ENCLOSURE 2

LICENSE APPLICATION CHAPTERS 1 – 13

NON-PROPRIETARY

TRISO

TRISO-X Fuel Fabrication Facility
Special Nuclear Material
License Application

Revision	:	**1**
Status	:	**Approved**
Release Date	:	**05-APR-2022**
Project	:	**Fuel**
NRC Docket No.	:	**70-7027**

SPECIAL NUCLEAR MATERIAL LICENSE

CHAPTER INDEX

ABBREVIATIONS AND ACRONYMS

This list contains the abbreviations and acronyms used in this document.

Abbreviation or Acronym	Definition
ALARA	As Low As Reasonably Achievable
ALI	Annual Limit on Intake
AHJ	Authority Having Jurisdiction
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASCE	American Society of Civil Engineers
BDC	Baseline Design Criteria
BS/BA	Bachelor of Science / Bachelor of Arts
CAA	Controlled Access Area
CAAS	Criticality Accident Alarm System
CEDE	Cumulative Effective Dose Equivalent
CFR	Code of Federal Regulations
CM	Configuration Management
DAC	Derived Air Concentration
DFP	Decommissioning Funding Plan
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
EPA	U.S. Environmental Protection Agency
ETSZ	East Tennessee Seismic Zone
FFF	Fuel Fabrication Facility
FHA	Fire Hazards Analyses
FNMCP	Fundamental Nuclear Material Control Plan
HALEU	High Assay Low Enriched Uranium
HPGe	High Purity Germanium
IAEA	International Atomic Energy Agency
IBC	International Building Code
ICPMS	Inductively Coupled Plasma Mass Spectrometry
ICRP	International Commission on Radiation Protection Publication
ISA	Integrated Safety Analysis
IROFS	Items Relied On For Safety
KPA	Kinetic Phosphorescence Analyzer
LA	License Application
LEU	Low Enriched Uranium
MBA	Material Balance Area

Release Date: 05-APR-2022

Abbreviation or Acronym	Definition
MC&A	Material Control and Accountability
MMI	Modified Mercalli Intensity
MOU	Memorandum of Understanding
NCRP	National Commission on Radiation Protection
NCS	Nuclear Criticality Safety
NFPA	National Fire Protection Association
NIST	National Institute of Standards and Technology
NMSS	Nuclear Materials Safety and Safeguards
NRC	U.S. Nuclear Regulatory Commission
OCA	Owner Controlled Area
OJT	On-the-Job Training
OSHA	Occupational Safety and Health Administration
PHA	Process Hazard Analyses
PM	Preventive maintenance
PSP	Physical Security Plan
QA	Quality Assurance
RCA	Radiologically Controlled Area
REM	Roentgen Equivalent Man
RPP	Radiation Protection Program
RSO	Radiation Safety Officer
RWP	Radiation Work Permits
SEP	Site Emergency Plan
SME	Subject Matter Expert
SNM	Special Nuclear Material
SRC	Safety Review Committee
TEDE	Total Effective Dose Equivalent
TRISO-X FFF	TRISO-X Fuel Fabrication Facility
U	Uranium
U-235	Uranium-235
U-238	Uranium-238
UL	Underwriters Laboratory
USGS	United States Geological Survey

NRC SPECIAL NUCLEAR MATERIAL LICENSE
SNM-XXXX
CHAPTER 1

GENERAL INFORMATION
Table of Contents

GENERAL INFORMATION

1.1 Facility and Process Information

The primary purpose of the TRISO-X Fuel Fabrication Facility (FFF) in Oak Ridge, Tennessee, is to manufacture coated particle fuel for the next generation of commercial nuclear reactors. The modular design of the process cells / areas anticipates additional manufacturing capabilities to satisfy the needs of a variety of fuel designs and reactors (e.g., pebble bed high temperature gas-cooled, prismatic gas-cooled, molten salt-cooled, accident tolerant fuel, nuclear thermal propulsion, and others). Nuclear materials enriched to less than 20 weight percent U-235 are utilized in the product manufacturing operations authorized by this license.

1.1.1 Site Description and Location

The TRISO-X site is located in the Horizon Center Industrial Park on property abutting portions of Renovare Boulevard, within the western limits of the City of Oak Ridge and in the northeastern portion of Roane County, Tennessee. The site is situated in an area dedicated and zoned for industrial development, on an approximately 110-acre greenfield site. Of the total acreage, approximately 60 acres are designated for manufacturing and administrative buildings, equipment yards, access roads, parking, and stormwater management. The site is situated at approximately latitude N 35° 57′ 41″ and longitude W 84° 22′ 13″.

The site location in northeastern Roane County is in the Valley and Ridge physiographic province. The regional topography near the site is typical of the Valley and Ridge province which is characterized by northeast–southwest trending ridges and intervening valleys. The site and other developed areas along State Route 95 (TN 95 – Oak Ridge Turnpike) to the northeast and southwest are located on relatively flat or slightly undulating terrain associated with the East Fork Poplar Creek Valley, while just northwest of the site, there is a steep incline to the top of Black Oak Ridge. Several other ridges oriented northeast to southwest are present within the vicinity of the site. The Poplar Creek Valley is the next valley north and parallels Black Oak Ridge. East Fork Ridge is located to the south and east of the site and is interrupted by the valley of Bear Creek and TN 95. Pine Ridge is located south and east of East Fork Ridge.

1.1.1.1 Population, Nearby Land Uses, and Transportation

A site location map, including the population centers located near the site, is shown in Figure 1-1. The closest major population center is the City of Oak Ridge which had a population of 31,402 as of April 1, 2020, per the United States Census Bureau website. The closest residents to the site are located in a residential development off Poplar Creek Road, approximately 0.6 miles northwest of the site boundary, separated from the site by Black Oak Ridge and areas of dense vegetation. There are also residential neighborhoods located to the east off TN 95, approximately 1.3 miles or more from the site. The North Boundary Greenway, a low-density recreational trail used for hiking and biking, borders the site boundary to the northwest.

The immediate area surrounding the site consists of rural wooded area and light commercial and industrial use buildings. The immediately adjacent one-story warehouse/office building located near the southern corner of the property is the corporate office of Philotechnics Inc., a radiological service and mixed and radioactive waste brokerage provider licensed by the State of Tennessee, does not manage, utilize, or store chemicals (hazardous materials) in quantities that pose hazards to the TRISO-X site. Renovare Boulevard, which borders the site to the southeast, is a two-lane divided roadway that provides access to the site and to other parcels within the Horizon Center Industrial Park.

Lands adjacent to the industrial park are predominantly undeveloped and forested, consisting of large tracts of U.S Department of Energy Oak Ridge Reservation land which border the site to the northwest and surround the industrial park in other directions, with the exception of the TN 95 roadway corridor to the east. The existing land use within one mile of the site consists of primarily industrial development and woodlands. Within a five-mile radius of the site, approximately 83 percent of the land is undeveloped (e.g., forest, pasture, wetland) and the remainder is developed. Other land uses within 5 miles of the site include heavy industrial, light industrial/manufacturing, commercial/office space, agricultural, and residential. The Methodist Medical Center of Oak Ridge is the nearest hospital, located approximately 9 miles from the site. The closest school to the site is Linden Elementary School, located approximately 5 miles from the site.

Transportation infrastructure near the site includes Renovare Boulevard; TN 95; two interstate highways – Interstate 40 and Interstate 75 – several Tennessee state highways; and local roads. The McGhee Tyson Airport, which serves public and military needs, is located 26 miles from Oak Ridge by road.

1.1.1.2 Meteorology

Oak Ridge is located in the broad Tennessee River valley between the Cumberland Mountains, which lie to the northwest, and the Great Smoky Mountains, to the southeast. The Cumberland Mountains moderate the local climate by retarding the flow of cold air from the north during winter. Both mountain ranges are generally oriented in a northeast-southwest direction. The valley between them is corrugated by broken ridges approximately 300 to 500 feet high and oriented parallel to the main valley in an approximate northeast-southwest direction.

The climate of Oak Ridge is classified as humid subtropical. The "subtropical" designation indicates that the region experiences a wide range of seasonal temperatures. Such areas are typified by significant temperature differences between summer and winter. The normal liquid equivalent annual precipitation in the Oak Ridge area is 50.91 inches, and the average annual snowfall is 5.9 inches. The normal daily minimum temperature in January is 28.9 degrees Fahrenheit (°F) and the normal daily maximum temperature in July is 88.4°F.

Direct deflection of the winds by terrain is a dominant mechanism that drives the winds in the Tennessee River valley. This mechanism acts approximately 50 – 60 percent of the time, resulting in winds that blow in directions generally along the approximate northeast-southwest axis of the valley. The distribution of prevailing monthly wind directions is bimodal, with winds from the northeast (50 – 60 degrees), or from the southwest (210 – 220 degrees). The mean annual wind speed is 2.8 miles per hour.

Severe storm conditions are infrequent in the Oak Ridge area, due to the area being south of most blizzard conditions, and too far inland to be affected by hurricane activity. Tornadoes generally occur more frequently in the western and middle portions of Tennessee; however, Eastern Tennessee experiences tornado outbreaks of varying magnitudes approximately every three to six years. In a four-county area around the site for the period 1950 to 2020, the highest intensity tornadoes were rated F3 as a result of storms on February 21, 1993. Due to the low frequency of tornadoes in this region, no specific design criteria relative to tornadoes are required by the International Building Code. Lightning risk at the site has been addressed through lightning protection systems as specified in the Fire Hazards Analysis as described in Chapter 7.

1.1.1.3 Hydrology

The site is categorized as upland; no water bodies or wetlands were identified within the site. The nearest water body to the site is East Fork Poplar Creek, the closest portions of which run in a southwest direction through the industrial park between TN 95 and Renovare Boulevard at an approximate elevation of less than 770 feet. East Fork Poplar Creek empties into Poplar Creek approximately 1.25 miles southwest of the site, and Poplar Creek empties into the Clinch River approximately 3 miles southwest of the site boundary.

Federal Emergency Management Agency *Flood Insurance Rate Map Number 47145C0130F, Panel 0130F, Roane County, Tennessee, and Incorporated Areas*, Effective Date September 28, 2007, shows the site to be in an area of minimal flood hazard (Zone X). The nearest section of detailed study for East Fork Poplar Creek is approximately 1.5 miles northeast of the site, with a 100-year base flood elevation of 783 feet at the downstream end of mapping. The closest Clinch River location to the site has a 100-year base flood elevation of 747 feet at the Poplar Creek outlet.

The floor in the TRISO-X FFF is located at an elevation of approximately 811 feet, and the elevation of Renovare Boulevard near the primary entrance to the site is approximately 776 feet. Therefore, the facility floor elevation ranges from 28 to 64 feet above the mapped 100-year base flood elevations of the nearest water bodies as described above, and the floor elevation is more than 35 feet above Renovare Boulevard. Renovare Boulevard is situated at a higher elevation than the lower lying unmapped portions of East Fork Poplar Creek closest to the site.

Four groundwater observation wells were installed on the site in fall of 2021 with total depths and screened intervals based on observed first water identified in the upper most bedrock during drilling. No water was identified in the shallow unconsolidated surficial sediments above the

bedrock. Total well depths range from approximately 39 feet to 75 feet below ground surface. The underlying bedrock in which the observation wells are completed is primarily comprised of dolomite and is the first type of bedrock encountered at all sites.

Depth to groundwater measurements taken at the four observation wells vary from approximately 10 to 57 feet below the top of the well casing. Groundwater elevation measurements and modeling indicate that groundwater generally flows in a southwest direction toward East Fork Poplar Creek. There are no known household, public, or industrial users of groundwater downgradient of the site.

1.1.1.4 Geology

The TRISO-X site is located within the Valley and Ridge Province, a long, narrow belt trending northeast to southwest that is bordered on the west by the Appalachian Plateau and on the east by the Blue Ridge Province. The province is expansive and extends from Vermont to Alabama. This physiographic province consists of a series of northeast/-southwest- trending synclines and anticlines composed of Early Paleozoic sedimentary rocks. Drainage patterns in the Valley and Ridge Province generally follow the northeast-southwest trend of topography. However, segments of major rivers cut across the regional topographic alignment following deeply entrenched, ancient stream courses. These include the Powell, Clinch, Holston, and French Broad rivers that join to form the Tennessee River after flowing many miles in northeast/southwest-trending valleys.

The Rome Formation and the Conasauga, Knox, and Chickamauga Groups and associated formations comprise the majority of the underlying bedrock of the Valley and Ridge Province. The site is underlain by limestones-dolomites of the Knox Group and limestones with interbedded shale, argillaceous limestone, mudstone, and wackestone associated with the Chickamauga Group.

According to the United States Geological Survey (USGS), the region containing the site may contain carbonate rocks that can become karstified. These folded and faulted carbonate rocks are Paleozoic in age and are subject to dissolution that may produce a range of features that include solution, collapse, cover-collapse sinkholes and caves. Karst features previously reported on lands adjacent to the site have included springs and sinkholes of various sizes. Based on the topography of the site, several shallow draws and depressions exist which may reveal karst features beneath the surface. Karst features are caused by dissolution of carbonate rocks and deep weathering along prevailing fractures and strike-oriented bedding, creating conduits and voids (open and/or clay-filled). Voids within the dolomite and limestone bedrock were encountered on the site during the geotechnical drilling program to support facility design. Bedrock was encountered during drilling at a minimum depth of 3.6 feet and a maximum depth of 50.0 feet.

The East Tennessee Seismic Zone (ETSZ) is the second most active zone in the eastern United States in terms of small magnitude (M<5) seismicity, second in frequency to the New Madrid seismic zone. Activity in the ETSZ has remained high for several decades with only a few events having magnitudes as large as M 4.6. Generally, earthquakes in the ETSZ produce minor or no damage: the largest observed earthquakes have produced only minor damage to buildings, typically chimney collapse, cracks in plaster, and broken windows, consistent with intensity VI on the Modified Mercalli Intensity (MMI) scale.

1.1.2 Facility Buildings and Structures

A site plan showing the location and arrangement of buildings is included as Figure 1-2. Security fencing along or near the property boundaries defines the Owner Controlled Area. The site includes 4 buildings for nuclear manufacturing, administrative offices, raw material preparation, and security.

The building code of record for the buildings on the site is the 2018 Edition of the *International Building Code*. The type of construction is classified as non-combustible. All handling, processing, and storage of licensed material occurs in the nuclear manufacturing building, which is sized for two process lines, each including similar process steps as outlined in Section 1.1.3. The design of the structures and facilities complies with seismic loadings based on the 2018 Edition of the *International Building Code* and American Society of Civil Engineers (ASCE) 7-16, *Minimum Design Loads and Associated Criteria for Buildings and Other Structures*, as appropriate for the geographic location of the site.

1.1.3 General Process Description

TRISO-X FFF manufacturing operations consist of receiving high assay low enriched uranium (HALEU) in the form of uranium oxide powder enriched to less than 20 weight percent U-235; converting the oxide into a uranyl nitrate solution, into gel spheres, and then into fuel kernels; and processing the fuel kernels through coating, overcoating, fuel form pressing, and carbonization. Coated particles and/or final fuel forms are removed from the process at the appropriate point and loaded into licensed shipping containers for shipment to other licensed facilities. These operations are supported by shipping and receiving, laboratory, quality control, research and development, uranium and chemical recovery, and waste disposal processes. Detailed facility and process descriptions are provided in the *TRISO-X Fuel Fabrication Facility Integrated Safety Analysis Summary.*

1.1.4 Raw Materials, Products, By-Products and Wastes

1. The feed material for the TRISO-X FFF is uranium oxide powder. The manufacturing, recovery, support, and waste packaging activities are supported by a number of non-radiological chemical materials stored in bulk quantities, as listed in the NRC-required Emergency Plan and ISA Summary.

2. Finished products containing licensed material include coated particles and final fuel forms in various shapes and configurations.
3. There are no byproducts as defined by 10 Code of Federal Regulations (CFR) 30.4 extracted or converted after extraction from the TRISO-X FFF for use in a commercial, medical, or research activity.
4. Uranium is recovered from non-conforming product materials, process solutions, and scrap materials by processing it into a form that is suitable for use as feedstock in the manufacturing process.
5. Process solutions contaminated with uranium that cannot be recovered/recycled are identified as liquid wastes. Liquid wastes are collected and sampled to determine appropriate handling/treatment steps. Treatment typically involves adjustment of pH, filtering, ion exchange, and/or precipitation. Precipitates are de-watered, and the solids are packaged for off-site disposal. If needed, liquid wastes that have been handled/treated can be sampled and discharged through an inline monitor to shipping packages or conveyances for off-site disposal. Used oils may also be sampled and containerized for shipment to a licensed disposal facility.
6. Airborne effluents are discharged to the atmosphere via a number of process stacks. HEPA filtration and dry scrubber systems are used as needed to remove radioactive particulates and chemicals from airborne effluents to assure compliance with 10 CFR 20 and applicable State of Tennessee regulations prior to discharge to the atmosphere.
7. Sanitary wastes are discharged through piping which goes to the City of Oak Ridge publicly owned treatment works. The inputs for the sanitary sewer system from the site include bathrooms and showers.
8. Solid waste materials include, but are not limited to, damaged and/or obsolete equipment, used ventilation filters and personal protective equipment, processing and waste treatment residues, and miscellaneous combustible wastes. Materials could be radiologically contaminated, non-contaminated, hazardous, or mixed (hazardous and radioactive). Solid waste materials are processed, recycled, and/or containerized for shipment to a licensed disposal facility.

1.2 Institutional Information

1.2.1 Corporate Identity

This application is filed by TRISO-X, LLC, a Delaware limited liability company, headquartered at 801 Thompson Avenue, Rockville, Maryland. TRISO-X, LLC is a wholly-owned subsidiary of X Energy, LLC, a Maryland limited liability company. TRISO-X, LLC is a privately held company and is not owned or controlled by a foreign corporation or government. The principal place of business and location of the licensed facility is as follows:

TRISO-X Fuel Fabrication Facility
(specific street address to be assigned by U.S. Post Office/City of Oak Ridge)
Oak Ridge, Tennessee 37830

1.2.2 U.S. Nuclear Regulatory Commission License Information

1. Docket Number: 70-7027
2. License Number: TBD
3. Period of License: 40 years

1.2.3 Financial Qualifications

A summary of financial qualifications that demonstrates the financial capability of TRISO-X, LLC to construct and operate the TRISO-X FFF has been submitted separately for NRC review. The financial arrangements to assure that decommissioning funds will be available are set forth in Chapter 10.

1.2.4 Type, Quantity, and Form of Licensed Material

The following types, maximum quantities, and forms of special nuclear materials (SNM) are authorized under 10 CFR 70, 30 and 40.

1. [redacted]**SRI** kilograms of U-235 contained in uranium enriched to less than 20%, in any chemical/physical form. Contaminants may include 10^{-7} grams of transuranic materials per gram of uranium, and 600 Becquerels of fission products per gram of uranium.
2. 350 grams of U-235 in any chemical/physical form and at any enrichment for use in measurement and detection instruments, check sources, and instrument response standards.
3. 350 grams of U-235 in any chemical/physical form and at any enrichment for use in research and development studies.
4. 25 millicuries of plutonium as counting and calibration standards and/or for use in research and development studies.
5. 300 millicuries of Cs-137 as sealed radioactive sources for use in measurement and detection instruments, check sources, instrument response standards, and counting and calibration standards.
6. 2 millicuries of any licensed material between atomic numbers 3 and 83 as sealed and unsealed radioactive sources for use in measurement and detection instruments, check sources, instrument response standards, and counting and calibration standards.
7. 1 microcurie of any licensed material between atomic numbers 84 and 95 as sealed and unsealed radioactive sources for use in measurement and detection instruments, check sources, instrument response standards, and counting and calibration standards.

1.2.5 Authorized Uses and Activities

This license authorizes the use of SNM for operations involving enriched uranium pursuant to 10 CFR Part 70 as listed in this section. This also includes the support activities related to the manufacture of SNM-containing products.

1. Manufacturing Operations
    a. Fuel Manufacturing – Conversion of uranium oxides to uranyl nitrate solutions, and fabrication of coated particles and final fuel forms containing uranium.
    b. Uranium Recovery – recycling/recovery of SNM from process scrap materials.
2. Laboratory Operations
    a. Chemical, instrumental, and physical analyses and testing on material consisting of and/or containing SNM.
    b. Preparation of any required samples or standards.
3. Research and Development Operations – Process, product, and other research and development activities using natural, source, and SNM compounds and mixtures in benchtop, laboratory-scale, and/or full-scale prototype equipment environments related to:
    a. Enriched uranium fuel designs.
    b. Manufacturing and processing technology and equipment.
    c. Recycling/recovery.
4. Waste Operations
    a. Volume reduction, treatment, packaging, and storage of liquid and solid wastes contaminated with or containing non-recoverable uranium.
    b. Treatment, packaging, and storage of hazardous or mixed waste.
    c. Shipment of wastes to licensed facilities for disposal.
5. Support Operations
    a. Receipt, handling, and storage of raw materials.
    b. Storage of licensed material compounds and mixtures in areas with containers arranged specifically for maintenance of radiological and nuclear safety.
    c. Storage of finished fuel products and the preparation of these products for transportation off-site.
    d. Decontamination of equipment and materials.
    e. Maintenance, repair, calibration, and/or testing of SNM processing equipment, instruments, auxiliary systems, contaminated equipment, and facilities.

1.2.6 Site Safeguards

Physical security at the TRISO-X FFF is described in the NRC-approved *TRISO-X Fuel Fabrication Facility Physical Security Plan*, and nuclear material control and accountability (MC&A) is described in the NRC-approved *TRISO-X Fuel Fabrication Facility Fundamental Nuclear Material Control Plan*. Both plans are maintained current in accordance with applicable regulations as outlined in Chapters 12 and 13. These plans detail the measures employed at the facility to detect

potential loss of, and mitigate the opportunity for theft of, SNM of Moderate Strategic Significance, in accordance with the applicable requirements of 10 CFR 73 and 10 CFR 74. Safeguards Information is controlled as described in the *TRISO-X Facility Safeguards Information Plan*.

1.2.7 Terminology / Definitions

Definitions for terms specific to a particular safety function may be given in the corresponding chapter on that function. The following definitions apply to terms used in this license:

Term	Definition
U-235 Enrichments	"Low enriched uranium", which is also known as "high assay low enriched uranium (HALEU)," is defined as any compound of uranium in which the enrichment in the isotope of uranium-235 is less than 20 percent by weight.
Nuclear Safety	Nuclear criticality safety
Will, Shall	A requirement.
Should	A recommendation.
May	Permission (optional), neither a requirement nor a recommendation.
Are	An existing practice for which there is a requirement to continue.
Frequencies	When audit, measurement, surveillance, and/or other frequencies are specified in license documents and approved procedures, the following time spans apply: ▪ Monthly – an interval not to exceed 40 days ▪ Quarterly – an interval not to exceed 4 months ▪ Semi-Annually – an interval not to exceed 7 months ▪ Annually – an interval not to exceed 15 months ▪ Biennially – an interval not to exceed 30 months ▪ Triennially – an interval not to exceed 45 months ▪ For time spans not covered above, an extension of 0.25 times the interval will apply.
Criticality Control or Criticality Safety Control	The administrative and technical requirements established to minimize the probability of achieving inadvertent criticality in the environment analyzed.

Term	Definition
Work Area Air Samplers	Stationary air samplers demonstrated to be representative of workers breathing air. If stationary air samplers have not been demonstrated to be representative, the results of lapel air samplers will constitute work area air samplers.
Equivalent Experience	For the purpose of meeting educational requirements described throughout the license, two (2) years experience is considered to be equivalent to one (1) year of post-secondary education. For example, eight (8) years of applicable experience will satisfy the requirement for a B.S. degree (4 years of post-secondary education).
Owner Controlled Area	A site area bounded by a fence designed to provide physical security, and which encompasses the Controlled Access Area. The area contains radioactive material processing, storage, and laboratory areas, as well as support functions.
Restricted Area	A site area in which individuals may be exposed to radiation or radioactive material at levels or concentrations in excess of that allowed for the general public (see definition in 10 CFR 20.1003). This could include any location on the site where the TRISO-X FFF is located, depending upon activities conducted and the exposure potential as evaluated by the safety function.
Radiologically Controlled Area	A site area where uncontained radioactive material is present, such that contamination levels are likely to be encountered in excess of acceptable levels for unrestricted use. This type of area, designated for contamination control purposes, requires various levels of protective clothing and other personnel protective actions. It could include any location within the Restricted Area, either on a permanent or temporary basis. This term is analogous to the 10 CFR 20.1003 defined term "controlled area...an area, outside of a restricted area, but inside the site boundary, access to which can be limited by the licensee for any reason."
Uncontrolled Area	A site area where radioactive materials may be handled in the form of sealed sources, in packages or closed containers, in small amounts (air samples, bioassay samples, etc.), or not at all. This type of area is designated for contamination control purposes and is not likely to have contamination at levels in excess of those acceptable for unrestricted use.

Term	Definition
Conditions Adverse to Safety	As used in Sections 2.2, 2.5.1, and 11.6, events that could have the potential to impact the safety of licensed activities, including equipment failures, malfunctions, or deficiencies; procedure problems, errors, or omissions; improper installations; non-conformances with regulatory requirements or commitments; quality-related issues; or a significant condition, such that if uncorrected, could have a serious effect on safety.

1.3 Special Exemptions and Special Authorizations

1.3.1 Changes to the License Application

Changes may be made to the License Application and/or to supporting documents referenced in the license without prior NRC approval provided that the following conditions are met:

1. The change does not decrease the level of effectiveness of the design basis as described in the License Application.
2. The change does not result in a departure from the methods of evaluation described in the License Application used in establishing the design basis.
3. The change does not result in a degradation of safety.
4. The change does not affect compliance with applicable regulatory requirements.
5. The change does not conflict with an existing license condition.
6. Within 30 days after the end of the calendar year in which the change is implemented, the licensee shall submit the revised chapters of the License Application to the Director, NMSS, using an appropriate method listed in 10 CFR 70.5(a), and a copy to the appropriate NRC Regional Office.

This authorization is consistent with the process for making changes under 10 CFR 70.72, *Facility Changes and Change Process*, and is further supported by Section C5, *Other Changes*, in NRC Regulatory Guide 3.74, *Guidance for Fuel Cycle Facility Change Processes*, January 2012.

1.3.2 Criticality Monitoring

10 CFR 70.24 requires a licensee authorized to possess SNM in stated amounts to maintain in each area in which such licensed SNM is handled, used or stored to employ a CAAS meeting the stated requirements.

Notwithstanding the requirements of 10 CFR 70.24, the licensee is granted an exemption from criticality monitoring requirements for SNM stored in authorized shipping containers complying with the requirements of the Code of Federal Regulations, Title 10, Part 71, and which are in

isolated arrays or on a transport vehicle and which are no more reactive than that approved for transport.

The requirements in 10 CFR 71.55, *General Requirements for Fissile Material Packages*, and 10 CFR 71.59, *Standards for Arrays of Fissile Material Packages*, ensure that arrays will remain subcritical under normal conditions and under accident conditions. The exemption does not affect the level of protection for either the health and safety of workers and the public or for the environment; nor does it endanger life or property or the common defense and security.

1.3.3 Posting and Labeling

10 CFR 20.1904(a) requires a licensee to ensure that each container of licensed material bears a durable, clearly visible label bearing the radiation symbol and the words: "CAUTION, RADIOACTIVE MATERIAL" or "DANGER, RADIOACTIVE MATERIAL". The label must also provide sufficient information (such as the radionuclide(s) present, an estimate of the quantity of radioactivity, the date for which the activity is estimated, radiation levels, kinds of materials, and mass enrichment) to permit individuals handling or using the containers, or working in the vicinity of the containers, to take precautions to avoid or minimize exposure.

Notwithstanding the requirements of 10 CFR 20.1904(a), the licensee is granted an exemption from affixing a label to each container of licensed material when entrances into each building in which radioactive materials are stored, used, or handled are posted with a sign stating "EVERY CONTAINER OR VESSEL WITHIN THIS AREA MAY CONTAIN RADIOACTIVE MATERIALS".

