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REVISION RECORD

REVISION NUMBER	DATE OF REVISION	PAGES REVISED	REVISION REASON
1.0	30APR95	All	Update to current operations.
2.0	28JUN96	iii, 6.8	Clarify Criticality Safety Basis for the compaction operation.
3.0	30AUG96	iii, 1.7, 1.9, 12.6, 12.7	Incorporate Safety Condition S-3 into Application; correct reference to Figure 1.3 instead of 2.3, to reflect expansion of the CAA in order to eliminate need for gate.
4.0	30SEP96	iii, 6.11, 6.12	Clarification of Criticality Safety Basis for the Pellet Stripping System Equipment and Hoods & Containment.
5.0	08NOV96	iii, 1.12, 3.18, and 3.19 (Reprinted all document pages in Microsoft Word format)	Incorporation of a definition, and incident notification criteria, recently approved by NRC Staff.
6.0	05MAY97	6.12 (Reprinted all document pages in Microsoft Word format.)	Clarify Evaluation Bounding Assumptions for Storage of Annular Pellets

CHAPTER 6.0

NUCLEAR CRITICALITY SAFETY

6.1 PROGRAM ADMINISTRATION

6.1.1 FACILITY PROCEDURES

(a) Plant Operating Procedures

Procedures, and procedure changes, impacting nuclear criticality safety will be reviewed and approved by the Nuclear Criticality Safety Function. First level managers will be responsible for assuring that such procedures are made available to appropriate personnel; through posting of limits, training programs, and/or other written, electronic or verbal notifications. Documentation of the process of review, approval, and operator signoff will be maintained electronically within the facility procedure control system. Specific details of the system are described in Chapter 3.0, Subsection 3.4.1 of this License Application.

(b) Regulatory Affairs Guidance Procedures

Regulatory-Significant procedures define the policies of the Regulatory Component, including nuclear criticality safety, and identify the requirements for implementation of applicable NRC regulations and license conditions. These guidance procedures will be issued by the Regulatory Component and approved by Regulatory Component Managers. Criticality safety issues will be addressed and communicated to first level managers for incorporation into plant operating procedures. Regulatory Component approvals will be required for all regulatory-significant procedures, and their changes, involving the handling, processing, storage, inspection, and/or movement of special nuclear materials. Specific details of the system are described in Chapter 3.0, Subsection 3.4.1 of this License Application.

(c) Posting of Limits and Controls

Posting refers to the placement of signs or painting of floor areas to summarize key criticality safety requirements and limits, to designate approved work and storage areas, or to provide instructions or specific precautions to personnel. Posting will occur at the entrance to work or storage areas such as hoods, zones, or modules where special nuclear material is handled, processed, or stored.

Storage postings will be located in conspicuous places and will include as appropriate; material type, container identification, number of items allowed; and, mass, volume, moderation, and/or spacing limits. Additionally, when administrative controls or specific actions/decisions by operators are involved, postings will include pertinent requirements identified within the configuration control and/or criticality safety evaluation.

Criticality safety precautions or prohibitions related to fire fighting such as use of water, fog nozzles, and high pressure sprays will be posted at the entrance to the affected area.

Postings will be approved and issued by the Nuclear Criticality Safety Function. Compliance with the requirements identified on postings will be documented by formal inspections and audits.

6.1.2 INSPECTIONS & AUDITS

Inspections and audits will be performed to document that plant operations are conducted in accordance with applicable license conditions, company policies, and written procedures. This program incorporates process, procedure, and program reviews as tools to evaluate the effectiveness of the criticality safety program. All such inspections and audits will be conducted and documented in accordance with a written procedure. Personnel involved in these reviews will be knowledgeable in nuclear criticality safety and the findings, recommendations, and observations will be reviewed by management within the Regulatory Component. After the review by the Regulatory Component has been completed, the findings, recommendations, and observations will be transmitted to appropriate managers. Specific details of the facility inspection and audit program are described in Chapter 3.0, Subsection 3.6.1 of this License Application.

Process reviews include inspections and audits of the conduct of operations within the facility and will be conducted on an annual frequency. The specific areas of interest for the annual review will be selected based on previous findings, trend analyses, and current operating conditions.

Procedure reviews will be conducted on a frequency corresponding to the appropriate procedure category. The steps within the procedure will be compared with the covered operation to demonstrate accuracy and representativeness. Specific details of the facility procedure review program frequency are described in Chapter 3.0, Subsection 3.4.1, Paragraph (c) of this License Application.

A Nuclear Criticality Safety program review will be conducted on an annual basis. The specific portions of the program evaluated during a particular review will be determined

based on previous findings, NRC inspection activities, current operating conditions, and date of last review. All portions of the program will be reviewed at least triennially. This provides a mechanism for assessing the effectiveness of all components of the nuclear criticality safety program on a revolving basis and maximizes the utilization of nuclear criticality safety personnel.

Nonroutine inspections and audits may be conducted at the discretion of the Nuclear Criticality Safety Function or may be performed as the result of an operation upset, floor observations, or external investigations.

