ENCLOSURE 2
ATTACHMENT 2

SHINE MEDICAL TECHNOLOGIES, INC.

SHINE MEDICAL TECHNOLOGIES, INC. APPLICATION FOR CONSTRUCTION PERMIT RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

PRELIMINARY SAFETY ANALYSIS REPORT CHANGES (MARK-UP)

99 pages follow

CHAPTER 5
COOLING SYSTEMS

CHAPTER 6
ENGINEERED SAFETY FEATURES

Acronyms and Abbreviations

Acronym/Abbreviation	Definition
°C	degrees Celsius
°F	degrees Fahrenheit
10 CFR	Title 10 of the Code of Federal Regulations
ac.	acre
AHA	acetohydroxamic acid
ALARA	as low as reasonably achievable
ANSI/ANS	American National Standards Institute/ American Nuclear Society
CAAS	criticality accident and alarm system
CAMS	continuous air monitoring system
CFR	Code of Federal Regulations
CP	Construction Permit
DBA	design basis accident
DOE	U.S. Department of Energy
ESF	engineered safety feature
ESFAS	engineered safety feature actuation system
gal.	gallon
GNEP	Global Nuclear Energy Partnership
ha	hectare
HAZOPS	hazards and operability study
HEU	highly enriched uranium
IE	initiating event
IF	irradiation facility
INL	Idaho National Engineering Laboratory
~~IROFS~~	~~items relied on for safety~~
ISA	integrated safety analysis

d. Shielding is used to minimize occupational exposures in normally-occupied areas of the facility from the radioactive materials contained within the process areas of the facility.
e. Ventilation systems for normally-occupied areas are separate from ventilation systems for areas containing radioactive materials. The ventilation system in the RCA is designed to pull air from the least contaminated areas to the most contaminated areas.
f. Areas, tanks, equipment, and piping that contain radioactive materials drain to criticality-safe sumps that are provided with leak detection to alert operators in the event of a breach of the primary fission product barrier or confinement areas.

1.2.3.2 Functional Safety Features

1.2.3.2.1 Radiological Safety

Shielding is used extensively to minimize personnel exposures. The IU cell walls are approximately six-foot thick reinforced concrete. This provides neutron and gamma shielding for workers above or adjacent to IU cells. The light water pool provides additional neutron and gamma shielding.

The concept of confinement is used in both the IF and the RPF to minimize the release and spread of contamination.

1.2.3.2.2 Criticality Safety

Criticality control is designed into the project by a combination of facility, systems, equipment, and processes. For operations outside the TSV, the facility is designed to meet the requirements of American National Standards Institute/American Nuclear Society (ANSI/ANS) 8.1 (ANS, 2007), including the double contingency principle. The hierarchy of controls is as follows:

a. The facility and equipment is designed so that significant quantities of fissionable material cannot be placed in a favorable configuration for criticality.
b. Engineered controls.
c. Administrative controls (e.g., limitations on allowed movements and processes involving special nuclear material [SNM]).

Fissile material is maintained in a shutdown state ($k_{eff} \leq 0.95$) in vessels and equipment in the facility except the TSV. The TSV is operated in a subcritical state.

1.2.4 DESIGN FEATURES AND DESIGN BASES

1.2.4.1 Design Bases for the SSCs

The SSCs at the SHINE facility are assigned a nuclear safety classification, as follows:

- Safety-related SSCs: Those SSCs that are relied upon to remain functional during normal conditions and during and following design basis events to ~~ensure~~assure:
  - The integrity of the primary system boundary (PSB)~~.~~;
  - The capability to shut down the TSV and maintain the target solution in a safe shut~~-~~down condition~~.~~;

- The capability to prevent or mitigate the consequences of accidents ~~that~~which could result in potential ~~off-site~~ exposures comparable to the applicable guideline exposures set forth in 10 CFR 20~~.~~;
- That the potential for an inadvertent criticality accident is not credible;
- That acute chemical exposures to an individual from licensed material or hazardous chemicals produced from licensed material could not lead to irreversible or other serious, long-lasting health effects to a worker or cause mild transient health effects to any individual located outside the owner controlled area; or
- That an intake of 30 mg or greater of uranium in soluble form by any individual located outside the owner controlled area does not occur.

- ~~Items relied on for safety (IROFS): Those SSCs, equipment, and activities of personnel that are relied upon to prevent or mitigate potential accidents at the facility that could exceed the performance requirements of 10 CFR 70.61 or to mitigate their potential consequences.~~

- Nonsafety-related: Those SSCs related to production and delivery of products or services that are not in the above ~~two~~ safety classification~~s~~.

The design bases and design for SHINE facility SSCs are addressed in Sections 3.1 and 3.5.

The SHINE production facility building is designed to withstand severe natural phenomena, including seismic events and tornado missiles. The building exterior wall structure is robust enough to remain intact following the impact of small aircraft, as defined in DOE (1996).

1.2.4.2 Functional Design Bases

1.2.4.2.1 Radiological Safety

A radiation protection program is provided to protect the radiological health and safety of workers and complies with the regulatory requirements in 10 CFR 19, 20, and 70. The Radiation Protection Program meets the requirements of 10 CFR 20, Subpart B, Radiation Protection Programs, and is consistent with the guidance provided in Regulatory Guide 8.2. This program is described in Subsection 11.1.2.

The radiation protection program includes an as low as reasonably achievable (ALARA) program. The facility's commitment to the implementation of an ALARA program is described in Subsection 11.1.3. The objective of the program is to make every reasonable effort to maintain personnel exposures to radiation as far below the dose limits of 10 CFR 20.1201 as is practical. The design and implementation of the ALARA program is consistent with the guidance provided in Regulatory Guides 8.2, 8.13, and 8.29. The operation of the SHINE facility is consistent with the guidance provided in Regulatory Guide 8.10.

Radiation monitoring and surveying are utilized to minimize the occupational dose to personnel. The program equipment and procedures are addressed in Subsection 11.1.4. This includes the use of area radiation monitors, continuous air monitors, the detection and monitoring of gaseous and liquid effluent release streams, control point monitoring, and the use of radiation surveys within the SHINE facility.

Occupational dose is also controlled and minimized through the use of dosimetry and exposure control. This includes the establishment of controlled areas within the facility, the use of access

Confinement is achieved through RV, radiological integrated control system (RICS), and biological shielding provided by the steel and concrete structures comprising the walls, roofs, and penetrations of the hot cells. Shielding of the hot cells is discussed in detail in Subsection 4b.2.2.

Confinement is also achieved by berms that limit the spread of hazardous chemical spills.

SSCs that perform an ESF function are classified as safety-related~~ or IROFS, as appropriate~~.

1.3.5 INSTRUMENTATION, CONTROL, AND ELECTRICAL SYSTEMS

The TSV process control system (TPCS) controls the operation of the TSV. The TSV is protected by the TSV reactivity protection system (TRPS). These are addressed in Sections 7a2.3 and 7a2.4, respectively.

Control and protection systems associated with the RPF are addressed in Section 7b.

Design features of the control consoles and display instrumentation, and the radiation monitoring systems for both the IU and the RPF, are addressed in Chapter 7. Radiation monitoring systems include the criticality accident and alarm system (CAAS), the radiation area monitoring system (RAMS), and the continuous air monitoring system (CAMS).

The SHINE facility has one common normal electrical supply system, which provides power to the IF, the RPF, and other support buildings. Power service is provided by the local utility via off-site feeds. A standby diesel generator provides power for asset protection to selected loads in the event of a loss of off-site power. These systems are described in Section 8a2.1.

Emergency electrical power for the SHINE facility is provided by a common emergency power system. A Class 1E uninterruptible electrical power supply system (UPSS) is provided for the facility. This system consists of two independent trains, each consisting of a 250 volts-direct current (VDC) battery system with associated charger, inverter, and distribution system. This system is described in Section 8a2.2.

1.3.6 TSV COOLING AND OTHER AUXILIARY SYSTEMS

Primary cooling for the TSV and related components is provided by the LWPS and the primary closed loop cooling system (PCLS). The TSV and related components are submerged in the light water pool. The LWPS is addressed in Sections 5a2.2 and 4a2.4. The PCLS is addressed in Section 5a2.2. The light water pool and primary closed loop cooling make-up system (MUPS) supports the LWPS and the PCLS. This system is addressed in Section 5a2.5.

Primary cooling for the RPF and removal of heat from both the LWPS and the PCLS is provided by the radioisotope process facility cooling system (RPCS). This system is discussed in Section 5a2.3.

Ventilation for both the IF and the RPF is provided by the RV. This ventilation system is described in Section 9a2.1.

Equipment and processes related to handling and storage of target solution are addressed in Section 9a2.2. Equipment and processes related to handling and storage of byproduct material and SNM are addressed in Section 9a2.5.

Figure 2.1-3 shows the boundaries and zones applicable to the project site. The square area near the center of the site within which all safety-related structures are located gives the rough location and size of the operations boundary in accordance with ANSI/ANS-15.7-1977 and ANSI/ ANS-15.16-2008. The Emergency Planning Zone is encompassed by the site boundary using the guidance in ANSI/ANS-15.16-2008, Regulatory Guide 2.6, Revision 1, 10 CFR 50.54, and Appendix E to 10 CFR 50.

The site boundary is the property line around the perimeter of the SHINE site in accordance with ANSI/ANS-15.7-1977 and ANSI/ANS-15.16-1982. The controlled area is the area within the site boundary in accordance with 10 CFR 20.1003~~ and 10 CFR 70.61(f)~~. In addition, the area directly under the facility operating license will be delineated by the site boundary.

Figure 2.1-4 shows the topography within the vicinity of the SHINE site. The finished site grade elevation is approximately 827 feet (ft.) (252 m) North American Vertical Datum of 1988 (NAVD 88). The project site and adjacent ground within a radius of approximately 1 mi. (1.6 km) is flat. Within a 5 mi. (8 km) radius from the SHINE site, topographic elevations range from approximately 755 ft. (230 m) NAVD 88 along the Rock River, to approximately 950 ft. (290 m) NAVD 88 to the east of the site (USGS, 1980). Therefore, the topography within a 5 mi. (8 km) radius ranges from approximately 72 ft. (21.9 m) below to approximately 123 ft. (37.5 m) above the SHINE site grade elevation.

The tallest building to be constructed at the project site is the production facility building, which at its highest point is approximately 58 ft. (17.7 m) above the site grade level. The top of the main exhaust stack for the production facility building is at 66 ft. (20.1 m) above the site grade level. Two buildings higher than 58 ft. (17.7 m) above ground level have been identified within 5 mi. (8 km) of the project site. These are St. Mary's Hospital, which is 78 ft. (23.8 m) high, and an associated clinic, which is 62 ft. (18.9 m) high. Both of these buildings are approximately 3.9 mi. (6.3 km) northeast of the SHINE site. However, given their distance from the site, neither of these buildings are expected to affect diffusion or dispersion of airborne effluents.

2.1.2 POPULATION DISTRIBUTION

This subsection describes the population distribution within 8 km (5.0 mi.) of the center point of the safety-related area at the SHINE site. The information includes estimates of the resident and transient populations for the most recent census year (2010) and projections of the resident and transient populations for the following future years:

- Year of submitting Construction Permit application (2013)
- Year of submitting Operating License application (2014)
- Five years after submitting Construction Permit application (2018)
- Five years after submitting Operating License application (2019)
- Approximate expected end of Operating License period (2045)
- Five years after approximate expected end of Operating License period (2050)

Estimates and projections of resident and transient populations around the project site are divided into five distance bands (concentric circles at 0 to 1 km (0 to 0.6 mi.), 1 to 2 km (0.6 to 1.2 mi.), 2 to 4 km (1.2 to 2.5 mi.), 4 to 6 km (2.5 to 3.7 mi.), and 6 to 8 km (3.7 to 5.0 mi.) from the center point of the safety-related area) and 16 directional sectors (with each directional sector centered on one of the 16 compass points). For each segment formed by the distance bands and directional sectors, the resident population was estimated using U.S. Census Bureau (USCB)

Acronyms and Abbreviations (cont'd)

Acronym/Abbreviation	Definition
gpm	gallons per minute
HAZOPS	Hazard and Operability Study
HCFD	hot cell fire detection and suppression system
HVAC	heating, ventilation, and air conditioning
Hz	hertz
I&C	instrument & control
ICBS	irradiation cell biological shielding
IEEE	Institute of Electrical and Electronics Engineers
IF	irradiation facility
IGS	inert gas control system
in.	inch
~~IROFS~~	~~item relied on for safety~~
ISA	Integrated Safety Analysis
IU	irradiation unit
kg/m^3	kilogram per cubic meter
kPa	kilopascal
kph	kilometers per hour
ksf	kilopounds per square foot
kW	kilowatt
kWh	kilowatt-hour
l	liters
l/min	liters per minute
lb/ft^3	pounds per cubic foot
LWPS	light water pool system

CHAPTER 3

DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

The design of the SHINE Medical Technologies, Inc. (SHINE) facility and systems are based on defense-in-depth practices. Defense-in-depth practices means a design philosophy, applied from the outset and through completion of the design, that is based on providing successive levels of protection such that health and safety are not wholly dependent upon any single element of the design, construction, maintenance, or operation of the facility. The net effect of incorporating defense-in-depth practices is a conservatively designed facility and systems that exhibit greater tolerance to failures and external challenges.~~ The risk insights obtained through performance of accident analysis can then be used to supplement the final design by focusing attention on the prevention and mitigation of the higher risk potential accidents.~~

3.1 DESIGN CRITERIA

The SHINE facility is licensed under Title 10 of the Code of Federal Regulations (10 CFR) Part 50. The following discussion describes the applicability of 10 CFR 50 to the primary structures, systems, and components (SSCs) within the facility.

A systems list is provided in Table 3.1-1.

~~Safety significant SSCs within the facility are separated into two classifications. The first classification relates to those SSCs which directly support the operation and/or safety of the irradiation facility (IF). Systems within this classification include those that are an active part of the irradiation process and also those that provide a confinement boundary of the IF. These SSCs must be capable of permitting safe operation and shutdown of the target solution vessel (TSV) and support systems during normal and abnormal credible events. These SSCs are designated as safety-related. The second classification relates to those SSCs that are outside the IF and in the radioisotope production facility (RPF) and involve handling, storage, processing, or transfer of special nuclear material (SNM), and/or hazardous chemicals. These SSCs are designated as items relied on for safety (IROFS).~~

The various codes and standards that are used as guidance for design of the facility SSCs are given in the Table 3.1-2. Unless otherwise noted, the version of the code listed is the version in effect six months prior to the Construction Permit application submittal date. Exceptions will be made in specific cases, e.g., the U.S. Nuclear Regulatory Commission (NRC) specifies an earlier version in a guidance document. If no version is specified, it is understood that the version is the version in effect six months prior to the Construction Permit application submittal date. NRC guidance documents used in the design of the SHINE facility are given in Table 3.1-3.

Design information for the complete range of normal operating conditions for various facility systems may be found throughout the PSAR.

Postulated initiating events and credible accidents that form the design bases for the SSCs located in the IF and the RPF are discussed in Chapter 13.

Chapters 6 and 7 discuss the design redundancy of SSCs to protect against unsafe conditions with respect to single failures of ESFs and control systems, respectively.

3.3 WATER DAMAGE

The design basis precipitations levels and flood levels and ground water levels for the SHINE facility are as follows:

- Design basis flood level: 50 feet (ft.) (15.2 meters [m]) below grade.
- Design basis precipitation level: at grade.
- Maximum ground water level: 50 feet (ft.) below grade.

Per Subsection 2.4.2.3, a local PMP event creates a water level about level with grade. The first floor of the building is at least 4 inches (in.) above grade; therefore, water will not infiltrate the door openings in the case of a local PMP event.

Per Subsection 2.4.3, a local PMF event creates a water level approximately 50 feet (ft.) (15.2 meters) below grade. The lowest point of the facility is 29 feet (ft.) (8.8 meters [m]) below grade; therefore, flooding does not cause any structural loading in the case of a local PMF event.

The impact of internal flooding is determined by the maximum flow rate and the volume of water available to feed the break. In many cases, no active response is assumed to terminate the flow and the entire volume of available water was assumed to spill into the SHINE facility. For water sources outside the building (fire water), automatic or operator actions are required to terminate the flow.

A water collection system is designed and installed to accommodate the total firefighting water volume. Water is not allowed to drain into the tank vaults or hot pipe trenches. Sloped floors and curbs are utilized to prevent water entry into these areas. Fire protection water is prevented from draining into the radioactive drain system with sloped floors and curbs.

All safety-related ~~and IROFS~~ equipment vulnerable to water damage is protected by locating it in flood-protective compartments and/or installing it at a minimum of 8 in. (20.3 cm) above the grade floor (elevation 0 ft. [0 m]), except for Fire Area 2 (see Figure 9a2.3-1), where this equipment is at a minimum of 12 in. (30.5 cm) above the grade floor.

3.3.1 FLOOD PROTECTION

This subsection discusses the flood protection measures that are applicable to safety-related SSCs for both external flooding and postulated flooding from failures of facility components containing liquid. A compliance review will be conducted of the as-built design against the assumptions and requirements that are the basis of the flood evaluation presented below. This as-built evaluation will be documented in a Flood Analysis Report.

3.3.1.1 Flood Protection Measures for Structures, Systems, and Components

Postulated flooding from component failures in the building compartments is prevented from adversely affecting plant safety or posing any hazard to the public. Exterior or access openings and penetrations into the SHINE facility are above the maximum postulated flooding level and thus do not require protection against flooding.

3.3.1.1.1 Flood Protection from External Sources

Safety-related components located below the design flood level are protected using the hardened protection approach described below. The safety-related systems and components are flood-protected because they are enclosed in a reinforced concrete safety-related structure, which has the following features:

a. Exterior walls below flood level are not less than 2 ft. (0.61 m) thick.
b. Water stops are provided in all construction joints below flood level.
c. Waterproof coating is applied to external surfaces exposed to flood level.
d. Roofs are designed to prevent pooling of large amounts of water in accordance with Regulatory Guide 1.102.