The exemption is authorized by law because there is no statutory prohibition on the proposed posting of a single sign indicating that every container in the posted area has the potential for internal contamination. To reduce unnecessary regulatory burden, the NRC issued a final rule in 2007 that, in part, modified 10 CFR 20.1905, *Exemptions to Labeling Requirements*, thereby exempting certain containers holding licensed material from the labeling requirements of 10 CFR 20.1904 if certain conditions are met. Although the 2007 rulemaking only applied to facilities licensed under 10 CFR 50 and 10 CFR 52, *Licenses, Certifications, and Approvals for Nuclear Power Plants*, the rationale underlying the rule supports the exemption request. Exempting TRISO-X from this requirement reduces licensee administrative and information collection burdens but serve the same health and safety functions as the current labeling requirements. Therefore, the exemption does not affect the level of protection for either the health and safety of workers and the public or for the environment; nor does it endanger life or property or the common defense and security.

1.3.4 ICRP-68 DAC and ALI Values

Derived air concentration (DAC) and the annual limit on intake (ALI) values based on the dose coefficients published in the International Commission on Radiation Protection Publication 68 (ICRP-68) may be used in lieu of the DAC and ALI values in Appendix B of 10 CFR 20 in accordance with approved procedures. See Chapter 4 for additional details.

The ICRP-68 guidance was promulgated after the 10 CFR 20, Appendix B criteria were established, and provides an updated and revised internal dosimetry model. Use of the ICRP-68 models provide more accurate dose estimates than the models used in 10 CFR 20 and allows TRISO-X to implement an appropriate level of internal exposure protection. In a Staff Requirements Memorandum dated April 21, 1999 (SECY-99-077), the Commission approved the staff granting exemptions based on the precedent set by the decision to authorize the use of models in ICRP Publication 68.

This exemption is in accordance with the As Low As is Reasonably Achievable (ALARA) principle, international standards on radiation protection, and does not conflict with established NRC dose limits. No new accident precursors are created by this exemption to allow modification to the values used to assess internal dose. There is no significant increase in the risk to workers or members of the public as a result of this exemption. The activities that are authorized by this exemption are in compliance with law and will not endanger life or property or the common defense and security.

1.3.5 ICRP-60 Organ Dose Weighting Factors

Tissue weighting factors listed in the International Commission on Radiation Protection Publication 60 (ICRP-60) may be used in lieu of the organ dose weighting factors in 10 CFR 20.1003 for effective dose assessments listed in ICRP-68 methodologies, in accordance with approved procedures.

The ICRP-60 guidance was promulgated in the same year that 10 CFR 20 organ dose weighting factors were established. Use of the ICRP-60 models provide more accurate dose estimates than the models used in 10 CFR 20 and allows TRISO-X to implement an appropriate level of internal exposure protection. In a Staff Requirements Memorandum dated April 21, 1999 (SECY-99-077), the Commission approved the staff granting exemptions based on the precedent set by the decision to authorize the use of models in ICRP Publication 68.

The underlying purpose of 10 CFR Part 20 is to ensure that occupational workers and members of the public are protected from radiation; that their doses, as a result of licensed activities, are within prescribed limits; and that their doses are ALARA.

This exemption is in accordance with the ALARA principle, international standards on radiation protection, and does not conflict with established NRC dose limits. No new accident precursors

are created by this exemption to allow modification to the values used to assess internal dose. There is no significant increase in the risk to workers or members of the public as a result of this exemption. The activities that are authorized by this exemption are in compliance with law and will not endanger life or property or the common defense and security.

1.3.6 Certain Unplanned Contamination Events

Notwithstanding the requirements of 10 CFR 70.50(b)(1), the licensee is granted an exemption from the requirement to report unplanned contamination events when the following conditions are met:

1. The event occurs in a restricted area in a building which is maintained inaccessible to the public by multiple access controls.
2. The area was controlled for contamination before the event occurred, the release of radioactive material is under control, and no contamination has spread outside the area.
3. Radiation safety personnel trained in contamination control are readily available.
4. Equipment and facilities that may be needed for contamination control are readily available.
5. The otherwise reportable unplanned contamination event is documented in the licensee's Corrective Action Program.

Chapter 4 describes the radiation protection program measures that keep worker exposures ALARA through: (a) approved radiation protection procedures and radiation work permits; (b) the use of ventilation systems, containment systems, and respirators to control exposure to airborne radioactive material; (c) the use of protective clothing to prevent the spread of surface contamination; (d) the use of surveys and monitoring programs to document contamination levels and exposures to workers; and (e) identification of items relied on for safety and management measures to maintain those items available and reliable.

In addition, (f) access to the site is restricted to individuals that have completed site-specific nuclear safety training requirements or individuals that are formally escorted; (g) during normal operations, trained and qualified radiation protection staffing is provided and readily available to support and respond to radiological conditions, and the staff is trained in contamination control procedures and techniques required for responding to a contamination event when needed; (h) appropriate radiation surveys are performed by qualified personnel during or after an unplanned contamination event as necessary to assess radiological conditions and provide the appropriate response, survey results are compared to specified action guides, appropriate actions are taken when contamination levels in excess of action levels are found and the affected area is decontaminated in a safe and timely manner, and survey records for contamination events are documented pursuant to 10 CFR 20.2103 and are available for review.

Based on the limited scope of the exemption, and the access and contamination controls, training, radiation surveys and other ALARA measures described in the application, granting the exemption as stated above does not endanger life or property. The exemption does not alter

reporting requirements for unplanned contamination events through other NRC requirements such as 10 CFR 20.2202, *Notification of incidents*, and 10 CFR 20.2203, *Reports of exposures, radiation levels and concentrations of radioactive material exceeding the constraints or limits*. In addition, the exemption does not involve information or activities that could impact the common defense and security.

Granting this exemption request is otherwise in the public interest because it promotes regulatory efficiency. The exemption relieves the licensee from a reporting requirement for unplanned contamination events that do not present a risk to public health and safety given the site-specific conditions and programs described above. Specifically, the exemption relieves the licensee from generating reports of contamination events in controlled areas where the release of radioactive material is under control and no contamination has spread outside the controlled area. Granting the exemption allows the licensee to focus the resources required to fulfill the reporting requirement on other activities. In addition, it relieves the NRC staff from receiving and processing reports which do not present a risk to public health and safety.

Therefore, the exemption does not affect the level of protection for either the health and safety of workers and the public or for the environment; nor does it endanger life or property or the common defense and security.

1.3.7 Release for Unrestricted Use

Limits developed by the NRC as specified in *Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material*, U.S. Nuclear Regulatory Commission, April 1993, may be used for decontamination and survey of surfaces or premises and equipment prior to abandonment or release for unrestricted use.

These guidelines are included as a regulatory acceptance criterion in NUREG-1520, *Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility*, as an acceptable method of demonstrating compliance with the radiation survey and monitoring requirements in 10 CFR Part 20. See Chapter 4 for additional details.

Figure 1-1: Site Location

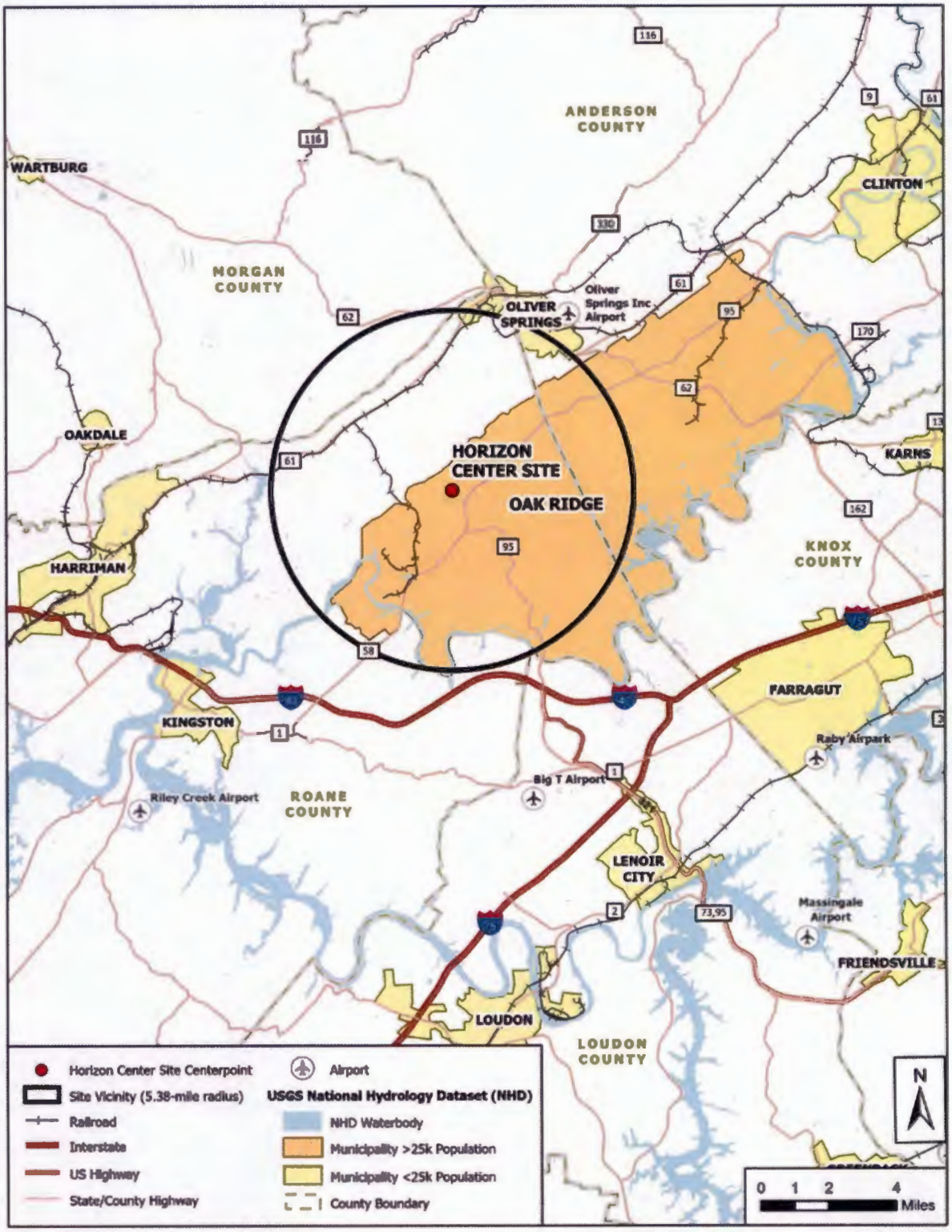
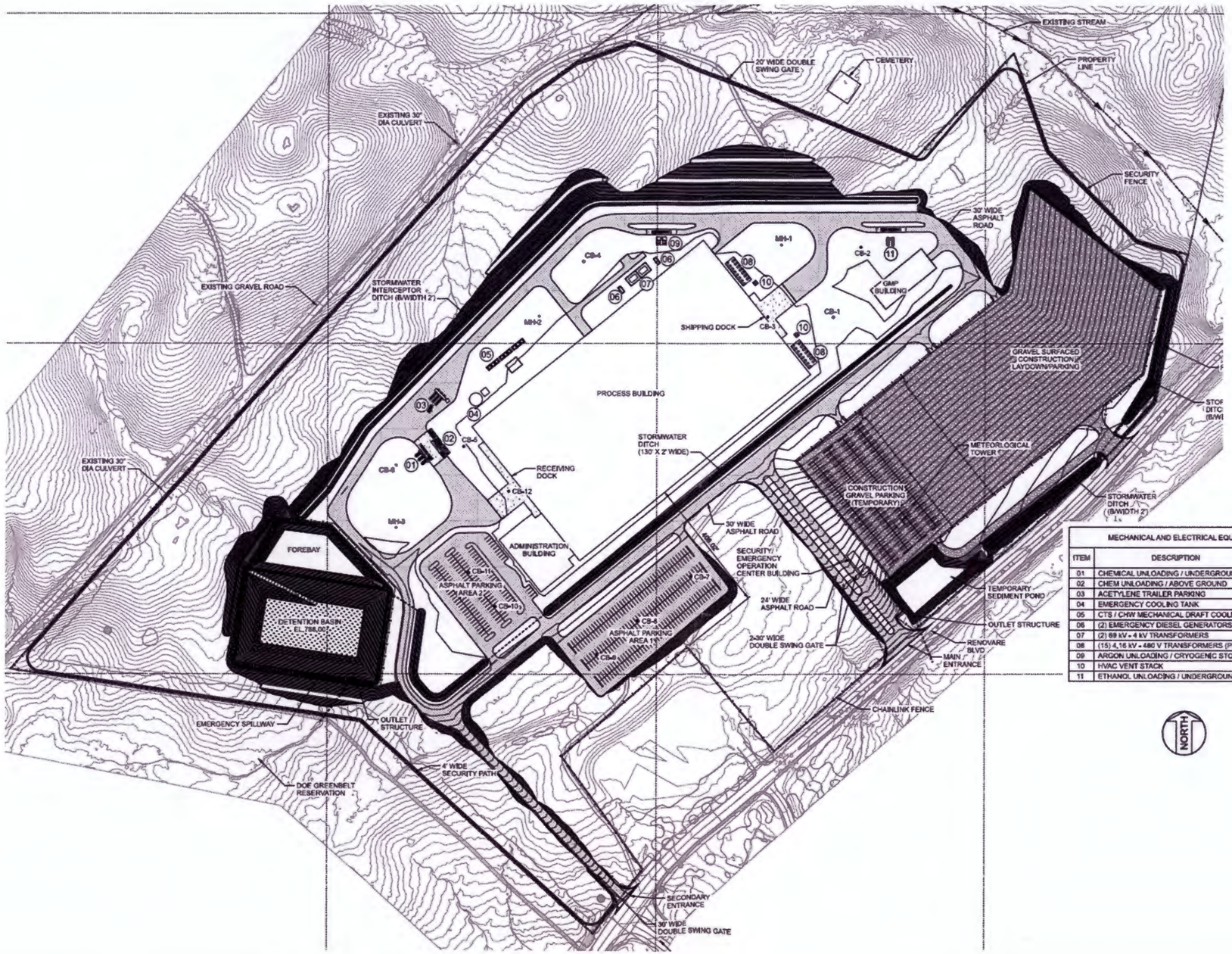


Figure 1-2: Site Plan



MECHANICAL AND ELECTRICAL EQUIPMENT	
ITEM	DESCRIPTION
01	CHEMICAL UNLOADING / UNDERGROUND TANKS
02	CHEM UNLOADING / ABOVE GROUND
03	ACETYLENE TRAILER PARKING
04	EMERGENCY COOLING TANK
05	CTS / CHW MECHANICAL DRAFT COOLING TOWERS
06	(2) EMERGENCY DIESEL GENERATORS
07	(2) 69 kV - 4 kV TRANSFORMERS
08	(15) 4.16 kV - 480 V TRANSFORMERS (PER LINE)
09	ARGON UNLOADING / CRYOGENIC STORAGE
10	HVAC VENT STACK
11	ETHANOL UNLOADING / UNDERGROUND STORAGE

NRC SPECIAL NUCLEAR MATERIAL LICENSE
SNM-XXXX
CHAPTER 2

ORGANIZATION AND ADMINISTRATION
Table of Contents

ORGANIZATION AND ADMINISTRATION

2.1 General Safety Policy and Responsibilities

It is TRISO-X's policy that radiation exposures to employees and the general public be kept ALARA. TRISO-X's policy further ensures that environmental protection measures are in place to control and monitor gaseous and liquid effluents and appropriately manage radioactive solid waste to ensure facility operations meet applicable regulations. Responsibility for safety in the various manufacturing lines, processes, and services is delegated to the lowest practical level of supervision. Safety is the responsibility of each supervisor within his/her own area. Through training and periodic retraining, each individual, regardless of position, is made aware that safety in his/her work area is ultimately his/her responsibility.

2.2 Site Organization

The TRISO-X organization provides the management, administrative, and technical capabilities for ensuring that design, construction, startup, modifications, and operations utilizing licensed materials are conducted in a manner that is protective of its workers, the public, and the surrounding environment, and remain in compliance with applicable Federal, State, and local regulations, licenses, and permits. This responsibility is implemented through the functional disciplines of manufacturing, engineering, regulatory affairs, and quality assurance, as described in the sections below, all of which have safety-related responsibilities. Figure 2-1 shows the current TRISO-X functional organization, including the independence of manufacturing, regulatory affairs, and quality assurance.

The management positions for each discipline together have the delegated responsibility for plant safety and for compliance with conditions of SNM licenses and with federal, state, and local regulations and laws governing operation of a nuclear facility in order to maintain a safe workplace for all employees. Each discipline management team is responsible for

- ensuring that all activities in their area are performed in a safe and effective manner;
- managing and directing operations within their discipline;
- ensuring that all operations under its guidance comply with safety and license conditions, requirements for quality-related safety activities, and safety-related configuration management requirements;
- being knowledgeable of the safety procedures and programs as they relate to their area of responsibility;
- developing, approving, and implementing procedures that incorporate safety and quality controls and limits commensurate with the particular operation involved; and
- ensuring that conditions adverse to safety are reported and investigated promptly, and that corrective actions are tracked to completion and, as applicable, monitored for effectiveness.

2.3 Organizational Responsibilities, Authority, and Qualifications

This section describes the functional responsibilities, education, and experience of key positions required by this license application.

Key personnel are those individuals who are responsible for safety and for safe operation of the site and include the plant manager and the discipline managers described in this section. Company policy requires written delegation of authority when senior managers are unavailable to perform their duties. The emergency plan delineates responsible management personnel and reporting relationships for handling site emergency situations.

The positions described in this section are intended to be generic in nature and do not reflect specific organizational or job titles. The responsibilities of the positions described may be fulfilled by one or more different organizational positions as long as the minimum position qualifications specified in this chapter are met. Similarly, functional areas shown in Figure 2-1 and described in this chapter may be grouped within their disciplines as needed to support the TRISO-X organization as long as the individual(s) responsible for the function(s) have a sufficient background to provide the capability for making sound safety and/or regulatory decisions. A combination of education and experience may be substituted for minimum qualifications described in this chapter if judged appropriate.

2.3.1 Plant Manager

The plant manager, or the discipline manager authorized to be his/her alternate, has the overall responsibility for the safety, security, quality, and operational aspects of all licensed activities conducted at the TRISO-X Fuel Fabrication Facility. Daily responsibility for licensed activities may be delegated in writing to one or more of the discipline manager positions specified in Sections 2.3.2 and 2.3.4.

The minimum qualifications for the plant manager are a BS/BA and/or advanced degree in science, engineering, or a technical field; at least five years of management experience in the nuclear industry and/or a nuclear-related field; and a general knowledge concerning the regulatory aspects of policies and procedures at the TRISO-X Fuel Fabrication Facility.

2.3.2 Manufacturing

The Manufacturing discipline is responsible for manufacturing-related activities involving the handling and processing of SNM, including developing operating procedures and maintaining facilities and equipment in a safe operating condition. This discipline includes activities associated with enriched uranium processing, transportation and waste management operations, and related equipment installation, start-up, and maintenance. This discipline manages the manufacturing technician work force and has line management responsibility for implementation of the safety programs and systems for conducting an active ALARA Program.

The minimum qualifications for a Manufacturing discipline manager are a BS/BA and/or advanced degree in science or engineering and at least five years of management experience in the nuclear industry and/or a nuclear-related field; and a sufficient background in manufacturing-related activities to provide the capability for making sound safety decisions.

The minimum qualifications for individual(s) responsible for manufacturing function(s) are a BS/BA and/or advanced degree in science or engineering and at least three years of experience in the nuclear industry and/or a nuclear-related field. He/she must have a sufficient background in manufacturing-related activities to provide the capability for making sound safety decisions.

2.3.3 Engineering

The Engineering discipline performs and/or provides oversight of activities involving design, construction, and/or installation of new and modified facilities and equipment; supplies maintenance and process engineering support; conducts activities associated with product research and development; assures that all equipment and facilities have appropriate safety controls and have been evaluated within the spirit and intent of ALARA; establishes configuration management (CM) as defined in Chapter 11 to ensure consistency among design and regulatory requirements, physical configuration, and facility configuration information; and maintains this consistency throughout the life of the facilities and activities until the point that CM is no longer needed.

The minimum qualifications for an Engineering discipline manager are a BS and/or advanced degree in engineering and at least five years of management experience in engineering-related activities, two years of which have been in the nuclear industry and/or a nuclear-related or other highly regulated field. He/she must have a sufficient background in manufacturing-related activities to provide the capability for making sound safety decisions.

The minimum qualifications for individual(s) responsible for engineering function(s) are a BS/BA and/or advanced degree in science or engineering and at least three years of experience in the nuclear industry and/or a nuclear-related or other highly regulated field.

2.3.4 Regulatory Affairs

The Regulatory Affairs discipline provides programs, procedures, and reviews to assure worker health and safety; environmental protection; and compliance with licenses and permits, including those related to transportation and disposal of licensed material. These activities are conducted with the ALARA principle in mind. Functional areas include nuclear criticality safety; radiation protection; environmental protection; industrial, chemical, and fire safety; integrated safety analysis; licensing; material control and accounting; security; and emergency preparedness. Emergency preparedness and response programs are supported by each functional area as needed. The integrated safety analysis (ISA) process is supported by each functional area providing ISA Team members as needed.

The Regulatory Affairs discipline monitors operations to ensure they are conducted in compliance with federal, state, and local regulations, and is authorized to suspend operations, approve re-start of operations, and/or require additional safety precautions when such measures are necessary in the interest of plant safety. The Regulatory Affairs discipline is administratively independent of the Manufacturing discipline, but both disciplines may report to a common management position.

The Regulatory Affairs discipline is responsible for overseeing the safety review committee as described in Section 2.4. The Chairperson of the safety review committee is considered to be a member of the Council and he/she may represent one of the disciplines/functions on the Council if approved by the plant manager, or designated alternate.

The minimum qualifications for a Regulatory Affairs discipline manager are a BS/BA and/or advanced degree in science or engineering and at least five years of management experience in the nuclear industry, a nuclear-related field, and/or in assignments involving regulatory activities. He/she must have appropriate understanding of the functional program(s) being managed.

2.3.4.1 Nuclear Criticality Safety Function

The nuclear criticality safety (NCS) function has responsibility for the development and implementation of a comprehensive nuclear criticality safety program, as defined in Chapter 5. Key responsibilities include the performance of NCS evaluations of applicable SNM operations and proposed changes to those operations; establishing NCS limits and controls based on those evaluations; assuring the proper incorporation of NCS engineered controls into design; assuring the proper incorporation of NCS limits and controls into applicable procedures and work instructions; and monitoring plant compliance with the NCS requirements through audits and assessments. The NCS function is administratively independent of Production and has the authority to shutdown potentially unsafe operations.

The minimum qualifications for the individual responsible for the nuclear criticality safety function are a BS/BA and/or advanced degree in science or engineering and at least three years of experience in nuclear criticality safety.

The minimum qualifications for nuclear criticality safety engineers are a BS/BA and/or advanced degree in science or engineering and successful completion of a formal internal training and qualification program that is defined in approved procedures.

2.3.4.2 Radiation Protection Function

The radiation protection function has responsibility for establishing and maintaining the radiation safety program necessary to ensure the protection of employees at the TRISO-X Fuel Fabrication Facility and the community, as defined in Chapter 4. Key responsibilities include management of the ALARA, dosimetry, and radiation monitoring and surveillance programs; analysis and approval of operations involving radiological safety and proposed changes to those operations;

establishing radiation protection criteria, procedures, and training programs to control contamination and exposure to individuals and the environment; and monitoring plant compliance with the radiological protection criteria through inspections and audits. Radiation monitoring includes measurement of airborne radionuclide concentration, contamination level, and external radiation levels; evaluation of the operational integrity and reliability of radiation detection instruments; and maintenance of records related to the radiation monitoring program. These tasks are accomplished through the use of health and safety technicians, health and safety technician supervisor(s), health physicists, and individual(s) responsible for the radiation protection function. The radiation protection function is administratively independent of production and has the authority to shutdown potentially unsafe operations.

The individual(s) responsible for the radiation protection function administer activities associated with radiological safety and has direct access to the plant manager (or equivalent) in vital matters of radiological safety. This includes monitoring and control of areas of airborne radioactivity, surface contamination, containment, ventilation, internal and external dosimetry, and bioassay services.

The minimum qualifications for the individual(s) responsible for the radiation protection function are a BS/BA and/or advanced degree in science or engineering, and at least three years of experience in radiation safety and have an understanding of the application and direction of radiation protection programs.

The minimum qualifications for a health physicist are a BS/BA and/or advanced degree in science or engineering, and at least one year of experience in health physics at a nuclear facility.

2.3.4.3 Environmental Protection Function

The environmental protection function has responsibility for establishing and maintaining the environmental protection program necessary to ensure the protection of the public and the environment, as defined in Chapter 9. Key responsibilities include identification of environmental requirements of federal, state, and local regulations governing TRISO-X operations; assurance of proper federal and state permits, licenses, and registrations for radiological and non-radiological discharges from the facility and waste handling and disposal activities; analysis and approval of operations involving potential environmental releases and proposed changes to those operations; establishing environmental protection criteria, procedures, and training programs to monitor gaseous and liquid effluents and waste handling and disposal; and monitoring plant compliance with the environmental protection criteria through inspections and audits. The environmental function is administratively independent of production and has the authority to shutdown potentially unsafe operations.

The minimum qualifications for the individual(s) responsible for the environmental protection function are a BS/BA and/or advanced degree in science or engineering, and at least three years of experience in applied health physics and/or environmental protection.

The minimum qualifications for an environmental protection analyst are a BS/BA and/or advanced degree in science or engineering, and at least one year of applied health physics or environmental protection experience.

2.3.4.4 Industrial, Chemical, and Fire Safety Functions

The industrial and chemical safety functions have responsibility for industrial hygiene or chemical safety, as defined in Chapter 6; industrial safety; and respiratory protection. Key responsibilities include analysis and approval of operations involving industrial safety and proposed changes to those operations; establishing industrial safety criteria, procedures, and training programs to protect the workers from industrial hazards; and monitoring plant compliance with the industrial safety/hygiene program through inspections and audits.

The fire safety function has responsibility for the fire protection program, as defined in Chapter 7. Key responsibilities include analysis and approval of operations involving fire safety and proposed changes to those operations; establishing fire safety criteria, procedures, and training programs to protect the workers from fire hazards; and monitoring plant compliance with the fire protection program through inspections and audits. The industrial, chemical, and fire safety functions are administratively independent of Production and have the authority to shutdown potentially unsafe operations.

The minimum qualifications for the individual(s) responsible for the industrial, chemical, and fire safety function(s) are a BS/BA and/or advanced degree, depending on functional assignment(s), in industrial hygiene, industrial safety, fire protection, or other appropriate field, and at least three years of experience related to functional assignment(s).

The minimum requirements for safety specialist positions are a BS/BA and/or advanced degree with specialized training, depending on functional assignment(s), in environmental health, fire protection, industrial safety/hygiene, chemical safety, or other closely related field, and at least one year of experience related to functional assignment(s).

2.3.4.5 Integrated Safety Analysis Function

The integrated safety analysis function has the overall responsibility for the ISA program, as defined in Chapter 3. Key responsibilities include the performance of chemical, radiological, and fire evaluations of applicable SNM operations and proposed changes to those operations; establishing IROFS based on those evaluations; assuring the proper incorporation of IROFS into applicable procedures and work instructions; coordinating updates to the ISA; and monitoring plant compliance with ISA requirements through inspections and audits. The ISA function also has responsibility for managing the Safety Related Equipment program for functionally testing IROFS on a periodic basis, as defined in Chapter 11.

The minimum qualifications for the individual responsible for the ISA function are a BS/BA and/or advanced degree in science or engineering and at least three years of experience in licensing, regulatory compliance, safety, and/or safety analysis in the nuclear or another highly regulated industry.

2.3.4.6 Licensing Function

The licensing function has overall responsibility for acquiring and maintaining safety-related licenses as required to operate the TRISO-X Fuel Fabrication Facility, as well as the broad responsibility for interface with regulatory agencies.

The minimum qualifications for the individual responsible for the licensing function are a BS/BA and/or advanced degree in science or engineering and at least three years of experience in licensing, regulatory compliance, safety, and/or safety analysis in the nuclear or another highly regulated industry.

2.4.3.7 Material Control and Accountability Function

The material control and accountability (MC&A) function maintains programs to assure that SNM is received, processed, stored, and transferred in accordance with federal regulations, and implements these functions through the areas of SNM safeguards, SNM accountability, shipping, and receiving.

