Westinghouse

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U.S. Nuclear Regulatory Commission
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[PUBLIC VERSION]

September 29, 2005

SUBJECT: WESTINGHOUSE LICENSE SNM-1107 RENEWAL APPLICATION (DOCKET 70-1151)

In accordance with 10CFR70.33, Westinghouse Electric Company requests renewal of our existing license, SNM-1107, which expires on November 30, 2005. This renewal application is being submitted in accordance with 10CFR70.21 and 70.22. The application has been revised per the consensus agreement at our meeting on September 12, 2005, and in subsequent conversations with your Staff.

Westinghouse also requests that existing License condition S-9 be deleted and that License condition SG-1.9 be revised to the following:

Not withstanding the requirement of section 2.1.1, block 6.b, of NUREG/BR-0006, which is incorporated via 10 CFR 74.15, to complete receiver's measurements of scrap receipts (following recovery processing) within 60 days of receipt, the licensee shall not be subject to any time limit relative to recovering and measuring UF_6 heels when the block 6.b action code N (of DOE/NRC Form 741) is used to book such receipts.

I hereby affirm that the statements made in this application are complete, true and correct to the best of my knowledge and belief. If you have any questions, please call me at (803) 647-3338.

Sincerely,

Nancy Blair Parr

Nancy Blair Parr
Licensing Manager

Enclosures
Enclosure 1, Application for Renewal of SNM-1107
Enclosure 2, Environmental Report

cc: U. S. Nuclear Regulatory Commission
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ENCLOSURE 1

APPLICATION FOR RENEWAL OF SNM-1107

[PUBLIC VERSION]

WESTINGHOUSE ELECTRIC COMPANY
NUCLEAR FUEL

APPLICATION FOR RENEWAL OF A SPECIAL NUCLEAR MATERIAL LICENSE FOR THE COLUMBIA FUEL FABRICATION FACILITY COLUMBIA, SOUTH CAROLINA

LICENSE NUMBER
SNM-1107

September 29, 2005 (Revision No. 0.0)

U.S. NUCLEAR REGULATORY COMMISSION
DOCKET 70-1151

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Tables and Figures

CHAPTER 1.0

GENERAL INFORMATION

1.1 GENERAL INFORMATION STRUCTURE

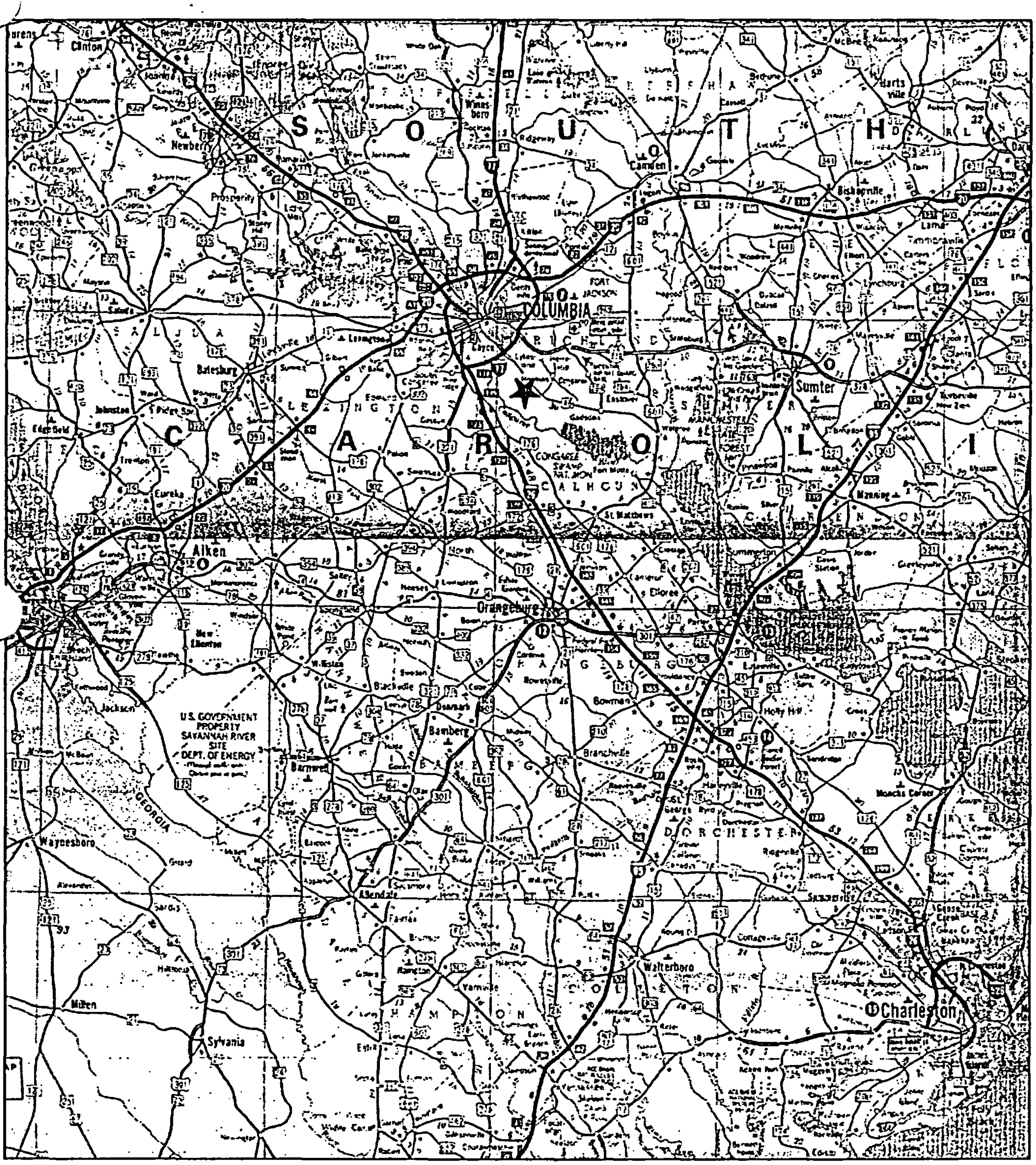
1.1.1 Site Description

The Westinghouse Columbia Fuel Fabrication Facility (CFFF) is located near Columbia, South Carolina, and is situated on a 1,158 (approximate) acre site in Richland County, some 8 miles southeast of the Columbia city limits, along State Highway 48 (Bluff Road). The region around the site is sparsely settled, and the land is characterized by timbered tracts and swampy areas penetrated by unimproved roads. Farms, single-family dwellings, and light commercial facilities are located mainly along nearby highways. A map of the surrounding area is presented in Figure 1.1.

The site is bordered by abutting properties, as presented in the Physical Security Plan described in Paragraph 1.1.2.1(e) of this Chapter. Of the total acreage, only 60 acres (about 5 percent) have been developed to accommodate the fuel fabrication buildings, holding ponds, parking and landscaped areas. Approximately 1098 acres of the site remain undeveloped. A map of the property boundary is presented in Figure 1.2. A site plan and site plan key are presented in Figures 1.3 and 1.4.

More details of the CFFF location, including proximity to nearby towns, industries, public facilities, the Congaree River, transportation links, and site topography are presented in the Site Emergency Plan described in Chapter 9.0 of this License Application. Extensive details of the site characterization are presented in the 1975 Environmental Evaluation Report described in Chapter 10.0 of this License Application and in subsequent updates.

Figure 1.1 CFFF Surrounding Areas



★ CFFF Site

1.1.2 Facility and Process Description

The CFFF is primarily engaged in the manufacture of fuel assemblies for commercial nuclear power plants, both pressurized water reactors (PWR) and boiling water reactors (BWR). The manufacturing operations to be authorized by this License Application consist of receiving low-enriched, less than or equal to 5.0 w/o U-235, uranium hexafluoride (UF_6); converting the UF_6 to produce uranium dioxide (UO_2) powder; and, processing the UO_2 powder through pellet pressing and sintering. These processes are followed by fuel rod loading and sealing, and fuel assembly fabrication. Manufacturing operations are governed by technically sound radiation and environmental protection, nuclear criticality safety, industrial safety and health, SNM safeguards, and quality assurance programs described in detail in this License Application.

The primary system used to convert UF_6 to UO_2 is the well known Ammonium Diuranate (ADU) process. ADU conversion equipment has been designed to receive and process uranium in enrichments up to 5.0 w/o U-235, through fuel assembly fabrication and shipping. These operations are supported by neutron absorber addition or coating, laboratory, scrap recovery, and waste disposal systems.

1.1.2.1 Site Utilities and Services

(a) Electrical Supply

The CFFF is served by a single, 115,000 volt, electrical supply line. At least four diesel-powered standby generators are provided and maintained to meet site emergency electrical power requirements in the event of a temporary outage of the normal supply source. The emergency power is automatically provided to crucial process equipment; emergency lighting systems; cooling system pumps; all fire alarm, hazard alarm, and other designated safety alarm systems; Conversion Control Room alarms, health physics sampling systems; and, emergency ventilation systems, including scrubbers.

(b) Water Supply

A 10-inch main from the Columbia Municipal Water Authority supplies water to the site.

(c) Gaseous and Liquid Effluent Management

Gaseous exhausts from process areas with potential for contamination are passed through HEPA filtration to remove entrained uranium particulates prior to discharge to the environment. Exhausts containing uranium in soluble form are passed through aqueous scrubbers preceding the HEPA filters. Following filtration, the gases are continuously sampled to enable analyses for demonstrating compliance with the limits specified in this License Application.

Liquid process wastes are treated prior to discharge to the Congaree River. Waste treatment, for removal of uranium, ammonia and fluorides, consists of filtration, flocculation, lime addition, distillation and precipitation (in a series of holding lagoons). Site sanitary sewage is treated in an extended aeration package plant prior to discharge, either directly or through a polishing lagoon. The discharged effluent is chlorinated, and mixed with treated liquid process waste, at the facility lift station. The combined waste is then passed through a final aerator, followed by pH adjustment as required, and subsequently pumped to the river via a 4-inch pipeline. Compliance with licensed discharge limits is verified by passing the waste streams through on-line monitoring systems; or, by manual sampling and analysis on a batch-basis. The treatment systems are designed with sufficient holdup capacity to assure that the discharge limits are continuously met.

Storm water from the site enters a system of drainage ditches and ultimately flows to the Congaree River.

(d) Solid Waste Storage and Disposal

Solid wastes are sorted into appropriate combustible and noncombustible fractions and are placed in specifically designated collection containers located throughout the work areas. (The wastes consist of paper, wood, plastic, metal, floor sweepings, and similar materials which are contaminated by, or contain, uranium.) Following a determination that the wastes are in fact properly sorted, the contents are transferred to a waste processing station.

Materials that are suited for complete survey may be decontaminated for free-release, or re-use, in accordance with provisions of this License Application. Combustible wastes are packaged in compatible containers, assayed for grams U-235, and stored to await incineration. Noncombustible wastes and selected combustible wastes are packaged in compatible containers, compacted when appropriate, measured to verify the uranium content, and placed in storage to await shipment for further treatment, recovery or disposal.

Administrative controls are in effect to assure that only authorized materials are packaged for disposal. These include verification of package contents, container security to minimize the probability of unauthorized additions to the containers, documentation of package contents, and routine over-checks to verify that the controls are effective.

(e) Site Safeguards

Physical Security at the CFFF is described in the NRC-approved *Physical Security Plan for the Columbia Fuel Fabrication Facility,* dated September 1, 1984, and subsequently revised in accordance with applicable regulations. Nuclear Material Control and Accountability (MC&A) at the CFFF is described in the NRC-approved *Fundamental Nuclear Material Control Plan for the Columbia Fuel Fabrication Facility*, dated April 1, 1987, and subsequently revised in accordance with applicable regulations. These Plans detail the measures employed at the facility to detect any potential loss of, and mitigate the opportunity for theft of, Special Nuclear Material (SNM) of Low Strategic Significance, in accordance with the applicable requirements of 10CFR73 and 74.

(f) Defense-in-Depth Design

For all new CFFF facilities, or new processes at existing facilities, the defense-in-depth design philosophy is implemented. For all existing facilities, the defense-in-depth design philosophy is implemented where practicable. An example of this philosophy is

(1) dispersible hazardous material work conducted in hoods, glove boxes, or other enclosures;
(2) the hoods, glove boxes, and other enclosures located within a Contamination Controlled Area;
(3) the Contamination Controlled Area located within the manufacturing building;
(4) The manufacturing building serviced by a HEPA filtered ventilation system;
(5) the ventilation system exhaust stacks located within the Controlled Access Area (CAA); and
(6) the CAA located within the Site Boundary.

(g) Instrumentation and Control Systems

For all new CFFF facilities, or new processes at existing facilities, a design philosophy that includes instrumentation and control systems to monitor and control the behavior of Items Relied on for Safety (IROFS) is implemented. For all existing facilities, a design philosophy that includes instrumentation and control systems to monitor and control the behavior of IROFS is implemented where practicable. This philosophy takes the form of a Safety Instrumented System (SIS). An example of a SIS would be a Programmable Logic Controller (PLC) as the logic solver with a connected level probe as the sensor, and a connected solenoid valve as the final element; such that, when the process liquid level reaches the level probe, the PLC shuts off the fluid input via the solenoid valve.

Figure 1.2 CFFF Property Boundaries

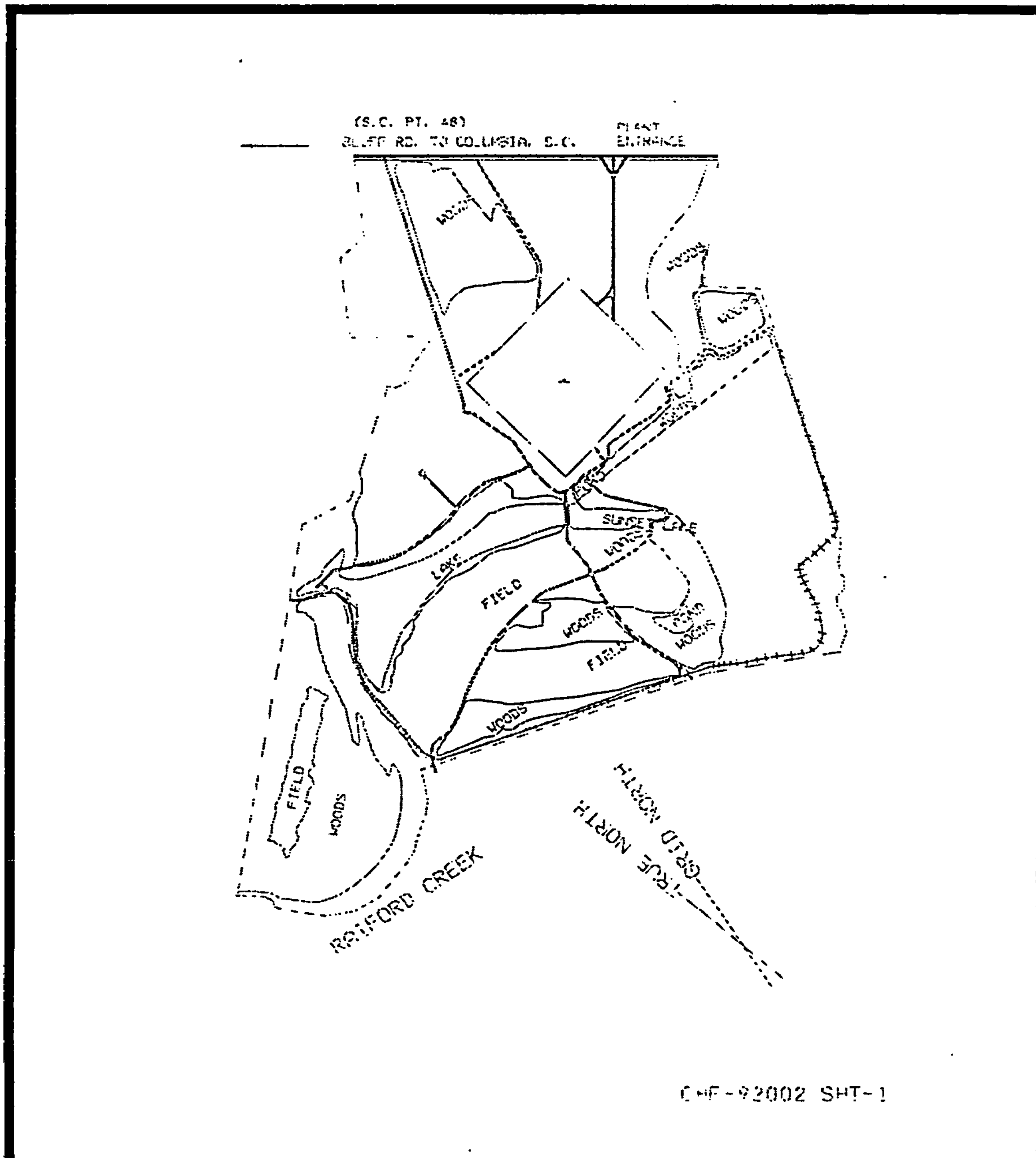


Figure 1.3 CFFF Site Plan

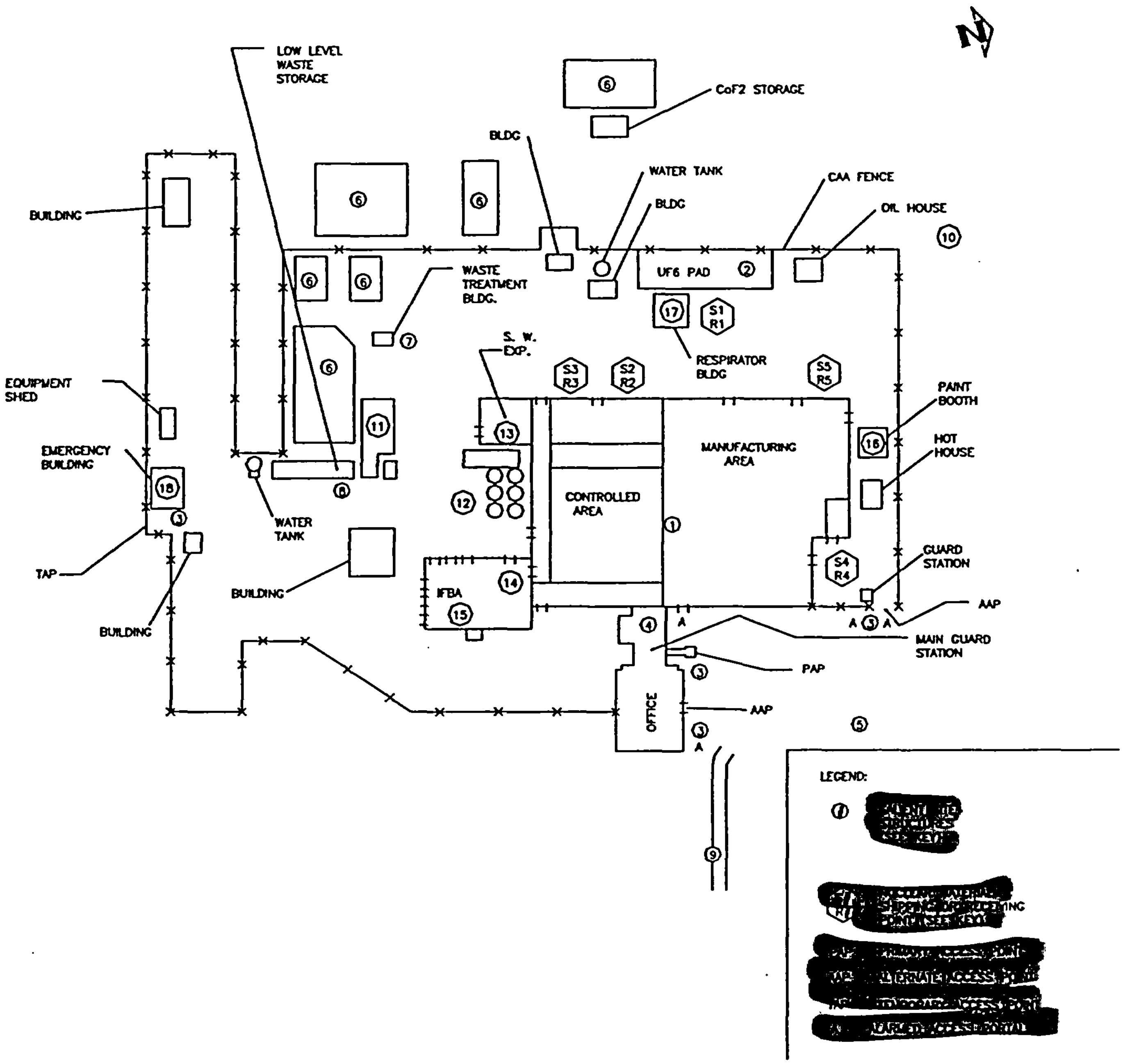


Figure 1.4 CFFF Site Plan Key

[redacted]
R1

[redacted]
R2

[redacted]

2: [redacted]

3: [redacted]

4: [redacted]

5: [redacted]

6: [redacted]

7: [redacted]

8: [redacted]

[redacted]

10 [redacted]

11: [redacted]

12 [redacted]

1 [redacted]

14 [redacted]

[redacted]

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[1] Nuclear Material Routinely Contained

1.1.3 Scope of Licensed Activities

Compliance with all applicable parts of Title 10, Code of Federal Regulations is required, unless specifically amended or exempted by NRC Staff.

(a) Authorized Activities

- Authorized activities at the Columbia Fuel Fabrication Facility include: (1) Receipt, handling, and storage of Special Nuclear Material as uranium hexafluoride, uranium nitrates, uranium oxides; and/or contained in pellets, fuel rods, fuel assemblies, samples, scrap, and wastes; (2) Receipt, handling, and storage of other licensed radioactive material; (3) Chemical conversion processing including vaporization and hydrolysis, precipitation and centrifugation, drying, calcining, comminution, and blending; (4) Fuel fabrication including powder preparation, die-lubricant addition, nuclear absorber addition, pelleting, sintering, grinding, pellet coating with nuclear absorbers, fuel rod loading and inspection, and final fuel assembly; (5) Quality assurance and control activities; (6) Analytical Services Laboratory operations including wet-chemistry and spectrographic methods; (7) Metallurgical Laboratory operations including sample preparation, polishing, testing, and examination; (8) Chemical Process Development operations including laboratory-scale process research, prototype development, and equipment check-out; (9) Mechanical Process Development operations including laboratory-scale research and development; (10) Health Physics Laboratory operations including sample preparation and analysis, instrument repair and calibration, respirator fit-testing, and bioassay sample and sealed source storage; (11) In-house and outsourced scrap recovery operations including scrap batch processing, solvent extraction, coated-pellet recovery, ash processing, scrap blending, and acid recovery; (12) UF_6 cylinder washing and decontamination, hydrostatic testing, and recertification; and, re-work of returned fuel assemblies; (13) Equipment and facility maintenance activities; (14) Facility, equipment, and protective clothing decontamination activities; (15) Waste storage and disposal preparation operations including HEPA filter testing, conversion liquid waste treatment, advanced waste-water treatment, lagoon storage, incineration, contaminated waste packaging for disposal, and calcium fluoride disposition; (16) Ancillary mechanical operations including non-radioactive component fabrication and assembly; and (17) Shipping container and over pack refurbishment.

- The CFFF may also perform work for other British Nuclear Fuels plc (BNFL) operations, or outside customers, which is within the authorized capabilities of the facility.

1.1.4 Material Possession Limits and Constraints

The following are the maximum quantities of nuclear materials that are possessed by the CFFF at any one time; and, the constraints on procurement, use, and transfer of such material.

(a) Material Possession Limits are: (1) 5-grams of U-233 in any chemical or physical form, limited to laboratory use as individual 1-gram maximum quantities in ventilated hoods, glove boxes, or other enclosures; (2) 350-grams of U-235, as uranium of any enrichment, in any chemical or physical form; (3)[redacted]ilograms of U-235 enriched to no greater than 5 weight-percent, in any chemical or physical form except metal; (4) 1.5-grams of Pu-238/239 as sealed sources; and (5) Transuranics and fission products, not to exceed 3,300 Bq alpha per KgU, or 440,000 Mev Bq gamma per Kg U (*i.e.*, the limits on alpha and gamma activity specified for "enriched reprocessed UF_6" in ASTM C996-96; *Standard Specification for Uranium Hexafluoride Enriched to Less Than 5% U-235*), and, not to exceed 5-grams of plutonium.

(b) Constraints on procurement, use, and transfer of nuclear materials are: (1) Procurement quantities are in accordance with continuing CFFF manufacturing needs; (2) Production, utilization, and/or significant loss is not authorized; and (3) Transfer is only as arranged with facilities authorized to receive and possess the materials.

1.1.5 Institutional Information

This Application requests a twenty year renewal of License SNM-1107, Docket 70-1151, for the Columbia Fuel Fabrication Facility (CFFF), located at 5801 Bluff Road in Columbia, South Carolina, and operated by Westinghouse Electric Company LLC (Westinghouse). Westinghouse is owned and controlled by British Nuclear Fuels plc (BNFL). In accordance with the requirements of 10CFR70.22(a)(1), additional institutional information is provided below.

20-year renewal

1.1.5.1 Applicant and State of Incorporation

Westinghouse Electric Company LLC; Delaware

1.1.5.2 Location of Principal Office

Monroeville, Pennsylvania

1.1.5.3 Names (Citizenships) and Addresses of Principal Officers

Stephen R. Tritch (USA)
President and Chief Executive Officer
Westinghouse Electric Company
Westinghouse Energy Center
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Pittsburgh, Pennsylvania 15230-0355

Mike J. Saunders (UK)
Senior Vice President
Nuclear Fuel
Westinghouse Monroeville Site
4350 Northern Pike
Monroeville, Pennsylvania 15146

Sandy D. Rupprecht (USA)
Vice President, U.S. Fuel
Westinghouse Columbia Site
P.O. Drawer R
Columbia, South Carolina 29250

Mark W. Fecteau (USA)
CFFF Site Manager
Westinghouse Columbia Site
P.O. Drawer R
Columbia, South Carolina 29250

1.1.5.4 Company Contact for Licensing Matters

Griff Holmes
Manager, Environmental Health and Safety
Westinghouse Energy Center
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355

1.1.5.5 Site Contact for Licensing Matters

Nancy B. Parr
Licensing Project Manager
Westinghouse Columbia Site
P.O. Drawer R
Columbia, South Carolina 29250

1.1.5.6 Additional Financial and Business Information

Additional financial and business information for Westinghouse Electric Company can be found on the Internet at www.westinghousenuclear.com.

1.1.6 Key Terms and Definitions

Throughout this License Application, the following terms are defined as indicated:

1.1.6.1 Active Engineered Controls

Safety Related Controls that require hardware and/or software assistance, but no operator action or other response, to be effective when called upon to ensure health, safety, and/or protection of the environment. Active Engineered Controls are preferred over Administrative Controls.

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1.1.6.2 Administrative Controls

Safety Related Controls that rely on an operator to perform an action or other response to be effective when called upon to ensure health, safety, and/or protection of the environment. Administrative Controls might or might not involve assistance by a computer or an alarm. Administrative Controls are the least preferred method of control.

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1.1.6.3 Alternative Actions

Tests, procedures or other practices that may be substituted for prescribed activities deemed appropriate by the Regulatory Component. In such case, a detailed analysis is performed and documented by the cognizant Regulatory Functions. The analysis includes a comparison of the proposed action with that specified in this License Application; and, a demonstration that action levels and limits are being met, and that health and safety of employees and the public, and quality of the environment is being protected.

1.1.6.4 Chemical Area

An area where uncontained radioactive material is processed, the probability of contamination on floors and accessible surfaces is high, and protective clothing is required. Examples include the UF_6 Bay, the Conversion Area, the Pelleting Area, the Rod Loading Area, *etc.*

1.1.6.5 Clean Area

An area where radioactive material, if present, is completely contained; and, there is negligible contamination on floors and accessible surfaces. Examples include the Machining Area, Grid Assembly Area, Final Assembly Area, Office Areas, and the Cafeteria.

1.1.6.6 Component

When used in an administrative context, this is an independent organizational unit that is distinguishable by its assigned responsibilities. Examples include the Engineering Component, the Manufacturing Component, the Quality Component, and the Regulatory Component.

1.1.6.7 Conduct of Operations

An alternate name for Management Measures, as defined in 10CFR70.4.

1.1.6.8 Contamination Controlled Area

An alternate name for the Chemical Area.

1.1.6.9 Controlled Area

The Controlled Area is the area between the Controlled Access Area and the Site Boundary.

1.1.6.10 Controlled Access Area

A physically defined area, represented on three sides by a seven-foot high barrier of Number-11 American Wire Gauge fabric-fence topped by three strands of barbed wire and a coil of razor wire, and represented on the fourth side by the Administration Building and Main Manufacturing Building. This area is the Controlled Access Area described in the CFFF Physical Security Plan.

1.1.6.11 Defense in Depth

A design philosophy that is based on providing successive levels of protection such that health and safety will not be wholly dependent upon any single element of the design, construction, maintenance, or operation of the facility.

1.1.6.12 Enrichment Limit

When used as an authorized enrichment limit, 5.0 weight-percent (w/o) U-235 means that, based on an enrichment measurement uncertainty no greater than 0.50 percent relative, the hypothesis that the true enrichment level is 5.0 w/o U-235 or less can not be rejected at the 0.05 level of significance.

1.1.6.13 Equivalent Experience

When used in a personnel qualification context for equating experience with education, eight-years of applicable experience is equivalent to a baccalaureate degree.

1.1.6.14 Fixed Location General Air Sample

Air samples used to assess general area radioactivity concentrations; and, to assess the adequacy of radioactive material confinement and containment within the processing areas of the facility; and, to establish airborne radioactivity areas.

1.1.6.15 Fixed Location Breathing Zone Representative Air Sample

Air samples used to assess and assign operator intakes of airborne radioactive materials.

1.1.6.16 Frequencies

When audit, measurement, surveillance, and/or other frequencies are specified in license documents (such as this License Application, the Physical Security Plan, the Fundamental Nuclear Material Control Plan, *etc.*), the following time spans apply:

(a) *Daily* means once each 24-hour period;
(b) *Weekly* means once each 7-consecutive-days;
(c) *Monthly* means 12-per-year, with each covering a span of 40-days or less;
(d) *Quarterly* means 4-per-year, with each covering a span of 115-days or less;
(e) *Semiannual* means 2-per-year, with each covering a span of 225-days or less;
(f) *Annual* means 1-per-year, with each covering a span of 15-months or less;
(g) *Biennial* means once every 2-years, with each covering a span of 30-months or less; and,
(h) *Triennial* means once every 3-years, with each covering a span of 45-months or less.

