CHAPTER 6

ENGINEERED SAFETY FEATURES

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Acronyms and Abbreviations

Acronym/Abbreviation	Definition
$\mu Ci/m^3$	microcuries per cubic meter
ACGIH	American Conference on Governmental Industrial Hygienists
AGS	American Glovebox Society
AHRI	Air-Conditioning Heating and Refrigeration Institute
AIHA	American Industrial Hygiene Association
ALARA	as low as reasonably achievable
AMCA	Air Movement and Control Association
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASHRAE	American Society of Heating, Refrigerating and Air-Conditioning Engineers, Inc.
ASME	American Society of Mechanical Engineers
BS	Bachelor of Science
Btu	british thermal unit
Btu/hr	british thermal units per hour
C	Celsius
CAAS	criticality accident and alarm system
CAM	continuous air monitor
CAMS	continuous air monitoring system
CFR	Code of Federal Regulations
DAC	derived air concentrations
DBA	design basis accident
DBE	design basis event
ESF	engineered safety feature

Acronyms and Abbreviations (cont'd)

Acronym/Abbreviation	Definition
ESFA	engineered safety feature actuation system
F	Fahrenheit
FDWS	facility demineralized water system
FHA	fire hazard analysis
FICS	facility integrated control system
FPC	facility process conditions
HCFPS	hot cell fire detection and suppression system
HEPA	high efficiency particulate air
HS&E	health safety & environment
HVAC	heating, ventilation, and air conditioning
I-125	iodine-125
I-131	iodine-131
IBC	International Building Code
ICBS	irradiation cell biological shielding
ICC	International Code Council
IF	irradiation facility
IMC	International Mechanical Code
ISG	interim staff guidance
IU	irradiation unit
k_{eff}	effective neutron multiplication factor
kWh	kilowatt-hour
LCO	limiting conditions for operations
LFL	lower flammability limit
LOOP	loss of off-site power
LSC	Life Safety Code

Acronyms and Abbreviations (cont'd)

Acronym/Abbreviation	Definition
LWPS	light water pool system
LWR	light water reactor
MCNP	Monte Carlo N-Particle
Mo-99	molybdenum-99
N-16	nitrogen-16
NCS	nuclear criticality-safety
NCSE	nuclear criticality-safety evaluation
NCSP	Nuclear Criticality Safety Program
NEMA	National Electrical Manufacturers Association
NFDS	neutron flux detection system
NFPA	National Fire Protection Association
NGRS	noble gas removal system
Np-239	neptunium-239
NRC	U.S. Nuclear Regulatory Commission
PCLS	primary closed loop cooling system
PFBS	production facility biological shield
PSAR	Preliminary Safety Analysis Report
PSB	primary system boundary
Pu-239	plutonium-239
PVVS	process vessel vent system
RAM	radiation area monitor
RAMS	radiation area monitoring system
RCA	radiologically controlled area
RDS	radioactive drain system
rem	roentgen equivalent man
RICS	radiological integrated control system

Acronyms and Abbreviations (cont'd)

Acronym/Abbreviation	Definition
RPF	radioisotope production facility
RV	radiologically controlled area ventilation system
RVSA	radiologically controlled area supply air
RVZ1	radiologically controlled area ventilation zone 1
RVZ2	radiologically controlled area ventilation zone 2
SMACNA	Sheet Metal and Air Conditioning Contractors National Association
SDD	system design description
SHINE	SHINE Medical Technologies, Inc.
SNM	special nuclear material
SRM	stack release monitoring
SSC	structure, system, or component
TOGS	target solution vessel off gas system
TPCS	target solution vessel process control system
TPS	tritium purification system
TRPS	target solution vessel reactivity protection system
TSV	target solution vessel
U-238	uranium-238
UFL	upper flammability limit
UN	uranyl nitrate
UNCS	uranyl nitrate conversion system
UPS	uninterruptible power supply
UPSS	uninterruptible electrical power supply system
UREX	uranium extraction
vol%	volume percent
Xe-133	xenon-133

CHAPTER 6

ENGINEERED SAFETY FEATURES

6a1 HETEROGENEOUS REACTOR ENGINEERED SAFETY FEATURES

The SHINE Medical Technologies, Inc. (SHINE) facility is not a heterogeneous reactor. Therefore, this section does not apply to the SHINE facility.

6a2 IRRADIATION FACILITY ENGINEERED SAFETY FEATURES

The engineered safety features (ESF) of this portion of the facility are those systems designed to provide safety and defense in depth during start-up, normal operations, and shutdown as well as mitigate the consequences of postulated accidents, in spite of the fact that these postulated accidents are unlikely. These features can be divided into active and passive categories and include the following functions:

Confinement

These systems provide active and passive protection against the potential release of radioactive material to the environment during normal conditions of operations and following a DBA. The confinement systems provide for active isolation of piping and heating, ventilation, and air conditioning (HVAC) systems penetrating confinement boundaries in certain post-accident conditions.

6a2.1 SUMMARY DESCRIPTION

The intent of this section is to identify ESFs and provide a cross reference to postulated DBAs and events which the ESF acts to mitigate. The subsequent sections provide a detailed review of the ESF initiation and response to the event.

Section 6a2.2 and its related subsections describe the ESFs, their modes of initiation and operation, and describe the structures, systems, and components (SSCs) which provide the features in detail.

Table 6a2.1-1 provides a reference summary of this information.

Table 6a2.1-1 Summary of IF Design Basis Accidents and ESF Provided for Mitigation

Engineered Safety Feature (ESF)	Irradiation Facility Design Basis Accidents Mitigated by ESF	SSCs which provide ESF	Detailed Description Section or Subsection
Confinement	• Mishandling or malfunction of target solution • Mishandling or malfunction of equipment affecting the PSB • Tritium Purification System Design Basis Accident	• Irradiation unit cells including penetration seals • RCA ventilation system Zone 1 (RVZ1) ductwork up to bubble-tight isolation dampers • Bubble-tight isolation dampers • Isolation valves on piping systems penetrating the irradiation unit cells • TOGS shielded cells including penetration seals • Engineered safety features actuation system (ESFAS) • Double-walled pipe used for TPS • TPS gloveboxes • TPS confinement system	6a2.2.1

6a2.2 IRRADIATION FACILITY ENGINEERED SAFETY FEATURES DETAILED DESCRIPTION

The ESFs are passive or active features designed to mitigate the consequences of accidents and to keep the radiological and chemical exposures to the public, the facility staff, and the environment within acceptable values. This section provides the details of design, initiation, and operation of ESFs that are provided to mitigate the design basis accidents (DBAs) tabulated in Section 6a2.1.