The minimum qualifications for key MC&A personnel are defined in the Fundamental Nuclear Material Control Plan.

2.3.4.8 Security Function

The security function is responsible for implementing the security program as defined in the Physical Security Plan and the Safeguards Information Plan.

The minimum qualifications for key Security personnel are defined in the Physical Security Plan.

2.3.4.9 Emergency Preparedness Function

The emergency preparedness function is responsible for implementing the emergency management program as defined in Chapter 8 and the Site Emergency Plan.

The minimum qualifications for key Emergency Preparedness personnel are defined in the Site Emergency Plan.

2.3.5 Quality Assurance

The Quality Assurance discipline assesses systematic programs for indoctrination and training of personnel performing quality-related safety activities; for specifying during the design phase the extent of quality assurance or confidence necessary for quality-related safety structures, systems, and components; and for performing audits, surveillances, and assessments of quality-

related safety activities. The quality assurance program is based on, but is not limited to, applicable requirements and guidance in ISO 9001:2015. The Quality Assurance discipline is administratively independent of operations and has no other duties or responsibilities unrelated to quality assurance that would interfere with carrying out the duties of this discipline.

The minimum qualifications for the Quality Assurance discipline manager are a BS/BA and/or advanced degree in science or engineering and at least five years of experience in quality assurance-related activities in the nuclear or another highly regulated industry.

2.4 Safety Review Committee

The safety review committee membership includes discipline managers, or individuals responsible for regulatory affairs functions that meet the qualifications of a discipline manager, of the following disciplines:

- Manufacturing;
- Engineering;
- Safety & Regulatory; and
- Safeguards.

The chairman, other committee members, and their alternates, are appointed by the plant manager, or the discipline manager authorized to be his alternate. At a minimum, the chairman is required to have the qualifications specified for an individual responsible for a regulatory affairs function, and the other committee members are required to have the qualifications specified for a discipline manager. Members of the safety review committee, as identified above, have completed training in incident investigation methods, and the completion of the training is documented.

The committee is responsible to the plant manager, or the discipline manager authorized to be his/her alternate, who retains overall authority for the approval or disapproval of committee actions.

The authority and responsibilities of the full safety review committee include the following:

- Reviewing proposed license changes affecting safety, physical security, emergency preparedness, and/or material control and accountability before the associated license amendment applications are submitted to the NRC.
- Reviewing the ALARA program for at least the following:
  - Trends in air activity,
  - Cumulative exposure,
  - Engineering design and personnel work practices.
- Working with the Regulatory Affairs discipline to implement the ALARA program.
- Reviewing results of safety inspections, audits, and investigations which the license requires be conducted.
- Reviewing all violations of regulations or license conditions having safety significance.

The committee will meet at the following frequencies:

- to discuss topics such as proposed license changes - as needed;
- to discuss ALARA considerations - at least semiannually;
- to review license-required safety inspections, audits, investigations, and violations of regulations or license conditions - at least quarterly.

Its proceedings, findings, and recommendations will be documented in writing and made available to the plant manager and discipline management. Such reports will be retained for at least five years.

Committee review of matters other than the bulleted items above may be conducted by either individual review or collectively at a meeting; however, individual members of the committee have the authority to request a meeting of the entire committee on any given matter.

2.5 Administration

2.5.1 Reporting of Potentially Unsafe Conditions or Activities

A problem identification system is available for any person at the TRISO-X Fuel Fabrication Facility to report potentially unsafe conditions or activities to the Regulatory Affairs discipline. Prompt reporting is expected so that conditions adverse to safety can be corrected as soon as practicable. The concern is entered in the system and processed through a screening committee with Regulatory Affairs discipline representation. The screening committee assigns the issue to an owner and defines follow-up investigation/evaluation requirements. Corrective actions are assigned and tracked to completion. The Corrective Action Program is discussed further in Chapter 11.

2.5.2 Management Measures

Management measures that ensure the reliability and availability of IROFS are established as described in Chapter 11.

2.5.3 Off-Site Emergency Response Resources

Written agreements with off-site emergency response organizations are described in Chapter 8.

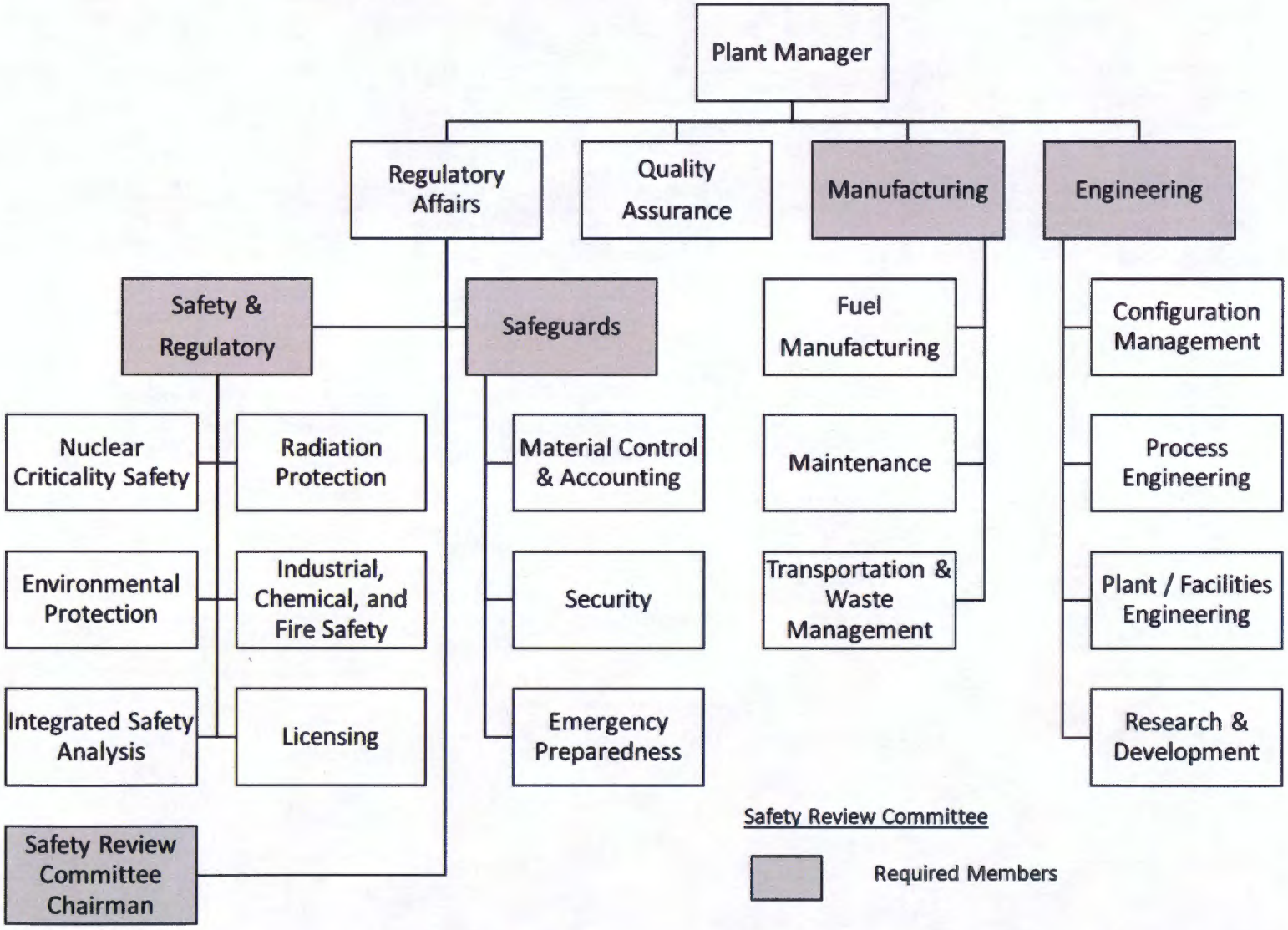
2.6 Transition from Design and Construction to Operations

The TRISO-X organization is also responsible for planning, organizing, and overseeing the initial testing and commissioning of the facility and equipment, including modifications in the future, using written plans and procedures. As the construction of systems is completed, they undergo functional and acceptance testing, as appropriate, as contained in approved procedures. Following successful completion of testing and commissioning, detailed transition plans and operational readiness reviews are used to confirm the equipment in each process area is

functionally tested and ready to operate, and the assigned staff is trained and ready to commence operations when authorized to do so.

The turnover will include physical systems and corresponding design information and records. Following turnover, the manufacturing organization will be responsible for system maintenance and configuration control. The design basis is maintained following the configuration management system described in Chapter 11, "Management Measures".

Figure 2-1: Functional Organization Chart



NRC SPECIAL NUCLEAR MATERIAL LICENSE
SNM-XXXX
CHAPTER 3

INTEGRATED SAFETY ANALYSIS
Table of Contents

INTEGRATED SAFETY ANALYSIS

3.1 Integrated Safety Analysis (ISA) Program and Commitments

TRISO-X maintains an ISA for the areas of the facility that involve or could impact the safe handling of SNM in accordance with 10 CFR 70 Subpart H and 10 CFR 70.65(a). The ISA Program establishes and maintains the safety program required by 10 CFR 70.62(a)(1) and consists of the following elements:

1) ISA Program commitments in this chapter,
2) ISA Summary documents, and
3) Supporting ISA information maintained at the facility.

3.1.1 Process Safety Information

Process safety information is compiled and maintained in sufficient detail to support the creation and updating of the ISA required by 10 CFR 70.62(b). The compilation of written process safety information includes information on the hazards, materials, technology, and equipment associated with each process. Process safety information can vary depending on the complexity of the operation, but it may include items such as piping and instrumentation diagrams (P&IDs), flow diagrams, process descriptions, and other aids that allow identification and understanding of the hazards associated with each process.

3.1.2 ISA Methods

The ISA conducted and maintained as required by 10 CFR 70.62(c) is a systematic analysis of TRISO-X FFF processes that identifies facility and external hazards and their potential for initiating credible accident scenarios; the consequences and likelihood of the credible accident scenarios; and the items relied on for safety (IROFS) needed to meet the performance criteria specified in 10 CFR 70.61. IROFS will be established and maintained such that they will be available and reliable as needed.

Credible accident scenarios are identified through PHAs using methodologies listed in NUREG-1513, "Integrated Safety Analysis Guidance Document," and the method is selected based on the complexity of the process to be analyzed and the severity of the hazards. Hazards reviewed include potential for inadvertent nuclear criticality; radiological exposures; and chemical, fire, and facility hazards that could increase radiological risk. Accident scenarios consider credible deviations from the normal operation of the process and failure of the IROFS for the process being reviewed, including consideration of human actions and errors that could lead to accidents of concern. Facility events external to the process being evaluated are also reviewed.

3.1.3 ISA Consequence Determination

For each credible accident scenario, the unmitigated consequences are evaluated using qualitative and/or quantitative methods, such as those described in NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook." Alternative methods and other industry accepted techniques may also be used to perform consequence calculations, provided the methods are appropriate to the process, the physical setting, and the specific condition being evaluated. Accident scenarios may be grouped if they will result in a similar consequence, such as an indoor spill of uranyl nitrate. Based on the evaluation results as compared to the 10 CFR 70.61 consequence thresholds, each credible accident scenario is assigned a consequence category of "Low," "Intermediate," or "High." Credible accident scenarios with the potential of resulting in a criticality are assumed to be "High" consequence events. The threshold criteria used to determine if unmitigated accident sequences have the potential to exceed the intermediate or high radiological or chemical consequence levels of 10 CFR 70.61(b) and (c) as summarized in Table 3-1 below.

Table 3-1
10 CFR 70.61 Radiological and Chemical Consequence Exposure Levels

Consequence Level	Radiological		Chemical (Note 1)	
	Worker	Public/Environment	Worker	Public/Environment
High	TEDE ≥ 100 rem	TEDE ≥ 25 rem	≥ CHEM3 ≥ 400 mg soluble U	≥ CHEM2 ≥ 30 mg soluble U
Intermediate	100 rem > TEDE ≥ 25 rem	25 rem > TEDE ≥ 5 rem 5000 x 10 CFR 20, Table 2, App. B limits averaged over 24-hour period	≥ CHEM2 < CHEM3 ≥ 150 mg and < 400 mg soluble U	≥ CHEM1 < CHEM2
Low	< Intermediate Levels	< Intermediate Levels	< Intermediate Levels	< Intermediate Levels

CHEM = AEGL, ERPG, or TEEL

Note 1: For chemical consequences, the Acute Exposure Guideline Levels (AEGLs) are used if available. If no AEGLs are available, Emergency Response Planning Guidelines (ERPGs) are used. If no ERPGs are available, Temporary Emergency Exposure Levels (TEELs) are used. If there are no AEGLs, ERPGs, or TEELs available, the ISA Summary identifies the methodology used to determine if the chemical poses an acute hazard. The soluble uranium intake limits are based on 10 CFR 70.61, ISG-14, NUREG-1391, and DOE-STD-1136-2017.

A risk assessment is performed for those credible accident scenarios with "Intermediate" or "High" consequences. Qualitative or quantitative risk assessment methods are used to determine the likelihood and risk of each credible accident scenario.

3.1.4 ISA IROFS Selection and Likelihood Determinations

IROFS are identified to prevent or mitigate each credible accident scenario such that the 10 CFR 70.61 performance criteria are met by "Intermediate" consequence events being unlikely and "High" consequence events being highly unlikely as defined in the ISA Summary. IROFS, in preferential order, may be engineered controls (passive or active), enhanced administrative controls (active features that prompt a person to take an action), or administrative controls (actions of people).

The Initiating or Enabling Event Frequency Index is a numerical value assigned to each accident sequence based on a qualitative assessment of how often the event is expected to occur considering industry acceptable values, engineering judgement, analytical data, and past experience. The frequency index numbers are defined in Table 3-2. The index may be one value higher or lower than listed with sufficient justification in the ISA documentation, and procedures provide additional guidance for determining Initiating/Enabling Event Frequency Index numbers.

Table 3-2
Initiating and Enabling Event Frequency Index Numbers (IE/EE)

Frequency Index	Description
-5	Not credible
-4	Physically possible, but not expected to occur
-3	Not expected to occur during the plant lifetime
-2	Not expected, but might occur in plant lifetime
-1	Expected to occur during plant lifetime
0	Expected to occur regularly during plant lifetime
1	A frequent event

Engineered and administrative controls that are needed to meet the performance requirements of 10 CFR 70.61 are designated as IROFS. Each IROFS is assigned an Effectiveness of Protection Index (EOPI) as specified in Table 3-3. The index is a numerical value assigned to each IROFS based on a qualitative assessment representing the credit given to IROFS in accident sequences, considering:

- Characteristics of the IROFS used to prevent or mitigate the accident of concern (safety margin, type of control, proper design, independence, reliability, availability, and other management measures)

- Industry acceptable values, engineering judgement, analytical data, and past experience
- Documented additional basis for administrative IROFS index numbers considering human reliability and failure probabilities
- Documented additional basis for failure rates of equipment and / or demand failure probabilities

The index may be one value higher or lower than listed in Table 3-3 with sufficient justification as documented in the ISA.

Table 3-3
IROFS Effectiveness of Protection Index (EOPI)

Effectiveness of Protection Index	Type of IROFS
-4	Protected by an inspected Passive Engineered Control, or an inherently safe active engineered control, with management measures applied to ensure availability and reliability.
-3	Protected by a functionally tested Active Engineered Control with management measures applied to ensure availability and reliability.
-2	Protected by a trained operator performing a routine task with an approved procedure; an enhanced administrative control; or an administrative control with margin of safety, independence, and management measures applied to ensure availability and reliability.
-1	Protected by a trained operator performing a non-routine task with approved written instructions.
0	No protection

If the initiating event involves an IROFS failure, the Effectiveness of Protection Index is assigned to the IROFS failure rather than the Initiating Event Frequency Index number. The duration of the IROFS failure is also considered based on the Duration Index provided in Table 3-4. The Duration Index is a numerical value assigned based on a qualitative assessment of how quickly the IROFS failure would be discovered considering engineering judgement, design information, past experience, and the associated management measures. The Duration Index is identified as "Dur" in the risk assessment.

Table 3-4 Duration Index (Dur)	
Duration Index	**Average Failure Duration**
1	More than 3 years
0	1 year
-1	1 month
-2	A few days
-3	8 hours
-4	1 hour
-5	5 minutes

In order to credit a Duration Index of less than -1, one of the following conditions must be satisfied so actions can be taken to address a failed IROFS:

- IROFS failure would be easily detected based on its location and type of failure (e.g., within frequent travel path)
- IROFS failure would be easily detected based on the activity that causes its failure (e.g., maintenance activity that causes a release of material from a vessel would be immediately recognized by the personnel performing the work)
- IROFS failure is identified by periodic tests or required surveillances corresponding to the selected duration (e.g., shift inspection, functional testing, etc.)

The Likelihood Index is determined by summing the Initiating/Enabling Event Frequency Index numbers, the Effectiveness of Protection Index for each IROFS, and when applicable, the Duration Index assigned to the accident sequence. If an accident sequence uses a Duration Index, the sequence needs to be reversed when the failure of one of the other IROFS can lead to a more positive index (i.e., results in a higher Likelihood Index).

For each accident sequence, the Total Likelihood Index is calculated as:

Likelihood Index T = IE/EE + EOPI for each IROFS + Dur (where applicable)

The Likelihood Category is determined using the Likelihood Index as specified in Table 3-5.

Table 3-5 Total Risk Likelihood Category	
Likelihood Index (T) (=sum of index numbers)	**Likelihood Category**
$T \leq -4$	**1**
$T = -3$	**2**
$T > -3$	**3**

3.1.5 ISA Risk Index Determination

The Risk Index is determined by multiplying the Likelihood Category (3.1.4) by the Consequence Category (3.1.3). The Risk Index is then compared to the Risk Matrix in Table 3-6 to determine if the Risk Index is acceptable or unacceptable.

Table 3-6
Risk Matrix with Risk Index Values

Severity of Consequences	Likelihood of Occurrence		
	Likelihood Cat. 1 Highly Unlikely	Likelihood Cat. 2 Unlikely	Likelihood Cat. 3 Not Unlikely
Consequence Cat. 3 High	Acceptable Risk 3	Unacceptable Risk 6	Unacceptable Risk 9
Consequence Cat. 2 Intermediate	Acceptable Risk 2	Acceptable Risk 4	Unacceptable Risk 6
Consequence Cat. 1 Low	Below Severity Threshold		

3.1.6 ISA Team Qualifications

PHAs are conducted by an ISA Team with membership commensurate with the process being reviewed. The team typically consists of a team leader; individuals knowledgeable of the process being analyzed; and individuals representing the safety discipline, including nuclear criticality safety, radiation protection, chemical safety, and fire safety; consistent with 10 CFR 70.62(c)(2). Team members may represent more than one functional area being evaluated. Disciplines that are not affected by the proposed process or change being evaluated do not require representation on the team.

The team leader is designated as the one member of the team who is knowledgeable in the methodologies used to conduct PHAs and ensures that the team members understand the methodology to be used.

3.1.7 ISA Change Management

Changes to the facility or its processes that impact the ISA are evaluated using a configuration management program that meets the requirements of 10 CFR 70.72, as described in Chapter 11, so that the ISA and its supporting documentation remain accurate and up-to-date.

Proposed changes to the facility or its processes are evaluated in accordance with the ISA Methods and ISA Team Qualifications described in this chapter. If a proposed change results in a new credible accident scenario being identified or increases the consequences and/or likelihood of a previously analyzed accident scenario, the existing IROFS and associated management measures are evaluated promptly for adequacy and new IROFS are identified or

changes are made, if required. IROFS with unacceptable performance deficiencies, as identified through the corrective action program or during updates to the ISA, are addressed.

The ISA Summary is updated at least annually by January 30th, incorporating changes made in the previous calendar year that affected the ISA Summary. The updated documents or pages are submitted to the NRC per 10 CFR 70.72(d)(3).

3.2 ISA Summary and ISA Documentation

The ISA Summary contains the following elements as required by 10 CFR 70.65(b):

- General description of the site;
- General description of the facility;
- Description of facility processes, hazards, and types of accident sequences;
- Demonstration of compliance with 10 CFR 70.61 performance requirements;
- Description of the ISA team qualifications and ISA methods;
- List of IROFS;
- Description of chemical consequence standards;
- List of sole IROFS (if any); and
- Definitions of the terms "credible," "unlikely," and "highly unlikely."

The ISA Documentation includes supporting information such as PHAs, Nuclear Criticality Safety Evaluations, Radiological / Chemical Accident Consequence Evaluations, and Fire Hazard Analyses. It also includes any completed 10 CFR 70.72 change management documentation that may not have yet been included in the annual update of the ISA Summary.

3.3 Management Measures

Management measures required by 10 CFR 70.62(d) to ensure the reliability and availability of each IROFS are established as described in Chapter 11.

3.4 Recordkeeping

TRISO-X documentation developed in accordance with Section 3.2 and as required by 10 CFR 70.62(a)(2) are maintained as records. TRISO-X also maintains records of failures of IROFS or management measures required by 10 CFR 70.62(a)(3). The TRISO-X records management program is described in Chapter 11.

3.5 Requirements for New Facilities or New Processes at Existing Facilities

As described in Section 11.1.2, design requirements are required to be developed, reviewed, approved, and documented for new facilities and processes/systems before input of SNM. The baseline design criteria identified in 10 CFR 70.64(a) are addressed for IROFS. As required by 10

CFR 70.64(b), the TRISO-X Fuel Fabrication Facility is designed using a defense-in-depth approach for protection against process-related accidents. To the extent practicable, the facility design considers preference for the selection of engineered controls over administrative controls to increase overall system reliability, and features that enhance safety by reducing challenges to IROFS are incorporated. The facility design addresses the baseline design criteria of 10 CFR 70.64(a) for new facilities and the control of process hazards.

Design Basis natural phenomena hazard (NPH) initiating events such as earthquakes, tornadoes, and high winds, and flooding that could cause adverse damage/consequences are demonstrated to be highly unlikely in the ISA documentation to meet 10 CFR 70.62(c)(iv), and the requirement of 10 CFR 70.64(a)(2) is met because the facility design provides for adequate protection against natural phenomena with consideration of the most severe documented historical events for the site.

If a planned new facility and/or new process meets the 10 CFR 70.72 criteria requiring a license amendment, the baseline design criteria of 10 CFR 70.64(a) will be applied to the control of process hazards. A defense-in-depth approach will be applied to higher risk accident sequences as required by 10 CFR 70.64(b).

NRC SPECIAL NUCLEAR MATERIAL LICENSE
SNM-XXXX
CHAPTER 4

RADIATION SAFETY
Table of Contents

RADIATION SAFETY

4.1 Radiation Protection Program

TRISO-X has established, maintains, and implements a Radiation Protection Program (RPP) in accordance with 10 CFR 20.1101 that is commensurate with the scope and extent of licensed activities and ensures compliance with applicable sections of 10 CFR 20. The purpose of the RPP is to maintain occupational and public doses below regulatory limits and as low as reasonably achievable (ALARA). Engineered controls (e.g., confinement, ventilation, equipment layout) provide primary radiation protection functions. Additional protection for workers is provided through an effective RPP that focuses on maintaining exposures to ionizing radiation ALARA. The guiding principles and requirements outlined in the RPP are reflected in programs and implementing procedures to ensure that:

- Radiation exposure to occupational workers and the public is maintained ALARA,
- Radiation protection staff are trained and qualified to carry out radiation protection procedures,
- Work is guided by and supplemented with detailed and effective radiation protection procedures and radiation work permits (RWPs),
- Personnel are trained in radiation protection principles, how to use tools and techniques to minimize exposure to radiation, and qualified to properly use Personal Protective Equipment (PPE),
- Facility ventilation and containment systems are designed to control airborne concentrations of radioactive material,
- A radiological survey and monitoring program is in place to document levels of radiation and contamination, and document occupational exposures to radiation,
- IROFS that limit high and intermediate consequences are identified, are consistent with regulatory performance criteria, have appropriate management measure in place to ensure they are available and reliable, and,
- Programs and procedures are in place that address records maintenance, corrective actions, and reporting requirements, as described in Chapter 11.

TRISO-X management commitment to keep exposures ALARA is documented in a formal policy statement that holds all levels of management and individual workers responsible for adhering to the company's ALARA policy. To ensure that this commitment is implemented without influence from production demands, the organizational structure reflects radiation protection staff reporting relationships that are independent from the facility's operations as discussed in Chapter 2. Radiation protection staff have clearly defined responsibilities, and possess the authority, and training, tools and equipment to carry out those responsibilities. As required by

10 CFR 20.1101(c), the RPP is reviewed on an annual basis and considers changes in the facility, technologies, and processes that could enhance the effectiveness of ALARA implementation.

4.2 ALARA Program

ALARA implementation is supported at the highest levels of management through written policy. These policies provide trained and qualified radiation protection staff with the authority to prevent practices that are not aligned with the ALARA philosophy. Management promotes safety in general, and radiation safety especially, by supporting a culture of interdependence, learning, employee ownership and input that is essential for improving safety and keeping exposures ALARA. Management is committed to providing facilities, operating and maintenance procedures, and equipment that ensures that exposures to radiation are ALARA. Additionally, ALARA principles are integrated into approved policies and procedures and are an integral component in the generation and use of Radiation Work Permits (RWPs).

The ALARA program is one of the several ways RP personnel interact with facility personnel. RP personnel are also involved in the preparation of RWPs. To prepare an RWP, RP personnel must interact with facility personnel to fully understand the activity and facility conditions to assess associated radiological hazards. RP personnel also interact with facility personnel with participating in safety audits.

TRISO-X's safety review committee, as described in Chapter 2, serves as the ALARA Committee. ALARA topics are reviewed and discussed at least annually to monitor employee dose and environmental release trends, identify areas for improvement, to set ALARA goals, implement required changes and review ALARA performance.

The Radiation Safety Officer (RSO) is responsible for overall radiation safety and ensuring that exposures are ALARA. Areas of responsibility include:

- Involvement in planning routine and non-routine work activities.
- Ensuring that work is guided by RWPs that effectively describe work requirements, precautions, and ALARA engineering controls.
- Ensuring that radiation protection instrumentation, equipment, and supplies are available, in good working order, and are properly used and maintained according to approved procedures based on manufacturers specifications.
- Auditing facility ALARA implementation (as part of the RPP annual review) to include:
    - Exposure record review
    - Radiological inspections (survey results)
    - Interviews with staff concerning application and implementation of ALARA
- Evaluating operating and maintenance procedures, equipment and facilities for modifications that could reduce occupational and public exposures.

- Tracking facility exposures and trends and proposing corrective actions.
- Assuring the proper implementation of radiological work controls.
- Evaluating the effectiveness of plans and procedures in maintaining occupational and public doses ALARA.

Radiation protection staff are responsible for overseeing the implementation of ALARA practices on an everyday basis. They are an essential resource for evaluating TRISO-X FFF processes for modifications and improvements for ALARA purposes.

4.3 Organization and Personnel Qualifications

Staff that are responsible for the management and implementation of the radiation protection program possess the education and training commensurate with their responsibilities. Responsibilities, and the authority necessary to implement those responsibilities, are delineated in policies and procedures. The management system and administrative procedures, including radiation protection staff education, experience, and training requirements are detailed in Chapter 2.

4.4 Procedures and Radiation Work Permits

Approved RPP implementing procedures incorporate radiation protection requirements found in 10 CFR 19, 20, 70, and 71, and other applicable regulations. The RPP is implemented through approved radiation safety procedures and inclusion of radiation protection requirements in operating procedures, equipment maintenance procedures, and RWPs to alert workers to special hazards or controls necessary for their protection. Radiation protection and ALARA principles are integrated into all facility process and maintenance procedures. The RSO reviews and approves all radiation protection procedures. As described in Chapter 11, the facility Document Control Process ensures that procedures are promptly modified when there are changes in technology or practices and guides procedure authorization and distribution. Training specifically addresses procedural changes. The following is a general (not comprehensive), list of tasks or topics associated with radiation safety for which procedures are established.