Waterproofing of foundations and walls of Seismic Category I structures below grade is accomplished principally by the use of water stops at expansion and construction joints. In addition to water stops, waterproofing of the SHINE facility is provided up to 4 in. (10.2 cm) above the plant ground level to protect the external surfaces from exposure to water. The flood protection measures that are described above also guard against flooding from on-site storage tanks that may rupture. Any flash flooding that may result from tank rupture drains away from the SHINE facility and causes no damage to facility equipment.

3.3.1.1.2 Compartment Flooding from Fire Protection Discharge

The total discharge from the failure of fire protection piping consists of the combined volume from any sprinkler and hose systems. The sprinkler system, if used, is capable of delivering a water density of 0.20 gallons per minute (gpm) (76 liters per minute [l/min]) over a 1500 square foot (ft^2) (139 square meters [m^2]) design area, therefore, the sprinkler system is calculated to have a flow rate of 300 gpm (1136 l/min). The hose stream is a manually operated fire hose capable of delivering up to 250 gpm (946 l/min). In accordance with NFPA 801 Section 5.10 (NFPA, 2008), the credible volume of discharge is sized for the suppression system operating for a duration of 30 minutes (min.), therefore, the total discharge volume is calculated as follows:

30 min. x (300 gpm [1136 l/min] + 250 gpm [946 l/min]) = 16,500 gallons (gal.) (62,459 liters [l]) = 2205 cubic feet ($ft.^3$) (62.44 cubic meters [m^3])

The depth of water is found by dividing the total discharge volume by the area. Water depth in the tank farm and supercell area:

2205 $ft.^3$ (62.44m^3) / 19,154 $ft.^2$ (1779.5 m^2) = 0.115 ft. (3.5 cm) = 1.38 in. (3.5 cm)

Therefore, the depth of water due to fire protection system discharge is less than the elevation that water sensitive safety-related ~~and IROFS~~ equipment is raised from the floor.

Water depth in Fire Area 2:
2205 $ft.^3$ (62.44 m^3) / 3600 $ft.^2$ (334.5 m^2) = 0.61 ft. (18.7 cm) = 7.35 in. (18.7 cm)

The elevation that water sensitive safety-related~~ and IROFS~~ equipment is raised from the floor is 12 inches (30.5 cm) in the irradiation area to provide greater margin. There is no fissile material stored in Fire Area 2, preventing the potential for inadvertent criticality.

Outside of the radiologically controlled area (RCA) there is limited water discharge from fire protection systems. Any water sensitive safety-related ~~and IROFS ~~equipment is installed a minimum of 8 in. (20.3 cm) above the floor slab at grade. The UPSS has two trains to provide redundancy. These trains are isolated from each other to prevent one train from being damaged by discharge of the fire protection system in the vicinity of the other train.

3.3.1.1.3 Compartment Flooding from Postulated Component Failures

Piping, vessels, and tanks with flooding potential in the safety-related portions of the SHINE facility are seismically qualified. There is no moderate-energy or high-energy piping where a break needs to be postulated. Water sensitive safety-related ~~and IROFS ~~equipment is raised at least 8 in. (20.3 cm) above the floor. The depth of water on the floor is limited to less than 8 in. (20.3 cm) by utilizing available floor space to spread the flood water and limiting the water volumes.

Analyses of the worst flooding due to pipe and tank failures and their consequences are performed in this subsection.

3.3.1.1.3.1 Potential Failure of Fire Protection Piping

The total discharge from the operation of the fire protection system bounds the potential water collection due to the potential failure of the fire protection piping.

3.3.1.1.3.2 Potential Failure of Light Water Pool

The light water pools in the irradiation unit cell area contain water filled to an elevation approximately equal to the top of the surrounding area floor slab. Given the robust design of the light water pool (approximately 6 ft. thick reinforced concrete) and the stainless steel liner, loss of a significant amount of pool water is not credible.

3.3.1.2 Permanent Dewatering System

There is no permanent dewatering system provided for the flood design.

3.3.2 STRUCTURAL DESIGN FOR FLOODING

Since the design PMP elevation is at the finished plant grade and the PMF elevation is approximately 50 feet (ft.) (15.2 meters [m]) below grade, there is no dynamic force due to precipitation or flooding. The lateral surcharge pressure on the structures due to the design PMP water level is calculated and does not govern the design of the below grade walls.

The load from build up of water due to discharge of the fire protection system in the RCA is supported by slabs on grade, with the exception of the mezzanine floor. Drainage is provided for the mezzanine floor in the RCA to ensure that the mezzanine slab is not significantly loaded. The mezzanine floor slab is designed to a live load of 125 pounds per square foot (610 kilograms per square meter). Therefore, the mezzanine floor slab is capable of withstanding any temporary water collection that may occur while water is draining from the mezzanine floor.

3.4.4 SEISMIC INSTRUMENTATION

3.4.4.1 Location and Description of Seismic Instrumentation

State-of-the-art solid-state digital instrumentation that enables the prompt processing of the data at the site is used. A triaxial time-history accelerometer is provided at essential locations.

3.4.4.2 Seismic Instrumentation Operability and Characteristics

The seismic instrumentation operates during all modes of facility operation. The maintenance and repair procedures provide for keeping the maximum number of instruments in service during facility operation.

The design includes provisions for inservice testing. The instruments are capable of periodic channel checks during normal facility operation and the capability for in-place functional testing.

3.4.5 SEISMIC ENVELOPE DESIGN FOR EXTERNAL HAZARDS

3.4.5.1 AIRCRAFT IMPACT ANALYSIS

The safety-related structures at the SHINE facility are evaluated for aircraft impact loading resulting from small aircraft which frequent the Southern Wisconsin Regional Airport (SWRA). The analysis consists of a global impact response analysis and a local impact response analysis.

The global impact response analysis is performed using the energy balance method, consistent with Department of Energy Standard DOE-STD-3014-2006 (DOE, 2006). The permissible ductility limit for reinforced concrete elements is in accordance with Appendix F of ACI 349 (ACI, 2007). The permissible ductility limit for truss members is determined from Chapter NB of AISC N690 (ANSI/AISC, 2012). The calculated values are then used to create the appropriate elastic or elastic-plastic load deflection curves. From these curves, the available energy absorption capacity of the structure at the critical impact locations is determined. The Challenger 605 was selected as the critical aircraft for the global impact analysis based on a study of the airport operations data. The Challenger 605 is evaluated as a design basis aircraft impact. The probabilistic distributions of horizontal and vertical velocity of impact are determined from Attachment E of Lawrence Livermore National Laboratory UCRL-ID-123577 (UCRL, 2007) to correspond to 99.5 percent of impact velocity probability distribution.

Over 20 impact locations are considered in the global evaluation. Each exterior wall that protects safety-related ~~or IROFS~~ equipment was evaluated for impacts at the center of the wall panel and at critical locations near the edge of the wall panel. Each roof that protects safety-related ~~or IROFS~~ equipment was evaluated for impacts at the end of the roof truss, at the center of the roof truss, at the center of the roof panel between trusses.

The local response evaluation was conducted using empirical equations in accordance with Department of Energy Standard DOE-STD-3014-2006 (DOE, 2006). The structure was shown to resist scabbing and perforation. A punching shear failure was not postulated based on Appendix F of ACI 349 (ACI, 2007). Scabbing and perforation thickness requirement was calculated using DOE-STD-3014-2006 (DOE, 2006).

3.5 SYSTEMS AND COMPONENTS

Certain systems and components of the SHINE facility are considered important to safety because they perform safety functions during normal operations or as required to prevent or mitigate the consequences of abnormal operational transients or accidents.

This section summarizes the design bases (DB) for design, construction, and operating characteristics of ~~all~~ safety-related (SR) ~~and items relied on for safety (IROFS)~~ structures, systems, and components (SSCs) in the SHINE facility.

In accordance with 10 CFR 50.2, design bases means that information which identifies the specific functions to be performed by a structure, system, or component of a facility and the specific values or ranges of values chosen for controlling parameters as "reference bounds" for design. These "reference bounds" are to include the bounding conditions under which SSCs must perform design basis functions. These bounding conditions may be derived from normal operation or any accident or events for which SSCs are required to function, including anticipated operational occurrences, design basis accidents, external events, natural phenomena, and other events specifically addressed in the regulations. These controlling parameter values may be (1) restraints derived from generally accepted "state-of-the-art" practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals.

The DB for systems and components required for safe operation and shutdown is established within the following three categories:

1. Design basis functions:

   a. ~~IROFS or s~~Safety-related functions performed by SSCs that are required to meet regulations, license conditions, orders, or technical specifications.
   b. Functions credited in the safety analysis to ensure safe shutdown of the facility is achieved and maintained, prevent potential accidents or mitigate the potential consequences of accidents which could result in consequences greater than applicable NRC exposure guidelines.

2. Design basis values:

   Values or ranges of values of controlling parameters established as reference bounds for the design to meet DB function requirements. These values may be:

   a. Established by an NRC requirement or,
   b. Derived from or confirmed by the safety analysis or,
   c. Selected by the designer from an applicable code, standard or guidance document.

3. Design basis criteria

Code-driven requirements established for the facility fall into the following categories (refer to Section 3.1 for a list of the codes, standards, and regulatory documents and references to the applicable sections for more detail):

a. Fabrication
b. Construction
c. Operations
d. Testing
e. Inspection
f. Performance
g. Quality

Codes include:

a. National consensus codes
b. National standards
c. National guidance documents

Systems and components within the IF are described in Section 3.5a. Systems and components within the RPF are described in 3.5b. There is some overlap of systems across these facility boundaries and they are discussed as appropriate to the limiting safety classification.

Sections 3.5a and 3.5b discuss the conditional application of Appendix A to 10 CFR 50 "General Design Criteria for Nuclear Power Plants," and 10 CFR 70.64 "Requirements for New Facilities or New Processes at Existing Facilities," as good design practice. Although not mandatory, these design criteria provide a rational basis from which to proceed. The Chapter 13 accident sequences for credible events define the design basis events (DBE). The SR ~~and IROFS~~ parameter limits for these events are detailed in Chapter 13. ~~Together, t~~The SR ~~and IROFS~~ parameter limits ensure that the associated DB, provided in this section, are met. Specific details on how the facility design or operation conforms to the DB are located in the individual sections of the PSAR.

Structures, systems, and components that are determined to have safety significance are designed, fabricated, erected, and tested commensurate with the criteria set forth in ANSI/ANS 15.8 R2005 (ANSI/ANS, 2005), "Quality Assurance Program Requirements for Research Reactors," as implemented by the SHINE Quality Assurance Program Description (QAPD). Appropriate records of the design, fabrication, erection, procurement, and testing of SSCs that are determined to have safety significance are maintained throughout the life of the plant.

The design addresses natural phenomena hazards, fire protection, environmental and dynamic effects, chemical protection, emergency capability (which includes: licensed material and hazardous chemicals; evacuation of on-site personnel; and on-site emergency facilities and services that facilitate the use of available off-site services), utility services, inspection, testing and maintenance, criticality safety, instrumentation and controls, and defense-in-depth. For a more detailed review of the application of individual DB, see PSAR Sections 3.2, 3.3, and 3.4 for natural phenomena hazards, environmental and dynamic effects; Chapter 8 for electrical utility services; Section 6b.3 for nuclear criticality safety in the production facility; Section 9a2.3 for fire protection; various system sections for chemical safety; Chapter 11 for radiation protection; and Chapter 12 for ~~management measures~~administrative controls for surveillance, maintenance, and testing.

SR ~~and IROFS~~ components and systems are qualified using the applicable guidance in the Institute of Electrical and Electronics Engineers (IEEE) Standard IEEE-323, 2003, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations." The qualification of each SR ~~and IROFS~~ component and system demonstrate that the SR ~~and IROFS~~ components and systems perform their safety function under the environmental and dynamic service conditions in which they are required to function and for the length of time the function is required. Additionally, nonsafety-related (NSR) components and systems are qualified to withstand environmental stress caused by environmental and dynamic service conditions under which their failure could prevent satisfactory accomplishment of the SR ~~or IROFS~~ safety functions.

Instrumentation and control (I&C) systems are provided to monitor variables and operating systems that are significant to safety over anticipated ranges for normal operation, for abnormal operation, for accident conditions, and for safe shutdown. These systems ensure adequate safety of process and utility service operations in connection with their safety function. Controls are provided to maintain these variables and systems within the prescribed operating ranges under all normal conditions. I&C systems are designed to fail into a safe state or to assume a state demonstrated to be acceptable if conditions such as loss of signal, loss of energy or motive power, or adverse environments are experienced.

The status and operation of SR hardware~~, SR and IROFS,~~ involving instrumentation that provides automatic prevention or mitigation of events are monitored by an integrated control system by means of an alarm. This integrated control system is appropriately isolated from SR ~~and IROFS~~ components. Consistent with IEEE-279-1971 (IEEE, 1971a), "Criteria for Protection Systems for Nuclear Power Generating Stations," the isolation devices are classified as part of the SR ~~or IROFS~~ boundary and are designed such that no credible failure at the output of the isolation device prevents the associated SR ~~or IROFS~~ component or system from meeting its specified safety function. The criteria contained in IEEE Standard 603-2009 (IEEE, 2009), "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," for separation and isolation of SR ~~and IROFS~~ systems and components are applied to the design. See Tables 7a2.2-1 and 7b.2-1.

Defense-in-Depth

The SHINE facility and system designs are based on defense-in-depth practices. The design incorporates a preference for engineered controls over administrative controls, independence to avoid common mode failures, and incorporates other features that enhance safety by reducing challenges to SR ~~and IROFS~~ components and systems. SR ~~and IROFS~~ systems and components are identified in Section 3.5a and 3.5b and are described in Chapters 4, 5, 6, 7, 8, and 9 as appropriate.

Single Failure

Mechanical, instrumentation, and electrical systems and components required to perform their intended safety function in the event of a single failure are designed to include sufficient redundancy and independence (in I&C systems this may be achieved by redundant channels and voted architecture) such that a single failure of any active component does not result in a loss of the capability of the system to perform its safety functions.

Mechanical, instrumentation and electrical systems and components are designed to ensure that a single failure, in conjunction with an initiating event, does not result in the loss of the system's ability to perform its intended safety function. The single failure considered is a random failure and includes any consequential failures in addition to the initiating event. An initiating event is a single occurrence, including its consequential effects, that places the plant or some portion of the plant in an abnormal condition. An initiating event and its resulting consequences are not a single failure. An initiating event can be a component failure, natural phenomenon, or external man-made hazard.

Active components are devices characterized by an expected significant change of state or discernible mechanical motion in response to an imposed demand upon the system or operation requirements. Examples of active components include switches, circuit breakers, relays, valves, pressure switches, motors, dampers, pumps, and analog meters.

Passive components are devices characterized by an expected negligible change of state or negligible mechanical motion in response to an imposed design basis load demand upon the system.

An active component failure is a failure of the component to complete its intended safety function(s) upon demand. Spurious action of a powered component originating within its automatic actuation of control systems is regarded as an active failure unless specific features or operating restrictions preclude such spurious action.

The design of SR systems (including protection systems) is consistent with IEEE Standard 379-2000 (IEEE, 2000), "Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems", and Regulatory Guide 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems."

The protection system is designed to provide two, three, or four instrumentation channels for each protective function and two logic train circuits. These redundant channels and trains are electrically isolated and physically separated in areas outside of the control room as required to avoid a common mode failure which affects redundant channels of trains. Thus any single failure within a channel or train does not prevent protective action at the system level, when required.

Design techniques such as physical separation, functional diversity, diversity in component design, and principles of operation, are used to the extent necessary to protect against a single failure.

3.5.1 CLASSIFICATION OF SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY

Systems and components in the SHINE facility are to be classified according to their importance to ~~nuclear~~ safety, quality levels, and seismic class. This section provides the top level guidance used in developing these classifications during preliminary design with the support of regulatory guidance reviews, HAZOPS, accident analysis, integrated safety analysis, and national consensus code requirements. Refer to Table 3.5-1 for a summary of SSC classifications developed facility-wide.

3.5.1.1 Nuclear Safety Classifications for SSCs

Certain SSCs of the SHINE facility are considered SR ~~or IROFS~~ because they perform safety functions during normal operations or as required to prevent or mitigate the consequences of abnormal operational transients or accidents. The purpose of this section is to classify SSCs according to the safety function they perform. In addition, design requirements are placed upon such equipment to ensure the proper performance of safety function, when required. A listing of these SSCs and their safety classifications are provided in Table 3.5-1.

SHINE uses a modified definition from 10 CFR 50.2 “Definitions” to develop the definition of SR SSCs, where appropriate~~, and utilizes a portion of 10 CFR 70.4 “Definitions” for the definition of IROFS SSCs~~.

3.5.1.1.1 Safety-related SSCs

Those SSCs that are relied upon to remain functional during normal conditions and during and following design basis events to ~~ensure~~assure:

a. The integrity of the primary system boundary;
b. The capability to shutdown the target solution vessel (TSV) and maintain the target solution in a safe shutdown (SSD) condition;~~ or~~
c. The capability to prevent or mitigate the consequences of accidents ~~that~~which could result in potential ~~off-site~~ exposures comparable to the applicable guideline exposures set forth in 10 CFR 20~~.~~;
d. That the potential for an inadvertent criticality accident is not credible;
e. That acute chemical exposures to an individual from licensed material or hazardous chemicals produced from licensed material could not lead to irreversible or other serious, long-lasting health effects to a worker or cause mild transient health effects to any individual located outside the owner controlled area; or
f. That an intake of 30 mg or greater of uranium in soluble form by any individual located outside the owner controlled area does not occur.

