose Rate Survey, Shipment Survey
Ion Chamber - High Range	1 mR - 1000 R	Emergency Monitoring
ALPHA SURVEY METERS	50 cpm - 2 x 10^6 cpm	Direct Personnel & Equipment Surveys
NEUTRON METERS	0.5 mR - 5 R	Special Dose Rate Surveys
LABORATORY INSTRUMENTATION		
Automatic air sample counter	N/A	Lab Analysis
Fixed geometry Geiger-Mueller counter	N/A	Lab Analysis
Scintillation Counter	N/A	Lab Analysis
In Vivo Lung Counter	N/A	Lung Deposition Measurements

CHAPTER 6.0

NUCLEAR CRITICALITY SAFETY

6.1 PROGRAM ADMINISTRATION

6.1.1 CRITICALITY SAFETY DESIGN PHILOSOPHY

The Double Contingency Principle as identified in nationally recognized American National Standard ANSI/ANS-8.1 (1983) is the fundamental technical basis for design and operation of processes within the GE-Wilmington fuel manufacturing operations using fissile materials. As such, "process designs will incorporate sufficient margins of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible." For each significant portion of the process, a defense of one or more system parameters is documented in the criticality safety analysis, which is reviewed and enforced.

The established design criteria and nuclear criticality safety reviews are applicable to:

- all new processes, facilities or equipment that process, store, transfer or otherwise handle fissile materials, and
- any change in processes, facilities or equipment which may have an impact on the established basis for nuclear criticality safety.

6.1.2 EVALUATION OF CRITICALITY SAFETY

6.1.2.1 Changes to Facility

As part of the design of new facilities or significant additions or changes in existing facilities, Area Managers provide for the evaluation of nuclear hazards, chemical hazards, hydrogenous content of firefighting materials, and mitigation of inadvertent unsafe acts by individuals. Specifically, when criticality safety considerations are impacted by these hazards, the approval to operate new facilities or make significant changes, modification, or additions to existing facilities is documented in accord

with established facility practices and conform to configuration management function 'Integrated Safety Analysis' (ISA) requirements described in Chapter 4.0.

Change requests are processed in accordance with configuration management requirements described in Chapter 3.0. Change requests which establish or involve a change in existing criticality safety parameters require a senior engineer who has been approved by the criticality safety function to disposition the proposed change with respect to the need for a criticality safety analysis.

If an analysis is required, the change is not placed into operation until the criticality safety analysis is complete and other preoperational requirements are fulfilled in accordance with established configuration management practices.

6.1.2.2 Role of the Criticality Safety Function

Qualified personnel as described in Chapter 2 assigned to the criticality safety function determine the basis for safety for processing fissile material. Assessing both normal and credible abnormal conditions, criticality safety personnel specify functional requirements for criticality safety controls commensurate with design criteria and assess control reliability. Responsibilities of the criticality safety function are described in Chapter 2.0.

6.1.3 OPERATING PROCEDURES

Procedures that govern the handling of enriched uranium are reviewed and approved by the criticality safety function.

Each Area Manager is responsible for developing and maintaining operating procedures that incorporate limits and controls established by the criticality safety function. Area Managers assure that appropriate area engineers, operators, and other concerned personnel review and understand these procedures through postings, training programs, and/or other written, electronic or verbal notifications.

Documentation of the review, approval and operator orientation process is maintained within the configuration management system. Specific details of this system are described in Chapter 3.0.

6.1.4 POSTING AND LABELING

6.1.4.1 Posting of Limits and Controls

Nuclear criticality safety requirements for each process system that are defined by the criticality safety function are made available to work stations in the form of written or electronic operating procedures, and/or clear visible postings.

Posting may refer to the placement of signs or marking of floor areas to summarize key criticality safety requirements and limits, to designate approved work and storage areas, or to provide instructions or specific precautions to personnel such as:

- Limits on material types and forms.
- Allowable quantities by weight or number.
- Allowable enrichments.
- Required spacing between units.
- Control limits (when applicable) on quantities such as moderation, density, or presence of additives.
- Critical control steps in the operation.

Storage postings are located in conspicuous places and include as appropriate:

- Material type.
- Container identification.
- Number of items allowed.
- Mass, volume, moderation, and/or spacing limits.

Additionally, when administrative controls or specific actions/decisions by operators are involved, postings include pertinent requirements identified within the criticality safety analysis.

6.1.4.2 Labeling

Where practical, process containers of fissile material are labeled such that the material type, U-235 enrichment, and gross weights can be clearly identified or determined. Deviations from this process include: large process vessels, fuel rods, shipping containers, waste boxes/drums, contaminated items, UF_6 cylinders

containing heels, cold trap cylinders, samples, containers of 1 liter volume or less, or other containers where labeling is not practical.

6.1.5 AUDITS & INSPECTIONS

6.1.5.1 Audits and Inspections

Details of the facility criticality safety audit program are described in Chapter 3.0. Criticality safety audits are conducted and documented in accordance with a written procedure and personnel approved by the criticality safety function. Findings, recommendations, and observations are reviewed with the Environment, Health & Safety (EHS) function manager to determine if other safety impacts exist. The findings, recommendations, and observations are then transmitted to Area Managers for appropriate action.

Routine surveillance inspections of the processes and associated conduct of operations within the facility, including compliance with operating procedures, postings, and administrative guidelines, are also conducted as described in Chapter 3.

6.1.5.2 Independent Audits

A nuclear criticality safety program review is conducted on a planned scheduled basis by nuclear criticality safety professionals independent of the GE-Wilmington fuel manufacturing organization. This provides a means for independently assessing the effectiveness of the components of the nuclear criticality safety program.

The audit team is composed of individuals recommended by the manager of the criticality safety function and whose audit qualifications are approved by the GE-Wilmington facility manager or Manager, EHS. Audit results are reported in writing to the manager of the criticality safety function, who disseminates the report to line management. Results in the form of corrective action requests are tracked to closure.

6.1.6 CRITICALITY SAFETY PERSONNEL

6.1.6.1 Qualifications

Specific details of the criticality safety function responsibilities and qualification requirements for manager, senior engineer, and engineer are described in Chapter 2.0.

6.1.6.2 Authority

Criticality safety function personnel are specifically authorized to perform assigned responsibilities in Chapter 2.0. All nuclear criticality safety function personnel have authority to shutdown potentially unsafe operations.

6.2 TECHNICAL PRACTICES

6.2.1 CONTROL PRACTICES

Criticality safety analyses identify specific controls necessary for the safe and effective operation of a process. Prior to use in any process, nuclear criticality safety controls are verified against criticality safety analysis criteria. The ISA program described in Chapter 4.0 implement performance based management of process requirements and specifications that are important to nuclear criticality safety.

6.2.1.1 Verification Program

The purpose of the verification program is to assure that the controls selected and installed fulfill the requirements identified in the criticality safety analyses. All processes are examined in the "as-built" condition to validate the safety design and to verify the installation. Criticality safety function personnel observe or monitor the performance of initial functional tests and conduct pre-operational audits to verify that the controls function as intended and the installed configuration agrees with the criticality safety analysis.

Operations personnel are responsible for subsequent verification of controls through the use of functional testing or verification. When necessary, control calibration and routine maintenance are normally provided by the instrument and calibration and/or maintenance functions. Verification and maintenance activities are performed per established facility practices documented through the use of forms and/or computer tracking systems. Criticality safety function personnel randomly review control verifications and maintenance activities to assure that controls remain effective.

6.2.1.2 Maintenance Program

The purpose of the maintenance program is to assure that the effectiveness of criticality safety controls designated for a specific process are maintained at the original level of intent and functionality. This requires a combination of routine maintenance, functional testing, and verification of design specifications on a periodic basis. Details of the maintenance program are described in Chapter 3.0.

6.2.2 MEANS OF CONTROL

The relative effectiveness and reliability of controls are considered during the criticality safety analysis process. Passive engineered controls are preferred over all other system controls and are utilized when practical and appropriate. Active engineered controls are the next preferred method of control followed by administrative controls. A criticality safety control must be capable of preventing a criticality accident independent of the operation or failure of any other criticality control for a given credible initiating event.

6.2.2.1 Passive Engineered Controls

These are physical restraints or features that maintain criticality safety in a static manner (i.e., fixed geometry, fixed spacing, fixed size, nuclear poisons, etc.). Passive engineered controls require no action or other response to be effective when called upon to ensure nuclear criticality safety. Assurance is maintained through specific periodic inspections or verification measurement(s) as appropriate.

6.2.2.2 Active Engineered Controls

A means of criticality control involving active hardware (e.g., electrical, mechanical, hydraulic) that protect against criticality. These devices act by providing predefined automatic action or by sensing a process variable important to criticality safety and providing automatic action (e.g., no human intervention required) to secure the system to a safe condition. Human intervention augmented by warning devices and interlocks that prevent continued operation may be used to sense a process variable. Assurance is maintained through specific periodic functional testing as appropriate. Active engineered controls are fail-safe (e.g., meaning failure of the control results in a safe condition).

6.2.2.3 Administrative Controls

Controls that rely for their implementation on actions, judgment, and responsible actions of people. Their use is limited to situations where passive and active control are not practical. Administrative controls may be proactive (requiring action prior to proceeding) or reactive (proceeding unless action occurs). Proactive administrative controls are preferred. Assurance is maintained through training, experience, and audit.

6.2.3 TABLE OF PLANT SYSTEMS AND PARAMETER CONTROLS

Table 6.0 identifies major process areas or support facility processes within the GE-Wilmington fuel manufacturing complex and support facilities. Table entries for each significant process item highlight the safety basis selected for the criticality safety analysis (CSA) and related worst credible contents (or bounding assumptions). Table column definitions are presented below:

AREA OR SYSTEM: A defined functional group of processes or pieces of equipment that operate as a single unit.

PROCESS SUBAREA OR EQUIPMENT: A defined subgroup of vessels, tanks, process and/or support equipment within an area that operate as a single unit.

BASIS FOR CRITICALITY SAFETY: The controlled parameters established within a CSA for nuclear criticality safety for the identified process subarea or equipment. For multiple parameter entries, the basis for nuclear criticality safety

established in the CSA may be based on the identified parameter(s), as appropriate, including the use of 'coupled' parameter control (e.g., mass/moderation).

> ***NOTE - To be included as section 1.3.15 in final License:*** *Changes from one parameter to another parameter for process subareas or equipment in which multiple (at least two) parameters are controlled are made in accordance with established change control measures and reported to the NRC within 90 days of completion. Changes to single parameter controlled processes or equipment from the identified parameter to a new parameter(s) will require NRC approval prior to the change being made.*

CSA BOUNDING ASSUMPTIONS: These are the values used for physical process parameters which are not directly controlled but represent the most reactive credible values for the system, process subarea, or equipment under consideration. As such, the CSA is performed to consider all process operations and credible upsets that fall within this range of assumptions. For items containing no bounding assumptions, all process operations and credible upsets must be analyzed within the CSA. The approved CSA may limit the operation of the system to levels more conservative than those permitted by the bounding assumptions.

In the following Table 6.0, unless otherwise specified, the enrichment limit for all processes are 5.0 wt. % U235 (or HiE), with the exception of conversion lines 1,2 , and 4 and related MSG lines 1-6 which are presently analyzed for 4.025 wt. % U235 (or LoE). When pails are used for product, 5-gallon cans may be used for LoE enrichments, while 3-gallon containers may be used for HiE material. All scrap material is treated as HiE.

Table 6.0 Plant Systems and Parameter Controls

AREA OR SYSTEM	PROCESS SUBAREA OR EQUIPMENT	BASIS FOR CRITICALITY SAFETY	CSA BOUNDING ASSUMPTIONS
Fuel Support: Storage Pads	UF_6 Cylinder Receipt and Storage	Enrichment	99.5 wt. % pure UF_6 $\leq$ 0.5 wt. % H_2O equivalent Optimal Interunit H_2O
	Scrap 3 and 5-gallon Container Storage	Geometry Mass	Homogeneous or Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	RA-Inner and Outer Container Storage	Geometry Moderation	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Waste Box Container Storage	Geometry/Mass Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	BU-J, BU-7, 7A Drum Storage	Geometry Mass Moderation *	Homogeneous or Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
Fuel Support: New Decon	Waste Box Load	Mass	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Oil Drum Load	Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
Chemical ADU Conversion System	UF_6 Cylinders	Moderation	99.5 wt. % pure UF_6 $\leq$ 0.5 wt. % H_2O equivalent Full Reflection
	Autoclave Vaporization	Moderation	99.5 wt. % pure UF_6 $\leq$ 0.5 wt. % H_2O equivalent Full Reflection
	Cold Trap System	Geometry Moderation	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Hydrolysis Receiver, Storage, and Scrubber Tanks	Geometry Concentration	Homogeneous UO_2F_2 Optimal H_2O Moderation Full Reflection
	Sump	Geometry Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Precipitation Tanks (Lines 1,2,4)	Geometry	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection

*** two out of any three control parameters required for criticality safety.**

AREA OR SYSTEM	PROCESS SUBAREA OR EQUIPMENT	BASIS FOR CRITICALITY SAFETY	CSA BOUNDING ASSUMPTIONS
	Precipitation Tanks (Lines 3, 5)	Geometry Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Dewatering Centrifugation	Geometry Mass	Homogeneous ADU or U_3O_8 Optimal H_2O Moderation Full Reflection Outside Containment
	Clarifying Centrifugation	Geometry Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Calcination	Geometry Geometry/Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Calciner Scrubber	Geometry Concentration	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	3 or 5-Gallon Product Container	Geometry Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	UO_2 Powder Pretreatment: Mill, Slug, Granulate (MSG)	Geometry or Mass Moderation	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	LoE and HiE UO_2 Powder Blending	Geometry Mass/Moderation	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	LoE Fluoride Effluent Vessels	Geometry Concentration	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Line 3 Accumulator/Permeate Vessels	Geometry Concentration	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Nitrate Quarantine Effluent Vessels	Geometry Concentration	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Powder Pack Screener	Geometry Moderation	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Powder Pack Product Container	Geometry Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	HVAC: Wet Areas	Geometry Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection

AREA OR SYSTEM	PROCESS SUBAREA OR EQUIPMENT	BASIS FOR CRITICALITY SAFETY	CSA BOUNDING ASSUMPTIONS
	HVAC: Dry Areas	Mass Moderation	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Exhaust Scrubber	Geometry/Mass Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Utilities: Steam , N_2, H_2, Dissoc. NH4, H_2O Supply	Mass	Backflow into large supply vessels prevented by backflow prevention measures, physical barriers, and/or process characteristics.
	REDCAP: Oxidation Feed Containers	Geometry Mass	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	REDCAP: Oxidation Furnace	Geometry Moderation	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	REDCAP: Oxidation Output Containers	Geometry Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	REDCAP: Oxidation Off-Gas System	Geometry Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Miscellaneous: 3 and 5-Gallon Container Floor storage	Geometry Mass	Homogeneous or Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Integration OXIDIZE 3 and 5-gal. Feed Containers	Geometry Mass	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Integration OXIDIZE 3 and 5-gal. Feed Container Storage	Geometry Mass Moderation } *	Heterogeneous UO_2 Optimal Interunit H_2O Moderation Full Reflection
	Integration: OXIDIZE Feed Hood	Geometry Mass	Homogeneous or Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Integration OXIDIZE Furnace	Geometry Moderation	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Integration RECYCLE Powder Outlet	Moderation	heterogeneous UO_2 Maximum Credible wt. % H_2O Full Reflection

*** two out of any three control parameters required for criticality safety.**

AREA OR SYSTEM	PROCESS SUBAREA OR EQUIPMENT	BASIS FOR CRITICALITY SAFETY	CSA BOUNDING ASSUMPTIONS
	Integration RECYCLE Blender	Moderation	Heterogeneous UO_2 Maximum Credible wt. % H_2O Full Reflection
	Integration RECYCLE DM-10 Vibromill	Moderation Mass	Heterogeneous UO_2 Maximum Credible wt. % H_2O Full Reflection
	Integration RECYCLE Unicone Container Storage	Moderation	Heterogeneous UO_2 Maximum Credible UO_2 Density Maximum Credible wt. % H_2O Optimal Interunit H_2O
	Integration RECYCLE 3-gal. Product Container Storage	Geometry Mass Moderation } *	Heterogeneous UO_2 Optimal Interunit H_2O Moderation Full Reflection
	Integration RECYCLE Powder Transfer Corridor	Moderation	Heterogeneous UO_2 Maximum Credible UO_2 Density Maximum Credible wt. % H_2O Full Reflection
Uranium Recovery Unit (URU) System	Fluoride Waste Process Vessels	Geometry Concentration	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Fluoride Waste Surge Vessel (V-106)	Concentration Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Radwaste Process Vessels	Geometry Concentration	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Nitrate Waste Process Vessels	Geometry Concentration	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Nitrate Waste Surge Vessel (V-103)	Concentration Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Oxidation Feed Containers	Geometry Mass	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Oxidation Furnace	Geometry	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Oxidation Furnace Boat Dump	Geometry Moderation	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection

*** two out of any three control parameters required for criticality safety.**

AREA OR SYSTEM	PROCESS SUBAREA OR EQUIPMENT	BASIS FOR CRITICALITY SAFETY	CSA BOUNDING ASSUMPTIONS
	Oxidation 3-gallon Container Storage	Geometry Mass Moderation *	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Oxidation Off-Gas System	Geometry Mass	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Dissolution: Can Dump Feed Conveyor	Geometry Mass Moderation	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Dissolution: Dissolvers, Pumps, Sumps, Filters, Piping	Geometry Concentration	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Oberlin Filter	Geometry Concentration	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Dissolution: NOX Scrubber	Concentration Mass	Homogeneous UO_2 On-Line Density Meter Full Reflection
	Counter-Current Leaching: Can Dump	Geometry Mass/Moderation	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Counter-Current Leaching: Leach Troughs, Pumps, Filters, Storage Tanks, Product Containers	Geometry Concentration	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Utilities: Steam, DI H_2O, Nitric Acid, Aluminum Nitrate	Mass	Backflow into large supply vessels prevented by backflow prevention measures, physical barriers, and/or process characteristics.
	Head-End Concentrator Process	Geometry Concentration	Homogeneous UNH Optimal H_2O Moderation Full Reflection
	Solvent Extraction Process	Geometry Concentration	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	UNH Product Storage Vessels	Geometry Concentration	Homogeneous UNH Optimal H_2O Moderation Full Reflection

*** two out of any three control parameters required for criticality safety.**

AREA OR SYSTEM	PROCESS SUBAREA OR EQUIPMENT	BASIS FOR CRITICALITY SAFETY	CSA BOUNDING ASSUMPTIONS
	Waste Solvent Drum Load	Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
Uranyl Nitrate Conversion (UCON) System	UNH LEM Tank Feed Tanks	Geometry Concentration	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	UCON: Precipitation Tanks	Geometry Mass	Homogeneous UNH Optimal H_2O Moderation Full Reflection
	UCON: Dewatering Centrifugation	Geometry Mass	Homogeneous ADU or U_3O_8 Optimal H_2O Moderation Full Reflection Outside Containment
	UCON: Clarifying Centrifugation	Geometry Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	UCON Process: Calcination	Geometry Geometry/Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
Waste Treatment Facility (WTF)	Fluoride Waste Barrens Surge Vessel (V-108)	Concentration Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Nitrate Waste Barrens Surge Vessel (V-104)	Concentration Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Centrifuge	Geometry Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Oberlin Filter	Geometry/Mass Concentration	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
Uranium Recovery from Lagoon Sludge (URLS) Facility Process	URLS Process Tanks	Concentration	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	URLS Process Non-Leach Filter Press	Geometry/Concent. Concentration	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	URLS Process Product Waste Container	Concentration Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
Waste Oxidation/ Reduction (Incineration) Facility	Incinerator Combustible Box Feed Containers	Mass (Box Monitor) Mass (E-Gun)	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection

AREA OR SYSTEM	PROCESS SUBAREA OR EQUIPMENT	BASIS FOR CRITICALITY SAFETY	CSA BOUNDING ASSUMPTIONS
	Incinerator	Mass (UPHOLD) Mass (INHOLD)	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Incinerator Product 3 or 5-Gallon Containers	Geometry Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
Dry Conversion Process (DCP) Conversion	UF_6 Cylinder Receipt and Storage	Enrichment	99.5 wt. % pure UF_6 $\leq$ 0.5 wt. % H_2O equivalent Optimal Interunit H_2O
	Vaporization Autoclave w/UF_6 Cylinder	Moderation	99.5 wt. % pure UF_6 $\leq$ 0.5 wt. % H_2O equivalent Full Reflection
	Vaporization Cold Trap System	Geometry Moderation	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Conversion: Reactor/Kiln	Moderation	Homogeneous UO_2 Maximum Credible UO_2 Density Maximum Credible wt. % H_2O Full Reflection
	Conversion: Powder Outlet Box	Moderation	Homogeneous UO_2 Maximum Credible UO_2 Density Maximum Credible wt. % H_2O Full Reflection
	Powder Outlet: Cooling Hopper	Moderation	Homogeneous UO_2 Maximum Credible UO_2 Density Maximum Credible wt. % H_2O Full Reflection
	Powder Transfer & Storage: Normal Product Container	Moderation	Homogeneous UO_2 Maximum Credible UO_2 Density Maximum Credible wt. % H_2O Full Reflection
	Powder Transfer & Storage: Out-of- Spec Moisture Product Container	Geometry Moderation	Homogeneous UO_2 Maximum Credible UO_2 Density Maximum Credible wt. % H_2O Full Reflection
	Homogenization	Moderation	Homogeneous UO_2 Maximum Credible UO_2 Density Maximum Credible wt. % H_2O Full Reflection