6.1.3 NUCLEAR CRITICALITY SAFETY PERSONNEL

The specific details of the Regulatory Component personnel position accountability and requirements are described in Chapter 2.0, Subsection 2.1.3, Paragraph (c) of this License Application. Key features specific to nuclear criticality safety are identified below:

(a) Requirements

The minimum qualifications for a Nuclear Criticality Safety Function Engineer or Manager will be a baccalaureate degree, with physical science or engineering emphasis; and, two years of experience in the nuclear industry. This will include at least one year of nuclear criticality safety experience. A Nuclear Criticality Safety Function Engineer will have demonstrated proficiency in nuclear criticality safety, and in the performance of the assigned position function. Specialized training and testing will be used as appropriate to document this experience and proficiency. Specific details are provided in Chapter 3.0, Subsection 3.4.2, Paragraph (b) of this License Application.

(b) Authority

Nuclear Criticality Safety Function Engineers and Managers will be empowered to review and approve facility procedures, perform validated nuclear criticality safety evaluations, set safety limits, and approve temporary operations. All Nuclear Criticality Safety personnel have authority to shutdown potentially unsafe operations. Nuclear Criticality Safety Function Engineers and Managers have authority to allow operations to restart once criticality safety issues have been resolved.

6.2 CONTROL METHODOLOGY AND PRINCIPLES

6.2.1 GENERAL CONTROL PROGRAM PRACTICES

The Double Contingency Principle (ANSI/ANS-8.1-1983 (R 1988)) will be the basis for design and operation of processes within the Columbia Fuel Fabrication Facility (CFFF) using special nuclear materials. Where practicable, all process designs will incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible. In those instances where at least two independent controls are utilized to prevent changes in one process condition, sufficient redundancy and diversity of controls will be utilized. For each significant portion of the process, a defense of one or more system parameters will be employed and documented within the Criticality Safety Evaluation. The defense is comprised of the set of 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.

Criticality Safety Evaluations (CSE) are utilized to identify the specific controls necessary for the safe and effective operation of a process. Nuclear criticality safety controls will be incorporated into the process design criteria documentation. Prior to use in any process, these controls will be undergo a functional verification process. A program for routine maintenance and testing will assure continued compliance.

(a) Verification Program

The purpose of the verification program will be to assure that the controls selected and installed match the requirements identified in the design criteria. All equipment will be examined in the "as-built" condition to validate the design and to verify the quality of the installation. In addition, a functional test will be performed to verify that the controls function as intended.

(b) Maintenance Program

The purpose of the maintenance program will be to assure that the controls designated for a specific process are maintained at the original level of implementation. This requires a combination of routine maintenance, functional testing, and verification of design specifications on a periodic basis. Specific details of this program are described in Chapter 3.0, Section 3.2 of this License Application.

Operations personnel will be responsible for the verification of controls through the use of functional tests. Assistance will be provided by Instrument Technician Functions as required. Control calibration and routine maintenance will normally be provided by the Instrument Technician Function. All verification and maintenance activities will be performed per detailed facility procedures and documented through the use of forms

and/or computer systems. Nuclear Criticality Safety Function personnel review all control verifications and maintenance activities and utilize a facility computer tracking system as the mechanism for tracking problems and documenting that corrective actions have been taken.

6.2.2 METHODS OF CRITICALITY SAFETY CONTROL

The relative effectiveness or reliability of controls will be considered during the Criticality Safety Evaluation process. Passive engineered controls will be preferred over all other system controls and will be utilized when available and appropriate. Active engineered controls will be the next preferred method of control and administrative controls will be the least preferred.

(a) Passive Engineering

These will be controls which require no action or other response to be effective when called upon to ensure nuclear criticality safety. Examples of passive engineered controls include safe geometry equipment such as structurally robust cylinders.

(b) Active Engineering

These will be controls which require an external signal and/or an electronic/mechanical action/operation to occur when called upon to ensure nuclear criticality safety. An example of an active engineered control is a shutoff valve actuated by an inline detector signal.

(c) Administrative

These will be controls which rely on user intervention and do not have the same level of reliability as engineered controls and will be least preferred. While such controls may be necessary, and hence acceptable, their use will be limited to process systems which do not lend themselves to engineered controls. Administrative controls include operator actions which are taken in accordance with a written procedure, operator verification of information with the assistance of computer terminals, actions taken in response to process alarms, etc.

6.2.3 TABLE OF PLANT SYSTEMS & PARAMETRIC CONTROLS

This table identifies the major area or system equipment, vessels, and containers within the CFFF. Table entries for each significant item highlight the safety basis selected for the Criticality Safety Evaluation (CSE) or Criticality Safety Analysis (CSA) and for the

bounding assumptions relevant to the analysis. An evaluation for unit interaction will be conducted and appropriate unit separation distances will be determined for any system containing moderated uranium or in cases where moderating materials are available. Table column definitions are presented below:

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

PROCESS OR EQUIPMENT: A defined subgroup of vessels, tanks, process and/or support equipment.