1.1.6.17 Function

When used in an administrative context, an individual (or individuals), designated by the Component Manager, acting in coordination with the other personnel of the component, having the capability, responsibility, and authority to make and implement decisions required to carry out assigned duties. Examples for the Regulatory Component include the Environmental Protection Function, the Radiation Safety Function, the Nuclear Criticality Safety Function, the Chemical Safety Function, the Fire Safety Function, the Safeguards Function, *etc.*

1.1.6.18 Integrated Safety Assessment (ISA)

An alternate name for Integrated Safety Analysis (ISA) as defined in 10CFR70.4.

1.1.6.19 Integrated Safety Assessment (ISA) Summary

An alternate name for Integrated Safety Analysis (ISA) Summary as defined in 10CFR70.4.

1.1.6.20 Items Relied On For Safety (IROFS)

A subset of Safety Significant Controls (SSCs), disclosed by the Integrated Safety Analysis, designated to prevent nuclear criticality accidents, and to prevent and/or mitigate high and intermediate consequence events.

1.1.6.21 License Annex

An alternate name for Integrated Safety Analysis (ISA) Summary as defined in 10CFR70.4.

1.1.6.22 Licensed Activity

That combination of personnel, plant, and equipment established by Westinghouse to carry out the processing of radioactive material at the CFFF, as authorized by this License Application.

1.1.6.23 May

Denotes implied permission by NRC Licensing Staff to take a stated action or course.

1.1.6.24 Passive Engineered Controls

Safety Related Controls that require no hardware and/or software assistance, or operator action or other response, to be effective when called upon to ensure health, safety, and/or protection of the environment. Passive Engineered Controls are the most preferred method of control.

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1.1.6.25 Portable Air Sample

An air sample that is not integrated into the CFFF's central air sample vacuum system.

1.1.6.26 Radiation Worker

Any individual who, in the course of employment, is likely to receive an annual occupational dose in excess of 100-millirem.

1.1.6.27 Regulatory-Significant Procedures

Those procedures that contain, in whole or in part, actions that are important to environmental protection, health, safety, and/or safeguards.

1.1.6.28 Restricted Area

Areas, controlled by the site access road Security Building, to which access is restricted by physical and/or administrative methods; and, which is monitored on a scheduled basis by the Security Function.

1.1.6.29 Safe Mass

The critical mass for a particular process or vessel given the credible material geometry for that process or vessel, and the Nuclear Criticality Safety bounding assumptions for the applicable material type (e.g., homogeneous UO_2) and reflection. Optimum moderation and material density are assumed.

1.1.6.30 Safety Margin Improvement Controls (SMICs)

A subset of Safety Related Controls, as specified by the cognizant Safety Functions, to increase the margin of health, safety, and protection of the environment.

1.1.6.31 Safety-Related

Relevant to systems crucial or important to safety; and, those systems that improve the margin of safety (*e.g.*, in the context of maintenance).

1.1.6.32 Safety Related Controls (SRCs)

The complete set of CFFF engineered and administrative controls designed to promote health and safety, and protection of the environment.

1.1.6.33 Safety-Significant

Relevant to systems crucial or important to safety (*e.g.*, in the context of quality assurance).

1.1.6.34 Safety Significant Controls (SSCs)

A subset of Safety Related Controls, as specified by the cognizant Safety Functions, to provide basic health and safety, and/or protection of the environment.

1.1.6.35 Unrestricted Area

An Area, access to which is neither limited nor controlled by the Security Function.

1.1.6.36 Will

Denotes a mandatory commitment to take a stated course or action.

CHAPTER 2.0

MANAGEMENT ORGANIZATION

2.1 MANAGEMENT ORGANIZATION STRUCTURE

British Nuclear Fuels plc (*aka* "BNFL") is comprised of two major Business Groups. One of these, the Nuclear Utilities Business Group (*aka* "Nuclear Utilities"), oversees operation of Westinghouse Electric Company LLC (*aka* "Westinghouse"). The Chief Executive of Nuclear Utilities, who also serves as President and Chief Executive Officer (CEO) of Westinghouse, reports to the Chief Executive of BNFL.

2.1.1 Organizational Responsibilities and Authorities

Westinghouse is comprised of several businesses. One of these, Westinghouse Nuclear Fuel (*aka* "Nuclear Fuel"), encompasses commercial activities directly relating to the development, manufacturing, and marketing of products contributing to the use of nuclear reactors for generation of electric power. The Senior Vice President of Nuclear Fuel reports to the President and CEO of Westinghouse.

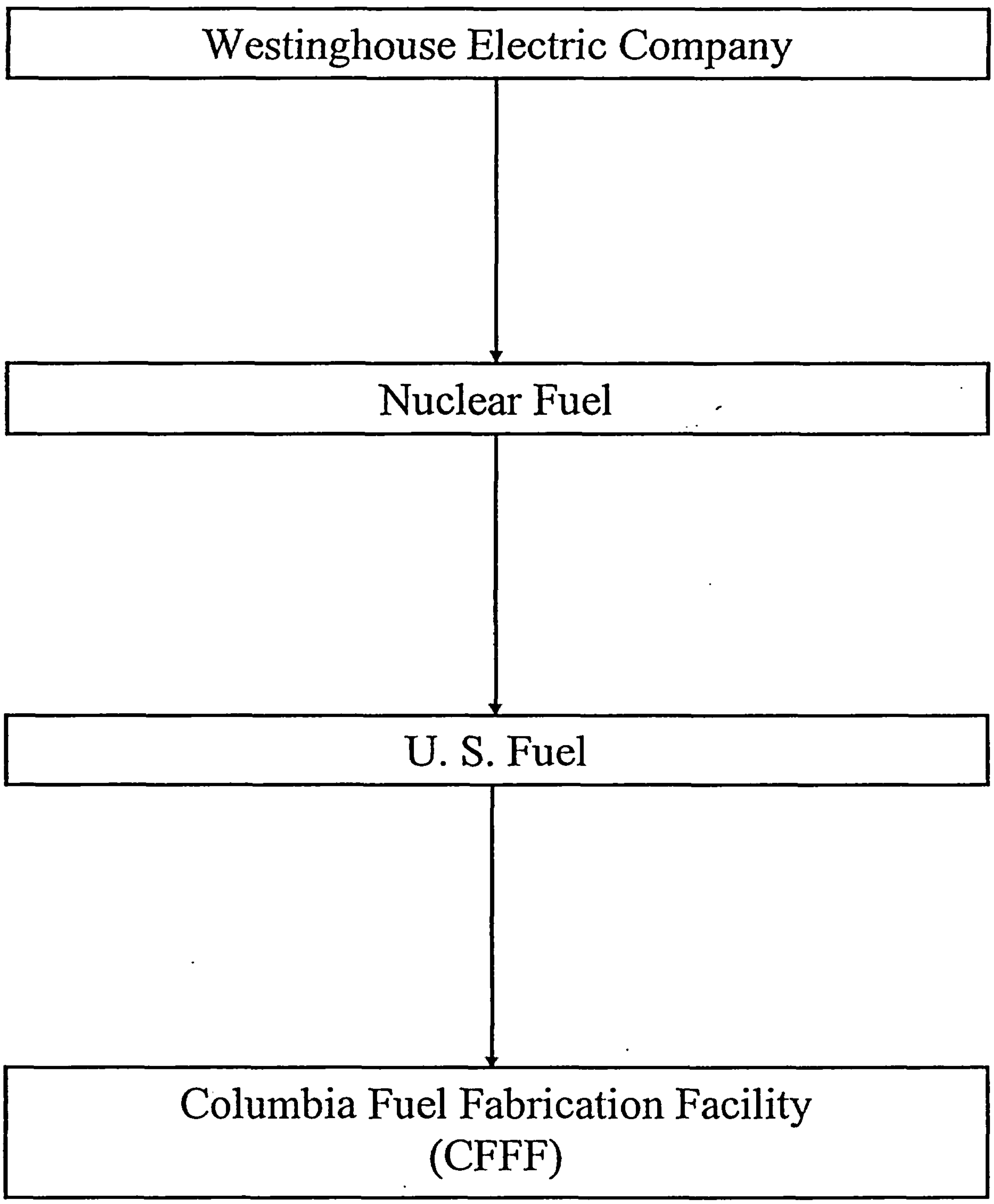
2.1.1.1 Organizational Operating Units

Within Nuclear Fuel, the primary responsibility for domestic fuel fabrication activities rests with U.S. Fuel. The Vice President of U.S. Fuel reports to the Senior Vice President of Nuclear Fuel. Within U.S. Fuel, the primary responsibility for fuel manufacturing operations rests with the Columbia Fuel Fabrication Facility (*aka* "CFFF"). The CFFF Plant Manager reports to the Vice President of U.S. Fuel. Figure 2.1 presents the Company Organization structure of Westinghouse.

2.1.1.2 Positions and Activities within Organizational Operating Units

Westinghouse management positions are covered by a written description, presenting the scope, duties, responsibilities and authorities for the position. Position descriptions are reviewed and approved by two higher levels of line management. These reviews determine that all key functions are covered, inter-relationships are clear, and conflicts are eliminated. Persons are selected to fill these management positions by evaluating their capability to perform the various activities specified in the position description.

Figure 2.1 Company Organization



Two higher levels of management, at minimum, must approve each selection or change of a management incumbent. Continuing quality performance of managers is assured through a formal program of annual reviews.

Operations at the CFFF are in accordance with the general operating philosophy and procedures that are employed in all Westinghouse plants and facilities. Basically, this philosophy provides that total responsibility for all phases of operations, including environmental protection, health, integrated safety, safeguards, and quality, follows the structured lines of organizational authority. Advisory and service groups are provided to assist line management in the evaluation of operations within their control; and, to provide measurements, determinations, and other information that aids in the analysis of specific operations and situations. However, such advice and service assistance in no way relieves an individual line manager from accountability for high quality operation of the function and facility or for ascertaining and assuring, through appropriate management channels, that adequate advice and service are being provided. Basic policies and procedures are established by line management with the review and approval of cognizant staff groups. Within the framework of these policies and procedures, the responsibility for making decisions at the operating level rests with the first level manager. A first level manager has the basic responsibility for operating controlled activities in a safe and compliant manner.

First level managers are responsible for ensuring that activities are conducted in accordance with operating instructions and for the guidance and direction of subordinate personnel. Written procedures, manuals, postings or other documents are prepared, which become the bases for performing specific operations. The first level manager cannot make unilateral changes in such documents without review and approval by cognizant staff groups. First level managers are also responsible for assuring that personnel under their jurisdiction receive adequate training.

The Regulatory Component participates in the orientation presentation for new employees. Fundamental radiation safety rules and policies, use of protective clothing and personnel monitoring devices, prevention of internal exposure, limiting external radiation exposure, nuclear criticality safety, and CFFF emergency procedures are among the topics covered. To acquaint a new employee with basic regulations, selected parts of Title 10, *Code of Federal Regulations*, are discussed. The cognizant first level manager assigns an experienced employee the responsibility for indoctrinating and training a new employee in the proper procedures and precautions for performing each specific job task. The first level manager then evaluates the progress of the new employee and gradually increases job assignments until complete requirements of the subject job description are fulfilled. Failure to achieve minimum performance requirements is cause for a change in assignment, or for release from employment. Periodic refresher training is conducted on-the-job by the employee's first level manager and/or by personnel from the Regulatory Component. As the need arises, changes in regulations, changes in operating conditions and/or practices, and changes in administrative policies are also covered.
To assure that all employees, who are not members of the emergency response organization, are aware of actions to take during an emergency situation, annual training

is provided. To keep emergency response personnel aware of the actions they must take during an emergency situation, emergency drills and exercises are conducted in alternate years. After each drill or exercise, appropriate plant personnel are informed of any shortcomings disclosed and subsequently instruct their personnel regarding any remedial actions required.

At the CFFF, all personnel involved in operations at the facility have the right and are actively encouraged to question and/or request a review of the safety or security of any operating task or procedure. All such concerns are investigated, assessed and resolved through the plant corrective action programs. Further, members of the Regulatory Component have the responsibility and authority to prohibit, through the cognizant first level manager, any situation that is believed to involve undue imminent hazard. Such terminated operations remain in a safe-shutdown state until the situation is reviewed with cognizant management, and there is a consensus resolution of the situation.

2.1.1.3 Position Accountability and Requirements

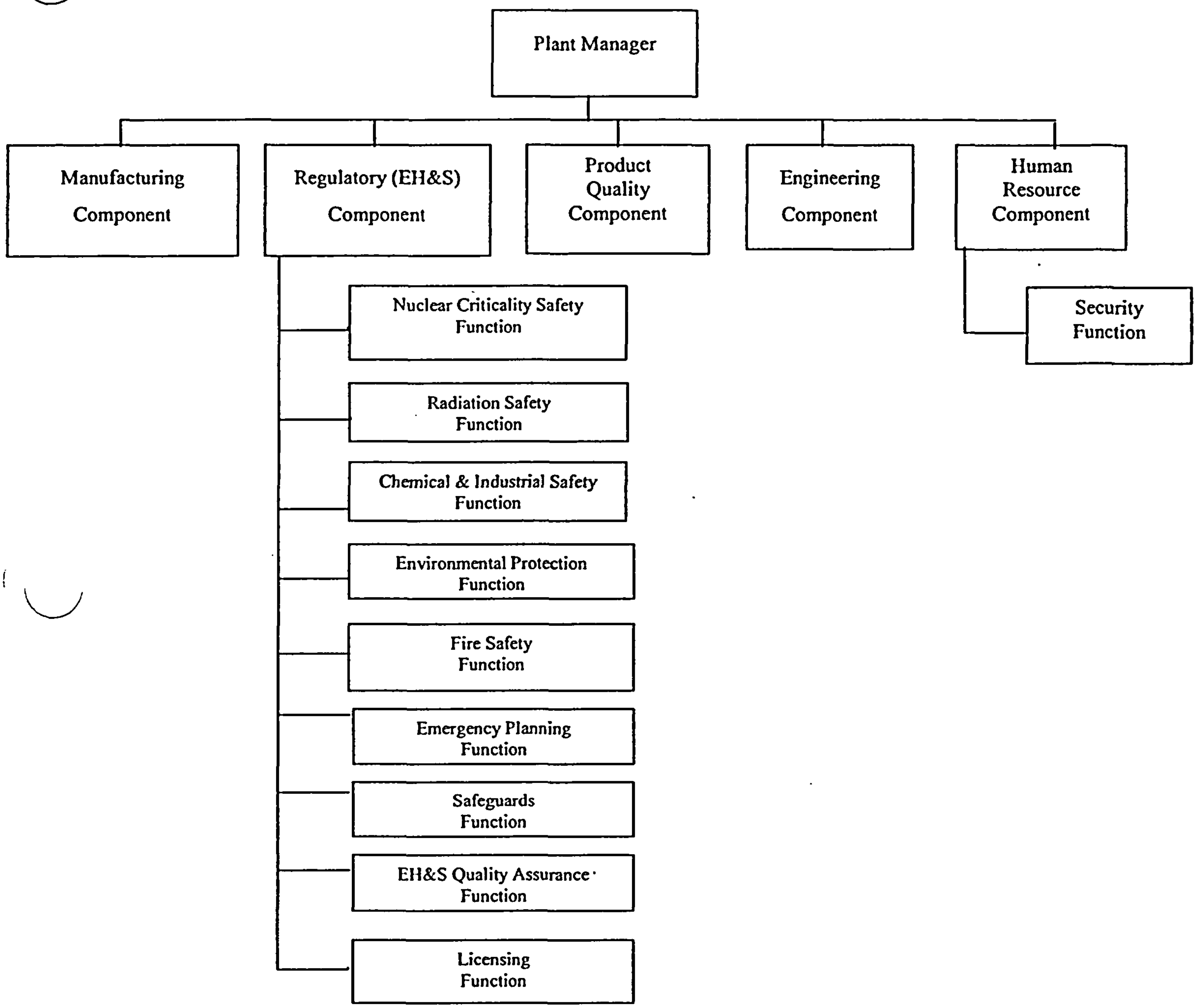
Administrative and managerial controls are in effect at all times to assure that decisions related to the operation of the CFFF are made at the designated level of accountability by individuals meeting the necessary authority and technical requirements. Figure 2.2 presents generic responsibilities within the CFFF organization structure.

(a) Plant Manager

The Plant Manager has overall accountability for all nuclear fuel manufacturing activities at the CFFF. This individual directs all activities of licensed operations and staff functions, either directly or through designated management personnel. This individual also coordinates any necessary support activities obtained from higher Westinghouse management and performs all assigned management activities in accordance with Westinghouse policies and higher management directives.

The minimum requirements for immediately assuming the position of CFFF Plant Manager are a baccalaureate degree or equivalent, five years of management experience in the nuclear business, and a broad general knowledge concerning the regulatory aspects of policies and procedures in effect at the CFFF. A Plant Manager-in-training that does not meet these minimum requirements formally designates an individual that does meet these requirements, to provide direct advice and consultation, until the minimum requirements are fully met. Typically, this designated advisor is the Senior Manager of the Regulatory Component.

Figure 2.2 CFFF Organization



(b) CFFF Managers

Component Managers (Senior Component Managers are typically called Plant Staff Managers, mid-level Component Managers are typically called Area Managers) have specific accountability for manufacturing, engineering, regulatory and product quality activities and operations involving licensed materials. To the extent practicable, the Regulatory Component is administratively independent of the Manufacturing, Engineering, and Quality Components. The Manufacturing Component conducts operations and maintenance activities required for the production of nuclear fuel. The Engineering Component provides technical support and design services related to processes and facilities used by the Manufacturing Component and others. The Regulatory Component is described below in paragraph (c) of this subsection. The Quality Component provides assurance, inspection, and analytical services in support of the Manufacturing Component and others. Component Managers plan, direct, and control such activities personally, or through subordinate management personnel; and, perform all assigned management duties in accordance with Westinghouse policy and higher management directives. A Component Manager might be responsible for more than a single work area; and, is directly accountable for the safe operation and control of activities in the work area(s). With appropriate support from cognizant service groups, Component Managers are responsible for environmental protection, health, integrated safety, quality, and safeguards in all areas over which they have authority.

First Level Managers (typically called Team Managers) normally supervise operations personnel. These Managers fulfill their responsibilities by assuring that all operations under their control are carried out in accordance with the radiation protection limits, nuclear criticality safety controls, processing procedures, schedules, and other instructions supplied by higher management.

All Component managers are knowledgeable in the operating procedures applicable to their work areas, including the application of the CFFF's safety programs, as they relate to controls and limitations on work activities, in assigned radiation and radioactive materials areas. Each manager of work areas where uranium is handled is knowledgeable in the application of the areas' nuclear criticality safety controls and other controls identified in the ISA. Managers are also knowledgeable in the occupational safety and health practices applicable to their areas of responsibility.

The minimum requirements for a position of Component Manager, is a baccalaureate degree, or equivalent, with a science or engineering emphasis; and, two years of experience in the nuclear business. A Component Manager-in-training that does not meet these minimum requirements has an individual, formally designated by the next highest level of management, to provide direct advice and consultation, until the minimum requirements are fully met. Typically, this designated advisor is an individual who formerly held the position, another Component Manager, or an individual (or individuals) experienced in the skills needed by the Component Manager-in-training.

The minimum requirements for a position of First Level Manager is a High School Diploma, or equivalent, and two years of experience in the nuclear business. A First Level Manager-in-training that does not meet these minimum requirements has an individual, formally designated by the next highest level of management, to provide direct advice and consultation, until the minimum requirements are fully met. Typically, this designated advisor is an individual who formerly held the position, another First Level Manager, or an individual (or individuals) experienced in the skills needed by the First Level Manager-in-training.

(c) Regulatory Component Managers and Engineering Functions

The Regulatory Component establishes requirements for environmental protection, radiation protection, nuclear criticality safety, occupational safety and health, emergency planning, and related licensed programs; and, for evaluating the effectiveness and compliance of these programs. The Regulatory Component is particularly responsible for assuring that these requirements have been evaluated and communicated to other Component management for incorporation into facilities, equipment, and procedures prior to their use for processing licensed material. Typical responsibilities of the Regulatory Component include:

- License and permit administration;
- Routine surveillance of operations;
- Inspection of licensed activities for compliance with applicable regulations, licenses and permits; and, documentation of these inspections, and actions to facilitate necessary corrective actions;
- Maintenance of CFFF regulatory plans;
- Maintenance of CFFF regulatory manuals;
- Maintenance of CFFF regulatory procedures;
- Conduct and maintenance of Integrated Safety Analyses;
- Review and approval of all CFFF procedures specifically related to environmental and radiation protection, nuclear criticality safety, occupational safety and health, and emergency planning;
- Review and approval of design drawings of equipment and layouts associated with the processing, handling and storage of licensed material;
- Verification of installed equipment for conformance to requirements for environmental and radiation protection, nuclear criticality safety, occupational safety and health, and emergency planning; and, for documentation of said conformance;
- Review of environmental and radiation protection, nuclear criticality safety, occupational safety and health, and emergency plan aspects of changes to equipment and operations associated with the processing, handling, and storage of licensed material;

- Training in, and monitoring the training effectiveness of, environmental and radiation protection, nuclear criticality safety, occupational safety and health, and emergency planning;
- Monitoring and reporting the effectiveness of the program for assuring that radioactivity, radiation, and hazardous material exposures are kept As Low As Reasonably Achievable (ALARA);
- Review and assessment of EH&S programs and performance; and,
- Review of regulatory violations and assurance of implementation of corrective actions.

The Regulatory Component is responsible for the establishment, conduct, and continuing evaluation of licensed activities to assure the protection of CFFF employees, the neighboring public, and the environment. In particular, for any processing change that could result in a credible consequence not previously evaluated, or in excess of one that that was previously evaluated, the Regulatory Component performs a safety analysis to assure that no off-site consequences, exceeding those specified by applicable regulations, could occur. Any process change for which the analysis indicates that a process upset could produce effects in excess of those previously evaluated is submitted for review and approval by appropriate NRC Staff, prior to implementation.

The Radiation Protection Program administered by the Regulatory Component includes, at minimum:

- The development of procedures to control contamination, exposure of individuals to radiation, and integrity and reliability of radiation detection instruments;
- The evaluation of radioactive effluents and material releases from the site;
- A robust subprogram for maintaining exposures to radiation and radioactive materials, and releases of radioactive materials to the environment, As Low As Reasonably Achievable (ALARA); and,
- The maintenance of required records and reports to document Radiation Protection Program activities.

The Nuclear Criticality Safety Program administered by the Regulatory Component includes, at minimum:

- The performance of process and equipment criticality safety evaluations before a new or modified fissile material operation is first operated;
- The determination of parametric controls and spacing requirements, based upon validated analytical or computational techniques, including computation of effective neutron multiplication factors for fissile material configurations;
- The conduct of audit and inspection services to assure operations are being conducted in accordance with approved nuclear criticality safety procedures and practices;
- The conduct of audits of the nuclear criticality safety program; and,

- The documentation and maintenance of process, equipment, and program reviews; of validated nuclear criticality safety evaluations; and, of operations equipment and procedure reviews, verifications, and approvals.

The Occupational Safety and Health Program administrated by the Regulatory Component includes, at minimum:

- The evaluation of potential physical, chemical, and fire hazards at the CFFF;
- The development and implementation of safety subprograms and procedures designed to minimize accidents and injury of employees;
- The procurement and maintenance of industrial safety protection and monitoring equipment;
- A robust subprogram for maintaining exposures to hazardous materials, and releases of hazardous materials to the environment, As Low As Reasonably Achievable (ALARA); and,
- The maintenance of required records and reports to document Occupational Safety and Health Program activities.

The minimum requirements for a position of a Regulatory Component Manager is a baccalaureate degree, or equivalent, with a science or engineering emphasis and two years of experience in assignments involving regulatory activities in the nuclear business. A Regulatory Component Manager-in-training that does not meet these minimum requirements has an individual, formally designated by the next highest level of management, to provide direct advice and consultation, until the minimum requirements, prescribed by an approved training checklist, are fully met. Typically, this designated advisor is an individual who formerly held the position, another Regulatory Component Manager, or an individual (or individuals) experienced in the skills needed by the Regulatory Component Manager-in-training. A Regulatory Component Manager has appropriate knowledge of health physics, nuclear criticality safety, and/or industrial safety and hygiene (typically demonstrated by completion of formal courses in one or more of the disciplines and/or by having prior work experience in one or more of the disciplines) and knowledgeable in administration of functional programs being managed.

The minimum requirements for a position of a Regulatory Function Engineer is a baccalaureate degree, or equivalent, with a science or engineering emphasis and two years of experience in positions involving assigned function activities, in the nuclear business. A Regulatory Function Engineer-in-training that does not meet these minimum requirements has an individual, formally designated by a Regulatory Manager, to provide direct advice and consultation until the minimum requirements prescribed by an approved training checklist are fully met. Typically, this designated advisor is an individual who formerly held the position, another Regulatory Function Engineer, or an individual (or individuals) experienced in the skills needed by the Regulatory Function Engineer-in-training. A Regulatory Function Engineer has knowledge in the quality execution of assigned function programs (typically demonstrated by formal performance reviews by a Regulatory Component Manager) and in administration of assigned functional programs.

2.1.1.4 Management of Organization Changes

Approved procedures are in place to assure that all relevant organizational changes within the Regulatory Component, and external to the Regulatory Component, are reviewed for impact on environmental and radiation protection, nuclear criticality safety, occupational safety and health, emergency preparedness, and other regulatory activities.

(a) It is the responsibility of each CFFF Component to submit all organizational changes involving managers and engineers, with assignments of regulatory importance, to the Regulatory Component so that the regulatory impact of the changes can be assessed. The assessment considers the structure of the organizational change; the capability and skills of personnel replacement(s); expectations of, and responsibilities of, the position(s); and any resultant changes to organizational responsibilities.

(b) It is the responsibility of the Regulatory Component to assess all Regulatory Component organizational changes so that the regulatory impact of the changes can be determined. The assessment considers the structure of the organizational change, as well as the capabilities and skills of the personnel involved.

(c) Organizational changes external to the Regulatory Component, involving personnel other than managers and engineers, are submitted to the Regulatory Component for assessment only if the responsible manager determines that environmental and radiation protection, nuclear criticality safety, occupational safety and health, emergency preparedness, and/or other regulatory activities could be impacted.

(d) Assessment considerations include both normal and off-normal operations (and any transitional phases), and potential for cumulative effects of organizational changes, as appropriate.

(e) The extent and detail of an assessment are commensurate with the level of risk for an adverse impact on regulatory activities determined for the organizational change.

(f) Like changes of personnel, and organizational changes that are built into a documented plan (*e.g.*, a Program Plan that prescribes reductions in manpower as assignments are completed) are outside the scope of these assessments.

(g) If a significant risk of an adverse impact on regulatory performance is identified, an organizational change is closely monitored using the Corrective Active Process, described in Section 3.8 of this License Application, until the risk is resolved.

(h) Organizational changes are reviewed prior to implementation, whenever practicable.

CHAPTER 3.0

CONDUCT OF OPERATIONS

Conduct of operations embraces the management measures that are implemented on a continuing basis to reasonably assure that Columbia Fuel Fabrication Facility (CFFF) activities for protection of the environment, health and safety of employees and the neighboring public are conducted to a high standard of quality. In particular, these management measures are applied to Safety Significant Controls (SSCs) to provide reasonable assurance that items relied on for safety are available and able to perform their functions when needed.

3.1 CONFIGURATION MANAGEMENT (CM)

To assure that facility or equipment design changes, and/or computer software modifications, do not have an adverse impact on environmental protection, health, safety, and/or safeguards programs at the CFFF, a formal review process has been established to analyze new structures, systems, and components, or modifications to existing structures, systems, and components, in order to reliably predict performance under normal operating conditions and potential process upsets. Structured safety analyses, conducted in accordance with the requirements of Chapter 4.0, specifically include examination of verified drawings and software (as applicable) under configuration management. Configuration management is a management measure controlled by the quality program described in Section 3.3 of this License Application. Periodic assessments are conducted to determine the program'S effectiveness and to correct any deficiencies.

3.1.1 CM Program Structure

The CFFF CM program is implemented in accordance with approved procedures for change management. These procedures define the review and approval processes for assuring that impacted structures, systems and components will continue to meet regulatory specification requirements. The procedures also specify the documentation required to maintain a current record of as built configurations.

3.1.1.1 One such procedure is an Engineering Component document that details CFFF configuration control. The purpose of this procedure is:

(a) To specify the process for implementation of proposed changes to all Plant manufacturing and inspection systems, facilities and utilities;

(b) To identify documentation requirements for maintaining records of current Plant conditions, and

(c) To define the review and approval processes necessary to ensure that specification requirements for manufacturing and inspection functions in a manner that:

- Is safe;
- Complies with applicable requirements, and
- Appropriately incorporates As Low As Reasonably Achievable (ALARA) considerations.

3.1.1.2 Another such procedure is a Regulatory Component document that details regulatory review of configuration change authorizations. The purpose of this procedure is:

(a) To establish an integrated process for providing the environmental protection, radiation safety, criticality safety, safeguards, chemical safety, and fire safety criteria associated with proposed modifications of, or additions to, existing hazardous material handling or storage systems, hazardous equipment, uranium processing systems, and ancillary facilities and operations.

(b) To assure all such modifications or additions implement the ALARA concept to minimize occupational radiation exposures and exposures to members of the public.

3.1.1.3 Another such procedure is a Product Assurance Component document that details computer software quality assurance. One purpose of this procedure is:

(a) To establish a process for ensuring that computer software that affects integrated safety or safeguards is appropriately qualified or verified before its application.

3.1.2 CM Program Implementation

3.1.2.1 The CM program is designed and implemented as an ancillary management measure in support of the facility's ISA such that it becomes an integral part of routine CFFF operations.

3.1.2.2 The following sequence of activities is used for all facility addition and/or change projects. The complexity of the project, and the issues involved determine the magnitude of effort afforded to each activity.

(a) Introduction - - An assigned project engineer provides information such as the project description and justification for the project.

(b) Documentation Updates - - Documentation (including drawings, procedures, and software) needing to be updated because of the project is compiled. For changes involving criticality safety, a new or revised criticality safety evaluation must be completed for the impacted process.

(c) Project Approvals - - Permission (signature and date) to proceed with the project is given by cognizant individuals (engineers and managers) from the Engineering and Manufacturing Components.