- Measuring and reporting occupational dose
- Measuring and reporting dose to the Public
- Radiation Surveys and Monitoring
- Posting and Labeling
- Access Control
- Air Sampling/Monitoring
- Care and Use of PPE
- Receiving and Opening Packages
- Storage and Control of Licensed Material
- Waste Management and Disposal

- Issuance and Termination of RWPs
- Investigation levels and action guides for external and internal exposure

The RWP system is documented in radiation protection procedures and are issued for specific tasks and for a specified period-of-time. The RSO (or designee) must review and approve RWPs. The RWP describes the tasks, work area known or potential radiological conditions, access, monitoring, and PPE requirements, and any special conditions or precautions. RWPs are posted at the work site, and personnel assigned to work under the RWP review the requirements prior to entry into the area.

4.5 Radiation Safety Training

TRISO-X management provides an effective radiation safety training program that meets regulatory requirements, that ensures that the working environment is safe, and ensures that employees and visitors understand the risks associated with exposure to radioactive materials. Training programs are designed and implemented to comply with the requirements of 10 CFR Parts 19 and 20. Chapter 11, Management Measures, addresses the training that ensures that administrative control IROFS are available and reliable.

Regulatory guidance used to develop the Radiation Protection Training Program includes:

- ANSI/HPS N13.36-2001 "Ionizing Radiation Safety Training for Workers"
- NCRP Report No. 134 "Operational Radiation Safety Training"
- ASTM E1168-95 (R2008) "Radiological Protection Training for Nuclear Facility Workers"
- Regulatory Guide 8.10, Rev. 2, August 2016, "Operating Philosophy for Maintaining Occupational and Public Radiation Exposures As Low As Is Reasonably Achievable"
- Regulatory Guide 8.13, Rev. 3, June 1999, "Instructions Concerning Prenatal Radiation Exposure"

A graded approach that is commensurate with radiation protection responsibilities is applied to Radiation Safety Training. Levels of training are based on regulatory requirements, the potential for radiation exposure, and the complexity of the task. Training and qualification of workers; including site orientation, general employee, radiation worker, radiation safety technician, and specialty training is provided commensurate with the hazard and planned activities. The level of training reflects prior training, personnel that are responsible for supervising others, and personnel that are directly or continuously supervised. To facilitate feedback that could assist in keeping exposures ALARA, training for Health Physics personnel responsible for performing surveys includes specific training on process related operations and tasks. Refresher training is provided every three years and addresses changes in policies, procedures, requirements, and the facility ISA.

A formal review and evaluation of the radiation protection training program for accuracy, effectiveness, and adequacy of the curriculum and instructors is performed at least every three years.

Training is provided to all personnel and visitors entering restricted areas that is commensurate with the potential radiological health risks associated with that individual's responsibilities. Training incorporates, where appropriate, the provisions in 10 CFR 19.12, and also includes, but is not limited to, training on the following topics.

- Radiation hazards and health risks,
- Radiation Safety principles, policies, and procedures,
- Safe storage, transfer, and handling of radioactive materials,
- Maintaining Radiation Dose ALARA,
- Access, egress controls, and escort procedures,
- Contamination control procedures, and PPE,
- ALARA and exposure limits,
- Internal and external exposure control and monitoring,
- Radiation exposure reports to individuals,
- Monitoring instruments,
- Emergency response, and
- Reporting responsibilities.

Further information on the training program is found in Chapter 11.

4.6 Ventilation and Respiratory Protection

The TRISO-X FFF designs incorporate ALARA principles for confinement and ventilation systems to limit airborne contamination levels. Engineered controls and features minimize potential inhalation of radioactive and other hazardous materials under all normal operating conditions; therefore, for normal operations, respiratory protection is not required. Equipment is maintained and tested to ensure systems operate when required and are within their design specifications, as described in Chapter 11. If engineering controls are not practical or feasible (certain maintenance operations for example) and the potential exists for levels of radioactive material to exceed those that define an airborne radioactivity area, monitoring is increased and intakes are limited by either control of access, limiting exposure times, or the use of respiratory protection.

4.6.1 Building / Area Ventilation

Barriers in the form of containment, ventilation, and filtration are designed to reduce discharges of radioactive material to ALARA levels. Appropriately sized ventilation is provided in areas of the facility where the potential exists for airborne concentrations of radionuclides to exceed the Derived Air Concentration (DAC) values during normal operations based on the dose coefficient values in ICRP Publication 68 and later. The design of the confinement ventilation system ensures the desired airflow during normal operations. Air that is recirculated is filtered through at least one stage of HEPA filtration.

The confinement and ventilation systems include the following critical design characteristics and functional measurements.

(1) Air flows are in the direction from areas with lower levels of potential contamination to areas with a higher potential for contamination.

(2) Areas where fume hoods and glove boxes are located are maintained at a negative pressure with respect to atmosphere during normal operation conditions.

(3) Ventilation ducts are designed to minimize accumulations of radioactive material.

(4) Fans are provided with variable frequency drives to allow maintenance of the flowrate as filter loading increases, and redundant capacity to service the full design load. Measurements are made to ensure that noise from the fans does not interfere with the ability to hear audible alarms.

(5) Process area air filtration includes a pre-filter bank and a HEPA filter bank in series. HEPA filters have 99.97% efficiency to filter out 0.3-micron particles from the air stream and fire resistance rating of UL 586. Filters are placed to facilitate maintenance and repair, and where possible, bag-in/out type filter housings are used to lessen personnel exposures.

4.6.2 Localized Ventilation

Where necessary, enclosures (i.e., hoods, gloveboxes, downdraft tables) or other localized ventilation designs are used to prevent the spread of airborne contamination within the facility and further limit the potential for intake by inhalation. In process areas, the design criteria for inward air flow through the open face of a containment enclosure used to handle radioactive material which has a propensity to suspend in air is at least 125 (+/-25) linear feet per minute (LFM). For operations, the inward air flow is maintained at least 100 (+/- 20) LFM.

For openings used to transfer containerized material or equipment, or for open face enclosures where excessive air flow interferes with sensitive analytical equipment or manufacturing process steps, the proper average face velocity and minimum rate of flow is established in approved procedures. Process enclosure air (from hoods, gloveboxes, etc.) is exhausted through the primary HEPA filter units to the atmosphere through monitored facility vent stacks.

The velocity of airflow at the entrance of all hoods and other enclosures and capture points are evaluated on a scheduled basis. Devices are provided to measure the differential pressure within a containment enclosure with respect to the outside atmosphere, except in containment enclosures where the nature of an operation makes this requirement impractical for processing purposes. Corrective action will be taken as soon as possible if the airflow is found to be deficient. Processes will cease operation in an enclosure if the average face velocity falls below 100 LFM, or if operability of enclosures is impaired.

4.6.3 Laboratory Ventilation

In laboratory areas, the design criteria for inward air flow through the open face of a containment enclosure used to handle radioactive material which has a propensity to suspend in air is in accordance with ANSI/AIHA Z9.5-2012 recommendations. The proper average face velocity and minimum rate of flow is established in approved procedures.

Any ventilated containment with an open door or port through which uncontainerized radioactive material is routinely handled is subject to these requirements; however, the intermittent opening of a door, glove port, etc. for the sole purpose of adding or removing containerized material or equipment does not constitute handling radioactive material with a propensity to suspend in air. In addition, any ventilated containment with an opening to the room which is high efficiency particulate air (HEPA) filtered for exhaust or over-pressurization protection is excluded from inward air flow requirements.

4.6.4 Respiratory Protection

When it is impractical to apply process or other engineering controls to limit concentrations of uranium in the air below those defined in 10 CFR 20.1201(e), other precautionary procedures, such as increased surveillance, limitation of exposure times, or provision of respiratory protective equipment are used to keep the intake of uranium by any individual within regulatory limits.

The respiratory protection program is developed according to the requirements detailed in 10CFR20 Subpart H – *Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas*. Appropriate protection factors consistent with 10 CFR 20 Appendix A are applied when calculating intake or the Committed Effective Dose Equivalent (CEDE).

Approved procedures guide the selection, fitting, issuance, maintenance, testing, training of personnel, monitoring, and recordkeeping for individual respiratory protection equipment and for specifying when such equipment is to be used. These procedures are revised to reflect changes in processes, the facility, or equipment that are significant enough to impact respirator use. Records are maintained as described in Chapter 11.

The Respiratory Protection Program requires that individuals must be medically qualified, trained (to include proper donning and doffing), and quantitatively fit tested to a specific respirator prior to initial use and on an annual basis. Procedures and training, as described in Chapter 11, include requirements for cleaning, inspection, and replacement of respirator parts.

4.7 Radiation Survey and Monitoring Programs

The facility RPP procedures detail a comprehensive survey program that implements operational surveys for radiological control. Operational surveys characterize workplace conditions, verify the effectiveness of engineering and administrative controls, evaluate changes in radiological conditions, identify areas requiring radiological posting, and assess radiological conditions during the performance of work. Training for Health Physics personnel responsible for performing

surveys ensures that RP personnel are sufficiently familiar with process activities so that precautions can be taken to minimize exposures.

Regulatory requirements related to monitoring are contained in 10 CFR 20 Subpart C, "Occupational Dose Limits"; Subpart D, "Radiation Dose Limits for Individual Members of the Public"; and Subpart F, "Surveys and Monitoring". Facility programs, policies and procedures guide the types of surveys, monitoring (including the air sampling and dosimetry programs), and evaluations for compliance with these subparts.

Types of surveys include external dose rate surveys, surface contamination surveys, air concentration surveys, surveys of personnel, PPE, equipment, waste, packages, and on containment/ventilation systems. Minimum survey and monitoring frequencies are shown in **Table 4-1** which was developed from Appendix B Reg Guide 8.24. Health Physics staff ensure that signs, labels, signals, other access controls, required notices to employees, copies of licenses, and other items are properly posted, legible, and operative, as required by 10 CFR Part 19, Notices, Instructions, and Reports to Workers: Inspection and Investigations, and 10 CFR Part 20 or specific license conditions.

Radiation survey and monitoring procedures address the procedure objectives, sampling processes, data analysis, equipment and instrumentation, measurement frequency, records, and reporting requirements, and required actions when measurements exceed regulatory or administrative or action levels. Survey documentation is retained in accordance with 10 CFR 20.2103. Survey and monitoring records are reported as mandated in 10 CFR Part 19 and 10 CFR Part 20 using the processes described in implementing procedures.

An adequate number of Radiological Protection instruments are maintained and used to perform radiological surveys and provide the necessary analytical results. Radiation Protection instrumentation includes: installed instrumentation such as a Criticality Accident Alarm System, Continuous Air Monitors, and Area Radiation Monitors; laboratory instrumentation such as Gas Flow Proportional Monitors for smear and air sample counting; stationary contamination monitors such as Hand and Foot Monitors, and Personal Contamination Monitors; and an array of portable instrumentation for radiation and contamination monitoring (alpha scintillators, beta-gamma plastic scintillators, Geiger-Muller type instruments, ionization chambers, and neutron counter). Available instruments are designed and manufactured to detect the radiation types of concern for the facility and that exhibit the level of sensitivity necessary to detect radiation at or below applicable action levels.

All radiation protection instrumentation is subject to a routine maintenance and calibration program to ensure that properly calibrated and operable instruments are available for use by health physics staff. Instrumentation is calibrated annually or following maintenance to NIST traceable standards and according to manufacturer's recommendation. All instrumentation is source checked prior to each day of use. The quality control requirements for operability checks are detailed in implementing procedures.

4.7.1 Radiation Surveys

The Health Physics Staff perform preoperational, routine, and special surveys of facility areas. All areas in which radioactive materials are stored or processed are identified prior to operations. Surveys are performed to identify areas in which personnel monitoring would be required according to the requirements of 10 CFR 20.1502(a)(1). The frequency and scope of routine surveys where radioactive materials are stored or processed, and where workers have access, are based on radiological content and potential dose rates.

4.7.2 Dosimetry Program

The dosimetry program incorporates measurements of external exposure, internal exposure, and air sampling data to complete the dose record for each monitored employee. The dose record is compared to the dose limits specified in Subpart C of 10 CFR 20. If engineering controls are not adequate to limit exposures to below regulatory levels, additional procedures such as limiting exposure time to licensed material or use of respiratory protection are implemented.

The External Dosimetry Program establishes the criteria for participating in the program, identifies target radiation types, establishes monitoring methods, assessments and recording criteria, specifies the dosimetry required, use, processing and evaluation, and documents administrative investigation and action levels.

The Internal Dosimetry Program is designed in compliance with 10 CFR 20.1201 "Occupational Dose Limits for Adults, 10 CFR 20.1204 "Determination of Internal Exposure", and the participation levels detailed in 10 CFR 1502(b) for adults, minors, and pregnant women. The program establishes the criteria for participating in the program, identifies the types of sampling, frequency of measurement and required detection levels, details the methods for measuring, assessing and recording intakes, details the evaluation and interpretation of the analytical results, and documents the facility investigation and action levels.

The monitoring requirements in 10 CFR Part 20 are summarized in **Table 4-2**. In accordance with the requirements of 10 CFR Part 20.1202, the TEDE is calculated by adding the Deep Dose Equivalent (DDE) to the CEDE for each person who requires both internal and external dose monitoring. The TEDE will not exceed the 10 CFR Part 20 dose limits. Investigation levels and action guides for external and internal exposure are established in RPP procedures.

4.7.2.1 External Dosimetry

Monitoring for exposure to external radiation is established according to the requirements in 10 CFR 20.1502(a) if external occupational dose is likely to exceed 10% of the dose limit appropriate for the individual (adult, minor, or declared pregnant woman). External radiation monitoring is also provided according to 10 CFR 20.1502(a)(3) for any individual entering a high or very high radiation area (areas requiring HRA and VHRA posting are not anticipated at the TRISO-X FFF). Beta-gamma sensitive thermoluminescent type dosimeters (TLDs) capable of measuring deep dose to the whole body, shallow dose to the skin or extremities, and dose to the lens of the eye are individually assigned for routine external exposure monitoring based on work area surveys,

occupancy time, or other exposure information such as area monitor results. Personnel dosimeters are provided, exchanged, and processed by a National Voluntary Laboratory Accreditation Program (NVLAP) accredited dosimetry vendor or supplier. Other types of dose measuring devices may be used including electronic dosimeters, direct-reading dosimeters, extremity dosimeters, neutron dosimeters and/or measurements made with portable radiation surveys instruments.

4.7.2.2 Internal Dosimetry

The TRISO-X internal dosimetry program establishes and manages a bioassay program to monitor and evaluate intakes from uranium, distributions of uranium within the body following intake, and the resulting radiation doses or possible chemical effects. The bioassay measurements are used to confirm the adequacy of radiological controls and to determine compliance with the occupational dose limits.

Facility personnel likely to receive greater than 10% of the applicable Annual Limit on Intake (ALI) values are monitored for intakes of radioactive material. Intakes are assigned to individuals based on air sampling (described in Section 4.7.2.3), urinalysis and/or in vivo lung counting. Intakes are converted to committed dose equivalent (CDE) and committed effective dose equivalent (CEDE) for the purposes of limiting and recording occupational doses

The appropriate routine bioassay frequency is based on (1) the potential exposure of the individual, (2) the retention and excretion characteristics of the radionuclide, (3) the sensitivity of the measurement technique, and (4) the acceptable uncertainty in the estimate of intake and committed dose equivalent. Bioassay measurements used for demonstrating compliance with the occupational dose limits are conducted often enough to identify and quantify potential exposures and resultant intakes that, during any year, are likely to collectively exceed 0.1 times the ALI. **Table 4-3** lists the minimum bioassay frequencies for bioassay program participants. Further evaluation, such as examination of airborne measurements or additional bioassay measurements, to obtain the best estimate of actual intake is performed if results exceed 0.02 times the annual limit on intake (ALI), or 40 derived air concentration (DAC) hours.

Urine sample concentrations are analyzed by either Kinetic Phosphorescence Analyzer (KPA) or Inductively Coupled Plasma Mass Spectrometry (ICPMS). Action levels are established in RPP procedures to prevent an individual from exceeding the occupational exposure limits specified in 10 CFR 20 and to protect against toxicological damage to the kidney. Control actions include temporarily restricting the individual from working in an area containing airborne radioactivity, and actions are taken as necessary to assure against recurrence. Nasal, saliva, urine and/or fecal samples, or in vivo chest counts may be collected from individuals when action limits are exceeded. Approved procedures define when referral to the corrective action program, as described in Chapter 11, is required.

Bioassay results are interpreted using industry standard methods and consensus models and may make use of incident-specific data and/or an exposed individual's personal characteristics.

4.7.2.3 Air Sampling and Monitoring Program

Air sampling is performed in all areas of the facility where dispersible forms of licensed materials are handled, stored, or processed and where concentrations of airborne radioactive materials could exceed 10% of a DAC. Air sampling is performed to evaluate airborne hazards whenever respiratory protective equipment is used to limit inhalation intakes, to evaluate containment and for posting airborne areas, to identify if an intake occurred and estimate the magnitude, and to provide an early warning of elevated air concentration.

As described in Chapter 1, the DAC is based on ICRP 66 and 68 which assumes an Activity Median Aerodynamic Diameter (AMAD) of 5 micrometers. If appropriate, TRISO-X may elect to adjust ALI and DAC values based on measured particle size data for particles that are larger than 5 AMAD.

Air sampling is performed using fixed-location samplers for basic evaluation of the exposure of workers, personal (lapel) samplers for supportive measurements and special studies, and air monitors for early warning of unexpected releases. Continuous air monitoring (CAMs) is performed where there is a reasonable potential for unintended releases to cause an intake exceeding 40 DAC-hours in a week or less. Placement of air sampling and monitoring equipment is based the purpose of the measurement (estimating worker intakes, verifying confinement of materials is effective, providing warning, detecting leaks, or determining if an airborne radioactivity area exist) and on airflow studies to determine the airflow patterns in the workplace and optimal placement of monitors. Alarm set points for CAMs meet radiation protection objectives, are defensible and documented, and are set as low as practical for the work being conducted without causing excessive false alarms.

Fixed air sampling results, lapel or other special air sampling results may be used to determine worker intake and to calculate CEDE in areas for which internal dose assessment is required. Where air sampling is used to comply with 10 CFR 20.1502(b) for measuring intake, evaluations demonstrate that the sampling is representative of the air breathed by the worker or correction factors applied where appropriate.

Air sampling frequency is performed in accordance with **Table 4-1**, Minimum Survey Frequency. At a minimum, areas posted as Airborne Radioactivity Areas have air samples changed at least once per day when production is ongoing in those areas. Other contaminated areas where licensed materials are handled, but are not Airborne Radioactivity Areas, and where the airborne levels could average greater than 10% DAC, have air samples changed weekly when work with licensed materials is ongoing. Air sampling and monitoring equipment such as continuous air monitors, portable high volume and/or lapel air samplers may be utilized where a fixed air sampling network is not practical.

Quality Assurance/Quality Control measures include periodic calibrations, and daily source and background checks. Airflow meters are calibrated according to manufacturer guidelines at least annually, and whenever repair or modification warrant.

Action levels for air sample results are established and documented in the RPP implementing procedures. Action levels are established associated with investigation, work suspension, and work restriction levels and their associated corrective actions. A measurement indicating an intake equal to or greater than 10 milligrams of Type F uranium in a week will result in a work restriction based on chemical toxicity limits.

4.7.3 Contamination Control

Contamination is controlled at the TRISO-X FFF by containment enclosures and ventilation and supplemented by area classification and access control associated with radiological postings, routine surveillance, protective clothing for personnel, and training.

4.7.3.1 Access Control

All areas that store, handle or process radioactive materials are classified as a Radiologically Controlled Area (RCA). Change facilities are provided for personal clothing, showering, and donning PPE. Areas within the RCAs are further controlled for access and egress as buffer areas, or for levels of radiation, contamination, or airborne contamination in accordance with the posting requirements in 10 CFR 20. Radiological postings inform workers of radiological conditions and requirements for entry/exit. Step-off pads and monitoring instruments are provided at PPE doffing locations.

4.7.3.2 Surveying of Surfaces

A routine contamination control program is established to evaluate and control surface contamination and to prevent unnecessary external or internal exposure of personnel to radiation. Surveys are performed on personnel leaving areas in which contamination could be present, materials and equipment designated for release from controlled areas, and on in-coming and out-going shipments containing radioactive material.

Routine surface contamination control surveys are performed for process and manufacturing areas, warehousing, and support facilities where licensed materials are stored, handled, or processed. Uncontrolled areas inside the facility are also surveyed periodically to ensure that radioactive materials are adequately confined in the RCAs. Minimum survey frequencies are presented in **Table 4-1**. Areas in which the potential for surface contamination is high, or the probability for human intake from resuspension is high, are surveyed more frequently.

Standardized methods for collecting and analyzing smear samples are established and employed to aid in comparison and establish trends for facility and equipment and material surveys. A diagram of each routinely surveyed area is used for recording survey results. Methods and instruments used for counting samples of removable surface contamination can detect alpha radiation from uranium at and below the levels specified in Appendix A, "Acceptable Surface Contamination Levels," to Regulatory Guide 8.24. Regulatory Guide 8.24 Appendix A is incorporated as **Attachment 1**.

The RPP requires that administrative action guidelines be established to assure that contamination levels and employee exposures are kept ALARA and within regulatory limits. Action guidelines are established to ensure appropriate corrective actions are taken for contamination control. Approved procedures detail the action guidelines, required actions to promptly address the contamination to a satisfactory resolution (cleaned or contained and labeled), and when referral to the corrective action program, as described in Chapter 11, is required.

4.7.3.3 Protective Clothing

PPE is required to minimize opportunities for personnel and their clothing to become contaminated. Protective clothing requirements are commensurate with anticipated work conditions. Monitors are available in areas where workers doff PPE to perform a survey (especially head, hands, and other exposed portions of the body) after they remove PPE and before they leave the RCA. Clothing surveyed and found to have less than 200 disintegrations per minute (dpm) per 100 cm^2 of uranium contamination is acceptable outside restricted areas.

Reasonable efforts are made to reduce skin contamination to background levels. Decontamination attempts under the direction of the radiation safety staff or medical consultant are repeated until (1) such attempts cease to achieve significant reductions, or (2) such attempts threaten to damage the skin, at which point the individual will be released from the RCA. Subsequent measurements will be made on the area to evaluate contamination levels over time. Exceptions may be made for emergencies and emergency drills.

4.7.3.4 Release for Unrestricted Use

Equipment, materials, and facilities that are evaluated for both fixed and removable contamination using guidance in "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source or Special Nuclear Material" as described in Chapter 1, and that do not exceed the levels represented in Appendix A, "Acceptable Surface Contamination Levels," to Regulatory Guide 8.24, may be released from restricted areas for unrestricted use (see **Attachment 1**).

Equipment and materials may be transferred between restricted areas through an un-restricted area if the exterior surfaces of the item or its container have smearable contamination levels less than **Attachment 1** levels. When contaminated items are transferred through un-restricted areas, the route will minimize transfer time and the possibility of accidental release.

4.7.3.5 Leak Testing of Sealed Sources

Non-exempt sealed sources used for calibration and quality control procedures are leak tested in accordance with Appendix C, "Leak Test Requirements," of Regulatory Guide 8.24. Regulatory Guide 8.24 Appendix C is incorporated as **Attachment 2**.

4.7.4 Environmental Monitoring

Effluent and environmental radiation monitoring are addressed in Chapter 9 of this application.

4.8 Additional Program Commitments

The following sections provide commitments to achieve compliance with the regulations in 10 CFR 20, Subpart L, 10 CFR 20, Subpart M, *Reports* and 10 CFR 70.74.

4.8.1 Records

In accordance with 10 CFR 20, Subpart L, *Records*, TRISO-X maintains records of the RPP, including but not limited to, occupational exposure of personnel to radiation, releases of radioactive materials to the environment, radiation survey results, and results of corrective action program referrals, radiation work permits and planned special exposures. Records associated with ALARA findings, employee training, personnel radiation exposures, and environmental activities are generated and retained in such a manner as to comply with the relevant requirements of 10 CFR 20 as part of the records management program described in Chapter 11.

4.8.2 Event Reporting

Approved procedures require reporting to the NRC, within the time specified in 10 CFR 20, Subpart M, *Reports*, and 10 CFR 70.74, any event resulting in an occupational exposure to radiation exceeding the dose limits in 10 CFR 20. Approved procedures contain instructions for when and how to report events to the NRC.

4.8.3 Annual Dose Monitoring Report

An annual report of the results of individual monitoring, as required by 10 CFR 20.2206(b), is submitted to the NRC.

4.9 Criticality Monitoring and Detection

The TRISO-X FFF criticality accident alarm system (CAAS) is designed, and a documented evaluation is maintained, to demonstrate that the system meets the requirements of 10 CFR 70.24, as well as ANSI/ANS-8.3-1997 (R2017), *Criticality Accident Alarm System*, with exceptions as noted in Regulatory Guide 3.71, *Nuclear Criticality Safety Standards for Nuclear Materials Outside Reactor Cores*, Revision 3 (2018). Monitoring is performed using radiation detectors (e.g., gamma, neutron) that are proportional to dose levels and not subject to the effects of saturation.

The criticality detection system consists of three essential parts: the readout module, alarm relay module, and the detectors. The detector collects light, or a charge caused by incident radiation. This light or charge is then conditioned and transmitted via multiconductor cable and displayed on the readout module.

A calibration check is performed for all units in service based on detector manufacturer's recommendations or if system monitoring indicates anomalous detector response, however not to exceed 5 years. Detectors are also response tested in accordance with internal procedures to

ensure continued operability. Periodically, the alarm is sounded for familiarity, training, or drills. Alarm audibility (e.g., siren, lights) is tested in accordance with approved procedures.

To meet regulatory requirements in 10 CFR 70.24 and to assure a limited number of false alarms, the system is set up with at least two detectors at each designated monitoring location. Alarm actuation is caused by at least two detectors at a location exceeding their alarm trip point, or by a single detector failure coupled with the second detector in alarm, which results in a plant-wide evacuation and worker accountability. The same logic would apply to a system with three detectors, with two of three alarming. Detector or other electronic component failure will result in a warning signal. This signal will initiate contingency measures which may include evacuation of personnel, suspension of operations, deployment of auxiliary monitoring equipment, and/or immediate system repair. The system is also provided with a backup power supply.

The placement of the radiation detectors are such that all areas of the facility where monitoring is required are covered. Typically, the alarm trip point is set at 20 mR/hr. Alarm set points may vary due to ambient radiation levels and measurement uncertainty. This trip point allows for minimization of an alarm from sources other than criticality. When the alarm trip point has been reached or exceeded, the system will produce an alarm throughout the facility which will continue regardless of the radiation level until manually reset. The alarm control and relay cabinets have limited access. Manual initiation of the alarm system is provided for testing. A warning signal is generated at the central control unit in the event of a detection system malfunction. Provisions are incorporated into the alarm system to allow appropriate testing and remote readouts are present at the central/secondary alarm stations that will alert personnel in the event of component failure.

The system is demonstrated to respond to a minimum criticality accident of concern. A criticality accident producing an absorbed dose in air of 20 rads at 2 meters within one minute is the limiting accident considered for the demonstration of the system response.

Detector placement is conservatively demonstrated by accounting for shielding from facility and process materials between a postulated accident and the detectors, as well as distance. The accident is evaluated from a number of locations to demonstrate the possible effects of attenuation. Common modeling codes are used to perform the evaluations (e.g., MCNP). Compliance is demonstrated if modeling results indicate that the postulated minimum accident of concern will result in an exposure rate exceeding the alarm set-point at a detector location.