CRITICALITY SAFETY BASIS: The parameters utilized within the CSE or CSA for defense of nuclear criticality safety for identified PROCESS OR EQUIPMENT items. When multiple entries are present, this means that the defense within the CSE or CSA may be based on either parameter as appropriate.

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

AREA OR SYSTEM	PROCESS OR EQUIPMENT	CRITICALITY SAFETY BASIS	EVALUATION BOUNDING ASSUMPTIONS
Storage Pad	UF_6 Cylinders	Moderation	▪ UF_6 Only ▪ $\leq$ 1 wt. % H_2O Equivalent ▪ Full Reflection
ADU Conversion	UF_6 Cylinder	Moderation	▪ UF_6 Only ▪ $\leq$ 1 wt. % H_2O Equivalent ▪ Partial Reflection
	Vaporizer	Geometry (Level Control)	▪ Homogeneous UO_2F_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Cold Trap System (ADU and IDR)	Geometry	▪ Homogeneous UO_2F_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection

bounding assumptions relevant to the analysis. An evaluation for unit interaction will be conducted and appropriate unit separation distances will be determined for any system containing moderated uranium or in cases where moderating materials are available. Table column definitions are presented below:

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

PROCESS OR EQUIPMENT: A defined subgroup of vessels, tanks, process and/or support equipment.

CRITICALITY SAFETY BASIS: The parameters utilized within the CSE or CSA for defense of nuclear criticality safety for identified PROCESS OR EQUIPMENT items. When multiple entries are present, this means that the defense within the CSE or CSA may be based on either parameter as appropriate.

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

AREA OR SYSTEM	PROCESS OR EQUIPMENT	CRITICALITY SAFETY BASIS	EVALUATION BOUNDING ASSUMPTIONS
Storage Pad	UF_6 Cylinders	Moderation	▪ UF_6 Only ▪ $\leq$ 1 wt. % H_2O Equivalent ▪ Full Reflection
ADU Conversion	UF_6 Cylinder	Moderation	▪ UF_6 Only ▪ $\leq$ 1 wt. % H_2O Equivalent ▪ Partial Reflection
	Vaporizer	Geometry (Level Control)	▪ Homogeneous UO_2F_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Cold Trap System (ADU and IDR)	Geometry	▪ Homogeneous UO_2F_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection

AREA OR SYSTEM	PROCESS OR EQUIPMENT	CRITICALITY SAFETY BASIS	EVALUATION BOUNDING ASSUMPTIONS
	Sump	Geometry (Level Control)	• Homogeneous UO_2F_2 • Optimum H_2O Moderation • Full Concrete Reflection On 5 Sides and Partial Reflection On 1 Side
	Hydrolysis	Geometry	• Homogeneous UO_2F_2 • Optimum H_2O Moderation • Partial Reflection
	Nitrate Vessel	Geometry	• Homogeneous UO_2F_2 • Optimum H_2O Moderation • Partial Reflection
	Precipitation	Geometry	• Homogeneous UO_2F_2 • Optimum H_2O Moderation • Partial Reflection
	Centrifugation	Geometry	• Homogeneous UO_2 • Optimum H_2O Moderation • Partial Reflection
	Drying	Geometry	• Homogeneous UO_2 • Optimum H_2O Moderation • Partial Reflection
	Elevator	Geometry	• Homogeneous UO_2 • Optimum H_2O Moderation • Partial Reflection
	Calcination	Geometry	• Homogeneous UO_2 • Optimum H_2O Moderation • Partial Reflection
	Comminution	Geometry	• Homogeneous UO_2 • Optimum H_2O Moderation • Partial Reflection
	Calciner Scrubber System	Geometry	• Homogeneous UO_2 • Optimum H_2O Moderation • Partial Reflection
	Conversion Liquid Effluent Tanks	Concentration Neutron Absorber	• Homogeneous UO_2 • $\leq$ 15 gms. U^{235} / Liter • Partial Reflection

AREA OR SYSTEM	PROCESS OR EQUIPMENT	CRITICALITY SAFETY BASIS	EVALUATION BOUNDING ASSUMPTIONS
IDR Conversion	UF_6 Cylinder	Moderation	▪ Homogeneous UF_6 ▪ ≤ 1 wt. % H_2O Equivalent ▪ Full Reflection
	Vaporizer	Geometry (Level Control)	▪ Homogeneous UO_2F_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Sump	Geometry (Level Control)	▪ Homogeneous UO_2F_2 ▪ Optimum H_2O Moderation ▪ Full Concrete Reflection On 5 Sides and Partial Reflection On 1 Side
	DE Chamber / Carbon Filters	Moderation	▪ Homogeneous UO_2 ▪ ≤ 10 wt. % H_2O Equivalent ▪ Partial Reflection
	Kiln	Moderation	▪ Homogeneous UO_2 ▪ ≤ 10 wt. % H_2O Equivalent ▪ Partial Reflection
	Check Hopper System	Geometry	▪ Homogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Conversion Liquid Effluent Tanks	Geometry	▪ Homogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
Powder Blending	Bulk Blender	Moderation	▪ Homogeneous UO_2 ▪ ≤ 1 wt. % H_2O Equivalent ▪ Partial Reflection
ADU Pelleting	Bulk Container	Moderation	▪ Homogeneous UO_2 ▪ ≤ 10 wt. % H_2O Equivalent ▪ Partial Reflection
	Powder Transport	Geometry Moderation	▪ Homogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection

	Compaction	Geometry (Level Control) Moderation	▪ Heterogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Granulation	Geometry Moderation	▪ Heterogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Pressing	Geometry	▪ Heterogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Sintering	Geometry	▪ Heterogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Pellet Grinding	Geometry	▪ Heterogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
Rods	Loading System	Geometry	▪ Heterogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Inspection	Geometry	▪ Heterogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
Final Assembly	Fabrication	Geometry Neutron Absorber	▪ Heterogeneous UO_2 ▪ Full Interstitial Moderation ▪ Full Reflection
	Inspection	Geometry Neutron Absorber	▪ Heterogeneous UO_2 ▪ Full Interstitial Moderation ▪ Full Reflection
	Washing	Geometry Neutron Absorber	▪ Heterogeneous UO_2 ▪ Full Interstitial Moderation ▪ Full Reflection
UF_6 Cylinder Washing	Cylinder System	Mass	▪ Homogeneous UO_2F_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Eduction System	Geometry	▪ Homogeneous UO_2F_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection

	System Storage Vessels	Geometry	▪ Homogeneous UO_2F_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Precipitation System	Geometry	▪ Homogeneous UO_2F_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Filtration System	Geometry	▪ Homogeneous UO_2F_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
URRS Dissolver	Dissolver Vessels	Geometry	▪ Homogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	System Filters	Geometry	▪ Homogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	UN Product Tanks	Geometry	▪ Homogeneous UN ▪ $\leq$ 1000 gms. U / Liter ▪ Partial Reflection
	System Support Tanks	Geometry	▪ Homogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
URRS Scrap Processing	Low Level Waste Processing System	Concentration Mass	▪ Homogeneous UO_2 ▪ $\leq$ 15 gms. U^{235} / liter ▪ Partial Reflection
	Incinerator System	Mass	▪ Homogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Ash Recovery System	Geometry Mass	▪ Homogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Liquid Honing System	Concentration Mass	▪ Homogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Ultrasonic Cleaning System	Concentration Mass	▪ Homogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection

	Shredder System	Mass	▪ Homogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Mop Water System Equipment	Geometry Mass	▪ Homogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
Solvent Extraction	Dissolver Equipment System	Mass	▪ Homogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Adjustment Equipment	Mass Concentration	▪ Homogeneous UN ▪ Optimum H_2O Moderation ▪ Partial Reflection
	System Filters	Geometry	▪ Homogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Solvent/Extraction Equipment	Geometry	▪ Homogeneous UN ▪ $\leq$ 1000 gms. U / Liter ▪ Partial Reflection
	Concentration Equipment	Geometry	▪ Homogeneous UN ▪ $\leq$ 1000 gms. U / Liter ▪ Partial Reflection
	Fluoride Stripping Equipment	Concentration	▪ Homogeneous UO_2F_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
IFBA	Mop Water System Equipment	Geometry	▪ Homogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Pellet Stripping System Equipment	Geometry	▪ Heterogeneous UO_2 ▪ $\leq$10 wt. % H_2O Equivalent ▪ Partial Reflection
	Pellet Coating System	Geometry	▪ Heterogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Rod Loading System	Geometry	▪ Heterogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection

	Inspection	Geometry	▪ Heterogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
URRS Waste Treatment	Advanced Wastewater Treatment System	Concentration Mass	▪ Homogeneous UO_2 ▪ 5 wt. % U ▪ Partial Reflection
	Storage Tanks	Concentration Mass	▪ Homogeneous UO_2 ▪ 5 wt. % U ▪ Partial Reflection
	Ammonia Recovery System	Concentration Mass	▪ Homogeneous UO_2 ▪ 5 wt. % U ▪ Partial Reflection
UN	Bulk Storage Tank System	Concentration	▪ Homogeneous UN ▪ $\leq$ 15 gms. U^{235} / liter ▪ Partial Reflection
	Storage Pad	Geometry Concentration	▪ Homogeneous UN ▪ $\leq$ 23 gms. U^{235} / liter ▪ Full Concrete Reflection On 5 Sides and Partial Reflection On 1 Side
Miscellaneous	Laboratories	Geometry Mass	▪ Heterogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Ventilation Systems	Geometry Mass	▪ Homogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Ventilation Systems	Moderation Mass	▪ Homogeneous UO_2 ▪ $\leq$10 wt. % H_2O Equivalent ▪ Partial Reflection
	Scrubber Systems	Geometry Concentration	▪ Homogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Hoods & Containment Heterogeneous Mat'l.	Geometry Mass	▪ Hetrogeneous UO_2 ▪ $\leq$10 wt. % H_2O Equivalent ▪ Partial Reflection
	Hoods & Containment Homogeneous Mat'l.	Geometry Mass	▪ Homogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection

Storage	Wet Material Containers	Geometry	▪ Homogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Full Reflection
	Dry Material Containers	Geometry Moderation	▪ Homogeneous UO_2 ▪ $\leq$10 wt. % H_2O Equivalent ▪ Partial Reflection
	Wet Material Containers	Geometry	▪ Heterogeneous UO_2 ▪ Optimum H_2O Moderation ▪ Partial Reflection
	Dry Material Containers	Geometry Moderation	▪ Heterogeneous UO_2 ▪ $\leq$10 wt. % H_2O Equivalent ▪ Partial Reflection
	Dry Material Containers (annular pellets only)	Geometry Moderation	▪ Heterogeneous UO_2 ▪ $\leq$15 wt. % H_2O Equivalent ▪ Partial Reflection
	Rod Storage	Geometry	▪ Heterogeneous UO_2 ▪ Full Interstitial Moderation ▪ Full Reflection
	Fuel Assembly Storage	Geometry Neutron Absorber	▪ Heterogeneous UO_2 ▪ Full Interstitial Moderation ▪ Full Reflection
	Pellet Cabinets	Geometry Neutron Absorber	▪ Heterogeneous UO_2 ▪ Full Interstitial Moderation ▪ Partial Reflection

6.2.4 CRITICALITY SAFETY CONTROLLED PARAMETERS

Nuclear criticality safety will be achieved by controlling one or more parameters of a system within subcritical limits with sufficient factors of safety as described in Subsection 6.2.1 of this License Application. The Criticality Safety Evaluation process will be used to identify the significant parameters affected within a particular system. All assumptions relating to process/equipment/material theory, function, and operation, including credible upset conditions, will be justified, documented, and independently reviewed.

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

(a) Geometry

Safe geometry defines the characteristic dimension of importance for a single unit of a specific geometric shape such that nuclear criticality safety will not be dependent on any other parameter except enrichment (e.g., optimal moderation, reflector thickness, or concentration).

Geometry defines the characteristic dimension of importance for a single unit of a specific geometric shape such that criticality safety will be maintained in conjunction with one or more other parameters such as material form, concentration, and/or reflection.

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

(b) Mass

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

Whenever mass control is established for individual rooms or groups of rooms, detailed records will be maintained for mass transfers into and out of these rooms. Establishment of mass limits will involve consideration of potential moderation, reflection, geometry, spacing, and material concentration. The evaluation will consider normal operations and expected process upsets for determination of the actual mass limit for the system and for the definition of subsequent controls. When only administrative controls are used for mass controlled systems, double batching is assumed to be the worst credible upset condition.

(c) Moderation

Moderation control may be used for nuclear criticality safety control of systems within the facility, and if utilized, will incorporate the criteria identified below:

(c.1) Controls will be established to ensure that the interstitial moderator is maintained within the analyzed system defined limits.

Two independent controls/measurements or the analysis of two independent samples will be utilized to document this compliance. The system for collecting, preparing, analyzing, and posting of results pertaining to sample evaluation will be designed to ensure the results obtained are independent.

(c.2) Controls will be established to prevent moderator from entering the system after initial loading has occurred. The minimum protection will be that two independent barriers preventing moderator from entering the system must fail before the system can be compromised. Examples include a combination of system containment/roof, system containment/pipes, double roofs, etc. A method for detection of failure of the outermost barrier will be established and a program to maintain the quality of the barrier will be in place and routinely inspected. All barriers will be tested for leakage during initial installation.

(c.3) The evaluation of a process or system under moderation control will include the establishment of limits for the ratio of maximum moderator to fissile material for both normal operating conditions and credible process upsets and this analysis will be supported by parametric studies. Transportation of materials outside of moderation control areas will be proceduralized. Maintenance and inspection programs will be defined and implemented through plant procedures. The quality and basis for selection of the barriers will be documented within the CSE process. Controls for the introduction and usage of moderating materials within areas that are under moderation control will be defined and approved by the Nuclear Criticality Safety Function.

(d) Interaction

Nuclear criticality safety evaluations will consider the potential effects of interaction. Methods utilized are:

(d.1) Non-Interaction Method

A non-interacting unit is defined as a unit that is under concentration control, contains fixed neutron absorbers, is under moderation control, or is spaced an approved distance from other units such that the multiplication of the subject unit is not increased.

Additionally, 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. For solid angle interaction analyses, a unit where the contribution to the total solid angle in the array is less than 0.005 steradians will also be considered non-interacting (provided the total of all such solid angles neglected is less than one half of the total solid angle for the system).

Transfer pipes no greater than a 2-inch diameter and ventilation lines may be excluded from interaction considerations provided that an evaluation is performed to document the basis for treatment and application as a non-interacting unit.