(d) Regulatory Reviews and Approvals - - The Regulatory Component then links the project to the appropriate ISA document identification. Cognizant engineers from the Regulatory Component (each safety and safeguards discipline) determine the need for their respective reviews of the project, perform such reviews if necessary, and approve the project subject to the results of their reviews. After completion of any action items, analyses, etc., Regulatory management then approves (signature and date) the project for startup.

(e) Project Close-Out - - After implementation of the project, Regulatory management's signature and date are required to approve project completion.

(f) Information provided by steps (a) through (e) is documented on a Configuration Change Control Form.

3.1.2.3 As described in Subsection 4.1.2.2 of this License Application, the Configuration Change Control Form and supporting information are filed with the applicable Baseline ISA, thus providing a substantially complete "living" framework for the facility safety basis.

3.2 MAINTENANCE

To keep safety-related systems and components at the Columbia Fuel Fabrication Facility (CFFF) in a condition of readiness such that they are likely to perform their desired function when called upon to do so, a maintenance program is implemented in accordance with approved procedures. Maintenance is a management measure controlled by the quality program described in Section 3.3 of this License Application.

3.2.1 Maintenance Program Structure

A basic purpose of the maintenance program is to ensure that Safety Significant Controls (SSCs), determined by the Integrated Safety Analysis (ISA) described in Chapter 4.0 of this License Application, are installed, tested, modified, and maintained in accordance with approved procedures. The ISA details the maintenance requirements.

The ISA also includes a table of SSCs that lists identification numbers for Operator Maintenance (OM) procedures (for maintenance activities performed by Operations Functions) and Preventive Maintenance (PM) procedures (for maintenance activities performed by Maintenance Functions). The table also lists applicable requirements for periodic maintenance, calibration, inspection, periodic functional testing, and post-repair/replacement testing. A portion of a typical table is presented in Figure 3.1.

The ISA Summary includes a table of SSCs that have been designated as Items Relied on for Safety (IROFS). This table lists the same maintenance information as the ISA table described above.

3.2.2 Maintenance Program Implementation

The basic element for implementation of the CFFF maintenance program is a computerized maintenance planning and control system. This system contains different modules that execute actions of interest to safety and safeguards.

3.2.2.1 The Equipment Module covers the following:

(a) Each piece of equipment is assigned a unique equipment number (record).

(b) All preventive maintenance for the equipment can be found under the equipment record.

(c) Equipment records of regulatory significance are assigned a special identification that will print out with all work orders written against the record.

Figure 3.1 Typical ISA Safety Significant Control Table

UN BULK STORAGE SYSTEM						
Control ID (P&ID Tag Number, if applicable)	•CONTROL • INITIATING EVENT ACCIDENT SEQUENCE PREVENTION/MITIGATION (Include parameter "Trigger Levels", if applicable) • ACTION EXPECTED	OP / OM / PM NUMBER(S)	PERIODIC MAINTENANCE/ CALIBRATION/ INSPECTION REQUIRED (YES,NO,N/A)	PERIODIC FUNCTIONAL TEST REQUIRED (YES,NO)	POST REPAIR/ REPLACE TEST REQUIRED (YES,NO,N/A)	SAFETY DISCIPLINE
PASSIVE ENGINEERED CONTROLS						
UN-301	• Guard Posts • Prevent vehicles from damaging tank or dike • Stop vehicles	OP N/A OM85018 PM N/A	YES	NO	NO	Env Prot
UN-501	• Flange Guards (safety shields) • Prevent personnel exposure to chemicals • Operator ensures that flange guards (safety shields) are functioning	COP-835512 OM85004 PM N/A	YES	NO	NO	Chem Safety
UN-502	• Tank wall • Prevent UN spills • Contain UN inside UN Storage Tanks	OP N/A OM N/A PM 85151	YES	NO	NO	Chem Safety
UN-901	• Diked pad • Prevent personnel exposure to UN and prevent UN discharge to environment • Contain UN spills	OP N/A OM85018 PM N/A	YES	NO	NO	Env Prot Chem Safety
ACTIVE ENGINEERED CONTROLS (e.g., INTERLOCKS)						
UN-101-1 (RT-736-A) UN-101-2 (RT-746-A)	• C4 Dissolver gamma monitors • Prevent discharge of UN with >5 gU235/l to UN Storage Tanks • Close Product Hold Tank discharge valve to UN Storage Tank and open Product Hold Tank recirculation valve	OP N/A MCP-202030 OM85017 PM N/A	YES	YES	YES	Crit Safety

3.2.2.2 **The Maintenance Module covers the following:**

(a) Work Orders for equipment maintenance are written in this module.

(b) PM procedures are written and maintained in this module. These are step-by-step instructions for performing preventive maintenance at an assigned frequency.

(c) A PM procedure that contains steps involving a SSC is designated "type SS", and requires approval by the Regulatory Component.

3.2.2.3 **The Inventory Module covers the following:**

(a) Each spare part in the storeroom has a unique record number.

(b) Each part has a minimum quantity assigned to maintain proper inventory levels.

(c) A part's use history can be viewed from this module to see usage trends.

(d) Parts of regulatory significance are assigned a special identification so they can undergo a required in-house receipt inspection prior to being released for storage and/or use.

3.2.2.4 **The Purchasing Module covers the following:**

(a) This module is used for ordering both stocked and non-stocked parts.

(b) A purchase requisition is created to be used to write a purchase order. (The purchase order includes any applicable Regulatory Component quality requirements, as described in Section 3.3 of this License Application.)

3.2.2.5 **The Calibration Module covers the following:**

(a) This module contains the data cards for all instruments that require calibration on a given frequency.

(b) Calibrations are performed pursuant to a "coming due" report that is generated from a "reports" menu in this module.

(c) The system maintains the next due date for a given instrument based on the required frequency on the applicable data card.

3.3 QUALITY ASSURANCE

Implementation of Regulatory Component Quality Assurance (QA) Program at uranium-processing fuel cycle plants is not explicitly required by regulation. However, Columbia Fuel Fabrication Facility (CFFF) management believes that some level of QA, appropriate to the type and magnitude of specific operations conducted at the CFFF and consistent with the degree of risk posed by these operations to workers, the public, and the environment (i.e., a "graded approach"), must be applied to all activities important to safety, safeguards, and protection of the environment.

The Westinghouse CFFF, Regulatory Component Quality Program/Policy Manual (QP Manual) describes management's commitment to the application of QA principles and criteria described in the *American National Standard Quality Assurance Program Requirements for Nuclear Facilities, ASME NQA-1* (NQA-1). The 18 basic requirements of NQA-1 have been applied in full to specified CFFF nuclear operations but are only intended to be selective for other applications, such as CFFF Regulatory Component. As stated in the NQA-1 forward, "The extent to which this document should be applied, either wholly or in part, will depend upon the nature and scope of the work to be performed and the relative importance (to safety, safeguards, and protection of the environment) of the items or services being produced. The extent of application is to be determined by the organization imposing this document."

The QP Manual is also consistent with CFFF management's commitment to mandatory application of principles and criteria described in its company policies and the *Westinghouse Electric Company Quality Management System* (QMS). Quality assurance criteria in 10CFR50 Appendix B, and in 10CFR71, apply to nuclear power reactors and suppliers of components and services for these reactors. The CFFF supplies both components (e.g., fuel assemblies) and means for transporting these components (i.e., shipping containers). The QMS describes how quality assurance is applied to these components and services.

The CFFF is licensed to possess and use special nuclear material (SNM) in the production of fuel assemblies. In this 10CFR70 environment, safety significant controls (SSCs), and certain instruments and services, are treated similarly to reactor basic components in the 10CFR50/71 environments. A major similarity is that quality assurance must be provided to ensure that SSCs, determined by the Integrated Safety Analysis (ISA) described in Chapter 4.0 of this License Application, are designed, installed, tested, modified, and maintained in accordance with approved procedures to guarantee their availability and reliability This is a basic purpose of the QP Manual.

The QP Manual applies to activities that affect the quality of items specified by and services supplied to the Regulatory Component. It defines the basic requirements applicable to such items and services that serve to protect workers, the public, and the environment. It serves as a directive for all EH&S Functions in establishing individual work instructions and implementing procedures. Additional Regulatory Component quality requirements, which supplement the QP Manual, may be developed to document and/or clarify specific quality commitments (e.g., the Regulatory Policy on *Application of Quality Assurance Program Criteria to Safety-Significant Controls*).

The CFFF Regulatory Component Component conforms to format and content of Standard Review Plans and other Regulatory Guidance at the discretion of CFFF management. However, Standard Review Plans and other Regulatory Guidance are not substitutes for regulations, and compliance with them is not required. Format and content different from those set out in Standard Review Plans and other Regulatory Guidance is acceptable if they provide an equivalent basis for the findings requisite to regulator actions.

3.3.1 QA Program Structure

The Regulatory Component quality program is structured to address the aforementioned QA criteria, namely:

(a) Organization;
(b) Regulatory Component Quality and Training Programs;
(c) Design Control;
(d) Procurement Document Control;
(e) Policies, Procedures, and Drawings;
(f) Document Control;
(g) Control of Purchased Material, Equipment and Services;
(h) Identification and Control of Materials, Parts and Components;
(i) Control of Special Processes;
(j) Inspection;
(k) Test Control;
(l) Control of Measuring and Test Equipment;
(m) Shipping/Receiving, Handling and Storage;
(n) Inspection, Test and Operating Status;
(o) Nonconforming Materials, Parts or Components;
(p) Corrective Action;
(q) EH&S Records;
(r) Audits and Compliance Inspections.

3.3.2 Graded Approach For Safety Systems

The "graded approach" to quality assurance is addressed as a part of performing a systematic ISA of hazards at the facility, including identification of the SSCs that are intended to prevent and/or mitigate the consequences of these hazards, as follows:

3.3.2.1 Quality Level A (A-6/A-5); High Consequence Systems ("Crucial")

These systems are crucial to safety and, therefore, receive rigorous attention to installation, operation and maintenance. They are defined by controlling the following performance indicators:

(a) Greater than or equal to 100 rem dose equivalent to a worker;
(b) Greater than or equal to ERPG3 chemical exposure to a worker;
(c) Greater than or equal to 25 rem dose equivalent to the offsite public;
(d) Greater than or equal to 30 milligrams soluble intake by the offsite public;
(e) Greater than or equal to ERPG2 chemical exposure to the offsite public.

Crucial safety systems receive full application of the QA program requirements to assure failure of their availability and reliability is highly unlikely. That is, each of the 18 criteria that could apply are specifically addressed.

3.3.2.2 Quality Levels B (B-4) and Safety Significant C (Css); Intermediate Consequence Systems ("Important")

These systems are important to safety and, therefore, receive appropriate attention to installation, operation and maintenance. They are defined by controlling the following performance indicators:

(a) Greater than or equal to 25 rem dose equivalent to a worker;
(b) Greater than or equal to ERPG2 chemical exposure to a worker;
(c) Greater than or equal to 5 rem dose equivalent to the offsite public;
(d) Greater than or equal to ERPG1 chemical exposure to the offsite public;
(e) Greater than or equal to 5000 times Table 2 Appendix B, 10CFR20 radioactivity release to the environment
(f) Loss of Nuclear Criticality Safety Double Contingency Protection.

Important safety systems receive selective application of the QA program requirements to assure failure of their availability and reliability is unlikely. That is, only the 18 criteria that the Regulatory Component determines should apply are specifically addressed.

3.3.2.3 Quality Level C; Safety Margin Improvement Systems

These systems have safety implications, but are neither crucial nor important (as defined above) to safety. They do not require specific application of quality assurance, and no extraordinary safety detail is applied. Safety margin improvement systems are installed, operated, and maintained in accordance with prudent industry practice.

3.3.3 QA Program Implementation

3.3.3.1 The program is designed and implemented such that it becomes an integral part of routine CFFF operations.

3.3.3.2 The program is performance based. That is, quality assurance decisions are based, to the extent practicable, on safety system performance histories.

3.3.3.3 The program's description is documented in a manual that specifies authority, responsibility and accountability for all program elements.

3.3.3.4 Program elements are conducted in accordance with approved, written procedures; training to these procedures is conducted to ensure the program operates effectively.

3.3.3.5 The program requires documented records to demonstrate conformance to program requirements.

3.3.3.6 The program includes checks and balances through appropriate functional separations and audits; however, routine quality assurance for safety systems may be performed by the functions responsible for operating the systems (i.e., quality-at-the-source).

3.3.3.7 The program embraces issue identification, remedial actions, and management control elements to ensure that deficiencies, deviations, and defective equipment and services are disclosed and corrected in a timely manner through utilization of the CFFF Corrective Action Program (CAPs).

3.3.3.8 Full implementation of the program is a forward-fitting process. That is, full implementation of the program does not becomes effective until the ISA for the affected area is complete, and the ISA Summary has been approved by NRC Staff. It is a bounding assumption that existing systems have been installed, operated and maintained in accordance with applicable requirements and accepted practices. Such systems are not back-fit except for system upgrade modification and/or actions arising from internal evaluations or external disclosures (such as NRC Information Notices, etc.). Such back-fitting is at the discretion of CFFF Management, as advised by the Regulatory Component.

3.4 PROCEDURES, TRAINING AND QUALIFICATION

At the Columbia Fuel Fabrication Facility (CFFF), procedures, training and qualification are integrated into a combined process to assure that safety and safeguards activities are being conducted by trained and qualified individuals, in accordance with Westinghouse policies and in accordance with commitments to Regulatory Agencies. Elements of this integrated process are developed by subject matter experts (SMEs), are reviewed and approved by cognizant individuals in affected Components, and are authorized for implementation by Component Management at a level that is responsible and accountable for the operations covered. Procedures, training, and qualification are management measures controlled by the quality program described in Section 3.3 of this License Application.

3.4.1 Procedure Structure

Operations to assure safe, compliant activities involving nuclear material are conducted in accordance with approved procedures. Approved procedures are maintained and controlled by an electronic training and procedures system. Approved procedures provide a basis for training of all personnel involved in operations with nuclear material at the facility.

Process Hazard Analyses (PHAs), conducted as described in Chapter 4.0 of this License Application, include reviews of applicable procedures. Administrative Safety Significant Controls (SSCs) are detailed in approved procedures.

3.4.1.1 Regulatory-Significant Procedure Structure

CFFF procedures are classified into three general categories:

(a) Category-1 Procedures

Category-1 procedures are for use by a Regulatory Component. Use of such procedures is to provide integrated safety and safeguards training and instructions for Regulatory Functions. They are prepared and approved for issuing by Regulatory Functions assigned by a Regulatory Component Manager; and, they are reviewed and approved for issuing by the Regulatory Component Manager, or an assigned designee.

Examples of Category-1 procedures subcategories include;

- Administration;
- Health Physics;
- Nuclear Criticality Safety;
- Environmental Protection;
- Safeguards;
- Instruments;
- Surveys;
- Dosimetry;
- Bioassay; and,
- Laboratory Practices

(b) Category-2 Procedures

Category-2 procedures are for use by individuals outside the Regulatory Component, and deal exclusively with regulatory practices. These procedures provide integrated safety and safeguards training and instructions for Engineering, Manufacturing, Quality and other Functions. They are used by these Functions in preparing Category-3 procedures. Category-2 procedures present regulatory guidance methodology acceptable to the Regulatory Component. They are prepared and approved for issuing by Regulatory Functions assigned by a Regulatory Component Manager, and they are reviewed and approved for issuing by the Regulatory Component Manager, or an assigned designee.

The Category-2 scope is similar to, and in many cases overlaps, that for Category-1, as applicable to use outside the Regulatory Component.

(c) Category-3 Procedures

Category-3 procedures are for use by individuals outside the Regulatory Component. These procedures provide training and instructions, including integrated safety and safeguards, for Engineering, Manufacturing, Quality and other Functions. They are prepared and approved for issuing by Component Functions assigned by a cognizant Component Manager, based on guidance from applicable Category-2 procedures and/or consultation with the Regulatory Functions. Category-3 procedures are reviewed and approved for issuing by the cognizant Component Manager, or an assigned designee. Category-3 procedures that include SSCs are reviewed and approved by the appropriate Regulatory Functions.

The Category-3 scope is determined by the cognizant Component Manager.

3.4.1.2 Issuance, Approval, and Communication of Procedure Content

Acceptable practices for integrated safety and safeguards activities are provided to Operations Components in procedures that are approved for electronic issue by the Regulatory Component. The content of these procedures is communicated to operations personnel by cognizant Component Management through incorporation into appropriate operating and/or quality procedures.

Regulatory-significant practices in operating and quality procedures, and changes to such practices, are approved for issuing by cognizant Components in accordance with documented instructions for procedure preparation, review and approval. Regulatory Component approvals are required for all aspects of procedures, and changes to such procedures, that direct the storage, handling, processing, inspection and/or other activities involving nuclear materials. Component Management is responsible for assuring and documenting that the content of these procedures is communicated to appropriate personnel through training, access to the electronic training and procedure system, and/or posting of instructions.

3.4.1.3 Procedure Review Frequencies

Maximum frequencies for technical reviews of regulatory-significant procedures are:

(a) Annual - - for Category-1 and Category-2 procedures, and

(b) Biennial - - for Category-3 procedures.

3.4.1.4 Procedure Use and Adherence

Compliance with approved procedures is a mandatory condition of continuing employment at the CFFF. The CFFF Employee Handbook specifically states that intentional violation of operating procedures is considered extremely serious and may result in immediate discharge. All employees are instructed to immediately stop any work activity that is not specifically covered by an approved procedure.

The formal internal reporting system described in Section 3.7 and the Corrective Action Process described in Section 3.8 of this License Application provide the means for employees to report inadequate procedures and/or the inability to follow procedures to their First Level Managers for corrective action.

First Level Managers enable and require compliance with all regulatory-significant procedures. This is accomplished by providing ready employee access to procedures, requiring documented procedure review and acknowledgement, and then evaluating employee performance with respect to procedure compliance on a continuing basis. Employees receive additional procedure training if determined necessary by the First Level Manager evaluations.

3.4.2 Training and Qualification Structure

Training is provided for everyone who works at the CFFF, commensurate with his or her duties. Formal training programs are developed and conducted as necessary to implement the training responsibilities described in Chapter 2.0 of this License Application and to enable procedure use and adherence. Such training programs are performance-based and, as such, address elements of job and task analyses, learning objectives, instructional methodologies, implementation, evaluation and feedback. The programs are structured such that specific training and qualification requirements are met prior to regulatory-significant positions being fully assumed or covered tasks being independently performed. Training and qualification records are maintained in accordance with Section 3.9 of this License Application.

3.4.2.1 General, Topical and Refresher Training

All new employees receive training in regulatory policies, general safety and safeguards practices, and emergency response. All new employees designated as radiation workers receive additional training relative to regulatory aspects concerning radiation and radioactive materials, risks involved in receiving low level radiation exposure, basic criteria and practices for radiation protection, maintaining radiation exposures and radioactivity in effluents As Low As Reasonably Achievable (ALARA), nuclear criticality safety, chemical and fire safety, and nuclear material safeguards. Facility visitors are provided with training commensurate with their visit's scope, and/or are escorted by trained employees.

Employees or visitors for whom respiratory protection devices might be required, receive pre-work training in the proper use of such devices. Employees designated to take part in emergency response receive training commensurate with their assigned activities during such response.

Radiation workers receive annual refresher training consisting on general regulatory topics. This training requires each employee to successfully pass an examination. The training addresses such subjects as:

(a) ALARA;

(b) General health physics practices;

(c) Health Physics rules and recommendations;

(d) Area-specific health physics practices;

(e) General nuclear criticality safety practices;

(f) Area-specific nuclear criticality safety practices;

(g) Industrial safety and hygiene, practices;

(h) Chemical Area work practices;

(i) Radiation risks;

(j) Fire safety practices;

(k) Emergency planning, and

(l) Safeguards.

Employees who are absent from the facility during scheduled regulatory refresher training receive such training within one month of their return to work.

3.4.2.2 Training and Qualification of Regulatory Function Engineers

In addition to the general, topical and refresher training requirements previously described, all Regulatory Function engineers receive training and documented qualification specific to their regulatory activities. This includes engineers who are subsequently assigned new or additional responsibilities. The purpose of the training is to enable the engineers to develop skills and abilities directed by the engineer's manager, who evaluates fundamental development opportunities on a case-by-case basis.

Upon assignment to the Regulatory Component, all engineers are assigned a peer trainer. The peer trainer is an experienced individual, assigned by management, to mentor and assist the new individual with both technical and non-technical indoctrination and training.

The engineer's manager acknowledges completion of the training program by documenting that the trainee is qualified to independently perform specific activities. Unqualified engineers cannot approve regulatory documents unless the document is co-signed by a qualified individual who takes responsibility for the document.

The engineer's manager also formally evaluates continuing performance of skills and abilities. Such evaluation may include:

(a) Reports of internal audits and compliance inspections conducted by the engineer;

(b) Feedback from training programs presented by the engineer, and/or

(c) Results of safety analyses and regulatory evaluations performed by the engineer.

Indoctrination, training and qualification of Regulatory Function engineers are performed in accordance with an approved procedure. This procedure provides specific processes to be performed and references checklists to be used.

3.4.2.3 Training and Qualification of Regulatory Operations Technicians

In addition to the general, topical and refresher training requirements previously described, all Regulatory Operations technicians receive training and documented qualification specific to their regulatory activities. Activities evaluated, on a case-by-case basis, by the technician's Manager may include:

- Developed skills and abilities;
- Applicable competency training;
- Documented acknowledgement of approved procedures, and/or
- Emergency preparedness training.

3.4.2.4 Non-CFFF Worker Risk Training

In addition to the general and topical training requirements previously described, all individuals who are non-CFFF workers performing ongoing activities in the CFFF controlled area are apprised of the risks associated with accidents involving nuclear material. This information is largely taken from applicable portions of the CFFF ISA documents.

3.5 HUMAN FACTORS

Human Factors concepts are employed at the Columbia Fuel Fabrication Facility (CFFF), in recognition of how the total job environment - - structures, equipment, training, and procedures - - shapes the expectations, thoughts, and decisions of employees who work with nuclear materials. The basis of Human Factors at the CFFF is the integrated Behavioral Safety & Human Performance Program. Human Factors is a management measure controlled by the quality program described in Section 3.3 of this License Application.

3.5.1 Behavioral Safety & Human Performance Program Structure

3.5.1.1 Behavioral Safety Process

Behavioral Safety is designed to influence the behavior of employees before accidents or incidents have an opportunity to occur. The goal of the process is everyone working together to eliminate accidents and incidents.

Observing workers is the key to Behavioral Safety. This involves trained observers looking for safe and unsafe behaviors by employees then giving immediate reinforcement or corrective feedback to the worker being observed.

3.5.1.2 Human Performance Process

Human Performance is based on an INPO model that provides a proven methodology to promote behaviors throughout the organization that support safe and reliable operations. The principles of Human Performance include:

(a) Humans are fallible.
(b) Error is predictable.
(c) Organization influences behavior.
(d) Behaviors are reinforced.
(e) Events are avoidable.

Human Performance tools are used to recognize error likely situations and prevent events from occurring. These tools include but are not limited to:

(a) Questioning attitude;

(b) Self check;

(c) Peer check;

(d) Pre-job briefing;

(e) Time out;

(f) Procedure use and adherence; and

(g) Personal safety assessment.

3.5.2 Behavioral Safety & Human Performance Program Implementation

CFFF employees are trained in Behavioral Safety & Human Performance concepts, commensurate with the level of their participation in the program. Trained observers then conduct systematic, documented observations that focus on high-risk or error-likely processes. Observations are documented and reviewed by appropriate personnel.

3.6 Compliance Inspections, Program Audits and Self-Assessments

Compliance inspections, program audits and self-assessments are conducted to assure that CFFF operations important to environmental protection, health, safety, and safeguards are properly documented, are conducted in accordance with such documentation and meet management expectations with respect to effectiveness. Compliance, inspections, audits, and self-assessments are integrated activities intended to self-identify and self correct issues such as process upsets and procedural inadequacies. Compliance inspections, program audits and self-assessments are management measures controlled by the quality program described in Section 3.3 of this License Application.

3.6.1 Compliance Inspections, Program Audits, and Self-Assessments Program Structure

3.6.1.1 Compliance Inspections

Compliance inspections are performed to assure that observed practices conform to approved implementation documentation (*e.g.*, procedures, handbooks, plans, *etc.*). Compliance inspections are normally conducted in work areas. Compliance inspections answer the question "is work being performed in accordance with approved implementation documentation?".

3.6.1.2 Program Audits

Program audits are performed to assure that intended work practices are properly reflected in approved implementation documentation (*e.g.*, procedures, handbooks, plans, *etc.*) and to objectively assess details of the effectiveness and proper implementation of regulatory programs (*e.g.*, Radiation Safety, Nuclear Criticality Safety, Chemical Safety, Fire Safety, Emergency Management, Environmental Protection, Safeguards, *etc.*). Program audits are normally conducted in administrative areas. Program audits answer the question "does approved implementation documentation properly reflect how work is being performed and does it meet requirements?"

3.6.1.3 Self-Assessments

Self-assessments are an evaluation of regulatory programs (or other areas of management interest), conducted by trained internal auditors and other individuals who are knowledgeable and/or experienced in the selected assessment subject, to provide management with an objective overview of the efficiency and effectiveness of specified regulatory activities. Self-assessments are normally conducted both in work areas and administrative areas. Self-assessments answer the question "are regulatory activities being effectively conducted in accordance with management expectations?".

3.6.2 Compliance Inspections, Program Audits, and Self-Assessments Program Implementation

3.6.2.1 Compliance Inspections

Compliance inspections are of two types:

(a) Informal inspections

Regulatory Component personnel conduct informal inspections of regulatory-significant performance in the course of their routine duties in CFFF work areas. Observed process upsets and procedural inadequacies are promptly reported to the cognizant First Level Manager for remedial action. Repeated upsets and inadequacies are reported to Regulatory management for subsequent reporting to increasingly higher levels of cognizant Management until effective remedial action has been taken. Such repeated upsets and inadequacies are documented in reports of relevant formal inspections (described below) to assure appropriate tracking and resolution.

(b) Formal inspections

An annual formal compliance inspection schedule is planned, documented, revised (as necessary), and implemented. Assigned Regulatory Component personnel conduct the formal inspections of regulatory-significant performance on a specified frequency.

Formal compliance inspection results are documented and reported to management having responsibility for the area being inspected. Inspectors schedule and conduct appropriate follow-up activities to ensure corrective actions were implemented effectively. These follow-up activities are also reported to applicable management.

3.6.2.2 Program Audits

An annual program audit schedule is planned, documented, revised (as necessary), and implemented. Assigned EH&S personnel, and/or external auditors selected by Regulatory Component Management, conduct the audits in accordance with an approved procedure and a pre-established checklist. A typical Program Audit Checklist is presented in Figure 3.3. Program Audits are led by appropriately qualified and certified individuals, and audit team membership may include personnel who have technical understanding of the programs being audited.

Audit results are documented and reported to management having responsibility for the program being audited. Auditors schedule and conduct appropriate follow-up activities to ensure corrective actions were implemented effectively.

These follow-up activities are also reported to applicable management.

3.6.2.3 Self-Assessments

Self-assessments are reviews conducted by individuals (within or external to the CFFF). The Assessment Team Leader, designated by Regulatory Component Management, selects a team of knowledgeable personnel who are not directly responsible for the portions of the CFFF regulatory activities they are to assess. These assessments are performed in accordance with an approved procedure. Results of these assessments are reported to management for disposition.

Another aspect of self-assessments is a summary evaluation of regulatory performance against a set of facility performance indicators or to meet regulatory requirements:

(a) Items documented in the formal program audit described in Section 3.6.2.2 of this Chapter;
(b) Process upsets and procedural inadequacies documented in formal compliance inspections described in Section 3.6.2.1 (b)
(c) CFFF Collective Dose Equivalent;
(d) CFFF Average Total Effective Dose Equivalent;
(e) Top Ten Facility Workers' Total Effective Dose Equivalents;
(f) Overexposures;
(g) Regulatory Agency Incident Notifications;
(h) Ratio of Recordable Incident Rate to SIC Code Average;
(i) Lost-Time Accidents as a Function of Facility Working Hours;
(j) Results of Special Nuclear Material Physical Inventory;
(k) Emergency Response Team Activations;
(l) Radioactivity Emissions in Gaseous Effluents;
(m) Radioactivity Emissions in Liquid Effluents;
(n) Radioactive Material Transportation Incidents; and,
(o) Regulatory Agency Violations.

On an annual basis, these performance indicators are summarized by the Regulatory Component and are formally presented to the Plant Manager for review. The Corrective Action Process (CAPs) described in Section 3.8 of this Chapter is used to document actions that need to be addressed, tracked and trended.

Figure 3.3 Typical Program Audit Checklist

PROGRAM AUDIT CHECKLIST

( ) What is the scope of the program?

( ) What is the program's purpose?

( ) Why is the program important?

( ) How important is this program relative to your many other programs?

( ) What is the basis (regulation, license, permit, commitment, ...) for the program?

( ) Has the program basis been translated into procedures?

( ) What are the procedures? (get copies)

( ) Who developed the procedures?

( ) What were the procedure developer's qualifications?

( ) Was there user input to development of the procedures?

( ) What is the quality of the procedures?

( ) How could the procedures be improved?

( ) How many individuals are trained and qualified to execute the procedures?

( ) How were they trained and qualified?

( ) How/where is the training and qualification documented?

( ) Could a well trained, but inexperienced, individual execute the procedure?

( ) What data is collected in executing the procedures?

( ) How is the data used?

( ) How is the data recorded?

( ) How/where are the records maintained?

( ) What are the program's major strengths?

( ) What are the program's major weaknesses?

( ) How can the program be improved?

3.7 INCIDENT INVESTIGATIONS

At the Columbia Fuel Fabrication Facility (CFFF), the organizational structure described in Chapter 2.0 of this License Application, supported by procedures approved in accordance with Subsection 3.4 of this Chapter, combine to provide for an incident investigation program that includes:

(a) A formal system for systematic reporting and investigation of abnormal occurrences (*i.e.*, process upsets and procedural inadequacies);
(b) Decision-making on corrective actions to fix the abnormal occurrence under investigation and to prevent recurrence of similar occurrences, and
(c) Follow-up to assure effectiveness of corrective and preventive actions.

To supplement this process, CFFF has in-place structured methodologies for determining and categorizing the apparent or root cause(s) of the failure(s) that led to investigated occurrences. Incident investigation is a management measure controlled by the quality program described in Section 3.3 of this License Application.