Table 4-1: Minimum Survey Frequencies

FACILITY AREAS	EXTERNAL RADIATION SURVEYS	AIR SAMPLING	REMOVABLE SURFACE CONTAMINATION SURVEYS
Uranium receiving, warehousing, and shipping	Monthly	Continuous air sampling; samples changed weekly and following any indication of release leading to airborne concentrations of uranium	Monthly and following any indication of release
Manufacturing processing areas, scrap recovery, waste handling, maintenance, and change rooms	Monthly	Continuous air sampling; samples changed each shift, following any change in equipment or process control, and following detection of any event that may have released uranium	Weekly and following any indication of release
Laboratories	Monthly	Continuous air sampling; samples changed each shift	Weekly
Final fuel form packaging and storage	Monthly	Continuous sampling; samples changed weekly	Monthly
Lunchrooms, cafeterias, snack bars, and vending machine areas	Quarterly	None	Monthly

Table 4-2: Summary of 10 CFR 20.1502 Monitoring Requirements

The use of individual monitoring devices for external dose is required:

- For adults who are likely to receive an annual dose in excess of any of the following (each evaluated separately):
  - 0.5 rem (0.005 Sv) deep-dose equivalent.
  - 1.5 rems (0.015 Sv) eye dose equivalent.
  - 5 rems (0.05 Sv) shallow-dose equivalent to the skin.
  - 5 rems (0.05 Sv) shallow-dose equivalent to any extremity.
- For minors who are likely to receive an annual dose in excess of any of the following (each evaluated separately):
  - 0.05 rem (0.5 mSv) deep-dose equivalent.
  - 0.15 rem (1.5 mSv) eye dose equivalent.
  - 0.5 rem (0.005 Sv) shallow-dose equivalent to the skin.
  - 0.5 rem (0.005 Sv) shallow-dose equivalent to any extremity.
- For declared pregnant women who are likely to receive an annual dose from occupational exposure in excess of 0.05 rem deep-dose equivalent, although the dose limit applies to the entire gestation period.
- Individuals entering a high or a very high radiation area.

Internal exposure monitoring (not necessarily individual monitoring devices) is required:

- For adults likely to receive in 1 year an intake in excess of 10% of the applicable ALIs for ingestion and inhalation.
- For minors and declared pregnant women likely to receive in 1 year a committed effective dose equivalent in excess of 0.05 rem (0.5 mSv).

Table 4-3: Routine Minimum Bioassay Frequencies

from ANSI/HPS N13.22-2013 Bioassay Programs for Uranium, Table 8 Minimum Frequencies of Bioassay

10 CFR 20 Solubility Class	Situation	Urine	In-vivo
D	Radiological	Monthly	Not normally used
W	Radiological	Quarterly	Annually
Special Y	Radiological	Quarterly	Annually
Y	Radiological	Not normally used	Annually
D and W	Chemical Toxicity	Monthly	Annually for Class W

Fecal bioassay frequency is quarterly for Class Y, or as needed for all Classes to address special situations such as incident / accident investigations.

Attachment 1

APPENDIX A

ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDES [a]	AVERAGE [b, c, e, f]	MAXIMUM [b, d, e, f]	REMOVABLE [b, c, e, f]
U-nat, U-235, U-238, and associated decay products	5,000 dpm α/100 cm^2	15,000 dpm α/100 cm^2	1,000 dpm α/100 cm^2
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, and I-129	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, and I-133	1,000 dpm/100 cm^2	3,000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-gamma (βγ) emitters (nuclides with decay modes other than alpha (α) emission or spontaneous fission) except Sr-90 and others noted above	5,000 dpm βγ/100 cm^2	15,000 dpm βγ/100 cm^2	1,000 dpm βγ/100 cm^2

a Where surface contamination by both α- and βγ-emitting nuclides exists, the limits established for α- and βγ-emitting nuclides should apply independently.

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

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

d The maximum contamination level applies to an area of not more than 100 square centimeter (cm^2).

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

f The average and maximum radiation levels associated with surface contamination resulting from βγ emitters should not exceed 0.2 millirad per hour (mrad/h) at 1 centimeter (cm) and 1.0 mrad/h at 1 cm, respectively, measured through not more than 7 milligrams per cm^2 of total absorber.

Attachment 2

APPENDIX C

LEAK TEST REQUIREMENTS

A. Each source shall be tested for leakage at intervals not to exceed 6 months. In the absence of a certificate from a transferor indicating that a test has been made within 6 months before transfer, the sealed source shall not be put into use until tested.

B. The test shall be capable of detecting the presence of 0.005 microcuries of contamination on the test sample. The test sample shall be taken from the source or from appropriate accessible surfaces of the device in which the sealed source is permanently or semipermanently mounted or stored. Records of leak test results shall be kept in units of microcuries and maintained for inspection by the Commission.

C. If the test reveals the presence of 0.005 microcuries or more of removable contamination, the licensee shall immediately withdraw the sealed source from use and shall cause it to be decontaminated and repaired by a person appropriately licensed to make such repairs or to be disposed of in accordance with the Commission's regulations. Within 5 days after determining that any source has leaked, the licensee shall file a report with the Director, Division of Fuel Cycle Safety and Safeguards, U.S. Nuclear Regulatory Commission (NRC), Washington, DC 20555. This report shall describe the source, test results, extent of contamination, apparent or suspected cause of source failure, and corrective action taken. A copy of the report shall be sent to the administrator of the nearest NRC regional office listed in Appendix D, "United States Nuclear Regulatory Commission Regional Offices," to 10 CFR Part 20, "Standards for Protection Against Radiation."

D. The periodic leak test required by this condition does not apply to sealed sources that are stored and not being used. The sources excepted from this test shall be tested for leakage before any use or transfer to another person unless they have been tested for leaks within 6 months before the date of use or transfer.

NRC SPECIAL NUCLEAR MATERIAL LICENSE
SNM-XXXX
CHAPTER 5

NUCLEAR CRITICALITY SAFETY
Table of Contents

NUCLEAR CRITICALITY SAFETY

5.1 Nuclear Criticality Safety Program and Philosophy

TRISO-X provides an effective nuclear criticality safety (NCS) program, including methodologies and technical practices, to support safe operation of the TRISO-X FFF. Controls and barriers that are designated as Items Relied on for Safety (IROFS) to prevent an inadvertent nuclear criticality are documented in NCS Evaluations (NCSEs) or in supporting Risk Assessment Evaluations for the respective NCSE and the Integrated Safety Analysis (ISA) Summary as appropriate.

TRISO-X provides for the appropriate management of the NCS program. The responsibilities and authorities of individuals that develop and implement the NCS program are also provided in Chapter 2. In addition, facility management measures are provided that support implementation and maintenance of the NCS program.

Subcriticality is maintained for all normal and credible abnormal conditions as required by 10 CFR 70.61(d). The Double Contingency Principle of ANSI/ANS-8.1-2014 (R2018), *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors,* formulates the preferred basis for the design and operation of special nuclear material (SNM) processes within the TRISO-X FFF. To support this overarching requirement, process designs incorporate sufficient factors of safety and controls provide sufficient redundancy and diversity to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible. The focus should be on understanding each credible change in process conditions and implementing the best overall control strategy to maintain subcriticality such that no single credible event or failure will result in a criticality accident. All credible events in which criticality is possible will be documented to be highly unlikely as required by 10 CFR 70.61(b). The NCSE will also document the basis that a change in a process condition is unlikely. When considering NCS accident sequences, guidance from Appendix A, ANSI/ANS-8.1-2014 (R2018), is used to support the evaluation of typical changes in process conditions.

TRISO-X FFF relies on passive, active, enhanced administrative, and administrative controls to maintain subcriticality. Where practicable, reliance is placed on passive equipment designs to maintain subcriticality rather than on administrative controls.

5.2 Organization and Administration of the NCS Program

TRISO-X provides an NCS program with sufficient resources to implement and maintain an effective program. The TRISO-X NCS program meets the regulatory requirements of 10 CFR 70 to ensure adequate protection against the consequences of accidental criticality events. The primary means of doing this is prevention by ensuring that processes remain subcritical under normal and credible abnormal conditions.

5.2.1 NCS Program Objectives

The TRISO-X NCS program includes the following objectives:

1. Perform and document NCS evaluations for new or changed processes and establish safety limits, controls, and procedures as necessary to ensure that processes will remain subcritical under normal and credible abnormal conditions.
2. Establish, as practicable, double-contingency protection and defense-in-depth measures; ensuring sufficient margins of safety and subcriticality to provide additional assurance that the likelihood of criticality will be acceptably low.
3. Establish and maintain a Criticality Accident Alarm System (CAAS) and emergency-response procedures to protect health and safety in the event criticality occurs.
4. Provide technical support to emergency response personnel in responding to and recovering from abnormal conditions and emergencies up to and including a criticality accident.
5. Verify the adequacy of NCS controls through audits and assessments of operations including verification of equipment configuration(s).
6. Ensure adequacy of NCS evaluations through peer reviews, assessments, and validation and verification of calculational methods.
7. Train and otherwise support operations in procedure development and implementation to ensure the safe handling of special nuclear material.
8. Support regulatory compliance regarding event reporting (10 CFR 70.50 and Appendix A to 10 CFR 70), comply with the facility change process (10 CFR 70.72), and participate in the performance and documentation of the facility's ISA (10 CFR 70.61 through 70.66) insofar as they pertain to criticality safety.

5.2.2 NCS Program Commitments

The TRISO-X NCS program shall meet the requirements of 10 CFR 70. The objectives of the program include:

1. An NCS program structure that is consistent with current industry practice as defined in ANSI/ANS-8.1-2014 (R2018) and ANSI/ANS-8.19-2014 (R2019), *Administrative Practices for Nuclear Criticality Safety*, including establishing the roles and responsibilities of key program personnel. Information regarding TRISO-X organization and administration is described in Chapter 2. Chapter 2 also includes the organizational positions, functional responsibilities, qualifications, and authorities of NCS management and staff who develop, organize, implement, and administer the NCS program.
2. Establish and maintain NCS safety limits, controls, procedures, and any NCS operating limits established for IROFS in fissile material processes and maintain management measures to ensure their continued reliability and availability to perform their intended safety function when required.
3. Support operations personnel through development of training, preparation of NCS postings, and other appropriate operator aids for key administrative controls (e.g., painted lines on the floor and warning lights), and review of procedures and operations to ensure they are unambiguous, easily understood, and readily achievable.

4. Evaluate modifications to the facility or safety program to ascertain their impact on criticality safety.
5. Require personnel to report defective NCS conditions to operations supervision and the NCS function. Management policies reinforce operators' stop-work authority and encourage the reporting of defective conditions.
6. Render processes safe or suspend operations upon loss of double contingency protection until such protection can be restored and assess the adequacy of the affected controls.

5.3 Management Measures Applied to the NCS Program

5.3.1 General Management Measures

Information regarding management measures programs is described in Chapter 11. These programs include the management measures identified in 10 CFR 70.62 used to implement and maintain the NCS program.

5.3.2 Employee Training

TRISO-X complies with the requirements of ANSI/ANS-8.19-2014 (R2019), and Section 7 of ANSI/ANS-8.20-1991 (R2020), *Nuclear Criticality Safety Training*, as they relate to NCS training. Training is provided to all personnel to recognize the Criticality Accident Alarm System (CAAS) signal and to evacuate promptly to a safe area. In addition, TRISO-X employees receive instruction training regarding the NCS Policy. Refer to Chapter 11 for additional discussion about training personnel regarding procedural compliance, stop-work authority, response to alarms, and reporting of defective conditions.

5.3.3 Training and Qualifications of NCS Staff

A formal training and qualification program is implemented and maintained for NCS staff consistent with guidelines presented in ANSI/ANS-8.26-2007 (R2016), *Criticality Safety Engineer Training and Qualification Program*, excluding Section 7.4. Elements of the program include on-the-job training, off-site NCS-related training courses, and mentoring by senior NCS engineers.

5.3.4 Auditing, Assessing, and Upgrading the NCS Program

TRISO-X complies with the requirements of ANSI/ANS-8.19-2014 (R2019) as it relates to NCS audits and assessments. Audits and assessments of site operations involving SNM are performed on a scheduled basis as defined in written approved procedures to confirm that activities are being conducted in accordance with nuclear criticality safety requirements including limits and controls.

As part of the audits and assessments program described in Chapter 11, NCS audits and assessments of facility activities are conducted on a scheduled basis as defined in approved procedures such that SNM processing or storage areas / facilities, evaluated under the scope of an NCSE, are reviewed at least triennially. The purpose of an audit is to determine that: (a) operations are conducted in compliance with license conditions, operating procedures, and

posted limits; (b) administrative controls and postings are consistent with NCSEs; (c) equipment and operations comply with NCSEs; and (d) corrective actions relative to findings of NCS inspections are adequate. The audit frequency for an area is based on its respective NCSE implementation date, and subsequent revisions will extend that audit requirement into its next cycle. The purpose of an audit is focused on compliance whereas an assessment is focused on improvement efforts. Audits may be performed under a formal QA program, as directed by that function, or under the scope of the nuclear criticality safety program.

Assessments are conducted to evaluate whether the nuclear criticality safety program meets the requirements of related ANSI/ANS-8 series standards. Facility walkthrough assessments are performed periodically by NCS for each of the SNM operations. Facility walkthrough assessments focus on field compliance with established NCS limits and controls. Assessment frequencies are based on the perceived criticality safety risk of an operation and may be performed more frequently for higher risk operations. The assessment program supports the NCS audit schedule by identifying problematic SNM operations that should be reviewed more frequently during the triennial period.

An independent assessment of the nuclear criticality safety program is conducted at least triennially.

Findings and observations from NCS audits and assessments are entered into the corrective action program and tracked until closure as described Chapter 11.

5.3.5 Procedures

TRISO-X commits to the requirements of ANSI/ANS-8.19-2014 (R2019) as it relates to operating procedures. As described in Chapter 11, procedures are provided for activities involving SNM, and the procedures incorporate safety limits and controls as appropriate. These procedures are reviewed and approved by the NCS function. During the review and approval process, the NCS staff may recommend or require modifications to the procedures to reduce the likelihood of occurrence of an inadvertent nuclear criticality.

5.3.6 NCS Reviews of New or Modified Equipment

Each proposed addition of new equipment or change to existing equipment used in the processing or storage of SNM, and any procedure changes resulting therefrom, is reviewed and approved by the NCS function as part of the ISA and Configuration Management processes described in Chapters 3 and 11. During the review and approval process, the NCS staff may recommend or require modifications to the design and/or to the procedures to reduce the likelihood of occurrence of an inadvertent nuclear criticality. NCS calculations and evaluations are incorporated into the configuration management program.

5.3.7 Posting of Nuclear Criticality Safety Limits

NCS requirements issued by the NCS function for each process system are made available to work areas in the form of written or electronic operating procedures. Clear, visible signs or notices may be posted at workstations or floor areas may be marked, as appropriate, to supplement the procedures by emphasizing specific limits and controls.

Posted nuclear criticality safety requirements and limits are defined by the nuclear criticality safety function and include, as appropriate:

1. Limits on material types and forms;
2. Allowable quantities by mass or number of items/containers;
3. Allowable enrichments;
4. Limits on reflecting materials;
5. Required spacing between units;
6. Control limits, when applicable, on quantities such as moderation, concentration/density, and the presence of additives.

5.3.8 ISA Process and ISA Summary

Refer to Chapter 3 for a discussion of the ISA process, including process hazard analyses and ISA Summary revisions.

5.3.9 Corrective Action Program

A corrective action program is implemented to document and manage NCS-related problems, observations, findings, investigations, corrective actions, and any unacceptable NCS-related performance deficiencies. Refer to Chapter 11 for a discussion of the corrective action program.

5.3.10 Records Management

Records of the NCS program are retained in accordance with the regulatory retention program. These records include NCS calculations and evaluations and documentation of corrective actions taken. Refer to Chapter 11 for a discussion of the records management program.

5.4 Emergency Notification, Planning, and Response

5.4.1 Criticality Accident Alarm System (CAAS)

A criticality accident alarm system (CAAS) is designed and installed to provide prompt detection and annunciation of an inadvertent nuclear criticality. The system utilizes an audible (e.g., siren) and/or visual signal (e.g., building entry and high-noise areas) to alert personnel in the area to evacuate and deter personnel from entering after an evacuation. The system is designed, and a documented evaluation is maintained, to demonstrate that the CAAS meets the requirements of 10 CFR 70.24, as well as ANSI/ANS-8.3-1997 (R2017), *Criticality Accident Alarm System*, with exceptions as noted in Regulatory Guide 3.71, *Nuclear Criticality Safety Standards for Nuclear Materials Outside Reactor Cores*, Revision 3.

The system is designed to remain operational during credible adverse events, and the system will alarm during credible failure modes. In the event of loss of normal power, emergency power is automatically supplied to the CAAS. Refer to Chapter 4 for a description of the CAAS.

If the CAAS is out of service, coverage is lost for an SNM operation, or system components are being tested or repaired, compensatory measures are established to ensure prompt personnel evacuation as documented in a written approved procedure. Compensatory measures may include suspended or limited movement of SNM in the affected area, limited personnel access, use of temporary detection equipment for personnel that must access the area during a CAAS outage, and/or monitoring of the criticality alarm panel. Compensatory measures and required evacuation procedures are approved and documented.

Employees and visitors are trained in responding to the CAAS annunciation. An ongoing aspect of this training is a quarterly annunciation on all shifts in which SNM operations are being conducted.

The need for CAAS coverage shall be evaluated for all activities in which the inventory of fissile materials in individual unrelated areas exceeds 700 g ^{235U} [ANSI/ANS-8.3-1997 (R2017)]. CAAS coverage of an SNM operation is documented during the process for NCS evaluation. In addition, any exceptions to CAAS coverage are also documented in NCS evaluations and are based on a conclusion in the NCSE that a criticality accident is non-credible specific to the area in which conduct of the operation is approved. Conclusions regarding non-credibility based entirely on fissile material present require at a minimum that the inventory of fissile material in the area is less than 700 g ^{235}U.

CAAS coverage is provided for SNM operations, except as specified in Chapter 1. In addition to the exemptions included in Chapter 1, CAAS coverage is not required for 1) areas where an evaluation has determined the risk of criticality is very low due to the amount or configuration of fissile material present, 2) materials and/or containers that satisfy the fissile material exceptions in 49 CFR 173 and/or 10 CFR 71, or 3) areas that contain less than 700 g ^{235}U. Areas that do not contain SNM operations do not require an NCSE and do not require CAAS coverage.

5.4.2 Portable CAAS

In the event an SNM operation requiring CAAS coverage is performed beyond the detection range of the established CAAS or CAAS coverage for an SNM operation is determined to not be available, a portable unit may be used to provide coverage. The portable unit has equivalent detection capabilities as the permanently installed units, however alarm annunciation is limited to the immediate area. Compensatory measures (e.g., use of facility public address system, and/or radio communication) are established to ensure prompt evacuation of personnel as documented in an approved procedure.

5.4.3 Emergency Management

Refer to Chapter 8 for a discussion of the emergency management program and emergency plan. Refer to Chapter 4 for a discussion of accident dosimetry. Guidance from ANSI/ANS-8.23-2019, *Nuclear Criticality Accident Emergency Planning and Response*, is also used for nuclear criticality accident emergency planning and response.

5.5 Methodologies and Technical Practices

5.5.1 Means of Control

The relative effectiveness and reliability of controls are considered during the nuclear criticality safety analysis process. Engineered controls or design features are preferred over administrative controls. Passive engineered controls or design features are preferred over all other system controls and are utilized when practicable and appropriate. Active engineered controls are the next preferred method of control. Administrative controls are the least preferred method of control; however, when administrative controls are deemed necessary, enhanced administrative controls are preferred over administrative controls.

1. Passive engineered controls (most preferred) use fixed design features or devices to maintain safe process conditions. No human intervention or action is required. Assurance is maintained through initial verification prior to operation. Periodic inspections are performed on those systems where credible changes in equipment dimensions may occur as determined by nuclear criticality safety evaluations. Assurance is also maintained through the configuration management program.

2. Active engineered controls use add-on, active hardware (e.g., electrical, mechanical), or moving parts to maintain safe process conditions. No human intervention or action is required during operation. Assurance is maintained through initial and periodic inspection, functional testing, and/or calibration. Active engineered controls detect an undesirable change in process conditions and automatically secure the system to a safe condition.

3. Enhanced administrative controls rely on human judgment, training, and personal responsibility for implementation and are augmented by warning devices (visual or audible) which require human action according to procedure. A visual or audible alarm alerts the operator to an undesirable change in process conditions, which requires human action or intervention in accordance with approved procedures to maintain or return the process to a safe condition. Alarm integrity and reliability is ensured by initial and periodic inspection or functional testing.

4. Administrative controls (least preferred) rely on human judgment, training, and personal responsibility for implementation when the control function is needed. The control is a

procedural human action that is required to maintain safe process conditions. Assurance is maintained through periodic verification, audit, or training.

5.5.2 Methods of Control

The following recognized control methods are also referred to as parameters, which may be controlled for nuclear criticality safety purposes (i.e., controlled parameters). When evaluating an SNM-bearing system for criticality safety, the procedure for preparation of an NCSE requires that each of the controlled parameters will be assumed to be at their optimum credible condition (i.e., most reactive credible condition) unless controls, including passive design requirements, are specified, and implemented to limit the parameters to certain values. When computer codes are used to determine the safety of a system, the values meet the k_{eff} limits of this chapter. Criticality safety may also be based on data or methods provided in industry accepted handbooks, reference documents, and/or experimental data. Maximum subcritical values may be used as provided in handbooks or standards (e.g., maximum subcritical values provided in ANSI/ANS-8.1).

1. Geometry – Geometry control is achieved by increasing neutron leakage by limiting the dimensions of defined geometrical shapes. Equipment relying upon favorable geometry for control include adequate factors of safety to ensure reliability under credible accident conditions. Before beginning an operation, all dimensions relied upon for geometry control are verified. The facility configuration management program is used to maintain these dimensions. Periodic inspections are performed on those systems where credible changes in equipment dimensions may occur as determined by nuclear criticality safety evaluations.

2. Spacing (or Unit Interaction) – Spacing (or Unit Interaction) control is a method of limiting the introduction of neutrons leaked from one SNM unit into a neighboring SNM unit by controlling the separation distance between units. Where spacing control is required, a passive engineered device (e.g., a spacer, fixed bumper, or permanent floor mounting) is the preferred method of control and is used where practicable. The structural integrity of any spacers/racks should be sufficient for normal and credible abnormal conditions. If not practicable, administrative controls may be utilized and should include such items as procedural instructions, postings, and visual indicators, as appropriate.
    a. Equipment, facilities, and individually subcritical units are effectively non-interacting or neutronically isolated when their surfaces are separated by any of the following:
        i. A 12-inch-thick layer of water, or
        ii. A thickness equivalent to a 12-inch thick layer of water, or
        iii. 12 feet of air, or
        iv. Sub-arrays separated by not less than the smallest dimension of the facing surfaces of the sub-arrays, or

v. The greatest distance across an orthographic projection of the largest of the fissile accumulations on a plane perpendicular to the line joining their centers, or
vi. 12 inches of solid concrete (block or poured).

b. The design conditions for interaction between multiple units or between arrays that experience neutron interaction will be based on values that can be demonstrated safe by one of the following methods:
   i. Solid Angle Method
   ii. Surface Density Method
   iii. Areal Density – When criticality safety is contingent only upon maintenance of a limited areal density of fissile material, controls will be implemented to ensure that the limit is not exceeded. The controls will limit the areal density to a safe value, which is defined to be no more than 45 percent of the minimum critical areal density.
   iv. Monte-Carlo Calculations (Each application of Monte-Carlo calculations must comply with the requirements of this chapter).
   v. American National Standard, ANSI/ANS-8.7-1998 (R2017), *Nuclear Criticality Safety in the Storage of Fissile Materials*
   vi. NRC and/or DOT packaging or transportation regulations (e.g., staging of packages in accordance with the Criticality Safety Index)

3. Volume – Volume control is a method of limiting the volume of SNM to an acceptable value. Equipment relied upon for volume control includes adequate factors of safety to ensure that a safe volume is maintained under credible accident conditions. Prior to the equipment being released for use, the volume of the equipment is verified. The facility configuration management program is used to maintain the volume. Periodic inspections are performed on those systems where credible changes in equipment volume may occur as determined by nuclear criticality safety evaluations. When the solution volume is measured (i.e., quantity of solution), appropriate instrumentation is used.

4. Neutron Absorber (Fixed/Soluble) – Neutron absorber control is a method of reducing the number of neutrons in a fissile material system available to cause a fission event, by introducing a parasitic neutron absorber (i.e., neutron poison) into the system. This method of control includes use of fixed or soluble neutron absorbers. When evaluating absorber effectiveness, neutron spectra are considered (e.g., cadmium is an effective absorber for thermal neutrons, but ineffective for fast neutrons).
   Fixed neutron absorber control is a method of increasing neutron absorption in material by placing a solid absorber (i.e., poison) in the system that may include the use of "poison fixtures" as well as taking credit for the neutron absorption properties of structural materials. For fixed neutron absorbers, the thickness of the absorber is measured and documented prior to first use. The composition of the absorber will be verified unless the chemical properties of the materials consist of standard structural materials (e.g.,

stainless steel, carbon steel, etc.). Controls, as necessary, are exercised to maintain the continued presence and the intended distribution and contribution of the absorber.
If fixed neutron absorbers are used, the requirements of ANSI/ANS-8.21-1995 (R2011), *Use Of Fixed Neutron Absorbers In Nuclear Facilities Outside Reactors*, are applied. If borosilicate glass Raschig rings are used, the requirements of ANSI/ANS-8.5-1996 (R2017), *Use Of Borosilicate-Glass Raschig Rings as a Neutron Absorber In Solutions Of Fissile Material*, are applied. If soluble neutron absorbers are used, the requirements of ANSI/ANS-8.14-2004 (R2016), *Use Of Soluble Neutron Absorbers In Nuclear Facilities Outside Reactors*, are applied. Fixed neutron absorber, including a moderator, use is prevalent in fissile material shipping containers and further subject to the requirements as specified in the package NRC certificate of compliance for use in the TRISO-X FFF.

5. Piece Count – Piece count is a method of limiting fissile material mass and/or geometry by limiting the number of containers and/or components with known amounts of SNM and/or fixed geometries at a given location.

6. Mass – Mass control is a method of limiting the amount of SNM at a given location to an acceptable value. Mass control may be used on its own or in combination with other control methods. When a given mass of material has been determined, a percentage factor is used to determine the mass percentage of SNM in that material. When fixed geometric devices are used to limit the mass of SNM, a conservative process density is used. When overbatching of SNM is credible, the largest mass resulting from a single failure is shown to be subcritical. Overbatching beyond double batching should be considered unless it requires multiple independent failures or is precluded by equipment capacity, availability of material, or other considerations. When the mass is measured, appropriate instrumentation is used (e.g., scales, non-destructive assay equipment).

7. Moderation – Moderation control is a method of limiting or excluding either interstitial (i.e., within the SNM) or interspersed (i.e., between SNM units) moderating materials or both. TRISO-X complies with the requirements of ANSI/ANS-8.22-1997 (R2016), *Nuclear Criticality Safety Based On Limiting And Controlling Moderators*, as it relates to limiting and controlling moderators. Nuclear criticality safety based on control of moderation requires that sources of moderation be identified and controlled. When designing physical structures for moderation control, the design should preclude the ingress of moderation. When developing firefighting procedures for use in a moderation-controlled area, restrictions should be placed on the use of moderator material. After evaluating all credible sources of moderation for the potential for intrusion into a moderation-controlled area/workstation, the ingress of moderation is precluded or controlled. When moderation is measured, the measurement is obtained by using instrumentation, calculation, or by visual inspection as appropriate. In addition, when the NCS organization requires samples to be taken and analyzed by the laboratory (e.g., wt. % H_2O) to

determine compliance with moderation limits, dual independent sampling methods will be employed. Shipper information may also be used as a basis for moderation content.

8. Concentration – Concentration control is a method of measuring and controlling the concentration of SNM in hydrogenous liquids to an acceptable value. When concentration control is utilized in an unfavorable geometry system, the concentration is determined by appropriate sampling and analysis techniques (e.g., dual independent sampling) and by instrumentation which has been properly maintained and calibrated (e.g., in-line monitor) prior to transfer to the unfavorable geometry system. The analysis will consider the solubility limits of the SNM composition and possible concentrating mechanisms (e.g., precipitation, evaporation, freezing, settling, chemical phase change) and controls are established, as necessary, to prevent or provide margin for such mechanisms. When a tank containing concentration-controlled solution is used, the tank is maintained closed to prevent unauthorized introduction of precipitating agents which could change the concentration.

9. Material Composition – Material composition (e.g., material type, density, heterogeneity, etc.) control is based on consideration of the physical, chemical, and/or nuclear properties of a material such that the ^{235}U density and neutron scattering/absorption of other materials within the compound are identified and understood (e.g., metal versus oxide versus nitrate, etc.). Manufacturing variability and measurement uncertainty are considered when using material specification as a method of control. Possible misidentification is considered for feed materials when using the feed material specification as control. Heterogeneous effects are only relevant for low-enriched (less than 6.0 wt. %) uranium processes, where all other parameters being equal, heterogeneous systems are more reactive than homogeneous systems, however this trend is not prevalent at 20 wt. % and the homogeneous modeled system is bounding for enrichments greater than 20 wt. %. With regard to density, when the density is measured, the measurement is obtained by the use of appropriate instrumentation.