According to the Final Interim Staff Guidance (ISG) Augmenting NUREG-1537, accident-initiating events (IEs) and scenarios, the following design basis accidents are to be addressed for the irradiation facility (IF):

a. Insertion of excess reactivity/inadvertent criticality
b. Reduction in cooling
c. Mishandling or malfunction of target solution
d. Loss of off-site power
e. External events
f. Mishandling or malfunction of equipment affecting the PSB
g. Large un-damped power oscillations (fuel temperature/void-reactivity feedback)
h. Detonation and deflagration in PSB
i. Unintended exothermic chemical reactions other than detonation
j. Primary system boundary system interaction events
k. Facility-specific events
    - Inadvertent exposure to neutrons from neutron driver
    - Irradiation facility fires
    - Tritium Purification System Design Basis Accident

The following design basis accidents (DBAs) do not have radiological consequences that require mitigation by ESFs:

a. Insertion of excess reactivity/inadvertent criticality
b. Reduction in cooling
c. Loss of off-site power
d. External events
e. Large un-damped power oscillations (fuel temperature/void-reactivity feedback)
f. Detonation or deflagration in PSB
g. Unintended exothermic chemical reactions other than detonation
h. Primary system boundary system interaction events
i. Facility-specific events
    - Inadvertent exposure to neutrons from neutron driver
    - Irradiation facility fires

The three DBAs requiring ESFs to mitigate the consequences are identified in Table 6a2.1-1.

6a2.2.1 CONFINEMENT

6a2.2.1.1 Introduction

Confinement is a term used to describe the low-leakage boundary that surrounds radioactive materials released during an accident and the associated RCA ventilation system (RV) components. Confinement systems are designed to localize release of radioactive material to controlled areas in normal operational states and mitigate the consequences of DBAs. Radiation protection control features such as adequate shielding and RV minimize hazards normally associated with radioactive materials. The principal design and safety objective of the confinement systems is to protect on-site personnel, the public, and the environment. The second design objective is to minimize the reliance on administrative or complex active engineering controls to provide a confinement system that is as simple and fail-safe as reasonably possible.

The target solution vessel (TSV), TSV dump tank, TSV off-gas system (TOGS), and associated components act as the primary pressure boundary and are safety-related SSCs. Together they act as the primary fission product boundary. The confinement boundary of the IU cell and TOGS shielded cell encloses the PSB.

The tritium purification system (TPS) double-walled piping and TPS confinement system provides confinement of the tritium supplied to the IU. The confinement boundary of the TPS gloveboxes encloses the TPS.

Confinement of the IU cells is achieved through the RV, ESFAS, and the biological shielding provided by the steel and concrete structures comprising the walls, roofs, penetrations of the IU cell and TOGS shielded cell. Shielding of the IU cells and TOGS shielded cells is discussed in detail in Section 4a2.5.

Confinement of the TPS gloveboxes is provided by the RV, ESFAS, and biological shielding comprising the walls, roofs, and penetration seals of the TPS gloveboxes. The biological shielding is described in Section 4a2.5.

6a2.2.1.2 Confinement System and Components

The ventilation system serving the IU cells, TOGS shielded cells, and TPS gloveboxes includes components whose functions are designated as nonsafety-related and safety-related. The ductwork, the isolation dampers, and the filter trains of RVZ1 are designated as safety-related components. Refer to Table 6a2.2-1 for a description of the system and component safety functions. Active confinement isolation components are required to operate as described below.

The IU cells, TOGS shielded cell, and TPS gloveboxes employ a combination passive-active confinement methodology. During normal operation, passive confinement is achieved through the contiguous boundary between the hazardous materials and the surrounding environment and is credited with confining the hazards generated as a result of DBAs.

This boundary includes the biological shield (created by the physical construction of the cell itself) and the extension of that boundary through the RCA Zone 1 (RVZ1) ventilation system. The extent of this passive confinement boundary extends from the upstream side of the intake high efficiency particulate air (HEPA) filter to the final downstream HEPA filter prior to exiting the building. For the TPS the confinement also includes the double-walled piping.

In the event of a DBA that results in a release in the IU cell, TOGS shielded cell, or TPS glovebox, radioactive material would be confined by the biological shield and physical walls of the cell itself. Each line that connects directly to the IU cell, TOGS shielded cell, or TPS glovebox atmosphere and penetrates the IU cell, TOGS shielded cell, or TPS glovebox is provided with redundant isolation valves to prevent releases of gaseous or other airborne radioactive material. Confinement isolation valves on piping penetrating the IU cell, TOGS shielded cell, or TPS glovebox are located as close as practical to the confinement boundary and active isolation valves are designed to take the position that provides greater safety upon loss of actuating power.

To mitigate the consequences of an uncontrolled release occurring within an IU cell, TOGS shielded cell, or TPS glovebox, as well as the off-site consequences of releasing noble fission products through the ventilation system prior to sufficient decay, the confinement barrier utilizes an active component in the form of bubble-tight isolation dampers (safety-related) on the inlet and outlet ventilation ports of each cell. This ESF effectively reduces the amount of ductwork in the confinement volume that needs to remain intact to achieve IU cell, TOGS shielded cell, or TPS glovebox confinement. These dampers close automatically (fail-closed) upon loss of power or receipt of a confinement isolation signal generated by the ESFAS. Following an initiating event, the ESFAS provides the confinement isolation signal that isolates the IU cells, TOGS shielded cell, or TPS glovebox. Refer to Subsection 7a2.5 for a description of the ESFAS.

A failure of the TPS outside the glovebox is mitigated by the TPS confinement system. The TPS confinement system uses isolation valves to stop a tritium leak outside the glovebox when a leak is detected.

Overall performance assurance of the active confinement components is achieved through factory and in-place testing. Duct and housing leak tests are performed in accordance with ASME N511 (ASME, 2008), with minimum acceptance criteria as specified in ASME AG-1 (ASME, 2009). Specific owner’s requirements with respect to acceptable leak rates are based on the safety analyses.

6a2.2.1.3 Functional Requirements

Active confinement components are designed to fail into a safe state if conditions such as loss of signal, loss of power, or adverse environments are experienced.

Mechanical, instrumentation, and electrical systems and components required to perform their intended safety function in the event of a single failure are designed to include sufficient redundancy and independence such that a single failure of any active component does not result in a loss of the capability of the system to perform its safety functions.