~~I3.5.1.1.2 Items Relied On For Safety~~

~~Items relied on for safety (IROFS) are those SSCs, equipment, and activities of personnel that are relied upon to prevent or mitigate potential accidents at the facility that would exceed the performance requirements of 10 CFR 70.61 or to mitigate their potential consequences. These SSCs are provided in Table 3.5-1.~~

3.5.1.1.2 NSR SSC

Nonsafety-related SSCs are those SSCs related to production and delivery of products or services that are not in the above ~~two~~ safety classification~~s~~.

3.5.1.2 Quality Assurance (Quality Group Classifications for SSCs)

Quality assurance requirements may be found in the SHINE QAPD.

The SHINE QAPD has been developed in accordance with ANSI/ANS 15.8-R2005 (ANSI/ANS, 2005), “Quality Assurance Program Requirements for Research Reactors,” and provides the following graded approach to quality.

3.5.1.2.1 QL-1

This quality level shall implement the full requirements of the QAPD in accordance with an approved Quality Assurance Plan (QAP). This quality level shall be applied to SR SSCs.

~~3.5.1.2.2 QL-2~~

~~This Quality Level shall be applied in conformance with an approved QAP and applies to the design of SSCs which are relied upon to limit the following in accordance with 10 CFR 70.61:~~

1. ~~The risk of nuclear criticality accidents with preventive controls and measures to ensure that under normal and credible abnormal conditions, all nuclear processes are subcritical, including use of an approved margin of sub-criticality for safety.~~

2. ~~The likelihood of occurrence of an event so that, upon implementation, the event is highly unlikely or its consequences are less than those listed below:~~
   - ~~An acute worker dose of 1.0 Sv (100 rem) or greater total effective dose equivalent (highly unlikely).~~
   - ~~An acute dose of 0.25 Sv (25 rem) or greater total effective dose equivalent to the public (highly unlikely).~~
   - ~~An intake to the public of 30 mg or greater of uranium in soluble form (highly unlikely).~~
   - ~~An acute chemical exposure to an individual from licensed material or hazardous chemicals produced from licensed material that could endanger the life of a worker or lead to irreversible or other serious, long-lasting health effects to the public (highly unlikely).~~

3. ~~The likelihood of occurrence of an event so that, upon implementation, the event is unlikely or its consequences is less severe than those listed below:~~
   - ~~An acute worker dose of 0.25 Sv (25 rem) or greater total effective dose equivalent (unlikely).~~
   - ~~An acute dose of 0.05 Sv (5 rem) or greater total effective dose equivalent to the public (unlikely).~~
   - ~~An acute chemical exposure to an individual from licensed material or hazardous chemicals produced from licensed material that could lead to irreversible or other serious, long-lasting health effects to a worker or mild transient health effect to the public (unlikely).~~
   - ~~A 24-hour averaged release of radioactive material outside the restricted area in concentrations exceeding 5000 times the values of Table 2 of the Appendix B to 10 CFR 20 (unlikely).~~

~~The performance criteria of 10 CFR 70.61 are formulated in terms of likelihood limits on each separate event sequence. The example guidelines given in Table 3.5-2 below are based on the acceptance criteria guidance on likelihood definitions provided by NUREG-1537, Chapter 13 which references NUREG-1520, Revision 1 for guidance. The subject table is extrapolated from that guidance.~~

~~Any risk or risk index method of likelihood evaluation using criteria as simple as those provided in the example above should not be relied on exclusively in deciding the acceptability of the likelihood of a given event sequence. Consideration of qualitative criteria, such as degree of defense in depth or independence of controls, may be used to alter decisions based on the example of simple semi-quantitative criteria presented here.~~

3.5.1.2.2 QL-~~3~~2

This quality level shall apply to NSR quality activities performed by the licensee, that are deemed necessary by SHINE to ensure the manufacture and delivery of highly reliable products and services to meet or exceed customer expectations and requirements.

3.5.2 SEISMIC CLASSIFICATION

Plant SSCs important to safety are designed to withstand the effects of a design basis earthquake (DBEQ) (see Section 3.4) and remain functional if they are necessary to ~~ensure~~assure:

1. The integrity of the primary system boundary~~.~~;
2. The capability to shutdown the TSV and maintain the target solution in a safe shutdown condition~~.~~;
3. The capability to prevent or mitigate the consequences of accidents ~~that~~which could result in potential ~~off-site~~ exposures comparable to the applicable guideline exposures set forth in 10 CFR 20~~.~~;
4. ~~The capability to prevent or mitigate potential accidents at the facility that could exceed the performance requirements in 10 CFR 70.61.~~That the potential for an inadvertent criticality accident is not credible;
5. That acute chemical exposures to an individual from licensed material or hazardous chemicals produced from licensed material could not lead to irreversible or other serious, long-lasting health effects to a worker or cause mild transient health effects to any individual located outside the owner controlled area;
6. That an intake of 30 mg or greater of uranium in soluble form by any individual located outside the owner controlled area does not occur; or

~~5~~7. They do not degrade the function and performance of any SR ~~or IROFS~~ SSC.

Plant SSCs, including their foundations and supports, that are designed to remain functional in the event of a DBEQ are designated as Seismic Category I, as indicated in Table 3.5-1.

Structures, components, equipment, and systems designated SR ~~or IROFS~~ (see Section 3.5.1.1 for a definition of safety classes) are classified as Seismic Category I.

SSCs co-located with Seismic Category I systems are reviewed and supported in accordance with II over I criteria. This avoids any unacceptable interactions between SSCs.

SSCs that must maintain structural integrity post-DBEQ, but are not required to remain functional are Seismic Category II.

All other SSCs that have no specific NRC regulated requirements are designed to local jurisdictional requirements for structural integrity and are Seismic Category III.

All Seismic Category I SSCs are analyzed under the loading conditions of the DBEQ and consider margins of safety appropriate for that earthquake. The margin of safety provided for safety class structures, components, equipment and systems for the DBEQ are sufficient to ensure that their design functions are not jeopardized. For further details of seismic design criteria refer to Section 3.4.

Table 3.5-1 System Classifications
(Sheet 1 of 4)

System Name	System Code	Highest Safety Classification Within System Scope[a]	Seismic Classification[b]	Quality Group
Safety-related (SR)~~and Items Relied on For Safety (IROFS)~~				
Facility Structure	FSTR	SR	Category I	QL-1
Subcritical Assembly System	SCAS	SR	Category I	QL-1
Light Water Pool System	LWPS	SR	Category I	QL-1
TSV Off-gas System	TOGS	SR	Category I	QL-1
Neutron Flux Detection System	NFDS	SR	Category I	QL-1
Noble Gas Removal System	NGRS	~~IROFS~~SR	Category I	~~QL-2~~QL-1
Process Vessel Vent System	PVVS	~~IROFS~~SR	Category I	~~QL-2~~QL-1
Criticality Accident and Alarm System	CAAS	~~IROFS~~SR	Category I	~~QL-2~~QL-1
~~Continuous Air Monitoring System~~	~~CAMS~~	~~IROFS~~	~~Category I~~	~~QL-2~~
Radiation Area Monitoring System	RAMS	SR	Category I	QL-1
Irradiation Cell Biological Shielding	ICBS	SR	Category I	QL-1
Radiologically Controlled Area (RCA) Ventilation (Zones 1, 2)	RVZ1	SR	Category I	QL-1
	RVZ2	~~IROFS~~SR	Category I	~~QL-2~~QL-1
TSV Reactivity Protection System	TRPS	SR	Category I	QL-1
Engineered Safety Features Actuation System	ESFAS	SR	Category I	QL-1
Uninterruptible Electrical Power Supply System	UPSS	SR	Category I	QL-1
Tritium Purification System	TPS	SR	Category I	QL-1
Radiological Integrated Control System	RICS	~~IROFS~~SR	Category I	~~QL-2~~QL-1
Hot Cell Fire Detection and Suppression System	HCFD	~~IROFS~~SR	Category I	~~QL-2~~QL-1
Mo Extraction and Purification System	MEPS	~~IROFS~~SR	Category I	~~QL-2~~QL-1
Target Solution Preparation System	TSPS	~~IROFS~~SR	Category I	~~QL-2~~QL-1

Table 3.5-1 System Classifications
(Sheet 2 of 4)

System Name	System Code	Highest Safety Classification Within System Scope[a]	Seismic Classification[b]	Quality Group
Uranyl Nitrate Conversion System	UNCS	~~IROFS~~SR	Category I	~~QL-2~~QL-1
Target Solution Cleanup (UREX)				
Thermal Denitration				
Production Facility Biological Shield System	PFBS	~~IROFS~~SR	Category I	~~QL-2~~QL-1
Radioactive Drain System	RDS	~~IROFS~~SR	Category I	~~QL-2~~QL-1
Radioactive Liquid Waste Evaporation and Immobilization	RLWE	~~IROFS~~SR	Category I	~~QL-2~~QL-1
Aqueous Radioactive Liquid Waste Storage	RLWS	~~IROFS~~SR	Category I	~~QL-2~~QL-1
RCA Material Handling Systems	RMHS	SR	Category I	QL-1
Other Facility Systems and Components				
Facility Instrument Air System	FIAS	NSR	Category III	~~QL-3~~QL-2
Facility Control Room	FCR	NSR	Category III	~~QL-3~~QL-2
Stack Release Monitoring	SRM	NSR	Category III	~~QL-3~~QL-2
Facility Fire Detection and Suppression	FFPS	NSR	Category III	~~QL-3~~QL-2
Neutron Driver Assembly System	NDAS	NSR	Category III	~~QL-3~~QL-2
Primary Closed Loop Cooling System	PCLS	NSR	Category II	~~QL-3~~QL-2
Primary Closed Loop Cooling and Light Water Pool Makeup System	MUPS	NSR	Category III	~~QL-3~~QL-2
Health Physics Monitors	HPM	NSR	Category III	~~QL-3~~QL-2
TSV Process Control System	TPCS	NSR	Category II	~~QL-3~~QL-2
Normal Electrical Power Supply System	NPSS	NSR	Category II	~~QL-3~~QL-2
Inert Gas Control	IGS	NSR	Category III	~~QL-3~~QL-2
Material Handling	MHS	NSR	Category II	~~QL-3~~QL-2
Solid Radioactive Waste Packaging	SRWP	NSR	Category II	~~QL-3~~QL-2

Table 3.5-1 System Classifications
(Sheet 3 of 4)

System Name	System Code	Highest Safety Classification Within System Scope[(a)]	Seismic Classification[(b)]	Quality Group
Material Control and Accountability System	MCAS	NSR	Category III	~~QL-3~~QL-2
Organic Liquid Waste Storage and Export	OLWS	NSR	Category II	~~QL-3~~QL-2
Radioisotope Process Facility Cooling System	RPCS	NSR	Category III	~~QL-3~~QL-2
Moly Isotope Product Packaging System	MIPS	NSR	Category III	~~QL-3~~QL-2
Standby Diesel Generator System	SDGS	NSR	Category II	~~QL-3~~QL-2
Radiologically Controlled Area (RCA) Ventilation Zone 3	RVZ3	NSR	Category III	~~QL-3~~QL-2
Facility Ventilation Zone 4	FVZ4	NSR	Category III	~~QL-3~~QL-2
Facility Integrated Control System	FICS	NSR	Category III	~~QL-3~~QL-2
Facility Potable Water System	FPWS	NSR	Category III	~~QL-3~~QL-2
Facility Compressed Air System	FCAS	NSR	Category III	~~QL-3~~QL-2
Facility Breathing Air System	FBAS	NSR	Category III	~~QL-3~~QL-2
Facility Inert Gas System	FIGS	NSR	Category III	~~QL-3~~QL-2
Facility Welding System	FWS	NSR	Category III	~~QL-3~~QL-2
Facility Roof Drains System	FRDS	NSR	Category III	~~QL-3~~QL-2
Facility Sanitary Drains System	FSDS	NSR	Category III	~~QL-3~~QL-2
Facility Data and Communications System	FDCS	NSR	Category III	~~QL-3~~QL-2
Facility Lightning Protection System	FLPS	NSR	Category III	~~QL-3~~QL-2
Facility Demineralized Water System	FDWS	NSR	Category III	~~QL-3~~QL-2
Facility Chilled Water Supply and Distribution System	FCHS	NSR	Category III	~~QL-3~~QL-2
Facility Acid Reagent Storage and Distribution System	FARS	NSR	Category II	~~QL-3~~QL-2
Facility Alkaline Reagent Storage and Distribution System	FLRS	NSR	Category II	~~QL-3~~QL-2

Table 3.5-1 System Classifications
(Sheet 4 of 4)

System Name	System Code	Highest Safety Classification Within System Scope[(a)]	Seismic Classification[(b)]	Quality Group
Facility Salt Reagent Storage and Distribution System	FSRS	NSR	Category II	~~QL-3~~QL-2
Facility Organic Reagent Storage and Distribution System	FORS	NSR	Category II	~~QL-3~~QL-2
Cathodic Protection System	CPS	NSR	Category III	~~QL-3~~QL-2
Emergency Lighting System	ELTG	NSR	Category II	~~QL-3~~QL-2
Facility Grounding System	FGND	NSR	Category III	~~QL-3~~QL-2
Lighting System	LTG	NSR	Category III	~~QL-3~~QL-2
Process Facility Wet Vacuum System	PFWV	NSR	Category III	~~QL-3~~QL-2
Process Facility Sampling System	PFSS	NSR	Category III	~~QL-3~~QL-2
Continuous Air Monitoring System	CAMS	NSR	Category III	QL-2

a) Safety classification accounts for highest classification in the system. Systems that are classified as safety-related may include both safety-related and nonsafety-related components. Only safety-related components will be used to satisfy the safety functions of the system, whereas nonsafety-related components can be used to perform non-safety functions. For example, there ~~safety classification for ventilation systems covers components within specified confinement boundaries only. There~~ are nonsafety-related components, such as fans, ~~which are NSR~~within the safety-related ventilation systems that perform nonsafety-related functions.

b) Seismic category may be locally revised to account for II over I design criteria and in order to eliminate potential system degradation due to seismic interactions.

3.5a.12.5 Radiation Monitoring System

Refer to Section 7a2.7 for a detailed discussion.

3.5a.12.5.1 RAMS Design Basis Functions

a) RAMS provide real time local and remote annunciation of area radiation in excess of preset limits.
b) RAMS must remain functional through DBAs.
c) Provide signal to initiate hot cell confinement.

3.5a.12.5.2 RAMS Design Basis Values

To be provided in the FSAR.

3.5a.12.6 Uninterruptible Electrical Power Supply

Refer to Section 8a2.2 for a detailed description.

3.5a.12.6.1 UPSS Design Basis Functions

a) Provide power when normal power supplies are absent.
b) Maintain power availability for a minimum of 120 minutes post-accident. The final mission time for the emergency power system will be determined as part of final design.
c) Remain functional through DBAs.

3.5a.12.6.2 UPSS Design Basis Values

a) The UPSS has a 30-year design life.

3.5a.12.7 Facility Structure

3.5a.12.7.1 FSTR Design Basis Functions

a) Provides a structure for ~~all~~ SR SSCs~~, IROFS,~~ and other systems.
b) FSTR provides protection from all external DBAs.

3.5a.12.7.2 FSTR Design Basis Values

a) Functions during and after normal operations, shutdown conditions, and DBAs.
b) FSTR has a 30-year design life.

Table 3.5a-1 Appendix A to 10 CFR 50 General Design Criteria Which Have Been Interpreted As They Apply to the SHINE Irradiation Facility
(Page 10 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 18—*Inspection and testing of electric power systems*. Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.	As Applied and Means of Compliance The SHINE facility electric power system important to safety, as identified by the ~~Integrated Safety Analysis (ISA)~~accident analysis (the UPSS), is designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems are designed with a capability to test periodically: 1) The operability and functional performance of the components of the systems, such as on-site power sources, relays, switches, and buses; and, 2) The operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the UPSS, the off-site power system, and the on-site power system. Refer to Section 6a and Chapter 8 for a detailed discussion.
Criterion 19—*Control room.* A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.	As Applied A control room is provided from which actions can be taken to operate the IUs safely under normal conditions and to maintain it in a safe condition under accident conditions. Means of Compliance The design of the control room permits safe occupancy during normal operational conditions. Radiation detectors, alarms, and emergency lighting are provided. Controls and instruments are available for equipment required to bring the facility to a safe shutdown condition. During accident conditions, operators will monitor safe shutdown status of the facility. Due to the inherently-safe design of the TSV and the other passive ESFs of the facility design, operator actions are not credited for achieving safe shutdown.

3.5b.1.2 Instrumentation and Electrical Equipment

Facility Seismic Category I instrumentation and electrical equipment (as identified in Table 3.5-1) is designed to resist and withstand the effects of the postulated DBEQ without functional impairment. The equipment remains operable during and after a DBEQ.

The magnitude and frequency of the DBEQ loadings that each component experiences are determined by its location within the plant. In-structure response curves at various building elevations have been developed to support design. Equipment such as batteries and instrument racks, control consoles, have test data, operating experience, and/or calculations to substantiate the ability of components and systems, to not suffer loss of function during or after seismic loadings due to the DBEQ.

This certification of compliance with the specified seismic requirements, including compliance with the requirements of IEEE Standard 344-1971 (IEEE, 1971b), “Recommended Practice for Seismic Qualifications of Class 1E Equipment for Nuclear Power Generating Stations,” is maintained as part of the permanent plant record in accordance with the SHINE QAPD.

3.5b.1.3 Radioisotope Production Facility ~~IROFS~~Safety-Related Mechanical Systems and Components

The ~~IROFS~~safety-related equipment and components within the radioisotope production facility are required to function during normal operations and during and following DBAs. This equipment is capable of functioning in the radioisotope production facility environmental conditions associated with normal operations and design basis accidents. Certain systems and components used in the ESF systems are located in a controlled environment. This controlled environment is considered an integral part of the ESF systems.