3.7.1 Incident Investigations Program Structure

3.7.1.1 Internal Reporting of Unusual, Safety-Related Occurrences

In accordance with approved procedures, a formal, computerized system is maintained by the Regulatory Component to enable all CFFF employees to report safety-related process upsets and procedure inadequacies to their First Level Managers for follow-up action. Such process upsets specifically include failures of Safety Significant Controls (SSCs) and/or Management Measures to execute their intended purpose. Procedural inadequacies include failure to have an approved procedure, inability to follow an approved procedure, and/or failure to follow an approved procedure. Employees are trained that the first step in internal reporting is to safely stop the work in process until the unusual occurrence is resolved. They are also trained to make appropriate notifications to process engineering and regulatory functions. (This reporting process is known as the "Redbook System" because, prior to its being computerized, it was a manual system involving forms that were completed and filed in red binders.)

3.7.1.2 Structured Unusual Occurrence Evaluation

In accordance with an approved procedure, all reported unusual occurrences are to be promptly evaluated, corrected and trended. This can be done in the "Redbook System," or if more detailed causal analysis methods are necessary to identify the causes and corrective actions to prevent recurrence of the event and to determine the extent of condition, the issue may be entered into the Corrective Action Process (CAPs), described in Section 3.8 of this Chapter.

3.7.1.3 Notification of Regulatory Agencies

In accordance with an approved procedure, cognizant regulatory agencies are promptly notified of significant unusual occurrences as required by the regulations.

3.7.2 Incident Investigations Program Implementation

3.7.2.1 Internal Reporting of Unusual Occurrences

Upon identifying a potential occurrence, the individual making the identification evaluates the occurrence for need to be formally reported. The flowchart for such an evaluation is presented in Figure 3.4.

The input screens for the computerized internal reporting process (aka "Redbook") request descriptive information from the originator reporting the occurrence, evaluation on whether the event involves a safety significant control and/or management measure, the immediate action taken, engineering review and final disposition. The final disposition is approved by the originator, the originator's supervisor, the engineering function, and the regulatory function.

3.7.2.2 Structured Unusual Occurrence Evaluation

Reported unusual occurrences are classified for evaluation in accordance with approved procedures. The Regulatory Component manager, along with the Manufacturing manager for the area involved and/or other managers as needed, review unusual occurrences to determine if structured evaluation is required.

If a structured evaluation is required, the event is input into the CAPs system described in Section 3.8 of this Chapter. Documentation in CAPs includes the following information about the event:

(a) Occurrence description;
(b) Causal analysis results and recommendations;
(c) Corrective actions;
(d) Corrective action status, including documentation of completed actions;
(e) Any additional information material to the event.

CAPs issues are kept available for review by internal and external auditors and cognizant Regulatory Agencies.

FLOWCHART FOR EVALUATION OF POTENTIAL REDBOOK EVENTS

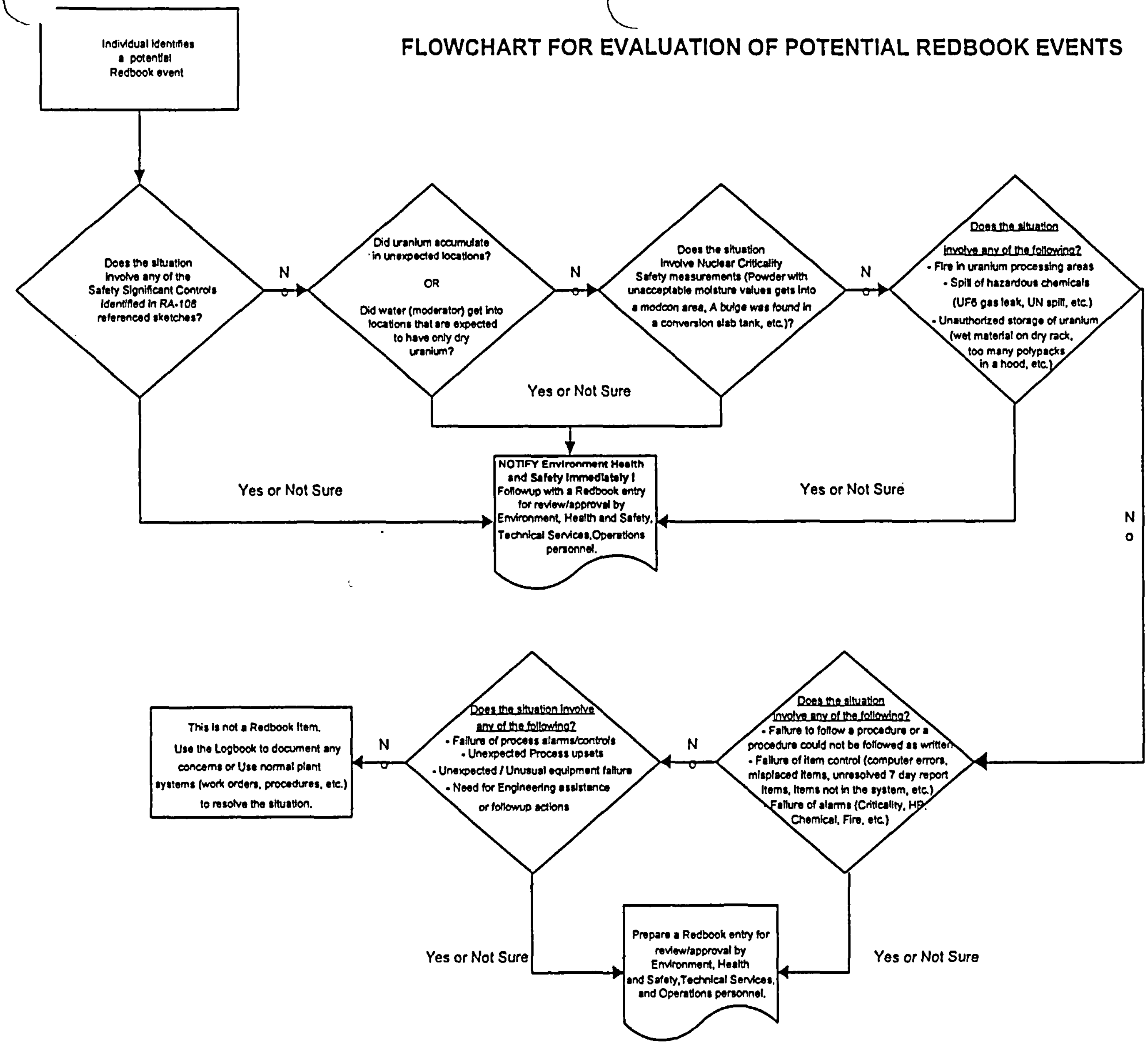


Figure 3.4 Flowchart for Evaluation of Potential Unusual Occurrence Reporting

3.7.2.3 Notification of Regulatory Agencies

The Regulatory Component Manager (or designee), makes the final call in determining if a reported unusual occurrence requires notification of Regulatory Agencies.

If notification is required, the NRC Operations Center is formally apprised of significant occurrences within prescribed time limits. The time limit "clock" starts when the assigned Emergency Coordinator classifies an occurrence, or when a cognizant Regulatory Function Engineer makes the initial "eyes-on" assessment of the errant condition, whichever comes first. Emergency Coordinators are authorized to make 1-hour notifications; and are authorized to make other notifications if a cognizant Regulatory Function Engineer cannot be contacted to make an initial "eyes-on" assessment in order to meet the time limit.

The NRC Operations Center is notified of the following types of occurrences:

(a) 1-Hour Notifications

- An inadvertent nuclear criticality;
- An acute intake, by a member of the offsite public, of 30 milligrams or greater of uranium in a soluble form;
- An acute chemical exposure to a member of the offsite public from licensed material, or hazardous chemicals produced from licensed material, that is greater than or equal to ERPG-2 limits.
- An acute chemical exposure to a worker from licensed material, or hazardous chemicals produced from licensed material, that is greater than or equal to ERPG-3 limits.
- An unusual occurrence such that no Items Relied on for Safety (IROFS) in the Integrated Safety Analysis (ISA) Summary remain available and reliable to control an accident sequence evaluated in the ISA; or,
- Loss of controls such that only one IROF in the ISA Summary remains available and reliable to prevent an inadvertent nuclear criticality, and has been in this state for greater than eight hours.

(b) 4-Hour Notifications

- An unusual occurrence that prevents immediate protective actions necessary to avoid exposures to radiation or radioactive materials, or releases of radioactive materials, that could exceed regulatory limits. (Notification is made as soon as possible.)

(c) 24-Hour Notifications

- An unplanned contamination occurrence that:

1. Involves a quantity of material greater than five times the lowest Annual Limit on Intake (ALI) for the material;
2. Results in a work area being unavailable for normal use for 24-hours, where the unavailability is due to the contamination (other than to allow isotopes with a half-life of less than 24-hours to decay prior to decontamination.)

- An occurrence in which safety-significant equipment is disabled or fails to function as designed when:
  1. The equipment is required to prevent exposures to radiation or radioactive materials, or releases of radioactive materials, that could exceed regulatory limits, or, to mitigate the consequences of an accident;
  2. The equipment is intended to be available and reliable when it is disabled or fails to function; and,
  3. No redundant equipment is available and operable to perform the intended function.
- An unplanned medical treatment at a medical facility when an individual has removable contamination, on clothing or person, that can spread into the medical facility.
- An occurrence in which a fire or explosion damages nuclear fuel, and the fuel's processing equipment or container is breached.
- An acute chemical exposure to a worker from licensed material, or hazardous chemicals produced from licensed material, that is greater than or equal to ERPG-2 limits.
- An acute chemical exposure to a member of the offsite public from licensed material, or hazardous chemicals produced from licensed material, that is greater than or equal to ERPG-1 limits;
- An unusual occurrence that results in the facility being in a state that was not analyzed, was improperly analyzed, or is different from occurrences analyzed in the ISA, that could have resulted in situations requiring 1-hour or 24-hour notification;
- Loss of IROFS, that could have resulted in situations requiring 1-hour or 24-hour notification;
- Any natural phenomenon or other external occurrence, including fires internal and external to the facility, that has affected or may have affected the intended safety function, or availability or reliability, of one or more IROFS; or,
- An occurrence that was considered in the ISA but was dismissed due to its likelihood; or was categorized as unlikely and whose unmitigated consequences would have resulted in situations requiring 1-hour or 24-hour notification had the IROFS not performed their intended safety function.
- Any unusual occurrence for which a news release is issued, or for which a report to non-NRC Regulatory Agencies is made, results in notification concurrent with the news release or Regulatory Agency report.

Unusual occurrences that require notification of the NRC Operations Center provide the following information, as it becomes available:

(a) The caller's name, position title, and call-back telephone number;

(b) Date, time, and location of the occurrence;

(c) A description of the occurrence including:
- Radiological or chemical hazards involved, including isotopes, quantities, and chemical and physical form of any material released;
- Actual or potential health and safety consequences to workers, the public, and the environment, including relevant radiological and chemical data for personnel exposures to radiation, radioactive materials, or hazardous chemicals mixed with or produced by licensed material;
- The accident sequence leading to the occurrence, and
- Whether the remaining structures, systems, components, and administrative controls relied on to prevent or mitigate potential accidents are available and reliable to perform their intended function.

(d) Any external conditions that are affecting the occurrence;

(e) Any actions taken in response to the occurrence;

(f) Status of the occurrence;

(g) Current and planned site status, including any declared emergency classification;

(h) Any non-NRC Regulatory notifications planned or made and,

(i) Status of any press releases planned or made.

Unusual occurrences that require notification of the NRC Operations Center are followed by a written report, within 30-days of the notification, that formally documents the following information (if available):

(a) The probable cause of the occurrence, including all factors that contributed to the occurrence;
(b) The name of the manufacturer of any safety-significant equipment that failed;
(c) Corrective actions taken or planned to prevent recurrence of similar occurrences in the future;
(d) The results of any evaluations or assessments and,
(e) Whether or not the occurrence was postulated and evaluated in the ISA.

This written report is typically faxed to the NRC Operations Center and appropriate NRC Region or Headquarters personnel.

3.8 CORRECTIVE ACTION PROCESS (CAPs)

The Columbia Fuel Fabrication Facility (CFFF) maintains a corrective action process that provides a structured, disciplined approach to detect, correct, and prevent recurrence of undesirable issues. CAPs employs a computerized system that tracks commitment owners and issue resolution, provides visibility and traceability, and fosters personal accountability.

The corrective action process is a management measure controlled by the quality program described in Section 3.3 of this License Application.

3.8.1 Corrective Action Program Structure

The CAPs process is an interaction between an issue originator, a Corrective Action Coordinator / Corrective Action Manager (CAC/CAM), a Corrective Action Review Board (CARB), and a Corrective Action Request Response Team (CART). The process begins with issue identification assessment, proceeds through analysis and correction, and ends with evaluation and closure. A CAPs Flowchart is presented in Figure 3.5.

3.8.2 Corrective Action Program Implementation

The CAPs computer program leads users through the steps in completing an Issue Report. A portion of a typical CAPs Issue Report is presented in Figure 3.6.

Figure 3.5 Corrective Action Process Flowchart

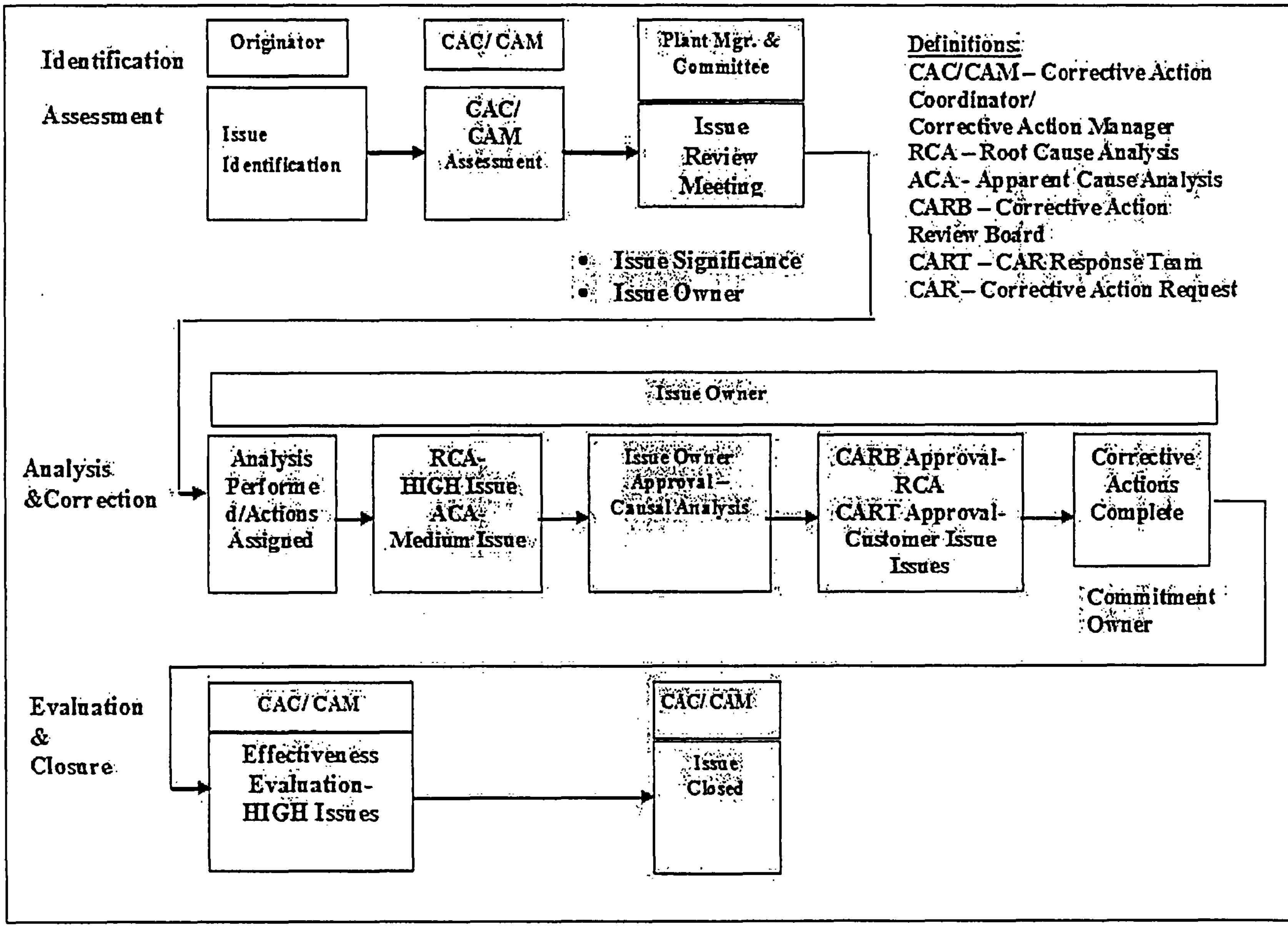


Figure 3.6 Format for Typical CAPs Issue Report

Westinghouse Electric CAPs — Issue Report

Step 1: Identification of Issue - (required information)

Created By: Natasha Wright
Created On: 12/06/2004
Phone: (803)647-3073

Organization
Originator's Business Unit: NFBU
Organization Level 2: US Fuel - Columbia Plant
Organization Level 3: Environmental Health & Safety
Originator's Group: NF - Columbia EH&S
Originator's Location: Columbia SC

Issue Title:

Issue Description:
State the issue (problem) ... include: Sequence of events (who, what, when and where); Consequences (cost, schedule, customer impact, safety significance)

QA/QC Controlled?:

Issue Raised By: ○ Employee ○ Regulatory Agency
○ Customer ○ Internal QA
○ Supplier

Does this issue potentially represent a condition adverse to nuclear safety pursuant to the requirements of Westinghouse Policy ... Procedure WEC 21.0? ○ Yes ○ No ○ I don't know

Is this issue potentially reportable to external authorities for environmental, industrial health & safety or other causes? ○ Yes ○ No ○ I don't know

Step 2: Identification of Issue - (optional information)

Suggested Significance Level: ○ High ○ Suggestion for Improvement
○ Medium ○ Best Practice
○ Watch/Trend

Recommended Resolution:
Suggested Issue Owner:
Suggested Responsible Org:
Background:
Date Issue First Occurred:
Affected Customer(s):
Affected Unit(s):
Affected Supplier(s):
Immediate Actions Taken:
Impacted Documents, Systems, Components, etc.:

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3.9 RECORD KEEPING AND REPORTING

The Columbia Fuel Fabrication Facility (CFFF) identifies, preserves, controls and destroys records in accordance with the guidelines, procedure, and practices set forth by Westinghouse. Records specifically required by applicable regulations are maintained in accordance with those regulations. Records data is reported as prescribed by applicable regulations. Record keeping and reporting are management measures controlled by the quality program described in Section 3.3 of this License Application.

3.9.1 Record Keeping and Reporting Program Structure

3.9.1.1 Records

Records include all those required by the regulations and the Quality Management System (QMS) in addition to regulatory correspondence, procedures, logs, reports, results of assessments, program audit and compliance inspection reports, commitments, *etc.*, whether or not required by regulatory agencies. Record custodians are identified, and their responsibilities are listed in an approved Records Flow Schedule (RFS) that also describes records to be retained, retention locations and retention time limits. Records and revisions to records are controlled by approved procedures.

All retained records are properly identified, including a "permanent" or "nonpermanent" classification, and can be retrieved in a timely manner. Records are protected against deterioration, damage or loss.

3.9.1.2 Reports

A detailed listing of reports required by regulatory agencies is maintained. Reports are submitted as required by the regulations. Details of reports and notifications related to abnormal occurrences (*i.e.*, process upsets and procedural inadequacies) are presented in Section 3.7 of this Chapter.

3.9.2 Record Keeping and Reporting Program Implementation

3.9.2.1 Record Keeping

The Records Flow Schedule contains detailed information of record types, separated into the following record names:

(a) Radiation Protection;
(b) Criticality;
(c) Environmental;
(d) Licenses / Permits;
(e) Procedures;
(f) Training;

(g) Safeguards;
(h) Safety;
(i) Emergency Preparedness; and,
(j) Miscellaneous.

In particular, the following record is maintained in response to the revised 10 CFR 70 regulation:

(a) A record documenting each discovery that an Item Relied on for Safety (IROFS), or Management Measure has failed to perform its function upon demand, or has degraded such that high or intermediate consequence events could occur.

All retained records are stored and maintained readily accessible in order to meet retrieval time restraints. This records retention system includes the capability to retrieve records within 24-hours for records generated within the preceding 12-months and within 7-calendar-days for older record generation periods.

Prudent measures of redundancy and protection are maintained such that acts of record alteration or inadvertent destruction do not foreclose the capability for reconstructing a complete and correct set of required records. In cases where such measures fail, and a particular record is lost or destroyed, a reconstruction may be generated using source data applicable to the time the subject record was originally created. When a record is just partially missing, all salvaged portions are attached to the reconstruction. If source data is not available for re-creating a missing record, the record may be reconstructed using inference to data relative to other records for similar information and time periods.

3.9.2.2 Reporting

A detailed listing of reports required by NRC regulations will be maintained and followed. This listing documents:

(a) Identification of the applicable regulations;

(b) Descriptions of the reports required, and

(c) Frequencies at which the reports must be submitted.

In particular, the following reports are submitted in response to the revised 10 CFR 70 regulation:

(a) For safety-related CFFF changes that do not require NRC Staff pre-approval, a report is submitted to NRC annually, within 30-days after the end of the calendar year during which the changes occurred, that contains a brief summary of all such changes.

CHAPTER 4.0

INTEGRATED SAFETY ANALYSIS (ISA)

4.1 ISA PROGRAM STRUCTURE

The Columbia Fuel Fabrication Facility (CFFF) develops and maintains an Integrated Safety Analysis (ISA) for the site. The ISA is a systematic examination of the facilities, processes, equipment, structures, and personnel activities to ensure that all relevant hazards that could result in unacceptable consequences have been evaluated, and that protective measures have been identified.

A document titled "Baseline Integrated Safety Analysis (ISA) and ISA Summary Handbook" provides details describing the key features and practices for (1) the conduct of a baseline ISA of the plant site and structures, (2) baseline system ISAs of plant operations, and (3) preparation of ISA summaries. It defines team organization and skills, analytical rules and assumptions, techniques, and deliverables required to enable an analysis to be performed. The document embraces all aspects of the CFFF ISA Plan and Schedule submitted to, and approved by, NRC staff in accordance with Section 70.62(c)(3)(i) of the Part 70 regulation. The original handbook and subsequent revisions are approved by the Regulatory Component Senior Manager.

In general, the ISA provides:

- a description of the structures, equipment, and process activities at the facility,
- an identification and systematic analysis of hazards at the facility,
- a comprehensive identification of potential accident/event sequences that would result in unacceptable consequences, and the expected magnitudes and likelihoods of those sequences,
- an identification and description of safety systems that are relied upon to limit or prevent potential accidents or mitigate their consequences; and,
- an identification of management measures taken to ensure the availability and reliability of identified safety systems.

A cross-reference of ISA Activities with Part 70 Regulatory Citations, NUREG-1520 Chapter 3 guidance, and CFFF Handbook guidelines is summarized in Table 4.1.

Table 4.1 Cross-Reference of Integrated Safety Analysis Activities

ISA Activity	10 CFR Part 70 Regulatory Citation	NUREG-1520, Chapter 3	CFFF Guidelines
Prepare site description	70.65(b)(1)	3.4.3.2(1)	1.2.1
Prepare facility description	70.65(b)(2)	3.4.3.2(2)	1.2.2
Describe monitoring and alarms	70.65(b)(4)	3.4.3.2(4C)	1.2.3
Describe compliance with baseline design criteria	70.64 (if applicable)	3.4.3.2(4D)	1.2.5
Describe ISA methods	70.65(b)(5)	3.4.3.2(5)	7.1.1 9.2
Define ISA Team requirements and describe ISA Team	70.65(b)(5)	3.4.3.2(5)	7.1.2
Define consequences of interest and consequence categories	70.65(b)(3)	3.4.3.2(3)	7.1.3
Define quantitative standards for acute chemical exposure	70.65(b)(7)	3.4.3.2(7)	7.1.3
Define frequency categories	70.65(b)(9)	3.4.3.2(9)	7.2.3
Develop process description	70.65(b)(3)	3.4.3.2(3)	2.1
Compile process safety information	70.62(b)	3.4.3.1	5.1
Identify hazards	70.65(b)(3)	3.4.3.2(3)	7.1.5
Conduct hazards analysis	--	--	7.1.6
Describe Safety Significant Controls (SSCs)	70.65(b)(6)	3.4.3.2(6)	7.2.5
Demonstrate compliance with 10 CFR 70.61	70.65(b)(6)	3.4.3.2(4) and (6)	7.2.4
Describe SSC management measures	70.65(b)(4)	3.4.3.2(4B) and (6)	7.2.5
List sole SSCs	70.65(b)(8)	3.4.3.2(8)	7.2.6

An ISA begins as a Baseline Document. This document identifies equipment and operations presenting hazards, and the control features that are relied upon for protection of the environment, and the health and safety of facility employees and the neighboring public. Any subsequent changes to the analyzed system are controlled by the CFFF Configuration Management Program and/or an electronic procedure management process. Configuration control data packages for such changes are filed with their respective Baseline ISAs which, taken together, provide a substantially complete "living" framework of the facility safety basis that is maintained on the CFFF site.

An ISA Summary (1) presents key aspects of the ISA in sufficient detail to enable an independent overview of the subject systems, and (2) provides reasonable assurance that operation of these systems will not lead to situations that would exceed the performance requirements specified in Section 70.61 of the Part 70 regulation. ISA Summaries are submitted to the NRC and are updated as appropriate to reflect any safety-significant changes. Sections 70.64 and 70.65 of the Part 70 regulation provide specific requirements for the content of the ISA Summary. These requirements are summarized as key components of the CFFF ISA Summary in Table 4.2.

Table 4.2 CFFF ISA Summary Key Components

CFFF ISA SUMMARY

KEY COMPONENTS

1.0 A general description of the site with emphasis on those factors that could affect safety. (Site and Structures ISA only. Other ISAs reference this section.)

2.0 A general description of the facility with emphasis on those areas that could affect safety, including an identification of the controlled area boundaries. (Site and Structures ISA only. Other ISAs reference this section.)

3.0 Concise description of each process analyzed in the ISA in sufficient detail to understand the theory of operation; and, for each process, the hazards that were identified in the ISA and a general description of the types of accident sequences.

4.0 Demonstration of compliance with the performance requirements of Section 70.61, including a description of management measures, monitoring and alarms, and other facility safety systems.

5.0 A description of the ISA Team, qualifications, and the methods used to perform the ISA.

6.0 A description of each Safety Significant Control (SSC) in sufficient detail to understand its function in relation to compliance with the performance requirements of Section 70.61.

7.0 A description of the quantitative standards used to assess the consequences of acute chemical exposure to licensed material or chemicals produced from licensed materials that are on site, or expected to be on site (Chemical Receipt, Handling and Storage ISA only).

8.0 A list of the SSCs that are the sole item preventing or mitigating an accident sequence that exceeds the performance requirements of Section 70.61.

9.0 A description of the definitions of unlikely, highly unlikely, and credible as used in the evaluations in the ISA.

4.1.1 The Handbook

The Baseline Integrated Safety Analysis (ISA) and ISA Summary Handbook consists of the following sections:

4.1.1.1 Site and Structures

This section describes methodology for preparing a description of the site with emphasis on those factors that could affect safety (*e.g.*, meteorology, seismology, *etc.*) and a description of the CFFF structures with emphasis on those areas that could affect safety, including an identification of the controlled area boundaries and facility safety systems (*e.g.*, moderation control barriers, emergency alarms, *etc.*). Information developed in accordance with this section is presented in appropriate parts of the ISA Summary.

4.1.1.2 Process Description

This section describes methodology for preparing a description of normal operation as it relates to each defined system. Information developed in this section, in conjunction with process theory and process equipment information developed in Sections 3.0 and 4.0, respectively, is presented in appropriate parts of the ISA Summary.

Section 2.0 is limited to the process itself. A narrative outline of the system equipment controls, with text references that detail normal operating boundaries (*e.g.*, compositions, concentrations, flows, safety-significant sampling, *etc.*), are typically included. Information such as schematic representations (flow diagrams) of the system, equipment interconnections, material types, or safety-related alarms/interlocks might be included if needed to present a clear understanding of process flows; but, the details concerning such items are included in other appropriate parts of the Baseline ISA.

4.1.1.3 Process Theory

This section describes methodology for preparing a narrative description of the normal process operating parameters in sufficient detail to understand the theory of operation. Process theory information includes (1) the ranges of conditions expected, (2) the hazards of the process, and (3) a general description of the types of accident sequences that could potentially occur. Descriptions of upset conditions that have potential for exceeding safety limits are typically included. References documenting the sources of the theory are also typically included.

4.1.1.4 Process Design and Equipment

This section describes the methodology for documenting the dimensions, construction materials, and design configuration of lines and vessels of each defined system. A narrative description of system transfer interconnections and a tabulation of relevant reference drawings are typically included.

4.1.1.5 Drawings and Procedures

This section describes the methodology for contributing to a compilation of the process safety information that will be maintained on site for use by the team of individuals performing a system's Process Hazard Analysis (PHA), identifying Safety Significant Controls (SSCs), performing safety analyses, and/or quantifying the risk of accident scenarios. Photographs of system/subsystem equipment that had relevance to (and were used during) the analysis process are typically included in the appropriate part of the Baseline ISA. Any other documents collected for review and/or information purposes are retained with the Baseline ISA as backup data.

4.1.1.6 Safety Analyses

This section describes the methodology for performing safety analyses in support of the Baseline ISA. First, each system is independently evaluated by environmental protection, radiation safety, nuclear criticality safety, safeguards, fire safety, and chemical safety functions. Then, to complete the analysis process, all applicable safety functions deliberate as a group to optimize safety controls and to provide recommendations to cognizant management for review and disposition.

4.1.1.7 Process Hazard Analysis and Accident Sequence Evaluation

This section describes the methodology for performing the Process Hazard Analysis (PHA) part of the ISA. The PHA is used to systematically identify and assess hazards, in order to evaluate the potential internal, external, and natural events that could cause identified hazards to develop into accidents.

This section also describes methodology for analyzing all credible accident sequences that have potential to result in intermediate or high consequence events. The purpose of analyzing these accident sequences is to identify the Items Relied on for Safety (IROFS) that ensure operations at the facility can meet the performance requirements specified in Section 70.61 of the Part 70 regulation. All accident sequences identified in the PHA that have an unmitigated consequence that is intermediate or high are carried forward for evaluation. This evaluation determines the severity of an accident's consequence on a linear scale from 0 (low) to 6 (high), and the overall likelihood of the accident's occurrence on a logarithmic (base 10) scale from 1 (not unlikely) to –4 (highly unlikely). The accident sequence risk evaluation process is summarized in Figure 4.1 and Table 4.3.