Mechanical, instrumentation, and electrical systems and components are designed to ensure that a single failure, in conjunction with an initiating event, does not result in the loss of the system’s ability to perform its intended safety function. The single failure considered is a random failure and any consequential failures in addition to the initiating event for which the system is required and any failures that are a direct or consequential result of the initiating event.

The design of safety-related systems (including protection systems) is consistent with IEEE Standard 379-2000 (IEEE, 2001)and Regulatory Guide 1.53 in the application of the single-failure criterion.

6a2.2.1.4 Confinement Components

The following components are associated with the secondary confinement barrier of the IU cells, TOGS shielded cell, or TPS glovebox, as previously described. Their specific materials, construction, and installation and operating requirements are evaluated based on the safety analysis.

Bubble-tight isolation dampers, designed, constructed and tested in accordance with ASME AG-1, Section DA “Dampers and Louvers” (ASME, 2009):

- Maintain their functional integrity.
- Maintain their rated leak-tightness following a seismic event.
- Maintain their structural integrity under fan shut-off pressure.
- Provide bubble-tight isolation upon receipt of a control signal or, in the event of loss of actuator power, by closure of actuator.
- Provide bubble-tight isolation when using manual actuator or when locked closed with power actuator removed.
- Relay damper full-open and full-closed position for control and indication by the use of limit switches.

Dampers are butterfly type, blade and frame fabricated of heavy-gage stainless steel. Total leakage is based on bubble solution test as outlined in ASME AG-1 2009, Section DA-5141 (ASME, 2009).

Ventilation ductwork and ductwork support materials meet the requirements of ASME AG-1, Article SA-3000 “Materials”. Supports are designed and fabricated in accordance with the requirements of ASME AG-1 2009, Section SA “Ductwork” (ASME, 2009).

Details of the TPS confinement system will be developed in final design and provided in the FSAR.

Low leakage seals are provided on each penetration through the IU cell, TOGS shielded cell, and TPS glovebox. For systems open to the IU cell, TOGS shielded cell atmosphere, or TPS glovebox, redundant isolation valves are provided.

6a2.2.1.5 Engineered Safety Feature Test Requirements

Engineered safety features are periodically tested to ensure that ESF components maintain operability and can provide adequate confidence that the system performs satisfactorily in service during postulated events.

To the extent possible, the ESFAS and the confinement ESF whose operation it initiates are designed to permit testing during plant operation. Testing actuation devices and actuated equipment may be done individually or in groups to avoid negative impact to plant operations.

6a2.2.1.6 Design Bases

For general discussion of the codes and standards used for the SHINE facility, see Chapter 3. For the design basis of the RVZ1, see Subsection 9a2.1.1. For the design basis of the IU cell and TOGS cell, see Section 4a2.5. For the design basis of the ESFAS, see Section 7a2.5. For the design basis of the TPS, see Section 9a2.7.1.

Potential variables, conditions, or other items that will be probable subjects of a technical specification associated with the IF confinement systems and components are provided in Chapter 14.

6a2.2.2 CONTAINMENT

The SHINE facility does not employ a containment feature. Due to the low temperature and power level of facility operations, the safety analysis demonstrates that confinement features are adequate to mitigate potential accidents. Therefore, Subsection 6b.2.1 is provided to describe the use of confinements as an ESF for the RPF.

Table 6a2.2-1 Irradiation Facility Confinement Safety Functions

System, Structure, Component	Description	Classification
RVZ1 IU cell isolation dampers	Provide confinement at IU cell and TOGS shielded cell boundaries	Safety-Related
ESFAS	Provides confinement isolation signal	Safety-Related
Isolation valves on piping systems	Provide confinement at IU cell and TOGS shielded cell boundaries, and TPS boundary	Safety-Related
IU cell and TOGS shielded cell including penetrations	Provide confinement	Safety-Related

6a2.3 EMERGENCY COOLING SYSTEM

Decay heat removal during accident scenarios is provided by the safety-related light water pool. No emergency core cooling system is required for the SHINE facility to mitigate the consequences of an accident. See Chapter 13.

6a2.4 IRRADIATION FACILITY ENGINEERED SAFETY FEATURES TECHNICAL SPECIFICATIONS

Potential variables, conditions, or other items that are probable subjects of a technical specification associated with the IF ESFs are provided in Chapter 14.

6a2.5 REFERENCES

ASME, 2008. In-Service Testing of Nuclear Air Treatment, Heating, Ventilating, and Air-Conditioning Systems, ASME N511-2007, American Society of Mechanical Engineers, February 2008.

ASME, 2009. Code on Nuclear Air and Gas Treatment, ASME AG-1-2009, American Society of Mechanical Engineers, September 2009.

IEEE, 2001. IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems, IEEE Standard 379-2000, Institute of Electrical and Electronics Engineers, September 2000, Reaffirmed March 2008.

6b RADIOISOTOPE PRODUCTION FACILITY ENGINEERED SAFETY FEATURES

6b.1 SUMMARY DESCRIPTION ENGINEERED SAFETY FEATURES

ESFs of this portion of the facility are those systems designed to mitigate the consequences of postulated accidents.

Confinement

These systems provide active and passive protection against the potential release of radioactive materials or chemicals to the environment following a design basis accident. The confinement systems provide for active isolation of piping and HVAC systems penetrating confinement boundaries in certain post-accident conditions.

The intent of this section is to identify ESFs and provide a cross reference to postulated DBAs and events which the ESF acts to mitigate. The subsequent sections provide a detailed review of the ESF initiation and response to the event.

Section 6b.2 and its subsections describe the ESFs, their modes of initiation and operation, and describe the structures, systems, and components (SSCs) which provide the features in detail.

Table 6b.1-1 provides a reference summary of this information.

Table 6b.1-1 Summary of RPF Design Basis Events and ESF Provided for Mitigation

Engineered Safety Feature (ESF)	Radioisotope Production Facility Design Basis Event Mitigated by ESF	SSCs which provide ESF	Detailed Description Section or Subsection
Confinement	• Critical equipment malfunction • Accidents with hazardous chemicals	• Hot cells including penetration seals • RCA ventilation system Zone 1 (including ductwork up to filters and filters) and Zone 2 • Bubble-tight isolation dampers • Tank vaults • Radiological integrated controls system (RICS) • Isolation valves on piping systems penetrating hot cells	6b.2.1

6b.2 RADIOISOTOPE PRODUCTION FACILITY ENGINEERED SAFETY FEATURES

The ESFs are passive or active features designed to mitigate the consequences of accidents and to keep the radiological and chemical exposures to the public, the facility staff, and the environment within acceptable values. This section provides the details of design, initiation, and operation of ESFs that are provided to mitigate the DBAs discussed in Section 6b.1. This includes chemical storage areas outside the RCA.