10. Enrichment – Enrichment control utilizes the inherent differences in critical attributes (critical dimensions, mass, etc.) of uranium at different enrichments of ^{235}U. A method of segregating enrichments is used to ensure differing enrichments will not be interchanged, or else the most limiting enrichment is applied to all SNM. When the enrichment needs to be measured, the measurement is obtained by using appropriate instrumentation (e.g., lab analysis, non-destructive assay equipment).

11. Reflection – Reflection control is a method of control which limits neutron return to an SNM-bearing system. Refer to the following reflection requirements:

a. General Reflection Requirements

Conservative reflection conditions are established when evaluating the criticality safety of individual units or arrays. If reflection conditions are uncontrolled, the maximum credible amount of water reflection (as established below) is considered when calculating system reactivity. Under certain conditions, however, materials such as concrete, carbon, and polyethylene may be more effective reflectors than water. The thickness and location of these types of reflectors are considered if any are present in thickness or quantity that exceeds the equivalent water reflection modeled. If it is credible for reflection conditions to exceed those used in the analysis of system reactivity, then reflection controls are implemented to maintain conditions to within the bounds of the analysis. Where positive barriers are used to maintain reflection control, the barriers are maintained through the configuration management and maintenance programs. In all cases, the impact of standard reflection conditions (e.g., the maximum credible amount of water reflection presented by adjacent water bodies) should be evaluated to determine if the presence of standard reflector conditions reduces system reactivity by decreasing interaction.

b. Single Portable Units

A single unit (e.g., container) is shown to be subcritical when reflected by at least 30 cm of close-fitting water unless:

i. reflector(s) more effective than water are within 30 cm of the unit, or
ii. where 30 cm of close-fitting water reflection is not credible, or
iii. reflection controls are implemented to maintain conditions to within the bounds of the analysis.

When reflectors more effective than water are within 30 cm of a single unit, the thickness and location of these reflectors are considered in the modeled configuration. When a reflector is offset from a single unit, subcriticality will be demonstrated for the thickness and material of the reflector at no more than the offset distance. Subcriticality for single unit reflection may be demonstrated with calculations or by reference to documented subcritical values (e.g., maximum subcritical values provided in ANSI/ANS-8.1).

c. Multiple Portable Containers

Unless controls are implemented to limit reflection, evaluation of multiple portable units (e.g., multiple containers) must be shown to be subcritical with at least a 15 cm thick, 183 cm tall, and 38 cm wide close-fitting water cuboid, representing a nearby operator, on one side of the containers, and at least 1.27 cm of tangential water reflection on the remaining vertical sides of the containers, representing operator hand contact. These reflectors are modeled as slabs or a

box that is tangential to the group of containers, or as a tangential annulus around the group of containers.

d. Enclosures/Gloveboxes

Subcriticality of enclosures is demonstrated with the accessible outer front, back, and sides of the enclosures reflected by at least 2.5 cm of close-fitting water. In addition, a minimum of at least two water cuboids (each cuboid consisting of a 15 cm thick, 183 cm tall, and 38 cm wide or equivalent block) representing nearby operators should be placed near components(s) in the system such that system reactivity is expected to be maximized. The thickness and location of reflectors that may be more effective than water (e.g., concrete, carbon, and polyethylene) should be evaluated in the model if they are located within 60 cm from the front, back, sides, and top of the enclosure or within 90 cm from the bottom of the enclosure.

Overall enclosure reactivity is calculated with portable containers (e.g., containers) located in positions expected to maximize system reactivity, considering any fixed spacing controls that may be present in the enclosure. At a minimum, at least one unit in the glovebox should be evaluated with vertical reflection provided by hands (1.27 cm thick water) in the glovebox (e.g., portable containers, equipment).

e. Arrays of Items

Arrays (one or more units spaced in one or more dimensions) of interacting items (e.g., columns, storage racks) are demonstrated to be subcritical using a minimum of at least two water cuboids (each cuboid consisting of a 15 cm thick, 183 cm tall, and 38 cm wide or an equivalent water body, sized to maximize reflection, that is at least 15 cm thick) representing nearby operators, should be placed near components(s) in the system such that system reactivity is expected to be maximized. The thickness and location of reflectors that may be more effective than water (e.g., concrete, carbon, and polyethylene) are also considered. The density of water interspersed between units within the array is varied from 0.0 to 0.01 grams per cubic centimeter, to bound conditions that may exist during fire sprinkler activation or up to full density water if full flooding is credible. Local reflection from thin water films on fissile items that might accumulate during sprinkler activation shall be conservatively represented by placing a 0.35-cm thick full density water layer on vertical surfaces and a 0.7-cm thick full density water layer on horizontal surfaces.

5.5.3 Computer Codes and Associated Safety Limits

Computer Codes

Computer codes may be used to calculate system reactivity (i.e., k_{eff}). Procedures for computer code validation for the TRISO-X FFF incorporate applicable requirements from ANSI/ANS-8.24-2017, *Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations*, as it relates to the validation process for computer codes, with exceptions and qualifications as noted in NRC Regulatory Guide 3.71, October 2018.

Nuclear criticality safety calculations are performed using approved and validated computer codes such as SCALE, MCNP, XSDRN, etc. The requirements for the verification and validation of computer codes are defined in approved procedures and incorporate the conservatisms as discussed herein. The validation report for a computer code is documented in a written approved report. The required content for a validation report is further discussed.

Validation establishes the suitability of the computer code system (hardware and software) for use in criticality safety evaluations by establishing the calculation margin, a margin of subcriticality, and the Upper Subcritical Limit (USL). The calculation margin is established by simulating benchmark experiments to estimate the systematic difference between the calculated multiplication factor (k_{eff}), and the measured k_{eff} of the benchmark experiments. The systematic difference is the code bias. Bias and bias uncertainty of the computer code system are included in the calculational margin. The calculational margin incorporates appropriate statistical methods that account for the behavior of bias and bias uncertainty (e.g., by considering trends in bias) throughout the range of parameters included in the collection of benchmark experiments. The area of applicability of the validation is a subset of the range of parameters included in the benchmark experiments. The calculational margin is based on widely accepted statistical methods such as the 95 percent confidence limit, the 95/95 lower tolerance limit, or a rank-and-percentile non-parametric method, applied to various subsets of the chosen benchmarks. The statistical method and subset of benchmarks that produce the largest calculation margin, as documented in the validation report, is used in determining the bounding USL.

Computer codes are validated to ensure that they calculate within acceptable ranges and that the assumptions are appropriate. The applicability of computer code validation is documented in nuclear criticality safety calculations and/or evaluations for the system being evaluated. The computer code validation reports are incorporated into the configuration management program.

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

a. Description of the methodology that is sufficiently detailed and clear to allow independent duplication of results, including the method used to select benchmark experiments, to determine the bias and bias uncertainty, and to determine the Upper Subcritical Limit.
b. Summary of the physical systems and area(s) of applicability that identifies the range of values for which valid results have been obtained.
c. Description of the methods used to extend the area(s) of applicability beyond the range of values covered by the benchmark experiments. Any such extension should be done by making use of trends in the bias, including accounting for any increased uncertainty due to the extrapolation.
d. Description of the benchmark experiments or benchmark sets used.
e. Description of the margin of subcriticality for safety and justification of its adequacy, including a statement of the minimum margin of subcriticality and any other factors that provide reasonable assurance of subcriticality.
f. Description of the controlled software and hardware used.
g. Description of any limitations on method use necessary to establish or ensure the validity of the method.
h. Description of the verification process and demonstration of acceptable results.

Changes to the code or cross section versions require additional code validation. Verification testing is performed during initial installation or when changes are made that could adversely impact the calculation process. Validation or verification are not required for changes that would not reasonably be expected to change calculation results. To ensure continued code performance, verification testing is performed and documented in applicable calculation results.

Safety Limits

The validation establishes the technical basis for the margin(s) of subcriticality based on the quantity and quality of benchmark data, and the overall similarity of the benchmark experiments to the systems being evaluated. A minimum margin of subcriticality of 0.02 in k_{eff} is used to establish the USL for criticality calculations for normal and abnormal conditions within the validated area of applicability and further subject to any limitations or restrictions documented in the TRISO-X FFF applicable Code Validation Report. The area of applicability may be extended beyond the range of the benchmarks by making use of trends in the data, and in accordance with applicable requirements of ANSI/ANS-8.24-2017, as documented in the Code Validation Report. An additional margin of subcriticality may be necessary in these extended ranges as documented in the Code Validation Report or in the evaluation of a specific process.

The minimum margin of subcriticality is justified based on both conservative modeling practices and conservatism in the validation methodology. For the TRISO-X FFF, conservative modeling practices include using an enrichment of 20 wt.% ^{235}U, assuming optimal moderation or saturated conditions for bounding material densities, using bounding fissile material compositions,

modeling geometric components to outer dimensions, and neglecting materials of construction of components unless shown to lead to a more reactive configuration. Some materials of construction are credited in fissile material shipping packages; however, these items are qualified under more stringent design and testing programs with materials of construction reduced (e.g., outer drum wall thickness) to conservatively bound any losses during the life of the package. Conservatism in the validation methodology includes the use of diverse statistical methods and the consideration of many diverse subgroups of the benchmark data. The choice of methods and data that produce the largest calculational margin shall be conservatively applied in determining the USL, including not crediting a positive bias. However, positive bias values will be included in the estimate of bias uncertainty and will conservatively increase bias uncertainty and decrease the associated USL. This same conservative approach will be applied to the validation of the code at other enrichment ranges.

The USL varies with the computer code system (hardware, codes, and cross sections) as the calculational margin varies. The USL is the k_{eff} of a critical system (i.e., 1.0) minus the sum of calculational margin and the margin of subcriticality. Systems that are evaluated within the area of applicability, and that have a $k_{eff} + 2\sigma$ less than or equal to the USL, demonstrate subcriticality.

The calculation of k_{eff} is accomplished using computer code systems that utilize Monte Carlo techniques to determine k_{eff} of a system. Computer models representing the geometric configuration and material compositions of the system are developed for use within the code. The development of appropriate models must account for or conservatively bound both normal and credible abnormal process conditions. The development of appropriate models shall also account for or conservatively bound both normal and credible abnormal process conditions.

For the initial design of the TRISO-X FFF, the SCALE-6.2.3 code with the V7-252-group ENDF/B-VII.1 cross section library is used on validated standalone workstations for criticality calculations. Future versions of the code or nuclear data or use of other industry-accepted codes (e.g., MCNP) will be controlled, verified, and validated with the same level of rigor, as discussed in this section.

When determining subcriticality based on computer code calculations, the following USL limits, regardless of the method employed, shall not be exceeded:

System	Upper Subcritical Limit
High-enriched systems (Uranium enriched in ^{235}U > than 20 wt. %)	0.95
High assay low enriched systems (Uranium enriched in $^{235}U \leq 20$ wt. % and > 10 wt. %)	0.96
Low-enriched systems (Uranium enriched in $^{235}U \leq 10$ wt. %)	0.97

The USL values are exact limit values, and do not imply that compliance need only be shown to 2 significant figures. The determination of the appropriate USL shall be documented in a validation report and lesser values shall be used based on variances in the calculated bias, bias uncertainty, margin of subcriticality, and the statistical method and subset of benchmark that produce the largest calculation margin. Compliance with these values shall include the calculational inaccuracies, such as Monte Carlo variance, by meeting the limit with a margin in the conservative direction of at least two standard deviations (i.e., $k_{eff} + 2\sigma$). Any rounding is in the conservative direction.

5.6 Sources of Criticality Data and Analytical Techniques

The NCS staff may use the following sources of criticality data and analytical techniques to support nuclear criticality safety analyses and evaluations:

1. J. H. Chalmers, G. Walker, and J. Pugh, *Handbook of Criticality Data*, UKAEA Handbook AHSB (S), 1965.
2. H. C. Paxton and N. L. Pruvost, *Critical Dimensions of Systems Containing ^{235}U, ^{239}Pu, and ^{233}U*, LA-10860, Los Alamos National Laboratory, 1986.
3. N. L. Pruvost and H. C. Paxton, *Nuclear Criticality Safety Guide*, LA-12808, Los Alamos National Laboratory, 1996.
4. R. D. Carter, G. R. Kiel, and K. R. Ridgway, *Criticality Handbook*, Volumes I, II, and III, ARH-600, Atlantic Richfield Hanford Company, 1968.
5. Gesellschaft für Anlagenund Reaktorsicherheit (GRS) gGmbH, *Handbuch zur Kritikalität*, GRS-380, 2019.
6. *Determination of H/U Ratios in UO2 Water and ADU-Water Mixtures*, JN-71-2, 1971.
7. B. T. Reardon and M. A. Jessee, Eds., *SCALE Code System*, ORNL/TM-2005/39, Version 6.2.3, March 2018.
8. American Nuclear Society, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*, ANSI/ANS 8.1-2014 (R2018).
9. American Nuclear Society, *Criticality Accident Alarm System*, ANSI/ANS-8.3-1997 (R2017).
10. American Nuclear Society, *Use Of Borosilicate-Glass Raschig Rings As A Neutron Absorber In Solutions Of Fissile Material*, ANSI/ANS-8.5-1996 (R2017).
11. American Nuclear Society, *Nuclear Criticality Safety in the Storage of Fissile Materials*, ANSI/ANS-8.7-1998 (R2017).
12. American Nuclear Society, *Use Of Soluble Neutron Absorbers In Nuclear Facilities Outside Reactors*, ANSI/ANS-8.14-2004 (R2016).
13. American Nuclear Society, *Administrative Practices for Nuclear Criticality Safety*, ANSI/ANS-8.19-2014 (R2019).
14. American Nuclear Society, *Nuclear Criticality Safety Training*, ANSI/ANS-8.20-1991 (R2020).
15. American Nuclear Society, *Use Of Fixed Neutron Absorbers In Nuclear Facilities Outside Reactors*, ANSI/ANS-8.21-1995 (R2011).

16. American Nuclear Society, *Nuclear Criticality Safety Based On Limiting And Controlling Moderators*, ANSI/ANS-8.22-1997 (R2016).
17. American Nuclear Society, *Validation of Neutron Transport Methods for Nuclear Criticality Safety Evaluations*, ANSI/ANS 8.24-2017.
18. American Nuclear Society, *Criticality Safety Engineer Training and Qualification Program*, ANSI/ANS-8.26-2007 (R2016).
19. Ronald Allen Knief, *Nuclear Criticality Safety Theory and Practice*. American Nuclear Society, 1993.
20. D. F. Hollenbach, *Nuclear Criticality Safety Data Book*, Rev. 2, DAC-EN-801768-A100, https://www.osti.gov/servlets/purl/1339414, 2016.
21. Gmelin-Institut für Anorganische Chemie, *Gmelin Handbook of Inorganic Chemistry, Supplement Volume A6, General Properties – Criticality*, 1983.
22. Japan Atomic Energy Research Institute, *Nuclear Criticality Safety Handbook*, JAERI-Review 95-013, 1995.
23. J. T. Thomas, Ed., *Nuclear Safety Guide, TID-7016, Revision 2*, NUREG/CR-0095, ORNL/NUREG/CSD-6, U.S. Nuclear Regulatory Commission, 1978.
24. *International Handbook of Evaluated Criticality Safety Benchmark Experiments*, NEA/NSC/DOC (95), Nuclear Energy Agency Science Committee, Organization for Economic Co-Operation and Development, 2020.

5.7 Requirements for New Facilities or New Processes at Existing Facilities

As required by 10 CFR 70.64(b), the facility is designed using a defense-in-depth approach for protection against inadvertent criticality. To the extent practicable, the facility design considers preference for the selection of engineered controls over administrative controls to increase overall system reliability, and features that enhance safety by reducing challenges to IROFS are incorporated. The facility design addresses the baseline design criteria of 10 CFR 70.64(a) for new facilities and criticality control.

As required by 10 CFR 70.64(a)(9), the TRISO-X Fuel Fabrication Facility design provides for criticality control including application of the double contingency principle.

If a planned new facility and/or new process meets the 10 CFR 70.72 criteria requiring a license amendment, the baseline design criteria of 10 CFR 70.64(a) will be applied to the control of criticality hazards. A defense-in-depth approach will be applied to higher risk accident sequences as required by 10 CFR 70.64(b).

NRC SPECIAL NUCLEAR MATERIAL LICENSE
SNM-XXXX
CHAPTER 6

CHEMICAL PROCESS SAFETY
Table of Contents

CHEMICAL PROCESS SAFETY

6.1 Chemical Process Safety Program

TRISO-X has established and maintains a chemical process safety program to assure that workers, the public, and the environment are adequately protected from the chemical hazards related to the storage, handling, and processing of licensed materials. With respect to NRC oversight, the program is implemented, and the facility is designed, to address chemical hazards associated with licensed materials and hazardous chemicals produced from licensed materials, and chemical risks of facility conditions that could affect the safety of licensed materials. The chemical safety element of the safety program required by 10 CFR 70.62(a) is evaluated and implemented through the ISA process described in Chapter 3.

Based on the 2013 Memorandum of Understanding (MOU) between the NRC and the Occupational Safety and Health Administration (OSHA), the NRC oversees chemical safety issues related to radiation risks of licensed materials, chemical risks of licensed materials, and facility conditions that affect or may affect the safety of licensed materials and thus present an increased radiation risk to workers. OSHA oversees facility conditions that do not affect or involve the safety of licensed materials.

6.2 Process Chemical Risk and Accident Sequences

6.2.1 Process Descriptions

A general process description of the production operations is provided in Chapter 1. This is supplemented by more detailed process system descriptions in the facility's ISA Summary as required by 10 CFR 70.65(b)(3) that include relevant information to provide a basic understanding of operation, allow an understanding of the chemical hazards, and support the development of potential accident sequences.

6.2.2 Identification and Evaluation of Chemical Accident Sequences

Potential accident sequences involving chemical hazards related to the safety of licensed materials are incorporated as part of the ISA. Accident sequence identification, consequence and likelihood determination, and risk assessment methods are discussed in Chapter 3 and the ISA Summary.

The ISA includes evaluations of chemical risks of licensed materials, risk of chemicals derived from licensed materials, and chemical risks introduced by facility conditions that could affect the safety of licensed materials. Analysis assumptions consider the maximum foreseeable inventories of chemicals at specific locations. Routine, non-routine, and credible abnormal operational scenarios are included in the analysis, along with conservative physical properties of the associated chemicals. Results of the evaluations are compared to the performance criteria in 10

CFR 70.61, and unmitigated scenarios that could result in Intermediate or High consequences are documented in the ISA Summary. IROFS and appropriate management measures are applied to Intermediate or High consequence scenarios to ensure that the performance criteria in 10 CFR 70.61 are met.

6.2.3 Chemical Consequence Estimates

Chemical consequence estimates are based on guidance in NUREG/CR-6410, where appropriate, for source term determination, release fractions, dispersion factors, meteorological conditions, and consequence modeling techniques. Alternative methods and other industry accepted techniques may also be used to perform consequence calculations, provided the methods are appropriate to the process, the physical setting, and the specific condition being evaluated. Safety Data Sheets are also reviewed when assessing chemical consequences. Dispersion models may be utilized to assess the consequences of accidental chemical release scenarios so that the results can be compared to the performance criteria in 10 CFR 70.61 as part of the ISA process. Source term and dispersion models are selected based on the chemical being evaluated and should provide for a conservative estimate of the potential consequences.

6.2.4 Chemical Exposure Standards

Chemical exposure standards used in support of assessing the consequences of an acute chemical exposure to an individual are identified in accordance with 10 CFR 70.65(b)(7). Per NUREG-1520, acceptable chemical exposure standards include, but are not limited to, Acute Exposure Guideline Levels (AEGLs) developed by the United States Environmental Protection Agency, Emergency Response Planning Guidelines (ERPGs) produced by the American Industrial Hygiene Association, and exposure limits established by OSHA. If no AEGLs or ERPGs are available, Temporary Emergency Exposure Limits developed by the United States Department of Energy may be used. These standards are documented in the ISA Summary.

6.3 IROFS and Management Measures

6.3.1 Chemical Process IROFS

IROFS are identified and implemented for chemical accident sequences that can lead to Intermediate or High consequence scenarios to ensure that the performance criteria in 10 CFR 70.61 are met. The IROFS and associated accident sequences are documented in the ISA Summary to demonstrate compliance with the performance criteria of 10 CFR 70.61. IROFS may be engineered controls (passive or active), enhanced administrative controls (active engineered features that alert a person to take an action), or administrative controls (actions of people).

6.3.2 Management Measures

Management measures that ensure the reliability and availability of IROFS are established as described in Chapter 11.

6.3.3 Chemical Process Safety Coordination with Emergency Management

IROFS identified in the ISA Summary credited to address chemical consequences may also be described in the Site Emergency Plan. The Site Emergency Plan described within Chapter 8 addresses response to and mitigation of accidents involving process chemicals.

6.4 Requirements for New Facilities or New Processes at Existing Facilities

As required by 10 CFR 70.64(b), the TRISO-X Fuel Fabrication Facility is designed using a defense-in-depth approach for protection against chemical accidents. To the extent practicable, the facility design considers preference for the selection of engineered controls over administrative controls to increase overall system reliability, and features that enhance safety by reducing challenges to IROFS are incorporated. The facility design addresses the baseline design criteria of 10 CFR 70.64(a) for new facilities and the control of chemical hazards.

As required by 10 CFR 70.64(a)(5), the design provides for adequate protection against chemical risks produced from licensed material, facility conditions that affect the safety of licensed material, and hazardous chemicals produced from licensed material. The design process includes review of opportunities to remove or reduce the hazards in chemical processing considering inherently safe design concepts – minimize, substitute, moderate, simplify. The proper handling, use, and storage of chemicals is addressed through approved procedures and Hazard Communication training.

If a planned new facility and/or new process meets the 10 CFR 70.72 criteria requiring a license amendment, the baseline design criteria of 10 CFR 70.64(a) will be applied to the control of chemical hazards. A defense-in-depth approach will be applied to higher risk accident sequences as required by 10 CFR 70.64(b).

NRC SPECIAL NUCLEAR MATERIAL LICENSE
SNM-XXXX
CHAPTER 7

FIRE SAFETY
Table of Contents

FIRE SAFETY

7.1 Fire Safety

The TRISO-X fire protection program is based on a combination of fire protection measures and systems. Such measures and systems are designed and maintained in accordance with industry standards and prudent industry practices. The standards and practices most often consulted are those of the National Fire Protection Association (NFPA).

The fire protection program is designed to minimize the potential for and provide reasonable protection against fire and explosive hazards associated with the processing, handling, and storage of licensed materials during normal operations, anticipated operational off-normal occurrences, and credible accidents. As part of the ISA process described in Chapter 3, area operations are evaluated for the potential for, and consequences of, fire and explosive hazards that could affect the safety of licensed materials and thus present an increased radiological risk. Where these consequences could exceed the performance requirements in 10 CFR 70.61, IROFS are assigned. These IROFS are identified and controlled as described in the ISA Summary and in approved procedures to ensure they remain available and reliable.

The TRISO-X fire protection program and general facility design were developed using NFPA 801, *Standard for Fire Protection for Facilities Handling Radioactive Materials*, 2014 edition. In addition to the Fire Hazards Analysis (FHA) prepared as required to comply with NFPA 801 and to support the ISA process, TRISO-X has a general fire safety program that includes general fire safety management measures, facility design requirements, and general fire protection and emergency response measures. The FHA confirms that an adequate level of fire protection and life safety has been achieved that includes defense in depth protection and conformance with national codes, standards, and federal regulations applicable to the facility's design and construction, and manufacturing operations.

7.2 Fire Safety Management Measures

7.2.1 Fire Safety Organization

Fire protection program organization, authorities, and responsibilities are described in Chapter 2. The Authority Having Jurisdiction (AHJ) for the fire protection program is held by the fire protection function unless mandated by local and/or state regulation, where the specifically required AHJ is utilized.

The equivalency concept may be applied in meeting the provisions of NFPA codes or standards, provided that technical documentation demonstrates equivalency, and that the method, system, or device is listed or approved for the intended purpose. Equivalency evaluation records are retained as described in Chapter 11, and records are made available for NRC review and inspection. Where required by code, equivalency evaluations are submitted to the AHJ.

If an IROFS-related NFPA code commitment cannot be met and an alternate method that provides an equivalent level of safety cannot be identified, a formal request to approve a deviation from the NFPA code commitment will be submitted to the NRC for review and inspection relative to effects on nuclear safety.

The configuration management program described in Chapter 11 assures that any facility changes are properly evaluated with regard to the impact upon fire safety and documented within the facility safety bases. New facilities and processes are reviewed by the facility safety review committee.

7.2.2 Fire Prevention

1. Employee Training – General fire safety awareness training is administered to each employee as part of their general employee training. Applicable IROFS training is provided as part of job specific training.
2. Facility Audits and Inspections – Periodic audits and inspections, as detailed in Chapter 11 and approved procedures, are performed for facilities containing licensed materials in a quantity and form that could cause at least an intermediate consequence as defined in 10 CFR 70.61 if totally consumed by fire. Items identified are entered into the facility's corrective action program and tracked to closure.
3. Fire Prevention Procedures – Approved procedures are maintained for the administration of the general fire prevention program. Fire safety procedures address areas such as the storage, handling, and control of combustible (including transient and work-generated), flammable, and pyrophoric materials; the review and issuance of permits for work performed in the facility to control sources of ignition, such as welding, cutting, brazing, soldering, and grinding; and fire prevention surveillance to ensure that TRISO-X maintains a readiness to extinguish fires and limit consequences of a fire through use of fire detection and suppression systems.

7.2.3 Inspection, Testing, and Maintenance of Fire Protection Systems

Procedural guidance is established for the inspection, testing, and maintenance of fire protection systems routinely performed by TRISO-X personnel. These procedures are applied to fire detection, warning, and suppression systems. Inspection, testing, and maintenance may also be conducted by outside vendors that specialize in the type of work required. Records of TRISO-X activities and of outside vendor activities are maintained as described in Chapter 11.

7.2.4 Emergency Response Organization

TRISO-X maintains an emergency response organization in accordance with 10 CFR 70.22. Approved procedures outline the overall emergency response program, including but not limited to staffing, training, drills and exercises, response measures, and offsite agency coordination. The Emergency Preparedness program, as described in Chapter 8, includes memorandums of

understanding between TRISO-X and off-site agencies, and addresses periodic off-site training and drills. The TRISO-X emergency response team is discussed in Section 7.6.7.

7.2.5 Pre-Fire Plans

TRISO-X maintains pre-fire plans for each building, or part thereof, that, if totally consumed by fire, could release licensed material in a quantity and form that could cause at least an intermediate consequence as defined in 10 CFR 70.61. These pre-fire plans provide information needed by fire-fighting personnel responding to the emergency and are located for ready access by the facility emergency response team and local fire departments who may respond to an emergency at the TRISO-X FFF. Where Criticality or adverse chemical reaction concerns may exclude water suppression techniques within certain process areas, or if detection and/or other suppression methods are selected to further mitigate the potential for criticality, this is addressed in the associated pre-fire plans. Pre-fire plans for buildings/areas that involve licensed materials are reviewed by NCS staff.

7.3 Fire Hazards Analysis

An FHA has been developed as required by NFPA 801 (2014 edition). This document was prepared for the manufacturing building that is involved in processing or storage of licensed materials in sufficient quantities and in a form that, if released in a fire, could result in an intermediate or high consequence event as defined in the ISA summary.

The FHA is a component of the ISA process, as described in Chapter 3. The FHA focuses on bounding fire scenarios within buildings that contain licensed materials and consider fire loading, the consequences and analysis of an unmitigated fire, and mitigating controls. Fire and/or explosion hazards which have the potential to create high or intermediate consequences as defined in 10 CFR 70.61 are controlled via the application of appropriate IROFS. These IROFS are documented in the ISA Summary as described in Chapter 3. Management measures that ensure the reliability and availability of IROFS are established as described in Chapter 11.