AREA OR SYSTEM	PROCESS SUBAREA OR EQUIPMENT	BASIS FOR CRITICALITY SAFETY	CSA BOUNDING ASSUMPTIONS
	Blending, Precompaction, Granulation	Moderation	Heterogeneous UO_2 Maximum Credible UO_2 Density Maximum Credible wt. % H_2O Full Reflection
	Tumbling: in Powder Container	Moderation	Heterogeneous UO_2 Maximum Credible UO_2 Density Maximum Credible wt. % H_2O Full Reflection
	Powder Pack Screener	Moderation	Heterogeneous UO_2 Maximum Credible UO_2 Density Maximum Credible wt. % H_2O Full Reflection
	Powder Pack Product Container	Geometry Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Utilities: N_2, H_2, H_2O Supply, Refrigerant	Mass	Backflow into large supply vessels not credible due to backflow prevention measures, physical barriers, and/or process characteristics.
	HF Effluent Recovery and Storage Vessels	Geometry Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Recycle Blender	Moderation	Heterogeneous UO_2 Maximum Credible UO_2 Density Maximum Credible wt. % H_2O Full Reflection
	Recycle Unicone Product Container/Storage	Moderation	Heterogeneous UO_2 Maximum Credible UO_2 Density Maximum Credible Internal wt. % H_2O Optimal Interunit H_2O
	Recycle 3-Gallon Product Container/ Storage	Geometry Mass Moderation *	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
Press Warehouse Facility Process	Conveyor Storage: 3 and 5-gallon Cans	Geometry Mass Moderation *	Homogeneous UO_2 Optimal Interunit H_2O Moderation Full Reflection
	Powder Dump Transfer Hopper/Chute	Geometry Moderation	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Pellet Presses	Geometry/Mass Moderation	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection

*** two out of any three control parameters required for criticality safety.**

AREA OR SYSTEM	PROCESS SUBAREA OR EQUIPMENT	BASIS FOR CRITICALITY SAFETY	CSA BOUNDING ASSUMPTIONS
	Press Lubricant Sump	Geometry Mass	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Press: Green Pellet Boat Product Container	Geometry Moderation	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	3-gallon Powder Cleanup Container	Geometry Mass	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Integration: PWDR-MRA Press Feed	Moderation	Heterogeneous UO_2 Maximum Credible wt. % H_2O Full Reflection
	Integration PWDR-MRA Container-Storage	Geometry/Mass Moderation	Heterogeneous UO_2 Maximum Credible UO_2 Density Maximum Credible wt. % H_2O Full Reflection
	Integration PWDR-MRA Powder Transfer Corridor	Moderation	Heterogeneous UO_2 Maximum Credible UO_2 Density Maximum Credible wt. % H_2O Full Reflection
Pellet Sintering System	Feed/Exit Conveyors	Geometry Moderation	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Sintering Furnace	Geometry Moderation	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
Pellet Grinding System	Feeder Hopper Bowl or Flat Feeder Table	Geometry Moderation	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Grinder	Geometry Moderation	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Grinder APITRON Filter	Geometry Moderation	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Grinder Swarf 3-Gallon Container	Geometry Moderation	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Grinder Hardscrap 3-Gallon Container	Geometry Mass	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection

*** two out of any three control parameters required for criticality safety.**

AREA OR SYSTEM	PROCESS SUBAREA OR EQUIPMENT	BASIS FOR CRITICALITY SAFETY	CSA BOUNDING ASSUMPTIONS
	Grinder Pellet Product Tray	Geometry Mass } * Moderation	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Pellet Transfer Cart	Geometry Moderation	Heterogeneous UO_2 Optimal Interunit H_2O Moderation Full Reflection
Rod Load, Out-Gassing, and Final Rod Welding System	Rod Load, Out-Gassing, and Final Rod Weld	Geometry Moderation	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Pellet Storage Cabinet	Geometry Moderation	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Rod Storage Cabinet	Geometry Moderation	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
Gadolinia Shop	Press, Sintering, Grinding, Rod Load, Rod Storage, & Outgas	Similar to UO_2 Shop Above	Similar to UO_2 Shop Above
	Gadolinia 3 and 5-Gallon Feed Containers	Geometry Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Gadolinia 3 and 5-Gallon Feed & Product Container Storage	Geometry Mass } * Moderation	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Gadolinia DM-10 Vibromill (MCA)	Geometry Moderation	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Gadolinia DM-3 Vibromill (MCA)	Mass Moderation	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Pellet Storage: Ministacker	Geometry/Mass Moderation	Heterogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Integration: Gadolinia MEZZ-MRA Unicone Feed Container	Mass Moderation	Homogeneous UO_2 Maximum Credible UO_2 Density Maximum Credible wt. % H_2O Full Reflection
	Integration Gadolinia MEZZ-MRA DM-10 Vibromill	Moderation	Heterogeneous UO_2 Maximum Credible wt. % H_2O Full Reflection

*** two out of any three control parameters required for criticality safety.**

AREA OR SYSTEM	PROCESS SUBAREA OR EQUIPMENT	BASIS FOR CRITICALITY SAFETY	CSA BOUNDING ASSUMPTIONS
	Integration Gadolinia MEZZ-MRA Rotary Slugger	Moderation	Heterogeneous UO_2 Maximum Credible wt. % H_2O Full Reflection
	Integration Gadolinia MEZZ-MRA Granulator	Moderation	Heterogeneous UO_2 Maximum Credible wt. % H_2O Full Reflection
	Integration: Gadolinia MEZZ-MRA 3 and 5-Gallon Feed & Product Container Storage	Geometry Mass Moderation } *	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Integration Gadolinia MEZZ-MRA Powder Transfer Corridor	Moderation	Heterogeneous UO_2 Maximum Credible UO_2 Density Maximum Credible wt. % H_2O Full Reflection
Bundle Assembly	Rod Trays	Geometry Mass	Heterogeneous UO_2 Optimal Interunit H_2O Moderation Full Reflection
	Rod Storage Cabinets	Geometry Moderation	Heterogeneous UO_2 Optimal Interunit H_2O Moderation Full Reflection
	Rod Tray Transfer Vehicle: "Big Joe"	Geometry Moderation	Heterogeneous UO_2 Optimal Interunit H_2O Moderation Full Reflection
	Magnetic and Passive Scanner: "MAPS"	Geometry Moderation	Heterogeneous UO_2 Optimal Interunit H_2O Moderation Full Reflection
	Bundle Accumulator: "BACC"	Geometry Moderation	Heterogeneous UO_2 Optimal Interunit H_2O Moderation Full Reflection
	Automatic Bundle Assemble Machine: "ABAM"	Geometry Moderation	Heterogeneous UO_2 Optimal Interunit H_2O Moderation Full Reflection
	Rod Scanner: "Fat Albert"	Geometry Moderation	Heterogeneous UO_2 Optimal Interunit H_2O Moderation Full Reflection
	Assembly Table	Geometry Moderation	Heterogeneous UO_2 Optimal Interunit H_2O Moderation Full Reflection
	Upender: Bundle and RA Container	Geometry Moderation	Heterogeneous UO_2 Optimal Interunit H_2O Moderation Full Reflection

*** two out of any three control parameters required for criticality safety.**

AREA OR SYSTEM	PROCESS SUBAREA OR EQUIPMENT	BASIS FOR CRITICALITY SAFETY	CSA BOUNDING ASSUMPTIONS
	Inspection Pit	Geometry Moderation	Heterogeneous UO_2 Optimal Interunit H_2O Moderation Full Reflection
	Bundle Storage: "Forest"	Geometry Moderation	Heterogeneous UO_2 Optimal Interunit H_2O Moderation Full Reflection
	RA Container: Transfer Port & RA Conveyor	Geometry Moderation	Heterogeneous UO_2 Optimal Interunit H_2O Moderation Full Reflection
	Rod Scanner: X-Ray-Unit	Geometry Moderation	Heterogeneous UO_2 Optimal Interunit H_2O Moderation Full Reflection
	Rod Inspection: Surface-Plate	Geometry Moderation	Heterogeneous UO_2 Optimal Interunit H_2O Moderation Full Reflection
	Rod Movement: One & Two-Tray Cart	Geometry Moderation	Heterogeneous UO_2 Optimal Interunit H_2O Moderation Full Reflection
	Container Storage: RA-Inner/Outer Storage	Geometry Moderation	Heterogeneous UO_2 Optimal Interunit H_2O Moderation Full Reflection
Decontamination & Volume Reduction Facility (DVRF)	Wash Down Areas, Sumps, Bag Filters	Geometry/Mass Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	Dust Hog	Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	HVAC	Geometry Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection
	3-Gallon Waste Container Storage	Geometry Mass	Homogeneous UO_2 Optimal H_2O Moderation Full Reflection

6.2.4 SPECIFIC PARAMETER LIMITS

The **safe geometry** values of Table 6.1 below are specifically licensed for use at the GE-Wilmington facility. Application of these geometries is limited to situations where the neutron reflection present does not exceed that due to full water reflection. Acceptable geometry margins of safety for units identified in this table are 93% of the minimum critical cylinder diameter, 88% of the minimum critical slab thickness, and 76% of the minimum critical sphere volume.

When cylinders and slabs are not infinite in extent, the dimensional limitations of Table 6.1 may be increased by means of standard buckling conversion methods; reactivity formula calculations which incorporate validated K-infinities, migration areas (M^2) and extrapolation distances; or explicit stochastic or deterministic modeling methods.

The **safe batch** values of Table 6.2 are specifically licensed for use at the GE-Wilmington facility. Criticality safety may be based on U235 mass limits in either of the following ways:

- If double batch is considered credible, the mass of any single accumulation shall not exceed a safe batch, which is defined to be 45% of the minimum critical mass. Table 6.2 lists safe batch limits for homogeneous mixtures of UO_2 and water as a function of U235 enrichment over the range of 1.1% to 15% for uncontrolled geometric configurations. The safe batch sized for UO_2 of specific compounds may be adjusted when applied to other compounds by the formula:

  kgs X = (kgs UO_2 • 0.88) / f

where.	**kgs X**	**= safe batch value of compound 'X'**
	kgs UO_2	**= safe batch value for UO_2**
	0.88	**= wt. % U in UO_2**
	f	**= wt. % U in compound X**

- Where engineered controls prevent over batching, a mass of 75% of the minimum critical mass shall not be exceeded.

Subject to provision for adequate protection against precipitation or other circumstances which may increase concentration, the following **safe concentrations** are specifically licensed for use at the GE-Wilmington facility:

- A concentration of less than or equal to one-half of the minimum critical concentration.
- A system in which the hydrogen to U235 atom ratio (H/U235) is greater than 5200.

Table 6.1 Safe Geometry Values

Homogeneous UO_2-H_2O Mixtures	Weight Percent U235	Infinite Cylinder* Diameters (Inches)	Infinite Slab* Thickness (Inches)	Sphere Volume* (Liters)
	2.00	16.70	8.90	105.0
	2.25	14.90	7.90	75.5
	2.50	13.75	7.20	61.0
	2.75	12.90	6.65	51.0
	3.00	12.35	6.25	44.0
	3.25	11.70	5.90	38.5
	3.50	11.20	5.60	34.0
	3.75	10.80	5.30	31.0
	4.00	10.50	5.10	29.0
	5.00	9.50	4.45	24.0
Homogeneous Aqueous Solutions	**Weight Percent U235**	**Infinite Cylinder Diameters (Inches)**	**Infinite Slab Thickness (Inches)**	**Sphere Volume (Liters)**
	2.00	16.7	9.30	106.4
	2.25	15.0	8.40	80.5
	2.50	14.0	7.80	66.8
	2.75	13.3	7.30	56.2
	3.00	12.9	7.00	49.7
	3.25	12.5	6.70	44.8
	3.50	12.1	6.50	41.0
	3.75	11.9	6.30	38.0
	4.00	11.7	6.00	34.9
	5.00	9.5	4.80	26.0
Heterogeneous Mixtures or Compounds	**Weight Percent U235**	**Infinite Cylinder Diameters (Inches)**	**Infinite Slab Thickness (Inches)**	**Sphere Volume (Liters)**
	2.00	11.10	5.60	35.7
	2.25	10.50	5.10	30.7
	2.50	10.10	4.80	27.3
	2.75	9.70	4.60	24.7
	3.00	9.40	4.40	22.6
	3.25	9.20	4.30	20.9
	3.50	9.00	4.20	19.2
	3.75	8.90	4.10	18.2
	4.00	8.80	4.00	16.9
	5.00	8.30	3.60	13.0

* These values represent 93%, 88% and 76% of the minimum critical cylinder diameter, slab thickness, and sphere volume, respectively. For enrichments not specified, smooth curve interpolation may be used.

Tabie 6.2 Safe Batch Values for UO_2 and Water*

Nominal Weight Percent U235	Homogeneous UO_2 Powder & Water Mixtures (Kgs UO_2)	Heterogeneous UO_2 Pellets & Water Mixtures (Kgs UO_2)	Nominal Weight Percent U235	Homogeneous UO_2 Powder & Water Mixtures (Kgs UO_2)	Heterogeneous UO_2 Pellets & Water Mixtures (Kgs UO_2)
1.10	2629.0	510.0	4.00	25.7	24.7
1.20	1391.0	341.0	4.20	23.7	22.9
1.30	833.0	246.0	4.40	21.9	21.4
1.40	583.0	193.0	4.60	20.2	20.0
1.50	404.0	158.0	4.80	19.1	18.8
1.60	293.3	135.0	5.00	18.1	18.1
1.70	225.0	116.0			
1.80	183.0	102.0			
1.90	150.6	96.5			
2.00	127.5	81.6			
2.10	109.2	73.1			
2.20	96.8	66.4			
2.30	84.3	61.0			
2.40	74.7	56.1			
2.50	68.9	52.1			
2.60	60.5	48.8			
2.70	56.6	45.4			
2.80	52.2	42.9			
2.90	47.6	40.1			
3.00	44.5	38.1			
3.20	38.9	34.1			
3.40	34.6	31.0			
3.60	31.1	28.5			
3.80	28.3	26.4			

***NOTE: These values represent 45% of the minimum critical mass. For enrichments not specified, smooth curve interpolation of safe batch values may be used.**

6.2.5 CONTROL PARAMETERS

Nuclear criticality safety is achieved by controlling one or more parameters of a system within established subcritical limits. The criticality safety review process is used to identify the significant parameters associated with a particular system. All assumptions relating to process equipment, material composition, function, and operation, including upset conditions, are justified, documented, and independently reviewed.

Identified below are specific control parameters that may be considered during the review process:

6.2.5.1 **Geometry** - Geometry may be used for nuclear criticality safety control on its own or in combination with other control methods. Favorable geometry is based on limiting dimensions of defined geometrical shapes to established subcritical limits. Structure and/or neutron absorbers that are not removable constitute a form of geometry control. At the GE-Wilmington facility, favorable geometry is developed conservatively assuming unlimited water or concrete equivalent reflection, optimal hydrogenous moderation, worst credible heterogeneity, and maximum credible enrichment to be processed. Examples include cylinder diameters, annular inner/outer dimensions, slab thickness, and sphere diameters.

Geometry control systems are analyzed and evaluated allowing for fabrication tolerances and dimensional changes that may likely occur through corrosion, wear, or mechanical distortion. In addition, these systems include provisions for periodic inspection if credible conditions exist for changes in the dimensions of the equipment that may result in the inability to meet established nuclear criticality safety limits.

6.2.5.2 **Mass** - Mass control may be used for a nuclear criticality safety control on its own or in combination with other control methods. Mass control may be utilized to limit the quantity of uranium within specific process operations or vessels and within storage, transportation, or disposal containers. Analytical or non-destructive methods may be employed to verify the mass measurements for a specific quantity of material.

Establishment of mass limits involves consideration of potential moderation, reflection, geometry, spacing, and material concentration. The criticality safety analysis considers normal operations and credible process upsets in determining actual mass limits for the system and for defining additional controls. When only

administrative controls are used for mass controlled systems, double batching is considered to ensure adequate safety margin.

6.2.5.3 **Moderation** - Moderation control may be used for nuclear criticality safety control on its own or in combination with other control methods. When moderation is used in conjunction with other control methods, the area is posted as a 'moderation control area'. When moderation control is the primary design focus and is designated as a the primary criticality safety control parameter, the area is posted 'moderation restricted area'.

When moderation is the primary criticality safety control parameter the following graded approach to the design control philosophy is applied in accordance with established facility practices (in decreasing order of restriction):

- At each enriched uranium interface involving intentional and continuous introduction of moderation (e.g., insertion of superheated steam into reactor), at least three controls are required to assure that the moderation safety factor is not exceeded. At least two of these controls must be active engineered controls.
- At enriched uranium interfaces involving intentional but non-continuous introduction of moderation at least three controls are required to assure that the moderation safety factor is not exceeded. At least one of these controls must be an active engineered control, unless a moderation safety factor greater than 3 is demonstrated.
- For situations where moderation is not intentionally introduced as part of the process, the required number of controls for each credible failure mode must be established in accordance with the double contingency principle.

When the maximum credible accident is considered, the safety moderation limit (i.e., % H_2O or equivalent) must provide sufficient factor of safety above the process moderation limit. This 'moderation safety factor', which is the ratio of the safety moderation limit to the process moderation limit, will normally be three or higher, but never less than two. The value of the moderation safety factor depends on the likelihood and time required for this system being considered to transition from the process moderation limit to the safety moderation limit.

In some cases, as described above, increased depth of protection may be required, but the minimum protection is never less than the following: two independent controls prevent moderator from entering the system through a defined interface and must fail

before a criticality accident is possible. The quality and basis for selection of the controls is documented in accordance with Integrated Safety Analysis process described in Chapter 4.0. Controls for the introduction and limited usage of moderating materials (e.g. for cleaning or lubrication purposes) within areas in which the primary criticality safety parameter is moderation are approved by the criticality safety function.

6.2.5.4 **Concentration (or Density)** - Concentration control may be used for nuclear criticality safety control on its own or in combination with other control methods. Concentration controls are established to ensure that the concentration level is maintained within defined limits for the system. When concentration is the only parameter controlled to prevent criticality, concentration may be controlled by two independent combinations of measurement and physical control, each physical control capable of preventing the concentration limit being exceeded in a location where it would be unsafe. The preferred method of attaining independence being that at least one of the two combinations is an active engineered control. Each process relying on concentration control has in place controls necessary to detect and/or mitigate the effects of internal concentration within the system (e.g., Dynatrol density meter, Rhonan density meter, etc.), otherwise, the most reactive credible concentration (density) is assumed.

6.2.5.5 **Neutron Absorber** - Neutron absorbing materials may be utilized to provide a method for nuclear criticality safety control for a process, vessel or container. Stable compounds such as boron carbide fixed in a matrix such as aluminum or polyester resin; elemental cadmium clad in appropriate material; elemental boron alloyed stainless steel, or other solid neutron absorbing materials with an established dimensional relationship to the fissionable material are recommended. The use of neutron absorbers in this manner is defined as part of a passive engineered control.

Credit may be taken for neutron absorbers such as gadolinia in completed nuclear fuel bundles (e.g., packaged and stored onsite for shipment) provided the following requirements are met:

- The presence of the gadolinia absorber in completed fuel rods is documented and verified using non-destructive testing; and the placement of rods in completed fuel bundles is documented in accordance with established quality control practices.

Credit may be taken for neutron absorbers that are normal constituents of filter media (e.g., natural boron) provided the following requirements are met:

- The failure or loss of the media itself also prevents accumulation of significant quantities of fissile material.
- The neutron absorber content is certified.

For fixed neutron absorbers used as part of a geometry control, the following requirements apply:

- The composition of the absorber are measured and documented prior to first use.
- Periodic verification of the integrity of the neutron absorber system subsequent to installation is performed on a scheduled basis approved by the criticality safety function. The method of verification may take the form of traceability (i.e. serial number, QA documentation, etc.), visual inspection or direct measurement.

6.2.5.6 **Spacing (or Unit Interaction)** - Criticality safety controls based on isolation or interacting unit spacing. Units may be considered effectively non-interacting (isolated) when they are separated by either of the following:

- 12-inches of full density water equivalent, or
- the larger of 12-foot air distance or the greatest distance across an orthographic projection of the largest of the fissile accumulations on a plane perpendicular to the line joining their centers.