According to Chapter 13 of the Final ISG Augmenting NUREG-1537, the following DBAs are to be addressed for the RPF:

a. Critical equipment malfunction.
b. Inadvertent nuclear criticality in the RPF.
c. RPF fire.
d. Accidents with hazardous chemicals.
e. External events.

These DBAs encompass LOOP and operator errors (See Section 13b).

The following DBAs do not have consequences that require mitigation by ESFs:

a. Inadvertent nuclear criticality in the RPF.
b. RPF fire.
c. External events.

The two DBAs requiring ESFs to mitigate consequences are identified in Table 6b.1-1.

6b.2.1 CONFINEMENT

6b.2.1.1 Introduction

Confinement describes the low-leakage boundary surrounding radioactive or hazardous chemical materials released during an accident and parts of RVZ1 and RVZ2. Confinement systems localize releases of radioactive or hazardous materials to controlled areas and mitigate the consequences of DBAs. Personnel protection control features such as adequate shielding and RV minimize hazards normally associated with radioactive or chemical materials. The principal design and safety objective of the confinement system is to protect the on-site personnel, the public, and the environment. The second design objective is to minimize the reliance on administrative or complex active engineering controls and provide a confinement system that is as simple and fail-safe as reasonably possible.

This subsection describes the confinement systems for the RPF. The RPF confinement areas include hot cell enclosures for process operations and trench and vault enclosures for process tanks and piping.

Confinement is achieved through RV, RICS, and biological shielding provided by the steel and concrete structures comprising the walls, roofs, and penetrations of the hot cells. Shielding of the hot cells is discussed in detail in Subsection 4b.2.

6b.2.1.2 Confinement System and Components

The RV serving the RCA, outside of the IF, includes components whose functions are designated as nonsafety-related and safety-related. The ductwork, the isolation dampers, and the filter trains of RVZ1 are designated as safety-related. Refer to Table 6b.2-1 for a description of the system and component safety functions. Active confinement isolation components are required to operate as described below.

The hot cells employ a combination passive-active confinement methodology. During normal operation, passive confinement is achieved through the contiguous boundary between the hazardous materials and the surrounding environment and is credited with confining the hazards generated as a result of DBAs.

This boundary includes the biological shield (created by the physical construction of the cell itself) and the extension of that boundary through the RVZ1. The intent of the passive boundary is to confine hazardous materials while also preventing the introduction of external energy sources that could disturb the hazardous materials from their steady-state condition. The extent of this passive confinement boundary extends from the upstream side of the intake HEPA filter to the final downstream HEPA filter prior to exiting the building.

In the event of a DBA that results in a release in the hot cells, radioactive material would be confined by the biological shield and physical walls of the cell itself. Each line that connects directly to the hot cell atmosphere and penetrates the hot cell is provided with redundant isolation valves to prevent releases of gaseous or other airborne radioactive material. Confinement isolation valves on piping penetrating the hot cell are located as close as practical to the confinement boundary and active isolation valves are designed to take the position that provides greater safety upon loss of actuating power.

To mitigate the consequences of an uncontrolled release occurring within a hot cell, as well as the off-site consequences of releasing fission products through the ventilation system, the confinement barrier utilizes an active component in the form of bubble-tight isolation dampers (safety-related) on the inlet and outlet ventilation ports of each hot cell. This ESF effectively reduces the amount of ductwork in the confinement volume that needs to remain intact to achieve hot cell confinement. These dampers close automatically (fail-closed) upon loss of power or receipt of a confinement isolation signal generated by the RICS. Following an initiating event, the RICS isolates the hot cells. Refer to Section 7b for a description of the RICS.

Overall performance assurance of the active confinement components is achieved through factory testing and in-place testing. Duct and housing leak tests are performed in accordance with ASME N511, with minimum acceptance criteria as specified in ASME AG-1 (ASME, 2009). Specific owner's requirements with respect to acceptable leak rates are based on the safety analyses.

6b.2.1.3 Functional Requirements

Active confinement components are designed to fail into a safe state if conditions such as loss of signal, loss of power, or adverse environments are experienced.

Mechanical, instrumentation, and electrical systems and components required to perform their intended safety function in the event of a single failure are designed to include sufficient redundancy and independence such that a single failure of any active component does not result in a loss of the capability of the system to perform its safety functions.

Mechanical, instrumentation, and electrical systems and components are designed to ensure that a single failure, in conjunction with an initiating event, does not result in the loss of the system's ability to perform its intended safety function. The single failure considered is a random failure and any consequential failures in addition to the initiating event for which the system is required and any failures that are a direct or consequential result of the initiating event.

The design of safety-related systems (including protection systems) is consistent with IEEE Standard 379-2000 and Regulatory Guide 1.53 in the application of the single-failure criterion.

6b.2.1.4 Confinement Components

The following components are associated with the confinement barriers of the hot cells, tank vaults and pipe trenches, as previously described. Their specific materials, construction, and installation and operating requirements are evaluated based on the safety analysis.

Bubble-tight isolation dampers, designed, constructed and tested in accordance with ASME AG-1, Section DA "Dampers and Louvers" (ASME, 2009):

- Maintain their functional integrity.
- Maintain their rated leak-tightness following a seismic event.
- Maintain their structural integrity under fan shut-off pressure.
- Provide bubble-tight isolation upon receipt of a control signal or, in the event of loss of actuator power, by closure of actuator.
- Provide bubble-tight isolation when using manual actuator or when locked closed with power actuator removed.

- Relay damper full-open and full-closed position for control and indication by the use of limit switches.

Dampers are butterfly type, blade and frame fabricated of heavy-gage stainless steel. Total leakage based on bubble solution test as outlined in ASME AG-1 2009, Section DA-5141 (ASME, 2009).

Ventilation ductwork and ductwork support materials meet the requirements of ASME AG-1, Article SA-3000 “Materials”. Supports are designed and fabricated in accordance with the requirements of ASME AG-1, Section SA “Ductwork” (ASME, 2009).

Low leakage seals are provided on each penetration through the hot cells. For systems open to the hot cell atmosphere, redundant isolation valves are provided.

6b.2.1.5 Engineered Safety Feature Test Requirements

Engineered safety features are tested to ensure that ESF components maintain operability and can provide adequate confidence that the system performs satisfactorily in service during postulated events.

To the extent possible, the RICS and the confinement ESF whose operation it initiates are designed to permit testing during plant operation. Testing actuation devices and actuated equipment may be done individually or in groups to avoid negative impact to plant operations.