7.4 Facility Design

7.4.1 Facility Design Criteria

The TRISO-X FFF is designed and constructed consistent with the requirements of NFPA 801 (2014 edition), as well as applicable state and local building, electrical, and fire codes in effect at the time of construction. The FHA and/or facility design documentation describes the type of construction, building materials, exterior openings, fire detection and alarm system, automatic fire suppression systems, and the fire water distribution system.

Design and construction criteria for facilities that process licensed materials include an evaluation to determine the proper methods to prevent, detect, extinguish, limit, and control fires and explosions. Fire-resistive and non-combustible materials are used as appropriate. Fire areas may

subdivide specified processes or materials involving significant fire hazards to confine the spread of fire to the area of origin, and fire barriers are provided as recommended by the FHA.

Criticality or adverse chemical reaction concerns may exclude automatic water suppression extinguishing systems from certain process areas, or detection and/or other suppression methods may be selected to further mitigate the potential for criticality where these areas have significant fire risks.

Electrical installation, ventilation, lightning protection, fire water runoff, life safety and worker egress, and firefighter access are considered. Physical security features designed for the protection of licensed material are reviewed to prevent or minimize the inadvertent delay of either worker egress or fire fighter access during emergency situations.

7.5 Process Fire Safety

Process fire safety is considered in the planning, design, and construction of new facilities and processes. In areas where fire and/or explosion hazards may impact licensed material, risks are evaluated and documented by the ISA process. The ISA evaluates the special fire risk associated with:

- Combustible, flammable, and pyrophoric process chemicals (solids, liquids, gases), in use and in storage,
- Exothermic reactions of uranium oxides, and
- High temperature and/or high pressure equipment.

The ISA summary identifies the fire-initiated release scenarios that may impact licensed material. Process-related fire hazards are controlled with IROFS to the extent necessary to meet the performance requirements of 10 CFR 70.61.

7.6 Fire Protection and Emergency Response

7.6.1 Water Supply

Fire protection water is provided to TRISO-X facilities through a stand-alone underground loop system fed by the City of Oak Ridge water distribution system. Details of the fire protection water supply are contained in the FHA, in pre-fire plans, and/or in approved procedures.

7.6.2 Hydrants, Standpipes, and Hose Houses

Multiple hydrants are provided throughout the fire protection loop. Locations are such that they allow ready access for quick use when needed to assist in firefighting. In areas where automatic water suppression extinguishing systems are excluded or are not preferred due to criticality or adverse chemical reaction concerns, standpipe hose cabinets are provided for use when authorized. Hose houses may also be provided in areas that are hard to access or are

inaccessible. Details of the hydrants, standpipes, and hose houses are contained in the FHA, in pre-fire plans, and/or in approved procedures.

7.6.3 Fixed Fire Protection Systems

Fixed fire protection systems, including clean agent and/or gaseous fire extinguishing and automatic sprinkler systems, are utilized throughout TRISO-X facilities. Selection of equipment considers the severity of the hazard, the type of activities performed in the area, the potential consequences of a fire, and the potential consequences of use of the suppression equipment (e.g., risk of an accidental criticality, or substantial electrical hazard). Details of the fixed fire protection systems are contained in the FHA, in pre-fire plans, and/or in approved procedures.

7.6.4 Fire Detection and Alarm Systems

TRISO-X facilities are equipped with fire detection and alarm systems that include initiating devices (e.g., smoke detection, heat detection, manual pull stations) and notification appliances (e.g., audible alarm, visual strobe). Selection of equipment considers the types of hazards in the area and the environment in which the equipment will be installed. Fire detection, alarm, and suppression systems are continuously monitored. Details of the fire detection and alarm systems are contained in the FHA, in pre-fire plans, and/or in approved procedures.

7.6.5 Portable Fire Extinguishers

Portable fire extinguishers are located throughout TRISO-X facilities for use by employees responding to incipient stage fires. Selection of equipment considers the class of fires that could occur in the area and capacity required. Details regarding the type and location of the portable fire extinguishers are contained in the FHA, in pre-fire plans, and/or in approved procedures.

7.6.6 Lightning Protection

TRISO-X facilities have been outfitted with lightning protection systems as specified in the FHA.

7.6.7 Emergency Response Team

TRISO-X maintains an emergency response team. The emergency response team is an organized group of employees who are knowledgeable, trained, and skilled in basic fire-fighting operations, first aid techniques, and emergency response. Training and education are provided for team members commensurate with those duties and functions that they are expected to perform. TRISO-X relies on support from offsite firefighting resources for fire and hazardous materials release response.

The TRISO-X Emergency Preparedness program, as described in Chapter 8, includes memorandums of understanding between TRISO-X and off-site agencies, and addresses periodic off-site training and drills. The TRISO-X emergency response organization is discussed in Section 7.2.4.

7.7 Requirements for New Facilities or New Processes at Existing Facilities

As required by 10 CFR 70.64(b), the TRISO-X Fuel Fabrication Facility is designed using a defense-in-depth approach for protection against fire and explosion accidents. To the extent practicable, the facility design considers preference for the selection of engineered controls over administrative controls to increase overall system reliability, and features that enhance safety by reducing challenges to IROFS are incorporated.

Consistent with the guidance provided in NFPA 801, the FHA and ISA demonstrate that the design, building construction, fire areas, life safety, and ventilation of the facility; process fire safety; fire detection and suppression systems; pre-fire planning; and manual fire suppression capability together form the basis for compliance with the BDC of 10 CFR 70.64(a) and the defense in depth requirements of 10 CFR 70.64(b).

As required by 10 CFR 70.64(a)(3), the design provides for adequate protection against fire and explosion by incorporating defense-in-depth concepts such that health and safety are not wholly dependent on any single element of the design, construction, maintenance or operation of the facility. This is accomplished by achieving a balance between preventing fires from starting; quickly detecting, controlling and promptly extinguishing those fires that do occur; and protecting structures, systems, and components such that a fire will not lead to an unacceptable consequence.

If a planned new facility and/or new process meets the 10 CFR 70.72 criteria requiring a license amendment, the baseline design criteria of 10 CFR 70.64(a) will be applied to the control of fire hazards. A defense-in-depth approach will be applied to higher risk accident sequences as required by 10 CFR 70.64(b).

NRC SPECIAL NUCLEAR MATERIAL LICENSE
SNM-XXXX
CHAPTER 8

EMERGENCY MANAGEMENT
Table of Contents

EMERGENCY MANAGEMENT

8.1 Emergency Plan

TRISO-X maintains an emergency plan in accordance with the requirements of 10 CFR 70.22(i)(1)(ii). Content of the plan is in accordance with 10 CFR 70.22(i)(3) and is consistent with U.S. Nuclear Regulatory (NRC) guidance in Regulatory Guide 3.67, *Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities*.

8.2 Emergency Plan Implementing Procedures

The requirements of the emergency plan are implemented through approved procedures and checklists.

8.3 Amendment of the Emergency Plan

The emergency plan is maintained as needed. TRISO-X may change the approved plan without NRC approval if the changes do not decrease the effectiveness of the plan. In accordance with 10 CFR 70.32(i), copies of changes that do not decrease effectiveness will be provided to the NRC and appropriate organizations within six months of making the changes. Proposed changes to the plan that decrease its effectiveness will not be implemented without prior approval by the NRC.

8.4 Agreements with Offsite Emergency Response Resources

Formal written agreements with appropriate offsite emergency response organizations (fire, police, medical, etc.) have been established and will be maintained to assure implementation of the emergency plan and emergency response procedures. These agreements are reviewed and renewed as described in the emergency plan and are maintained on file at the TRISO-X Fuel Fabrication Facility.

8.5 Requirements for New Facilities or New Processes at Existing Facilities

As required by 10 CR 70.64(a)(6), and as described in the emergency plan, the TRISO-X Fuel Fabrication Facility design provides for emergency capability to maintain control of licensed material and hazardous chemicals produced from licensed material, evacuation of onsite personnel, and onsite emergency facilities and services that facilitate the use of available offsite services.

If a planned new facility and/or new process meets the 10 CFR 70.72 criteria requiring a license amendment, the baseline design criteria of 10 CFR 70.64(a) will be applied to emergency capability.

NRC SPECIAL NUCLEAR MATERIAL LICENSE
SNM-XXXX
CHAPTER 9

ENVIRONMENTAL SAFETY
Table of Contents

ENVIRONMENTAL SAFETY

9.1 Environmental ALARA

TRISO-X has established and maintains an environmental safety program to maintain concentrations of radioactive materials in facility effluents and the surrounding environment as low as reasonably achievable (ALARA). The TRISO-X ALARA program is described in Chapter 4. Environmental releases are limited and monitored such that compliance with the public dose limits of 10 CFR 20.1301 and the effluent limits of 10 CFR 20.1302 can be achieved and demonstrated. These objectives are supported by performing routine measurements and calculations, comparing results to action levels, and reporting results to facility management and the NRC, as appropriate. Internal action levels are implemented through approved procedures to provide early identification of potential problems and prevent exceedance of established guidelines. If action levels are exceeded, investigations are initiated to identify the cause, and appropriate corrective action(s) are taken to minimize the likelihood of recurrence as part of the corrective action program outlined in Chapter 11.

The environmental safety program implementing procedures ensure compliance with 10 CFR 20 Subparts B, *Radiation Protection Programs*, D, *Dose to the Public*, F, *Surveys and Monitoring*, K, *Waste Disposal*, L, *Records*, and M, *Reports*, that address effluent control and treatment. The program includes provisions for the monitoring of the facility environment, including ambient air, surface water, ground water, soils, and vegetation, that could be affected by facility effluents. The TRISO-X ISA Summary and Environmental Report provide additional information. Chapter 2 of this application addresses staff qualifications of individuals responsible for the environmental safety program.

9.2 Gaseous Effluent Control

Operating and engineered controls are used as necessary to ensure that environmental airborne concentrations of radioactive materials attributable to gaseous effluents are constrained and resultant radiological doses to members of the public comply with the concentration limits and public dose limit specified in 10 CFR 20.1101(d), consistent with guidance in Regulatory Guide 4.20. Dose calculations are performed using nationally recognized methods.

Dose calculations as well as environmental concentrations in 10 CFR 20, Appendix B, Table 2 for members of the public may be modified based on ICRP 66 and 68 as described in Chapter 1, assuming an Activity Median Aerodynamic Diameter (AMAD) of 5 micrometer.

9.2.1 Gaseous Effluent Sampling

Continuous representative sampling is performed in stacks exhausting air with potential concentrations of radioactive materials that are significant with respect to the site's compliance with 10 CFR 20. Samples are collected and analyzed for particulate radioactive material on a scheduled basis as defined in approved procedures using methods and frequencies appropriate for the effluent medium and the radionuclide(s) being sampled. Effluents are sampled unless periodic sampling or other means have established that radioactivity in the effluent is insignificant and will remain so.

Gaseous effluent sampling is performed during manufacturing operations involving licensed materials. Sampling of exhaust air stacks is not required when the underlying ventilation system has been shut down in conjunction with a cessation of the processing of licensed materials in the affected ventilated spaces. Any passive emissions of radioactive materials are abated by the continued presence of the HEPA filters in place. Approved procedures define action levels to ensure that compliance with applicable limits is maintained.

9.2.2 High-Efficiency Particulate Absolute (HEPA) Filtration

HEPA filtration is used on stacks exhausting air that potentially contain radioactive materials that are significant with respect to the site's compliance with 10 CFR 20. Exhaust air is passed through at least one stage of HEPA filtration prior to release from the stack. Fire-resistant HEPA filters that are certified by the manufacturer as meeting HEPA efficiency specifications are used.

The adequacy of final HEPA filter installations is verified by in-place testing prior to initiating operations with radioactive materials in the following instances:

- Startup of a new facility
- Following replacement of final filters
- After maintenance work on the final filter bank that could have foreseeable adverse impacts on their effective operation
- After exposure of the final filters to a condition or agent that may have adversely impacted their effective operation, if deemed necessary based on visual/operational inspection.

9.2.3 Final HEPA Filter Surveillance

Measures as described in approved procedures are taken to conservatively monitor the potential onset of, or adverse emissions impacts from HEPA filter deterioration. These measures include the following:

- Periodic inspection of HEPA filters;
- Periodic measurement of differential pressures across HEPA filter banks; and
- Stack emissions monitoring that establish action levels triggering notifications to the maintenance/engineering organization and performance of HEPA filter inspections at

measured offgas radionuclide concentrations set below applicable modified 10 CFR 20 Appendix B effluent limits, and levels above which the process would be shut down.

Final HEPA filter installations are equipped with pressure differential measuring/indicating devices. Measured differential pressures are used to evaluate the need for filter changeout/maintenance.

9.3 Liquid Effluent Control

No liquid effluents are planned for radiological process streams in the TRISO-X FFF. Design of the facility, along with operating and engineered controls, is used as necessary to ensure that radiological liquid effluent discharges to the environment do not occur. Approved procedures define action levels to ensure that compliance with applicable limits is maintained.

9.3.1 Wastewater Collection/Treatment

Process solutions generated by process systems and equipment are recycled to the maximum extent practical. Process solutions contaminated with uranium that cannot be recovered/ recycled are identified as liquid wastes. Liquid wastes are collected and sampled to determine appropriate handling/treatment steps. Treatment typically involves adjustment of pH, filtering, ion exchange, and/or precipitation. Precipitates are de-watered, and the solids are packaged for off-site disposal. If needed, liquid wastes that have been handled/treated can be sampled and discharged through an inline monitor to shipping packages or conveyances for off-site disposal.

Sanitary sewer discharges to the City of Oak Ridge sewer system from facility restrooms and non-radiological process streams related to equipment blowdowns, flushes, and cleaning activities are conducted in accordance with a locally-issued permit. Used oils may also be sampled and containerized for shipment to a licensed disposal facility.

9.4 Waste Management

The TRISO-X Waste Management Program (WMP) is consistent with EPA and NRC guidance to meet the requirements in 10 CFR 20.1406, and is designed to minimize facility generated waste. The WMP is endorsed and supported by upper management, details waste streams and the waste characterization process, and is evaluated on a scheduled basis for improvement. The WMP procedures and facilities for waste handling, staging for shipment, and monitoring result in safe and timely disposition of materials.

Solid waste disposal preparation facilities, with sufficient capacity and capability to enable processing, packaging, and transfers of solid waste to licensed treatment and/or disposal sites in accordance with the regulations, are provided and maintained as required to support the operation of the TRISO-X Fuel Fabrication Facility.

9.5 Environmental Monitoring

TRISO-X conducts a routine environmental surveillance program. Surface environmental media and groundwater samples are collected from strategic locations in the surrounding environs and analyzed for pertinent constituents of concern. Baseline levels of radionuclides in media surrounding the facility established through sampling and analysis are used for comparison for future environmental monitoring events. Action levels and actions to be taken if levels are exceeded are specified for each environmental medium and radionuclide as defined in approved procedures.

The program provides additional validation of effluent monitoring systems, early detection and response to a negative trend in environmental data, and support data in the event of a release of radioactive material. Environmental dosimeters are located onsite to monitor ambient external doses and to assist with the assessment of potential accidents. Environmental dosimeter data are used to monitor external dose rates in unrestricted areas, determine doses to members of the public, and demonstrate compliance with regulatory dose limits. Doses to members of the public are calculated per 10 CFR 20.1302(b)(1), and may include considerations for the amount of time a member of the public is actually present or potentially present at a given location.

A summary of typical sampling activities is included in Table 9-1. Maps of sampling locations will be provided once recommendations are available from the Environmental Report. It is expected that the locations for sampling of ambient air, soil, and vegetation will be concentrated along the predominant wind directions. Four groundwater observation wells are installed on the site. Groundwater elevation measurements and modeling indicate that groundwater generally flows in a southwest direction toward East Fork Poplar Creek. There are no known household, public, or industrial users of groundwater downgradient of the site.

Figure 9-1: Environmental Monitoring Parameters

Type of Sample	Analyses	Typical Sampling Frequency
Air Effluent Discharge Points	Gross Alpha/Beta	Continuous (collection weekly)
Ambient Air	Gross Alpha/Beta	Monthly
Groundwater	Gross Alpha/Beta Isotopic Uranium	Quarterly
Soil	Gross Alpha/Beta	Semi-annually
Vegetation	Gross Alpha/Beta	Semi-annually

9.6 Program Management

Quantities of radioactive material in air and liquids released from the facility are reported to the NRC on a semi-annual basis as required by 10 CFR 70.59, *Effluent monitoring reporting requirements.*

Approved procedures outline sampling techniques, sample processing and analysis methodologies, quality assurance, and other necessary information to validate analytical results and maintain a viable program.

Sample analysis may be performed either on-site or off-site. In all cases, analytical techniques for sample analysis of each medium are appropriate for the quantities and types of radionuclides present at the facility and are sensitive enough to ensure adequate detection and quantification based on media radiological content and limits. Quality control procedures, for on-site or off-site analysis, detail the periodic checks necessary to demonstrate the operability of the instrumentation used for analysis. Analytical results are reported in a timely manner so that staff can determine the appropriate response to established action levels.

NRC SPECIAL NUCLEAR MATERIAL LICENSE
SNM-XXXX
CHAPTER 10

DECOMMISSIONING
Table of Contents

DECOMMISSIONING

10.1 Decommissioning Funding Plan (DFP)

TRISO-X maintains a DFP that contains the elements called for in 10 CFR 70.25(e)(1), including a decommissioning cost estimate; a description of the methods used to assure funds for decommissioning are available when needed; a means for adjusting the cost estimate and associated funding levels periodically over the life of the facility; and, when applicable, certification that financial assurance has been provided in an amount that covers the current estimate for decommissioning. The DFP addresses the decommissioning of facilities with potential for contamination with licensed materials at the TRISO-X Fuel Fabrication Facility site. The DFP describes how facility design and procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment and generation of radioactive waste, and will facilitate eventual decommissioning as required by 10 CFR 20.1406(a).

10.2 Decommissioning Cost Estimate

Consistent with 10 CFR 70.25(e)(2), the decommissioning cost estimate will be reviewed and updated at an interval not to exceed three years.

10.3 Financial Assurance for Decommissioning

Financial assurance for decommissioning will be provided by the methods authorized in 10 CFR 70.25(f) and will be in place prior to introducing licensed material. TRISO-X will provide the NRC with signed originals of the financial instruments obtained to satisfy the requirements of 10 CFR 70.25(f).

10.4 Recordkeeping for Decommissioning

In accordance with 10 CFR 70.25(g), records will be maintained that are important to the decommissioning of the TRISO-X Fuel Fabrication Facility until such time as the facility is released for unrestricted use.

NRC SPECIAL NUCLEAR MATERIAL LICENSE
SNM-XXXX
CHAPTER 11

MANAGEMENT MEASURES
Table of Contents

MANAGEMENT MEASURES

As specified in 10 CFR 70.62(d), management measures are applied to Items Relied on For Safety (IROFS) to provide reasonable assurance that the IROFS are designed, implemented, and maintained to ensure they are available and reliable to perform their functions when needed. The ISA Summary identifies IROFS applied to facility systems and activities to assure they function to satisfy the performance requirements of 10 CFR 70.61. IROFS may be engineered controls (passive or active), enhanced administrative controls (active features that prompt a person to take an action), or administrative controls (actions of people). Management measures are applied to IROFS using a graded approach based on the type of control and may also consider the reduction of risk credited to that control and/or the item's importance in meeting the performance requirements. Methods used to select and assign management measures to IROFS are documented in approved procedures.

11.1 Configuration Management (CM)

A formal review and approval process is used to evaluate modifications to systems and components to ensure that configuration changes do not adversely impact currently implemented IROFS and to ensure new processes meet the performance requirements of 10 CFR 70.61. The CM program captures formal documentation governing the design, safety bases, and continued modification of the site, structures, processes, systems, equipment, components, selected computer programs, personnel activities, and supporting management measures.

11.1.1 CM Program

The TRISO-X CM Program controls facilities and processes so safety bases are maintained, and changes are evaluated and documented according to approved procedures consistent with 10 CFR 70.72 in Section 11.1.4 and the license application change process discussed in Chapter 1. The CM process provides assurance that consistency is established and maintained between facility design, operational requirements, physical configuration, and facility documentation. CM provides oversight and control of design information, safety information, and records of modifications that might impact the ability of IROFS to perform their functions when needed.

Engineering is responsible for the implementation and ongoing management of the CM Program. All TRISO-X personnel and organizations are responsible for complying with the CM Program objectives and implementing Program requirements as an integral part of their respective functional areas of operation.

11.1.2 Design Requirements

Configuration control is accomplished during design using procedures for controlling design, preparation, review, and approval. Design requirements and associated design bases are established and maintained during design, construction, and operations.

For new facilities and processes/systems, design requirements are required to be developed, reviewed, approved, and documented before input of SNM. The baseline design criteria (BDC) identified in 10 CFR 70.64(a) are addressed for IROFS. The preferred design approach is used to the extent practical to select engineered controls over administrative controls. New facility and system design is also based on defense in-depth practices in accordance with 10 CFR 70.64(b) to enhance safety by reducing challenges to IROFS. Design requirements and documents are prepared by the engineering organization. Prior to approval, the design documents are reviewed for adequacy, accuracy and completeness as per approved procedures. Changes to design documents or the ISA are subject to the change control processes as described in Chapter 1 and Section 11.1.4

11.1.3 Document Control

Procedures are established to control the preparation and issuance of documents. This includes creation, revision, storage, tracking, distribution, and retrieval of applicable information, to include but not limited to, manuals, instructions, drawings, procedures, design documents, specifications, plans, and other documents that pertain to the CM function. Measures are established to ensure documents, including revisions, are adequately reviewed, approved, and released for use by authorized personnel. An electronic document management system is used both to file facility records and to make available the latest revision (i.e., the controlled copy) of design documents. Controlled documents are maintained until cancelled or superseded.
As part of the configuration management program, refer to Section 11.7 for further discussion of the document control and records management procedures.

11.1.4 Change Control

The objective of the change control process is to maintain consistency among design requirements, the physical configuration, and the related facility documentation (including the ISA). The process is used to ensure facility configuration documentation changes are properly reviewed, approved and implemented to assure that all impacts of proposed changes are identified and evaluated, design requirements (and bases) are maintained or appropriately revised, and changes are coordinated across the affected organizations and personnel responsible for activities and programs at the TRISO-X FFF.

Types of changes are defined and may range from replacement with identical design that are authorized as part of normal maintenance, to new or different designs that require specified review and approval. Major changes include substantial modifications to existing licensed facilities and/or new processes, new licensed facilities, or new processes in existing licensed facilities. Any change requiring a license amendment is also considered a major change. The change control process is implemented via approved procedures to which appropriate personnel are trained.

The change control process assures that the following items are addressed prior to implementing a change as required by 10 CFR 70.72(a):

1) The technical basis for the change;
2) The impact of the change on safety, health, and control of licensed material;
3) Modifications to existing drawings, procedures, and training;
4) Authorization requirements for the change;
5) For temporary changes, the approved duration (e.g., expiration date) of the change;
6) The impacts or modifications to the ISA, ISA Summary, nuclear criticality safety evaluation, or other safety program information, developed in accordance with 10 CFR 70.62 and/or 10 CFR 70.64; and
7) An evaluation as to whether or not a license amendment must be approved by the NRC prior to implementation of the change in accordance with 10 CFR 70.72(c).

Final documentation of the change approval is maintained, and the applicable documentation is made available to the affected personnel. Per 10 CFR 70.72(d)(2), a brief summary of major changes that required revision of the applicable safety or environmental bases will be submitted within 30 days after the end of the calendar year during which the changes occurred.

11.1.5 Assessments

Periodic audits and/or assessments of the configuration management program are conducted in accordance with the requirements in Section 11.5 for the purpose of evaluating the program's effectiveness and to correct deficiencies. The results of these assessments are documented and maintained in accordance with approved procedures.

11.2 Maintenance

The maintenance program is designed to ensure that IROFS are maintained in a manner to ensure they are available and reliable to perform their intended function when needed. The maintenance program consists of the following key program elements, including management systems that provide scheduling and documentation of these elements when applied to IROFS:

1) Surveillance and Monitoring,
2) Corrective Maintenance,
3) Preventive Maintenance, and
4) Functional Testing.

Maintenance procedures and instructions are an integral part of the Maintenance program as described in Section 11.4.3.

11.2.1 Surveillance and Monitoring

The surveillance and monitoring program is implemented to monitor the current and long-term performance of IROFS.

Surveillance activities include preventive maintenance (11.2.3) and functional testing (11.2.4) that are performed on a scheduled basis, and follow-up to corrective maintenance (11.2.2). Documentation of surveillances is prepared as per approved procedures. Frequencies of surveillances are based on the type and safety significance of the IROFS, as well as manufacturer's recommendations. The results of surveillances are trended to support the determination of performance trends for IROFS and can lead to changes to maintenance frequencies, if appropriate. Maintenance procedures also prescribe compensatory measures, if appropriate for surveillance tests of IROFS that can only be performed while the equipment is out of service.

IROFS found to be out-of-tolerance or unable to perform their intended function are reported in a timely manner through the corrective action program discussed in Section 11.6. Reports of IROFS failures are entered into the corrective action program which provides a means to evaluate the failure, identify the cause of failure, and assign appropriate corrective actions to be initiated. Records of IROFS performance issues and corrective actions are maintained within the maintenance and corrective action programs, as applicable. Records for failures of IROFS are maintained in accordance with 10 CFR 70.62(a)(3) within the corrective action program.

11.2.2 Corrective Maintenance

Corrective maintenance is performed using a systematic, integrated, and controlled approach to ensure that IROFS and other systems necessary for the safe operation of the facility are properly repaired and restored to service in a manner that maintains facility safety and the function of the safety system. Maintenance activities are performed on IROFS in a manner that minimizes or eliminates the recurrence of unacceptable performance deficiencies.

Corrective maintenance is authorized, initiated, and documented through a formally established process that includes steps requiring coordination between the maintenance and operating organizations. The process also includes an evaluation to determine if IROFS performance have been, or may be, adversely affected by the equipment failure/malfunction or the ensuing maintenance and whether post-modification functional testing of IROFS is required.

11.2.3 Preventive Maintenance

Preventive maintenance (PM) is performed in a preplanned and scheduled manner to refurbish or overhaul IROFS to ensure that they continue to perform their intended function. PM activities are appropriately balanced against the objective of minimizing unavailability of IROFS. Periodic calibrations are conducted where recommended by manufacturer or industry guidance. After conducting PM, and before returning a safety control to service, a functional test may be required to provide reasonable assurance that the safety control performs as designed and provides the safety action expected.

A schedule for performing PM on IROFS is maintained as specified in approved procedures, and frequencies are established based on operating history, manufacturer and industry guidance,

feedback from surveillance and maintenance activities, and/or recommendations from the corrective action program.

11.2.4 Functional Testing

Functional testing of IROFS is performed using approved written instructions prior to startup of facilities or process operations involving IROFS (pre-operational testing) and at periodic intervals during operations. This is intended to provide reasonable assurance that the safety control performs as designed and provides the desired safety action. Functional test instructions and frequencies are based on operating history, manufacturer and industry guidance, risk assessment, feedback from surveillance and maintenance activities, and/or recommendations from the corrective action program. During process operations, compensatory measures are used as appropriate while functional testing is performed on IROFS.

Administrative controls that are identified as IROFS are documented in approved procedures. Administrative controls are assured to available and reliable during operations by applying the applicable measures addressed in this chapter (e.g., procedures, training and qualifications).

11.2.5 Maintenance Records

The results of Surveillance and Monitoring, Corrective Maintenance, Preventive Maintenance, and Functional Testing for IROFS are documented, and the documentation is maintained as "records pertaining to safety" as specified in Section 11.7.

11.3 Training and Qualification

The Training and Qualification Program provides workers with the knowledge and skills to safely perform their job function, recognize the importance of IROFS, effectively deal with the hazards of the workplace, implement proper control and accounting of SNM, and properly respond to emergency situations. The qualification aspect of this program ensures that operations and maintenance are performed only by properly trained personnel.

Requirements and methods for the training and qualification programs are approved by TRISO-X management, who also provide ongoing evaluation of the effectiveness of the programs.

This training typically falls into one of the following categories:

1) General safety training not specific to a particular workstation or activity;
2) Training to assure proper performance for positions and work activities that are relied on for safety, in particular those designated as IROFS; and
3) Proper control and accounting of SNM.