For Solid Angle interaction analyses, a unit where the contribution to the total solid angle in the array is less than 0.005 steradians is also considered non-interacting (provided the total of all such solid angles neglected is less than one half of the total solid angle for the system). Transfer pipes of 2 inches or less in diameter may be excluded from interaction consideration, provided they are not grouped in close arrays.

Techniques which produce a calculated effective multiplication factor of the entire system (e.g., validated Monte Carlo or S_n Discrete Ordinates codes) may be used. Techniques which do not produce a calculated effective multiplication factor for the entire system but instead compare the system to accepted empirical criteria, (e.g., Solid Angle methods) may also be used. In either case, the criticality safety analysis must comply with the requirements of Sections 6.1.1 and 6.3.

6.2.5.7 **Material Composition (or Heterogeneity)** - The criticality safety analysis for each process determines the effects of material composition (e.g., type, chemical form, physical form) within the process being analyzed and identifies the basis for selection of compositions used in subsequent system modeling activities.

It is important to distinguish between homogeneous and heterogeneous system conditions. Heterogeneous effects within a system can be significant and therefore must be considered within the criticality safety analysis when appropriate. Evaluation of systems where the particle size varies take into consideration effects of heterogeneity appropriate for the process being analyzed.

6.2.5.8 **Reflection** - Most systems are designed and operated with the assumption of 12-inch water or optimum reflection. However, subject to approved controls which limit reflection, certain system designs may be analyzed, approved, and operated in situations where the analyzed reflection is less than optimum.

In criticality safety analysis, the neutron reflection properties of the credible process environment are considered. For example, reflectors more effective than water (e.g., concrete) are considered when appropriate.

6.2.5.9 **Enrichment** - Enrichment control may be utilized to limit the percent U-235 within a process, vessel, or container, thus providing a method for nuclear criticality safety control. Active engineered or administrative controls are required to verify enrichment and to prevent the introduction of uranium at unacceptable enrichment levels within a defined subsystem within the same area. In cases where enrichment control is not utilized, the maximum credible area enrichment is utilized in the criticality safety analysis.

6.2.5.10 **Process Characteristics** - Within certain manufacturing operations, credit may be taken for physical and chemical properties of the process and/or materials as nuclear criticality safety controls. Use of process characteristics is predicated upon the following requirements:

- The bounding conditions and operational limits are specifically identified in the criticality safety analysis and, are specifically communicated, through training and procedures, to appropriate operations personnel.

- Bounding conditions for such process and/or material characteristics are based on established physical or chemical reactions, known scientific principles, and/or facility-specific experimental data supported by operational history.
- The devices and/or procedures which maintain the limiting conditions must have the reliability, independence, and other characteristics required of a criticality safety control.

Examples of process characteristics which may be used as controls include:

- Conversion and oxidation processes that produce dry powder as a product of high temperature reactions.
- Experimental data demonstrating low moisture pickup in or on uranium materials that have been conditioned by room air ventilation equipment.
- Experimental/historical process data demonstrating uranium oxide powder flow characteristics to be directly proportional to the quantity of moisture present.

6.3 CONTROL DOCUMENTS

6.3.1 CRITICALITY SAFETY ANALYSIS (CSA)

In accordance with ANSI/ANS-8.19 (1984), the criticality safety analysis is a collection of information that "provides sufficient detail clarity, and lack of ambiguity to allow independent judgment of the results." The CSA documents the physical/safety basis for the establishment of the controls. The CSA is a controlled element of the Integrated Safety Analysis (ISA) defined in Chapter 4.0.

The CSA addresses the specific concerns (event sequences) of nuclear criticality safety importance for a particular system. A CSA is prepared or updated for each new or significantly modified unit or process system within the GE-Wilmington facility in accordance with established configuration management control practices defined in Chapter 3.0.

The scope and content of any particular CSA reflects the needs and characteristics of the system being analyzed and includes applicable information requirements as follows:

- **Scope** - This element defines the stated purpose of the analysis.
- **General Discussion** - This element presents an overview of the process that is affected by the proposed change. This section includes as appropriate; process description, flow diagrams, normal operating conditions, system interfaces, and other important to design considerations.
- **Criticality Safety Controls/Bounding Assumptions** - This element defines a minimum of two criticality safety controls that are imposed as a result of the analysis. This section also clearly presents a summary of the bounding assumptions used in the analysis. Bounding assumptions include; worst credible contents (e.g., material composition, density, enrichment, and moderation), boundary conditions, interunit water, and a statement on assumed structure. In addition, this section includes a statement which summarizes the interface considerations with other units, subareas and/or areas.
- **Model Description** - This element presents a narrative description of the actual model used in the analysis. An identification of both normal and credible upset (accident condition) model filenaming convention is provided. Key input listings and corresponding geometry plot(s) for both normal and credible upset cases are also provided.
- **Calculational Results** - This element identifies how the calculations were performed, what tools or reference documents were used, and when appropriate, presents a tabular listing of the calculational result and associated uncertainty (e.g., Keff + 3σ) results as a function of the key parameter(s) (e.g., wt. fraction H_2O). When applicable, the assigned bias of the calculation is also clearly stated and incorporated into both normal and/or accident limit comparisons
- **Safety During Upset Conditions** - This element presents a concise summary of the upset conditions considered credible for the defined unit or process system. This section include a discussion as to how the established nuclear criticality safety limits are addressed for each credible process upset (accident condition) pathway.
- **Specifications and Requirements for Safety** - When applicable, this element presents both the design specifications and the criticality safety requirements for correct implementation of the established controls. These requirements are incorporated into operating procedures, training,

maintenance, quality assurance as appropriate to implement the specifications and requirements.

- **Compliance** - This element concludes the analysis with pertinent summary statements and includes a statement regarding license compliance.
- **Verification** - Each criticality safety analysis is verified in accordance with section 6.3.2.5 by a senior engineer approved by the criticality safety function and who was not involved in the analysis.
- **Appendices** - Where necessary, a summary of information ancillary to calculations such as parametric sensitivity studies, references, key inputs, model geometry plots, equipment sketches, useful data, etc., for each defined system is included.

6.3.2 ANALYSIS METHODS

6.3.2.1 Keff Limit

Validated computer analytical methods may be used to evaluate individual system units or potential system interaction. When these analytical methods are used, it is required that the effective neutron multiplication factors for credible process upset (accident) conditions are less than or equal to 0.97 including applicable biases and calculational uncertainties, that is:

Keff + 3σ - bias $\leq$ 0.97 (accident conditions).

Thus, the established delta-k safety margin used at the GE-Wilmington facility is 0.03.

Normal operating conditions include maximum credible conditions expected to be encountered when the criticality control systems function properly. Credible process upsets include anticipated off-normal or credible accident conditions and must be demonstrated to be critically safe in all cases in accordance with Section 6.1.1. The sensitivity of key parameters with respect to the effect on Keff are evaluated for each system such that adequate criticality safety controls are defined for the analyzed system.

6.3.2.2 Analytical Methods

Methodologies currently employed by the GE-Wilmington criticality safety function include hand calculations utilizing published experimental data (e.g., ARH-600 handbook), Solid Angle methods (e.g., SAC code), and Monte Carlo codes (e.g., GEKENO, GEMER) which utilize stochastic methods to solve the 3D neutron transport equation. Additional Monte Carlo codes (e.g., Keno Va and MCNP) or S_n Discrete Ordinates codes (e.g., ANISN or XSDRNPM) may be used after validation as described in subparagraph (c) below.

GEKENO (Geometry Enhanced KENO) is a multigroup Monte Carlo program which solves the neutron transport equation in 3-dimensional space. The GEKENO criticality program utilizes the 16-energy group Knight-Modified Hansen Roach cross-section data set, and a potential scattering σ_p resonance correction to compensate for flux depression at resonance peaks. GEKENO is normally used for homogeneous systems. For infinite systems, K_∞ can be calculated directly from the Hansen Roach cross-sections using the program KINF.

GEMER (Geometry Enhanced MERit) is a multigroup Monte Carlo program which solves the neutron transport equation in 3-dimensional space. The GEMER criticality program is based on 190-energy group structure to represent the neutron energy spectrum. In addition, GEMER treats resolved resonances explicitly by tracking the neutron energy and solving the single-level Breit-Wigner equation at each collision in the resolved resonance range in regions containing materials whose resolve resonances are explicitly represented. The cross-section treatment in GEMER is especially important for heterogeneous systems since the multigroup treatment does not accurately account for resonance self-shielding.

6.3.2.3 Validation Techniques

Experimental critical data or analytical methods which have been validated (benchmarked) by comparison with experimental critical data in accordance with criteria described in section 4.3 of ANSI/ANS 8.1 (1983) are used as the basis for validation. An analytical method is considered validated when the following are established:

- the type of systems which can be modeled
- the range of parameters which may be treated
- the bias, if any, which exists in the results produced by the method.

Currently GEMER is validated against 123 critical experiments and GEKENO is validated against 56 critical experiments. Both validations produce a bias fit as a function of H/U235 atom ratio. This fit is established against the lower limit of the 3-sigma confidence band (see Figures 6.1 and 6.2). The bias (K_{calc} - 1.0) is applied over its negative range and assigned a value of zero over its positive range. The range of applicability covers all compounds in use at GE-Wilmington and enrichments up to 5.0 % wt. % U235.

FIGURE 6.1 - GEMER BIAS DETERMINATION, PARTICLE WEIGHT

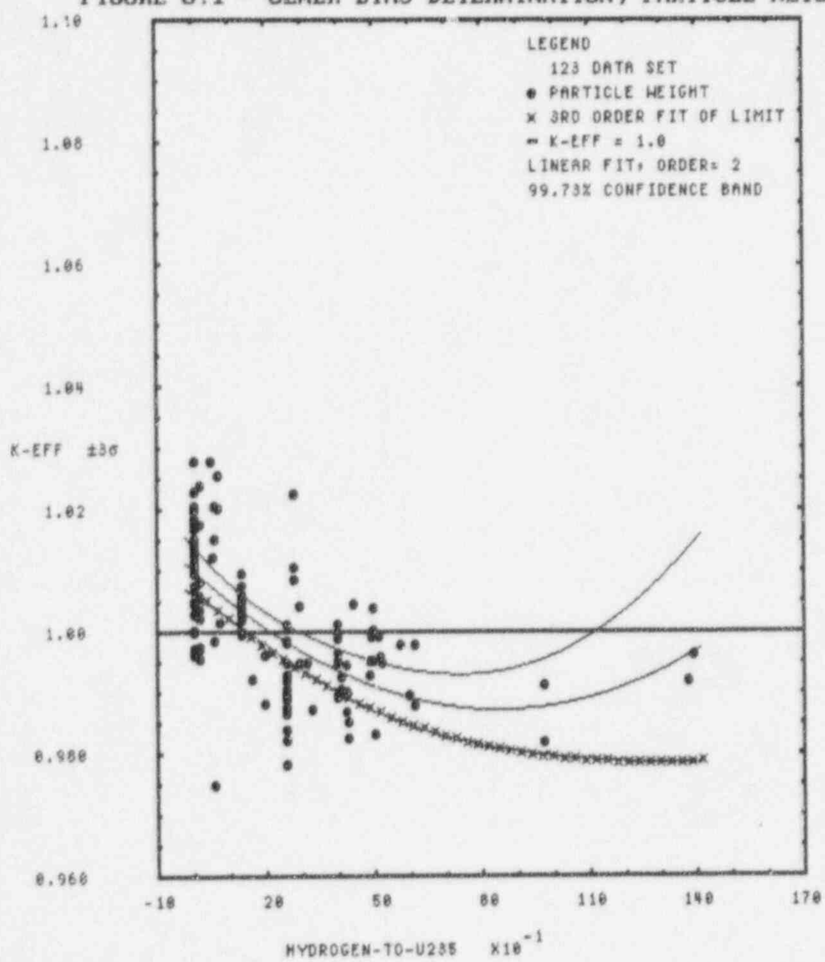
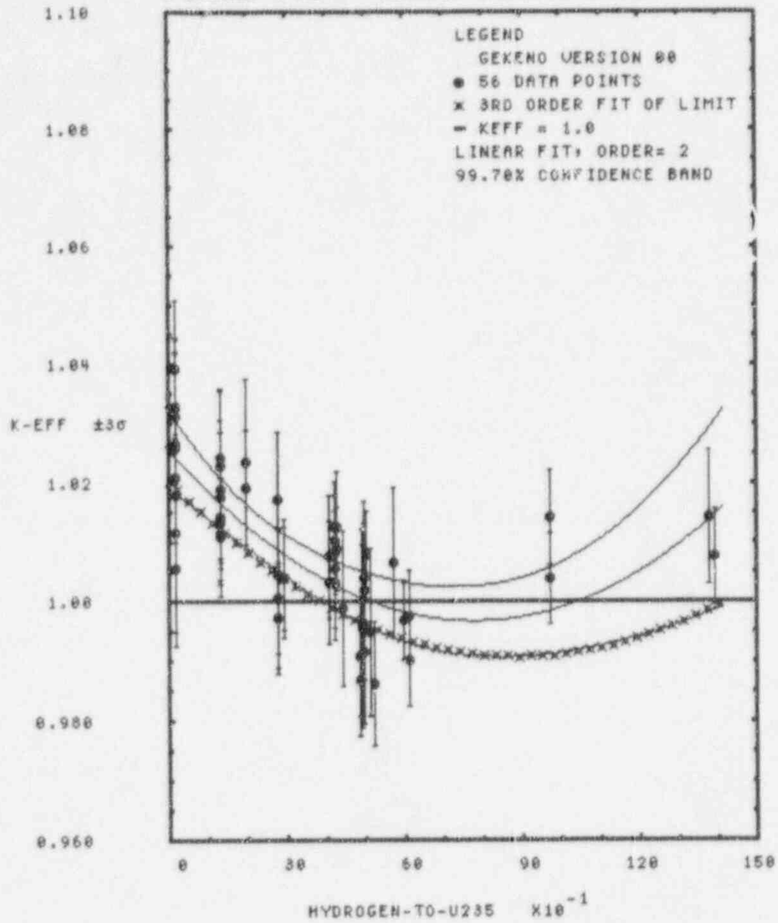


FIGURE 6.2 - GEKENO BIAS CALCULATION



6.3.2.4 Computer Software & Hardware Configuration Control

The software and hardware used within the criticality safety calculational system is configured and maintained so that change control is assured through the authorized system administrator. Software changes are conducted in accordance with an approved configuration control program described in Chapter 3.0 that addresses both hardware and software qualification.

Software designated for use in nuclear criticality safety are compiled into working code versions with executable files that are traceable by length, time, date, and version. Working code versions of compiled software are validated against critical experiments using an established methodology with the differences in experiment

and analytical methods being used to calculate bias and uncertainty values to be applied to the calculational results.

Each individual workstation is verified to produce results identical to the development workstation prior to use of the software for criticality safety calculations demonstrations on the production workstation.

Modifications to software that may affect the calculational logic require re-validation of the software. Modifications to hardware or software that do not affect the calculational logic are followed by code operability verification, in which case, selected calculations are performed to verify identical results from previous analyses. Deviations noted in code verification that might alter the bias or uncertainty requires re-qualification of the code prior to release for use.

6.3.2.5 Technical Reviews

Independent technical reviews of proposed criticality safety control limits specified in criticality safety analyses are performed. A senior engineer within the criticality safety function is required to perform the independent technical review.

The independent technical review consists of a verification that the neutronics geometry model and configuration used adequately represent the system being analyzed. In addition, the reviewer verifies that the proposed material characterizations such as density, concentration, etc., adequately represent the system. He/She also verifies that the proposed criticality safety controls are adequate.

The independent technical review of the specific calculations and computer models are performed using one of the following methods:

- Verify the calculations with an alternate computational method.
- Verify the calculations by performing a comparison to results from a similar design or to similar previously performed calculations.
- Verify the calculations using specific checks of the computer codes used, as well as, evaluations of code input and output.
- Verify the calculations with a custom method.

Based on one of these prescribed methods, the independent technical review provides a reasonable measure of assurance that the chosen analysis methodology and results are correct.

6.4 CRITICALITY ACCIDENT ALARM SYSTEM

6.4.1 SPECIFICATIONS

The criticality accident alarm system radiation monitoring unit detectors are located to assure compliance with appropriate requirements of ANSI/ANS-8.3 (1986). The location and spacing of the detectors are chosen to avoid the effect of shielding by massive equipment or materials. Spacing between detectors is reduced where high density building materials such as brick, concrete, or grout-filled cinder block shield a potential accident area from the detector. Low density materials of construction such as wooden stud construction walls, asbestos, plaster, or metal-corrugated panels, doors, non-load walls, and steel office partitions are disregarded in determining the spacing.

6.4.2 OPERATION

The criticality accident alarm system initiates immediate evacuation of the facility. Employees are trained in recognizing the evacuation signal. This system, and proper response protocol, is described in the Radiological Contingency and Emergency Plan for GE-Wilmington.

6.4.3 MAINTENANCE

The nuclear criticality alarm system is a safety-significant system and is maintained through routine calibration and scheduled functional tests conducted in accordance with internal procedures. In the event of loss of normal power, emergency power is automatically supplied to the criticality accident alarm system.

CHAPTER 7.0

CHEMICAL SAFETY

7.1 CHEMICAL SAFETY PROGRAM

It is the policy of GE-Wilmington to provide a safe and healthy work place by minimizing the risk of chemical exposure to employees and members of the general public. The chemical safety program is applicable to the chemicals associated with the authorized activities in Chapter 1 and include UF_6 and hydrofluoric acid as well as any other chemicals which may directly or indirectly affect the nuclear safety of these activities. The GE-Wilmington chemical safety program is documented in written, approved practices that are followed, and ensures that processes and operations comply with applicable federal and state regulations pertaining to chemical safety.

Hazard evaluations are performed on nuclear and non-nuclear operations within the nuclear manufacturing operations where the potential exists for hazardous chemicals to be used in such a manner that they could effect the nuclear safety program. This ensures appropriate controls are in place for adequate protection of the general public and safe use by employees, and that the use of chemicals does not create potential conditions that adversely effect the handling of licensed nuclear materials.

Employees using hazardous materials are trained to ensure safe handling, use, and disposal.

7.2 CONTENTS OF CHEMICAL SAFETY PROGRAM

The following management control elements are incorporated into GE-Wilmington chemical safety program:

7.2.1 CHEMICAL SAFETY IN INTEGRATED SAFETY ANALYSIS

Considerations of chemical safety for hazardous materials as described in this Chapter are incorporated in GE-Wilmington's Integrated Safety Analysis program. This program includes UF_6 and hydrofluoric acid. GE-Wilmington's Integrated Safety Analysis Program is explained in detail within Chapter 4.0.

7.2.2 CHEMICAL APPROVAL / EVALUATION

Prior to new hazardous materials being brought on-site or used in a process, they are approved through the environmental protection function and the chemical and fire safety function. The formal approval process consists of evaluations of the following potential hazards:

- Physical Hazards
- Health Hazards
- Fire / Explosive Hazards
- Potential Impact on handling of licensed nuclear material

The conclusions of this approval process may dictate the following assurance of chemical process safety:

- New procedures or changes in existing procedures
- Maintenance programs for control related equipment
- Configuration management
- Emergency Planning
- Training

7.2.3 LABELING & IDENTIFICATION

Hazardous materials or conveyance systems are labeled or identified to meet applicable regulations. The proper identification of hazardous materials decreases the likelihood of improper use, handling and disposal reducing potential negative consequences.

7.2.4 EMPLOYEE TRAINING & AWARENESS

Radiation workers receive nuclear safety training and other job related training (Chapter 3, Section 3.4) which includes safety information related to chemicals associated with nuclear material and chemicals in the area which could impact the nuclear safety of the process.

7.2.5 INCIDENT CLASSIFICATION & INVESTIGATION

GE-Wilmington's incident classification and investigation program is discussed in Chapter 3.0.

7.2.6 CONDUCT OF OPERATIONS

Other elements of the chemical safety program are included in Chapter 3.0, "Conduct of Operations".

CHAPTER 8.0

FIRE SAFETY

GE-Wilmington fire protection is achieved by appropriate combinations of fire prevention measures and response systems. Such measures and systems are designed and maintained in accordance with federal, state, and local codes, industry standards and prudent practices. The National Fire Protection Association (NFPA) is the most common standard and practice used as guidance.

8.1 FIRE PROTECTION PROGRAM RESPONSIBILITY

The Emergency Organization is comprised of functional groups capable of assisting and/or advising in the prevention, handling and controlling of emergency situation. The structure of the Emergency Organization is detailed within the Radiological Contingency and Emergency Plan for GE-Wilmington.

8.2 FIRE PROTECTION PROGRAM

Fire hazard analysis is incorporated into the GE-Wilmington's Integrated Safety Analysis (ISA) program and/or site process reviews. The ISA program is described in Chapter 4.0.