6b.2.1.6 Design Bases

For general discussion of the codes and standards used for the SHINE facility, see Chapter 3. For the design basis of the RVZ1 and RVZ2, see Subsection 9a2.1.1. For the design basis of the hot cells, see Section 4b.2. For the ESF-related design basis of the RICS, see Section 7b.4.1.

Potential variables, conditions, or other items that will be probable subjects of a technical specification associated with the RPF confinement systems and components are provided in Chapter 14.

6b.2.2 CONTAINMENT

The SHINE RPF does not employ a containment feature. Due to the low temperature and power level of facility operations, the safety analysis demonstrates that confinement features are adequate to mitigate potential accidents. Subsection 6b.2.1 is provided to describe the use of confinements as an ESF for the RPF.

6b.2.3 EMERGENCY COOLING SYSTEM

There is no emergency cooling system associated with the RPF side of the SHINE facility; therefore, this subsection is not applicable.

6b.2.4 RADIOISOTOPE PRODUCTION FACILITY ENGINEERED SAFETY FEATURES TECHNICAL SPECIFICATIONS

Potential variables, conditions, or other terms that are probable subjects of a technical specification associated with the RPF ESFs are provided in Chapter 14.

Table 6b.2-1 Radioisotope Production Facility Confinement Safety Functions

System, Structure, Component	Description	Classification
RVZ1 hot cell isolation dampers, ductwork up to filters and filters	Provide confinement isolation at hot cell boundaries	SR
RVZ2 isolation dampers, ductwork up to filters and filters	Provide confinement isolation at RCA boundary	SR
RICS	Provides confinement isolation signal	SR
Isolation valves on piping systems	Provide confinement at hot cell boundaries	SR
Hot cells, tank vaults, and pipe trenches	Provides confinement	SR

6b.3 NUCLEAR CRITICALITY CONTROL

The nuclear criticality safety program (NCSP) for the radioisotope production facility (RPF) is in accordance with applicable American National Standards Institute/American Nuclear Society (ANSI/ANS) standards, as endorsed by Regulatory Guide 3.71. Except for irradiation operations in the TSV, the facility is designed to meet the requirements of the following standards and NRC Regulatory Guide:

- ANSI/ANS-8.1-1998 (R2007), “Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors” (ANSI/ANS, 2007a) and ANSI/ANS-8.19-2005, “Administrative Practices for Nuclear Criticality Safety” (ANSI/ANS, 2005b) (see Regulatory Guide 3.71), for organization and administration.
- ANSI/ANS-8.1-1998 (R2007), “Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors” (ANSI/ANS, 2007a) Appendix A will be used to identify nuclear criticality safety (NCS).
- ANSI/ANS-8.3-1997 (R2012), “Criticality Accident Alarm System” (ANSI/ANS, 2012a), as modified by Regulatory Guide 3.71, “Nuclear Criticality Safety Standards for Fuels and Materials Facilities.”
- ANSI/ANS-8.7-1998 (R2007), “Guide for Nuclear Criticality Safety in the Storage of Fissile Materials” (ANSI/ANS, 2007b).
- ANSI/ANS-8.10-1983 (R2005), “Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement” (ANSI/ANS, 2005a).
- ANSI/ANS-8.20-1991 (R2005), “Nuclear Criticality Safety Training” (ANSI/ANS, 2005c).
- ANSI/ANS-8.21-1995 (R2011), “Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors” (ANSI/ANS, 2011).
- ANSI/ANS-8.23-2007 (R2012), “Nuclear Criticality Accident Emergency Planning and Response” (ANSI/ANS, 2012b).
- ANSI/ANS-8.24-2007 (R2012), “Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations” (ANSI/ANS, 2012c).
- ANSI/ANS-8.26-2007 (R2012), “Criticality Safety Engineer Training and Qualification Program” (ANSI/ANS, 2012d).

There is no deviation from any standards or requirements that would require the development of equivalent requirements or standards in the medical isotope production area. These criteria do not apply to the operation of the target solution vessels.

Elements comprising the NCSP are presented in Table 6b.3-2. Listed with each program element are the corresponding ANSI/ANS standards and regulations that will serve as the program basis.

The RPF will be designed, constructed, and operated such that a nuclear criticality event is prevented. The NCSP will implement the guidance provided in ANSI/ANS-8.19-2005 (ANSI/ANS, 2005b). Nuclear criticality safety at the facility will be ensured by designing the facility and SSCs with safety margins such that safe conditions will be maintained under normal and abnormal operations and design basis accidents. The facility is designed with a preference for NCS controls in the following order:

a. Passive engineered.
b. Active engineered.
c. Enhanced administrative.

d. Simple administrative.

An accidental criticality is highly unlikely because the SHINE facility is designed with passive engineered design features, including the use of neutron absorbers.

The analysis, consequences, and safety controls of such an event have been described in Subsection 13b.2.5. Administrative controls ensure the reliability and availability of the safety controls are adequate to maintain subcriticality.

Heterogeneous effects are not considered applicable because the uranium enrichment is less than 20 percent.

Management of the Nuclear Criticality Safety Program

The NCS criteria are used for managing criticality safety and include adherence to the double contingency principle, as stated in ANSI/ANS-8.1-1998 (R2007) (ANSI/ANS, 2007a). The double contingency principle states “process design should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.” Each process that has accident scenarios that could result in an inadvertent nuclear criticality at the SHINE facility meets the double contingency principle. Process design incorporates sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

The NCS program establishes criteria for the administration of those operations in which there exists a potential for nuclear criticality accidents. Responsibilities of management, supervision, and the nuclear criticality safety staff are defined below. Objectives and characteristics of operating and emergency procedures are also defined. The emergency procedures will include reporting criteria and report content requirements. Reports will be issued based on whether the criticality controls credited were lost (i.e., they were unreliable or unavailable to perform their intended safety functions), irrespective of whether the safety limits of the associated parameters were actually exceeded. Training in criticality safety will be provided to individuals who handle nuclear material at the facility. The training is based upon the training program described in ANSI/ANS-8.20-1991 (R2005) (ANSI/ANS, 2005c) and ANSI/ANS-8.26 (R2012) (ANSI/ANS, 2012d).

Other aspects of the NCS program include:

- Providing distinctive NCS postings in areas, operations, work stations, and storage locations relying on administrative controls for NCS.
- Requiring personnel to perform activities in accordance with written, approved procedures when the activity may affect NCS. Unless a specific procedure deals with the situation, personnel shall take no action until the NCS staff has evaluated the situation and provided recovery procedures.
- Requiring personnel to report defective NCS conditions to the NCS program management.