Figure 4.1 Accident Sequence Risk Evaluation Process

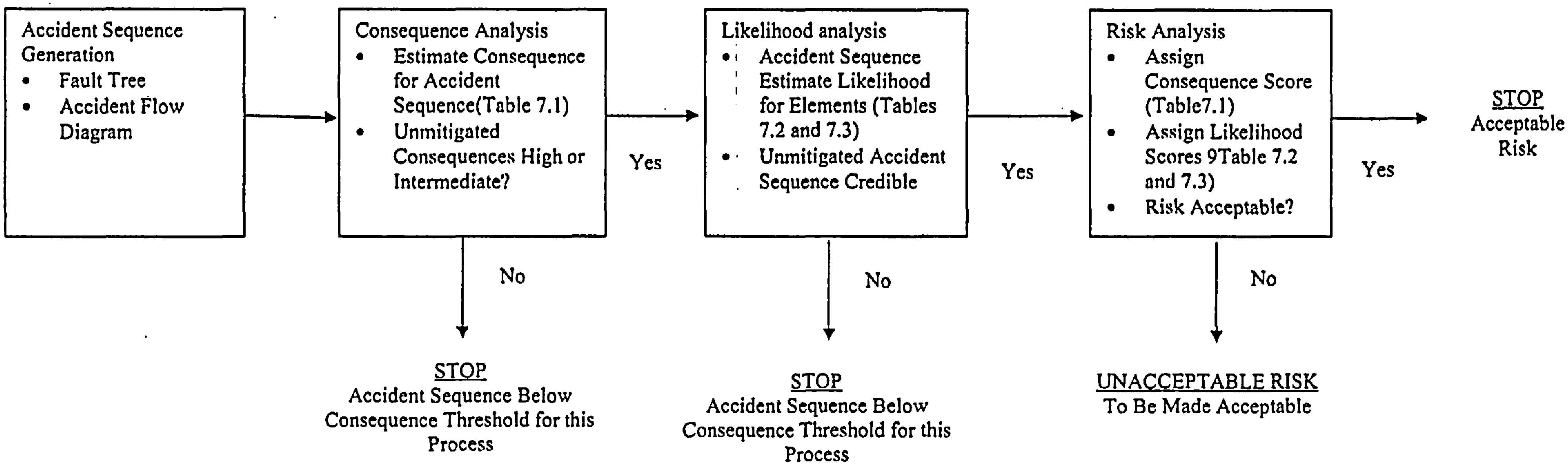


Table 4.3 Risk Analysis Table

				Chemical Consequence	Fire Consequence	Criticality Consequence	Radiological Consequence
Score	Performance Indicator	Quality Level	Qualitative Descriptor	Effects from Chemical Hazards Exposure[1]	Effects from Fire Hazards Exposure[1]	Effects from Criticality Hazards Exposure	Effects from Radiological Hazards Exposure
6	• Greater than or equal to 100 rem dose equivalent to a worker, and/or • Greater than or equal to ERPG-3 chemical exposure to a worker, and/or • Greater than or equal to 25 rem dose equivalent to the offsite public, and/or • Greater than or equal to 30 milligrams soluble uranium intake by the offsite public, and/or • Greater than or equal to ERPG-2 chemical exposure to the offsite public	A	Very High Multiple fatalities	Acute chemical exposure to an individual from licensed material or hazardous chemicals produced from licensed material that could cause death to multiple workers or permanently disable a member of the public at the site boundary	Fire that could cause commensurate radiological, chemical, or criticality consequences	Occurrence of a criticality	Lethal radiation dose
5			High Fatality or multiple permanent health effects	Acute chemical exposure to an individual from licensed material or hazardous chemicals produced from licensed material that could endanger the life of the worker or lead to irreversible or other serious long-lasting health effects to a member of the public at the site boundary	Fire that could cause commensurate radiological or chemical consequences	N/A	Lethal radiation dose

<table>
<tr><th colspan="4"></th><th>Chemical Consequence</th><th>Fire Consequence</th><th>Criticality Consequence</th><th>Radiological Consequence</th></tr>
<tr><th>Score</th><th>Performance Indicator</th><th>Quality Level</th><th>Qualitative Descriptor</th><th>Effects from Chemical Hazards Exposure[1]</th><th>Effects from Fire Hazards Exposure[1]</th><th>Effects from Criticality Hazards Exposure</th><th>Effects from Radiological Hazards Exposure</th></tr>
<tr><td>4</td><td rowspan="2">• Greater than or equal to 25 rem dose equivalent to a worker, and/or
• Greater than or equal to ERPG-2 chemical exposure to a worker, and/or
• Greater than or equal to 5 rem dose equivalent[2] to the offsite public, and/or
• Greater than or equal to ERPG-1 chemical exposure to the offsite public, and/or
• Greater than or equal to 5,000 times Table 2, Appendix B, 10 CFR 20 radioactivity release[3] to the environment, and/or
• Loss of nuclear criticality safety double contingency protection</td><td>B</td><td>Intermediate

Permanent loss of function/limb or multiple lost-time injury</td><td>Acute chemical exposure to an individual from licensed material or chemicals produced from a licensed material that could lead to irreversible or serious long-lasting effects to a worker or mild transient health effects to a member of the public at the site boundary</td><td>Fire that could cause commensurate radiological or chemical consequences</td><td>N/A</td><td>Exposure of worker or member of the public substantially in excess of 10 CFR 20 limits</td></tr>
<tr><td>3</td><td>C_{ss}</td><td>Medium

Restricted/lost-time work injury or multiple medical treatment cases</td><td>Chemical accident that could result in exceeding radiological criteria</td><td>Fire that could cause commensurate radiological or chemical consequences</td><td>Loss of double contingency protection</td><td>Exposure of worker or member of the public in excess of 10 CFR 20 limits</td></tr>
</table>

				Chemical Consequence	Fire Consequence	Criticality Consequence	Radiological Consequence
Score	Performance Indicator	Quality Level	Qualitative Descriptor	Effects from Chemical Hazards Exposure[1]	Effects from Fire Hazards Exposure[1]	Effects from Criticality Hazards Exposure	Effects from Radiological Hazards Exposure
2 1 0		C	Anticipated process upset consequences that are controlled and remediated by licensed safety programs				

[1] Does not include plant conditions that result in an occupational risk, but do not affect the safety of licensed radioactive materials.
[2] From exposure of a hypothetical individual at the site boundary, due to an airborne radioactivity release to the environment.
[3] Concentration of radioactivity in liquid released to ground water on site or surface water off site.

4.1.1.8 License Compliance Verification

This section describes the methodology for (1) presenting a listing of any License Commitments specific to the defined system, and (2) providing documentation that potentially applicable commitments were reviewed during the safety analyses of the system and were confirmed to be properly addressed.

4.1.1.9 Appendices (Optional)

This section describes methodologies useful for development of supporting information for each defined system. Topics addressed may include:

(a) Consequence analyses;
(b) PHA methods;
(c) Reducing fault trees to accident sequences;
(d) Data sources and calculation methods;
(e) Example analyses for accident sequence risk estimation using the fault tree approach;
(f) Example analyses for accident sequence risk estimation using the accident flow diagram approach;
(g) Nuclear criticality safety parametric studies;
(h) Checklists; and,
(i) Risk ranking Integrated Safety Analyses.

4.1.1.10 ISA Review Form

This section describes the methodology for documenting results of the group deliberations to optimize safety controls and to provide any consensus recommendations to cognizant management for review and disposition, as described in Section 6.0.

4.1.1.11 Photographs (Optional)

This section describes the methodology for inclusion of photographs used by the ISA Team in developing the ISA or the ISA Summary.

4.1.1.12 Preparation of ISA Summaries

This section describes the methodology for preparing ISA Summaries for submittal to the NRC. The ISA Summary provides information to the NRC that provides reasonable assurance that the CFFF has performed a systematic evaluation of facility hazards and has identified credible accident sequences, Items Relied on for Safety (IROFS) and management measures that satisfy the performance requirements specified in Section 70.61 of the Part 70 regulation. The ISA Summary also provides information to demonstrate that credible accidents that result in a release of licensed material, a nuclear criticality event, or any other exposure to radiation resulting from the use of licensed material, that exceed the criteria stated in 10 CFR 70.61, are "unlikely" or "highly unlikely" to occur, as appropriate. Accident sequences having unmitigated consequences that will not exceed the 70.61 performance requirements, once identified as such, are not reported in the ISA Summary.

4.1.2 The ISA

The Integrated Safety Analysis (ISA) is developed in accordance with methods acceptable to Columbia Fuel Fabrication Facility (CFFF) management, as approved by the Handbook. Depending on when a specific system ISA was developed during the multiyear CFFF ISA development process, any specific ISA may or may not embrace a given activity described in the Handbook. However, if a given activity was embraced, it was performed as described. A notable exception to this latitude is Handbook Subsection 7.2 (*Accident Sequence Evaluation)*. Subsection 7.2 activities are specific commitments to the Nuclear Regulatory Commission (NRC) and must be executed, as described, for each ISA.

4.1.2.1 System ISAs

Baseline ISAs for the following systems make up the CFFF ISA:

(a) Site and Structures;
(b) Plant Ventilation;
(c) Chemicals Receipt, Handling, and Storage;
(d) Nuclear Material Storage;
(e) ADU Conversion;
(f) ADU Bulk Powder Blending;
(g) Pelleting;
(h) ADU Fuel Rod Manufacturing;
(i) Burnable Absorber Fuel Processing;
(j) Burnable Absorber Fuel Rod Manufacturing;
(k) Final Assembly
(l) Scrap Uranium Processing;
(m) UF6 Cylinder Washing;
(n) Safe Geometry Dissolver;
(o) Solvent Extraction;
(p) Uranyl Nitrate Bulk Storage Tanks;
(q) Hoods and Containment;
(r) URRS Wastewater Treatment;
(s) Low Level Radioactive Waste Processing; and,
(t) Laboratories.

4.1.2.2 ISA Maintenance

ISAs are maintained current through implementation of the Configuration Management program described in Section 3.1 of this License Application. A Baseline ISA consists of all documentation that might extend from an original Criticality Safety Assessment, through Criticality Safety Evaluations, to the final, fixed in time, ISA document (for which an original ISA Summary was submitted to NRC pursuant to the ISA Plan and Schedule submitted to, and approved by, NRC staff in accordance with Section 70.62(c)(3)(i) of the Part 70 regulation). All subsequent changes that might affect the Baseline ISA are reviewed by the same safety disciplines that were involved in preparation of the Baseline ISA. If safety analyses are required for the change, they are performed to the current standards required for the Baseline ISA. Summary details of the change, including required approvals, are documented on a Configuration Change Control Form that is filed with the applicable Baseline ISA, thus providing a substantially complete "living" framework for the facility safety basis.

4.1.3 The ISA Summary

The Integrated Safety Analysis (ISA) Summary is developed in accordance with methods acceptable to Columbia Fuel Fabrication Facility (CFFF) management, as approved by the Handbook. Handbook subsection 7.2 (*Preparation of ISA Summaries)* activities are specific commitments to the Nuclear Regulatory Commission (NRC) and must be executed, as described, for each ISA Summary.

4.1.3.1 ISA Summary Content

The ISA Summary includes the following information:

(a) Site

The site description focuses on those factors that could affect safety, such as geography, meteorology (*e.g.*, high winds and flood potential), seismology, demography, and nearby industrial facilities and transportation routes.

(b) Facility

The facility description focuses on features that could affect potential accidents and their consequences. Examples of such features include facility location, facility design information, and the location and arrangement of structures on the facility site.

(c) Processes, Hazards, and Accident Sequences

The process description addresses each process that was analyzed as part of the ISA. This description also includes a discussion of the hazards (and interactions of hazards) for each process and the accident sequences that could result from such hazards, and for which the unmitigated consequences could exceed the performance requirements of 10 CFR 70.61.

(d) Demonstration of Compliance with 10 CFR 70.61

For each applicable process, the following information, developed in the ISA, is presented to demonstrate compliance with the performance requirements of 10 CFR 70.61:

1. Postulated consequences and comparison to the consequence levels identified in the performance requirements, as well as information (such as inventory and release path factors) supporting the results of the consequence evaluation.
2. Information showing how CFFF established the likelihoods of accident sequences that could exceed the performance requirements of 10 CFR 70.61.
3. Information describing how designated Items Relied on for Safety (IROFS) protect against accident sequences that could exceed the performance requirements of 10 CFR 70.61.
4. Information on management measures applied to IROFS.
5. Information on how the criticality monitoring requirements of 10 CFR 70.24 are met.
6. When applicable, how the baseline design criteria of 10 CFR 70.24 are addressed.

(e) Team Qualifications and ISA Methods

A discussion of the ISA Team's qualifications and ISA methods used is presented. Specific examples of the application of ISA methods is included as necessary to demonstrate appropriate selection and use.

(f) List of Items Relied on for Safety (IROFS)

The Items Relied on for Safety, relative to all intermediate and high consequence accidents, are listed and described in sufficient detail to understand their safety functions.

(g) Chemical Consequence Standards

Site specific quantitative standards (*i.e.,* ERPG levels) are identified for assessing the chemical consequences specified in 10 CFR 70.61.

(h) List of Sole IROFs

Any Item Relied on for Safety that is the only control for preventing or mitigating an accident, for which the consequences could exceed the performance requirements of 10 CFR 70.61, are listed and described.

(i) Likelihood Definitions

The ISA Summary includes definitions of the terms "credible", "unlikely", and "highly unlikely", as used in the ISA.

4.1.3.2 ISA Summary Maintenance

ISA Summaries are submitted to the NRC Licensing Staff, and are maintained as current, stand-alone documents. Whenever CFFF regulatory management makes a decision to approve a substantive change to the ISA Summary, this requires NRC pre-approval under 10CFR70.72. The NRC Licensing Project Manager is apprised and an amendment request is submitted. Whenever the CFFF makes change to the ISA Summary that does not require NRC pre-approval under 10CFR70.72, changed pages to update the ISA Summary are submitted to the NRC annually, within 30 days after the end of the calendar year during which the change occurred.

CHAPTER 5.0

RADIATION SAFETY PROGRAM

5.1 RADIATION SAFETY PROGRAM STRUCTURE

The Columbia Fuel Fabrication Facility (CFFF) maintains a Radiation Safety Program for the site. A primary purpose of the Radiation Safety Program is to assure that exposure of workers to radiation and radioactive materials is kept As Low As Reasonably Achievable (ALARA).

5.2 RADIATION SAFETY PROGRAM

Definitions:

5.2.1 The Derived Airborne Concentration (DAC) and Annual Limit on Intake (ALI) referenced in this chapter, and used to calculate Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE), are based on the dose coefficients in ICRP Publication No. 68.

ALARA

5.2.2 The Columbia Fuel Fabrication Facility (CFFF) implements and maintains a Radiation Safety Program which assures that exposure of workers to radiation and radioactive materials is kept As Low As Reasonably Achievable (ALARA).

5.2.3 The Regulatory Component maintains the occupational doses and doses to members of the public ALARA by:

- Generating specific ALARA requirements and goals.
- Including ALARA requirements in operating procedures.
- Assigning responsibility and authority for implementing ALARA requirements to first level managers.
- Incorporating and approving ALARA considerations in the design of new or modified facilities and equipment.
- Including ALARA principles and requirements in required training sessions.

5.2.4 The appropriate Senior Component Management maintains oversight of the CFFF commitment to ensure exposures to radiation and radioactive materials remain As Low As Reasonably Achievable (ALARA).

5.2.5 Short-term ALARA progress is tracked by a formal quarterly evaluation and documentation of performance indicators, by the Regulatory Component. This is reported to management as appropriate.

5.2.6 Long-term ALARA progress is tracked by a formal annual evaluation and documentation of performance indicators, by the Regulatory Component. This is reported to management.

5.2.7 Implementation of the ALARA process is used to satisfy the 10CFR20.1101(c) requirement for annual review of radiation protection program content and implementation.

Radiation Work Permits:

5.2.8 A Radiation Work Permit is required for all temporary configuration changes (including approval duration); and, for all work for which safety requirements are not specifically covered by an approved procedure and the following conditions are met:

1) Release of detectable contamination outside of a Contamination Controlled area might result in contamination of personnel or equipment.

(2) The average local concentration of radioactive contaminants is predicted to exceed 50-percent of Derived Air Concentration (DAC).

(3) The deep dose equivalent is predicted to exceed 100 millirem in a week.

(4) The Total Effective Dose Equivalent is predicted to exceed 10-percent of the 10CFR20 limit.

5.2.9 Only personnel who have completed required safety training are be assigned to work under an RWP.

5.2.10 A copy of an RWP is made available to personnel working under the permit, and the work will only be conducted as specified in the approved permit.

5.2.11 The Regulatory Component specifies and approves applicable protection requirements for the work to be performed.

Ventilation Systems:

5.2.12 Ventilation control systems are installed and used whenever they are determined to be required by the Radiation Safety Function, based on measurements or evaluations.

5.2.13 Ventilation systems are designed and operated to assure adequate control of radioactive dust and particulate matter, and will are monitored and corrected as needed on a routine basis specified by the Radiation Safety Function. Air flows are typically maintained from non-chemical process area to chemical process area. Whenever adverse air flows are detected, corrective actions are taken as soon as practicable.

5.2.14 During work operations, ventilation systems servicing primary enclosures where uncontained radioactive material is handled provide minimum face velocities of 100-linear feet per minute. All enclosure velocities are tested quarterly; and all systems which fail to meet the velocity criteria are either corrected immediately or shut down until corrected.

5.2.15 Gloveboxes or similar enclosures are used when containment by conventional ventilation hoods is not possible or is not practical.

- These systems are designed and operated at a negative pressure with respect to room air, unless positive pressure is specifically approved by the Radiation Safety Function.
- These systems are equipped with instrumentation for measuring differential pressure.
- The operability of instrumentation is checked periodically.

5.2.16 When positive pressure enclosures are required for a purpose specifically approved by CFFF management, they will be designed and operated according to control criteria approved by the Radiation Safety Function, including monitoring on a routine basis. The following criteria apply:

- The gloveboxes are designed for high integrity containment and moisture control.
- The gloveboxes are operated at a nominal positive internal pressure; and, in-plant air sampling is used to verify containment of radioactive material.
- Internal atmospheres are continuously re-circulated through HEPA filters.
- Alarms are provided to indicate when pressure exceeds the pre-set positive pressure limit.
- An interlock, or other pressure relief device, is provided to exhaust the glovebox with a sufficient factor of safety to ensure its continuing integrity.

5.2.17 Ventilation hoods and gloveboxes are constructed primarily of metal, and use glass and/or Class-I fire rated plastic for viewing areas.

5.2.18 Ventilation ducts are designed to minimize accumulations of radioactive material, and are inspected on a frequency commensurate with the potential for accumulation.

5.2.19 Exhausts from hoods, gloveboxes, and similar enclosures are passed through HEPA filtration that is monitored on a routine basis to assure they meet maximum differential pressure limits approved by the Radiation Protection Function. The HEPA filters are replaced using one or more of the following criteria:

- A routine schedule.
- Airborne radioactive concentrations.
- Hood velocity
- Differential pressure (8 inches of water for negative pressure systems and 4-inches of water for positive pressure systems)
- Particulate penetration

5.2.20 Exhausts from re-circulating process-air cleaning systems either have their HEPA filters penetration tested, or are sampled for airborne radioactive concentrations on at least a quarterly basis. Maintenance is performed on systems found to exceed 25-percent Derived Air Concentration (DAC).

5.2.21 The effectiveness of final HEPA filters, in process ventilation equipment and containment systems, is determined by in-situ testing using particulate penetration methods or other means approved by the Radiation Safety Function. The testing is performed following each filter change.

5.2.22 Adequacy of containment and ventilation controls is determined by continuous air sampling. Action activity levels are approved by the Radiation Safety Function.

Air Sampling:

5.2.23 All areas where exposure to airborne radioactive material is a risk are monitored using air sampling.

- Air samplers used to estimate operator Committed Effective Dose Equivalent are located in or around the worker's breathing zone.
- Air samplers used to monitor the effectiveness of containment and/or ventilation are located where they will detect deterioration in these controls.

5.2.24 The breathing zone representativeness for fixed or portable air samplers is:

- Determined in accordance with Section 3 of Regulatory Guide 8.25, "Air Sampling in the Workplace".
- Confirmed at least annually or whenever substantive changes are made, in accordance with Section 3 of Regulatory Guide 8.25.

5.2.25 Air samples are changed out on a frequency specified by the Radiation Safety Function.

- Fixed air samplers are typically changed out at least once each working shift during normal operations, unless area airborne concentrations justify a less frequent schedule.
- Samples are allowed time for natural activity to decay and are analyzed on measurement equipment calibrated with sources traceable to national standards.
- Samples suspected of reflecting elevated airborne events are counted as soon as practicable for investigation purposes.
- Lapel samples are used to supplement and/or test fixed samples.

5.2.26 Air sampling practices provide for investigation and/or special sampling, if the radioactivity concentration outside of containment structures exceeds action levels specified by the Radiation Safety Function.

5.2.27 All new operations, or substantive modifications to existing equipment are evaluated to assess the need for air sampling.

5.2.28 Air flow measurement devices on air samplers are routinely verified for proper adjustment and proper operation by the Radiation Safety Function.

Contamination Control:

5.2.29 Contamination surveys are performed to assure that maximum acceptable limits are not exceeded. Maximum acceptable limits and minimum survey frequencies for floors and other readily accessible surfaces are specified in Figure 5.1.

Figure 5.1

CONTAMINATION SURVEY LIMITS AND FREQUENCIES

AREA TYPE	ALPHA ACTIVITY ON SMEAR *	MINIMUM FREQUENCY
Change Rooms, and Eating/Vending Areas	50	Weekly
Clean Area	200	Monthly
Contamination Controlled Area	5000	Biweekly

*Units of Disintegrations-Per-Minute Per 100-Square-Centimeters

5.2.30 Approved smear measurement techniques are used to survey floors and other readily accessible surfaces. The following criteria applies to contamination surveys:

- All new operations are subject to increased surveillance.
- Average contamination is based on areas not greater than 10-square meters.
- Decontamination is required within three working shifts whenever the average contamination exceeds the limits.
- Decontamination is required immediately whenever the average contamination exceeds five times the limit.
- Decontamination is required immediately whenever the contamination is found in clean areas.
- Verification surveys are performed to assure decontamination below limits.
- An alpha smear measurement technique is used, that is capable of detecting 25-disintegrations-per-minute per sample, at a 90-percent confidence level, when surveying clean areas, change rooms, and eating and vending areas.

5.2.31 Specific portions of a Contamination Controlled Area might be assigned higher limits and/or frequencies, provided a documented evaluation by the Radiation Safety Function has demonstrated that collective protective measures for the subject area can assure compliance with licensed and regulatory requirements. Examples include areas where contamination does not represent the potential for becoming airborne or being tracked, and areas where decontamination is impractical (e.g., under process equipment, hoods, etc).

5.2.32 Contamination surveys are performed on radioactive material received from other facilities in compliance with 10CFR20.1906 with the following clarifications:

- The three hour "clock" referenced in 10CFR20.1906, as it applies to the contents of the van, begins when the tamper indicating seal is broken for radioactive material received in an enclosed dry van with tamper indicating seal.
- For all other receipts of radioactive material, the survey process will be initiated, but not necessarily completed, within the time prescribed by 10CFR20.1906 and continued uninterrupted until completed.

Access Control:

5.2.33 Access to areas in which radioactive materials are used or stored are controlled.

5.2.34 Personnel are authorized to enter Contamination Controlled Areas, by virtue of management approval in accordance with the CFFF Physical Security Plan, only after completing required radiation protection training.

5.2.35 Access points to Contamination Controlled Areas are provided with change rooms and/or step-off pads. Each such access point defines an uncontaminated side and a potentially contaminated side, with the step-off area dividing the two sides.

5.2.36 Each access point to the Contamination Controlled Area is posted in accordance with 10CFR20.1902, with the exception of 10CFR20.1902(e). In lieu thereof, a sign bearing the legend "Every container or vessel in this area may contain radioactive material" is posted at entrances to each such area in which radioactive materials are used, or stored.

5.2.37 Access to Contamination Controlled Areas, including the Chemical Manufacturing Area and other areas involved in the processing and storage of unencapsulated radioactive material (i.e., not contained in a sealed source, a fuel rod, a shipping container, or other type of strong, tight container), requires the use of protective clothing.

5.2.38 Protective clothing is provided for personnel entering the Contamination Controlled Area. This includes such apparel as labcoats, coveralls, shoecovers, safety shoes, and/or other specified garments consistent with an individual's work assignment. Street clothing, of persons to be dressed completely in protective clothing, is stored on the uncontaminated side of the change line. Used protective clothing is stored on the contaminated side of the change line until collected for laundering. Contamination limits for protective clothing are consistent with the limits in Figure 5.1.

5.2.39 Personnel survey instruments are provided in change rooms and at step-off pads, for use by personnel leaving Contamination Controlled Areas. The instruments are checked, for proper operation, at a frequency approved by the Radiation Safety Function.

5.2.40 Instructions are posted at exit points from Contamination Controlled Areas, which describe survey techniques, procedures for decontamination, and what to do in the event of survey instrument malfunction.

External Exposure:

5.2.41 Adults likely to receive greater than 0.5 REM, in a year from sources external to the body are monitored by personnel dosimeters.

5.2.42 Personnel dosimeters, supplied by a NAVLAP-certified commercial supplier, are issued to trained users to measure external beta-gamma and x-radiation dose.

5.2.43 Neutron detection capability is maintained and evaluated at least quarterly.

5.2.44 Personnel dosimeters are evaluated on a frequency specified by the Radiation Safety Function.

Internal Exposure:

5.2.45 Adults likely to receive greater than 10-percent of the applicable Annual Limit on Intake (ALI) values, are monitored for intakes of radioactive material.

5.2.46 Suitable and timely measurements of radioactive material in work area air, and/or measurements of radionuclides in the body, and/or measurements of radionuclides excreted from the body, are used to monitor intakes by individuals.

- The primary method of determining Committed Effective Dose Equivalent (CEDE) is by measuring the concentration of radioactive material in work area air.
- In-vitro samples, collected during work restrictions, may be used to determine CEDE in place of work area air analysis.

5.2.47 Work restrictions and diagnostic evaluations are initiated when air sample results indicate an individual may have received a single significant intake of:

- Greater than 40 DAC-Hours exposure to non-transportable compounds of uranium.
- Greater than 20 DAC-Hours exposure to transportable compounds of uranium.

5.2.48 Work restrictions without diagnostic evaluations are imposed when individuals exceed administrative limits or 80 % of applicable annual limits (i.e., 0.8 ALI, 1600 DAC-Hours, 4.0 REM CEDE, 4.0 REM TEDE, 4.0 REM DDE, 40 REM CDE, etc.).

5.2.49 Diagnostic evaluations include in-vitro and in-vivo analyses to support air sampling measurements in determining CEDE and to demonstrate compliance with occupational dose equivalent limits in 10CFR20.

5.2.50 A bioassay capability is maintained to evaluate the effectiveness of contamination control and personnel protection practices, to evaluate intakes of radioactive material that exceed action levels in Section 5.2.47; and, to determine compliance with applicable occupational dose equivalent limits.

- The bioassay program conforms to guidance provided in Regulatory Guide 8.9; "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program."
- Routine in-vitro bioassay samples (urinalysis) are collected and evaluated, at least annually, to track and evaluate retention of radioactive material individuals.
- Routine in-vivo bioassay (lung burden) is performed, at least annually, to track and evaluate retention of radioactive material in individuals. In-vitro analysis is used in place of lung burden measurements for claustrophobic individuals.
- Initial baseline and termination bioassay evaluations are performed when pratical.

Calculating Total Dose:

5.2.51 Internal and external occupational radiation doses are combined in accordance with criteria in 10CFR20; and, applicable guidance contained in Regulatory Guide 8.7, "Instructions for Recording and Reporting Occupational Radiation Exposure Data" and Regulatory Guide 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses."

5.2.52 Radiation dose to the embryo/fetus is calculated in accordance with applicable guidance in Regulatory Guide 8.36, "Radiation Dose to the Embryo/Fetus."

Respiratory Protection:

5.2.53 When engineered and/or administrative controls are not practical for protecting individuals from intakes of radioactive material, respiratory protection is provided for use in accordance with an approved policy statement specified by the Radiation Protection Function.

5.2.54 Respiratory protection equipment is used in accordance with written procedures which cover:

- Respirator selection, fitting, issuance, maintenance and testing.
- Supervision and training of personnel.
- Monitoring, including air sampling and bioassay.
- Recordkeeping.
- Use of process or other engineering controls, instead of respirators.
- Routine, non-routine and emergency use of respirators
- Periods of respirator use, and relief from respirator use.

5.2.55 The respirator protection policy includes the following elements.

- Only respiratory devices certified by the National Institute for Occupational Safety and Health / Mine Safety and Health Administration (NIOSH/MSHA) are used.
- Individuals using respiratory protection are trained in accordance with the criteria in 10CFR20, Subpart H.
- Respiratory protection factors from 10CFR20, Appendix A, or more conservative protection factors based on the results of quantitative fit tests, are used when assigning actual radioactive material intakes to individuals.
- Personnel authorized to use respiratory protection equipment are fit-tested annually.
- Personnel authorized to use respiratory protection equipment are trained in the applicable requirements biennially.
- Determination by a physician prior to the initial fitting of respirators, and periodically at a frequency determined by a physician, that the individual user is medically fit to use the respiratory protection equipment.

- Personnel are required to test respirators for operability immediately prior to each use.

Instrumentation:

5.2.56 Instruments used for radiation protection measurements have capabilities to cover the range of use as follows; however, more than one instrument might need to be utilized to cover the specified range:

(a) Portable Survey Instruments:

- Alpha: 100 to 1.0E06 Disintegrations Per Minute;
- Beta-Gamma: 0.1 Millirem per hour to 300 Rem per hour;
- Neutron: 0.5 to 5 Millirem per hour.

(b) Laboratory Assay Instruments:

- Alpha: 10-percent of Derived Air Concentration (DAC) values for sampling periods of 8-hours or more.

5.2.57 Radiation protection instruments are calibrated on a routine schedule established by the Radiation Safety Function. The schedule requires calibration,

- Following initial instrument acquisition.
- ~~Following major repairs.~~
- At least semiannually.