Routine inspection and testing of the fire protection system are conducted by GE-Wilmington personnel under the direction of the manager of the site security & emergency preparedness function. Responsibility for maintenance, operation, and engineering of the fire protection system and equipment is specified in written, approved GE-Wilmington practices.

The fire protection program equipment is maintained as part of the formal, planned preventative maintenance program at GE-Wilmington.

Review and control of modifications of the facility or processes to minimize fire hazards is part of configuration management described in Chapter 3.0.

An approved cutting and welding procedure known as a hot work permit is provided to control welding and torch cutting activities as a means of fire prevention.

Basic fire protection training is provided as needed. Additionally new employees and contractors are trained during orientation programs. The emergency response team is given documented training as part of the emergency preparedness program described in Chapter 9.0.

A system is provided to enable reporting of fire incidents to management. Fire alarm pull stations are strategically located throughout the facility. Areas with potential fire hazards are equipped with appropriate fire detection and/or suppression systems.

In order to ensure emergency response readiness a comprehensive emergency exercise is conducted on an annual basis.

8.3 ADMINISTRATIVE CONTROLS

Audits and inspections, which include fire protection, are performed, as follows:

- Internal formal quarterly audits, supplemented by informal inspections.
- Independent auditors perform fire protection, prevention, and inspections of the facility. Action plans are developed to address findings arising from such inspections.

8.4 BUILDING CONSTRUCTION

8.4.1 EXISTING BUILDING

The existing building's original design is in accordance with the local, state, federal and national codes, standards and/or regulations in effect at the time of construction. The building and appurtenances used to process and store hazardous materials are designed to provide containment of such material under the conditions of fire and explosion.

8.4.2 DRY CONVERSION PROCESS FACILITY (DCP)

The building's design is in accordance with the local, state, federal and national codes, standards and/or regulations. The building and appurtenances used to process and store hazardous materials are designed to provide containment of such material

under the conditions of fire and explosion. Recognizing the requirement for moderation restriction, the DCP facility is compartmentalized with 1.5 hour fire walls to control the spread of fire using appropriate techniques.

8.5 VENTILATION SYSTEMS

Ventilation systems are designed to perform the following functions in the event of a fire:

- Air supply closed and air exhaust will continue
- Automatic closing of fire dampers and doors

8.6 PROCESS FIRE SAFETY

Potential fire hazards are determined, evaluated, and controlled by internal and external personnel using industry accepted methods, analysis, and procedures.

8.7 FIRE DETECTION AND ALARM SYSTEMS

8.7.1 Areas where fire or explosion hazards are present, automatic detection equipment is installed. Equipment such as the following is utilized:

- Smoke Detectors
- Heat Detectors
- Hydrogen Detectors (DCP only)
- Hydrogen Fluoride Detectors (DCP only)

8.7.2 Audible fire alarms are installed in specified locations throughout the facility. Such alarms are monitored by a continuously manned, central control station that indicates fire detection system and zone status.

8.7.3 Manual fire alarm actuators (pull-boxes) are installed in appropriate locations throughout the facility and serve to activate a coded fire alarm.

8.8 FIRE SUPPRESSION EQUIPMENT

GE-Wilmington fire protection system is designed in accordance with the NFPA.

Selection of equipment for suppression of fire takes into account the severity of the hazard, the type of activity to be performed, the potential consequences of a fire, and the potential consequences of use of the suppression equipment (including, risk of accidental criticality).

Automatic sprinkler systems are specifically excluded from areas where moderation control is a principal nuclear criticality safety concern.

Portable fire extinguishers, of sufficient capacity, quantity and type of suppression agent used, are available and maintained throughout the facility.

8.9 FIRE PROTECTION WATER SYSTEM

- The fire protection water system is supplied by site water wells.
- Prime components of the fire protection system are as follows:
    - Elevated tank capable of supplying dedicated water to the fire protection system.
    - Ground level fire protection reservoir with a dry hydrant connection.
    - Pump back up system with automatic startup capabilities for supplying the fire protection loop from the retention basin with water at adequate pressure.
    - A jockey pump to maintain sufficient pressure on the fire protection system.
    - A pump under the water tower with automatic startup and manual stop.
    - A fire main loop around the prime production facilities.
    - A series of branch headers supplying fire protection water to sectionalized sprinkler system in each building.

- A supervised alarm and warning system providing full time coverage of prime fire protection safety auxiliaries such as sprinkler system supply valve closing, sprinkler system water flow, fire pump operations, smoke detection operation, etc.
- Fire hose on reels connected to the primary fire protection system.

8.10 RADIOLOGICAL CONTINGENCY AND EMERGENCY PLAN (RC&EP)

GE-Wilmington maintains plans that provide information needed by fire-fighting personnel responding to an emergency. This plan is described in Chapter 9.0.

8.11 EMERGENCY RESPONSE TEAM

Fire training of the Emergency Response Team is conducted for the response to incipient stage fires in accordance with emergency planning requirements. Local volunteer fire departments are contacted for more serious fires which include structural fires.

CHAPTER 9.0

RADIOLOGICAL CONTINGENCY AND EMERGENCY PLAN

GE-Wilmington shall maintain and execute the response measure in the Radiological Contingency and Emergency Plan as specified in Safety License Condition S-3 of Materials License SNM-1097; or as further revised by the licensee consistent with 10 CFR 70.32(i). The Radiological Contingency and Emergency Plan incorporates the requirements established by the Emergency Planning and Community Right-to-Know Act of 1986, Title III, Publication L 99-499.

GE-Wilmington will make no changes to the Radiological Contingency and Emergency Plan which would decrease its effectiveness without prior approval of the NRC.

Changes, which do not decrease the effectiveness of the Radiological Contingency and Emergency Plan, will be reported within six months of the change to the Chief, Licensing Branch, Division of Fuel Cycle Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

The requirements of the Radiological Contingency and Emergency Plan are implemented through approved documented procedures maintained by GE-Wilmington.

CHAPTER 10.0

ENVIRONMENTAL PROTECTION

10.1 AIR EFFLUENT CONTROLS AND MONITORING

Air effluent control systems are designed and operated to assure compliance with regulatory requirements. The adequacy of air effluent controls for operations that might result in the exhausting of radioactive materials, is verified by representative stack sampling to demonstrate compliance with the regulations. Samples are collected and analyzed so as to be representative of the discharges during production operations. Adequate controls and evaluations are in place to monitor, assess and take necessary protective actions that may be needed for circumstances not explicitly treated. The ventilation and exhaust systems are described in Chapter 5.0.

10.2 LIQUID TREATMENT FACILITIES

A liquid treatment facility, with sufficient capacity and capability to enable treatment, sampling, analysis, and discharge of liquids in accordance with the regulations, is provided and maintained in proper working order during the operation of the plant.

Compliance with NRC 10 CFR 20 effluent radioactivity limits for discharge of liquids to the unrestricted area is assured by on-line gamma energy monitoring or other appropriate controls. Quarantine tanks, diversion tanks, and filtration operations are provided to assure the liquid is below regulatory-driven limits. Process RadWaste and laundry streams are released from the Uranium Recovery Unit (URU) to the final process basins. The nitrate-bearing liquids from process areas are directed to the Waste Treatment facility for final treatment and then transferred off-site to a nearby paper manufacturer. The ammonium fluoride-bearing liquids are released from process areas and directed to the Waste Treatment facility for ammonia recovery. The remaining liquids are filtered and then discharged to the final process basins. The discharges from these operations are controlled to assure uranium concentration in the final process effluent is less than 5 ppm in one day or less than 0.2 ppm daily average for a month. Assurance is provided that uranium levels are in compliance with 10 CFR 20.1301 and 1302, thereby meeting the unrestricted release limit.

A continuous proportional sample of process liquid effluent release to the Northeast Cape Fear River is collected. The sampling program design is such that, typically, a daily composite is analyzed for uranium content; a weekly composite of this sample is analyzed for gross alpha activity and gross beta activity; and the determination of technetium 99 is performed on a composite sample which is a collection of weekly samples over a six month period.

Nitrate-bearing liquid, which is produced as a result of tubing etching processes and uranium-processing operations, is transferred to a nearby paper company. This nitrate solution is used as a nutrient in their biological waste treatment facility. Monitoring of these shipments includes sampling of each shipment and composite samples of daily shipments. Lower Limits of detection are (a) 10 ppm U per single truck (shipment); (b) 0.02 ppm U per daily composite sample; and (c) 5 pCi/l gross alpha.

In the dry process for converting UF_6 to UO_2, hydrogen fluoride dissolved in water is generated. This hydrofluoric acid is collected in a bulk storage tank facility to await shipment. Material containing 3 parts per million uranium (or greater concentration) is not released for shipment. The total volume produced will vary based upon production load.

10.3 SOLID WASTE MANAGEMENT FACILITIES

Solid waste management facilities, with sufficient capability to enable preparation, packaging, storage, and transfers to licensed disposal sites in accordance with the regulations, is provided and maintained in proper operating condition as required to support the operation of the plant. Combustible wastes may be incinerated on site.

10.4 PROGRAM DOCUMENTATION

The GE-Wilmington facility licensed activity prepared an Environmental Report dated January 1, 1974, revised July 1983, revised May 1989, and supplemented April 1996. Future Environmental Report updates will be prepared and submitted to the NRC Licensing Staff on a schedule contingent upon the operating term of the license. The review and updating will be concurrent with each renewal application.

10.4.1 MINIMUM PROGRAM IMPLEMENTATION

The GE-Wilmington facility environmental monitoring program includes the elements illustrated in Figure 10.1. Analytical sensitivities (minimum detection levels) are illustrated in Figure 10.1. Action levels will be included in documented procedures for environmental monitoring parameters as appropriate so that internal review and other actions are initiated. Such action levels provide guidance in assuring compliance within 10 CFR 20 limits. Locations of (a) air sampling sites; (b) vegetation and soil sampling points; (c) surface water monitoring points; and (d) monitoring wells are illustrated in Figures 10.2, 10.3, 10.4, and 10.5, respectively. For monitoring wells found not to contain water at time of sampling, an evaluation is performed by the EHS function to determine if alternate well sampling data may be used or other assessments will be used. These program elements, analytical sensitivities, and/or locations may be changed without prior NRC Licensing Staff approval, provided: (1) a documented evaluation by the EHS function demonstrates that the changes will not decrease the overall effectiveness of the environmental monitoring program; and, (2) the documented evaluation is maintained on file at the facility and the changes are submitted to the NRC Licensing Staff in the subsequent Environmental Report update.

10.4.2 REPORTING PROGRAM RESULTS

Radioactivity in releases of radioactive materials in gaseous and liquid effluents from the facility will be reported to the NRC Staff on a semi-annual basis.

10.5 EVALUATIONS

The EHS function performs a periodic evaluation of vendors contracted to analyze environmental samples. The evaluations consider applicable methods such as "spike' and "replicate sample" submittals.

10.6 OFF-SITE DOSE

Compliance with NRC 10 CFR 20, Subpart D, and EPA 40 CFR 190 regulations for off-site dose requirements to the maximally exposed individual is demonstrated by assuring that the off-site annual dose does not exceed 25 mREM. Additionally,

compliance with 10 CFR 20.1101(d) regulations for off-site dose projections due to air emissions is demonstrated by assuring that off-site annual dose (due to air emissions of radioactivity) does not exceed 10 mREM.

10.7 ALARA

Compliance and the ALARA concept are inherent in the Environmental program in terms of comprehensive monitoring, analysis, and evaluation of air emissions, liquid effluents and disposition of solid waste. Management controls, quality assurance and program implementation provide (1) representative measurements of radioactivity in the highest potential exposure pathways and (2) verification of the accuracy of the effluent monitoring program of those environmental exposure pathways. Trends are assessed using monitoring results to evaluate plant operations, in terms of "control-at-the-source" of contamination and the containment of radioactivity; the projections of potential dose to off-site populations; and the detection of any unanticipated pathways for the transport of radionuclides within the environment. Monitoring with periodic evaluations are summarized and presented to senior management on an annual basis.

FIGURE 10.1
GE-NE ENVIRONMENTAL MONITORING PARAMETERS

TYPE OF SAMPLE	ANALYSES	TYPICAL SAMPLING FREQUENCY	ROUTINE MINIMUM DETECTION LEVEL	
Air Particulates - Point Sources	Alpha	Continuous (Collection Weekly)	1.0E-12 microcuries per milliliter	
Ambient Air - On-Site	Alpha	Continuous (Collection Weekly)	0.5E-15 microcuries per milliliter	
Process Liquid At On-Site Discharge Point	Uranium Content Alpha; Beta	Daily; Weekly	0.02 parts per million - uranium	
			3.0E-8 microcuries per milliliter alpha	5E-8 microcuries per milliliter beta
Ground Water - On-Site	Uranium Content Alpha; Beta	Monthly; Quarterly	0.02 parts per million - uranium	
			5 picocuries per liter - alpha	20 picocuries per liter - beta
River Water - Upstream and Downstream of Site Discharge	Alpha; Beta	Monthly	5 picocuries per liter alpha	20 picocuries per liter beta
Sediment - Above Site Dam	Uranium	Annually	0.02 parts per million - uranium	
Soil - On-Site	Uranium	Semi-Annually	0.02 parts per million - uranium	
Vegetation - On-Site	Fluoride	Semi-Annually	1.0 parts per million - fluoride	

Locations of Ambient Air Sampling Sites

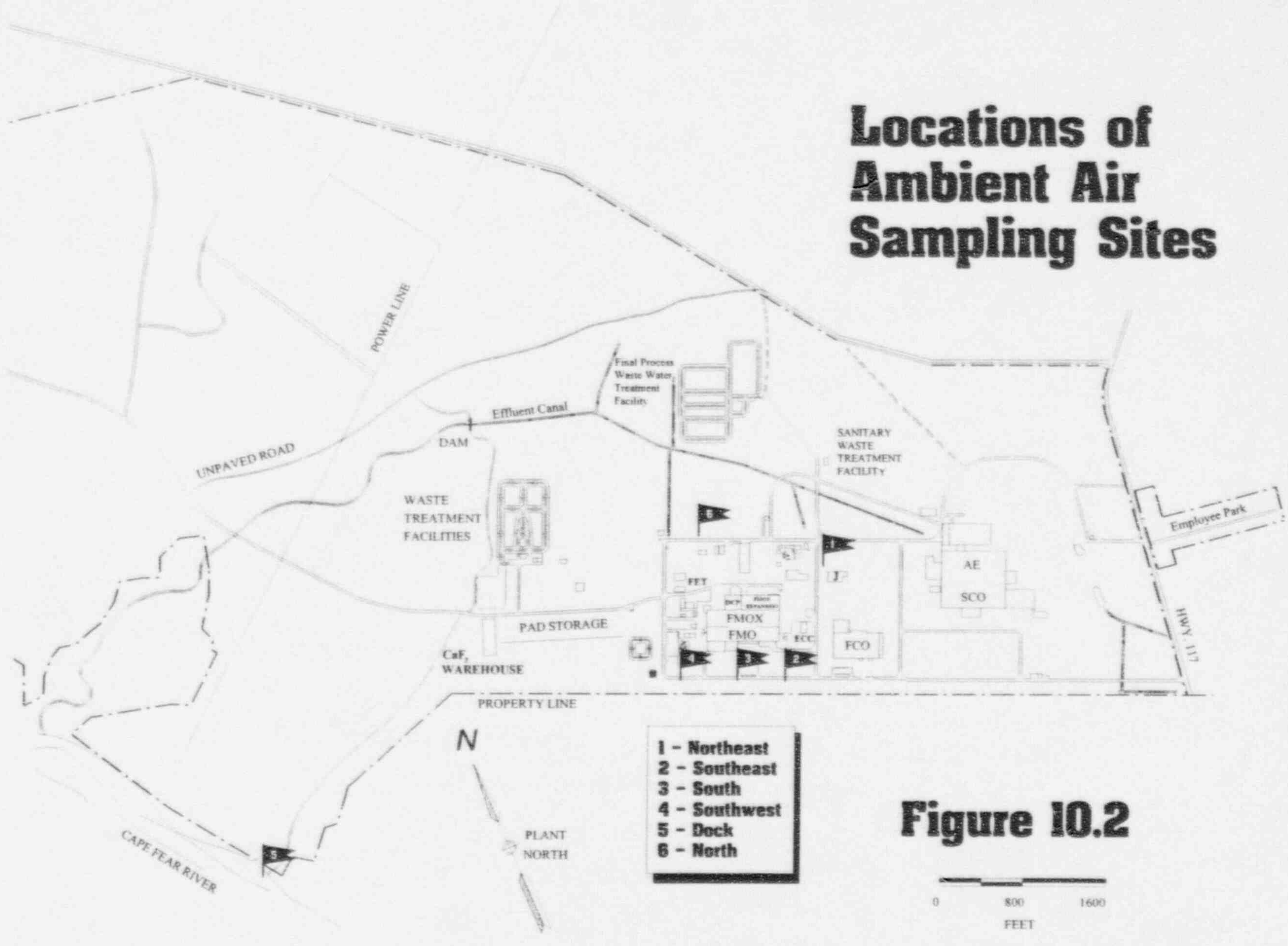


Figure 10.2

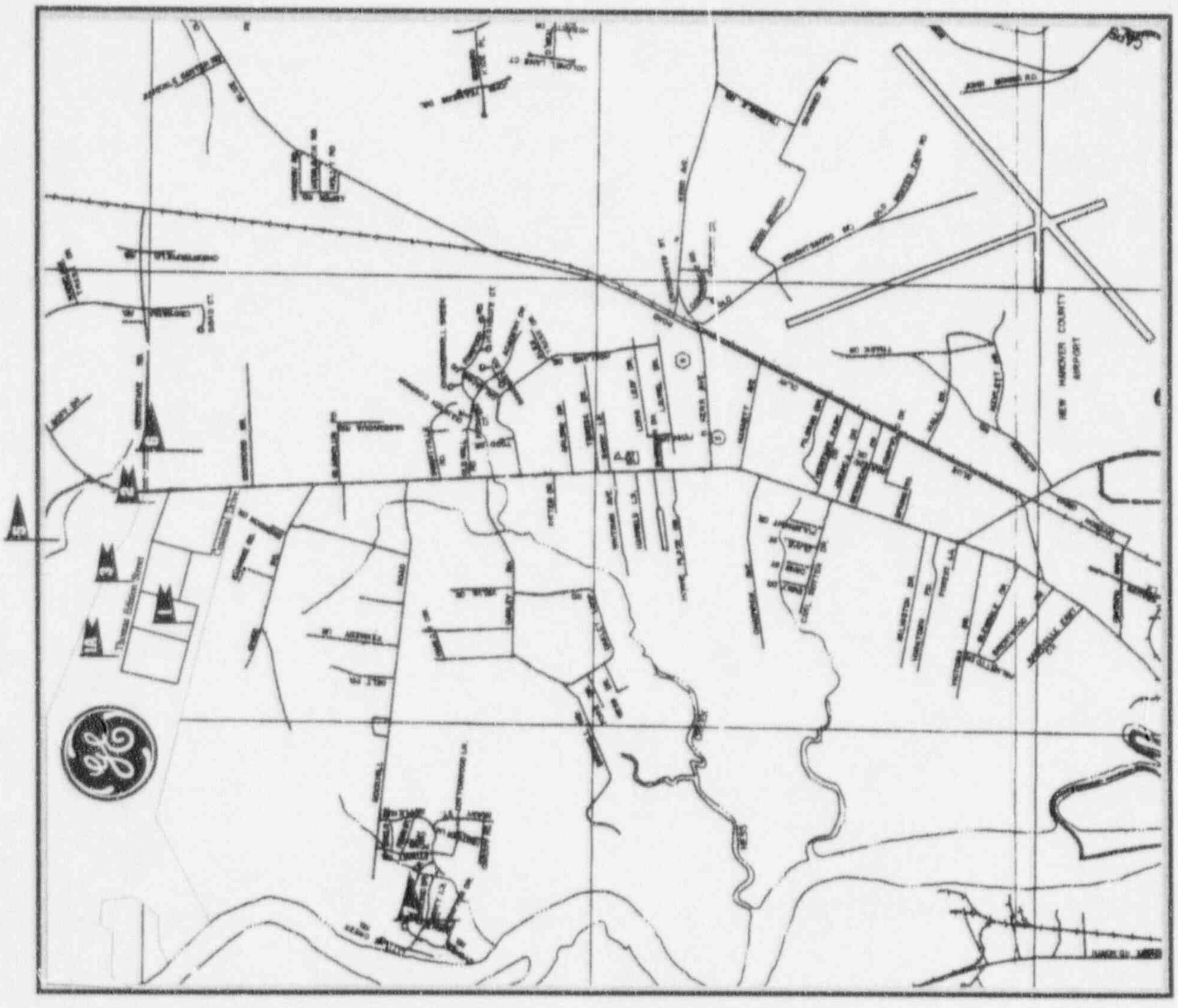
Locations of Environmental Soil Sampling Sites