- Describing organizational positions, experience of personnel, qualifications of personnel, and functional responsibilities.

- Designating an NCS program director who will be responsible for implementing the NCS program.

- Promptly detecting any NCS deficiencies by means of operational inspections, audits, or investigations and reporting those deficiencies. Records of deficiencies will be retained and corrective actions will be documented.

- A program for evaluating the criticality significance of NCS events and an apparatus in place for making the required notification to the NRC Operations Center. Qualified individuals make the determination of significance for NCS events. The determination of loss or degradation of double-contingency protection is made against the license.

The training program is developed and implemented with input from the criticality safety staff, training staff, and management. The training focuses on the following:

- Knowledge of the physics of NCS.
- Analysis of jobs and tasks to determine the knowledge a worker must have to perform tasks efficiently.
- Design and development of learning objectives based upon the analysis of jobs and tasks that reflect the knowledge, skills, and abilities needed by the worker.
- Implementation of revised or temporary operating procedures.
- Training all personnel to recognize the CAAS signal and to evacuate promptly to a safe area.
- Providing instruction and training regarding the policy for NCS organization procedures.

Management responsibilities include:

- Overall responsibility for safety of operations.
- Demonstration of continuing interest in NCS.
- Formulation of NCS policy and communication of the policy to all employees.
- Involvement in operations with fissile material.
- Assignment of responsibility and delegation of authority to implement established policy.
- Responsibility for criticality safety that is assigned in a manner corresponding to that for other safety disciplines.
- Provision of a criticality safety staff appropriate for the scope of operations and experienced in the interpretation of criticality data.
- Provision of administrative oversight, such as an NCS organization or a Criticality Control Committee.
- Provision of administrative and organizational assurance that the SHINE NCS organization is independent of operations to an extent practical.
- Establishment of a review system to monitor the NCSP.
- Periodic participation in assessments or audits of the overall effectiveness of the NCSP.
- Provision of a committee that will assist management in achieving the objectives of the NCSP.

Supervisory responsibilities include:

- Safety of operations under their control.
- Demonstration of knowledge of aspects of NCS related to operations under their control.
- Assurance that personnel under their supervision have an understanding of procedures and safety considerations such that personnel may be expected to perform their functions without undue risk.
- Development or participation in the development and maintenance of written procedures that have an impact on NCS.
- Verification of compliance with NCS specifications prior to the use of new or modified equipment. Verification may make use of inspection reports or other features of the quality control system.
- Maintenance of conformance with good safety practices, including unambiguous identification of fissile materials and transient combustibles.

Nuclear criticality safety staff responsibilities include:

- Provision of technical guidance for the design of equipment and processes, and for the development of operating procedures.
- Maintenance of familiarity with current developments in NCS standards, guides and codes.
- Assistance to supervision, on request, for personnel training.
- Conduct audits of NCS practices and compliance with procedures, as directed.
- Examination of procedural violations, deficiencies, and mishaps for possible improvement of safety practices and reporting findings to management.
- Response to potential NCS non-compliance and defective NCS conditions.
- Reporting of potential NCS non-compliance and defective NCS conditions to management.

The NCS staff member must understand and have experience in the application of NCS programs. The NCS engineer is responsible for implementation of the NCSP. The education requirements of NCS staff members will be addressed in the FSAR.

<u>Process Descriptions</u>

Process descriptions of criticality safety for normal operations will be developed for the FSAR.

<u>Operating Procedures</u>

The purpose of operating procedures is to facilitate the efficient and safe conduct of operations. Procedures should be organized and presented for convenient use by operators. The operating procedures will include:

- Controls and limits significant to the NCS of the operation.
- Periodic revisions, as necessary.
- Procedures will be supplemented by NCS limits posted on equipment or incorporated in operating check lists or flow sheets.
- Periodic review of active procedures by supervision.

The operating procedures will meet the intent of ANSI/ANS-8.19-1996.

Planned Response to Criticality Accidents

The criticality accident and alarm system (CAAS) will be used as described in Section 7b.6. The CAAS provides for detection and annunciation of criticality accidents. Emergency procedures shall be prepared and approved by management. On-site and off-site organizations that are expected to respond to emergencies shall be informed of conditions that might be encountered. These procedures shall clearly designate evacuation routes. Evacuation should follow the quickest and most direct routes practical. These routes shall be clearly identified. Procedures shall include assessment of exposure to individuals. In addition, personnel assembly stations, outside the areas to be evacuated, shall be designated. Means to account for personnel shall be established. Arrangements shall be made in advance for the care and treatment of injured and exposed persons. The possibility of personnel contamination by radioactive materials shall be considered.

Personnel shall be trained in evacuation methods and informed of routes and assembly stations. Drills shall be performed at least annually. Instrumentation and procedures are provided for determining the radiation in the evacuated area following a criticality accident. Information is collected in a central location.

Emergency procedures for each area in which special nuclear material is handled, used, or stored will be maintained to ensure that all personnel withdraw to an area of safety upon the sounding of the alarm. The procedures include conducting drills to familiarize personnel with the evacuation plan, designation of responsible individuals for determining the cause of the alarm, and placement of radiation survey instruments in accessible locations for use in such an emergency. The current procedures for each area will be retained as a record for as long as licensed special nuclear material is handled, used, or stored in the area. The superseded portion of the procedures will be retained for three years after the portion is superseded.

Fixed and personnel accident dosimeters are provided in areas that require a CAAS. These dosimeters are readily available to personnel responding to an emergency, and there is a method for prompt on-site dosimeter readouts.

NCS Change Management

Changes that involve or could affect SNM will be evaluated under 10 CFR 50.59. Such changes include: new design, operation, or modification to existing SSCs; computer programs; processes; operating procedures; or administrative controls. Prior to implementing the change, it shall be determined that the entire process will be subcritical (with approved margin for safety) under both normal and credible accident scenarios. The above process for NCS change management is in accordance with ANSI/ANS-8.19-1996 (R2005) (ANSI/ANS, 2005b).

6b.3.1 CRITICALITY-SAFETY CONTROLS

The scope of this section is NCS controls for the RPF and it is applicable to SSCs of the facility in which uranium is present in sufficient mass for a nuclear criticality accident to be credible (in certain conditions). Criticality safety controls are listed in Table 13b.2.5-1. The uranium contained within the TSV and TSV dump tank is outside the scope of this section as the TSV and TSV dump tank are within the IF. The CAAS is not a control from perspective of criticality control; however, the CAAS is considered safety-related.