5.2.58 Alpha counting instruments used in the Radiation Safety Laboratory are checked each working day, when in use to determine:

- Background activity.
- Statistical Control using a certified source.

5.2.59 Instrument calibration records are maintained for a period of at least three years.

5.2.60 Operability of portable survey instruments is determined prior to use.

Radiation Safety Analyses

5.2.61 The Radiation Safety Analyses are comprehensive assessments, which identify controls required to maintain an adequate margin of safety.

5.2.62 The analyses consist of individual radiological accident sequences analyzed using the accident flow diagram method. The sequence is traced through the event to arrive at a consequence of interest. Each identified pathway defines an initiating event and protective measure failures that collectively represent an accident sequence.

5.2.63 The Radiation Safety Analyses are one of the evaluation methods of the ISA safety analyses described in Chapter 4.0 of this License Application. The level of detail for a particular analysis is based on the complexity of the initial system, and subsequent proposed changes to the system. Thus, the scope and content of a Radiation Safety Analyses are customized to reflect the particular characteristics and needs of the specific system.

5.2.64 Radiation Safety Analyses are maintained current through implementation of the Configuration Management program described in Section 3.1 of this License Application. The Radiation Safety Analysis portion of a Baseline ISA consists of all documentation that might extend from an original facility Radiation Safety Assessment, through Radiation Safety Evaluations, to the final, fixed in time, ISA document (for which an original ISA Summary was submitted to NRC pursuant to the ISA Plan and Schedule submitted to, and approved by, NRC staff in accordance with Section 70.62(c)(3)(i) of the Part 70 regulation). All subsequent changes that might affect the Baseline ISA are reviewed by the Radiation Safety Function. If a Radiation Safety Analysis is required for the change, it is performed to the same standards required for the baseline analysis. Summary details of the change, including required approvals, are documented on a Configuration Change Control Form that is filed with the applicable Baseline ISA, thus providing a substantially complete "living" framework for the facility radiation safety basis.

Audits and Compliance Inspections:

5.2.65 Program and process assessments are conducted to compare established Radiation Safety standards to CFFF performance.

- Program assessments take the form of program audits.
- The complete Radiation Safety Program is assessed on a triennial frequency.
- Process assessments take the form of compliance inspections that evaluate implementation of radiation safety requirements.
- The complete set of operations making up the CFFF ISA is assessed on a five year frequency.
- Results of the program and process assessments are documented and maintained for NRC Staff review and inspection.

CHAPTER 6.0

NUCLEAR CRITICALITY SAFETY (NCS) PROGRAM

6.1 NCS PROGRAM STRUCTURE

The Columbia Fuel Fabrication Facility (CFFF) maintains a Nuclear Criticality Safety (NCS) Program for the site. A primary purpose of the NCS Program is to designate the controls and barriers that are relied upon to prevent criticality in operations with special nuclear material (SNM). The NCS Program meets the requirements of ANSI/ANS-8.19(1996), as it applies to organization and administration.

6.1.1 General Control Program Practices

The Double Contingency Principle of ANSI/ANS-8.1(1998) is the basis for design and operation of processes using SNM within the CFFF. Double Contingency Protection means that all process designs incorporate sufficient margins of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible. The preferred approach to demonstrate double contingency is to control two independent parameters. In those instances where multiple controls are used to prevent changes in a single parameter (e.g., mass, moderation, or configuration) and Double Contingency Protection exists by way of multiple process upsets before a criticality accident is possible, sufficient redundancy and diversity of controls are used to ensure that at least two process upsets remain independent.

For each process within a system, a defense of one or more controlled parameters is employed and is documented within the process Criticality Safety Evaluation (CSE). The defense consists of the bounding assumptions, criticality safety limits, and criticality safety constraints that, as a set, are uniquely sufficient to maintain the minimum subcritical margin against an initiating event.

CSEs are performed to identify the specific limits and controls necessary for the safe and effective operation of a process. Types of NCS controls and their relative preference for use are described in Section 6.1.2. The NCS controls are included as part of the process design criteria. Passive engineered controls are verified at time of installation and, where appropriate, are entered into the management measures programs for routine inspection and maintenance to assure their reliability and availability. Active engineered controls undergo an operational verification process prior to first use in any system, to assure reliability of intended function, and are entered into the management measures programs for routine testing and maintenance to assure continued availability. Periodic inspection of passive controls, and testing of active controls, is implemented through approved procedures. Any such controls that are not functionally tested or replaced on a regular schedule are specifically identified, and the reason for not testing or routinely replacing is documented. Administrative controls are implemented through approved procedures. The reliability and effectiveness of administrative controls are assured through procedure reviews, training, experience, audits, and compliance inspections.

6.1.2 Control Methods

The relative effectiveness and reliability of NCS controls are considered during the CSE process. Passive engineered controls are preferred over other types of controls, and are used whenever practicable (i.e., when such controls can be implemented, would not cause excessive restriction of operations, and are not cost-prohibitive when compared to their benefits). Active engineered controls are the next preferred method of control. Administrative controls are the least preferred method of control; and, their use is limited to process systems which, in the judgment of the Nuclear Criticality Safety Function, do not provide sufficient benefit for the cost that would be associated with any potential engineered controls. The choice of a particular control will be justified in the appropriate CSE identifying the control. Use of active engineered controls and administrative controls (as opposed to passive engineered controls) will be justified similarly.

(a) Passive Engineered Controls

These are controls that require no operator action or other response to be effective when used to assure nuclear criticality safety. Examples of such controls are favorable geometry equipment and moderation control water barriers.

(b) Active Engineered Controls

These are controls that use a sensed signal or condition to automatically initiate effective actions when called upon to assure nuclear criticality safety. An example of such a control is a shutoff valve actuated by an inline detector signal.

(c) Administrative Controls

These are controls that rely on an operator to perform effective actions to assure nuclear criticality safety. Examples of administrative controls are: actions taken in accordance with a written procedure, verification of information with the assistance of a computer terminal, and actions taken in response to an alarm.

6.1.3 Controlled Parameters

Nuclear criticality safety is achieved by controlling one or more parameters of a system within subcritical limits, with sufficient factors of safety, in conformance with the Double Contingency Principle. Specific controlled parameters that are considered during the CSE process are described below. The following apply to each parameter:

(a) The CSE process is used to identify the significant parameters affected within a particular system.

(b) For each parameter, the optimum (i.e., most reactive) condition for each parameter is assumed, unless 1) it is demonstrated that less reactive conditions are the worst case credible conditions, or 2) appropriate controls (IROFS) are established to maintain the parameter within the assumed limits.

(c) All assumptions relating to process / equipment / material theory, function, and operation (including credible upset conditions) are justified, documented, and independently reviewed. In addition, the most reactive credible dimensional and material composition tolerances are assumed.

Details of the various CFFF systems and their parametric controls are described in the CFFF Integrated Safety Analysis (ISA). Crucial items relied on for safety (IROFS) used to control NCS parameters are listed in the ISA Summary provided for each system. This listing provides the type (passive, active, or administrative) of control, the control's function, and key management measures (availability / reliability tests) applied to each control.

6.1.3.1 Mass

(1) Mass control is used to limit the quantity of uranium within specific process operations or vessels; within storage, transportation, and disposal containers; and within a room or groups of rooms. Mass control is used both on its own and in combination with other parametric controls.

(2) An evaluation to establish mass limits involves consideration of all appropriate criticality safety parameters and will be documented accordingly. The evaluation also considers normal operations and expected process upsets to determine the operating mass limit and the controls necessary to prevent reaching the safety limit. When only administrative controls are used for mass-controlled systems, double batching is generally assumed to be the worst credible single-upset condition, but this must still be justified in the appropriate CSE. Analytical or non-destructive measurement methods are employed to determine the mass of a specific quantity of material.

(3) Whenever mass control is established for a room or group of rooms, detailed records are maintained to document mass transfers into and out of the rooms.

(4) When using a single parameter mass limit derived from experimental data, the mass is limited to no more than 45% of the mass limit when double-batching is credible, and no more than 75% of the mass limit when double-batching is not credible.

6.1.3.2 Moderation

(1) Moderation control is used both on its own and in combination with other parametric controls.

(2) Moderation control includes those controls required to exclude moderator from a system, those controls required to restrict the amount of moderator in a system, and/or those controls required to detect the presence of moderator in a system.

(3) Moderation controls (IROFS) are established to ensure that the interstitial moderator is maintained within the analyzed system's documented limits, for normal operation and expected process upsets. The most reactive credible densities for interstitial moderator are modeled.

(4) When moderation control is used as the sole controlled parameter, the operations are conducted in a "moderator control area," and the guidelines of ANSI/ANS-8.22(1997) are used. In addition, the following requirements are applied:

- Minimum protection requires that two independent barriers (e.g., roofs) must fail before moderation control can be compromised. Management measures to maintain the quality of a barrier, including routine inspections, are required. All outermost barriers are tested for leakage as part of initial barrier installation.

- Procedures are established to control the introduction of moderating materials to, and use of moderating materials in, areas under moderation control. Procedures are established to ensure removal of all uncontrolled or unauthorized moderator prior to releasing a moderation controlled system for production. Procedures are established to prevent uncontrolled or unauthorized moderator from entering the system after special nuclear material (SNM) loading (e.g., into a bulk container) has occurred.

- Two independent measurements (i.e., two separate samples measured on two different instruments, or on the same instrument but separated by a standard control check), and/or two independent samples (i.e., two samples taken by two different people at different times using different sampling methods), are used to establish material moderator content. The process for sample collection, preparation, analysis, and posting of results is designed to ensure the results obtained are independent.

- Procedures are established for transportation of moderation controlled materials outside of moderator control areas. The basis for selection of route barriers, to prevent accidental exposure to moderators, will be documented within the applicable CSEs. Management measures to maintain the quality of route barriers, particularly routine inspections, are required.

(5) When moderation control is used in addition to one or more other controlled parameters, the guidelines of ANSI/ANS-8.22(1997) are used, with one exception: a "moderator control area" will not be formally designated, in order to avoid diluting the significance of the designation, with respect to processes that rely only on moderation control.

6.1.3.3 Concentration

(1) Concentration control is used both on its own and in combination with other parametric controls.

(2) Concentration controls (IROFS) established to maintain a system within documented limits will be evaluated in a CSE and shown to be reliable and independent.

(3) The determination of concentration limits and controls will consider precipitation, evaporation, freezing, settling, heterogeneity and chemical phase change events.

(4) When determining concentration, and concentration is the only controlled parameter, two independent controls/measurements, or the analysis of two independent samples (taken by two different people or instruments), will be used. As required by the implementing CSE, sample analysis or measurement will be performed by two different instruments, or by the same instrument separated by a standard control check.

6.1.3.4 Geometry / Volume

(1) Geometry control is used to limit the shape, configuration or volume of SNM within specific process operations and vessels; and, within storage transportation, and disposal containers. Geometry control is used both on its own and in combination with other parametric controls.

(2) Definitions for achievement of geometry control:

- Favorable geometry means establishing the characteristic dimensions of importance for a single unit of a specified shape such that criticality safety will be maintained in conjunction with one or more other constraints (e.g., material form, material concentration, reflection, enrichment, etc.). At the CFFF, the other parameter constrained is often enrichment. Since enrichment will be maintained at or below the maximum licensed enrichment for CFFF, such favorable geometry dimensions are considered the equivalent of safe geometry dimensions.

- Safe geometry means establishing the characteristic dimensions of importance for a single unit such that criticality safety will be maintained without any other constraints.

- Level control means detecting (e.g., through use of level probes) or removing (e.g., through use of overflow holes or slots) material in/from a non-favorable geometry vessel at a specific level, such that material accumulation within the vessel is limited to a favorable height. When level is credited as a controlled parameter, appropriate analyses will be performed to demonstrate the adequacy of the controls.

(3) Geometry controlled systems are analyzed and evaluated for fabrication tolerances and dimensional changes that might occur through corrosion, wear, or mechanical distortion.

(4) When using critical dimension limits derived from experimental data, the margins of safety are no more than 90% of the critical cylinder diameter, 85% of the minimum critical slab thickness, and 75% of the minimum critical sphere volume.

(5) Geometry controls will be maintained through management measures that include procedure reviews, training, experience, audits, and compliance inspections. Where appropriate, passive geometry controls are entered into the management measures programs for routine inspection and maintenance to assure their reliability and availability.

6.1.3.5 Material Composition and Process Characteristics

(1) Within specific manufacturing operations, credit is taken for physical and chemical properties of the process, and/or materials in the process, as nuclear criticality safety controls.

(2) When credit is taken for process characteristics, the bounding assumptions and process / operational limits are documented in the applicable CSE and are communicated to cognizant operations personnel through training and procedures.

(3) Utilization of process and/or material characteristics as controls is based on known scientific principles, established physical properties or chemical reactions, and/or experimental data supported by CFFF operational history.

(4) The applicable CSE for each system documents the effects of material composition within the process being evaluated and documents the basis for composition selection in subsequent system modeling for analysis.

6.1.3.6 **Enrichment**

(1) Enrichment control is used in combination with all other parametric control methods.

(2) Control of enrichment to less than the licensed limit is used to limit the percent of U-235 in a process, vessel, or container. Active engineered and/or administrative controls are required to verify enrichment, and to prevent the introduction of uranium at unacceptable enrichments, within the defined system.

6.1.3.7 **Heterogeneity**

(1) When applicable, significant effects of material heterogeneity within a system are documented within the applicable CSE.

(2) Nuclear criticality safety calculations have demonstrated that for particle sizes $\leq$ 150 microns in diameter, the material can be considered homogeneous.

(3) For particle sizes greater than 150 microns in diameter, an evaluation will take into account the effects of heterogeneity specific to the process being analyzed.

6.1.3.8 **Neutron Absorbers**

(1) Neutron absorbing materials (aka "poisons") are used to provide nuclear criticality safety control for processes, vessels, and containers. When so used, the absorbers will be solid (i.e., fixed) materials (e.g., borosilicate-glass Raschig rings, gadolinium plates, borated stainless steel, etc.) or solution (e.g., boric acid with a minimum concentration to assure adequate subcriticality).

(2) When Raschig rings are used, their use and maintenance is in accordance with ANSI/ANS-8.5(1996), with the following exceptions (for use in basic solutions):

- System pH is maintained $\geq$ 7, but $\leq$ 11.
- System temperature is maintained $\leq$ 60 degrees (Celsius).

(3) For fixed absorbers other than Raschig rings, in addition to the guidance of ANSI/ANS-8.21(1995), the following requirements apply:

- The absorber composition is measured, and documented in the applicable CSE, prior to first use.

- The presence and condition of the absorber in the process, vessel, or container is verified on a frequency documented in the applicable CSE. Methods of verification include traceability (e.g., unique serial number), visual inspection, and/or specific measurements.

6.1.3.9 Reflection

Credible reflection conditions will be considered in the determination of all system limits and controls. The terms "full reflection" and "partial reflection" are defined as 12-inches and 1 inch of water equivalent, respectively. Other potential reflection conditions will be evaluated and justified, as appropriate. When less than full reflection is assumed, it shall be demonstrated that the reflection conditions modeled are the most reactive credible conditions, or appropriate controls (IROFS) will be established to maintain reflection within the applicable limits.

6.1.3.10 Interaction / Spacing

NCS analyses will consider the potential effects of interaction. The following general guidance will be utilized in the evaluation:

- Units may be considered non-interacting when they are separated by a 12-foot air distance or by 12 inches of full density water equivalent material.
- The interaction of units not meeting the above criteria will be evaluated using approved and validated methods. This includes calculations with validated computer codes (XSDRN, KENO, MCNP, etc.), standards (ANS-8 series limits) and approved hand calculation methods.

Spacing controls will be maintained through management measures that include procedure reviews, training, experience, audits, and compliance inspections. Where appropriate, passive spacing controls are entered into the management measures programs for routine inspection and maintenance to assure their reliability and availability.

6.1.4 Criticality Safety Documentation

6.1.4.1 Criticality Safety Calculation Notes (Calc Notes)

(1) Calc notes may be used to document criticality safety computer and hand calculations.

(2) Calc notes can be referenced in CSEs.

(3) Calc notes can be used to document parametric studies that may be referenced by multiple CSEs.

6.1.4.2 Criticality Safety Evaluation (CSE)

(1) The CSE is a comprehensive nuclear criticality safety evaluation of each component within a defined system. The evaluation identifies controlled parameters for the system, establishes bounding assumptions for other system parameters, and identifies the Safety Significant Controls necessary to ensure double contingency. Calculations and sensitivity studies are performed as necessary to identify the margin of subcriticality.

(2) The CSE serves as the primary documentation that Double Contingency Protection exists for the system, when controls are applied to the parameters that prevent each contingency from occurring.

(3) In the CSE, the reliability of each control is evaluated, and potential common mode failures are considered. Margin of safety is also addressed.

(4) As part of the CSE process, criticality accident sequences are evaluated by teams of process, operations and criticality safety experts. These accident sequence evaluations are documented in the CSE and serve as input to the ISA fault trees that are used to demonstrate that each accident sequence is highly unlikely to occur.

(5) CSEs are performed in accordance with guidelines provided in the CFFF procedure for CSE generation.

(6) CSEs must be reviewed by a qualified Criticality Safety Technical Reviewer (see Section 6.1.6), and must be approved by Criticality Safety management and appropriate plant operations management, or designates.

(7) CSEs serve as the "living" documentation of the plant criticality safety basis and, as such, are maintained current through implementation of the CFFF Configuration Management program.

(8) "Record" copies of CSEs must be maintained in accordance with CFFF document control requirements.

6.1.5 Analytical Methods

Validated computation methods are used to calculate the k_{EFF} of individual pieces of equipment, and to calculate equipment interactions. Conditions evaluated include normal operations, anticipated process upsets, and credible abnormal operations. When using nationally-accepted standards or handbook data, appropriate margins will be employed as dictated by the requirements of the process. If the data is not from a nationally-recognized source, appropriate validation of the data will be performed before it is employed in a CSE.

6.1.5.1 Analytical Codes

Criticality safety calculations are performed using the approved and validated computer codes such as SCALE, MCNP, XSDRN, etc.

6.1.5.2 Limits of k_{EFF}

Based on the results of calculations, the sensitivity of key parameters are evaluated to determine the effect on k_{EFF}, and to assure that adequate controls have been provided to demonstrate a sufficient margin of safety for the analyzed system.

(1) For normal operations, and anticipated process upsets, a sufficient margin of safety is defined as a 95/95 k_{EFF} that is = 0.95 when all applicable biases and computational uncertainties are taken into account.

(2) For credible abnormal configurations a sufficient margin of safety is defined as a 95/95 k_{EFF} that is = 0.98 when all applicable biases and computational uncertainties are taken into account.

(3) A 95/95 k_{EFF} that includes all applicable biases and computational uncertainties is demonstrated using the following equation:

$$95/95\ k_{EFF} = k_s + 2s_s + (\text{bias} + \text{uncertainty})$$

where:

k_s is the calculated multiplication factor, using a validated computation method;

s_s is the k_s standard deviation for that computation method; and,

(bias + uncertainty) is the appropriate value from the validation performed for that computation method, determined as described in Subsection 6.1.5.3. Note that a negative bias will not be credited (i.e., a bias that reduces the value of the calculated k_{EFF}).

6.1.5.3 Validation Techniques

Computational methods will be validated in accordance with guidelines of ANSI/ANS-8.1(1983). The specific validation method used will be described in the appropriate validation documents, and may include nationally-recognized methods such as those documented in NUREG/CR–6361 ("Criticality Benchmark Guide for Light Water Reactor Fuel in Transportation and Storage Packages") and NUREG/CR-6698 ("Guide for Validation of Nuclear Criticality Safety Calculational Methodology").

Validation reports will be prepared, reviewed, and approved by qualified individuals for each combination of computational method (e.g., code), cross-section library, computer platform, and analytical area of applicability (e.g., homogenous UO_2 versus heterogeneous UO_2), as appropriate. In all cases, each validation report, or the calculation note documenting an analysis using a specific computational method, shall include the following:

(1) demonstration of the adequacy of the margin of safety for subcriticality by assuring that the margin is large compared to the uncertainty in the calculated value of k_{EFF}

(2) demonstration that the calculation of k_{EFF} is based on a set of variables whose values lie in a range for which the methodology used to determine k_{EFF} has been validated; or demonstration that trends in the bias support the extension of the methodology to areas outside the areas of applicability.

(3) a description of the specific validation method used, including reference to input data, area of applicability, and discussion of the applicable uncertainties.

6.1.5.4 Computer Hardware and Software Control

(1) Validation and verification are completed, documented and independently reviewed before:

- Use of specific hardware and software systems utilizing specific cross section libraries;
- Use of analytical codes;
- Use of the methodology; and,
- Qualification and re-qualification of the codes.

(2) The configuration of the hardware platform used in support of software for criticality safety calculations is maintained such that only authorized system administrators are allowed to make system changes. System changes are conducted in accordance with an approved configuration control program that addresses both hardware and software qualification. System operability verification is used for alerting users to any changes that might impact the operation of codes on the platform.

(3) Software on the platform that is designated for use in criticality safety calculations is compiled into working code versions, with executable files that are traceable with respect to length, time, and version.

(4) Modifications to hardware or software that are essential to the calculation process are followed by code operability verification. In such cases, selected calculations are performed to verify results are not substantially different to those from pre-modification analyses. Any deviations disclosed by code verification, that might alter the bias or uncertainty; require re-qualification of the code prior to continued use.

6.1.6 Technical Review

A qualified NCS technical reviewer (TR) performs an independent verification of all criticality safety evaluations and calculations that support limits specified in a safety analysis. The TR verifies that a proposed calculation geometry model and configuration adequately represents the system being analyzed. The TR also verifies that proposed material characterizations (e.g., density, concentration, etc.) adequately represent the system. The minimum required qualification for a TR will be identified in appropriate CFFF procedures.

The verification of such evaluations and calculations uses one (or more) of the following processes:

(1) Verification using an alternate computer code and/or hand calculations.

(2) Verification by performing a comparison with prior results for a similar, approved calculation and/or a similar configuration.

(3) Verification by using a technical verification checklist, including checks of the computer code used, and evaluation of code input and output.

(4) Verification using a custom method, including detailed information that describes the custom methodology.

6.1.7 Posting of Limits and Controls

Posting includes placement of signs and/or physical identification (e.g., using tape, paint, etc.) of floors, to designate approved work and storage areas. Postings provide information and/or specific precautions to supplement operating procedures.

Appropriate postings are placed at the entrance to work and holding areas (e.g. equipment, rooms, etc.) where fissile material is processed or stored. Criticality safety precautions or prohibitions (e.g., approved moderator limits, approved fire-fighting methods, etc.) are posted at entrances to affected areas. Storage postings are conspicuously located at entrances to holding areas (i.e., at such locations that it would be unlikely that personnel could enter an area without seeing the postings); and, include (as applicable) information such as material type, container identification, number of containers allowed, controlled parameter limits, and spacing requirements.

Postings are approved and issued by the Nuclear Criticality Safety Function. First level managers are responsible for assuring that their cognizant personnel are aware of, and understand, posted information.

6.1.8 Criticality Accident Alarm System (CAAS)

The CAAS initiates immediate evacuation of the facility in response to detection of a potential criticality accident. The CAAS, and the proper response protocol, is detailed in the CFFF Emergency Plan and Emergency Procedures.

The CAAS radiation monitoring detectors are located to pursue conformance to the guidance of ANSI/ANS-8.3(1997) (as modified by Regulatory Guide 3.71), and compliance with 10CFR70.24. Location and spacing of the detectors are chosen to minimize the effect of shielding by massive equipment or materials of construction. Spacing is reduced where high-density materials (e.g., concrete, cinder block, brick, etc.) are located between a potential accident source and a detector. Low-density materials (e.g., wooden construction walls, non-load walls, office panel walls, metal-corrugated panels, doors, plaster, etc.) are disregarded when determining CAAS spacing.

If the CAAS is out-of-service for more than four hours, all movement and processing of fissile material is prohibited in the affected area until the alarm service is restored; or, an equivalent level of protection (e.g. continuously attended portable detection instruments, with the capability to issue area-wide emergency communications), approved by the Nuclear Criticality Safety Function, is provided. Four hours without CAAS coverage is considered an acceptable risk, since all criticality accident sequences are demonstrated to have frequencies no greater than highly unlikely, and thus the probability of having an inadvertent criticality during any four hour interval is considered to be incredible. Suspension of fissile material movement and processing will be directed and enforced by the plant emergency response team. Routine testing, calibration, and/or maintenance of the system is permitted without suspension of fissile material movement or processing.

Employees and visitors are trained in responding to the alarm signal, which is a continuous warbling siren. An ongoing aspect of this training is a weekly test of the signal on all working shifts.

6.1.9 Audits and Compliance Inspections

Program and process assessments are conducted to compare established NCS standards to CFFF performance. The assessments take the form of program audits and compliance inspections, as described in Chapter 3.0, Section 3.6. These assessments meet the guidelines of ANSI/ANS-8-19(1996), as it relates to audits and assessments.

Program assessments take the form of program audits. Specific portions of the NCS program evaluated during a particular assessment are based on previous internal audit findings, external audit findings, NRC inspection activities, current operating conditions, and time since last assessment. Program audits schedules are developed annually, with the complete NCS program assessed on a triennial frequency. Results of the assessments are documented and maintained for NRC Staff review and inspection.

Process assessments take the form of compliance inspections that evaluate implementation of NCS requirements (e.g., conformance to the applicable CSE container spacing, following procedures and postings, etc.) for CFFF operations. Frequency of inspections is based on previous internal inspection findings, NRC inspection results, incidents (those reported, and those requiring notification), configuration management activities, and time since last assessment. Formal compliance inspection schedules are developed annually, with the complete set of operations making up the CFFF ISA assessed on a five-year frequency. Results of the assessments are documented and maintained for NRC Staff review and inspection.

6.1.10 Procedures, Training, and Qualification

At the CFFF, procedures, training and qualification are integrated into a combined process to assure that safety and safeguards activities are being conducted by trained and qualified individuals, in accordance with Westinghouse policies and in accordance with commitments to Regulatory Agencies. This process is described in Chapter 3, Section 3.4, and meets the guidelines of ANSI/ANS-8.19(1996) and ANSI/ANS-8.20(1991), as they relate to training, procedures, and the requirement that no single, inadvertent departure from a procedure could cause an inadvertent criticality.

CHAPTER 7.0

CHEMICAL SAFETY PROGRAM

7.1 CHEMICAL SAFETY PROGRAM STRUCTURE

The Columbia Fuel Fabrication Facility (CFFF) maintains a Chemical Safety Program for the site. A primary purpose of the Chemical Safety Program is to assure that exposure of workers to hazardous chemicals, in particular those that contain licensed nuclear material or are produced from licensed nuclear material, is kept As Low As Reasonably Achievable (ALARA). An extensive detail of how much of this is done in practice is documented in an Integrated Safety Analysis (ISA) and ISA Summary titled "Chemicals Receipt, Handling, and Storage Systems."

7.1.1 Program Basis

7.1.1.1 Chemical Safety Program activities are spread out among various CFFF organizations, procedures, manuals and other documentation. This widespread approach demonstrates how chemical safety concepts are incorporated into all aspects of CFFF activities, at all levels of the organization.

7.1.1.2 The Process Safety Management (PSM) regulation (29 CFR 1910.119) is the basis for CFFF Chemical Safety Program elements for all consequence levels (low, intermediate, and high).

7.1.1.3 The Chemical Safety Program addresses the following elements:

(a) Employee Participation;
(b) Policies and Programs;
(c) Organization and Responsibilities;
(d) Inspections, Audits and Appraisals;
(e) Design Base Documentation;
(f) Process Hazard Analysis;
(g) Operating Procedures;
(h) Training;
(i) Maintenance and Surveillance;
(j) Chemical Storage and Handling;
(k) Chemical Release and Response;
(l) Hazard Communication;
(m) Contractors;
(n) Pre-Startup Safety Review;
(o) Hot Work Permit;
(p) Management of Change;
(q) Incident Investigation;
(r) Receipt and Shipment of Chemicals;
(s) Hazardous Waste and Chemical Disposal;

(t) Trade Secret;
(u) Fire Prevention;
(v) Chemical Labeling; and,
(w) Medical Services and First Aid.

7.1.1.4 A cross reference matrix is maintained that identifies the specific elements of the Chemical Safety Program and links them to applicable compliance documentation.

7.1.2 Program Practices

7.1.2.1 The CFFF Chemical Safety Program is designed to assure that all processes and operations comply with applicable federal and state regulations pertaining to chemical safety.

7.1.2.2 The Chemical Safety Program is implemented to assure that hazards associated with the risk posed by chemicals used at the CFFF are evaluated, and that appropriate measures are taken to assure all operations are performed in a safe manner.

7.1.2.3 Appropriate facilities, equipment, and procedures for the safe storage and handling of hazardous chemicals are maintained at the CFFF. Face velocity requirements for enclosures whose primary control function relates to chemical fumes, mists, and dusts are specified by the Chemical Safety Function.

7.1.2.4 Employees using hazardous chemicals are specifically trained in procedures for safe handling and disposal of them.

7.1.2.5 The Chemical Safety Program includes evaluations of:

(1) Potential physical, chemical, and/or fire hazards;

(2) Development and implementation of safety programs and procedures designed to minimize accidents and injuries to employees;

(3) Purchase and maintenance of protection and monitoring equipment; and,

(4) Maintenance of appropriate records and reports.

7.1.2.6 The Site Emergency Plan and Implementing Procedures, described in Chapter 9.0 of this License Application, detail the manner in which the CFFF responds to any accidental release of hazardous chemicals.

7.1.3 Performance and Documentation of Analyses

7.1.3.1 Hazard and Operability (HAZOP) Analysis, What-If/Checklist, and/or other recognized methods are used to systematically evaluate the safety of chemical operations at the CFFF. The hazard evaluation method selected is based on the complexity of the process being analyzed.

7.1.3.2 Hazards to be evaluated are based on the nature of the chemicals involved, the process conditions (flow, temperature, pressure, concentration, etc.), personnel experience, and information about previous incidents in the facility. The evaluation is used to ensure that adequate safety margin is present in each chemical process. For areas where additional safety controls might be required, an action plan is developed for increasing the safety margin of the process, in accordance with CFFF priorities and resources.