On Site 1 - 3

Off Site 5 - 7

PLANT NORTH

N



0 1500 3000 FEET

Figure 10.3a

Locations of Ditch and Vegetation Sampling Sites

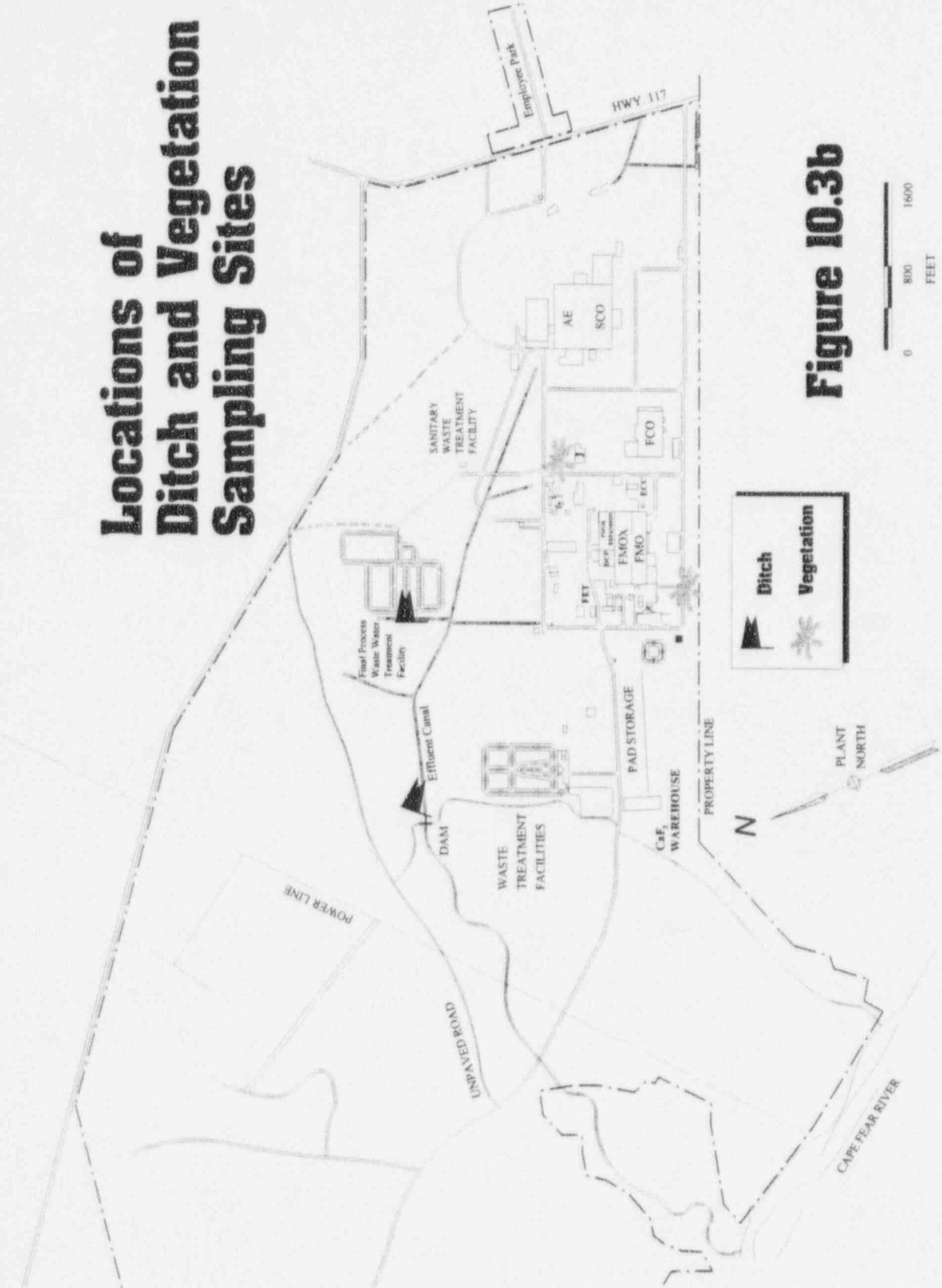


Figure 10.3b

Subject : Sample Collection from Site Dam

Location of Site Dam Sampling

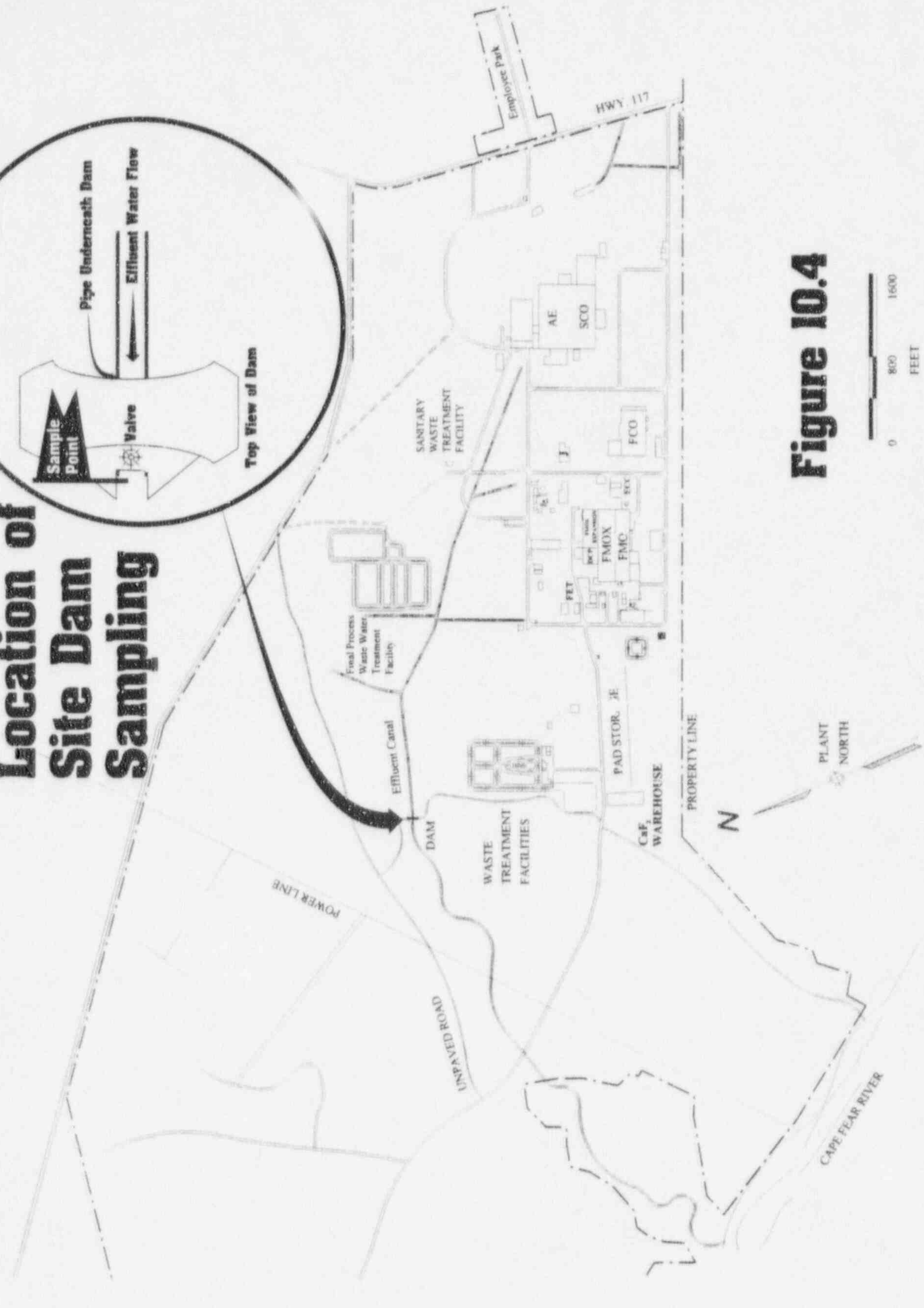


Figure 10.4

Subject : Sample Collection from Monitoring Wells

Locations of MW Series Wells around Waste Treatment Facility

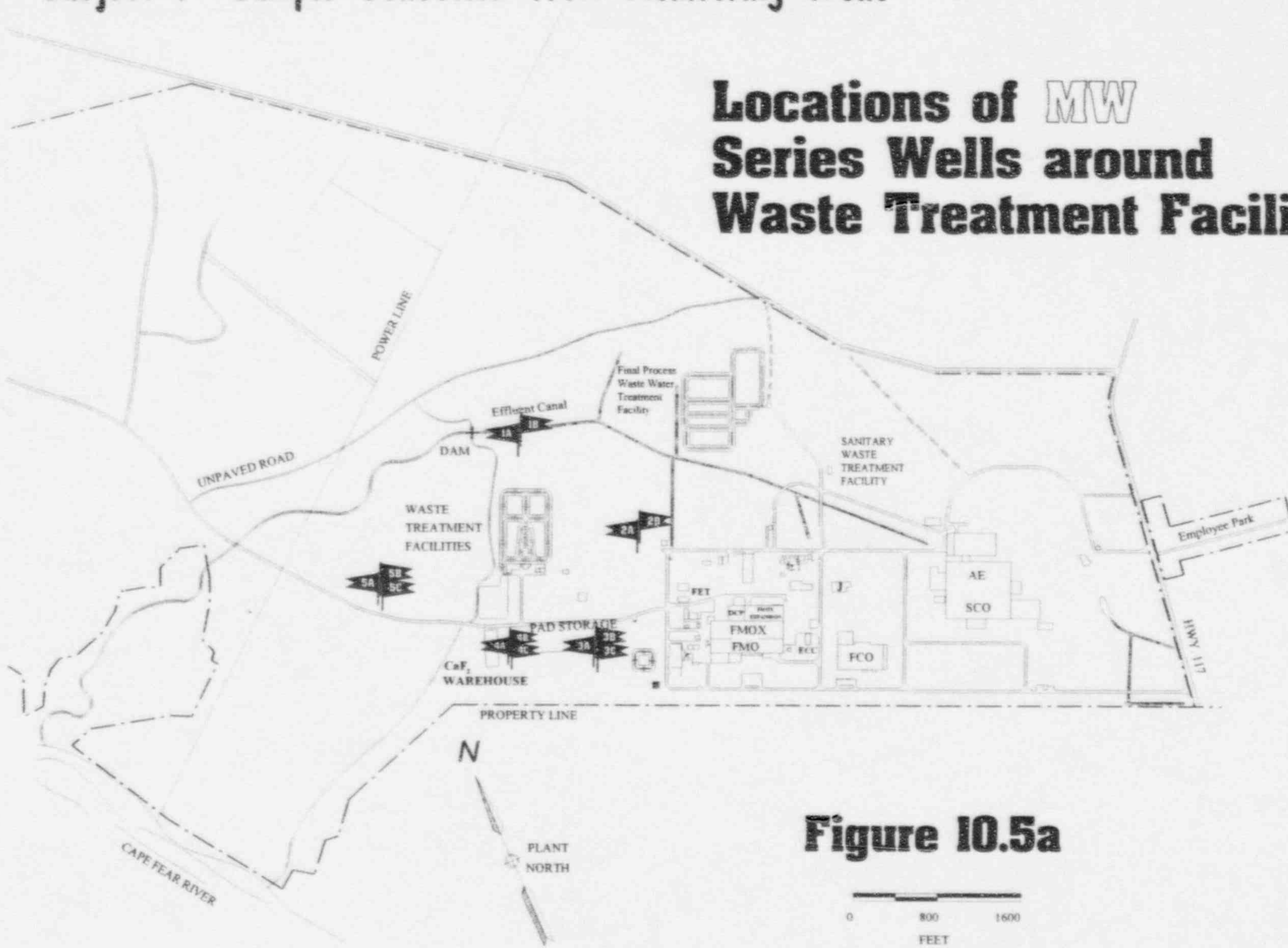


Figure 10.5a

Subject : Sample Collection from Monitoring Wells

Locations of PL Series Wells around Final Process Waste Water Treatment Facility

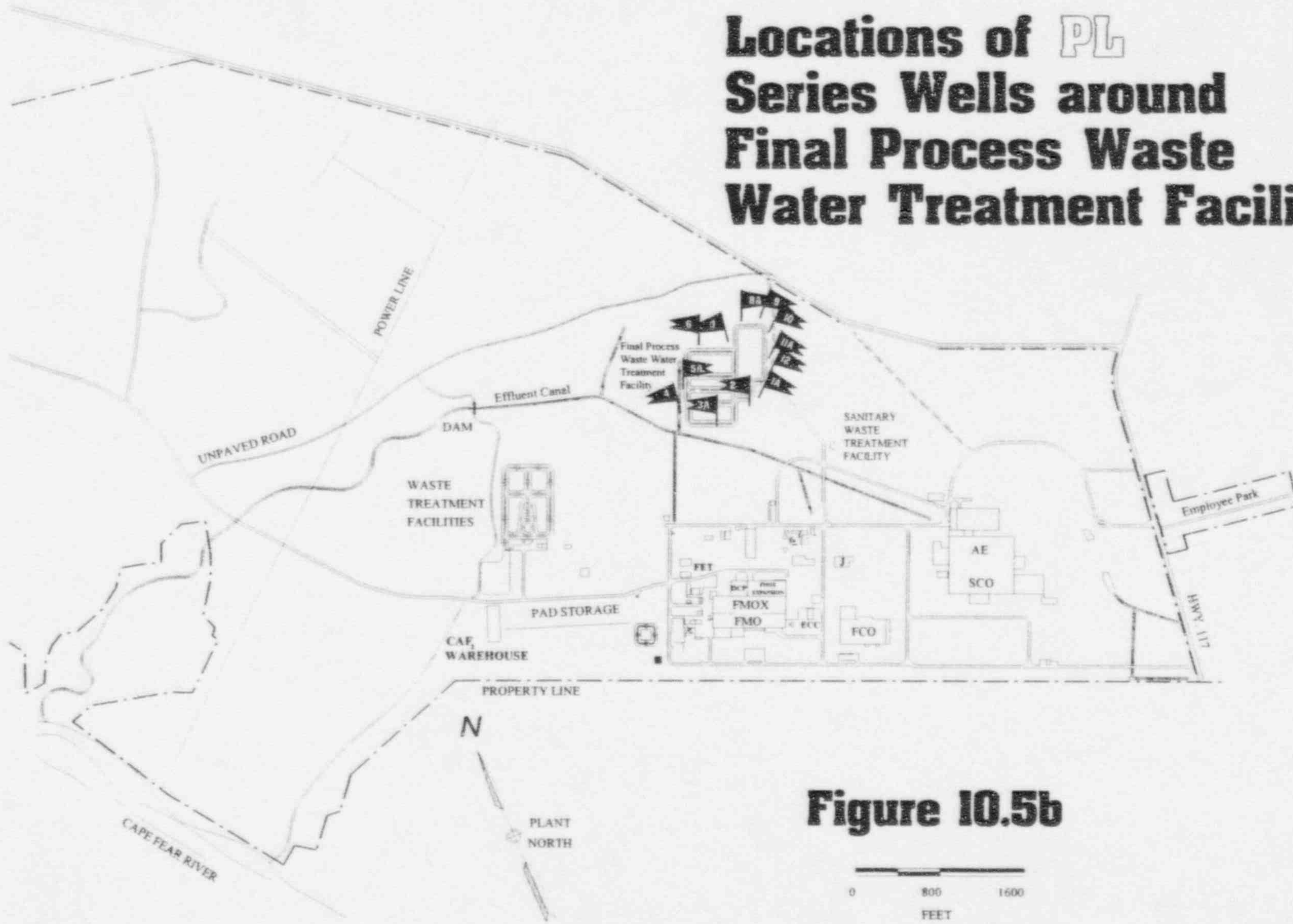


Figure 10.5b

Locations of WT Series Wells around Waste Treatment Facility

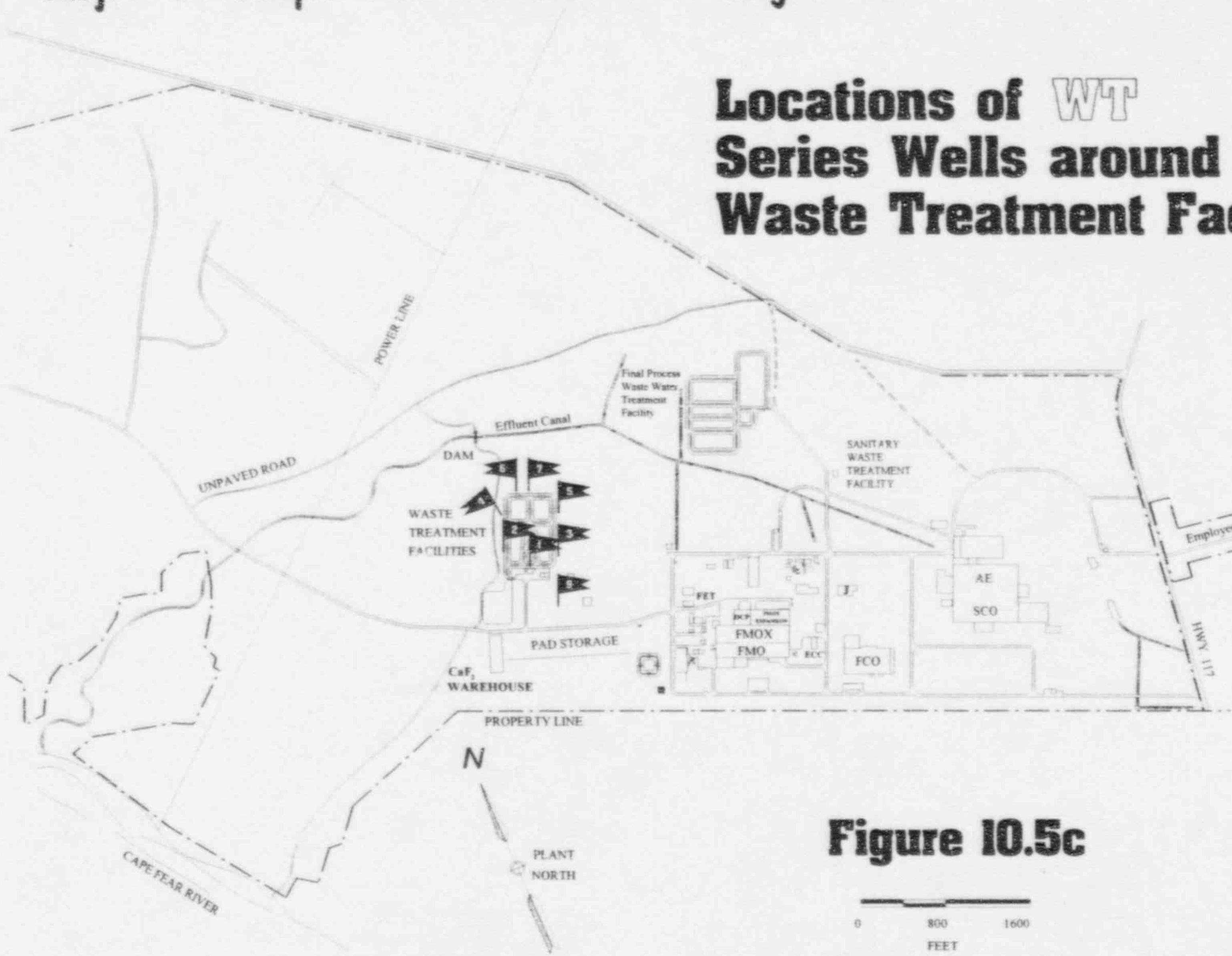


Figure 10.5c

Subject : Sample Collection from Monitoring Wells

Locations of F Series Wells around FMO / FMOX and FCO Areas

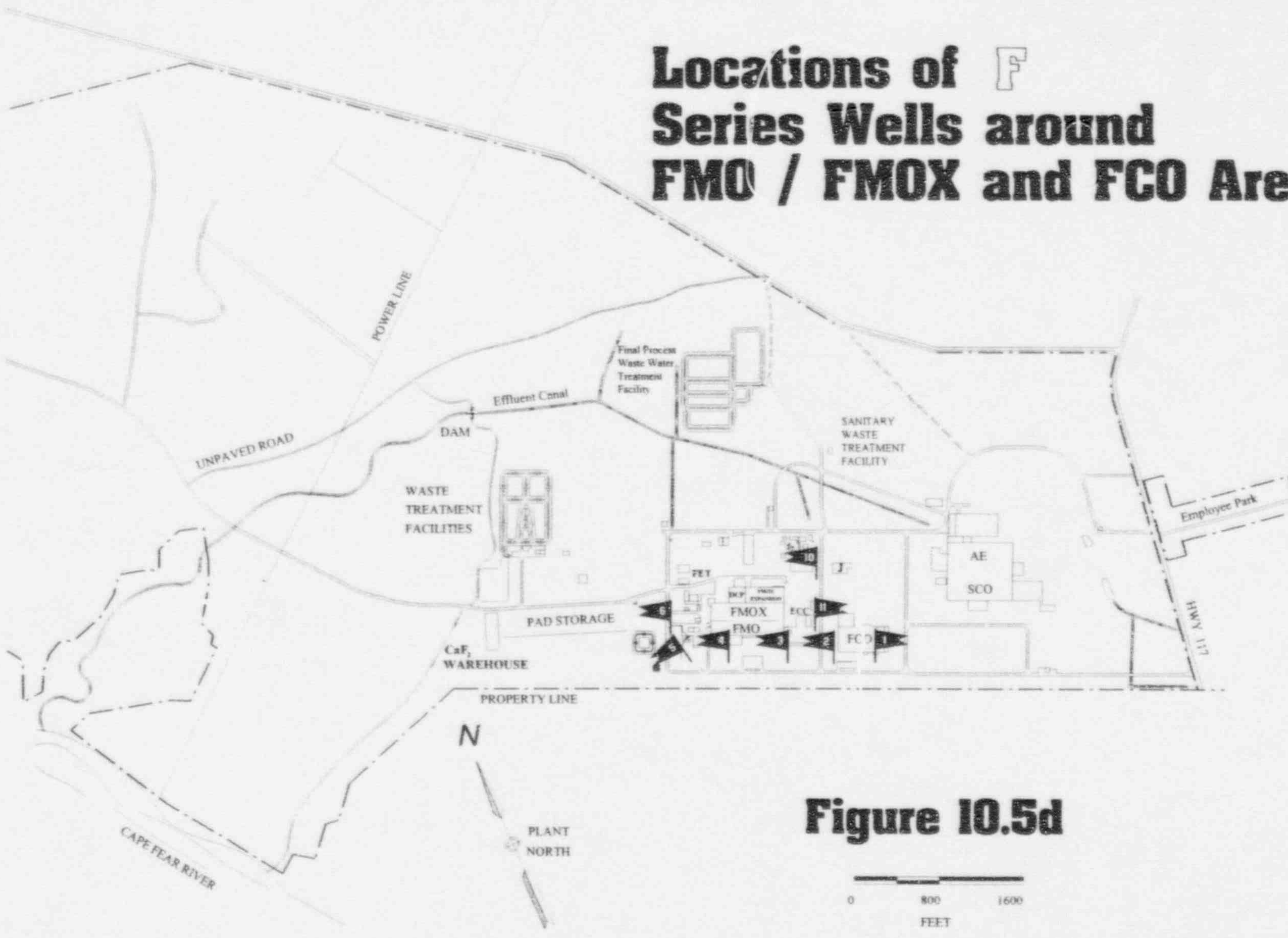


Figure 10.5d

Subject : Sample Collection from Monitoring Wells

Locations of CW Series Wells around Waste Treatment Facilities

Compliance Wells Approximately 500 feet from Facilities

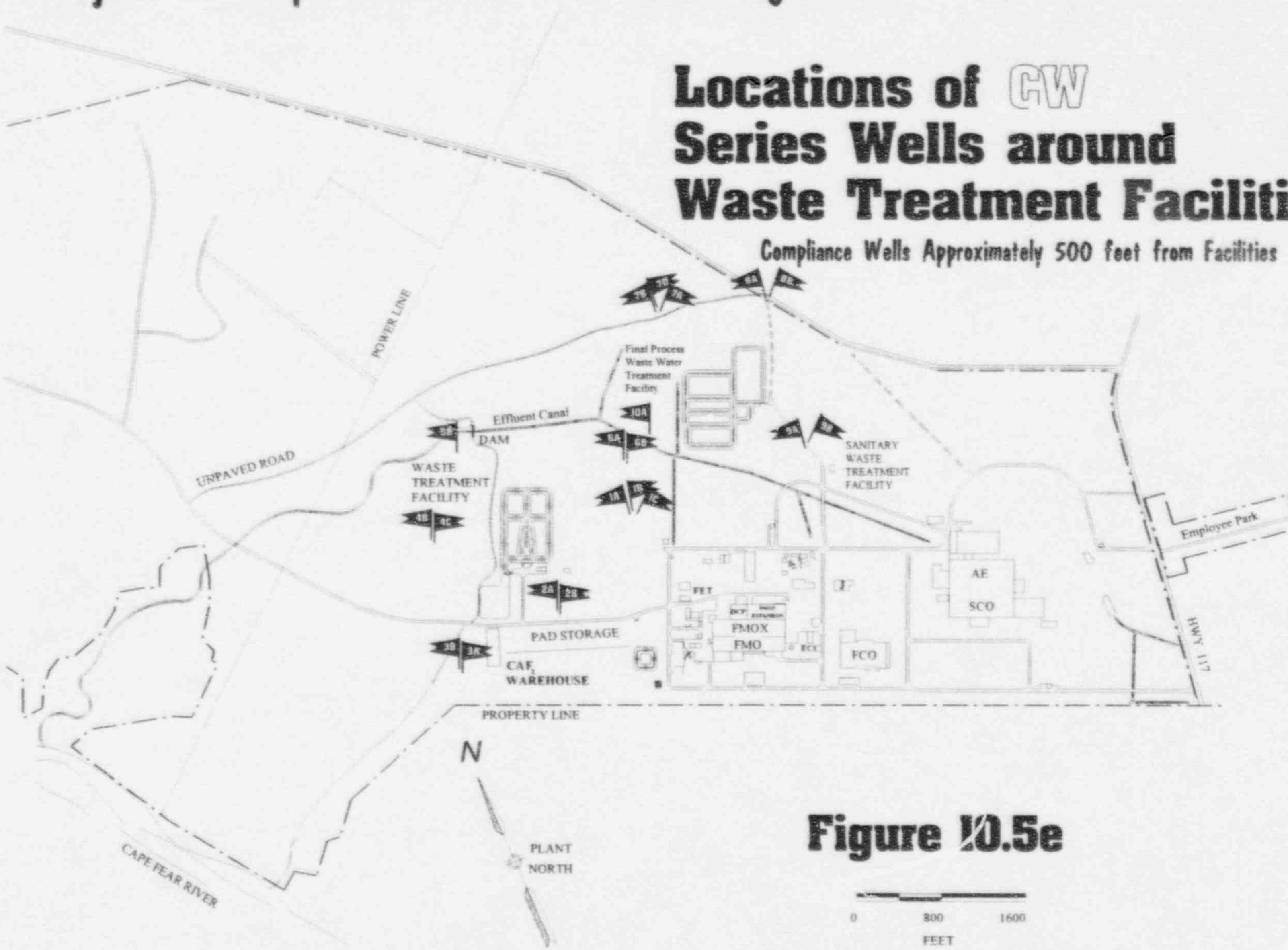


Figure 10.5e

Locations of FX Series Wells along Western Perimeter of FMO / FMOX Buildings

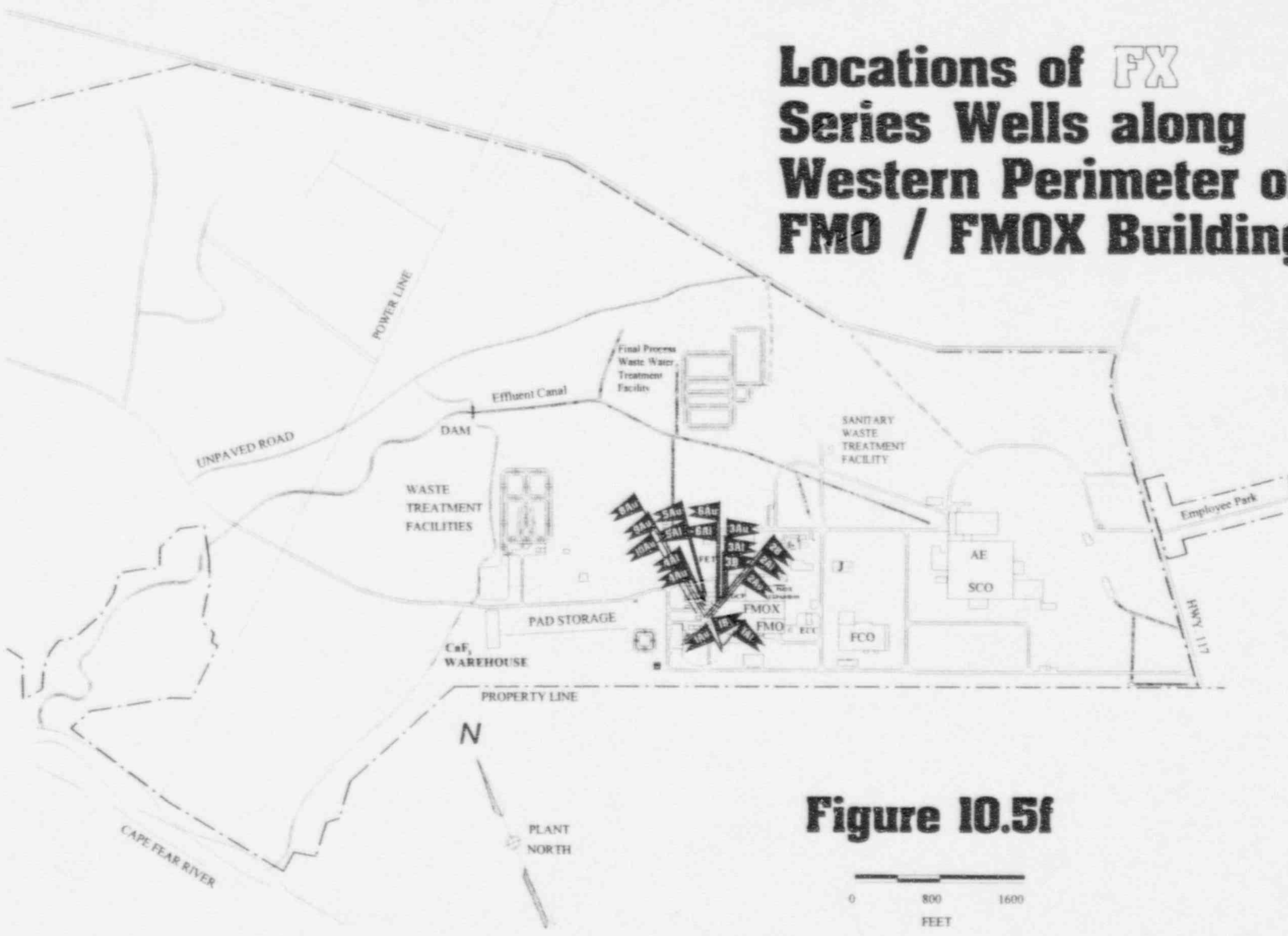


Figure 10.5f

Subject : Sample Collection from Monitoring Wells

Locations of WW Series Plant Water Supply Wells

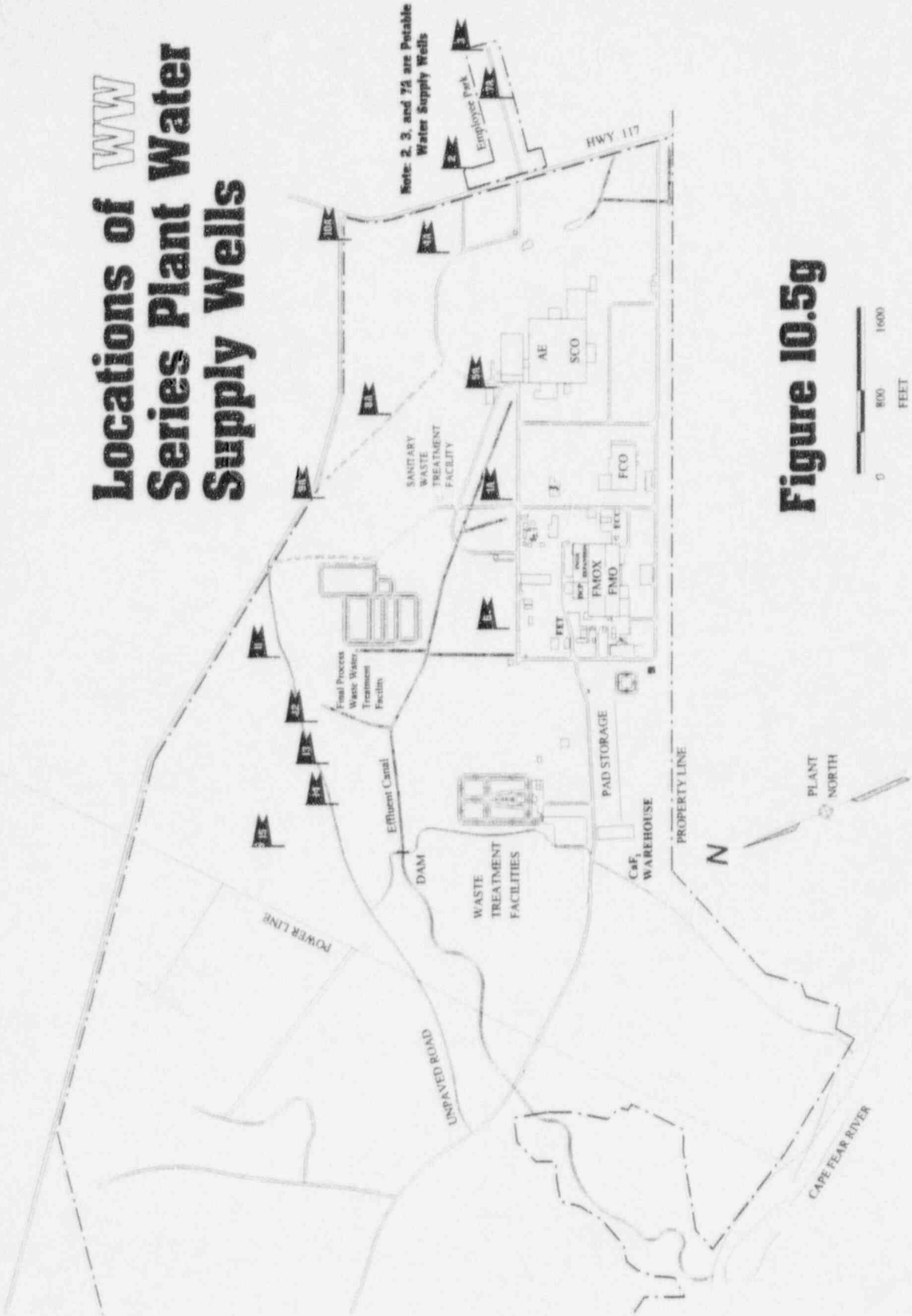


Figure 10.5g

CHAPTER 11.0

DECOMMISSIONING

At the end of plant life, GE-Wilmington shall decommission the facilities and site in accordance with the Decommissioning and Closure Plan dated December 18, 1996, as revised in accordance with regulatory provisions. The Decommissioning and Closure Plan was originally approved by the NRC on December 11, 1981.

GE Nuclear Energy's self-guarantee submittals dated March 14 and 24, 1994, were approved by the NRC on April 29, 1994.

The financial commitment submitted by GE Nuclear Energy with the Decommissioning and Closure Plan identified in the December 18, 1996 version continues to assure that funds will be available for decommissioning and will be revised in accordance with regulatory provisions.

APPENDIX

REFERENCE LETTERS

DATE	TITLE
12/20/89	TP Winslow to LC Rouse - Request to Transfer Quantities of CaF_2 for Beneficial Reuse
09/24/92	TP Winslow to JWN Hickey - Request to Transfer Test Quantities of CaF_2 for Evaluation
02/27/96	RJ Reda to RC Pierson - Request to Transfer Test Quantities of HF
05/27/97	RJ Reda to MF Weber - Response to NRC Questions Relating to Criticality Safety, ISA and Administrative Changes
05/29/97	RJ Reda to MF Weber - Request to Delay the Biennial Emergency Exercise

GE Nuclear Energy

December 20, 1989

Mr. L. C. Rouse, Chief
Fuel Cycle Safety Branch
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Rouse:

Subject: LICENSE AMENDMENT REQUEST (REVISION #25) - REVISED

References: 1) NRC License SNM-1097, Docket #70-1113
2) License Amendment Request dtd October 27, 1989

GE Nuclear Fuel and Components Manufacturing hereby resubmits License Amendment Request (Revision # 25), originally dated 10/27/89, to include additional information that will clarify our submittal per your request. This request is to authorize the transfer of quantities of industrial waste treatment products (primarily calcium fluoride) for beneficial reuse without continuing NRC controls. This material contains low level amounts of uranium less than 30 picocuries per gram (pCi/gm) on a dry weight basis and will be used as a mixer with steel flux forming materials in the production of steel.

Attachment 1 of this letter describes the requested activities, the decision criteria, authorized recipient, and proposed controls.

Attachment 2 is a description of the requested revision and Attachment 3 is the revised pages to our SNM license.

Pursuant to 10 CFR 170.31, a check for $150 was enclosed with the original submittal.

GE personnel would be pleased to discuss this matter further as you may deem necessary.

Very truly yours,

GE NUCLEAR ENERGY

GAMallett for
T. Preston Winslow, Manager
Licensing & Nuclear Materials Management

Attachments

cc: Region II - S. D. Ebneter

ATTACHMENT 1

PURPOSE