3.5b.1.4 Qualification Methods

3.5b.1.4.1 Instrumentation and Electrical Systems and Components

Environmental qualification of ~~IROFS~~safety-related electrical equipment is demonstrated by tests, analysis or reliance on operating experience. Testing is the preferred method of qualification. Qualification testing is accomplished either by tests on the particular equipment or by type tests performed on similar equipment under environmental conditions at least as severe as the specified conditions.

The equipment is qualified for normal and accident environments. Qualification data is maintained as part of the permanent plant record in accordance with the SHINE QAPD.

3.5b.1.4.1.1 Mechanical Systems and Components

Environmental qualification of safety-related mechanical systems and components is demonstrated by tests, analysis or reliance on operating experience. Testing is the preferred method of qualification. Qualification testing is accomplished either by tests on the particular equipment or by type tests performed on similar equipment under environmental conditions at least as severe as the specified conditions.

The equipment is qualified for normal and accident environments. Qualification data is maintained as part of the permanent plant record in accordance with the SHINE QAPD.

h) Monitor the ESF and ensure that ESF actuations go to completion.
i) The portion of RICS that monitors and controls ~~IROFS~~safety-related components and initiates ESF actuations must remain functional during DBAs.
j) Initiate the actuation of isolation dampers for RPF hot cells or glove boxes upon receipt of signals from the HCFD system.

3.5b.1.11.2 RICS Design Basis Values

a) Function during and after normal operations, shutdown conditions, and design basis accidents.
b) RICS has a 30-year design life.
c) RICS actuates the necessary isolation functions within the time required by the safety analysis with acceptable margin.

3.5b.1.12 Molybdenum Extraction and Purification System

3.5b.1.12.1 MEPS Design Basis Functions

a) Receive irradiated target solution, preheat solution, and extract the Mo.
b) Purify the Mo and prepare for transfer to the moly isotope product packaging system.
c) Concentrate the Mo solution through adsorption and evaporation.
d) Prevent inadvertent criticality through inherently safe design of equipment.

3.5b.1.12.2 MEPS Design Basis Values

a) Prevent inadvertent criticality and maintain primary fission product boundary during and after normal operations, shutdown conditions, and DBAs.
b) MEPS has a 30-year design life, with the exception of the columns, filters, and purification glassware, which are replaced frequently.

3.5b.1.13 Target Solution Preparation System

3.5b.1.13.1 TSPS Design Basis Functions

a) Prepare target solution (uranyl sulfate) by reacting uranium oxide and sulfuric acid.
b) Dissolve uranium metal in nitric acid to form uranyl nitrate and transfer the uranyl nitrate to UNCS.
c) Receive and store uranium oxide from the UNCS in storage racks to be used in preparation of future target solutions.
d) Prevent inadvertent criticality through inherent design of equipment.
e) Transfer and meter target solution into the TSV.

3.5b.1.13.2 TSPS Design Basis Values

a) Prevent inadvertent criticality and maintain primary fission product boundary during and after normal operations, shutdown conditions, and DBAs.
b) TSPS has a 30-year design life.

Table 3.5b-1 Baseline and General Design Criteria for Radioisotope Production Facility (Sheet 1 of 5)

BASELINE DESIGN CRITERIA 10 CFR 70.64	As Applied to SHINE
(1) *Quality standards and records.* The design must be developed and implemented in accordance with management measures, to provide adequate assurance that items relied on for safety will be available and reliable to perform their function when needed. Appropriate records of these items must be maintained by or under the control of the licensee throughout the life of the facility.	As Applied SSCs important to safety are designed, fabricated, erected, tested, operated, and maintained to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they are identified and evaluated to determine their applicability, adequacy, and sufficiency and are supplemented or modified as necessary to ensure a quality product in keeping with the required safety function. A quality assurance program is established and implemented in order to provide adequate assurance that these SSCs satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection and testing of SSCs important to safety are maintained by or under the control of SHINE throughout the life of the facility. Means of Compliance The SHINE facility uses a graduated Quality Assurance Program which links quality classification and associated documentation to safety classification and linked to the manufacturing and delivery of highly-reliable products. The quality classification and safety classifications are listed in this chapter. The SHINE QAPD provides details of the procedures to be applied. Refer to Chapter 12 for further discussion.
(2) *Natural phenomena hazards*. The design must provide for adequate protection against natural phenomena with consideration of the most severe documented historical events for the site.	As Applied and Means of Compliance SSCs important to safety are designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety function~~ classified as IROFS~~. The design bases for these SSCs reflect: a) Appropriate consideration of the most severe of natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time which the historical data have been accumulated. b) Appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena. c) The importance of the safety functions to be performed. The design basis and criteria are discussed in this subsection.

Table 3.5b-1 Baseline and General Design Criteria for Radioisotope Production Facility (Sheet 2 of 5)

BASELINE DESIGN CRITERIA 10 CFR 70.64	As Applied to SHINE
(3) *Fire protection.* The design must provide for adequate protection against fires and explosions	As Applied and Means of Compliance SSCs important to safety are designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials are used wherever practical throughout the unit, particularly in locations such as the confinement and control room. Fire detection and suppression systems of appropriate capacity and capability are provided and designed to minimize the adverse effects of fires on SSCs important to safety. Firefighting systems are designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these SSCs. Where necessary within zoned areas or where criticality and access are an issue, required systems are manually initiated by operations after review of a detection signal. The facility fire protection system and HCFD is designed such that a failure of any component of the system will not impair the ability of ~~IROFS~~SR SSCs to safely shut down and isolate the RPF or limit the release of radioactivity to the environment. Refer to Chapter 9 and Subsections 6a2.2.7 and 6b.2.6 for further discussion.
(4) *Environmental and dynamic effects.* The design must provide for adequate protection from environmental conditions and dynamic effects associated with normal operations, maintenance, testing, and postulated accidents that could lead to loss of safety functions.	As Applied and Means of Compliance SSCs important to safety are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. Due to the low temperature and low pressure nature of the SHINE processes, dynamic effects due to pipe rupture and discharging fluids are not applicable to the SHINE facility.
(5) *Chemical protection*. The design must provide for adequate protection against chemical risks produced from licensed material, facility conditions which affect the safety of licensed material, and hazardous chemicals produced from licensed material.	As Applied and Means of Compliance Chemical protection is provided by confinement isolation systems, liquid retention features, and the use of appropriate personal protective equipment (PPE). Refer to Subsection 6b.2.1 for further information.

Table 3.5b-1 Baseline and General Design Criteria for Radioisotope Production Facility (Sheet 3 of 5)

BASELINE DESIGN CRITERIA 10 CFR 70.64	As Applied to SHINE
(6) *Emergency capability*. The design must provide for emergency capability to maintain control of: (i) Licensed material and hazardous chemicals produced from licensed material; (ii) Evacuation of on-site personnel; and (iii) Onsite emergency facilities and services that facilitate the use of available offsite services.	As Applied and Means of Compliance SHINE will develop and maintain emergency procedures for each area that contains licensed material and hazardous chemicals produced from licensed material. These procedures include provisions for the evacuation of all personnel to an area of safety in the event of an alarm. The procedures also include conducting drills to familiarize personnel with the evacuation plan, designation of responsible individuals and organizations for the disposition of licensed material, evaluation of the cause of the alarm, and the placement of facilities, systems, instruments, tools and materials for use in such an emergency. The SHINE facility includes a Fire Brigade and Hazmat Response Area that provides a command center for the use of available off-site emergency services and personnel.
(7) *Utility services*. The design must provide for continued operation of essential utility services.	As Applied and Means of Compliance The SHINE facility provides a standby diesel generator for asset protection of selected systems. Refer to Section 8b for detailed information.
(8) I*nspection, testing, and maintenance*. The design of items relied on for safety must provide for adequate inspection, testing, and maintenance, to ensure their availability and reliability to perform their function when needed.	As Applied and Means of Compliance SHINE has provided access and controls for testing, maintenance and inspection of ~~IROFS~~SR SSCs. This is a general practice that is applied differently throughout the facility. Refer to Sections 4b, 6b, 7b and 9b for detailed information.
(9) *Criticality contro*l. The design must provide for criticality control including adherence to the double contingency principle.	As Applied and Means of Compliance SHINE includes criticality-safe by geometry process vessels, criticality-safe storage, as well as other passive engineering and administrative controls. Compliance with the requirements of criticality control including adherence to the double-contingency principle are described in detail in Section 6b.3.

Table 3.5b-1 Baseline and General Design Criteria for Radioisotope Production Facility (Sheet 4 of 5)

BASELINE DESIGN CRITERIA 10 CFR 70.64	As Applied to SHINE
(10) *Instrumentation and controls*. The design must provide for inclusion of instrumentation and control systems to monitor and control the behavior of items relied on for safety. (b) Facility and system design and facility layout must be based on defense-in-depth practices.[1] The design must incorporate, to the extent practicable: (1) Preference for the selection of engineered controls over administrative controls to increase overall system reliability; and (2) Features that enhance safety by reducing challenges to items relied on for safety. [1] As used in 10 CFR 70.64, Requirements for new facilities or new processes at existing facilities, defense-in-depth practices means a design philosophy, applied from the outset and through completion of the design, that is based on providing successive levels of protection such that health and safety will not be wholly dependent upon any single element of the design, construction, maintenance, or operation of the facility. The net effect of incorporating defense-in-depth practices is a conservatively designed facility and system that will exhibit greater tolerance to failures and external challenges. The risk insights obtained through performance of the integrated safety analysis can be then used to supplement the final design by focusing attention on the prevention and mitigation of the higher-risk potential accidents.	As Applied and Means of Compliance The SHINE facility and system designs are based on defense-in-depth practices. The design incorporates a preference for engineered controls over administrative controls, independence to avoid common mode failures, and incorporates other features that enhance safety by reducing challenges to ~~IROFS~~SR components and systems. System descriptions identify their associated ~~IROFS~~SR components and systems and additional design and safety features that provide defense-in-depth. Instrumentation and control (I&C) systems are provided to monitor and control the behavior of ~~items relied on for safety~~SR SSCs. These systems ensure adequate safety of process and utility service operations in connection with their safety function. Controls are provided to maintain these variables and systems within the prescribed operating ranges under all normal conditions. I&C systems are designed to fail into a safe state or to assume a state demonstrated to be acceptable if conditions such as loss of signal, loss of energy or motive power, or adverse environments are experienced. ~~IROFS~~SR SSCs are identified in Section 3.5 and are described in Chapters 4, 5, 6, 7, and 8 as appropriate.
Appendix A to 10 CFR 50 General Design Criteria as Stated	**As Applied to SHINE**
Criterion 61—*Fuel storage and handling and radioactivity control*. The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.	As Applied and Means of Compliance The target solution storage and handling, radioactive waste, and other systems that may contain radioactivity are designed to ensure adequate safety under normal and postulated accident conditions. These systems are designed: 1) With a capability to permit appropriate periodic inspection and testing of components important to safety. 2) With suitable shielding for radiation protection. 3) With appropriate confinement and filtering systems. Decay heat in the target solution does not require active cooling in the RPF. Refer to Sections 4b.4 and 9b.2 for detailed information.

Acronyms and Abbreviations (cont'd)

Acronym/Abbreviation	Definition
gpm	gallons per minute
gU/L	grams of uranium per liter
H_2	hydrogen gas
H_2SO_4	sulfuric acid
HNO_3	nitric acid
hp	horsepower
HVAC	heating, ventilation and air conditioning
HVPS	high voltage power supply
I	iodine
IF	irradiation facility
in.	inch
in. w.c.	inches of water column
in^3/min	cubic inches per minute
~~IROFS~~	~~item relied on for safety~~
IU	irradiation unit
k_{eff}	effective neutron multiplication factor
kg	kilogram
kg/batch	kilograms per batch
kg/h	kilograms per hour
kg/L	kilograms per liter
KI	Kurchatov Institute
$KMnO_4$	potassium permanganate
kPa	kilopascal
Kr-85	krypton-85

4b.1 FACILITY AND PROCESS DESCRIPTION

4b.1.1 INTRODUCTION

This chapter describes the design of the RPF and the processes employed within it. The primary function of the facility is to extract, purify, package, and ship medical isotopes.

4b.1.2 FACILITY DESCRIPTION

The facility design includes a number of intrinsic safety features that represent good engineering practice for nuclear processing facilities.

- Radioactive material processing areas and related SSCs are located in Category I structures designed to survive design basis earthquake loadings and other external events. Additionally, SSCs co-located with ~~IROFS~~safety-related SSCs are reviewed and supported in accordance with II over I criteria. This avoids any unacceptable interactions between SSCs that could degrade the confinement of radioactive or chemically hazardous materials and result in an uncontrolled release.
- Radioactive materials are contained within piping and processing systems. These systems are located within shielded hot cells and vaults. These hot cells are ventilated with systems that are independent of the occupied zone ventilations.
- Radioactive material transfers between hot cells and vaults are minimized to the extent possible.
- Tanks, piping, and equipment that contain fissile material are designed to be criticality-safe by geometry.
- Piping systems for radioactive liquid transfers between processing areas are contained within hardened, shielded pipe chases.
- Hot cells, tank vaults, and shielded pipe chases drain to sumps that include leak detection to alert operators to a breach of primary confinement.
- Tanks include overflow lines that are hard-piped to the criticality-safe collection tank in the low point of the building.
- Operating areas are monitored with the continuous air monitoring system (CAMS), radiation area monitoring system (RAMS), and a fail-safe criticality accident alarm system (CAAS).
- Hot cells are isolated from the building heating, ventilation, and air conditioning (HVAC) system upon detection of a leak, to prevent the spread of contamination.
- Radioactive materials are excluded from normally occupied areas, except for transfers within suitably shielded containers.

4b.1.3 PROCESS DESCRIPTION

The RPF has been divided into a number of systems that represent discrete areas for design development. The systems associated with processing are:

- Target solution preparation system (TSPS).
- Molybdenum extraction and purification system (MEPS).
- Uranyl nitrate conversion system (UNCS).
- Noble gas removal system (NGRS).
- Process vessel vent system (PVVS).
- Radioactive liquid waste evaporation and immobilization (RLWE).

Acronyms and Abbreviations (cont'd)

Acronym/Abbreviation	Definition
in.	inch
IU	irradiation unit
~~IROFS~~	~~item relied on for safety~~
kg	kilogram
km	kilometer
kW	kilowatt
lb.	pound
lb/min	pounds per minute
LCO	limiting condition for operation
LFL	lower flammable limit
LLL	low liquid level
LWPS	light water pool system
m	meter
MEPS	molybdenum extraction and purification system
mi.	mile
MM	million
MUPS	light water pool and primary closed loop cooling make-up system
N-16	nitrogen-16
NDAS	neutron driver assembly system
PCLS	primary closed loop cooling system
PFD	process flow diagram
PSAR	preliminary safety analysis report
psig	pounds per square inch gauge
PVVS	process vessel vent system
RCA	radiologically controlled area
RDS	radioactive drain system

5a2.3 SECONDARY COOLING SYSTEM

5a2.3.1 DESIGN BASIS

The secondary cooling system provides heat removal from the primary cooling system during normal TSV/IU operation and shut down. The secondary cooling system for the IUs is the RPCS, which also provides cooling water to the process systems.

This section focuses on the RPCS requirements with respect to the primary cooling system, which includes the LWPS and PCLS cooling loops, but discusses the other process systems the RPCS provides cooling to as well. There are eight IUs, each with an LWPS and PCLS cooling loop.

The RPCS provides cooling water to equipment within the RCA boundary in the facility. The RPCS is a closed loop cooling system that circulates cooling water to process system users within the RCA boundary and transfers the absorbed heat to the FCHS via a plate and frame heat exchanger within the RCA boundary. The RPCS is not a safety-related ~~or IROFS~~ system. If flow in the RPCS is interrupted leading to temperature rise in the process systems, process equipment requiring cooling is shut down until normal operation flow and temperatures can be reestablished. Safety is not adversely affected for other process systems if RPCS cooling is lost.

See Figure 5a2.3-1 for the process flow diagram of the RPCS. Table 5a2.3-1 gives specifications for the system.

Heat is generated by the SCAS and NDAS target chamber and is absorbed by the LWPS and PCLS cooling loops for each IU. The NDAS utilizes RPCS cooling for other components located within the IU cell. The LWPS water circulates in a cooling loop that includes a heat exchanger where it interfaces with the RPCS. The PCLS cooling water circulates in a closed cooling loop that includes a heat exchanger where it interfaces with the RPCS.

The heat in the RPCS is transferred to the FCHS, where it is then dissipated to the environment.

5a2.3.2 PROCESS FUNCTIONS

The process functions of the RPCS are:

- Remove at a minimum [Proprietary Information] per IU of heat from the LWPS and at a minimum [Proprietary Information] per IU of heat from the PCLS during full-power irradiation. The total heat the RPCS removes from the eight IU LWPS and PCLS loops is approximately [Proprietary Information].
- Remove a total of [Proprietary Information] from RCA systems.
- Maintain physical integrity of system pressure boundary.
- Maintain water quality to reduce corrosion and scaling.
- Maintain higher pressure than the primary cooling systems.

CHAPTER 6

ENGINEERED SAFETY FEATURES

Table of Contents

Acronyms and Abbreviations

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

Acronyms and Abbreviations (cont'd)

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

6b RADIOISOTOPE PRODUCTION FACILITY ENGINEERED SAFETY FEATURES ~~AND ITEMS RELIED ON FOR SAFETY~~

6b.1 SUMMARY DESCRIPTION ENGINEERED SAFETY FEATURES

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

Confinement

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

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

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

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

6b.2.1 CONFINEMENT

6b.2.1.1 Introduction

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

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

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

Confinement is also achieved by berms to confine the spills of hazardous chemicals.

6b.2.1.2 Confinement System and Components

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

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

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

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

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

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

Berms employ a passive confinement methodology. Passive confinement is achieved through a continuous boundary between the hazardous materials and the surrounding area. In the event of an accidental release, the hazardous liquid is confined to limit the exposed surface area of the liquid.

6b.2.1.3 Functional Requirements

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

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

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

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

Berms are designed to hold the entire contents of the container in the event that the container fails.