The major controlling parameters used in the facility are geometry control and/or limitations on the mass. Nuclear criticality safety evaluations (NCSEs) and analyses are used to identify the significant parameters affected within a particular system. Assumptions relating to process, equipment, material function, and operation, including credible accident scenarios, are justified, documented, and independently reviewed. Where possible, passive engineered controls are used to ensure NCS.

The use of geometry as a controlled parameter is acceptable because all dimensions and nuclear properties that use geometry control are verified.

Interaction and neutron absorption may also be used as controlling parameters. Interaction as a controlling parameter is acceptable if engineered controls are used to maintain a minimum separation between units. Enhanced administrative controls are used where engineered controls are not feasible.

The use of neutron absorption as a controlled parameter is acceptable by following ANSI/ANS-8.21-1995 (R2011) (ANSI/ANS, 2011), “Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors.” Neutron spectra are considered in the evaluation of absorber effectiveness.

Nuclear Criticality Safety Evaluations

Nuclear criticality safety evaluations will be performed to ensure that nuclear processes will remain subcritical during both normal and accident scenarios. An approved margin of subcriticality will be included in these evaluations. Nuclear criticality safety is analyzed for the design features of the plant system or component and for the operating practices that relate to maintaining NCS. The analysis of individual systems or components and their interaction with other systems or components containing enriched uranium is performed to ensure the NCS criteria are met.

Nuclear criticality safety analyses also meet the following:

- NCS analyses are performed using acceptable methodologies.
- Methods are validated and used only within demonstrated acceptable ranges.
- The analyses adhere to the methodology standards in ANSI/ANS-8.1-1998 (R2007) (ANSI/ANS, 2007a).
- The reference manual and documented reviewed validation report will be kept at the facility.
- The reference manual and validation report are incorporated into the Configuration Management Program.

- The NCS analyses are performed in accordance with the methods specified and incorporated in the configuration management program.
- Nuclear criticality safety controls and controlled parameters ensure that under normal and credible abnormal conditions, all nuclear processes are subcritical, including use of an approved margin of subcriticality for safety that is used.
- Process specifications incorporate margins to protect against uncertainties in process variables and against a limit being accidentally exceeded (ANSI/ANS, 2007a).
- Use of ANSI/ANS-8.7-1998 (R2007) (ANSI/ANS, 2007b), as it relates to the requirements for subcriticality of operations, the margin of subcriticality for safety, and the selection of controls.
- ANSI/ANS-8.10-1983 (R2005) (ANSI/ANS, 2005a), as modified by Regulatory Guide 3.71, as it relates to the determination of consequences of NCS accident sequences, is used.
- If administrative k_{eff} margins for normal and credible accident scenarios are used, NRC pre-approval of the administrative margins will be sought.
- Subcritical limits for k_{eff} calculations such that the margin is large compared to the uncertainty in the calculated values, and includes adequate allowance for uncertainty in the methodology, data, and bias to ensure subcriticality are used.
- Studies to correlate the change in a value of a controlled parameter and k_{eff} value are performed. The studies include changing the value of one controlled parameter and determining its effect on another controlled parameter and k_{eff}.
- The calculation of k_{eff} is based on a set of variables within the method's validated area of applicability.
- Trends in the bias support the extension of the methodology to areas outside the area or areas of applicability.

The NSCE procedure addresses requirements for:

- Normal case operating conditions.
- Nuclear criticality hazard identification.
- Hazard identification method.
- Hazard identification results.
- Nuclear criticality hazard evaluation.
- Nuclear criticality parameter discussions.
- Nuclear criticality safety controls (passive design features, active engineered features and administrative controls).
- Nuclear criticality safety peer review requirements.

Geometry Tanks

Each of the tanks within the scope of this section features criticality safety controls that meet the double-contingency principle, i.e., an inadvertent criticality is not credible unless two independent and unlikely events occur simultaneously. The first criticality safety control is that each tank, with the exception of the tanks associated with liquid waste processing, is criticality safe by geometry or by the combination of geometry and a layer of neutron absorbing material integral to the tank construction. The second, independent criticality-safety control is that the most reactive concentration of uranium in any tank results in $k_{eff} \leq 0.95$, based on MCNP analyses. MCNP is a validated code for calculating reactivity and dosimetry. Small amounts of fissile plutonium-239 (Pu-239), resulting from activation of uranium-238 (U-238) and subsequent decay of neptunium-239 (Np-239) during fuel solution irradiation, are accounted for in the MCNP calculations, even if the measured concentration of plutonium is negligible.

Piping

Pipe runs are single-parameter criticality-safe by geometry, with pipe diameters limited based on MCNP calculations, and with consideration of material and spacing relative to other uranium, reflecting, and moderating materials.

In the event of a leak in a double-walled pipe located in the pipe trenches, the target solution in the pipe trench would drain into a criticality-safe by geometry tank.

Tank Vaults

Constraints on the facility footprint prevent limiting potential maximum spills in tank vaults to single-parameter criticality safe depths. Therefore, each tank vault is connected, via a non-valved gravity drain into the criticality safe sump catch tank, which is criticality-safe by geometry.

Limits

Safe geometry limits for each of the following SSCs are established:

- Uranium and uranium oxide receipt and storage: storage container’s diameter and center-to-center spacing.
- Uranium dissolution: process vessels and piping diameter and center-to-center spacing.
- Target Solution Hold Tank (1-TSPS-03T): configuration (e.g., pencil or donut) and dimensions.
- TSV Dump Tank (1-SCAS-01T): configuration (e.g., pencil or donut) and dimensions.
- UREX (uranium extraction) recovery system: process vessels and piping diameter and center-to-center spacing.
- Denitration system: process vessels and piping diameter and center-to-center spacing.
- Criticality-safe sump catch tank (1-RDS-01T): configuration and dimensions.

Mass Limits

The mass of uranium present in an SSC is determined by analytical measurements and process calculations. Conservative administrative limits for each operation are specified in the operating procedures. Whenever mass control is established for a container, records are maintained for mass transfers into and out of the container. Establishment of mass limits for a container involves consideration of potential moderation, reflection, geometry, spacing, and enrichment. The evaluation considers normal operations and credible abnormal conditions for determination of the operating mass limit for the container and for the definition of subsequent controls necessary to prevent reaching the safety limits.

Mass limits, expressed as total mass of uranium or as uranium concentration, for each of the following SSCs will be established:

- Uranium and uranium oxide receipt and storage: total mass.