GE is reviewing methods for recycling materials that are now disposed of as waste. GE proposes to transfer one of these industrial waste products for beneficial reuse, having less than 30 pCi/gram on a dry weight basis, to the identified company for the manufacture of briquettes to be used in the production of steel.

REQUEST

GE hereby requests an amendment to License SNM-1097 to authorize free release to Cametco, Inc., without continuing NRC controls of calcium fluoride waste treatment products in which the concentration of uranium is less than 30 pCi/gram on a dry weight basis. We are requesting this amendment to authorize distribution of calcium fluoride to the briquette manufacturer to be mixed with other steel flux forming materials, briquetted, and further distributed to steel manufacturers in the production of steel. Chemical separation of the uranium from the waste would not be permitted.

BACKGROUND

The chemical conversion of uranium hexafluoride (UF_6) to uranium dioxide (UO_2) results in an aqueous waste containing ammonium fluoride (NH_4F) and a very low concentration of soluble uranium. This aqueous waste is treated with lime ($Ca(OH)_2$) to precipitate the fluoride ion and capture the remaining small amounts of uranium. This results in an insoluble calcium fluoride (CaF_2) precipitate.

The CaF_2 is filtered from the waste stream and the filtered liquid is pumped to the lagoons where it is discharged after processing.

The dewatered CaF_2 solids contain less than 30 pCi of uranium per gram on a dry weight basis. Currently, these solids are transported off-site to a waste burial facility for disposal as described in SNM-1097, Section 1.8.5.2.

INTRODUCTION

The material to be shipped will be limited to calcium fluoride waste treatment products that have been dewatered and dried.

Prior to shipment, the material will be analyzed to assure that the uranium concentration limit is not exceeded. Material use by the recipient will be limited to those that preclude the chemical separation of uranium from the matrix.

The method of transportation will be a covered transport trailer.

STEEL INDUSTRY APPLICATION

Calcium fluoride is used as a fluxing agent in the steel making process. The calcium fluoride from naturally occurring ore (fluorspar) is made into briquettes by several manufacturers. The fluxed impurities in the steel making process end up as a slag for subsequent disposal. Fluorspar contains natural uranium ranging from 2 to 10 pCi/g.

The uranium concentration in calcium fluoride generated at the GE-Wilmington facility is effectively the same as the natural calcium fluoride (fluorspar) used as a fluxing agent in the manufacture of steel. GE-Wilmington's CaF_2 contains uranium in the 2-30 pCi/gram range.

DECISION CRITERIA

The environmental impact of calcium fluoride generated at the GE-Wilmington facility will not differ significantly from that generated by natural sources.

RADIOLOGICAL EVALUATION OF THE FREE RELEASE CALCIUM FLUORIDE TO STEEL-FLUXING BRIQUETTES MANUFACTURERS

The largest potential for radiation exposure due to the re-use of CaF_2 by Cametco, Inc. is in the manufaturing of the briquettes themselves. After the briquettes are manufactured the trace

quantities of uranium are encapsulated in the briquettes. The radiation doses from the manufacture and use of steel contaminated with trace amounts of uranium has been extensively evaluated in NUREG 0518 - Draft Environments Statement Concerning Proposed Rulemaking Exemption from Licensing Requirements for Smelted Alloys Containing Residual Technetium - 99 and Low-Enriched Uranium (USNRC, October 1980). This report indicates that there are no significant radiological problems for individual workers or members of the general public in the use of residually contaminated steel.

An analysis was made of the potential radiological impact of the use of CaF_2 on the workers at the Cametco facility. The data for the analysis was obtained by touring the Cametco facility and from discussions with Cametco management. The facts and assumptions for the analysis are as follows:

1. Cametco produces 13,000 tons of briquetted product per year.

2. GE shipments of CaF_2 with less than 30 pCi/g of up to 5% enriched uranium would be no more than 1000 tons per year.

3. It is assumed that GE CaF_2 would be present in the airborne dust in proportion to its mass fraction of the total Cametco briquette production.