6b.2.1.4 Confinement Components

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

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

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

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

d. Simple administrative.

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

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

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

Management of the Nuclear Criticality Safety Program

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

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

Other aspects of the NCS program include:

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

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

Planned Response to Criticality Accidents

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

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

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

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

NCS Change Management

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

6b.3.1 CRITICALITY-SAFETY CONTROLS

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

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

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

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

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

Nuclear Criticality Safety Evaluations

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

Nuclear criticality safety analyses also meet the following:

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

- The NCS analyses are performed in accordance with the methods specified and incorporated in the configuration management program.
- Nuclear criticality safety controls and controlled parameters ensure that under normal and credible abnormal conditions, all nuclear processes are subcritical, including use of an approved margin of subcriticality for safety that is used.
- Process specifications incorporate margins to protect against uncertainties in process variables and against a limit being accidentally exceeded (ANSI/ANS, 2007a).
- Use of ANSI/ANS-8.7-1998 (R2007) (ANSI/ANS, 2007b), as it relates to the requirements for subcriticality of operations, the margin of subcriticality for safety, and the selection of controls. ~~required by 10 CFR 70.61(d).~~
- ANSI/ANS-8.10-1983 (R2005) (ANSI/ANS, 2005a), as modified by Regulatory Guide 3.71, as it relates to the determination of consequences of NCS accident sequences, is used.
- If administrative k_{eff} margins for normal and credible accident scenarios are used, NRC pre-approval of the administrative margins will be sought.
- Subcritical limits for k_{eff} calculations such that the margin is large compared to the uncertainty in the calculated values, and includes adequate allowance for uncertainty in the methodology, data, and bias to ensure subcriticality are used.
- Studies to correlate the change in a value of a controlled parameter and k_{eff} value are performed. The studies include changing the value of one controlled parameter and determining its effect on another controlled parameter and k_{eff}.
- The calculation of k_{eff} is based on a set of variables within the method's validated area of applicability.
- Trends in the bias support the extension of the methodology to areas outside the area or areas of applicability.

The NSCE procedure addresses requirements for:

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

<u>Geometry Tanks</u>

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

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

Program Element Name	Requirement Basis for Element
Criticality Safety Policy	Not applicable
Verification & Validation of NCS Codes	ANSI/ANS 8.1-1998
Nuclear Criticality Safety Evaluations	ANSI/ANS 8.1-1998 ANSI/ANS 8.19-1996 ~~10 CFR 70.61~~
Nuclear Criticality Training and Qualifications	ANSI/ANS 8.20-1991 R1999 10 CFR 19.12 ~~10 CFR 70.61~~ ~~10 CFR 70.62~~
Implementation of Criticality Safety Controls and Limits	ANSI/ANS 8.19-1996
Configuration Control / Change Control	10 CFR 50.59 ~~10 CFR 70.72~~
Audits and Inspections	ANSI/ANS 8.19-1996
Criticality Safety Non-Compliances: Investigating, Reporting, Tracking, and Trending	ANSI/ANS 8.19-1996 10 CFR 19.12 ~~10 CFR 70.61~~ ~~10 CFR 70.62~~
Criticality Safety Guidelines for Fire Fighting	Not applicable
Emergency Preparedness Plan and Response Procedures Manual	ANSI/ANS 8.23-1997 ANSI/ANS 8.19 ~~10 CFR 70.52~~
Criticality Detector and Alarms System	ANSI/ANS 8.3-1997 10 CFR 19.12 10 CFR 70.24
Testing and Calibration of Active-Engineered Controls	Not applicable
Criticality Safety Controls Program	Not applicable

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Acronyms and Abbreviations

Acronym/Abbreviation	Definition
1oo2	one out of two voting
2oo3	two out of three voting
A/E	alarm/events
AHR	aqueous homogeneous reactor
ANS	American Nuclear Society
BWR	boiling water reactor
CAAS	criticality accident alarm system
CAMS	continuous air monitoring system
CGD	commercial grade dedication
COTS	commercial off-the-shelf
CSSD	cold safe shut down
DCS	digital control system
DBE	design basis event
~~DID~~	~~defense in depth~~
EPRI	Electric Power Research Institute
ESD	emergency shut down
ESF	engineered safety feature
ESFAS	engineered safety feature actuation system
FFPS	facility fire protection system
FMEA	failure modes and effects analysis
FICS	facility integrated control system
GDC	general design criteria
GOTS	government off-the-shelf
HCFD	hot cell fire detection and suppression system
HFE	human factors engineering

Acronyms and Abbreviations (cont'd)

Acronym/Abbreviation	Definition
HMI	human machine interface
IEC	International Electrotechnical Commission
IEEE	The Institute of Electrical & Electronic Engineers
IF	irradiation facility
~~IROFS~~	~~items relied on for safety~~
~~ISA (1st occurrence)~~	~~integrated safety analysis~~
ISA ~~(2nd occurrence)~~	Instrumentation, Systems, and Automation Society
ISO	International Organization for Standardization
IU	irradiation unit
k_{eff}	effective neutron multiplication factor
LCO	limiting conditions for operation
LSSS	limiting safety system settings
LWPS	light water pool system
M	subcritical multiplication factor
MEPS	molybdenum extraction and purification system
MSV	mean square voltage
MUPS	light water pool makeup and purification system
NDAS	neutron driver assembly system
NDI	non-developmental items
NFDS	neutron flux detection system
NRC	United States Nuclear Regulatory Commission
OIT	operator interface terminal
PCLS	primary closed loop cooling system

The TPCS is discussed in greater detail in Section 7a2.3.

The TRPS and NFDS are discussed in greater detail in Section 7a2.4.

ESFAS is discussed in greater detail in Section 7a2.5.

The hardware and software descriptions including software flow diagrams for digital computer systems, description of how the operational and support requirements will be met, and a description of the methodology and acceptance criteria to establish and calibrate trip or actuation setpoints or interlock functions will be provided in the FSAR.

7a2.2.4 SYSTEM PERFORMANCE ANALYSIS

The IF instrumentation and controls have the capability to trip the IU and isolate the IU cell and TOGS shielded cell. The TRPS functions as the safety-related protection system for the PSB and performs the protective actions. The ESFAS performs the protective actions for isolation of the IU cell and TOGS shielded cell. This analysis discusses the safety-related TRPS design criteria and design basis.

Potential variables, conditions, or other items that will be probable subjects of technical specifications associated with the IF instrumentation and control systems are provided in Chapter 14.

7a2.2.4.1 IU Trip Design Basis

This section discusses design basis information for the IU trip functions and the ESFAS actuation, including those required by Section 4 of IEEE-603-2009. The IU trip is a protective function and is part of the overall protection and safety monitoring systems for the IF. The specific equipment design basis for the instrumentation and equipment used for the IU trip functions are discussed in Section 7a2.4. The ESFAS is a mitigative system and is part of the overall protection and safety monitoring systems for the IF. The specific equipment design basis for the instrumentation and equipment used for the ESFAS actuation are discussed in Section 7a2.5.

The following discussion relates to the design bases utilized for monitoring specific signal values for IU trips and ESFAS actuation, the requirements of performance, the requirements for specific modes of operation for the TSV and the documents generating the basis.

7a2.2.4.1.1 Safety Functions and Corresponding Protective/Mitigative Actions for Design Basis Events

Citation - Section 4a and 4b of IEEE-603-2009

A preliminary ~~integrated safety analysis (ISA)~~accident analysis has been completed and the results are detailed in Section 13a2. Conditions that result in an IU trip are discussed in Subsections 13a2.1.2, 13a2.1.3, 13a2.1.4, 13a2.1.8, and 13a2.1.9. These subsections correlate the accident condition to the IU trip.

7a2.2.4.1.2 Variables Monitored to Control Protective/Mitigative Action

Citation - Section 4d of IEEE-603-2009

The following variables are monitored for an IU trip or ESFAS actuation:

- TSV cover gas hydrogen concentration
- Neutron flux, source range and high range
- Primary closed loop cooling system (PCLS) temperature
- Status of manual emergency shut down (ESD) pushbuttons
- PCLS flow
- High radiation in RCA ventilation system

TRPS is discussed in Section 7a2.4.

ESFAS is discussed in Section 7a2.5.

7a2.2.4.1.3 Variable Monitored Having Spatial Dependence

Citation - Section 4f of IEEE-603-2009

The neutron flux is measured by three different flux detectors located nominally 120 degrees apart, surrounding the TSV, being located in the light water cooling pool. Variations in flux between these three detectors will be able to be observed by the operators.

7a2.2.4.1.4 Range of Transient and Steady-State Conditions During Normal, Abnormal, and Accident Conditions

Citation - Section 4g of IEEE-603-2009

Ranges of transient and steady-state conditions will be provided in the FSAR.

Discussions of trips are provided in Subsection 7a2.4.1.1

7a2.2.4.1.5 Functional Degradation of Safety System Performance

Citation - Section 4h of IEEE-603-2009

This section of the IEEE-603 describes what constitutes system malfunctions for safety-related and nonsafety-related devices. The safety-related systems are designed to consider those conditions having the potential for functional degradation in performance as described in ~~the ISA Summary and~~ Chapter 13. Manual ESD pushbuttons are also provided to allow analog IU cell trip independent of the DCSs.

7a2.2.4.2 Analysis

7a2.2.4.2.1 TSV Trip Function Conformance to Applicable Criteria

The TRPS performs an IU trip as a protective function as part of the protection system. The criteria for equipment selection are discussed in Section 7a2.4. The following discussions relate to conformance to criteria for the IU trip function.

7a2.2.4.2.1.1 General Functional Requirement Conformance

Citation - Section 5 of IEEE-603-2009, GDC-13, GDC-20

The TRPS initiates safe shutdown when the system detects an abnormal event. IU trips are discussed in Subsection 7a2.4.1.1. The monitored values and subsequent trips were determined in the ~~ISA~~accident analysis and provide a means to mitigate or reduce the consequences of the design basis event (DBE) to acceptable levels.

7a2.2.4.2.1.2 Single Failure Criterion Conformance

Citation - Section 5.1 of IEEE-603-2009, IEEE-379-2000

A postulated single failure in the TRPS or the ESFAS does not prevent an IU trip. This is accomplished by utilizing redundant or triplicate measurement devices, having multiple paths from measurement sources, utilizing diversity in measurement, and having system designs based on single failure criteria. The equipment requirements are discussed in Sections 7a2.4 and 7a2.5.

7a2.2.4.2.1.3 Independence for Control and IU Trip Conformance

Citation - Section 5.6 of IEEE-603-2009, GDC-24

The TRPS is a separate system, independent from the TPCS. The TRPS and TPCS are located in separate fire areas. See Section 9a2.3.

Where measured values come from shared components, such as the NFDS, there are appropriate electrical isolation modules in place. Electrical isolation is utilized for connections to the TRPS that could compromise the ability to perform its safety function.

IEEE 7-4.3.2-2010 provides the design criteria for safety systems including data processing function for interconnected computers.

The ESFAS is designed as two separate independent trains that are physically separated. The safety-related ESFs are separated per IEEE-384. The actuation of any one ESFAS train results in isolation of the IU and TOGS shielded cells and a subsequent IU trip and safe shutdown. ESFAS is independent of the TPCS and TRPS.

Table 7a2.2-1 Design Criteria for TSV Instrumentation and Control Systems (Sheet 1 of 12)

Design Criteria as Cited with Summary Intent	Design Bases As Applied to SHINE
IEEE Std 379-2000 (R2008) **IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems** **Abstract:** Application of the single-failure criterion to the electrical power, instrumentation, and control portions of nuclear power generating safety systems is covered in this standard. **Keywords:** actuator, cascaded failure, common-cause failure, design basis event, detectable failure, effects analysis, safety system, single-failure criterion, system actuation, system logic	As Applied This standard is applied to the design of the target solution vessel reactivity protection system (TRPS), engineered safety features (ESFs), ESF actuation system (ESFAS), and other systems, structures, or components (SSCs) that are identified as safety-related for the IF. Means of Compliance The TRPS uses a platform that has previously been approved for use by the NRC in safety systems. It is based on triple modular redundancy (TMR) of power supplies, processors, and input/output channels. Controls that are deemed safety-related identified in Section 13a ~~or the ISA Summary~~ are evaluated against the single-failure compliance for which the TRPS performs a monitor or control function. TRPS can also be tripped manually from the control room by an operator. The NFDS is similar to systems that have been deployed in research reactors throughout the United States. The present design of the NFDS utilizes three detectors, each on a separate channel. These channels are voted 2oo3 internally in the NFDS. The NFDS system then provides two sets of triplicate relays for use and input to the TRPS and TPCS that an out of operating limits excursion has occurred. The ESFAS system is designed as a redundant system designated as trains A and B. The ESFAS design incorporates safety relays whose function is to interrupt power to the active ESFs that have been identified in the Section 6a or ~~the ISA Summary~~13a. A trip of either train A or B performs the ESFAS function, and brings the IU cell in question to a safe isolation state. The ESFAS can be tripped automatically from the monitored redundant sensing devices that are part of the ESFAS. It can also be tripped manually from the control room by an operator.

Table 7a2.2-1 Design Criteria for TSV Instrumentation and Control Systems (Sheet 2 of 12)

Design Criteria as Cited with Summary Intent	Design Bases As Applied to SHINE
IEEE Std 577-2004 **IEEE Standard Requirements for Reliability Analysis in the Design and Operation of Safety Systems for Nuclear Facilities** **Abstract:** This standard sets forth minimum acceptable requisites for the performance of reliability analyses for safety related systems of nuclear facilities when used to address the reliability requirements identified in regulations and other standards. The requirement that a reliability analysis be performed does not originate with this standard. However, when reliability analysis is used to demonstrate compliance with reliability requirements, this standard describes an acceptable response to the requirements. **Keywords:** nuclear facilities, reliability analysis, safety systems	As Applied This standard is applied to the design of the TRPS, ESFAS, and other instrumentation SSCs that are identified as safety-related for the SHINE facility. Means of Compliance For safety functions identified in the Section 6a or 13a ~~or the ISA Summary~~, a reliability analysis of the proposed design solution is performed. This can be qualitative or quantitative in nature as described in the standard.
IEEE Std 603-2009 **IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations** **Abstract:** Minimum functional and design criteria for the power, instrumentation, and control portions of nuclear power generating station safety systems are established. The criteria are to be applied to those systems required to protect the public health and safety by functioning to mitigate the consequences of design basis events. The intent is to promote appropriate practices for design and evaluation of safety system performance and reliability. Although the standard is limited to safety systems, many of the principles may have applicability to equipment provided for safe shutdown, post accident monitoring display instrumentation, preventive interlock features, or any other systems, structures, or equipment related to safety. **Keywords:** actuated equipment, associated circuits, Class 1E, design, failure, maintenance bypass, operating bypass, safety function, sense and command features, sensor	As Applied This standard is applied to the design of the TRPS, ESFAS, and other SSCs that are identified as safety-related for the SHINE facility. It describes the minimum functional and design criteria for safety systems. It does not describe what systems are to be determined as safety systems. Means of Compliance For safety functions identified in the Section 6a or 13a ~~or the ISA Summary~~, the design conforms to the practices detailed in the standard. Exception The SHINE facility is not a nuclear power reactor and does not have all of the systems detailed in this standard. The intent of this standard is followed.

Table 7a2.2-1 Design Criteria for TSV Instrumentation and Control Systems (Sheet 3 of 12)

Design Criteria as Cited with Summary Intent	Design Bases As Applied to SHINE
IEEE Std 384-2008 **IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits** **Abstract:** The independence requirements of the circuits and equipment comprising or associated with Class 1E systems are described. Criteria for the independence that can be achieved by physical separation and electrical isolation of circuits and equipment that are redundant are set forth. The determination of what is to be considered redundant is not addressed. **Keywords:** associated circuit, barrier, Class 1E, independence, isolation, isolation device, raceway, separation	As Applied This standard is applied to the design of the TRPS, ESFAS, and other instrumentation SSCs that are identified as safety-related for the SHINE facility. It describes the minimum criteria for separation and independence of systems in a physical way. It does not describe what systems are to be separate and independent, only a means to do so. Means of Compliance For safety functions identified in the Section 6a or 13a~~ or the ISA Summary~~, the design conforms to the practices detailed in the standard. Exception The SHINE facility is not a nuclear power reactor and does not have all of the systems detailed in this standard. The intent of this standard is followed.
IEEE Std 323-2003 **IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations** **Abstract:** The basic requirements for qualifying Class 1E equipment and interfaces that are to be used in nuclear power generating stations are described in this standard. The principles, methods, and procedures described are intended to be used for qualifying equipment, maintaining and extending qualification, and updating qualification, as required, if the equipment is modified. The qualification requirements in this standard, when met, demonstrate and document the ability of equipment to perform safety function(s) under applicable service conditions including design basis events, reducing the risk of common-cause equipment failure. **Keywords:** age conditioning, aging, condition monitoring, design basis events, equipment qualification, harsh environment, margin, mild environment, qualification methods, qualified life, radiation, SR function, significant aging mechanism, test plan, test sequence, type testing	As Applied This standard defines the methods for equipment qualification when it is desired to qualify equipment for the applications and the environments to which it may be exposed. This standard is generally utilized for qualification of Class 1E equipment located in harsh environments, and for certain post-accident monitoring equipment, but it may also be utilized for the qualification of equipment in mild environments. Means of Compliance For safety functions identified in the Section 6a or 13a~~ or the ISA Summary~~, the design conforms to the practices detailed in the standard for those systems determined to be Class 1E and located in harsh environment. This describes those SSCs that reside within the IU cell. Not all of the safety components reside in the IU cell. As an example, isolation valve actuators that reside in the IU cell with electrical components are scrutinized with this standard, but the controlling system ESFAS that resides outside the IU cell does not require the same level of exposure per the application of this standard. The standard is applied in a graded approach based on the location of the equipment.