(d.2) Solid Angle Method

Solid angle criteria may be used to determine the total solid angle subtended by each unit in an array, for interaction calculations. The solid angle criteria of TID-7016 (Rev. 2) will apply, as supplemented by reflector conditions of no more than full water reflection on six sides of the array or no more than the equivalent of full concrete reflection on three sides of the array.

The solid angle validation manual will be used for guidance in performing these calculations. Solid angles will be determined by the point-to-plane method. The multiplication factor of a single unit will be determined from validated computer calculations.

(d.3) Monte Carlo Method

Individual unit multiplication and array interaction may be evaluated using computer codes (e.g., XSDRN, KENO, MCNP, etc.) for which validations have been documented. When array interaction is evaluated in this manner, the maximum allowed K_{eff}, including all biases and uncertainties at a one-sided 95/95 confidence level, will not exceed 0.95.

(e) Enrichment

Enrichment control may be utilized to limit the percent U^{235} within a process, vessel, or container, thus providing a method for nuclear criticality safety control. Active engineered and/or administrative controls will be required to verify enrichment and to prevent the introduction of uranium at unacceptable enrichment levels within a defined system.

In cases where enrichment control is not utilized, the maximum credible enrichment will be assumed.

(f) Concentration & Density

Concentration control may be used for nuclear criticality safety control of systems within the facility, and if utilized, will incorporate the following criteria:

(f.1) Controls will be established to ensure that the concentration level is maintained within the analyzed system defined limits.

Two independent controls/measurements or the analysis of two independent samples will be utilized to document this compliance. The system for collecting, preparing, analyzing, and posting of results pertaining to sample evaluation will be designed to ensure the results obtained are independent.

(f.2) Controls will be established to prevent concentration within the system after initial loading has occurred. These events might include evaporation, precipitation, freezing, settling, or chemical phase changes. Each system will have in place controls necessary to detect and/or mitigate the effects of internal concentration within the system.

(f.3) For those systems which utilize concentration as a controlled parameter for ensuring nuclear criticality safety, the CSE's will demonstrate the solubility limits of the SNM composition, identify possible precipitating agents to the operators through procedures and ensure appropriate precautions are being taken to ensure such agents are not introduced, and provide a positive means of preventing unwanted transfers if a possibility exists for precipitating agents to be transferred via connected processes.

(g) Reflection

Many systems are designed and operated with the assumption of full reflection. However, certain system designs will be analyzed, approved, and operated in situations where reflection is less than full. Two such methods (partial and bare reflection) may be employed. Partial reflection is defined as 1 inch H_2O equivalent in systems where equipment location or design limits the placement of moderating materials, including humans, near the specific system. Bare reflection (no moderator considered) will be utilized in systems where equipment location or design impose physical limits on the ability to place moderating materials, including humans, near the specific system.

For all system evaluations, the neutron reflection properties of the credible

process environment will be analyzed. This may limit the available use of partial or bare neutron reflection methodology for specific systems.

(h) Neutron Absorber

Neutron absorbing materials may be utilized to provide a method for nuclear criticality safety control for a process, vessel or container. If used, these will take the form of solid materials such as borosilicate-glass Raschig rings, gadolinium plates, borated stainless steel, or other solid neutron absorbing materials. The use of neutron absorbers in this manner will be defined as a passive engineered control.

When Raschig rings are used to control nuclear criticality safety, their use and maintenance will be in accordance with ANSI/ANS-8.5-1986, with the following exceptions for basic solutions:

- System PH maintained ≤ 11; and
- System temperature maintained ≤ 60 degrees centigrade.

For other fixed neutron absorbers, the following requirements apply:

- The composition of the absorber will be measured and documented prior to first use; and,
- The presence of the absorber in a process, vessel or container will be verified on an annual basis. The method of verification may take the form of traceability (i.e. serial number,etc.), visual inspection or direct measurement.

(i) Composition

The CSE for each system will determine the effects of material composition within the process being analyzed and identify the basis for composition selection pertinent to subsequent system modeling activities.

(j) Heterogeneity

The effects of heterogeneity within a system can be significant and will be considered within the CSE when appropriate. (It is therefore important to distinguish between homogeneous and heterogenous systems, especially at lower enrichments, where lumping of the uranium can have an adverse impact on the

nuclear criticality safety of a system.)

The primary concern is the determination of the particle size that will be permitted in a uniform system such that the system will still be considered homogeneous from the viewpoint of nuclear criticality safety. A review of nuclear criticality safety calculations has demonstrated that for systems where the particle size is less than or equal to 150 microns, the process can be considered homogeneous. For systems where the particle size is greater than or equal to 1000 microns, the system is purely heterogeneous. The evaluation of systems where the particle size falls within the range of 150 - 1000 microns will take into consideration the effects of heterogeneity appropriate for the process being analyzed.