4. For the purpose of estimating potential exposure levels, it is assumed that the average worker inhales 24 grams of dust per year. Dust levels at the Cametco facilities vary significantly through out the course of a normal work day. Workers wear a North Model 7170 Dust Respirator at their discretion. The 24 gram quantity is derived by assuming a worker inhales dust at a level just below the 10 mg/m³ silica dust limit recommended by the American Conference of Governmental Industrial Hygienists in the Threshold Limit Values and Biological Indicies for 1988-89. Since there is no specific limit for CaF_2 or other briquette raw materials, the silica dust limit is commonly used as a surrogate limit for non-toxic respirable dust. An average breathing rate of 1.2 m³/hr. for 2000 hours per year is assumed.

5. A dose conversion factor of 62.5 rems/µCi is derived by equating the 10 CFR 20 appendix B limit for insoluble uranium-234 of 1×10^{-10} µCi/ml times 1.2 m^3/hr times 2,000 hours/year and equating the product with 15 rems dose to the lungs (the critical organ).

Combining these facts and assumptions the potential exposure to an individual worker can be calculated as follows:

$$24 \text{ g dust} \times \frac{1{,}000 \text{ t GE } CaF_2}{13{,}000 \text{ t of briquetted product}}$$

$$\times \frac{30 \text{ p Ci}}{\text{g GE } CaF_2} \times \frac{\mu Ci}{10^6 \text{ pCi}} \times \frac{62.5 \text{ rems}}{\mu Ci} = 3.5 \text{ mrems}$$

A dose of 3.5 mrems per year is significantly less than the 40 CFR 190 limit of 25 mrems per year, and using a conventional risk factor of 2×10^{-4} adverse health effects per rem, corresponds to a risk level most would consider insignificant.

AUTHORIZED RECIPIENT

The following company is requested to be authorized to receive industrial waste treatment products as described in this request:

Cametco Inc.
600 Duquesne Blvd.
Pittsburgh, PA 15211

CRITERIA FOR SHIPMENT

1. Materials shall be limited to industrial waste treatment products (primarily calcium fluoride) and other homogeneous mixtures in which the uranium constituents are less than 30 picocuries per gram (dry basis).

2. The recipient shall be appraised of the typical chemical content of the materials, including uranium, and the limitations of its use and distribution.

3. Material use and distribution shall be limited to those that preclude the chemical separation of uranium from the matrix and entry of the product into the human food chain.

4. Materials shall be appropriately sampled and analyzed to assure that the shipments of CaF_2 contain less than 30 pCi/gram activity.

5. The following table summarizes actual data of samples from calcium fluoride shipped for burial.

Average Frequencies of pCi/g in CaF_2
(Based on Data from 1989)

Range pCi/g	Number of Data Points Wet Basis	Number of Data Points Dry Basis*	Number of Data Points Dry Basis (Worst Case)**
0-5	85	74	67
6-10	1	9	11
11-15	4	3	5
16-30	0	3	3
> 30	0	1***	4***
	90	90	90

*This material contains an average of 47.03% solids with a one sigma of 4.66.

**Conversion to dry basis at an average of 47.03% solids minus two times sigma.

***These cases would not be shipped to the vendor.

CONCLUSION

Disposal of low activity concentrations of the industrial waste treatment products (primarily CaF_2) by this alternative means will not pose an undue risk to the public health and safety.

This beneficial use of this material will have no adverse effects on members of the public or the environment. In addition, there will be a positive environmental impact from this approach because the calcium fluoride will be beneficially used instead of buried in a landfill.

GE Nuclear Energy

September 24, 1992

Director
Office of Nuclear Materials Safety & Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. John W. N. Hickey, Chief
Fuel Cycle Safety Branch
OWFN, Room 6H6, Mail 6H3

Dear Sir:

Subject: License Amendment Request (Revision #36)

Reference: NRC License SNM-1097, Docket # 70-1113

GE Nuclear Fuel & Components Manufacturing hereby submits a license amendment request to authorize the transfer of test quantities of calcium fluoride containing low level amounts of uranium to vendors for evaluation of potential usage.

Attachment 1 of this letter describes the material to be shipped, recipients, planned testing activities, maximum quantities, and benefits.

Attachment 2 is a description of the requested revision and Attachment 3 is the revised pages to our SNM license.

Please contact me on (919) 675-5461 if you require additional information or would like to discuss this matter further.

Very truly yours,

GE NUCLEAR ENERGY

GR Mallett for

T. Preston Winslow, Manager
Licensing & Nuclear Materials Management

/sbm

Attachment

cc: TPW-92-133

ATTACHMENT 1

Purpose

The following information is provided in support of GE's Nuclear Fuel & Components Manufacturing (NF&CM) request for NRC authorization to ship quantities of dried calcium fluoride to potential buyers for testing without continuing NRC controls. The activity concentration is sufficiently low as to justify its beneficial use rather than transport to a disposal facility.

Background

The chemical conversion of uranium hexafluoride (UF_6) to uranium dioxide (UO_2) results in an aqueous waste containing ammonium fluoride (NH_4F) and a very low concentration of soluble uranium. This aqueous waste is treated with lime ($Ca(OH)_2$) to precipitate the fluoride ion and capture the remaining small amounts of uranium. This results in an insoluble calcium fluoride (CaF_2) precipitate.

The CaF_2 is filtered from the waste stream and the filtered liquid is pumped to the lagoons where it is discharged after processing.

The remaining CaF_2 solids are dried. These solids contain less than 30 pCi of uranium per gram on a dry weight basis. Currently, these solids are approved for transport to a waste burial facility for off-site disposal as described in SNM-1097, Section 1.8.5. or for beneficial reuse in briquette manufacturing for the steel industry, as described in SNM-1097, Section 1.8.12.

Description of Material

The test material to be shipped will be limited to dried calcium fluoride containing not greater than 30 pCi per gram of uranium on a dry weight basis. Each test quantity will not exceed 1 gram U^{235} total per recipient. Prior to shipment, the material will be analyzed to assure that this concentration limit is not exceeded. Feasibility tests by recipients will be limited to those that preclude the chemical separation of uranium from the matrix.

Recipients of the Material

The following potential customers are the designated recipients of the described material for the initial feasibility tests:

Customer/Manufacturer

1. Armstrong Glass Co., Inc.
   1320 Ellsworth Industrial Dr., NW
   Atlanta, GA 30318

2. The Paul Wissmach Glass Co., Inc.
   420 Stephen Street
   Paden City, WV 26159

3. Kokomo Opalescent Glass Co., Inc.
   1310 South Market St.
   Kokomo, IN 46902

Description of Planned Testing

The CaF_2 shall be mixed with the recipient's glass batch material in the glass manufacturing process.

Glass from these processes is not intended to be used in products designed to contain food or liquid for human consumption. In addition, test activities and end use of the material will be limited to those that do not allow chemical separation of the uranium or entry of the product into the food chain.

Maximum Quantity per Vendor

The maximum amount of CaF_2 required to be shipped to each potential customer shall not exceed 2,000 pounds.

Request

NF&CM requests permission to ship test quantities of calcium fluoride as described herein without continuing controls to the potential customers. These recipients will evaluate and test the material for usefulness in their process and perform feasibility studies to ultimately allow beneficial use of this material.

The potential radiological exposures from using CaF_2 pursuant to this request are estimated to be less than 10 mRem per year.

Benefit

This proposed action will lead to an authorization for the recycle of excess CaF_2 instead of having to dispose of it in a landfill. In addition, it will decrease the usage of natural CaF_2, a non-renewable material source, currently being used in glass manufacturing.

GE Nuclear Energy

February 27, 1996

Mr. R. C. Pierson
Licensing Branch, NMSS
U.S. Nuclear Regulatory Commission
Mail Stop T 8-D-14
Washington, D.C. 20555-0001

Subject: License Amendment Request - HF Samples, Revision 53

References: Docket 70-1113, NRC License SNM-1097

Dear Mr. Pierson:

GE's Nuclear Energy Production (NEP) facility in Wilmington, NC, hereby submits a license amendment request to authorize the transfer of test quantities of hydrofluoric (HF) acid potentially containing trace amounts of uranium to potential buyers/customers and/or laboratories for their evaluation of the material.

Attachment 1 of this letter describes the material to be shipped, recipients and maximum quantities.

Attachment 2 is a description of the requested revision and Attachment 3 is the revised pages to our SNM license.

Please contact Rick Foleck on (910) 675-6299 if you require additional information or would like to discuss this matter further.

Sincerely,

GE NUCLEAR ENERGY

Ralph J. Reda

R. J. Reda
Manager
Fuels and Facility Licensing

/zb
attachments

cc: RJR-96-023

ATTACHMENT 1

Purpose:

The following information is provided in support of GE's Nuclear Energy Production (NEP) facility in Wilmington, NC request for NRC authorization to ship quantities of hydrofluoric (HF) acid to potential buyers/customers and/or laboratories who are typically involved in analyzing this type of material for testing without continuing NRC controls. The activity concentration is sufficiently low as to justify its beneficial use.

Background:

GE is facilitizing to replace the current ADU UF6 to UO2 conversion process technology with a more direct and efficient UF6 to UO2 conversion process. The process technology is being purchased from Franco-Belge de Fabrication de Combustibles (FBFC). They have over 15 years successful operation of the technology. A coproduct of this process technology is the production of nominal 50% HF at the rate of approximately 0.8 kg of 50% HF per kg UO2. The HF is of commercial grade and contains only traces of uranium averaging slightly above 0.1 PPM. The purity and concentration of this coproduct make it attractive for a number of industrial uses.

Description of Material:

A sample batch of approximately 30 liters of nominal 50% HF was obtained from FBFC. This HF was generated by the same UF6 to UO2 conversion process that GE has purchased and is installing at the Wilmington, NC location. The uranium concentration was checked and found to be equal to or less than 0.1 PPM (0.1mg/liter). The enrichment is less than 5%. GE is developing a customer base for sale of the HF produced by our new conversion operation and the samples are needed as a part of product evaluation and testing by the potential users.

The sample material has been shipped from France to the US as non-nuclear material since under French and international guidelines it is non-nuclear. It will be subdivided here at the Wilmington Plant and transferred to the companies in accord with US chemical regulations, DOT regulations and the conditions of our NRC license. In the future, the HF samples may originate from either FBFC or from our facility.

The uranium concentration must be equal to or less than 3 PPM based on weight, and the enrichment will be equal to 6% or less. These values are based on process history of the French plant, our process history, or the samples will be checked to assure these limits are met.

Recipients of the Maten. l:

Recipients of the HF samples shall be limited to potential buyers/customers and/or laboratories who are typically involved in analyzing this type of material for the purpose of conducting purity verification and feasibility tests.

Description of Planned Tests:

The HF will be analyzed to evaluate the nature and content of the impurities present with regard to the particular sensitivities of the processes for which intended use applies.

The material may also be evaluated under actual or simulated process conditions to verify that it performs within the process requirements.

None of the tests result in the material being consumed by humans nor used in products used on or in the body or in the food chain.

Maximum Quantity per Customer:

Individual sample transfers will contain not more than 5 liters of HF (typically 2 liters). These shipments will be made in accordance with US Department of Transportation regulations for this type and form of HF.

The receiver(s) will be notified that the HF potentially contains traces of uranium enriched to no more than 6%. They will be advised that this is not a nuclear hazard, but will be advised that the material should be handled carefully and in such a manner so as not to be consumed by humans nor used in products used on or in the body or in the food chain.

Request:

GE-NEP Wilmington, NC facility requests permission to ship the test quantities of HF as described herein, without continuing NRC controls. The recipient(s) will test and evaluate the material to determine the acceptability and compatibility in processes as a part of a program to market this coproduct.

The potential radiological exposures from using this material for these tests and evaluations as requested are estimated to be far less than 10 mRem per year.

GE Nuclear Energy

910 675-5000

May 27, 1997

Mr. M. F. Weber, Licensing Branch, NMSS
U.S. Nuclear Regulatory Commission
Mail Stop T 8-D-14
Washington, DC 20555-0001

Subject: License Renewal - Response to Request for Additional Information (TAC No. L10079)

Reference:
(1) NRC License SNM-1097, Docket 70-1113
(2) License Renewal Application, 4/5/96
(3) Submittal, RJ Reda to ED Flack, 5/6/96
(4) Submittal, RJ Reda to RC Pierson, 5/14/96
(5) Letter, RC Pierson to RJ Reda, 7/18/96
(6) Submittal, RJ Reda to RC Pierson, 8/30/96
(7) Submittal, RJ Reda to ED Flack, 9/26/96
(8) Letter, MA Lamastra to RJ Reda, 10/2/96
(9) Submittal, RJ Reda to MA Lamastra, 11/22/96
(10) Application, RJ Reda to MF Weber, 12/16/96
(11) Letter, MA Lamastra to RJ Reda, 12/17/96
(12) Submittal, RJ Reda to MF Weber, 2/5/97
(13) Letter, MA Lamastra to RJ Reda, 2/10/97
(14) Submittal, RJ Reda to MF Weber, 2/19/97
(15) Submittal, RJ Reda to MF Weber, 2/25/97
(16) Letter, MA Lamastra to RJ Reda, 3/5/97
(17) Submittal, RJ Reda to MF Weber, 3/27/97
(18) Submittal, RJ Reda to MF Weber, 3/28/97
(19) Letter, MA Lamastra to RJ Reda, 5/6/97
(20) Letter, MA Lamastra to RJ Reda, 5/14/97
(21) Letter, RJ Reda, to MA Lamastra, 5/21/97

Dear Mr. Weber:

GE's Nuclear Energy Production (NEP) facility in Wilmington, N.C., hereby transmits the enclosed information in response to the above referenced letter dated 5/6/97. The response includes information discussed at the management meeting on 5/20/97 at NRC Headquarters, including a subsequent telephone discussion on 5/22/97 regarding concentration control. This information is being provided in support of our license renewal request.

Attachment 2 contains (1) a description of the changes made to the license renewal by page and section, and (2) the page changes to our license renewal application for pages contained in the Table of Contents, Chapter 1.0, Chapter 2.0, Chapter 3.0, Chapter 4.0, Chapter 6.0 and Chapter 7. Each chapter is provided in its entirety for easy replacement. Each page within the chapter that contains a change is indicated with a horizontal line (|) in the right hand column to show where a change has taken place. All replacement pages contain the date of this submittal (5/27/97) and are shown as revision zero.

Six copies of this submittal are being provided for your use.

Please contact Charlie Vaughan on (910) 675-5656 or me on (910) 675-5889, if you have any questions or would like to discuss this matter further.

Sincerely,

GE NUCLEAR ENERGY

Ralph J. Reda

Ralph J. Reda, Manager
Fuels & Facility Licensing

/zb
Attachments

cc: RJR-97-065
L. A. Reyes, Region II Administrator
G. L. Troup, NRC-Atlanta
M. Fry, State of NC

ATTACHMENT 1

Response to Request for Additional Information Contained in
Letter from MA Lamastra to RJ Reda Dated May 6, 1997

Response to NRC Request for Specific Comments and Additional Information Required for GE-NEP's License Renewal Application

Please provide the following information:

1. *In an NRC letter dated December 17, 1996, the Fuel Cycle Licensing Branch (FCLB) commented that the qualifications of most positions have decreased compared to the existing license. GE was requested to demonstrate or explain why such a decrease in the overall experience of the staff would not adversely affect the safety of plant operations. GE's response dated February 5, 1997, stated in part that GE management was responsible and accountable for the safe operation of the plant, GE had in place a management system for identifying job function and selecting qualified individuals, and that the minimum requirements are generally consistent with other like facilities. Accordingly, GE made no changes in Section 2.2.1 of the application.*

   FCLB agrees that GE is ultimately responsible for the safe operation of the plant. However, 10 CFR 70.23, "Requirements for the Approval of Applications" states, in part, that an application for a license will be approved, if the Commission determines that the applicant is qualified by reason of training and experience. GE's proposed minimum qualification is basically a B.S. degree and two years experience or a high school diploma with 5 years experience for both staff and supervisor positions. FCLB has also reviewed the current minimum levels of training and experience at other fuel fabrication facilities and determined that GE's minimum requirements would be the least and significantly less than those of the average facility. Accordingly, we see no basis for the reduction in qualifications for staff and supervisory personal and request that for each safety-significant position (radiation protection, criticality safety, chemical safety, fire protection, environmental safety etc.) that the minimum qualifications be upgraded to at least the requirements of the current license or a position by position justification.

   In accord with the RAI, GE has modified Sections 2.2.1.2, 2.2.1.3, 2.2.1.4, 2.2.1.8 and 2.2.1.9 to be consistent with the comments and discussions regarding minimum qualifications.

2. *The RAI dated March 5, 1997, questioned the definition of "practices" as used in Chapter 3.0 of the license application. As described in the license application, these practices should be maintained, controlled and/or approved in the same manner as procedures.*

   GE provided an acceptable response. However, in order to convey the information provided in the response of March 27, 1997, GE should add a statement to the license application that approved policies, practices and procedures will be followed. An acceptable statement would include wording similar to the following:

   Licensed material processing or activities will be conducted in accordance with properly issued and approved practices and procedures (P&P), plant practices, or operating procedures.

   GE has modified Section 3.9 to include the requested clarification in wording.

3. *Section 4 of your application, should be modified to include a schedule for completing the ISA for the balance of plant and a schedule for submitting a revised ISA summary for the DCP. The schedule should include milestones for a final completion date and intermediate dates for those systems most important for safety. In addition, a clear commitment to complete the proposed ISA summary for each system should be provided. Further, and most importantly, GE should commit to maintaining available and reliable systems equipment and controls that are most important to safety based on the ISA results. Enclosure 2 and Comment 4 identifies the types of information that a summary should include.*

   At the management meeting at NRC Headquarters on May 20, 1997, RAI's 3 and 4 were discussed in relation to the outline for preparing ISA summaries for the NRC and the content of those summaries. GE also presented a schematic overview of the role of the ISA in the proposed safety program as well as an overview of the type and flow of information generated by the ISA process.

   As a result of these discussions it was mutually agreed that GE and the NRC both have work to do to fine tune the outline with regard to what constitutes an acceptable ISA summary. This work will start after the license is renewed with working sessions to review the records generated by the ISA process, information needs by the NRC and a critique of the current ISA summary. In relation to the schedule to complete ISAs for the balance of the plant it was mutually agreed that this summary definition work had to be completed

on a schedule that will support the schedule GE is committing to for that work.

GE's schedule is keyed to the issuance of a renewed license, clarification of the summary detail required and systematic and timely feedback of the NRC's critique of ISA summaries as they are submitted to the NRC.

GE prefers that the formal commitments for this work be called out in a letter or a license condition as opposed to modifying Chapter 4 - the reason being that this is a onetime effort and it appears that it would be better identified that way.

Based on the above detail the schedule that GE proposes is as discussed on May 20, 1997, and is as follows:

• **NRC Issue New Facility License**	**June 1, 1997**
• **GE Finalizes Master ISA Implementation Plan**	**August 1997**
• **Resolution of ISA Summary Issues**	**July 1998[1]**
• **Complete Fabrication and GAD Shop**	**December 1998**
• **Complete Uranium Recovery Operations**	**December 1999**
• **Complete Balance of Nuclear Operations**	**December 2000**
• **Complete NRC Review and Reconciliation**	**July 2001[2]**
• **Begin Revalidation of DCP**	**July 2001[3]**

[1]-The ISA summary issues must be resolved on a schedule that identifies acceptable performance by GE with sufficient lead time to prepare the required summaries.
[2]-The schedule assumes that the NRC reviews the ISA summaries as they are submitted and keeps them moving on a fairly level loaded schedule rather than letting them all bunch up at the end.
[3]-The commitment in the currently proposed license is to revalidate ISAs at a minimum of once every five years.

GE requests that the schedule commitment be such that small shifts in the schedule are accommodated since the concept is that the work is reasonably well level loaded over the period and that it is all finished by July of 2001.

4. *CRITICALITY CONTROLS AND THE ISA*

In the renewal application, when committing to perform an Integrated Safety Analysis (ISA) and to provide a summary description of it, GE should provide a commitment to provide the following information in the summary:

For each accident scenario (identified in the ISA) that, without preventative controls or mitigation could result in a nuclear criticality, information should be provided identifying the criticality controls established to prevent that scenario and evaluating their adequacy. Specifically, for each scenario state:

1. *The controls formally established to prevent it;*

2. *The set of controls for the process, for the scenario identified, meet established acceptance criteria for adequacy (There may be a single generic statement, e.g., "Unless otherwise indicated all controls have been determined to meet acceptance criterion xx of Practices and Procedures document PP-yy, i.e., double contingency."); and*

3. *The measures, such as configuration control, maintenance, and training needed to assure the reliability of these controls.*

The renewal should also contain a commitment to establish and maintain the controls identified in the ISA and to provide the measures to assure their reliability and conformance to the acceptance criteria. There should also be a commitment for maintaining the ISA current as changes to the processes are made.