Table 7a2.2-1 Design Criteria for TSV Instrumentation and Control Systems (Sheet 5 of 12)

Design Criteria as Cited with Summary Intent	Design Bases As Applied to SHINE
IEEE Std 338-2012 **IEEE Standard for Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems** **Abstract:** The standard provides criteria for the performance of periodic surveillance testing of nuclear power generating station safety systems. The scope of periodic surveillance testing consists of functional tests and checks, calibration verification, and time response measurements, as required, to verify that the safety system performs its defined safety function. Post-maintenance and post-modification testing are not covered by this document. This standard amplifies the periodic surveillance testing requirements of other nuclear safety-related IEEE standards. **Keywords:** functional tests, IEEE 338, periodic testing, risk-informed testing, surveillance testing	As Applied This standard is applied to the design of the TRPS, ESFAS, and other instrumentation SSCs that are identified as safety-related for the SHINE facility. It describes the methods and criteria for establishing a periodic surveillance program. It does not describe what systems are to be separate and independent, only a means to do so. Means of Compliance For safety functions identified in the Section 6a or 13a~~ or the ISA Summary~~, the design conforms to the practices detailed in the standard.
IEEE Std 497-2010 **IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations** **Abstract:** Criteria are established in this standard for variable selection, performance, design, and qualification of accident monitoring instrumentation, and include requirements for display alternatives for accident monitoring instrumentation, documentation of design bases, and use of portable instrumentation. **Keywords:** accident monitoring, display criteria, selection criteria, type variables	As Applied The purpose of this standard is to establish selection, design, performance, qualification and display criteria for accident monitoring instrumentation. It provides guidance on the use of portable instrumentation and examples of accident monitoring display configurations. Means of Compliance For those monitoring functions determined to be required for the health and safety of public or workers during normal operation and for post-design base accident, the design conforms to the practices detailed in the standard.

Table 7a2.2-1 Design Criteria for TSV Instrumentation and Control Systems (Sheet 9 of 12)

Design Criteria as Cited with Summary Intent	Design Bases As Applied to SHINE
IEEE Std 1028-2008 **IEEE Standard for Software Reviews and Audits** **Abstract:** Five types of software reviews and audits, together with procedures required for the execution of each type, are defined in this standard. This standard is concerned only with the reviews and audits; procedures for determining the necessity of a review or audit are not defined, and the disposition of the results of the review or audit is not specified. Types included are management reviews, technical reviews, inspections, walk-throughs, and audits. **Keywords:** audit, inspection, review, walk-through	As Applied This standard provides minimum acceptable requirements for systematic software reviews. This standard describes organizational means for doing a review and documenting the findings. The TRPS is designed as a DCS, the TRPS has safety-related functions, and this standard applies to the development of the software for that system. Means of Compliance The TRPS is developed utilizing this standard.
ANS-10.4-2008 **Verification and Validation of Non-Safety-Related Scientific and Engineering Computer Programs for the Nuclear Industry** **Abstract:** This standard provides guidelines for the V&V of non-safety related scientific and engineering computer programs developed for use by the nuclear industry. The scope is restricted to research and other non-safety-related, noncritical applications. **Keywords:** software integrity level, software life cycle, V&V, validation, verification	As Applied The purpose of software V&V is to help the development organization build quality into the software during the software life cycle. This standard describes a means to verify and validate the software development for the nonsafety-related systems. It is utilized for any software development in the SHINE facility that is not safety significant, i.e. nonsafety-related. Means of Compliance Nonsafety-related software is developed utilizing this standard.
ISA 67.04.01-2006 **Setpoints for Nuclear Safety-Related Instrumentation** **Abstract:** This standard defines the requirements for assessing, establishing, and maintaining nuclear SR and other important instrument setpoints associated with nuclear power plants or nuclear reactor facilities. **Keywords:** Setpoint, drift, analog channel, reliability analysis	As Applied This standard is applied to the design of the TRPS and other instrumentation SSCs that are identified as safety-related for the SHINE facility. It describes the methods and criteria for establishing setpoints utilized in safety-related systems and maintaining the documentation thereafter. Means of Compliance For any safety function identified in the Section 6a or 13a ~~or the ISA Summary~~ that is designed with inherent setpoints, the design conforms to the practices detailed in the standard.

Table 7a2.2-2 IF Verification Matrix Design Criteria, Bases, Description (Sheet 6 of 10)

Verification Matrix	Design Criterion	Design Basis Applicability	Detailed Section Reference	Functional Means
29	10CFR 50, Appendix A, GDC 13 Instrumentation and control	1) TRPS 2) NFDS 3) ESFAS 4) TPCS 5) TRPS display 6) TPCS display 7) TRPS end devices 8) ESFAS end devices 9) TRPS manual trip 10) ESFAS manual operator panel shutdown	1) 7a2.4 2) 7a2.4 3) 7a2.5 4) 7a2.3 5) 7a2.6 6) 7a2.6 7) 7a2.4 8) 7a2.5 9) 7a2.4 10) 7a2.5	This criterion directly addresses the instrument and controls. The instrumentation and controls provide control, protection, and means to safely mitigate the identified events described in ~~the ISA Summary~~Section 13a.
30	10CFR 50, Appendix A, GDC 15 Reactor coolant system design	1) TRPS 2) TRPS display 3) TRPS end devices 4) TRPS manual trip	1) 7a2.4 2) 7a2.6 3) 7a2.4 4) 7a2.4	This criterion directly addresses the coolant system. The instrumentation and controls provide control, protection, and means to safely mitigate the identified events described in ~~the ISA Summary~~Section 13a.
31	10CFR 50, Appendix A, GDC 16 Containment design	1) ESFAS 2) ESFAS end devices 3) ESFAS manual operator panel shutdown	1) 7a2.5 2) 7a2.5 3) 7a2.5	This criterion directly addresses the containment system. The SHINE facility does not have containment, but has confinement per NUREG-1537 definitions. The ESFAS provides monitoring of radioactivity in the IU and TOGS shielded cell ventilation. In the event radiation levels exceed predetermined values, the control and protection shall shutdown and isolate the process.
32	10CFR 50, Appendix A, GDC 17 Electric Power Systems	1) TRPS 2) NFDS 3) TRPS display 4) TRPS end devices	1) 7a2.4 2) 7a2.4 3) 7a2.6 4) 7a2.4	This criterion directly addresses the electric power system. The instrumentation and control and active ESFs are designed to fail-safe with loss of power. Upon complete loss of electric power, the de-energized state for the target solution in the TSV is to be directed to the geometrically-safe dump tank. The safety-related UPSS provides power upon loss of utility power to safety-related systems that allow for monitoring and maintaining a safe shutdown condition. See Table 3.5a-1.

Table 7a2.2-2 IF Verification Matrix Design Criteria, Bases, Description (Sheet 7 of 10)

Verification Matrix	Design Criterion	Design Basis Applicability	Detailed Section Reference	Functional Means
33	10CFR 50, Appendix A, GDC 19 Control Room	1) TRPS 2) NFDS 3) ESFAS 4) TPCS 5) TRPS display 6) TPCS display 7) TRPS end devices 8) ESFAS end devices 9) TRPS manual trip 10) ESFAS manual operator panel shutdown	1) 7a2.4 2) 7a2.4 3) 7a2.5 4) 7a2.3 5) 7a2.6 6) 7a2.6 7) 7a2.4 8) 7a2.5 9) 7a2.4 10) 7a2.5	This criterion directly addresses the control room for the facility. The instrumentation and controls provide control, protection, and means to safely mitigate the identified events described in ~~the ISA Summary~~Section 13a. It affords the ability to have a manual initiated shutdown for the operator.
34	10CFR 50, Appendix A, GDC 20 Protection system functions	1) TRPS 2) NFDS 3) ESFAS 4) TRPS display 5) TRPS end devices 6) ESFAS end devices 7) TRPS manual trip 8) ESFAS manual operator panel shutdown	1) 7a2.4 2) 7a2.4 3) 7a2.5 4) 7a2.6 5) 7a2.4 6) 7a2.5 7) 7a2.4 8) 7a2.5	This criterion directly addresses the protection systems for the facility. The TRPS provides protection as the safety-related protection system and the ESFAS provides mitigation as the isolation system. Both systems automatically trip upon appropriate signal which safely mitigates the identified events in ~~the ISA Summary~~Section 13a. The TRPS and the ESFAS include the ability to have a manual initiated shutdown and isolation for the operator.
35	10CFR 50, Appendix A, GDC 21 Protection system reliability and testability	1) TRPS 2) NFDS 3) ESFAS 4) TRPS display 5) TRPS end devices 6) ESFAS end devices 7) TRPS manual trip 8) ESFAS manual operator panel shutdown	1) 7a2.4 2) 7a2.4 3) 7a2.5 4) 7a2.6 5) 7a2.4 6) 7a2.5 7) 7a2.4 8) 7a2.5	This criterion directly addresses the protection systems and the ability to tolerate single failure of any component and the requirement for online surveillance of any channel used for safety. The system is presently designed as either dual or triplicate sensing for the safety-related sensing measurements. For the triplicate systems, namely analog channels, there is online surveillance. For the dual redundant systems, namely discrete inputs, there is to be a periodic surveillance schedule developed for the facility.

Specified CAMs/RAMs also trigger the ESFAS or RICS. The ESFAS or RICS incorporates safety relays whose function is to interrupt power to the active ESFs that have been identified in ~~the ISA Summary~~Chapter 13. The ESFAS or RICS can be tripped manually from the control room by an operator.

Refer to Subsection 7a2.2.4 for discussion of safety function performance analysis.

Potential variables, conditions, or other items that will be probable subjects of technical specifications associated with the radiation monitoring instrumentation are provided in Chapter 14.

7b RADIOISOTOPE PRODUCTION FACILITY INSTRUMENT & CONTROL SYSTEM

7b.1 SUMMARY DESCRIPTION

Within the SHINE facility, the RPF houses the extraction, purification, packaging, target solution preparation and cleanup, and waste treatment systems. The systems are enclosed predominately by hot cells and glove box designs. The RICS provides for monitoring and control of ~~items relied on for safety (IROFS) and ESFs~~safety-related components (including ESFs) within the RPF. The RICS also provides process monitoring and control of the nonsafety-related systems within the RPF.

7b.1.1 RICS DESCRIPTION (~~IROFS~~SR/ESF)

The RICS is a DCS that monitors and controls ~~IROFS and ESFs~~SR components (including ESFs) within the RPF. When the monitored safety parameters exceed normal conditions, the RICS provides mitigative action by activating the ESF for the affected area. The ESFs in the RPF provide isolation functionality and alert operators of potential contamination events. The RICS can isolate one or any combination of the isolable hot cells in the RPF. The RICS also initiates the ESF isolation between ventilation zones in the RCA. This system is further described in Subsection 7b.2.3 and Section 7b.4.

7b.1.2 RICS DESCRIPTION (PROCESS CONTROL)

The RICS performs as the overall production process controller. It monitors and controls the required instrumented functions within the RPF. This includes monitoring of process fluid transfers and controlled inter-equipment pump transfers of process fluids. This system is further described in Section 7b.3.

7b.1.3 RADIATION MONITORING

The RPF utilizes CAMS, RAMS, and the criticality accident alarm system (CAAS) for continuous monitoring of processes. The CAMS, RAMS, and CAAS are strategically placed throughout the RPF to alert personnel of any potential radiation hazards. The CAMS, RAMS, and CAAS monitor the RPF for radiation and perform alarming in the control room and locally at locations throughout the RPF. The CAAS is further described in Section 7b.6. The RAMS and CAMS are further described in 7a2.7.

Specific CAMs/RAMs channels provide input to RICS for ESF functions.

7b.1.4 CONTROL ROOM AND INSTRUMENT DISPLAYS

The SHINE RPF is monitored and controlled from a centralized control room. The RICS has separate dedicated annunciation, alarming, and operator interface displays. The RICS operator panels and displays are electrically isolated and independent components. Within the control room, there are RICS consoles that are redundant in nature and can be operated simultaneously and independently. From these consoles, an operator can assess the state of a hot cell and other process enclosures within the RPF. The operator can view and trend essential measurement values from the operator interface display. From the RICS operator workstation, the operator controls many of the RPF processes that are not performed through manual means (such as radioisotope purification). The operator is provided real-time data from the essential measurements used to control and monitor the RPF process on the RICS displays. This system is further described in Section 7b.5.

Table 7b.2-1 Design Criteria for the RPF Instrumentation and Control System (Sheet 1 of 13)

Design Criteria as Cited with Summary Intent	Design Bases As Applied to SHINE
IEEE Std 379-2000 (R2008) **IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems** **Abstract**: Application of the single-failure criterion to the electrical power, instrumentation, and control portions of nuclear power generating safety systems is covered in this standard. **Keywords:** actuator, cascaded failure, common-cause failure, design basis event, detectable failure, effects analysis, safety system, single-failure criterion, system actuation, system logic	As Applied This standard is applied to the design of the RICS, ESFs, and other instrumentation SSCs that are identified as ~~IROFS~~SR in the RPF. Means of Compliance The RICS is a DCS designed, rated, and approved for use in safety instrumented systems as determined by ISA 84.00.01. The system will use a safety PLC as recognized by IEC 61508 conforming to a system based on redundant power supplies, processors, and input/output channels. Controls that are classified ~~IROFS~~SR in the Section 6b and 13b~~ or the ISA Summary~~ for the RPF are evaluated against the single failure criteria. Exception NUREG-1537 allows for sharing and combining of systems and components with justification. This does not negate the single failure criterion; it makes an evaluation of systems mandatory where this would be a proposed design solution, which will be a function of reliability and risk.
IEEE Std 577-2004 **IEEE Standard Requirements for Reliability Analysis in the Design and Operation of Safety Systems for Nuclear Facilities** **Abstract:** This standard sets forth minimum acceptable requisites for the performance of reliability analyses for safety related systems of nuclear facilities when used to address the reliability requirements identified in regulations and other standards. The requirement that a reliability analysis be performed does not originate with this standard. However, when reliability analysis is used to demonstrate compliance with reliability requirements, this standard describes an acceptable response to the requirements. **Keywords:** nuclear facilities, reliability analysis, safety systems	As Applied This standard is applied to the design of the RICS, ESFs, and other instrumentation SSC that is identified as ~~IROFS~~SR for the SHINE facility. Means of Compliance For ~~IROFS~~SR functions identified in the Section 6b and 13b~~ or the ISA Summary~~, the design performs a reliability analysis of the proposed design solution. This can be qualitative or quantitative in nature as described in the standard.

Table 7b.2-1 Design Criteria for the RPF Instrumentation and Control System (Sheet 2 of 13)

Design Criteria as Cited with Summary Intent	Design Bases As Applied to SHINE
IEEE Std 603-2009 **IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations** **Abstract**: Minimum functional and design criteria for the power, instrumentation, and control portions of nuclear power generating station safety systems are established. The criteria are to be applied to those systems required to protect the public health and safety by functioning to mitigate the consequences of design basis events. The intent is to promote appropriate practices for design and evaluation of safety system performance and reliability. Although the standard is limited to safety systems, many of the principles may have applicability to equipment provided for safe shutdown, post accident monitoring display instrumentation, preventive interlock features, or any other systems, structures, or equipment related to safety. **Keywords:** actuated equipment, associated circuits, Class 1E, design, failure, maintenance bypass, operating bypass, safety function, sense and command features, sensor	As Applied This standard is applied to the design of the RICS, ESFs, and other instrumentation SSCs that are identified as ~~IROFS~~SR for the RPF. It describes the minimum functional and design criteria for safety systems. It does not describe what systems are to be determined as safety systems. Means of Compliance For ~~IROFS~~SR functions identified in ~~the~~Section 6b or 13b~~ or the ISA Summary~~, the design conforms to the practices detailed in the standard. Exception The SHINE facility is not a nuclear power reactor and does not have all of the systems detailed in this standard. The intent of this standard is followed.
IEEE Std 384-2008 **IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits** **Abstract**: The independence requirements of the circuits and equipment comprising or associated with Class 1E systems are described. Criteria for the independence that can be achieved by physical separation and electrical isolation of circuits and equipment that are redundant are set forth. The determination of what is to be considered redundant is not addressed. **Keywords:** associated circuit, barrier, Class 1E, independence, isolation, isolation device, raceway, separation	As Applied This standard is applied to the design of the RICS, ESFs, and other instrumentation SSCs that are identified as ~~IROFS~~SR for the RPF. It describes the minimum criteria for separation and independence of systems in a physical way. It does not describe what systems are to be separate and independent, only a means to do so. Means of Compliance For any ~~IROFS~~SR function identified in Section 6b or 13b~~ or the ISA Summary~~, the design conforms to the practices detailed in the standard. Exception The SHINE facility is not a nuclear power reactor and does not have all of the systems detailed in this standard. The intent of this standard is followed.

Table 7b.2-1 Design Criteria for the RPF Instrumentation and Control System (Sheet 3 of 13)

Design Criteria as Cited with Summary Intent	Design Bases As Applied to SHINE
IEEE Std 323-2003 **IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations** **Abstract:** The basic requirements for qualifying Class 1E equipment and interfaces that are to be used in nuclear power generating stations are described in this standard. The principles, methods, and procedures described are intended to be used for qualifying equipment, maintaining and extending qualification, and updating qualification, as required, if the equipment is modified. The qualification requirements in this standard, when met, demonstrate and document the ability of equipment to perform safety function(s) under applicable service conditions including design basis events, reducing the risk of common-cause equipment failure. **Keywords:** age conditioning, aging, condition monitoring, design basis events, equipment qualification, harsh environment, margin, mild environment, qualification methods, qualified life, radiation, SR function, significant aging mechanism, test plan, test sequence, type testing	As Applied This standard defines the methods for equipment qualification when it is desired to qualify equipment for the applications and the environments to which it may be exposed. This standard is generally utilized for qualification of Class 1E equipment located in harsh environments, and for certain post-accident monitoring equipment, but it may also be utilized for the qualification of equipment in mild environments. Means of Compliance For ~~IROFS~~SR functions identified in Section 6b or 13b~~ or the ISA Summary~~, the design conforms to the practices detailed in the standard for those systems determined to be Class 1E and located in harsh environment. This includes consideration of those SSCs that reside within the hot cells and process enclosures. Not all of the safety-related instrumented functions reside in hot cells or process enclosures. As an example, isolation dampers that reside in the hot cell with electrical components are scrutinized with this standard, but the controlling system RICS that resides outside the hot cell does not require the same level of exposure per the application of this standard. The standard is applied in a graded approach based on the location of the equipment.