(k) Process Characteristics

Within certain manufacturing operations, credit may be taken for physical and chemical properties of the process and/or materials as nuclear criticality safety controls. When so utilized, this credit will be predicated upon the following requirements:

(k.1) The bounding assumptions will be defined through the CSE process and operational limits will be identified within the Criticality Safety Analysis (CSA); and, will be specifically communicated, through training and procedures, to appropriate manufacturing personnel.

(k.2) Utilization of such process and/or material characteristics will be based on established physical or chemical reactions, known scientific principles, and/or facility-specific experimental data supported by operational history.

Examples of such an application include:

- Conversion and oxidation processes that produce dry powder ($\leq$ 10 wt. % H_2O) as a product of high temperature reactions.
- Experimental data demonstrating low moisture pickup ($\leq$ 3 wt. % H_2O) from room air in ventilation equipment.
- Experimental/historical process data demonstrating uranium oxide powder flow characteristics to be directly proportional to the quantity of moisture present.

6.2.5 MOVABLE NON-FAVORABLE GEOMETRY (NFG) CONTAINERS

Movable NFG container usage within the CFFF will be rigorously controlled and utilized only when other practical methods are unavailable. Prior to use of a movable NFG container in the Chemical Manufacturing Area, a comprehensive analysis will be performed. The key components of this analysis follow:

(a) Manufacturing personnel will provide justification for the use of the movable NFG container.

(b) Controls will be identified and implemented to provide assurance that proposed movements of an NFG container can be performed safely.

(c) A Criticality Safety Analysis will be completed and approved prior to use of the movable NFG container within the facility.

6.3 ALARM SYSTEM

6.3.1 SPECIFICATIONS

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

Should the nuclear criticality alarm system be out of service for a time period exceeding four hours, all movements of SNM will cease until the alarm service has been restored or special monitoring, approved by the Nuclear Criticality Safety Function, will be implemented. Routine testing, calibration and/or maintenance of the system will be permitted with no suspension of SNM movements.

6.3.2 OPERATION

The nuclear criticality alarm system initiates immediate evacuation of the facility. Employees will be trained in recognizing the evacuation signal, which is a continuous-sounding siren. This system, and proper response protocol, is detailed in the CFFF Site Emergency Plan.

6.3.3 MAINTENANCE

The nuclear criticality alarm system will be a safety-related system and will be maintained through routine calibration and scheduled functional tests conducted in the manner described in Subsection 6.2.1 of this License Application. In the event of loss of normal power, emergency power will be automatically connected to the system.

6.4 CONTROL DOCUMENTS

6.4.1 CRITICALITY SAFETY EVALUATION (CSE)

The Criticality Safety Evaluation is essentially a subset of the Integrated Safety Assessment (ISA) defined in Chapter 4.0 of this License Application. The CSE identifies and documents the basis of nuclear criticality safety for a particular system. A CSE is prepared or updated for each new or significantly modified system within the CFFF. The level of detail for a particular CSE will be determined based on the complexity of the system or proposed changes and will be documented by the Nuclear Criticality Safety Function Engineer and approved by the Nuclear Criticality Safety Function Manager. The same process will be used to identify the CSE requirements for an existing system modification. Therefore, the scope and content of any particular CSE reflects the needs and characteristics of the system being analyzed and includes appropriate information from the following:

(a) Process Description

This section presents a precise narrative definition of normal operation as it relates to each defined system. It also provides a schematic representation of the system and a narrative outline of the system transfer interconnections, with text references that detail normal operating boundaries (e.g., composition, concentrations, flows, and sampling). References are provided to all relevant drawings and procedures; and, photographs, diagrams, tables, and charts depicting crucial system and subsystem equipment.

(b) Process Theory

This section presents a narrative description of the normal process operating parameters including the ranges of conditions expected, for each defined system. It also provides descriptions of upset conditions which have the potential for exceeding established safety limits. References are provided documenting the sources of the process theory.

(c) Process Design and Equipment

This section presents the dimensions, construction materials, and design configuration of process equipment, vessels, and transfer lines of each defined system. It also provides a precise narrative definition of subsystem equipment controls and features, as related to the defined system, and a tabulation of relevant reference drawings.

(d) Drawings and Operating Procedures

This section presents a complete reference listing of all documents used in performing the formal Process Hazards Analysis, and describes their relationship to the evaluation process, for each defined system. It also provides photographs of the system/subsystem equipment that had relevance to (and were used during) the analysis process. (All other documents collected for review and/or information purposes will be retained as part of the Data Pack for the system's Criticality Safety Evaluation.)

(e) Safety Analysis

This section constitutes the results of a comprehensive nuclear criticality safety review for each component within the defined system. The evaluation reviews the criticality safety control reliability through documentation of the bases identified for each control. Reliability will be systematically assessed through identification of the relative strengths and weaknesses of each control including the consideration for potential common mode failures. Included in the evaluation will be a summary of facility practices to ensure the quality of safety significant measurements for the system, as well as, summary documentation of the controls within each component of the system.

The Safety Evaluation incorporates portions of the Criticality Safety Analyses (CSA) which will be included within the CSE Appendix. The CSA defines the safety limits for each component within the system.