Based on our discussions and agreements reached at the Management Meeting in Washington on May 20, 1997, the content of the first part of RAI #4 dealing with certain aspects of the ISA is to be considered as a part of resolving the operational details of the ISA.

GE has modified Section 4.1 to include the additional commitments requested in this RAI (also see answer to RAI #3).

5. *VALIDATION OF CRITICALITY EVALUATIONS AT ENRICHMENTS EXCEEDING FIVE PERCENT*

Benchmarks do not exist for the conduct of criticality evaluations of commitments in the range of 5-10 percent. Accordingly, for each specific process where uranium enriched to greater than five percent is to be used, provide a validation study including a criticality safety analysis and evaluation. This validation study

should establish for the specific cases calculated, (1) the case and data used are valid, and (2) that the specific quantitative method for setting margins of K_{eff} to account for uncertainties and biases. This quantitative method for margins should address both normal and accident conditions. The current additional margins of K_{eff} less than .97 for normal and .95 for accident conditions, in the absence of additional experiments or information, are inadequate to account for the uncertainties in extrapolation much beyond 5 percent enrichment.

GE has decided to withdraw our request for processing material enrichments up to 6.0 wt. % U235, at this time. Accordingly, Chapter 6, section 6.3.2.3 of our license renewal has been modified to exclude validation justification for processing higher than 5.0 wt. % U235. Similarly Sections 1.1 and 1.2.2 have been modified to limit the authorized enrichment for production to 5% (currently authorized at 6%).

Our future plans will include a separate license amendment addressing the 5-10% enrichment range for both GEKENO, GEMER.

At our Management Meeting of May 20, 1997, we also provided an overview of the direction our business is heading with high burn-up fuel and the implication for higher enrichments over the next few years.

6. *TABLE OF PLANT SYSTEMS AND PARAMETER CONTROLS*

Table 6.0, page 6.9 does not appear to be complete. Accordingly, please provide information on missing areas and systems. Specifically, add to this table the Dry Conversion Process Integration Facility, including transfer corridors.

Table 6.0 has been modified to call out the same level of detail for the integration of the DCP to the balance of the fuel manufacturing process as was used for the other process steps.

7. *CRITICALITY CONTROLS FOR TRANSFER CORRIDOR ADJACENT TO LAUNDRY*

Please provide information describing potential criticality scenarios identified for the Dry Conversion Process Integration Facility Transfer Corridor adjacent to the laundry. Have all scenarios that could introduce water unexpectedly into the corridor such as washing machine overflows, pipe breaks, drains plugged, etc., been identified, controls established, and the likelihood of causing a failure of moderation control evaluated to be acceptably low?

Yes, all credible external water source ingress pathways have been considered.

The model for the criticality accident is a non-uniform distribution of moderator in 1000 kilograms of uranium oxide powder enriched to 5 wt. percent U-235. The powder is assumed to be non-homogeneous with respect to particle sizes up to 1500 microns. Neutron reflection at the boundary of the model is twelve inches of water to represent the worst case. The non-uniform moderator limit calculated in CSA 1320.02, Rev. 03, is 9.3 kg. of water.

Normal operating condition is uranium oxide powder not exceeding 5 wt. percent enrichment in U-235 that is contained in a water (spray) resistant, strong stainless steel powder container. The transfer corridor is dry and the powder container is attended at all times while in the transfer corridor. The two concurrent contingencies required for a criticality accident to be possible in the Integration Transfer Corridor are loss of containment of the uranium oxide powder and accumulation of moderator. The controls for the Integration Transfer Corridor are categorized as two control systems designed to make the occurrence of either of these two contingencies unlikely.

Loss of containment of the uranium oxide powder is prevented by controls that make significant water ingress to the container or powder spillage from the container unlikely. The powder containers are built to a specification that requires the container be able to withstand 15 psi pressurization, normal handling stresses, and remain essentially dry when subjected to a moderate pressure water spray. The individual control specifications and requirements include the container fabrication drawing specification, operator procedural requirements, and information management systems that authorize container movements. During movement through the Integration Transfer Corridor, attendance of the powder container by an operator who is capable of immediately moving the powder container is required. The normal operating condition is to transfer the container through the transfer corridor without delay. Leaving the container unattended during the transfer, as a result of an actual or simulated emergency that requires immediate building evacuation, is a mitigating circumstance that may result in degradation of a control, but is considered an acceptably low safety risk. This condition is temporary and does not represent an immediate risk of a criticality accident.

Accumulation of moderator in the transfer corridor is prevented by controls that make it unlikely for water from either spraying or spreading over the floor into the corridor to occur. Credible sources of water are identified within the laundry room, overhead process piping, and rain water from the

external environment. The operator observes the condition of the Integration Transfer Hallway and is trained not to move the container through standing water on the floor or a water spray into the area.

Characteristics of the Integration Transfer Corridor are summarized as follows:

- **A single barrier roof over the transfer corridor is an acceptable barrier because of the requirement to directly transfer the powder container (minimize time present in the transfer corridor).**
- **Process piping that normally contains moderator and passes through the Integration Transfer Corridor is encased in a shroud that drains any leakage outside the transfer corridor.**
- **Rooms that normally containing moderator, such as the laundry room, are separated by a wall or doors to prevent ingress of water spray and other physical barriers that restrict the movement of the large powder transfer container.**
- **Gravity drains in the laundry room are physically at a lower elevation than the corridor and direct any accumulation of water on the floor from equipment (or fire protection system) away from the transfer corridor moderation restricted area.**

Drains, walls, and other physical barriers that are identified as important to nuclear criticality safety, are identified and marked to indicate their importance. These requirements are identified and documented in appropriate nuclear criticality safety analyses.

8. *The RAI dated March 5, 1997, questioned whether use of chemicals followed the OSHA Process Safety Management Standard (29 CFR 1910.119) in Section 7.1 (page 7.1). NRC also stated that elements of the Chemical Safety Program for UF_6 and hydrofluoric acid should be included in the license application.*

GE provided a general response, noting that the regulations are implemented through internal procedures as typified by GE's internal safety procedure 303 Safety Considerations in Design. However, elements of the Chemical Safety Program for UF_6 and hydrofluoric acid were not included or referenced in the license application.

Because UF_6 is licensed material and is used daily at the facility and hydrofluoric acid is an offgas produced by the processing of this licensed material, these chemicals should be specifically discussed in the Chemical Safety Program. Release of these materials could affect the availability and reliability of safety

controls relied on for plant safety. An acceptable approach to resolve the staff's concerns would be to include the following language in Chapter 7.0, Section 7.1:

> *This chemical safety program is applicable to the chemicals associated with the authorized activities in Chapter 1 and include UF_6 and hydrofluoric acid as well as any other chemicals which may directly or indirectly affect the nuclear safety of these activities.*

The management control elements of the chemical safety program of UF_6 and hydrofluoric acid should also be included in the license application. This means that management control elements of the GE-Wilmington Chemical Safety Program (as described in Section 7.2 of License Application) that apply to UF_6 and hydrofluoric acid should be included in the application. An acceptable commitment would be as follows:

> *The Chemical Safety Program for UF_6 and hydrofluoric acid utilize the following elements: Integrated Safety Analysis and Conduct of Operations.*

Exact placement in the license application is left to the discretion of the licensee but Chapter 7.0 - Chemical Safety appears to be the best chapter.

GE has modified Sections 7.1 and 7.2.1 to add the words of clarification identified in RAI #8.

9. *In Section 6.2.5.1, page 6.23 of the renewal application it is stated that "Structure and/or neutron absorbers that are not removable constitute a form of geometry control...". Since the use of the term "geometry control" for fixed absorbers has the potential for confusion, what measures will be taken to assure the proper assessment and maintenance of fixed neutron absorbers? Do plant procedures mandate compliance with the measures of ANSI/ANS 8.21 and with ANSI/ANS 8.1 section 4.2.3?*

Section 6.2.5.1 states, "...Favorable geometry is based on limiting dimensions of defined geometrical shapes to established subcritical limits. GE then considers structure and/or neutron absorbers that are not removable constitute a form of geometry control...". This means that for structures which includes neutron absorbers as an integral element (versus removable elements such as sleeves etc.) of their configuration, we consider the nuclear poison a subset of geometry control. These structures are included in the periodic verification requirements of section 6.2.5.5.

GE internal procedures do not mandate verbatim compliance with ANSI/ANS 8.21 and ANSI/ANS 8.1 section 4.2.3, however, our programs and

procedures conform to the intent of requirements and guidance expressed in these standards.

10. *What is the technical basis for the safe batch rule of section 6.2.4 embodied in the equation:*

kgs UO_2 x 0.88 = kg X • f, where f = wt. % U in compound X and kgs UO_2 is the safe batch size for UO_2?

GE acknowledges that there is a problem with the notation, and we have re-written the equation, for clarity, to read the following:

kgs X = (kgs UO_2 • 0.88) / f

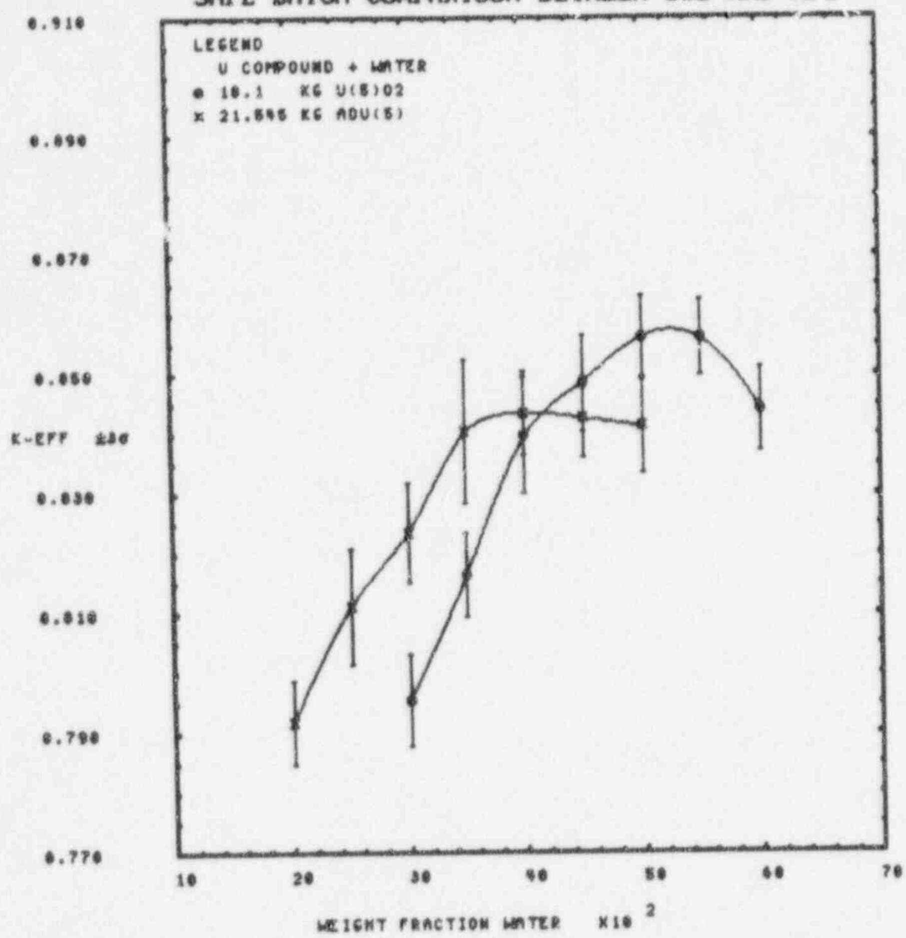
where,	**kgs X**	**= safe batch value of compound 'X'**
	kgs UO_2	**= safe batch value for UO_2**
	0.88	**= wt. % U in UO2**
	f	**= wt. % U in compound X**

Section 6.2.4 of our license application has been modified accordingly.

The safe batch tables in the current SNM-1097 license and in the renewal application are for uranium dioxide (UO2). The mass includes the oxygen which represents approximately 11.85% of the UO2 mass. Therefore, if we consider another uranium compound and replace the mass contribution from oxygen by the mass contribution from the non-uranium constituents of the other compound, we will still have a safe batch providing that the other uranium compound is neutronically less reactive.

UO2 is the most reactive form of uranium processed at this facility. Therefore, if we apply the equation in question to ammonium diuranate (ADU), we would expect the resulting mass to produce a k-effective less than that resulting from the safe batch mass of UO2. This is shown in the attached figure for 18.1 Kg of UO2 and 21.545 Kg of ADU, each at 5.00% U235 enrichment. The 21.545 was determined by multiplying 18.1 by 0.88144 and dividing by 0.74049, the respective U-factors for UO2 and ADU. UO2 is also more reactive than the other uranium compounds characteristic of our processes [U3O8, UO2F2, UO2(NO3)2 - 6H2O]. While uranium metal is not a compound, we recognize that it can be more reactive than UO2.

SAFE BATCH COMPARISON BETWEEN UO2 AND ADU



11. *With respect to Table 6.1, "Safe Geometry Values", what is the significance of the missing values for cylinder diameters for Homogeneous Aqueous Solutions? What methods are to be used in this case, if not this table? In particular, what method was used to determine the diameter of UF_6 hydrolysis columns? For cases where the enrichment exceeds 5%, provide additional information showing how these values have been validated and that they incorporate sufficient margins to account for uncertainties. Are they validated by comparison experiments?*

    For values in Table 6.1 at enrichments less than 5%, a comparison to the most recently peer reviewed guidance, Norman L. Pruvost and Hugh C. Paxton, LA-12808 Nuclear Safety Guide, Sept. 1996 (formerly TID-7016), shows several values that differ in the non-conservative direction. These are noted below. Please provide additional justification for these values or adopt those from LA-12808.

Differences between LA-12808 and Table 6.1:

	Enrich.	*Table 6.1*	*LA-12808*
Homogeneous UO2 H20 cylinder diam.	*2%*	*16.70 in.*	*16.50 in.*
Homogeneous UO2 H2O slabs	*2%*	*8.90 in.*	*8.82 in.*
"	*3%*	*6.25 in.*	*6.10 in.*
"	*4%*	*5.10 in.*	*4.96 in.*
"	*5%*	*4.45 in.*	*4.37 in.*
Homogeneous aqueous solutions, slabs	*4%*	*6.00 in.*	*5.94 in.*
Homogeneous UO2 & water, Kgs UO2	*4%*	*25.7 Kg.*	*25.5 Kg.*
"	*5%*	*18.1 Kg.*	*16.0 Kg.*
Heterogeneous UO2 pellets & water, Kgs UO2	*3%*	*38.1 Kg.*	*36.1 Kg.*
"	*4%*	*24.7 Kg.*	*20.3 Kg.*
"	*5%*	*18.1 Kg.*	*13.9 Kg.*

The values were included in the 1979 version of SNM - 1097 but not in the current version. Currently, while they are acceptable limits, they are generally used very little since our current practice is to utilize discreet models in most cases. Some of the older criticality analysis for the facility are based on these values. The values are as follows:

wt% U235	**Inf Cyl Dia**	**wt.% U235**	**Inf Cyl Dia**
2.00	**16.7 in**	**3.25**	**12.5 in**
2.25	**15.0 in**	**3.50**	**12.1 in**
2.50	**14.0 in**	**3.75**	**11.9 in**
2.75	**13.3 in**	**4.00**	**11.7 in**
3.00	**12.9 in**	**5.00**	**9.5 in**

These values represent 93% of the minimum critical dimension, and will be added to Table 6.1.

The 10-inch Schedule 40 hydrolysis receiver and storage tanks are modeled explicitly and analyzed using the GEKENO Monte Carlo program. The analysis is documented in "CSA - EVALUATION OF HYDROLYSIS AT 5%", performed under Change Request 89.0224.

The questions regarding enrichment are discussed under RAI #5. Based on that discussion, the tables are modified by deleting values above 5.00%.

The remainder of this item deals with comparison of single parameter limits between Table 6.1 of the GE license, and the recently published LA-12808.

Specifically, LA-12808 Table 8 (solutions) and Table 9 (homogeneous and heterogeneous oxides) are based on work originating from H. K. Clark (Table 8 - NSE vol. 81, pp. 351-378, 1982 and Table 9 from DP-1014, 1966, receptively). Both sets of data were obtained using analytical techniques normalized to appropriate critical experiments, however, both include some degree of uncertainty in the results. The small differences identified in the RAI for dimensional limits appear to be within the uncertainty and are not believed to be significant.

The differences in comparison of safe batch values is significant and occurs because the numbers do not represent the same thing. The safe batch values in Table 6.2 represent 45% of the minimum critical mass. The corresponding values in the RAI are obtained by taking 45% of the subcritical limits reported in Table 9 of LA-12808. These limits were taken from DP-1014 tables of "safe" values which are defined in DP-1014 to correspond to a k-effective value of 0.98.

Significant differences are noted between the safe batch values in Table 6.2 for Heterogeneous UO2 Pellets & Water Mixtures and the safe batch values inferred from LA-12808. Some of the difference can be attributed to the 45% of the minimum critical value (GE) verses the 45% of the subcritical value (LA-12808) explained in the previous paragraph. Another significant difference is due to the data in Table 6.2 being for pellets and the data in LA-12808 being for optimum diameter rods. Since the optimum rod diameter for mass limited systems of 5% enriched UO2 is about 1/8-inch diameter and pellets is much larger (1/3 to 2/5-inch diameter) the safe batch table for pellets does not apply for smaller dimensions than the pellet diameter.

In any case, the standard practice at GE is to explicitly model mass limited 'heterogeneous systems' containing fuel diameters smaller than pellets using Monte Carlo (e.g., GEMER) analytical methods.

12. *Concerning Section 6.4.1 page 6.36, is the location and spacing of criticality monitoring alarms such that the system meets the requirements of either 10 CFR 70.24 (a)(1) or (a)(2)? Specifically, is coverage of all areas by two detectors provided? Are there any areas of the facility that will not be covered by detectors meeting the requirements? Is SNM ever handled, used, or stored in these areas?*

Yes, GE's criticality monitoring accident alarm system meets the requirements of 10 CFR 70.24 (a) (1), except for the special authorizations stipulated in Section 1.3.11 of the renewal application.

Outline for the ISA Summary for GE Balance of Plant

I. *The areas of review for each system should be listed e.g., radiological safety, criticality safety, chemical safety, fire protection, and industrial safety, etc.*

II. *For each system, a description of how the ISA teams are formed, type of membership, minimum qualification of members, how areas of review were integrated, and management and QA oversight.*

III. *A list of specific written plant procedures, techniques, and computer based tools used by the ISA teams to perform the ISA for each system.*

IV. *A list of the segments that the system was broken into to perform the ISA and why.*

V. *A description of how the sequences of the work was performed by the ISA team e.g., identify the hazards, determine the consequences and likely frequency, identify the controls which prohibit or mitigate the consequences, establish a risk ranking (frequency x consequence) unmitigated and mitigated.*

VI. *A description of how the ISA team determined the consequences of the event or condition.*

VII. *Based on the ISA process, provide a list of the most important process segments and the controls relied upon to prevent and/or mitigate an incident. The incident description should include a list of the initiating event (internal or external), the unmitigated consequences of the resulting accident, and the necessary level of quality and reliability established for each control.*

VIII. *Summary matrix of accident sequences plotted by consequence versus probability (qualitative).*

GE Nuclear Energy

May 29, 1997

Mr. M. F. Weber, Licensing Branch, NMSS
U.S. Nuclear Regulatory Commission
Mail Stop T 8-D-14
Washington, DC 20555-0001

Subject: Request to Delay the Biennial Emergency Exercise

Reference: (1) NRC License SNM-1097, Docket 70-1113
(2) Radiological Contingency & Emergency Plan (RC&EP)

Dear Mr. Weber:

The purpose of this letter is to request a delay in the biennial exercise required in our Radiological Contingency & Emergency Plan (RC&EP) from June 30, 1997 until August 31, 1997. The delay is necessitated by the critical startup activities associated with the Dry Conversion Process and the availability of federal, State and local agencies. The exact date will be coordinated with Region II and the other support agencies.

The biennial exercise was originally scheduled for June of this year, based, in part, on the expectation that the new DCP critical startup phase would then be completed. The DCP startup phase has unavoidably extended into the proposed June exercise time period

The advantages of rescheduling the exercise have been discussed with Region II personnel who are supportive of the proposed delay.

The request can be accommodated as a provision of the soon to be issued license, however, if this is not possible a separate authorization will be acceptable.

Please contact me at (910) 675-5889 or Charlie Vaughan, at (910) 675-5656, if you have any questions or would like to discuss this matter further.

Sincerely,

GE NUCLEAR ENERGY

Ralph J. Reda, Manager
Fuels & Facility Licensing

/zb

cc: RJR-97-067
GL Troup