Table 7b.2-1 Design Criteria for the RPF Instrumentation and Control System (Sheet 5 of 13)

Design Criteria as Cited with Summary Intent	Design Bases As Applied to SHINE
IEEE Std 338-2012 **IEEE Standard for Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems** **Abstract:** The standard provides criteria for the performance of periodic surveillance testing of nuclear power generating station safety systems. The scope of periodic surveillance testing consists of functional tests and checks, calibration verification, and time response measurements, as required, to verify that the safety system performs its defined safety function. Post-maintenance and post-modification testing are not covered by this document. This standard amplifies the periodic surveillance testing requirements of other nuclear safety-related IEEE standards. **Keywords:** functional tests, IEEE 338, periodic testing, risk-informed testing, surveillance testing	As Applied This standard is applied to the design of the RICS, ESFs, and other instrumentation SSCs that are identified as ~~IROFS~~SR for the RPF. It describes the methods and criteria for establishing a periodic surveillance program. It does not describe what systems are to be separate and independent, only a means to do so. Means of Compliance For ~~IROFS~~SR functions identified in Section 6b or 13b~~ or the ISA Summary~~, the design conforms to the practices detailed in the standard.
IEEE Std 497-2010 **IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations** **Abstract:** Criteria are established in this standard for variable selection, performance, design, and qualification of accident monitoring instrumentation, and include requirements for display alternatives for accident monitoring instrumentation, documentation of design bases, and use of portable instrumentation. **Keywords:** accident monitoring, display criteria, selection criteria, type variables	As Applied The purpose of this standard is to establish selection, design, performance, qualification, and display criteria for accident monitoring instrumentation. It provides guidance on the use of portable instrumentation and examples of accident monitoring display configurations. Means of Compliance For those monitoring functions determined to be required for the health and safety of public or workers during normal operation and for DBAs, the design conforms to the practices detailed in the standard.

Table 7b.2-1 Design Criteria for the RPF Instrumentation and Control System (Sheet 7 of 13)

Design Criteria as Cited with Summary Intent	Design Bases As Applied to SHINE
IEEE Std 828-2012 **IEEE Standard for Configuration Management in Systems and Software Engineering** **Abstract:** This standard establishes the minimum requirements for processes for Configuration Management (CM) in systems and software engineering. The application of this standard applies to any form, class, or type of software or system. This revision of the standard expands the previous version to explain CM, including identifying and acquiring configuration items, controlling changes, reporting the status of configuration items, as well as software builds and release engineering. Its predecessor defined only the contents of a software configuration management plan. This standard addresses what CM activities are to be done, when they are to happen in the life cycle, and what planning and resources are required. It also describes the content areas for a CM Plan. The standard supports ISO/IEC/IEEE 12207:2008 and ISO/IEC/IEEE 15288:2008 and adheres to the terminology in ISO/IEC/IEEE Std 24765 and the information item requirements of IEEE Std 15939. **Keywords:** change control, configuration accounting, configuration audit, configuration item, IEEE 828, release engineering, software builds, software configuration management, system configuration management	As Applied This standard describes configuration management processes to be established, how they are to be accomplished, who is responsible for doing specific activities, when they are to happen, and what specific resources are required. The RICS is designed as a DCS, the RICS has ~~IROFS~~SR functions, and this standard applies to system development, specifically software development. Means of Compliance The RICS software is developed utilizing this standard for safety function implementation.

Table 7b.2-1 Design Criteria for the RPF Instrumentation and Control System (Sheet 8 of 13)

Design Criteria as Cited with Summary Intent	Design Bases As Applied to SHINE
IEEE Std 1028-2008 **IEEE Standard for Software Reviews and Audits** **Abstract:** Five types of software reviews and audits, together with procedures required for the execution of each type, are defined in this standard. This standard is concerned only with the reviews and audits; procedures for determining the necessity of a review or audit are not defined, and the disposition of the results of the review or audit is not specified. Types included are management reviews, technical reviews, inspections, walk-throughs, and audits. **Keywords:** audit, inspection, review, walk-through	As Applied This standard provides minimum acceptable requirements for systematic software reviews. This standard describes organizational means for doing a review and documenting the findings. The RICS is designed as a DCS, the RICS has ~~IROFS~~SR functions, and this recommended standard applies to the development of the software for those systems. Means of Compliance The RICS is developed utilizing this standard.
ANS-10.4-2008 **Verification and validation of non-safety-related scientific and engineering computer programs for the nuclear industry** **Abstract:** This standard provides guidelines for the V&V of non-safety related scientific and engineering computer programs developed for use by the nuclear industry. The scope is restricted to research and other non-safety-related, noncritical applications. **Keywords:** software integrity level, software life cycle, V&V, validation, verification	As Applied The purpose of software V&V is to help the development organization build quality into the software during the software life cycle. This standard describes a means to verify and validate the software development for the nonsafety-related systems. It is utilized for software development in the SHINE facility that is not safety significant, i.e. not safety-related.~~ or IROFS.~~ Means of Compliance Nonsafety-related software is developed utilizing this standard.
ISA 67.04.01-2006 **Setpoints for Nuclear Safety-Related Instrumentation** **Abstract:** This standard defines the requirements for assessing, establishing, and maintaining nuclear SR and other important instrument setpoints associated with nuclear power plants or nuclear reactor facilities. **Keywords:** Setpoint, drift, analog channel, reliability analysis	As Applied This standard is applied to the design of the RICS and other instrumentation SSCs that are identified as ~~IROFS~~SR for the SHINE facility. It describes the methods and criteria for establishing setpoints utilized in safety systems and maintaining the documentation thereafter. Means of Compliance For ~~IROFS~~SR functions identified in Section 6b or 13b~~ or the ISA Summary~~ that is designed with inherent setpoints, the design conforms to the practices detailed in the standard.

Table 7b.2-1 Design Criteria for the RPF Instrumentation and Control System (Sheet 9 of 13)

Design Criteria as Cited with Summary Intent	Design Bases As Applied to SHINE
ISA 84.00.01-2004 Part 1 **Functional Safety: Safety Instrumented Systems for the Process Industry Sector - Part 1: Framework, Definitions, System, Hardware and Software Requirements** **Abstract:** This International Standard gives requirements for the specification, design, installation, operation and maintenance of a safety instrumented system, so that it can be confidently entrusted to place and/or maintain the process in a safe state. This standard has been developed as a process sector implementation of IEC 61508. **Keywords:** Safety Instrumented System, Safety Integrated Level, Safety Instrumented Function, SIS, SIL, SIF	As Applied This standard can be applied to the design of safety systems for the SHINE facility, but was specifically developed for the Industrial Process Sector. The standard is made up of three parts. Part 1 lays the ground work for the safety life cycle, overall structure of safety systems, definitions utilized, and an approach to implementing safety system design engineering. The physical hardware of the RICS is a product of design based on this standard and IEC 61508. Any ~~IROFS~~SR function required to be implemented by the RICS are evaluated utilizing the Part 1, 2, and 3 of this standard. The intent is to have the same reliability and risk reduction demonstrated utilizing systems in the RICS that are more readily available from the process industry, but having the same or higher documented and tested ability to reduce risk as fulfillment through other channels. Means of Compliance For the ~~IROFS~~SR functions required of the RICS, this standard is utilized for the design and implementation.
ISA 84.00.01-2004 Part 2 **Functional Safety: Safety Instrumented Systems for the Process Industry Sector - Part 2: Guidelines for the Application of ANSI/ISA-84.00.01-2004 Part 1 (IEC 61511-1 Mod) – Informative** **Abstract:** This International Standard gives requirements for the specification, design, installation, operation and maintenance of a safety instrumented system, so that it can be confidently entrusted to place and/or maintain the process in a safe state. This standard has been developed as a process sector implementation of IEC 61508. **Keywords:** Safety Instrumented System, Safety Integrated Level, Safety Instrumented Function, SIS, SIL, SIF	As Applied This standard can be applied to the design of safety systems for the SHINE facility, but was specifically developed for the Industrial Process Sector. The standard is made up of three parts. Part 2 provides guidance on the specification, design, installation, operation and maintenance of safety instrumented functions and related safety instrumented systems as defined in ISA 84.00.01, Part 1. ~~IROFS~~SR functions required to be implemented by the RICS are evaluated utilizing the Part 1, 2, and 3 of this standard. The intent is to have the same reliability and risk reduction demonstrated utilizing systems in the RICS that are more readily available from the process industry, but having the same or higher documented and tested ability to reduce risk as fulfillment through other channels. Means of Compliance For ~~IROFS~~SR functions that are required of the RICS, this standard is utilized for the design and implementation.

Table 7b.2-1 Design Criteria for the RPF Instrumentation and Control System (Sheet 10 of 13)

Design Criteria as Cited with Summary Intent	Design Bases As Applied to SHINE
ISA 84.00.01-2004 Part 3 **Functional Safety: Safety Instrumented Systems for the Process Industry Sector - Part 3: Guidance for the Determination of the Required Safety Integrity Levels – Informative** **Abstract:** This International Standard gives requirements for the specification, design, installation, operation and maintenance of a safety instrumented system, so that it can be confidently entrusted to place and/or maintain the process in a safe state. This standard has been developed as a process sector implementation of IEC 61508. **Keywords:** Safety Instrumented System, Safety Integrated Level, Safety Instrumented Function, SIS, SIL, SIF	As Applied This standard can be applied to the design of safety systems for the SHINE facility, but was specifically developed for the Industrial Process Sector. The standard is made up of three parts. Part 3 provides information on: • The underlying concepts of risk, the relationship of risk to safety integrity. • The determination of tolerable risk. • A number of different methods that enable the safety integrity levels for the safety functions to be determined. ~~IROFS~~SR functions required to be implemented by the RICS are evaluated utilizing the Part 1, 2, and 3 of this standard. The intent is to have the same reliability and risk reduction demonstrated utilizing systems in the RICS that are more readily available from the process industry, but having the same or higher documented and tested ability to reduce risk as fulfillment through other channels. Means of Compliance For ~~IROFS~~SR functions that are required of the RICS, this standard is utilized for the design and implementation.

Table 7b.2-1 Design Criteria for the RPF Instrumentation and Control System (Sheet 13 of 13)

Design Criteria as Cited with Summary Intent	Design Bases As Applied to SHINE
Regulatory Guide 1.53, Rev.2, 2003 **Application of the Single-Failure Criterion to Safety Systems** **Abstract:** Provides methods acceptable to the NRC staff for satisfying the NRC's regulations with respect to the application of the single-failure criterion to the electrical power, instrumentation, and control portions of nuclear power plant safety systems. **Keywords:** IEEE Std 379, Single-Failure Criterion	As Applied The approach provides guidance for applying single-failure criterion to safety-related instrumentation and control systems. Some end-devices utilized by the RICS are identified as ~~IROFS~~SR. These systems and components are evaluated via this regulatory guide. Means of Compliance The RICS, ESFs, and ~~IROFS~~SR end-devices are evaluated with this document.
Regulatory Guide 1.97, Rev.4, 2006 **Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants** **Abstract:** This regulatory guide to describe a method that the NRC staff considers acceptable for use in complying with the agency's regulations with respect to satisfying criteria for accident monitoring instrumentation in nuclear power plants. **Keywords:** IEEE Std 497, accident monitoring	As Applied The approach provides guidance for accident monitoring in the SHINE facility. These systems and components are evaluated via this regulatory guide. Means of Compliance The RICS, CAAS, CAMS, and RAMS are designed with the use of document.
Regulatory Guide 5.71, 2010 **Cyber Security Programs for Nuclear Facilities** **Abstract:** This regulatory guide provides an approach that the NRC staff deems acceptable for complying with the Commission's regulations regarding the protection of digital computers, communications systems, and networks from a cyber attack as defined by 10 CFR 73.1. **Keywords:** Cyber Security, 10 CFR 73.54(a)(2), design basis threat	As Applied The approach provides guidance for accident monitoring in the SHINE facility. These systems and components are evaluated via this regulatory guide. Means of Compliance The RICS and RICS HMI are designed with the use of document.

7b.2.3 SYSTEM DESCRIPTION

The RPF instrumentation and controls are composed of four basic blocks or systems: RICS, active ESFs, radiation monitoring systems, and operator interface displays and terminals. These systems provide an interface for the operator for monitoring and control. The RICS is a DCS that functions independently and is electrically isolated from other systems in the RCA. The RICS initiates active ESF mitigative responses and controls ~~IROFS~~SR and non-safety components. The ESFs in the RPF provide isolation functionality. The RICS can isolate one or any combination of the isolable cells in the RPF. The RICS also initiates the ESF isolation between ventilation zones in the RCA. The radiation monitoring systems consist of the RAMS, CAMS, and the CAAS. The radiation monitoring is discussed in Section 7b.6. The base status of radiation monitoring is shown on the RICS display system.

7b.2.3.1 RICS Description

The RICS utilizes a high integrity controller specified for use in a safety instrumented system for the process industry. The RICS is sectionalized into two parts. One section is dedicated to monitoring ~~parameters~~instruments that are ~~IROFS~~SR. This same section provides the control action to initiate the active ESFs within the RPF. This platform forms the basis for the monitoring and control for ~~IROFS~~SR SSCs and to activate the ESFs. The system utilizes dual redundancy with high self-diagnostic functionality for the internal design of subsystems within the controller system to meet the stringent demands.

The other section of the RICS supports the activities required to perform the various functional operational modes in the RPF.

7b.2.3.2 Control Room

The RPF control room is integrated with the IF control room and described in Section 7a2.6.

7b.2.3.2.1 Operator Interface Description

The operator has direct visualization of critical values and has the ability to input control functions to the RICS. The RICS dedicated displays perform the following functions:

a. Static display shows critical measurement values and performs the function of annunciator panel. This is a fixed display panel that does not provide any interactive control functionality.

b. Alarm / event annunciator display panel. This panel displays any event or alarm that is defined for the process. This display allows the operator to acknowledge current events and alarms, and forms the historical record for events.

c. Dynamic interface display panel or HMI. This panel allows the operator a means to perform tasks, change modes, enable/disable overrides, and essentially anything that requires an operator input to allow, perform, or modify a task or event.

The set of displays are arranged in a workstation group. This group comprises the displays and the keyboard/mouse that are utilized to interface with the system.

Figure 7b.2-1 shows the graphic representation of the display description above for the RPF. The screen development follows the design criteria in Section 7b.2.1 and design basis as defined in Section 7b.2.2.

There is one workstation that consists of the RICS and facility integrated control system (FICS) displays in the control room. The FICS is classified as nonsafety-related and provides monitoring and control for the facility systems including lighting, HVAC, specialty gas and compressed air distribution systems, water systems, power distribution, and facility communication systems.

Additionally, the IF workstations each have RICS display and interface sections to provide the operator the ability to monitor inter-facility processes. The RICS HMI systems are fully functional and have the capability to control the RPF systems in the case of an emergency. The RICS HMIs allow for usage simultaneously. Figure 7a2.6-1 shows the graphic representation of the display description described above for the IF.

7b.2.3.2.2 Process Area

In addition to HMI systems in the control room, the RPF has operator interface terminals (OITs) located adjacent to the individual hot cell and glovebox systems. These systems communicate key parameters about each system to the operator such an internal temperature and differential pressure from the RICS. Alarms and ESF activations that occur for the individual hot cell are displayed on the OITs. For some locations, the OIT is utilized to setup and initiate inter-system process fluid transfers in the RPF. The OIT handles pump and flow controls where they are required.

7b.2.4 SYSTEM PERFORMANCE ANALYSIS

The RPF instrumentation and controls monitor the RPF processes and ESFs when required. The ~~IROFS~~SR components are managed by the RICS. The RICS provides the central decision making processor that evaluates monitored parameter from various plant instrumentation as well as from the radiation monitoring systems of the CAMS, CAAS, and RAMS. The analysis herein discusses safety as it relates to the ~~IROFS~~SR components design criteria and design basis.

Potential variables, conditions, or other items that will be probable subjects of technical specifications associated with the RPF instrumentation and control systems are provided in Chapter 14.

7b.2.4.1 RPF Trip and Alarm Design Basis

The design basis information for the RICS trip functions are based on the following two requirements~~ from 10 CFR 70. They are~~:

1) *Double contingency principle* means that process designs should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible~~ (baseline design criteria of 10 CFR 70.64(9))~~.

2) The safety program shall ensure that each ~~item relied on for safety~~SR SSC will be available and reliable to perform its intended safety function when needed~~ and in the context of the performance requirements of this section (performance requirements of 10 CFR 70.61(e))~~.

The RICS trip and alarm annunciation are protective functions and are part of the overall protection and safety monitoring systems for the RPF. The specific equipment design basis for the instrumentation and equipment used for the RICS trip and alarming functions are discussed in Section 7b.2.2.

The following discussion relates to the design bases utilized for monitoring specific signal values for RPF trips and alarms, the requirements of performance, the requirements for specific modes of operation of RPF and RICS and the design criteria documents generating the basis noted as a citation.

7b.2.4.1.1 Safety Functions and Corresponding Protective/Mitigative Actions for Design Basis Events

Citation - Section 4a and 4b of IEEE-603-2009

The results of the ~~ISA~~accident analysis for the RPF SSCs are discussed in Section 13b. Conditions that require monitoring and the subsequent action to be taken are detailed in Section 13b. ~~IROFS~~SR components identified in Section 13b, including the ESFs described in 6b, are monitored and controlled by RICS, as required.