7.1.3.3 The physical design and implementation of chemical operations at the CFFF is evaluated to identify deviations from the intended operation, which could result in potential hazards or operational concerns. These hazards include the following, when applicable:

(1) Potential for criticality safety incidents;
(2) Potential to violate a License commitment;
(3) Potential for personnel exposure or injury; and/or,
(4) Potential for radioactive contamination, release of chemicals to the atmosphere, fire or explosion.

7.1.3.4 Chemical Safety Analysis

(a) Analysis Performance

(1) The Chemical Safety Analysis is a comprehensive assessment of each component within a defined system. The analysis identifies controls required to maintain a sufficient margin of safety.

(2) Chemical accident sequences are analyzed using the accident flow diagram format. In this format, the analyst traces each sequence through the diagram (starting with the initiating event) to arrive at a consequence of interest. Each identified pathway defines an initiating event and protective measure failures that collectively represent an accident sequence.

(b) Analysis documentation

(1) The Chemical Safety Analysis is one of the ISA safety analyses described in Chapter 4.0 of this License Application. The level of detail for a particular analysis is based on the complexity of the initial system, and subsequent proposed changes to the system. Thus, the scope and content of a Chemical Safety Analysis are customized to reflect the particular characteristics and needs of the system being analyzed.

(2) Chemical Safety Analyses are maintained current through implementation of the Configuration Management program described in Sections 3.1 and 4.1 of this License Application. If a Chemical Safety Analysis is required for a proposed change, it is performed to the current standards required for the baseline analysis. Summary details of the change, including required approvals, are documented on a Configuration Change Control Form that is filed with the applicable Baseline ISA, thus providing a substantially complete "living" framework for the facility chemical safety basis.

7.1.4 Audits and Compliance Inspections

Program and process assessments are conducted to compare established chemical safety standards to CFFF performance. The assessments take the form of program audits and compliance inspections, as described in Chapter 3.0, Section 3.6, of this License Application.

7.1.4.1 Program assessments take the form of program audits. Specific portions of the Chemical Safety Program, evaluated during a particular assessment, are based on previous internal audit findings, external audit findings, NRC inspection activities, current operating conditions, and time since last assessment. The Chemical Safety Program is assessed on a triennial frequency. Results of the assessments are documented and maintained for NRC Staff review and inspection.

7.1.4.2 Process assessments take the form of compliance inspections that evaluate implementation of chemical safety requirements (*e.g.*, personal protective equipment, following procedures and postings, *etc.*) for CFFF operations (*i.e.*, Site and Structures, ADU Conversion, Solvent Extraction, *etc.*). The frequency of inspections is based on previous internal inspection findings, NRC inspection results, incidents (those reported, and those requiring notification), configuration management activities, and time since last assessment. The complete set of operations making up the CFFF ISA is assessed on a five year frequency. Results of the assessments are documented and maintained for NRC Staff review and inspection.

CHAPTER 8.0

FIRE SAFETY PROGRAM

The Columbia Fuel Fabrication Facility (CFFF) maintains a robust Fire Safety Program for protection of the site. A primary purpose of this Fire Safety Program is to assure that the opportunity for fires in and about the facility is kept As Low As Reasonably Achievable (ALARA). Fire protection is achieved by combinations of fire protection measures and systems. Such measures and systems are designed and maintained in accordance with industry standards and prudent industry practices. The standards and practices most often consulted are those of the National Fire Protection Association (NFPA).

8.1 FIRE SAFETY PROGRAM STRUCTURE

In the early 1990's, a multi-component, Engineering and Regulatory team was empowered by facility management to formally evaluate the CFFF Fire Safety Program, using as guidance the Nuclear Regulatory Commission (NRC) *Branch Technical Position on Fire Protection for Fuel Cycle Facilities*. The team provided the results of the evaluation, and a proposed program structure, to Engineering and Regulatory Component management. Based on this evaluation, with input from prior and subsequent evaluations, the Fire Safety Program has been basically defined as consisting of the following elements:

8.1.1 Basic Fire Protection

8.1.1.1 Fire Safety Program management organization, authorities, and responsibilities conform to the structure presented in Chapter 2.0 of this License Application.

8.1.1.2 The CFFF is designed to provide protection against fires and explosions that could affect the safety of licensed materials and thus present an increased radiological risk.

8.1.1.3 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. Criticality concerns/controls restrict use of water for fire suppression in identified plant areas.

8.1.1.4 The Security Function is responsible for announcing alarms and alerting personnel to fire incidents through use of the facility public address system. Following announcement of an alarm, instructions are given to instruct personnel of any necessary protective actions to be taken.

8.1.1.5 An approved cutting and welding procedure, welder training program, and hot work permits are provided to control torch use activities.

8.1.1.6 Flammable liquids are retained in containers and/or cabinets designed for such purpose, and additional precautions are taken as specified by the Fire Safety

Function. Non-routine use of flammable materials is controlled by the same precautions used for routine use of such materials.

8.1.1.7 Periodic fire emergency drills are conducted as part of the Emergency Management Program described in Chapter 9.0 of this License Application. An emergency exercise, that includes facility evacuation, is conducted on a biennial basis. At times prescribed by the Fire Safety Function, a fire scenario is included as part of such an exercise.

8.1.1.8 Review and control of modifications of the facility or processes to minimize fire hazards is implemented as described in Section 3.1 of this License Application.

8.1.1.9 A fire protection preventive maintenance program is in place, and relevant documentation is maintained for the maintenance activities, as described in Section 3.2 of this License Application. Inspection, testing, and maintenance of fire protection equipment is covered by this program.

8.1.1.10 The initial CFFF fire hazard analysis is documented in the *Westinghouse Nuclear Fuel Columbia Site Evaluation Report* (March, 1975). A supplement to this analysis is documented by Impell Corporation in the *Fire Hazard Analysis for Westinghouse Nuclear Fuel Columbia Plant* (June, 1987). Current fire hazard analyses are found in the Pre-Fire Plans for the various areas of the facility and in the ISA Fire Safety Analyses, as described in Chapter 4.0 of this License Application. Fire safety controls, instruments, and services are included in the Quality Assurance Program as described in Section 3.3 of this License Application.

8.1.1.11 Basic fire protection training is covered in new-hire and contractor orientation programs as described in Section 3.4 of this License Application. An Emergency Response team is given extensive additional training.

8.1.1.12 Approved procedures, as described in Section 3.4 of this License Application, define reporting guidelines and investigation requirements for fire incidents.

8.1.1.13 Approved procedures also prescribe the housekeeping practices for the facility. Good housekeeping techniques are practiced at the facility as an integral part of the Human Performance culture described in Section 3.5 of this License Application.

8.1.1.14 The Fire Safety Program is periodically evaluated through program audits, compliance inspections and self-assessments, as described in Section 3.6 of this License Application. Resolution of significant findings is tracked by the Corrective Action Program, as described in Section 3.8 of this License Application.

8.1.1.15 A formal system is provided to enable reporting of fire incidents to First Level Management for action, as described in Section 3.7 of this License Application.

8.1.1.16 Fire Safety Program records are maintained, as described in Section 3.9 of this License Application.

Details of these and other Fire Safety Program elements are presented in the balance of this Section.

8.1.2 Building Construction

The construction standards for the CFFF manufacturing areas were those that prevailed at the time the areas were originally constructed. The building structural members were built using non-combustible, or limited combustible materials. Whenever the building structure is expanded, or otherwise modified, prevailing NFPA code requirements are met.

8.1.2.1 To minimize exposure to fire risk, the facility employs guidance from applicable NFPA standards.

8.1.2.2 To enable rapid personnel egress from buildings in the event of a fire, the facility employs guidance from the NFPA 101 standard.

8.1.2.3 Electrical installations and wiring also conform to applicable industry standards, e.g., NEPA 70.

These areas will conform to the following, as specified by the Fire Safety Function:

(1) location and manning requirements;
(2) fire barrier ratings;
(3) fire detection requirements;
(4) sprinkler, or other fire suppression method, specifications;
(5) container and containment specifications;
(6). wiring grades;
(7) combustible material inventory controls; and/or,
(8) housekeeping practices.

8.1.2.4 Smoke vents are located in the mechanical manufacturing areas.

8.1.2.5 Hidden (concealed) spaces are routinely checked as part of monthly compliance inspections.

8.1.2.6 Water drainage is addressed by properly sized floor drains; and, sumps are installed where specified by the Fire Safety Function.

8.1.2.7 Lightning protection of steel buildings is maintained by use of grounding straps; and, equipment specified by the Fire Safety Function is also grounded.

8.1.3 Ventilation System

8.1.3.1 Facility heating and ventilation systems are designed for fire protection.

8.1.3.2 Space heating furnaces are built to industry and NFPA 70 standards.

8.1.3.3 Fire barrier penetrations employ fire dampers designed to specifications.

8.1.3.4 Automatic closing is required for fire doors and dampers.

8.1.3.5 Class 1 UL-586 (or equivalent) final HEPA filters are used.

8.1.4 Process Fire Safety

8.1.4.1 Principal chemicals used at the facility are evaluated for their fire hazards, and their control is specified by the Fire Safety Function. In particular, the following chemicals are so controlled:

(a) Anhydrous ammonia;
(b) Hydrogen;
(c) Nitric acid;
(d) Sulfuric acid;
(e) Natural gas; and
(f) Fuel oil- diesel.

Uses of such chemicals conform to:

(1) hazard recognition by handlers;
(2) training in safe handling and spill prevention techniques;
(3) storage;
(4) containment;
(5) maintenance;
(6) leak testing; and/or,
(7) safety shut-off valve verifications,

as specified by the Fire Safety Function.

8.1.4.2 Processes involving use of flammable gases are not introduced to the facility until they are evaluated, and their controls have been specified by the Fire Safety Function. In particular, the following controls are applied to flammable gas processes:

(a) Construction, installation, operation and maintenance of bulk gas storage, loading and dispensing systems are in accordance with prudent industry practice;

(b) Combustible gas analysis is performed prior to hot (open flame) work, as specified on work permits;

(c) Sintering furnaces are provided with flame curtains designed to continually burn off excess hydrogen gas upon release of furnace atmosphere. Process interlocks are employed to assure proper operation of the flame curtains; and,

(d) Sintering furnaces have been upgraded to meet the NFPA 86 standards in effect at the time of the upgrade.

8.1.4.3 Processes involving use of flammable and combustible liquids are not introduced to the facility until they are evaluated, and their controls have been specified by the Fire Safety Function. In particular, the following controls are applied to flammable and combustible liquid processes:

(a) Flammable and combustible liquid storage systems are designed and maintained as specified by the Fire Safety Function;

(b) Construction, installation, operation and maintenance of bulk liquid storage, loading and dispensing systems are in accordance with prudent industry practice;

(c) Above ground storage tanks are provided with emergency relief vents in accordance with industry standards;

(d) Supports for aboveground storage tanks are protected from potential exposure to fires; and,

(e) Indoor storage of flammable and combustible liquids is evaluated and appropriate fire extinguishers are kept immediately available.

8.1.4.4 The fire hazard in handling of uranium oxides has been evaluated. Non-combustible materials are specified for powder handling systems where the potential for spontaneous exothermic reaction needs to be considered. Where high density polypropylene containers are used for storage and transport of active uranium oxides, operators are trained to recognize hazardous powder characteristics and are instructed how to monitor for exothermic reactions in such containers.

8.1.4.5 Machining operations on combustible metals at the facility are evaluated for their fire hazards, and appropriate controls are specified by the Fire Safety Function. In particular, the following operations involving potential for zirconium metal fines are controlled by approved procedures:

(a) Fuel rod repair stations;
(b) Final fuel assembly loaders;
(c) Laser welders;
(d) Zirconium grid strap production areas;
(e) Mechanical development laboratories; and,
(f) Tool rooms.

Such areas conform to containment, ventilation, filtration and/or fire extinguisher requirements, as specified by the Fire Safety Function.

8.1.4.6 The Facility Incinerator

The facility incinerator is isolated from the rest of the facility by a rated fire barrier. Incinerator exhaust is passed through a water media for cooling and dust separation. The exhaust is then routed through a filtration and sampling system prior to release to the environment.

8.1.4.7 Boilers and boiler-furnaces are evaluated, and their controls are specified by the Fire Safety Function. In particular, the following controls have been applied:

(a) Boilers are contained in non-fire-rated boiler houses that are physically separated from manufacturing buildings;

(b) Fuel storage tanks are separated from boiler houses; and, fuel lines are marked for identification and are located to minimize damage potential; and,

(c) Construction and operation of boiler-furnaces is in accordance with industry standards.

8.1.4.8 Stationary combustion engines are evaluated, and their controls are specified by the Fire Safety Function. In particular, the following controls have been applied:

(a) Stationary combustion engines are located in rooms constructed of non-combustible materials;

(b) Engine exhaust systems are designed to prevent ignition of combustible material by contact with hot metal surfaces, or by leaking exhaust gases or sparks;

(c) Engine rooms are configured such that process-generated dusts and flammable vapors cannot enter;

(d) Engine rooms are ventilated to minimize accumulation of combustible vapors. The ventilation systems are automatically activated when engines are started;

(e) Emergency generator areas located inside the main building are protected by a sprinkler fire suppression system; and,

(f) Fire pump storage tanks are constructed in accordance with industry standards.

8.1.4.9 Hoods and gloveboxes have been evaluated for fire hazards, and their controls are specified by the Fire Safety Function. In particular, the following controls have been applied:

(a) Hoods and gloveboxes are constructed primarily of metal, using glass and/or fire resistant plastic for viewing areas. The plastic conforms to a Class-I fire rating; and,

(b) Explosive mixtures in gloveboxes are prevented, using inert gas or dry air atmospheres when required.

8.1.4.10 Fire protection methods for laboratories handling radioactive materials are in accordance with industry standards.

8.1.5 Fire Detection and Alarm Systems

8.1.5.1 Automatic fire detectors are installed in areas with a substantial combustible loading and/or in areas with infrequent occupancy, as specified by the Fire Safety Function, unless such areas are covered by automatic fire suppression systems.

8.1.5.2 Automatic flammable vapor/gas detectors are installed for hydrogen systems, unless such systems have been evaluated and it has been determined by the Fire

Safety Function that potential for leakage is minimal and/or sufficient dilution air is present to prevent formation of explosive mixtures.

8.1.5.3 Audible fire alarms are installed in locations throughout the facility, and supplementary visual alarms are installed in high noise areas, as specified by the Fire Safety Function. These alarms are supervised by a continuously manned, central control station that monitors fire detection system and zone status.

8.1.5.4 Manual fire alarm actuators (pull-boxes) are installed in specified locations throughout the facility, as specified by the Fire Safety Function.

8.1.6 Fire Suppression Equipment and Services

8.1.6.1 Fire Suppression Equipment

(a) 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 (*e.g.*, risk of an accidental criticality, or substantial electrical hazard).

(b) Multiple 6-inch fire hydrants, with 2.5-inch hose connectors, are located at strategic locations about the facility site.

(c) Multiple 1.5-inch standpipes are strategically located throughout the facility. Standpipe and hose systems are selected and designed in accordance with industry standards. Standpipe and hose systems have readily accessible hose outlet locations.

(d) Automatic sprinkler systems are selected and designed in accordance with industry standards. Automatic sprinkler systems are specifically excluded from areas where moderation control is specified by the Nuclear Criticality Safety Function as a principal controlled parameter, and/or in areas with a high concentration of energized electrical equipment.

(e) Portable fire extinguishers, with sufficient capacity and proper type of suppression agent, are available and maintained throughout the facility. Portable fire extinguishers are selected and deployed in accordance with industry standards.

8.1.6.2 Fire Suppression Services

(a) Water supply for fire protection systems is assured. The 10-inch water main that supplies process and drinking water to the site also supplies two water tanks, with a combined capacity of 450,000 gallons available for use in fire fighting. The tanks are checked weekly, and topped-off with water as required. (Based upon historical data, a minimum water volume of 85-percent of tank capacity has thus been maintained.)

(b) Fire pump installations are designed to deliver water to hydrants, standpipes, and sprinkler systems. Fire pump #2 is rated at 1,500 gallons-per-minute flow at 125 pounds-per-square-inch pressure.

(c) Alternative power for fire pumps is provided. Diesel pumps are test-started on a weekly frequency and two sets of batteries are provided for back-up starting. Emergency response personnel are trained to start the pumps manually.

(d) The water distribution system is designed such that failure of a single component will not disable the supply of fire suppression water to the facility.

8.1.7 Fire Emergency Response Team

8.1.7.1 The Fire Emergency Response Team is organized, and fire fighting equipment is maintained, as part of the Emergency Management Program described in the Site Emergency Plan and Procedures, as presented in Chapter 9.0 of this License Application.

8.1.7.2 Training to enable high quality performance of duties in response to facility fires is provided to the Team as part of the Emergency Management Program described in the Site Emergency Plan and Procedures, as presented Chapter in 9.0 of this License Application.

8.1.8 Pre-Fire Plans

8.1.8.1 The CFFF maintains ready for use, and on file for inspection by Regulatory Agencies, comprehensive Pre-Fire Plans that provide the strategic and tactical information needed by fire-fighting personnel when responding to an emergency.

8.1.8.2 Pre-Fire Plans include the following information:

(a) Division of the facility into logical planning areas.
(b) Site sketches that identify:

- Locations of areas;
- Response Team assembly points;
- Assembly point coverage areas; and,
- Locations of fire hydrants.

(c) Assignment of basic Response Team responsibilities, and Team checklists.
(d) Listings of fire detection and protection devices.
(e) Details of:

- Area description;
- Expected occupancy;
- Potential locations for trapped occupants;
- Potential disabled personnel that might require emergency assistance;
- Information about Area utilities;
- Construction information;
- Schedule for Plan updates;
- Basic information on hazardous materials in the area;
- Fire-fighting strategy considerations; and
- Supplementary information specified by the Fire Safety Function.

8.1.8.3 Pre-Fire Plans (and revisions to the Plans) are prepared and maintained by the Fire Safety Function. Copies of the Plans are made available to the off-site fire department most likely to respond to a call for assistance.

8.1.9 Fire Hazard Analyses

8.1.9.1 Performance and Documentation of Analyses

(a) Analysis Performance

(1) The Fire Safety Analysis is a comprehensive assessment of each component within a defined system. The analysis identifies controls required to maintain a sufficient margin of safety.

(2) Fire accident sequences are analyzed using the accident flow diagram format. In this format, the analyst traces each

sequence through the diagram (starting with the initiating event) to arrive at a consequence of interest. Each identified pathway defines an initiating event and protective measure failures that collectively represent an accident sequence.

(b) Analysis documentation

(1) The Fire Safety Analysis is one of the ISA safety analyses described in Chapter 4.0 of this License Application. The level of detail for a particular analysis is based on the complexity of the initial system, and subsequent proposed changes to the system. Thus, the scope and content of a Fire Safety Analysis are customized to reflect the particular characteristics and needs of the system being analyzed.

(2) Fire Safety Analyses are maintained current through implementation of the Configuration Management program described in Sections 3.1 and 4.1 of this License Application. If a Fire Safety Analysis is required for a proposed change, it is performed to the current standards required for the baseline analysis. Summary details of the change, including required approvals, are documented on a Configuration Change Control Form that is filed with the applicable Baseline ISA, thus providing a substantially complete "living" framework for the facility fire safety basis.

8.1.10 Audits and Compliance Inspections

Program and process assessments are conducted to compare established fire safety standards to CFFF performance. The assessments take the form of program audits and compliance inspections, as described in Chapter 3.0, Section 3.6, of this License Application.

8.1.10.1 Program assessments take the form of program audits. Specific portions of the Fire Safety Program, evaluated during a particular assessment, are based on previous internal audit findings, external audit findings, NRC inspection activities, current operating conditions, and time since last assessment. The Fire Safety Program is assessed on a triennial frequency. Results of the assessments are documented and maintained for NRC Staff review and inspection.

8.1.10.2 Process assessments take the form of compliance inspections that evaluate implementation of fire safety requirements (*e.g.*, control of combustible materials, following procedures and postings, *etc.*) for CFFF operations (*i.e.*, Site and Structures, ADU Conversion, Solvent Extraction, *etc.*). Frequencies of inspections are based on previous internal inspection findings, NRC inspection results, incidents (those reported, and those requiring notification), configuration management activities, and time since last assessment. The complete set of operations making up the CFFF ISA is assessed on a five year frequency. Results of the assessments are documented and maintained for NRC Staff review and inspection.

CHAPTER 9.0

EMERGENCY MANAGEMENT PROGRAM

The Columbia Fuel Fabrication Facility (CFFF) maintains a comprehensive Emergency Management Program with facilities, equipment and processes for protecting workers, the public and the environment. This program ensures control of licensed material, capability to evacuate personnel, and availability of emergency measures and facilities. The program is documented in an approved Site Emergency Plan and Procedures. At minimum, the Plan and Procedures are reviewed annually to ensure that the overall emergency preparedness program is being properly maintained.

9.1 EMERGENCY MANAGEMENT PROGRAM STRUCTURE

9.1.1 Site Emergency Plan

CFFF emergency preparedness practices are described in the latest revision of the Site Emergency Plan, submitted to NRC Staff, approved in accordance with applicable regulations, and maintained as prescribed by regulatory requirements. The Plan addresses the following emergency preparedness criteria:

(a) Facility Description;
(b) Engineered Safeguards for Abnormal Operations;
(c) Types of Accidents and Classifications;
(d) Response Management System;
(e) Mitigation of Consequences and Assessment of Releases;
(f) Emergency Response Facilities and Equipment;
(g) Maintaining Emergency Preparedness Capability;
(h) Records and Reports;
(i) Safe Shutdown, Recovery, and Plant Restoration; and,
(j) Hazardous Chemicals.

9.1.2 Emergency Procedures

Implementing procedures, approved in accordance with CFFF policy, contain detailed instructions on emergency response, and emergency personnel activities based on practices required by the Site Emergency Plan. These procedures clearly define duties, responsibilities, action levels, and actions to be taken by each functional individual or group in response to emergency situations. Copies of Emergency Procedures, and subsequent changes to them, are issued to personnel responsible for emergency response activities. The procedures address the following emergency preparedness criteria:

(a) Emergency Response Organization;
(b) Emergency Response Team;
(c) Equipment and Supplies;
(d) Evacuation, Accountability, and General Response;
(e) Classification;
(f) Communication;
(g) Notification;
(h) Biological Threat;
(i) Bomb Threat (Package or Object);
(j) Bomb Threat (Telephone or Correspondence);
(k) Civil Disturbance;
(l) Criticality;
(m) Explosion;
(n) Fire;
(o) Hazardous Material Release;
(p) Hazardous Weather;
(q) Loss of Utilities;
(r) Oil Spill;
(s) Radioactive Powder or Liquid Release;
(t) Transportation Accident; and,
(u) UF6 Release.
(v) Local Law Enforcement Agency Incident Response Plan; and,
(w) Notification Guidelines for NRC and Other Agencies.

CHAPTER 10.0

ENVIRONMENTAL PROTECTION

10.1 ENVIRONMENTAL PROTECTION PROGRAM STRUCTURE

The Columbia Fuel Fabrication Facility (CFFF) maintains an Environmental Protection Program for the site. A primary purpose of the Environmental Protection Program is to assure that exposure of the public and the environment, to hazardous materials used in facility operations, is kept As Low As Reasonably Achievable (ALARA).

CFFF prepared an Environmental Evaluation Report dated March 1975, that was subsequently updated in revisions dated April 1983, April 1990 and December 2004. Also an extensive update of much of the information in the March 1975 report was documented in an Integrated Safety Analysis (ISA) and ISA Summary titled "CFFF Site and Structures." Annual reviews of Environmental Protection Program data is documented in the ALARA Reports described in Section 5.1.14 of this License Application.

10.1.1 Effluent Air Control

For operations that might result in exhausting radioactive materials to unrestricted areas, the adequacy of air effluent controls is determined by representative stack-sampling, to demonstrate compliance with applicable regulations. Such sampling is performed continuously during production operations involving licensed materials. Samples are collected and analyzed daily.

If radioactivity in gaseous effluents exceeds 1,500 microcuries per calendar quarter, a report is prepared and submitted to NRC Staff within 30-days of the end of the quarter in which the excess occurred. This report identifies the cause of exceeding the limit and the corrective actions taken to reduce release rates. The report is submitted to NRC Headquarters with a copy to NRC Region II. Subsequently, if any parameters important to a dose assessment in the original report are found to have changed, a follow-up report is submitted within 30-days of disclosure which describes the changes in parameters and includes an estimate of the resultant change in dose commitment.

In the event that a calculated Total Effective Dose Equivalent (TEDE) to any member of the public in a calendar year could exceed a limit of 100 millirem, immediate steps are taken to reduce emissions to levels that will bring the TEDE back below the limit.

10.1.2 Liquid Waste Treatment

Liquid waste treatment facilities, with sufficient capacity and capability to enable holdup, treatment, sampling, analysis, and discharge of liquid wastes in accordance with applicable regulations, are provided and maintained in proper operating condition.

Control of radioactivity in the process liquid waste stream is achieved by operation of two serial treatment systems:

(a) A continuous in-line gamma spectroscopy monitor and quarantine tank filtration system within the chemical controlled area of the main Plant building; and,

(b) An Advanced Wastewater Treatment Facility (for removing uranium to ALARA levels) that is external to the building.

The first system is installed following quarantine tanks, diversion tanks, and filtration operations. This system assures that the process liquid waste stream, being transferred from the internal chemical controlled area to the external treatment area, meets the discharge limit in approved operating procedures. This limit is nominally less than 30 parts per million uranium (equivalent to 7.2 E-05 microcuries per milliliter at a specific activity of 2.4 microcuries per gram of uranium). When the liquid has successfully passed the scan for discharge from the first system, it is transferred from the in-plant final pump-out tank to the second system for further uranium removal.

The second system assures that uranium in the discharge is removed to a nominal limit of less than 0.5 parts per million uranium (equivalent to 1.2 E-06 microcuries per milliliter at a specific activity of 2.4 microcuries per gram of uranium). Approved operating procedures implement ALARA and assure that applicable 10CFR20 discharge limits are met.

Miscellaneous liquid wastes are filtered and sampled on a batch basis to assure uranium is effectively removed to levels that will enable conformance to ALARA goals.

Quiescent settling in the North, South, East, and West Lagoons further reduce uranium levels in liquid wastes prior to final discharge to the Congaree River. A continuous, proportional sample of the liquid effluent released to Congaree River is collected. A 30-day composite of this sample is analyzed for recording the gross alpha and beta activity and isotopic uranium content of the final discharge.

Any violation of the CFFF's NPDES Permit is reported to NRC Region II Staff within 15-days of confirmation of the violation. If the Permit is revoked, or if Permit conditions are revised, NRC Headquarters Staff is promptly notified.

10.1.3 Solid Waste Disposal

Solid waste disposal preparation facilities, with sufficient capacity and capability to enable processing, packaging, and transfer of solid wastes to licensed treatment or disposal sites, in accordance with applicable regulations, are provided and maintained in proper operating condition.

10.1.4 Environmental Monitoring

The CFFF environmental monitoring program includes the sampling criteria presented in Figure 10.1. (Note: For wells found not to contain water at the time of sampling, an evaluation is performed by the Environmental Protection Function to determine if alternate well data can be used to represent the dry well; or, if a new well must be dug.) Typical program analytical sensitivities are as presented in Figure 10.2. Locations of air, vegetation and soil monitoring stations, locations of surface water monitoring stations, and locations of monitoring wells are as presented in Figures 10.3, 10.4, and 10.5, respectively. Action levels for sample results are established by approved procedures.

These sampling criteria, sensitivities, and/or locations can be changed without prior NRC Staff approval provided:

(a) A documented evaluation by the Environmental Protection Function demonstrates that the changes will not decrease the overall effectiveness of the environmental monitoring program; and,

(b) The changes are submitted to NRC Staff as part of the subsequent annual update of Section 10.1 of this Chapter to enable opportunity to inspect the evaluation.

10.1.5 Periodic Reporting of Surveillance Data

Quantities of radioactive material in air and liquids released from the facility are reported to NRC Staff, in accordance with applicable regulatory guidance and regulations, on a semiannual basis.

10.1.6 Off-Site Dose Control

Compliance with 10CFR20 (NRC) and 40CFR190 (EPA) requirements, for off-site dose to the maximally exposed individual, is assured by demonstrating that no such potential annual dose exceeds 25 millirem. Dose calculation methodology includes models that have been evaluated and approved by the Environmental Protection Function and that have been recognized by the appropriate regulatory agencies.

Figure 10.1 Environmental Sampling Criteria

TYPE OF SAMPLE	LOCATIONS	ANALYSES	SAMPLING FREQUENCY
Air Particulates	Four	Alpha	Continuous (Collection Weekly)
Surface Water	Three	Alpha; Beta	Quarterly
Well Water[1]	Ten	Alpha; Beta; Ammonia	Semi- Annually
River Water	Three	Alpha	Quarterly
Sediment	One	Alpha; Beta; Uranium	Annually
Soil	Four	Alpha; Beta; Uranium	Annually
Vegetation[2]	Four	Alpha; Beta; Fluoride	Annually
Fish	One	Alpha; Beta; Uranium	Annually

[1]If gross alpha concentration exceeds 15 pCi/l, isotopic analyses for uranium will be conducted. If gross beta exceeds 50 pCi/l, beta/gamma scans are conducted. If a monitoring well exceeds a mean concentration of 30 pCi/l of total uranium, the result will be provided to cognizant NRC staff.

[2]If a vegetation gross alpha activity result exceeds 15 pCi/gram, an additional sample will be collected.