11.3.1 General Safety Training

The Training and Qualification Program requires that all personnel who are granted unescorted access to the owner-controlled area receive formal safety and security orientation training. Safety orientation training covers facility security and safety rules, radiological, nuclear criticality, chemical, fire, and environmental safety topics as appropriate to the job function of the individuals being trained. In addition, this training covers proper response to emergencies.

Continuing training is conducted in these areas as necessary to maintain employee proficiency. The content of safety training is evaluated on a scheduled basis, as appropriate for the subject of the training, to ensure it remains current and relevant.

11.3.2 Training and Qualification for Activities Involving the Handling of SNM

The Training and Qualification Program includes work training for operating personnel and others who directly handle greater than laboratory sample quantities of special nuclear material. Facility specific activities are correlated with applicable supporting procedures and training materials. Work training typically includes classroom, on-the-job, and guided-work-experience training necessary to provide the desired knowledge and/or skill. It covers the operating procedures, alarms, emergency response actions, special nuclear material controls and accounting, and radiological, nuclear criticality, industrial, and environmental safety controls and limits specific to the particular work assignment.

Work training includes appropriate reinstruction for previously qualified individuals prior to implementation of a process change or procedural modification. When changes are made relative to safety or emergency response requirements, provisions are made to assure that affected employees are appropriately informed and instructed on the changes. Work training is evaluated, and necessary recurrent training / retraining / requalification is identified and documented. Additional details about the work training program are provided in approved procedures.

The Training and Qualification Program provides for the instruction and training of mechanics involved in maintenance activities. The type and level of training is commensurate with the job assignments.

Organization and Management of Training

The responsibility for the assurance of properly trained and qualified personnel resides with the discipline management team and pertinent line management. Support to line management for the development, implementation, and administration of the facility Training and Qualification Program is provided by the Training function. Implementation of the Training and Qualification Program is accomplished in accordance with approved procedures. All training is conducted by, or under the supervision of, individuals recognized by management as possessing the necessary knowledge and skills to conduct the training. Exemptions from training are only authorized as described in approved procedures.

Training records are maintained to support management information needs and provide required information on each individual's training and qualification. The records are maintained in accordance with approved procedures.

Identification of Activities Requiring Training

Positions impacting the availability/reliability of IROFS are assessed, based on a graded approach that considers the hazards and the safety responsibilities associated with each position. Input from subject matter experts, with support from the training function, is utilized as appropriate.

Position Training Requirements

Objectives and requirements for training programs are jointly agreed upon by management based upon facility needs and input provided by the training function and the appropriate discipline. Each position involving personnel assigned to SNM process operations is evaluated to determine the specific requirements that apply to the defined job function. Personnel must remain current on the defined set of requirements to maintain job qualifications.

Bases for Training

The objective of training is to ensure safe and efficient operation of the facility and compliance with applicable established regulations and requirements. Learning objectives are established for those positions/activities impacting the safety and security of licensed material operations, and in particular the availability/reliability of designated IROFS. Objectives include, as applicable, the knowledge skills, and abilities the trainee should demonstrate; the conditions under which required actions will take place; and the standards of performance the trainee should achieve upon completion of the training activity.

Training Materials

Lesson plans, computer-based training, and other training guides (for self-study, classroom, and on-the-job training) developed for activities relied on for safety and security are based on learning objectives developed from specific job performance requirements. Information provided, reviewed, and approved by subject matter experts is included in the content of training elements with clearly defined objectives. The lesson plans also provide reasonable assurance that training is conducted in a reliable and consistent manner. Lesson plans, guides and other training materials are reviewed and approved before their issuance and use. The CM Program provides a means to assure that design changes and modifications to IROFS are accounted for in the training and that personnel are instructed using current procedures.

Evaluation of Trainee Accomplishment

Trainee understanding and command of learning objectives are evaluated. The evaluation may be accomplished through a combination of observation/skills demonstration, written tests, or oral examinations. The results of trainee evaluations are documented.

On-the-Job Training (OJT)
OJT requirements for activities relied on for safety and listed in the ISA Summary, if applicable, are specified as part of pertinent position training requirements. Completion of OJT may be demonstrated by actual task performance (preferred) or task simulation. OJT is conducting by qualified individuals using current training materials. Completion of OJT is demonstrated through actual task actions (or simulation) using conditions encountered during the performance of assigned duties including the use of references and tools, and equipment conditions reflecting the actual task to the extent practicable. Completion of OJT requirements are documented.

Training Program Review
The effectiveness of the Training and Qualification Program is assessed on a periodic basis. Work assignments involving the handling of SNM are evaluated for needed recurrent training and/or reevaluation of qualification activities. Improvements and changes are made to training as needed to correct any deficiencies or performance problems.

11.3.3 Personnel Qualification
The minimum qualifications for key management and technical professional staff positions are described in Chapter 2. Qualifications for personnel who conduct activities involving the handling of SNM are described in Section 11.3.2.

11.4 Procedure Development and Implementation

Activities involving the handling of SNM and/or IROFS are conducted in accordance with approved procedures as defined in this section. Procedures address the following activities: design, configuration management, procurement, construction, operations, radiation safety, maintenance, waste management, quality assurance, training and qualification, audits and assessments, incident investigations, records management, nuclear criticality safety, fire safety, chemical process safety, environmental protection, and reporting requirements. Procedures also address the requirements contained within the Site Emergency Plan, Fundamental Nuclear Material Control Plan, and Physical Security Plan. Procedures are classified into the general categories of operating, general safety and emergency, and maintenance. Administrative procedures are used for activities that support the process operations, and do not include activities involving the handling of SNM and/or operating IROFS.

The process for the development, management, and implementation of procedures is defined in approved procedures. These procedures address how procedures are developed, reviewed, approved, distributed, revised, and deleted. Each procedure contains an identifying number, title, revision number, and date. The system ensures that the most current revisions of procedures are readily available to workers within their work areas (operating procedures), or in a centralized location accessible to all affected personnel, that any necessary training and qualification requirements are identified, and that the timeframe for which the procedure is valid is defined.

Procedures are approved by appropriate management personnel who are responsible for the activity governed by the procedure. Changes and/or revisions to procedures covering licensed material operations and/or IROFS are reviewed by the regulatory affairs functions, as appropriate, in accordance with the requirements of the CM program, as discussed in Section 11.1, to ensure that all associated activities and documentation (safety analyses, reviews, testing, training, etc.) are completed before procedural changes are implemented.

If any aspect of a procedure is unclear or incorrect as written, personnel are authorized to safely stop the operation and/or activity and contact management. In the event of an unusual incident, accident, significant operator error, equipment malfunction, or system modification, the applicable procedures are evaluated and revised as necessary.

11.4.1 Operating Procedures

Operating procedures are documents written to authorize the processing of radioactive material; and within these documents, detailed instructions for operation of equipment used in the process or activity, instructions for disposition of radioactive wastes, and limits and controls established for safety and regulatory purposes, including IROFS, are identified.

Operating procedures include the required actions and limits for startup, operation, and shutdown; actions necessary to prevent or mitigate accidents identified in the ISA Summary; and responses to alarms and applicable off-normal conditions, including failure of an IROFS. Operating procedures include provisions to place process operations in a safe condition if a step of the procedure cannot be performed as written. Workplace posting of limits and controls, training, and other communication devices are used, if appropriate, to enhance comprehension and understanding of operating procedures.

During operating procedure development, the technical accuracy is verified. Changes to existing operating procedures are evaluated to determine if the scope of the change warrants a walk-down and/or an independent verification/validation. New operating procedures are validated by operations staff to ensure that they can be performed as written.

11.4.2 General Safety and Emergency Procedures

General safety procedures outline health and safety practices that help maintain occupational radiation exposures at levels as low as reasonably achievable (ALARA). These procedures are generally applicable on a facility-wide basis to include safe work practices to control processes with licensed material, IROFS, and hazardous materials. Included in this category are the Emergency Plan implementing procedures and the Criticality control procedures. General safety procedures are reviewed and approved by the applicable regulatory affairs functions.

11.4.3 Maintenance Procedures

Maintenance of facility structures, systems and components is performed in accordance with approved procedures, documented instructions, checklists, or drawings appropriate to the circumstances that conform to applicable codes, standards, specifications, and other appropriate criteria. Maintenance program procedures ensure that corrective and preventive maintenance as well as functional testing are implemented for IROFS; that reviews for accuracy and completeness are conducted for work to be performed; and require the affected organizations to be notified prior to performing the maintenance work and at completion of the work. Procedures provide compensatory measures for IROFS that may be degraded or taken out-of-service during maintenance activities.

Procedures require work to be controlled through review of the planned work by the applicable regulatory affairs functions. The maintenance program identifies qualifications of personnel authorized to perform maintenance, specifications for replacement components as covered by CM, requirements for post maintenance testing, required records management of maintenance activities, and safe work practices applicable to the work to be performed.

11.4.4 Temporary Procedures

Approved temporary procedures are used when permanent procedures do not exist to:

1) Direct operations during testing, maintenance, and facility modifications;
2) Provide guidance in unusual situations not within the scope of permanent procedures; or,
3) Provide assurance of orderly and uniform operations for periods of short duration when the facility, a system, or a component is performing in a manner not covered by existing permanent procedures or has been modified or extended in such a manner that portions of existing procedures do not apply.

Temporary procedures are controlled, reviewed, and approved as specified by a written procedure and will not change an ISA except as authorized under 10 CFR 70.72. The review and approval process required for temporary procedures is the same as for other procedures, and a timeframe is defined for which the procedure is valid.

11.4.5 Periodic Reviews of Procedures

Procedures governing activities relied on for safety involving the handling of SNM and/or IROFS are reviewed periodically to ensure content remains current and relevant and that administrative IROFS remain available and reliable. The review frequency is defined in approved procedures and may be graded based on importance to safety. Emergency procedures are reviewed per the Emergency Plan required in Chapter 8. Safeguards procedures are reviewed per the Fundamental Nuclear Material Control Plan required in Chapter 12. Security related procedures are reviewed per the Physical Security Plan required in Chapter 13. The corrective action program (Section 11.6) includes provisions to assess the role of procedures in adverse conditions or events evaluated within the program. Corrections of procedural deficiencies are tracked to completion within the system.

11.5 Audits and Assessments

A program is in place for conducting audits and assessments of activities significant to facility safety, safeguards, and environmental protection that identifies responsibility for:

1) Determining the appropriate utilization of internal and/or external personnel for particular audit and assessment activities.
2) Assuring audit and assessment personnel have the expertise and background sufficient to successfully conduct audit and assessment activities.
3) Assuring audit and assessment personnel are sufficiently independent of the area being reviewed.
4) Verifying the utilization of an effective corrective action program to address findings of audits and assessments.

Audits and assessments are conducted for the areas of radiation safety, nuclear criticality safety, chemical safety, fire safety, environmental protection, quality assurance, configuration management, maintenance, training and qualification, procedures, incident investigation, and records management. The areas of emergency management, safeguards, and security are also audited and assessed in accordance with the Emergency Plan, Fundamental Nuclear Material Control Plan, and Physical Security Plan. Approved procedures and guidance used to plan, schedule, and perform the audits and assessments contain the following information:

- Activities to be reviewed.
- Frequency of reviews.
- Applicable guidance to be used in conducting the reviews.
- Responsibilities for each phase of the reviews.
- Instructions for recording the results, and recommending and approving actions to be taken.
- The levels of management to which results are reported.

Results, including findings and observations, are captured in the corrective action program. Corrective actions to prevent recurrence are assigned to owners, documented, and tracked to completion in accordance with the requirements specified in the corrective action program.

11.5.1 Internal Audits

Internal audits are compliance-based evaluation activities with an objective of verifying compliance of operations with established regulatory requirements, license commitments, and standard industry practice. Audits also ensure that administrative IROFS remain available and reliable to perform their intended safety function over extended periods of operation.
Members of the regulatory affairs functions, as described in Chapter 2, perform audits of activities involving the handling of SNM, including support areas, on a scheduled basis as defined in approved procedures.

Members of the Quality Assurance discipline periodically audit facility programs as directed by plant management.

11.5.2 Independent Assessments

Independent assessments are performance-based evaluation activities conducted to assess the effectiveness of health, safety, and environmental compliance functions in achieving their designated purpose, particularly in providing reasonable assurance of the availability and reliability of IROFS. The assessments are conducted using offsite groups or individuals not involved in the licensed activity.

11.6. Incident Investigations and Corrective Action

A corrective action program is implemented through approved procedures to investigate and document events for operations involving special nuclear materials, including those required to be reported under 10 CFR 70.50, 70.62, and 70.74. Events, including those with conditions adverse to safety, are reported, investigated, tracked, and corrective actions are assigned through a formal corrective action program. A graded, risk-based approach is used to establish the requirements for determining specific or generic root cause(s) and generic implication(s) of events.

Events are reviewed and classified based on the safety significance and regulatory compliance, including the impact on the health and safety of the public and the environment; impact on reliability or availability of safety controls and/or; and impacts to regulatory commitments.

11.6.1 Conduct of Incident Investigations

A graded, risk-based approach is applied to the assignment of the level of investigation; and, based on severity or potential severity of the event, the investigation may be conducted by one or more individual(s). Levels of investigation, as well as reviews and approvals, are assigned for events in accordance with approved procedures. Corrective actions are developed, documented, approved, and implemented. Measures to prevent recurrence and/or to control the affected work in progress may also be taken. Procedural guidance for conducting an investigation defines responsibilities for investigators and approvers; general methods for conduct of investigations; and requirements for report preparation, approval, and distribution. A graded, risk-based approach is applied to prioritize completion of corrective actions so that conditions adverse to safety are corrected as soon as practicable. The process used to monitor corrective actions also includes verification of completion, and as applicable, reviews of effectiveness and management attention for those corrective actions deemed ineffective.

Corrective actions generated from investigations are used to make corrections and improvements (i.e., "lessons learned") necessary to prevent or minimize single or common-mode failures. Details of the accident event sequence(s) are compared with accident sequence(s)

already considered in the ISA, and the ISA Summary will be modified to include evaluation of the risk associated with accidents of the type experienced.

Auditable records and documentation related to events, investigations, and root cause analysis are maintained as described in approved procedures. Procedures require maintenance of all documentation relating to events for two years (or for the life of the operation), whichever is longer. This documentation will also be used as part of a "lessons learned" program that may be applied to future operations of the facility.

11.7 Records Management

A records management system, as applied to licensed regulatory and quality assurance activities, is maintained in accordance with approved procedures.

Information related to occupational exposure of personnel to radiation, releases of radioactive materials to the environment, and other pertinent activities, are maintained in such a manner as to demonstrate compliance with license conditions and the relevant regulatory requirements of 10 CFR 20.

All records pertaining to safety are retained for at least two years unless longer retention is required by other regulatory or license specifications. For example, records of major changes implemented under 10 CFR 70.72 will be maintained until termination of the license. Major changes are defined in 11.1.4.

Records management procedures (a) assign responsibilities for records management, (b) specify the authority needed for records retention or disposal, (c) specify which records must have controlled access and provide the controls needed, (d) provide for the protection of records from loss, damage, tampering, theft, or during an emergency, and (e) specify procedures for ensuring that the records management system remains effective.

A functional organization is in place to ensure prompt detection and correction of deficiencies in the records management system or its implementation. The records management procedures provide the following instructions to ensure that:

- Records are prepared, verified, characterized, and maintained.
- Records are legible, identifiable, and retrievable for their designated lifetimes.
- Records are protected against tampering, theft, loss, unauthorized access, damage, or deterioration for the time they are in storage; and,
- Procedures are established and documented specifying the requirements and responsibilities for record selection, verification, protection, transmittal, distribution, retention, maintenance, and disposition.

Records are categorized by their relative importance to safety and/or regulatory compliance to identify record protection and storage needs and to designate the retention period for individual kinds of records. Records of IROFS failures are kept and updated in accordance with 10 CFR 70.62(a)(3). The decommissioning recordkeeping requirements of 10 CFR 70.25(g) are addressed in Chapter 10.

11.8 Other Quality Assurance (QA) Elements for IROFS

The quality system consists of the organizational structure, procedures, processes, and resources needed to implement quality management. The following elements, as appropriate, are applied on individual projects using a graded approach based on the degree of importance to safety.

1. Organization and Responsibilities
   Chapter 2 provides the commitments associated with the organizational structure, authority, and responsibilities to ensure that activities involving the handling of SNM and/or IROFS are performed safely and in compliance with license and regulatory requirements.

2. Quality Assurance Program
   The QA Program is based on, but is not limited to, applicable requirements and guidance in ISO 9001:2015, under the overall responsibility of the Quality Assurance discipline. Aspects of this program may be applied to IROFS using a graded approach based on the degree of importance to safety.

3. Design Control
   Design control is an element of the Configuration Management Program as described in Section 11.1 and in approved procedures.

4. Procurement Document Control
   Procurement documents include those necessary requirements to ensure that IROFS will be of the desired quality. These include the following, as appropriate:

   - Scope of work – description of services or items being procured.
   - Basic technical requirements including drawings, specifications, codes, and industrial standards with applicable revision data, test and inspection requirements, special requirements such as for designing, fabricating, cleaning, identification marking, erecting, packaging, handling shipping and storage.
   - QA requirements – the extent to which will depend upon the type and use of the item or services being procured.
   - Requirements for the control of nonconformances and changes, including provisions to control and report nonconformance and changes to products being delivered.
   - Requirements on sub-tier suppliers.

- Procurement documents and changes thereto are reviewed to ensure they include the appropriate requirements.
- Applicable 10 CFR 21, *Reporting of Defects and Noncompliance*, reporting requirements (if any).

5. Instructions, Procedures, and Drawings
Section 11.4 includes the commitment that "activities involving the handling of SNM and/or IROFS are conducted in accordance with approved procedures". This section also describes the process for developing and implementing procedures. Drawings are controlled under the Configuration Management Program as described in Section 11.1.

6. Document Control
A process is in place for developing, implementing, and revising documents to provide reasonable assurance that the appropriate documents are in use (refer to Sections 11.1 and 11.4). Document changes are reviewed for adequacy and approved for implementation by authorized personnel.

7. Control of Purchased Items and Services
The procurement of IROFS is controlled to ensure conformance with documented requirements. The controls provide the following, as appropriate: supplier (source) evaluation and selection; evaluation of objective evidence of quality furnished by the supplier; and examination of items or services upon delivery or completion.

Sourcing activities are planned and documented to ensure a systematic approach to the procurement process. Supplier selection is based, in part, on an evaluation of the supplier's capability to provide items or services in accordance with the requirements of sourcing documents.

Supplier nonconformances may be identified either by TRISO-X or by the supplier. Nonconforming items are not released for use until the nonconforming condition is reviewed and accepted by TRISO-X and implementation of the disposition is verified, except where otherwise controlled and documented according to approved procedures. Records of supplier nonconformance are maintained.

8. Identification and Control of Items
Controls are established to assure that only correct and accepted items are used or installed. Identification is maintained on the items or in documents traceable to the items, or in a manner that assures identification is established and maintained as described in this section.

Where specified, items having a limited operating or shelf life are identified and controlled to preclude use of items whose operating life or shelf life has expired.

Procedures provide for item identification consistent with the planned duration and conditions of storage.

9. Control of Special Processes

Where identified by Engineering, special processes affecting quality of items and services are controlled. Procedures, instructions, drawings, checklists, work orders, or other appropriate means control processes. These means assure that special process parameters are controlled, and the special environmental conditions are maintained.

10. Inspection

Acceptance testing and/or inspection is a part of the Configuration Management Program which ensures that IROFS meet specified requirements prior to initial use. The Surveillance and Monitoring, Preventive Maintenance, and Functional Testing functions, as described in Section 11.2, provide assurance that IROFS continue to meet specified requirements by assuring that these testing and inspection activities are scheduled and implemented.

11. Test Control

Acceptance testing and/or inspection is a part of the Configuration Management Program which ensures that IROFS meet specified requirements prior to initial use. The Surveillance and Monitoring, Preventive Maintenance, and Functional Testing elements, as described in Section 11.2, provide assurance that IROFS continue to meet specified requirements by assuring that these testing and inspection activities are scheduled and implemented.

12. Control of Measuring and Test Equipment

Measuring and Test Equipment (M&TE) used in activities affecting the availability or reliability of IROFS are controlled, calibrated, and adjusted at specified intervals to maintain equipment performance within required limits. Policies, plans, and procedures ensure that devices and standards used for measurement, tests, and calibration activities are of the proper type, range, and accuracy. Calibration control is not necessary for commercial devices such as rulers, tape measures, levels, and stop watches. A list of devices is established to identify those items within the calibration control system. This identification listing includes, as a minimum, the due date of the next calibration and any use limitations (when calibrated for limited use).

M&TE is calibrated at specified intervals or prior to use against equipment having a known valid relationship to nationally recognized standards. If no nationally recognized standard exists, the basis for calibration is documented. M&TE is properly handled and stored to maintain accuracy. When M&TE is found to be out of calibration, as-found data are recorded, and an evaluation is made and documented as to the validity of previous inspection, test results, and of the acceptability of items previously inspected or tested.

Out-of-calibration devices are tagged or segregated and are not used until recalibrated. When M&TE is consistently found to be out of calibration, it is repaired or replaced. Calibrations are also performed when personnel performing measurements and tests deem the accuracy of the equipment suspect. Records are maintained and equipment is suitably marked or otherwise identified to indicate its calibration status

13. Item Handling, Storage, and Shipping
Materials and equipment are handled, stored, and shipped in accordance with design and procurement requirements to protect against damage, deterioration, or loss. Special coverings, equipment, and protective environments are specified and provided where necessary for the protection of particular items from damage or deterioration.

14. Inspection, Test, and Operating Status
Acceptance testing and/or inspection is a part of the Configuration Management Program which ensures that IROFS meet specified requirements prior to initial use. The Surveillance and Monitoring, Preventive Maintenance, and Functional Testing elements, as described in Section 11.2, provide assurance that IROFS continue to meet specified requirements by assuring that these testing and inspection activities are scheduled and implemented. The Configuration Management and Purchasing Programs have provisions for identifying and controlling items, including IROFS, to provide reasonable assurance that incorrect or defective items are not used.

15. Control of Nonconforming Items
Items and related activities that do not conform to specified requirements are controlled to prevent inadvertent installation or use. Nonconforming items are segregated, when practical. When segregation is impractical or impossible due to physical conditions (for example, size, weight, or access limitations), other measures are employed to preclude inadvertent use of the item.

Nonconforming items are reviewed and dispositioned. Further processing, delivery, installation, or use of the nonconforming item is controlled pending an evaluation and approved disposition by personnel as authorized in approved policies, plans, and/or procedures, and documented notification to affected organizations is provided.

The responsibility and authority for the evaluation and disposition of nonconforming items is defined.

Nonconformance documentation identifies the nonconforming item, describes the nonconformance, contains the disposition and any re-inspection requirements, and contains the appropriate signatures approving the disposition.

16. Corrective Action
Reports of conditions adverse to safety are promptly identified and entered into the Corrective Action Program (see Section. 11.6), which provides a means to evaluate the

problem, identify the cause of the problem, assign appropriate corrective actions to be initiated, and track the corrective actions to closure. Prompt identification and effective corrective actions should provide reasonable assurance that repetition of the problem will be minimized.

17. Quality Assurance Records
The Records Management Program, as described in Section 11.7, has provisions for the identification, retention, retrieval, and maintenance of records that furnish evidence of the control of quality of IROFS.

18. Audits
Section 11.5 includes the commitments for scheduling and implementing audits and assessments.

NRC SPECIAL NUCLEAR MATERIAL LICENSE
SNM-XXXX
CHAPTER 12

MATERIAL CONTROL AND ACCOUNTING OF SPECIAL NUCLEAR MATERIAL
Table of Contents

MATERIAL CONTROL AND ACCOUNTING OF SPECIAL NUCLEAR MATERIAL

12.1 Fundamental Nuclear Material Control Plan

TRISO-X maintains a Fundamental Nuclear Material Control Plan (FNMCP) for licensed Category II – Fuel facilities/licensees authorized to possess Special Nuclear Material (SNM) of moderate strategic significance, as defined in 10 CFR 74.4. 10 CFR 70.22(b)requires that the application contain a full description of the program for control and accounting of SNM in the licensee's possession authorized by the license. The FNMCP demonstrates how compliance with the applicable requirements of 10 CFR 74.31, 74.33, 74.41, or 74.51 is accomplished. Based upon the possession limits identified in the license application, the requirements contained within 10 CFR 74.41, nuclear material control and accounting for special nuclear material of moderate strategic significance, is applicable to the TRISO-X Fuel Fabrication Facility. The FNMCP provides the full description of this program.

The TRISO-X FMNCP addresses commitments regarding the material control and accounting (MC&A) program in the following areas:

1. Management structure and personnel qualification and training
2. Measurement systems
3. Measurement-control system
4. Use of statistics to ensure requirements are met
5. Conduct of periodic physical inventories and reconciliation of book records to the results of the physical inventories
6. Item-control system
7. Shipper/receiver comparisons
8. Independent assessment of the MC&A program
9. Tamper-safing (Use of Tamper Indicating Devices)
10. Designation of material balance areas, item-control areas, and custodians
11. Resolving indications of loss, theft, diversion, or misuse of SNM
12. Assisting in the investigation and recovery of missing SNM
13. Recordkeeping

12.2 Fundamental Nuclear Material Control Plan Implementation

The requirements of the FNMCP are implemented through approved procedures and controlled software used for the accounting of SNM.

12.3 Amendment of the Fundamental Nuclear Material Control Plan

The Fundamental Nuclear Material Control Plan is maintained as needed. TRISO-X may modify this FNMCP without receiving prior NRC Approval provided the change does not decrease the effectiveness of the MC&A program. When such a change is made, the licensee notifies the NRC of the change within six months of the date the change is implemented by providing a report containing a description of the change and justification of why the change does not decrease the effectiveness of the program, in accordance with 10 CFR 74.32(c)(2)(ii). The licensee obtains NRC approval of modifications to the plan in accordance with 10 CFR 70.34 if the modification would result in a decrease the effectiveness of the MC&A program. The policies used to control changes to the FNMCP and ensure that only the current version is used are described in Chapter 11. Proposed changes to the plan that decrease its effectiveness will not be implemented without prior approval by the NRC.

NRC SPECIAL NUCLEAR MATERIAL LICENSE
SNM-XXXX
CHAPTER 13

PROTECTION OF SPECIAL NUCLEAR MATERIAL
Table of Contents

PROTECTION OF SPECIAL NUCLEAR MATERIAL

13.1 Physical Security Plan

TRISO-X maintains a Physical Security Plan (PSP) for licensed Category II – Fuel facilities/licensees authorized to possess special nuclear material (SNM) of moderate strategic significance, as defined in 10 CFR 74.4. The PSP maintains a physical protection system, as required by 10 CFR Part 73, "Physical Protection of Plants and Materials," and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material." The physical protection system provides reasonable assurance that activities involving the protection of SNM are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety. The PSP demonstrates compliance with the applicable requirements. The PSP provides the full description of the security program.

TRISO-X maintains a Safeguards Information Plan (SIP) that establishes, implements, and maintains an information protection system for Safeguards Information (SGI) generated for the TRISO-X Fuel Fabrication Facility to satisfy the requirements outlined in 10 CFR Part 73 and ensure SGI is protected from unauthorized disclosure.

13.2 Physical Security and SGI Plan Implementation

The requirements of the PSP are implemented through passive and active engineered features and approved procedures. The requirements of the SIP are implemented through approved procedures.

13.3 Amendment of the Plans

The Physical Security Plan is maintained as needed. TRISO-X may modify the PSP and/or SIP without receiving prior NRC Approval provided the change does not decrease the effectiveness of the security plan in accordance with 10 CFR 70.32(e). When such a change is made, the licensee notifies the NRC of the change within two months of the date the change is implemented by providing a report containing a description of the change and justification of why the change does not decrease the effectiveness of the security plan. The licensee obtains NRC approval of modifications to the plan in accordance with 10 CFR 70.34 if the modification would result in a decrease the effectiveness of the security plan. The policies used to control changes to the PSP and SIP that ensure that only the current version is used are described in Chapter 11. Proposed changes to a plan that decrease its effectiveness will not be implemented without prior approval by the NRC.