- Uranium dissolution: uranium concentration.
- TSV Feed Tank (1-TSPS-03T): uranium concentration.
- TSV Dump Tank (1-SCAS-01T): uranium concentration.
- Uranium extraction (UREX): uranium concentration.
- Thermal denitration system: uranium concentration.
- Waste processing systems: uranium concentration and total mass.

Criticality Accident Alarm System

The SHINE CAAS will be designed to meet the following:

- The facility CAAS is capable of detecting a criticality that produces an absorbed dose in soft tissue of 20 rads of combined neutron and gamma radiation at an unshielded distance of 2 meters from the reacting material within one minute. Two detectors cover each area needing CAAS coverage.
- The CAAS will use gamma and neutron sensitive radiation detectors that energize clearly audible alarm signals if accidental criticality occurs.
- The system complies with ANSI/ANS-8.3-1997 (R2012) (ANSI/ANS, 2012a), as modified by Regulatory Guide 3.71.
- The CAAS is appropriate for the facility for the type of radiation detected, the intervening shielding, and the magnitude of the minimum accident of concern.
- The CAAS is designed to remain operational during design basis accidents.
- The CAAS is clearly audible in areas that must be evacuated or there are alternative notification methods that are documented to be effective in notifying personnel that evacuation is necessary.
- Operations are rendered safe, by shutdown and quarantine, if necessary, in any area where CAAS coverage has been lost and not restored within a specified number of hours. The number of hours will be determined on a process-by-process basis, because shutting down certain processes, even to make them safe, may carry a larger risk than being without a CAAS for a short time. Compensatory measures (e.g., limiting access, halting SNM movement, or restoring CAAS coverage with an alternate instrument) when the CAAS system is not functional will be determined.
- Emergency power will be provided to the CAAS via the UPSS.

6b.3.2 TECHNICAL SPECIFICATIONS

Potential variables, conditions, or other items that are probable subjects of specification associated with the NCSP are provided in Chapter 14.

Table 6b.3-1 Tanks Subject to Criticality-Safety Controls

Tag #	Description	# Each	Criticality Safe	Type
1-TSPS-01T	Uranyl Sulfate Preparation Tank	2	Y	Annular
1-TSPS-02T	Uranium Metal Dissolution Tank	1	Y	Pencil
1-TSPS-03T	Target Solution Hold Tank	8	Y	Annular
1-SCAS-01T	TSV Dump Tank	8	Y	Ring
1-MEPS-01T	Mo Extraction Column	3	Y	Pencil
1-UNCS-01T	Uranyl Nitrate Conversion Tank	2	Y	Annular
1-UNCS-02T	UREX Feed Tank	2	Y	Harp
1-UNCS-03T	Technetium Removal Column	1	Y	Cylinder
1-UNCS-04T	Solvent Hold Tank	1	Y	Annular
1-UNCS-05T	Raffinate Hold Tank	2	Y	Harp
1-UNCS-06T	Recycle UN (Uranyl Nitrate) Hold Tank	1	Y	Harp
1-UNCS-07T	UN Evaporator Vessel	1	Y	Process
1-UNCS-08A	Thermal Denitrator	1	Y	Tank type will be provided in FSAR
1-UNCS-09T	Recycle Target Solution Tank	3	Y	Annular
1-RDS-01T	Criticality Safe Sump Catch Tank	1	Y	Harp

Table 6b.3-2: Nuclear Criticality Safety Program Elements and Requirement Bases

Program Element Name	Requirement Basis for Element
Criticality Safety Policy	Not applicable
Verification & Validation of NCS Codes	ANSI/ANS 8.1-1998
Nuclear Criticality Safety Evaluations	ANSI/ANS 8.1-1998 ANSI/ANS 8.19-1996
Nuclear Criticality Training and Qualifications	ANSI/ANS 8.20-1991 R1999 10 CFR 19.12
Implementation of Criticality Safety Controls and Limits	ANSI/ANS 8.19-1996
Configuration Control / Change Control	10 CFR 50.59
Audits and Inspections	ANSI/ANS 8.19-1996
Criticality Safety Non-Compliances: Investigating, Reporting, Tracking, and Trending	ANSI/ANS 8.19-1996 10 CFR 19.12
Criticality Safety Guidelines for Fire Fighting	Not applicable
Emergency Preparedness Plan and Response Procedures Manual	ANSI/ANS 8.23-1997 ANSI/ANS 8.19
Criticality Detector and Alarms System	ANSI/ANS 8.3-1997 10 CFR 19.12 10 CFR 70.24
Testing and Calibration of Active-Engineered Controls	Not applicable
Criticality Safety Controls Program	Not applicable

6b.4 REFERENCES

ANSI/ANS, 2005a. Criteria for Nuclear Criticality-Safety Controls in Operations With Shielding and Confinement, ANSI/ANS-8.10-1983 (R2005), ANSI/ANS, 2005.

ANSI/ANS, 2005c. Nuclear Criticality-Safety Training, ANSI/ANS-8.20-1991 (R2005), ANSI/ANS, 2005.

ANSI/ANS, 2007a. Nuclear Criticality-Safety in Operations with Fissionable Materials Outside Reactors, ANSI/ANS-8.1-1998 (R2007), ANSI/ANS, 2007.

ANSI/ANS, 2007b. Guide for Nuclear Criticality-Safety in the Storage of Fissile Materials, ANSI/ANS-8.7-1998 (R2007), ANSI/ANS, 2007.

ANSI/ANS, 2005b. Administrative Practices for Nuclear Criticality-Safety, ANSI/ANS-8.19-2005, ANSI/ANS, 2005.

ANSI/ANS, 2011. Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors, ANSI/ANS-8.21-1995 (R2011), 2011.

ANSI/ANS, 2012a. Criticality Accident Alarm System, ANSI/ANS-8.3-1997 (R2012), 2012.

ANSI/ANS, 2012b. Nuclear Criticality Accident Emergency Planning and Response, ANSI/ANS-8.23-2007 (R2012), 2012.

ANSI/ANS, 2012c. Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations, ANSI/ANS-8.24-2007 (R2012), 2012.

ANSI/ANS, 2012d. Criticality Safety Engineer Training and Qualification Program, ANSI/ANS-8.26-2007 (R2012), 2012.

ASME, 2007. Testing of Nuclear Air Treatment Systems, ASME N510, American Society of Mechanical Engineers, 2007.

ASME, 2009. Code on Nuclear Air and Gas Treatment, ASME AG-1, American Society of Mechanical Engineers, 2009.

IEEE, 2000. Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems, IEEE 379, Institute of Electrical and Electronics Engineers, 2000.

IMC, 2012. International Mechanical Code, IMC, 2012.