Figure 10.2 Typical Environmental Program Radiological Analytical Sensitivities

TYPE OF SAMPLE	ANALYSES	TYPICAL QUANTITY	NOMINAL MINIMUM DETECTION LEVEL
Air Particulates	Alpha	571 Cubic Meters	2.0E-15 Microcuries Per Milliliter
Surface Water	Alpha	1 Liter	2.2E-9 Microcuries Per Milliliter
	Beta	1 Liter	2.5E-8 Microcuries Per Milliliter
Well Water	Alpha	1 Liter	2.2E-9 Microcuries Per Milliliter
	Beta	1 Liter	2.5E-8 Microcuries Per Milliliter
River Water	Alpha	1 Liter	2.2E-9 Microcuries Per Milliliter
	Beta	1 Liter	2.5E-8 Microcuries Per Milliliter
Sediment	Alpha	100 Grams	1.0 Picocuries Per Gram
	Beta	100 Grams	3.0 Picocuries Per Gram
	Uranium	100 Grams	0.5 Picocuries Per Gram
Soil	Alpha	100 Grams	1.0 Picocuries Per Gram
	Beta	100 Grams	3.0 Picocuries Per Gram
	Uranium	100 Grams	0.5 Picocuries Per Gram
Vegetation	Alpha	100 Grams	1.0 Picocuries Per Gram
	Beta	100 Grams	3.0 Picocuries Per Gram
Fish	Alpha	30 Grams	1.0 Picocuries Per Gram
	Beta	30 Grams	3.0 Picocuries Per Gram
	Uranium	1 Kilogram	0.5 Picocuries Per Gram

Figure 10.3 Locations of Air, Vegetation, and Soil Monitoring Stations

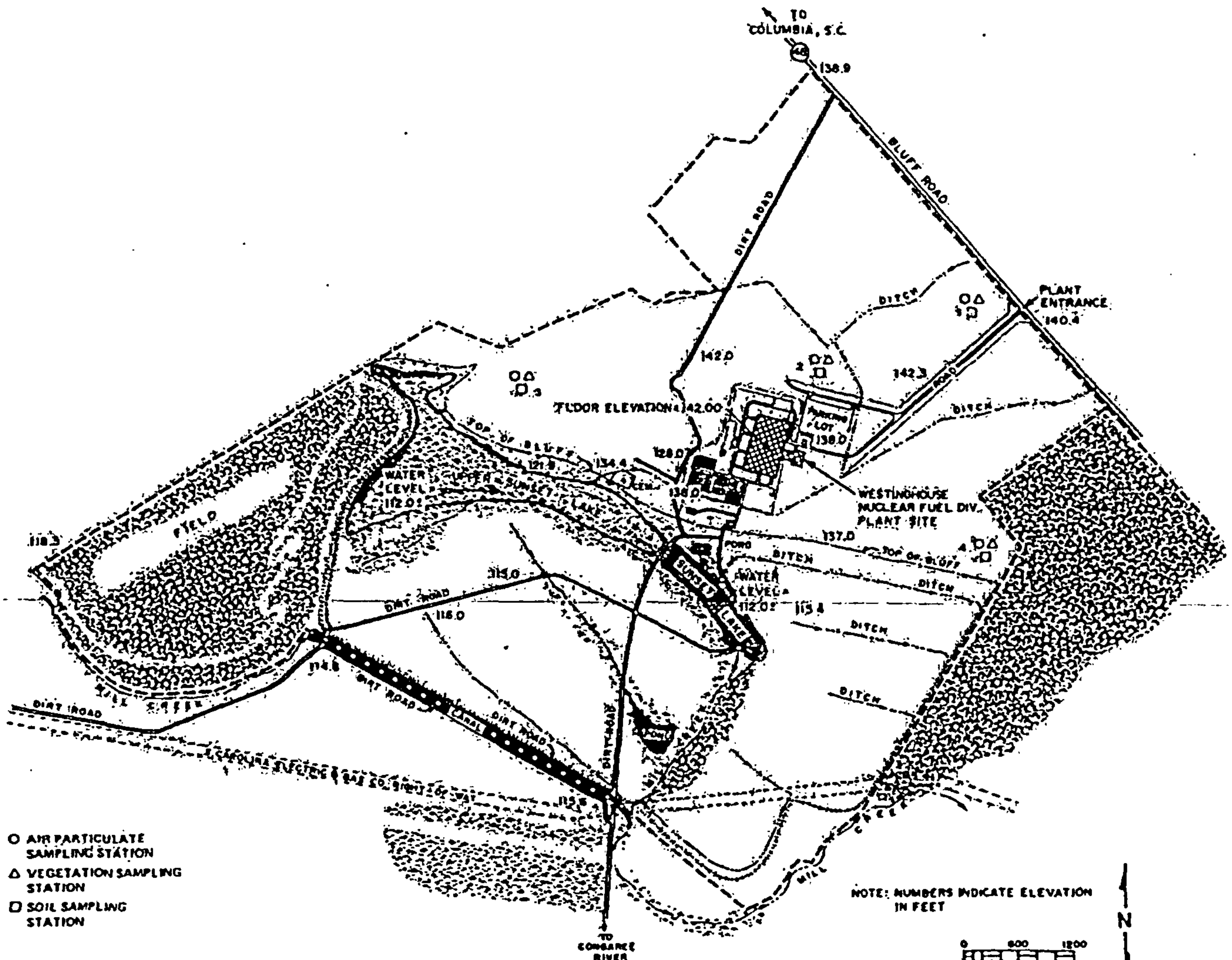


Figure 10.4 Locations of Surface Water Monitoring Stations

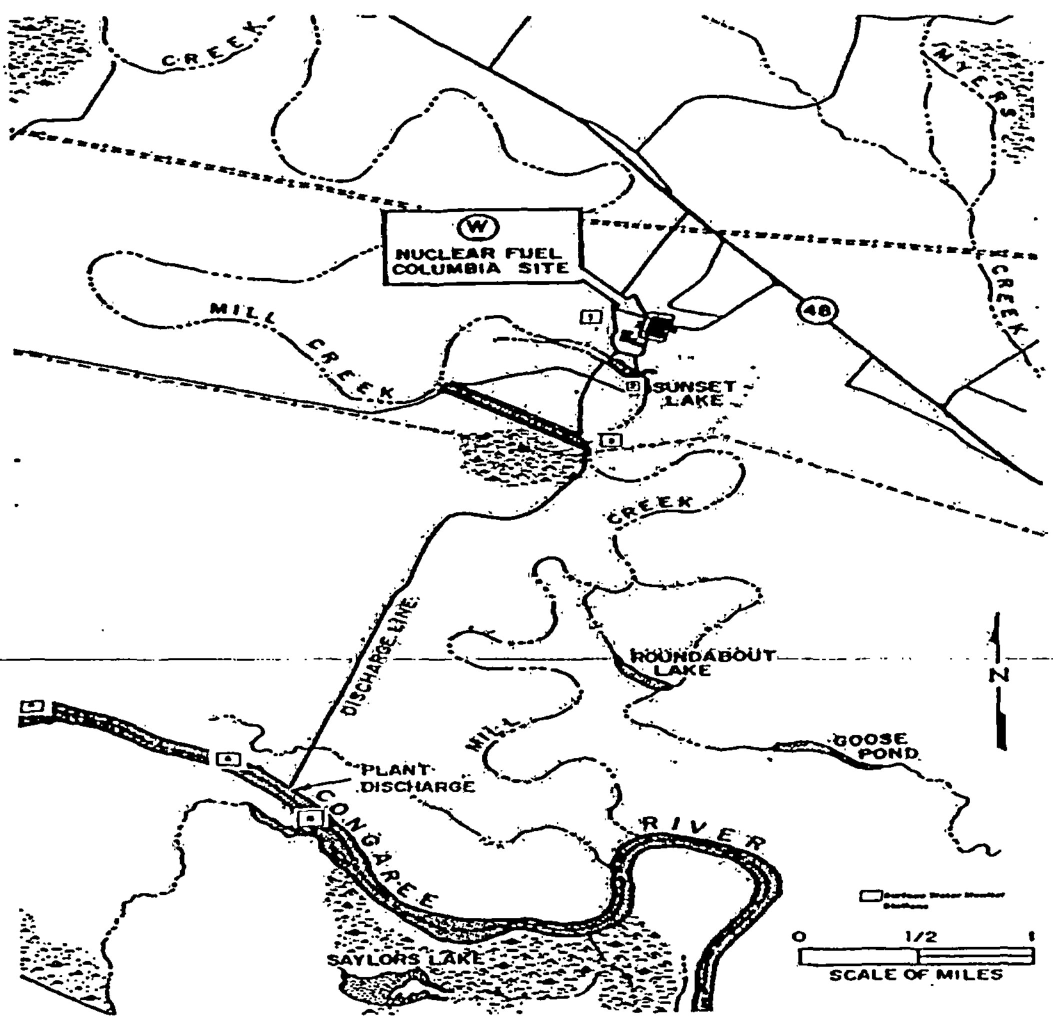
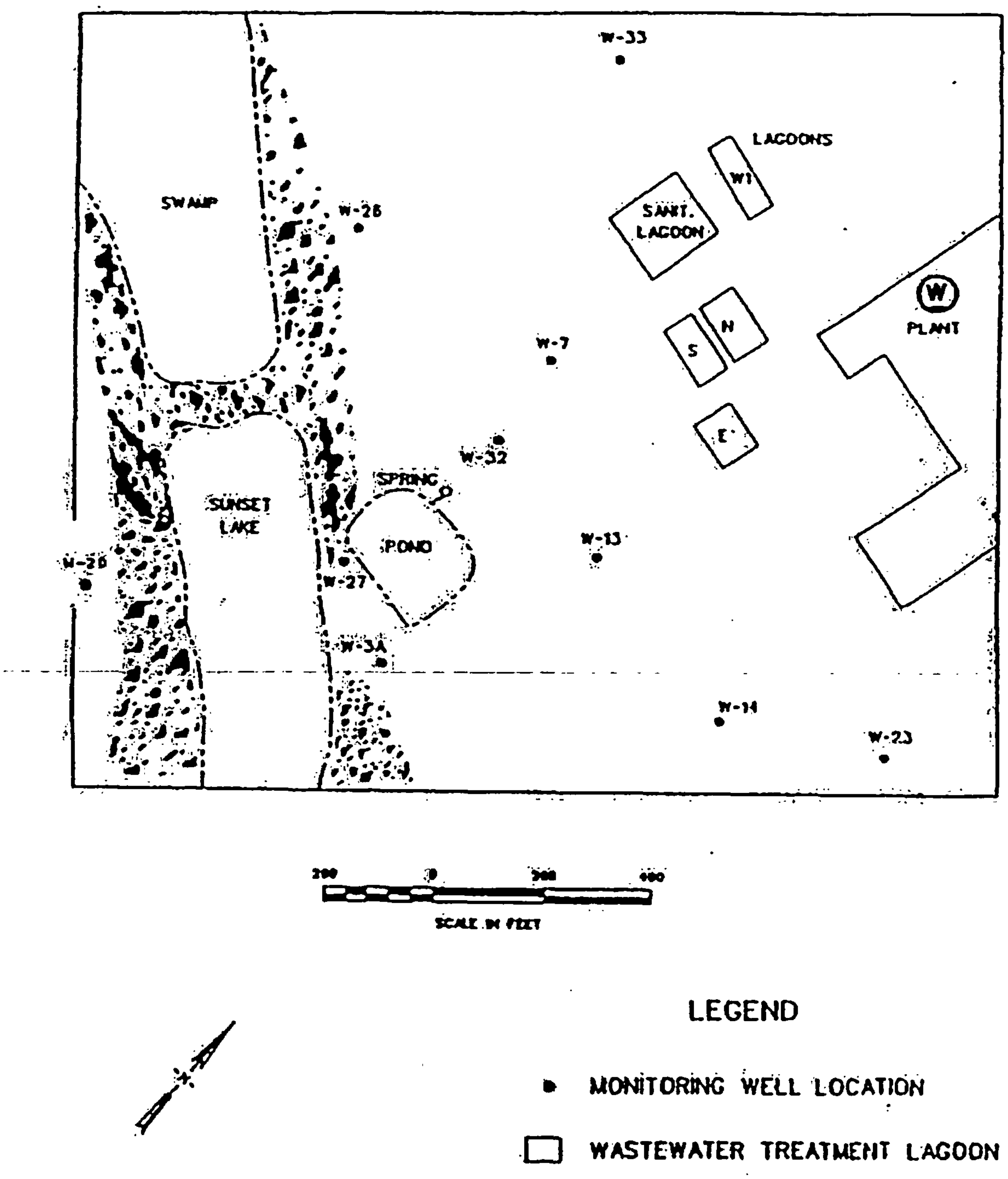


Figure 10.5 Locations of Monitoring Wells



10.1.7 Performance and Documentation of Analyses

10.1.7.1 Environmental Protection Analysis

Analysis performance

The Environmental Protection Analysis is a comprehensive assessment of each component within a defined system. The analysis identifies controls required to maintain a sufficient margin of safety.

Environmental accident sequences are analyzed using the accident flow diagram format. In this format, the analyst traces each sequence through the diagram (starting with the initiating event) to arrive at a consequence of interest. Each identified pathway defines an initiating event and protective measure failures that collectively represent an accident sequence.

Analysis documentation

The Environmental Protection Analysis is one of the ISA safety analyses described in Chapter 4.0 of this License Application. The level of detail for a particular analysis is based on the complexity of the initial system, and subsequent proposed changes to the system. Thus, the scope and content of an Environmental Protection Analysis are customized to reflect the particular characteristics and needs of the system being analyzed.

Environmental Protection Analyses are maintained current through implementation of the Configuration Management program described in Sections 3.1 and 4.1 of this License Application. If an Environmental Protection Analysis is required for a proposed change, it is performed to the current standards required for the baseline analysis. Summary details of the change, including required approvals, are documented on a Configuration Change Form that is filed with the applicable Baseline ISA, thus providing a substantially "living" framework for the facility Environmental Protection basis.

10.1.8 Audits and Compliance Inspections

10.1.8.1 Program and process assessments are conducted to compare established environmental protection standards to CFFF performance. The assessments take the form of program audits and compliance inspections, as described in Chapter 3.0, Section 3.6, of this License Application.

Program assessments take the form of program audits. Specific portions of the Environmental Protection Program, evaluated during a particular assessment, are based on previous internal audit findings, external audit findings, NRC inspection activities, current operating conditions, and time since last assessment. The Environmental Protection Program is assessed on a triennial frequency. Results of the assessments are documented and maintained for NRC Staff review and inspection.

Process assessments take the form of compliance inspections that evaluate implementation of environmental protection requirements (*e.g.*, effluent controls, following procedures and postings, *etc.*) for CFFF operations (*i.e.*, Site and Structures, ADU Conversion, Solvent Extraction, *etc.*). Frequency of inspection is based on previous internal inspection findings, NRC inspection results, incidents (those reported, and those requiring notification), configuration management activities, and time since last assessment. The complete set of operations making up the CFFF ISA is assessed on a five year frequency. Results of the assessments are documented and maintained for NRC Staff review and inspection.

10.1.8.2 The Regulatory Component performs a biennial audit of vendors used to analyze environmental samples. Such audits are also performed if substantive program anomalies are disclosed. The audits consider the need for "spike" and/or "replicate sample" submittals, as part of evaluation of a vendor's capability and quality control effectiveness.

CHAPTER 11.0

DECOMMISSIONING PLANNING

11.1 DECOMMISSIONING PLANNING STRUCTURE

To assure adequate financial resources are available to decommission the Columbia Fuel Fabrication Facility (CFFF) at the end of its useful life, a conceptual decommissioning plan (*Cost Estimate to Terminate License SNM-1107*), and a decommissioning funding plan and financial assurance mechanism, have been prepared and are maintained current.

11.1.1 Conceptual Decommissioning Plan

In support of the *Cost Estimate to Terminate License SNM-1107*, a dedicated document file is maintained. This file includes the following record categories:

(a) Correspondence Chronological File;
(b) Historic Conceptual Plan(s) and Cost Estimate(s);
(c) Historic Facility Radiological Information;
(d) NRC Guidance Documents;
(e) EPA Guidance Documents;
(f) Decommissioning Plan Shell;
(g) Current Conceptual Plan and Cost Estimate; and,
(h) Financial Assurance.

The file includes a records log-out/return process that provides for information on:

(a) Date;
(b) Out to; and,
(c) File number or name out.

Each record category is clearly marked "Warning, these decommissioning records must not be removed or destroyed without the written approval of the Regulatory Component."

Copies of the most recent *Cost Estimate to Terminate License SNM-1107* are maintained by the Engineering Component and the Regulatory Component. The Engineering Component maintains an electronic copy that contains the Westinghouse position on decommissioning in the following file structure:

(a) Executive Summary;
(b) Project Summary;
(c) Project Description;
(d) Estimate Configuration;
(e) Assumptions;
(f) Westinghouse Staff;
(g) Demolition Labor Rate;

(h) Subcontract - Consumables
(i) Wash-Down Estimate;
(j) Labor Factors;
(k) Material Density and Pack Factors;
(l) Inflation Factors;
(m) Major Cost Drivers;
(n) Overhead Piping Density;
(o) Structure Data Sheets;
(p) Equipment Data Sheets; and,
(q) Major Drivers.

The *Cost Estimate to Terminate License SNM-1107* is reviewed for need to update on a triennial basis.

11.1.2 Decommissioning Funding Plan and Financial Assurance Mechanism

(a) Decommissioning Funding Plan

The decommissioning funding plan is a cost estimate for decommissioning the CFFF at the end of its useful life. To substantiate the cost of decommissioning, the Westinghouse position is maintained on the following cost estimating tables:

- Planning and Preparation;
- Decontamination and/or Dismantling of Radioactive Facility Components;
- Packaging, Shipping, and Disposal of Radioactive Wastes;
- Restoration of Contaminated Areas on Facility Grounds;
- Final Radiation Survey; and,
- Site Stabilization and Long-term Surveillance.

The decommissioning cost estimate is submitted to NRC Staff for acceptance and acknowledgement in accordance with prevailing requirements or directives.

(b) Financial Assurance Mechanism

Westinghouse has established a financial assurance mechanism, to support the projected cost of CFFF decommissioning, in accordance with the provisions of 10CFR70.25. The financial assurance mechanism is submitted to NRC Staff for acceptance and acknowledgement in accordance with prevailing requirements or directives. By a letter dated May 15, 2000, Westinghouse submitted a revised decommissioning cost estimate for License SNM-1107. By a letter dated December 15, 2000, NRC Staff acknowledged the submittal, and instructed Westinghouse to proceed in correcting the financial assurance instrument to more closely reflect the updated cost estimate. By a letter dated January 15, 2001, Westinghouse submitted the revised financial assurance mechanism. The most recent triennial update of the cost estimate to terminate License SNM-1107 was completed on December 17, 2003 and is on file at the CFFF.

CHAPTER 12.0

AUTHORIZATIONS AND EXEMPTIONS

12.1 AUTHORIZATIONS

12.1.1 Authorization to Make Changes to License Commitments

(a) CHANGES REQUIRING PRIOR APPROVAL

Westinghouse shall not make changes to the License Application that decrease the effectiveness of commitments, without prior NRC approval. For these changes, Westinghouse will submit to the NRC, for review and approval, an application to amend the License. Such changes will not be implemented until approval is granted.

(b) CHANGES NOT REQUIRING PRIOR APPROVAL

Upon documented completion of an Integrated Safety Assessment for a facility or process, as described in Chapter 4.0 of this License Application, Westinghouse may make changes in the facility or process as presented in the License Application, or conduct tests or activities not presented in the Application, without prior NRC approval, subject to the following conditions:

1. There is no degradation in the safety commitments in the License Application.

2. The change, test, or activity does not impair the Westinghouse ability to meet all applicable Federal regulations.

3. The change, test, or activity does not conflict with any condition specifically stated in the License.

Records of such changes shall be maintained, including technical justification and management approval, in dedicated datapacks to enable NRC inspection upon request at the facility. A report containing a description of each such change, and appropriate revised pages to the License Application, shall be submitted to the NRC within three months of implementing the change.

12.1.2 Authorization for Leak-Testing Sealed Plutonium Sources

The following procedure shall be authorized for leak-testing sealed plutonium sources at the licensed activity:

- Each sealed plutonium source in use shall be leak-tested at least semi-annually. In absence of a certificate from the supplier indicating that such a test has been performed within six month prior to transfer to the licensed activity, the subject sealed plutonium source shall not be put into use until leak-tested.

- Sealed plutonium sources that are stored, and are not being used, shall be exempt from the leak-test requirement. Such stored sources shall be leak-tested prior to any use in, or transfer from, the licensed activity unless such a test has been performed within the six months preceding the date of use or transfer.

- The leak-test shall be capable of detecting the presence of 0.005-microcuries, or more, of alpha contamination on a smear-test sample. The smear-test sample shall be taken directly from the sealed source, or from appropriate accessible surfaces of the device in which the source is mounted or stored.

- Records of leak-test results shall be kept in units of microcuries, or other units directly convertible to microcuries by multiplication using a recognized constant; and, the records shall be maintained for review by the NRC Staff.

- If a leak-test reveals the presence of 0.005-microcuries limit, the licensed activity shall file a report with the NRC Staff Headquarters which describes the subject source, the leak-test results, the extent of any related contamination, the apparent cause of failure, and corrective actions taken. A copy of this report shall also be sent to the NRC Region II Staff.

12.1.3 Authorization for Possession at Reactor Sites

The licensed activity may possess unirradiated fuel assemblies, at nuclear reactor facilities anywhere within the United States, for the purpose of loading them into shipping packages, and delivery to an authorized carrier for transport in accordance with the regulations. Operation incident to such loading shall be subject to the control of a licensed activity representative, approved by the Manager of the Regulatory Affairs Component, who shall assure that the completed transport package complies with all requirements of the regulations.

For such operations, the licensed activity shall be exempted from conditions of Title 10, Code of Federal Regulations, Part 70.24; CRITICALITY ACCIDENT REQUIREMENTS, provided:

- As finished fuel assemblies are resolved from their approval storage facilities, they shall be constrained in an arrangement that is no more reactive that that which they will assume in the shipping package.

- The total number of fuel assemblies in process at anyone time shall not exceed the maximum authorized contents of the packaging being loaded.

- If two fuel assemblies are in movement at the same time, a 12-inch minimum edge-to-edge separation shall be maintained between them; and, only one fuel assembly at a time shall be loaded into the shipping package.

- Loaded packages shall be stored in the approved shipping array, pending delivery to a carrier.

- No more than the maximum number of packages authorized for a single shipment shall be loaded and possessed, in conduct of such operations by the licensed activity, at any one location.

12.1.4 Authorization for Use at Off-Site Locations
(WITHDRAWN)

12.1.5 Authorization for Transfer of Hydrofluoric Acid

Pursuant to Title 10 Code of Federal Regulations, Part 20.2002; *Method for Obtaining Approval of Proposed Disposal Procedures*, aqueous hydrofluoric acid containing trace quantities of uranium may be transferred to non-licensed receivers provided the following conditions are met:

- Prior to first unrestricted sale or other transfer of the subject material to each receiver, a detailed plan for such sale or transfer shall be submitted to the NRC Staff for review and approval.

- Prior to transfer of the hydrofluoric acid from Westinghouse, each shipment must be representatively sampled and analyzed; and the following maximum permissible concentrations shall not be exceeded: A uranium enrichment of 5 w/o U-235; A uranium concentration of 3-parts-per-millino by weight; and, and HF concentration, in the acid solution, of 55-percent by weight.

- Particular attention shall be paid to each sale or transfer to assure that the hydrofluoric acid is not to be used for any purpose resulting in human consumption.

12.1.6 Authorization for Transfers as Non-Regulated Material

Pursuant to Title 10, Code of Federal Regulations, Part 20.2002; *Method for Obtaining Approval of Proposed Disposal Procedures*, industrial waste treatment products from the licensed activity, such as calcium fluoride and other homogenous mixtures in which the mean concentration of uranium constituents does not exceed 30-picocuries per gram, may be released without continuing NRC licensing controls, to receivers for off-site calcium fluoride drying and briquette manufacturing, or for cement or brick manufacturing, or to disposition at a chemical disposal site or industrial landfill. Calcium fluoride so released to off-site manufacturers shall contain a minimum of 60-percent solids. Prudent efforts shall be made to reduce the radioactive contents of all such transferred materials to level as low as reasonably achievable.

A sampling plan shall be implemented to characterize the industrial products in accordance with NUREG/CR-2082; MONITORING FOR COMPLIANCE WITH DECOMMISSIONING TERMINATION SURVEY CRITERIA, as follows:

- The estimation of the population mean for uranium concentration shall be representative of the industrial products being transferred;

- The sample size used to calculate the mean uranium concentration value shall be determined such that the 95-percent confidence limit for the value is less than 25-percent of the value;

- The sampling plan is to provide a minimum confidence level of 95-percent that the true mean uranium concentration value, determined for the industrial to be transferred, is less than the maximum permissible limit of 30-picocuries per gram of dry material.

- Records pertaining to the release of such materials, including identities of receivers, shall be maintained for review by the NRC Staff.

12.1.7 Authorization to Release Contaminated Records

The licensed activity may abandon or dispose of small quantities of radioactive materials that are present as minor contamination on certain papers, notebooks, computer print-outs, films, and/or similar items retained for record purposes. No licensed controls shall be required for final disposition of such records, and they may randomly be mingled with, and/or disposed of as, other records, provided:

- Prior to transfer from contamination control areas at the licensed activity, a documented survey instrument measurement shall conclude that the following limits are not exceeded: Average uranium-alpha contamination of 220-disintegrations-per-minute per 100-square-centimeters; Maximum uranium-alpha contamination of 2200-disintegrations-per-minute per 100-square-centimeters. Average beta-gamma emitter contamination of 660-disintegrations-per-minute per 100-per-square-centimeters; Maximum beta-gamma emitter contamination of 6600-+disintegrations-per-minute per 100-square-centimeters.

- Such records shall be kept in locations that are used primarily for record storage and/or disposal.

12.1.8 Authorization to Release for Unrestricted Use

Licensed activity material and equipment may be released from contamination areas on-site to clean areas on-site, or from on-site possession or use to unrestricted possession or use off-site; provided, such releases are subject to all applicable conditions of the NRC Staff's April 1993 document entitled; *Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material.*

12.1.9 Authorization to Use ICRP 68

DAC and ALI values based on the dose coefficients published in ICRP Publication No. 68 may be used in lieu of the DAC and ALI values in Appendix B of 10 CFR Part 20 in accordance with internal procedures.

12.2 EXEMPTIONS

12.2.1 Exemption from Prior Commitments

All commitments made to NRC Staff prior to the approval date of this License Application shall be no longer binding upon Westinghouse, following approval of this License Application, unless re-imposed as License Conditions.

12.2.2 Exemption from Individual Container Posting

Notwithstanding the requirement of paragraph (a) of Title 10, Code of Federal Regulations, Part 20.1904; *Labeling Containers,* the license activity shall be exempted from the requirement that "each container of licensed material bears a durable clearly visible label"; provided, in lieu thereof, a sign bearing the legend "EVERY CONTAINER OR VESSEL IN THIS AREA MAY CONTAIN RADIOACTIVE MATERIAL" is posted at each entrance to areas for buildings in which radioactive materials are used or stored, from areas in which such materials are not used or stored. Regarding storage of radioactive material outside the Fuel Manufacturing Building, the number of posted buildings and size of posted areas shall be minimized to the extent practicable, consistent with manufacturing and storage requirements.

12.2.3 Exemption from Respirator Use Reporting

Notwithstanding the requirement of paragraph (d) of Title 10, Code of Federal Regulations, Part 20.1703; *Use of Individual Respiratory Protection Equipment*, since use of respiratory protection has been ongoing at the Columbia Fuel Fabrication Facility, continuing use shall be exempted from the requirement to "notify, in writing, the Regional Administrator of the appropriate Nuclear Regulatory Commission Regional Office listed in Appendix D at least 30-days before the date that respiratory protective equipment is first used" under provisions of the April 30, 1995 License Renewal Application approval.

12.2.4 Exemption from Shallow-Dose Equivalent Tissue Depth

Notwithstanding the requirement of Title 10, Code of Federal Regulations, Part 20.1003, *Definitions, "Shallow-Dose Equivalent"*, the licensed activity shall be exempted from the requirement that the Shallow-Dose Equivalent is taken as the dose equivalent at a tissue depth of 0.007-centimeter (7 mg/cm^2), when this dose equivalent is measured for the finger. In lieu thereof, for finger doses, the Shallow-Dose Equivalent shall be taken as the dose equivalent at a tissue depth of 0.038-centimeter (38 mg/cm^2). This applies to both the assessment of finger doses and for determining compliance with the finger dose limit.

12.2.5 Exemption from Criticality Monitoring System Requirements

Notwithstanding the requirement of Title 10, Code of Federal Regulations, Part 70.24, the licensed activity shall be exempted from the "monitoring system" requirements in the areas, and under the conditions specified below:

Office and conference room areas, chemistry laboratories, metallurgical laboratories, development laboratories, health physics counting rooms, and machine shop – provided that:

- Each such area shall be remote from other operations with special nuclear material.
- Each such area shall be administratively limited to 1000-grams of U235; and, for chemistry laboratories, an additional 5 grams of U233.

Low concentration storage areas in which containers have uranium in quantities representing no more than 350-grams of U235 per package and no more than 5 grams of U235 in any 10 liters of package; or, no more than 50-grams of U235 per container and no more than an average of 5 grams of U235 per 10 liters of package – provided that:

- Each such area qualifies for appropriate nuclear isolation with respect to other areas where special nuclear material is more concentrated.

The limits established above represent values that are below the maximum subcritical limits as established in numerous technical references, including LA-12809, ARH-600, LA-10860, ANSI/ANS-8.1-1998, and the limits presented in the *Handbook for the Conduct of Nuclear Criticality Safety Activities at the Columbia Fuel Fabrication Facility.* These limits apply to all aspects of the operation, including expected upset conditions.

Storage areas in which the only special nuclear material present is contained in authorized packages as defined in 49CFR173 – provided that:

- The maximum number of containers permitted in each such area shall be unlimited for low specific activity packages.

- The maximum number of packages bearing FISSILE labels stored in any one storage area must be limited so that the total sum of the criticality safety indices in any individual group of such packages does not exceed 50. Groups of such packages must be stored so as to maintain a spacing of at least 6m (20 feet) from all other groups of such packages.

12.2.6 Exemption from Packaged Radioactive Material Monitoring Requirements

Notwithstanding the requirement of 10 CFR 20.205(b) to monitor the external surfaces of packaged radioactive material receipts for radioactive contamination, the licensed activity is exempted from such requirement relative to flatbed trailer shipments of fuel assemblies received from the General Electric Company for interim storage purposes only, provided the constraints, conditions and controls committed to in a letter, dated November 30, 1993, (identification # NRC-93-036), are satisfied; and further provided that the total number of such fuel assemblies stored at the site at any given time does not exceed 250.

12.2.7 Exemption for Electronic Submissions

Notwithstanding the requirements of 10CFR 70.5, communications or reports concerning the regulations in Part 70 and any application filed under these regulations may be submitted electronically.

12.2.8 Exemption From the Transportation Requirements for Certain Fissile Material

The licensed activity is exempt from fissile material classification and from the fissile material package standards of 10CFR71.55 and 10CFR71.59 for the transport of certain bulk materials contaminated with U235. Concentration limits, stated as the ratio of U235 to non-fissile material, are established that provide control parameters adequate to ensure nuclear criticality safety for shipments. This exemption has already been approved for Westinghouse Licensee SNM-33 on April 15, 2002.