(f) Process Hazards Analysis

This section presents each defined system's Hazards and Operability Analysis (HAZOP) Table or What If/Checklist Results, Fault Trees, and Event Trees. (This will constitute an integrated analysis that includes Nuclear Criticality Safety.) Essential elements of the HAZOP Table or What If/Checklists Results will include a listing of each process/system upset and deviation disclosed in the analysis, the significant causes and consequences of each such process/system

upset or deviation, and the controls in place to prevent each cause and/or mitigate each consequence. The complete Analyses Report, providing the detailed results of the analysis in a narrative format, will be retained as part of the documentation for each system's Criticality Safety Evaluation.

(g) License Compliance

The criticality safety basis and the bounding assumptions generated during the CSE process will be reviewed for compliance with Subsection 6.2.3 of this License Application.

(h) Appendix

This section presents a summary of ancillary information (such as calculations, parametric sensitivity studies, references, etc.) for each defined system.

6.4.2 CRITICALITY SAFETY ANALYSIS (CSA)

The CSA is comprised of a review of criticality safety controls to identify the minimum requirements necessary to ensure nuclear criticality safety. Expert determination is utilized to evaluate system reliability and to document the adequacy and effectiveness of each control.

6.4.3 ANALYSES METHODOLOGIES

(a) K_{eff} Limit

Validated computer analytical methods will be utilized to evaluate individual vessels or potential vessel interaction. The computed K_{eff} for normal operating conditions and expected process upsets will be less than or equal to 0.95 including applicable biases and calculational uncertainties. Credible operating conditions include conditions expected to be encountered during routine operations, process upsets, and credible accident situations. The sensitivity of key parameters with respect to the effect on K_{eff} will be evaluated for each system such that adequate system controls are defined for the analyzed system.

(b) Analytical Analysis

Criticality calculations are currently performed using the AMPX modules NITAWL-S and XSDRNPMS for cross-section generation and KENO Va for reactivity calculations, but additional Monte Carlo codes such as MCNP may be used after validation as described in Paragraph (c) below. These methods have

been benchmarked to various critical experiments to verify their applicability.

The current design method starts with 227 energy group cross-sections generated from ENDF/B-V data. The AMPX system codes, NITAWL-S and XSDRNPMS, will be used for cross-section library processing. The NITAWL-S program performs the self-shielded resonance cross-section corrections that are appropriate for each particular geometry. The Nordheim Integral Treatment will be used. Energy and spatial weighting of the cross-sections will be performed by the XSDRNPMS program, a one-dimensional transport code. These multi-group cross-section sets will then be used as input to KENO Va, a three dimensional Monte Carlo code.

(c) Validation Techniques

Nuclear criticality safety analyses conducted utilizing computerized methodology will be validated in accordance with the criteria described in Section 4.3 of ANSI/ANS-8.1-1983.

(d) Computer Software & Hardware Configuration Control

The configuration of the hardware "calculational platform" used in the support of software for nuclear criticality safety calculations will be maintained such that only authorized system administrators will be allowed to make system changes. System changes will be conducted in accordance with an approved configuration control program that addresses both hardware and software qualification. System operability verification will be performed to alert users to any changes that would impact the operation of "codes" on the calculational platform.

Software designated for use in nuclear criticality safety calculations on the calculational platform will be compiled into working code versions with executable files that are traceable by length, time, date, and version. Working code versions of compiled software will be qualified on the basis that physical critical experiments were modeled using an established methodology with the differences in experiment and analysis being used to calculate bias and uncertainty values to be applied to the obtained results.

Modifications to hardware or software that is essential to the calculational process will be followed by code operability verification, in which case, selected calculations will be performed to verify identical results from previous analyses. Deviations noted in code verification that might alter the bias or uncertainty will require re-qualification of the code prior to continued use.

(e) Technical Review

Independent technical reviews of criticality safety assessments, criticality safety evaluations, or calculations in support of limits specified in CSA's or CSE's will be performed. The Nuclear Criticality Safety Function Manager will assign a qualified reviewer the task of performing the independent technical review.

The technical reviewer will verify that the proposed calculational geometry model and configuration adequately represent the system being analyzed. In addition, the reviewer will verify that the proposed material characterizations such as density, concentration, etc., adequately represent the system.

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

- Verify the calculations with an alternate computational method.
- Verify the calculations by performing a comparison to results from a similar design or to similar previously performed calculations.
- Verify the calculations using a Technical Review Suggestion List for guidance. This method will include specific checks of the computer codes used, as well as, evaluations of code input and output.
- Verify the calculations with a custom method. Provide detailed information that describes the chosen methodology.

After the technical review has been completed, the original system analysis and the information provided in the technical review will be reviewed and approved by the Nuclear Criticality Safety Function Manager.

(f) Solid Angle Method

Solid angle analysis methods may be utilized within the CFFF as described in Subsection 6.2.4, Paragraph (d.2) of this License Application.