7b.2.4.1.2 Variable Monitored to Control Protective/Mitigative Action

Citation - Section 4d of IEEE-603-2009

The following variables are monitored for RPF trip for isolation:

- The hot cell fire detection and suppression system (HCFD) is monitored for actuation. If tripped, the hot cell is isolated by the ventilation inlet and outlet dampers.
- The facility fire protection system (FFPS) is monitored for actuation. If tripped, the RCA confinement zone of the affected area is isolated by the zone bubble-tight dampers. This is not a~~n~~ ~~IROFS~~SR function.
- Hot cell gamma detectors are monitored in the hot cell. If acceptable gamma levels are exceeded, the hot cell is isolated by the ventilation inlet and outlet bubble-tight dampers.

The following is a preliminary list of variables to be monitored in the RPF for alarming to eliminate or reduce the exposure for the operator. The final list of variables to be monitored will be provided in the FSAR.

- Hot cell temperature – internal environment.
- Hot cell pressure – internal environment.
- Uranyl nitrate conversion system (UNCS) outlet temperature – process upset.
- Radioactive drain system (RDS) sump level – contamination exposure.
- Primary vessel vent system (PVVS) pressure – internal environment.
- PVVS flow – internal environment.
- RCA confinement zone pressure – contamination exposure.

- RCA radiation levels (CAMS, RAMS, and CAAS) – contamination or direct radiation exposure.
- Process stream pH – process upset.
- Process stream radiation – process upset.
- Valve positions within the RPF – process upset.

Tasks for the RICS are discussed in Section 7b.3.

7b.2.4.1.3 Functional Degradation of Safety System Performance

Citation - Section 4h of IEEE-603-2009

This section of the IEEE-603 describes what constitutes system malfunctions for safety-related ~~(IROFS)~~ and nonsafety-related devices. The ~~IROFS~~SR components are designed to consider those conditions having the potential for functional degradation in performance as described in ~~the ISA Summary and~~ Chapter 13.

The manual soft trip pushbuttons on the RICS HMI allow the operator to accomplish immediate isolation of the hot cells, shielded cells, or ventilation zones as they deem necessary.

Single channel failure is covered by redundant end measurement. ~~No safety measurement of an IROFS is made with less than two independent devices.~~

7b.2.4.2 Analysis

7b.2.4.2.1 RICS Trip Function Conformance to Applicable Criteria

The RICS performs a trip as a protective function as part of the RPF safety analysis. The design criteria for selection are discussed in Subsections 7b.2.1 and 7b.2.2. The following discussions relate to conformance to criteria for the RICS trip function.

7b.2.4.2.2 General Functional Requirement Conformance

Citation - Section 5 of IEEE-603-2009, GDC-13, GDC-20

The RICS initiates and controls the ESF activation and isolation when the system detects an off-normal event appropriate for activation. RICS trips are discussed in Section 7b.4. These monitored values and subsequent trips are a result of the preliminary accident analysis in Section 13b and provide a means to mitigate or reduce the consequences from the DBA to acceptable levels.

7b.2.4.2.3 Single Failure Criterion Conformance

Citation - Section 5.1 of IEEE-603-2009, IEEE-379-2000

A postulated active single failure in the RICS or ~~IROFS~~SR SSCs does not prevent the RICS from performing its protective action. This is accomplished by utilizing redundant measurement devices, and having systems design based on single failure criteria. The design criteria are discussed in Subsection 7b.2.1.

Table 7b.2-2 RPF Verification Matrix Design Criteria, Bases, Description (Sheet 1 of 7)

Verification Matrix	Design Criterion	Design Basis Applicability	Detailed Section Reference	Functional Means
1	IEEE-379 Single Failure Criterion	1) RICS 2) ESFs 3) RICS display 4) RICS ~~IROFS~~SR end devices 5) ESFs manual isolation	1) 7b.2.3 2) 7b.4 3) 7b.5 4) 7b.4 5) 7b.4	1) Safety DCS pre-approved platform for SIS 2) Redundant independent isolation components 3) Redundant operator interface workstations 4) Redundant sensor 5) Alternative manual means for ESF initiation
2	IEEE-577 Reliability Analysis Criterion	1) RICS 2) ESFs 3) RICS display 4) RICS ~~IROFS~~SR end devices 5) ESFs manual isolation	1) 7b.2.3 2) 7b.4 3) 7b.5 4) 7b.4 5) 7b.4	1) Safety DCS pre-approved platform for SIS 2) Redundant independent isolation components 3) Redundant operator interface workstations 4) Redundant sensor 5) Alternative manual means for ESF initiation
3	IEEE-603 Standard Criteria Safety Systems	1) RICS 2) ESFs 3) RICS display 4) RICS ~~IROFS~~SR end devices 5) ESFs manual isolation	1) 7b.2.3 2) 7b.4 3) 7b.5 4) 7b.4 5) 7b.4	Subsection 7b.2.4 for detailed discussion
4	IEEE-384 Independence of Class 1E Equipment & Circuits	1) RICS 2) ESFs 3) RICS display 4) RICS ~~IROFS~~SR end devices 5) ESFs manual isolation	1) 7b.2.3 2) 7b.4 3) 7b.5 4) 7b.4 5) 7b.4	For preliminary design, the independence of equipment is sufficient to meet IEEE-603 and IEEE-379.
5	IEEE-323 Qualifying Class 1E Equipment	1) RICS 2) ESFs 3) RICS display 4) RICS ~~IROFS~~SR end devices 5) ESFs manual isolation	1) 7b.2.3 2) 7b.4 3) 7b.5 4) 7b.4 5) 7b.4	This standard is for selecting and qualifying equipment. RICS, ESFs, and selected ~~IROFS~~SR end devices are required to be qualified for Class 1E use.
6	IEEE-344 Recommended Practice for Seismic Qualification	1) RICS 2) ESFs 3) RICS display 4) RICS ~~IROFS~~SR end devices 5) ESFs manual isolation	1) 7b.2.3 2) 7b.4 3) 7b.5 4) 7b.4 5) 7b.4	This standard is for selecting and qualifying equipment. RICS, ESFs, and selected ~~IROFS~~SR end devices are required to be qualified for Class 1E use.
7	IEEE-338 Criteria for the Periodic Surveillance Testing of Safety Systems	1) RICS 2) ESFs 3) RICS display 4) RICS ~~IROFS~~SR end devices 5) ESFs manual isolation	1) 7b.2.3 2) 7b.4 3) 7b.5 4) 7b.4 5) 7b.4	This standard is for selecting equipment, general design criteria that must be considered during design.

Table 7b.2-2 RPF Verification Matrix Design Criteria, Bases, Description (Sheet 2 of 7)

Verification Matrix	Design Criterion	Design Basis Applicability	Detailed Section Reference	Functional Means
8	IEEE-497 Criteria for Accident Monitoring Instruments	1) RICS 2) ESFs 3) RICS display 4) RICS ~~IROFS~~SR end devices 5) CAAS 6) RAMS 7) CAMS	1) 7b.2.3 2) 7b.4 3) 7b.5 4) 7b.4 5) 7b.6 6) 7b.6 7) 7b.6	This standard is for selecting accident monitoring equipment (specifically target towards radiation monitoring and annunciation), general design criteria that must be considered during design.
9	IEEE-7.4.3.2 Criteria for Digital Computers in Safety Systems	1) RICS 2) RICS display 3) OIT displays	1) 7b.2.3 2) 7b.5 3) 7b.5	Programming software for the RICS must follow the criteria. Programming must comply with the Software Quality Assurance Plan developed as part of the commitment to the Design Criteria outlined herein and in this standard. The software and hardware utilized for the displays for the RICS and the OIT must also follow the guidelines set forth in this standard. The equipment selected to date and the path forward allow for a successful completion following IEEE-7.4.3.2.
10	IEEE-828 Configuration Management in Systems and Software Engineering	1) RICS 2) RICS display 3) OIT displays	1) 7b.2.3 2) 7b.5 3) 7b.5	Part of the overall SQAP commitment of IEEE 7-4.3.2.
11	IEEE-829 Software and System Test Documentation	1) RICS 2) RICS display 3) OIT displays	1) 7b.2.3 2) 7b.5 3) 7b.5	Part of the overall SQAP commitment of IEEE 7-4.3.2.
12	IEEE-1012 Criteria for Software Verification and Validation	1) RICS 2) RICS display 3) OIT displays	1) 7b.2.3 2) 7b.5 3) 7b.5	Part of the overall SQAP commitment of IEEE 7-4.3.2.
13	IEEE-1028 Software Reviews and Audits	1) RICS 2) RICS display 3) OIT displays	1) 7b.2.3 2) 7b.5 3) 7b.5	Part of the overall SQAP commitment of IEEE 7-4.3.2.
14	ANS-10.4 Verification and Validation for non-safety software	1) RICS 2) RICS display 3) OIT displays	1) 7b.2.3 2) 7b.5 3) 7b.5	Part of the overall SQAP commitment of IEEE 7-4.3.2.
15	ISA 67.04.01 Setpoints for Nuclear Safety-Related Instruments	1) RICS 2) RICS ~~IROFS~~SR end devices	1) 7b.2.3 2) 7b.4	Part of the overall design commitment.

Table 7b.2-2 RPF Verification Matrix Design Criteria, Bases, Description (Sheet 3 of 7)

Verification Matrix	Design Criterion	Design Basis Applicability	Detailed Section Reference	Functional Means
16	ISA 84.00.01, Part 1,2,&3 Functional Safety: Safety Instrumented Systems for the Process Industry Sector	1) RICS 2) RICS display 3) OIT displays	1) 7b.2.3 2) 7b.5 3) 7b.5	This standard is utilized to design and develop the nonsafety-related systems as it relies on safety, reliability, and functionality.
17	NUREG-0700, Rev. 2 Human-System Interface Design Review Guidelines	1) RICS 2) RICS display 3) OIT displays	1) 7b.2.3 2) 7b.5 3) 7b.5	This standard is utilized to design and develop the safety-related and nonsafety-related systems as it pertains to control room arrangement, screen developments, and Operator Interface.
18	NUREG/CR-6463 Review Guidelines on Software Languages for Use in Nuclear Power Plant Safety Systems	1) RICS	1) 7b.2.3	This guideline is utilized to design, develop, and review the safety-related software.
19	NUREG/CR-6090 PLC and applications in Nuclear Reactor Systems	1) RICS	1) 7b.2.3	This guideline is utilized to design, develop, and review the safety-related and nonsafety-related software.
20	EPRI TR-106439, Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications	1) RICS display 2) OIT display	1) 7b.5 2) 7b.5	This standard is utilized to design and develop the safety-related systems as it pertains to obtaining software / hardware for the RICS, operator interface displays, and data acquisition systems.
21	Reg Guide 1.152 Criteria for use of Computers in Safety Systems	1) RICS 2) RICS display 3) OIT display	1) 7b.2.3 2) 7b.5 3) 7b.5	1) Redundant safety PLC platform 2) RICS redundant HMI workstations 3) Operator interface workstations
22	Reg Guide 1.53 Single Failure Criterion Evaluation for Safety Systems	1) RICS 2) ESFs 3) RICS display 4) RICS ~~IROFS~~SR end devices 5) ESFs manual isolation	1) 7b.2.3 2) 7b.4 3) 7b.5 4) 7b.4 5) 7b.4	1) High integrity safety PLC 2) Redundant channels for ESFs 3) Redundant operator interface workstations 4) Redundant sensor 5) Alternative manual means for ESFs initiation

Table 7b.2-2 RPF Verification Matrix Design Criteria, Bases, Description (Sheet 4 of 7)

Verification Matrix	Design Criterion	Design Basis Applicability	Detailed Section Reference	Functional Means
23	Reg Guide 5.71 Cyber Security Programs for Nuclear Facilities	1) RICS 2) RICS display 3) OIT display	1) 7b.2.3 2) 7b.5 3) 7b.5	Requires design approach and implementation.
24	10CFR 50, Appendix A, GDC 2 Natural Phenomena	1) ESFs manual operator isolation 2) RICS manual soft trip	1) 7b.4 2) 7b.2.4	This criterion defines that safety-related systems are designed to handle natural phenomena. RICS is capable of performing its safety functions during and after the external events described in Chapter 13. The RICS has the ability for the operator to manually choose to isolate and trip the EFFs in the RPF should external events dictate.
25	10CFR 50, Appendix A, GDC 4 Environmental and dynamic effects design bases.	1) RICS 2) ESFs 3) RICS display 4) RICS ~~IROFS~~SR end devices 5) CAAS 6) RAMS 7) CAMS	1) 7b.2.3 2) 7b.4 3) 7b.5 4) 7b.4 5) 7b.6 6) 7b.6 7) 7b.6	This criterion is met by choosing qualified equipment, testing, and surveillance. The criteria are described in Subsections 7b.2.1 and 7b.2.2.
26	10CFR 50, Appendix A, GDC 5 Sharing of structures, systems, and components	1) RICS 2) ESFs 3) RICS display 4) RICS ~~IROFS~~SR end devices 5) CAAS 6) RAMS 7) CAMS	1) 7b.2.3 2) 7b.4 3) 7b.5 4) 7b.4 5) 7b.6 6) 7b.6 7) 7b.6	This criterion is met by having independent redundant systems with defense-in-depth. The basis of design addresses this in Subsections 7b.2.1, 7b.2.2, and 7b.2.3. This effort is completed in detailed design.
27	10CFR 50, Appendix A, GDC 10 Reactor design	N/A	N/A	This criterion is addressed with having the RCS controlling the process steps of the reactivity process and the RPS as a separate independent system monitoring for off-normal events. This criterion does not apply to the SHINE RPF.
28	10CFR 50, Appendix A, GDC 12 Suppression of reactor power oscillations	N/A	N/A	This criterion is addressed to traditional BWR or PWR reactor systems. This criterion does not apply to the SHINE RPF.
29	10CFR 50, Appendix A, GDC 13 Instrumentation and control	1) RICS 2) ESFs 3) RICS display 4) RICS ~~IROFS~~SR end devices 5) CAAS 6) RAMS 7) CAMS	1) 7b.2.3 2) 7b.4 3) 7b.5 4) 7b.4 5) 7b.6 6) 7b.6 7) 7b.6	This criterion directly addresses the instrument and controls. The ICS instrumentation and controls provide control, protection, and means to safely mitigate the identified events described in ~~the ISA Summary~~Section 13b.

Table 7b.2-2 RPF Verification Matrix Design Criteria, Bases, Description (Sheet 5 of 7)

Verification Matrix	Design Criterion	Design Basis Applicability	Detailed Section Reference	Functional Means
30	10CFR 50, Appendix A, GDC 15 Reactor coolant system design	N/A	N/A	This criterion directly addresses the coolant system. This criterion does not apply to the SHINE RPF.
31	10CFR 50, Appendix A, GDC 16 Containment design	1) RICS 2) ESFs 3) RICS display 4) RICS ~~IROFS~~SR end devices 5) CAAS 6) RAMS 7) CAMS	1) 7b.2.3 2) 7b.4 3) 7b.5 4) 7b.4 5) 7b.6 6) 7b.6 7) 7b.6	This criterion directly addresses the containment system. The SHINE facility does not have containment, but has confinement per NUREG-1537 definitions. The RICS provides monitoring of radioactivity and fire in the cell and hot cell ventilation. In the event there is measured radioactivity in excess of predetermined values, the RICS isolates the process system.
32	10CFR 50, Appendix A, GDC 17 Electric Power Systems	1) RICS 2) ESFs 3) RICS display 4) RICS ~~IROFS~~SR end devices 5) CAAS 6) RAMS 7) CAMS	1) 7b.2.3 2) 7b.4 3) 7b.5 4) 7b.4 5) 7b.6 6) 7b.6 7) 7b.6	This criterion directly addresses the electric power system. The RICS and active ESFs are designed to fail-safe with loss of off-site power. Upon loss of off-site power, the de-energized state for the ESFs is the safe-state. There is a safety-related UPSS that provides power upon loss of off-site power that allows for monitoring.
33	10CFR 50, Appendix A, GDC 19 Control Room	1) RICS 2) ESFs 3) RICS display 4) RICS ~~IROFS~~SR end devices 5) CAAS 6) RAMS 7) CAMS	1) 7b.2.3 2) 7b.4 3) 7b.5 4) 7b.4 5) 7b.6 6) 7b.6 7) 7b.6	This criterion directly addresses the control room for the facility. The RICS provides control, protection, and means to safely mitigate the identified events described in ~~the ISA Summary~~Section 13b from the control room. It affords the ability to have a manual initiated isolation for the operator.
34	10CFR 50, Appendix A, GDC 20 Protection system functions	1) RICS 2) ESFs 3) RICS display 4) RICS ~~IROFS~~SR end devices 5) CAAS 6) RAMS 7) CAMS	1) 7b.2.3 2) 7b.4 3) 7b.5 4) 7b.4 5) 7b.6 6) 7b.6 7) 7b.6	This criterion directly addresses the protection systems for the facility. The RICS provides protection as the initiator for ESF mitigative isolation of the systems in the RPF. The RICS automatically initiates a protective action upon appropriate signal which safely mitigates the identified events described in ~~the ISA Summary~~ Section 13b. The RICS and the ESFs include the ability to have a manual initiated isolation from the operator.

Table 7b.2-2 RPF Verification Matrix Design Criteria, Bases, Description (Sheet 6 of 7)

Verification Matrix	Design Criterion	Design Basis Applicability	Detailed Section Reference	Functional Means
35	10CFR 50, Appendix A, GDC 21 Protection system reliability and testability	1) RICS 2) ESFs 3) RICS display 4) RICS ~~IROFS~~SR end devices 5) CAAS 6) RAMS 7) CAMS	1) 7b.2.3 2) 7b.4 3) 7b.5 4) 7b.4 5) 7b.6 6) 7b.6 7) 7b.6	This criterion directly addresses the protection systems and the ability to tolerate single failures of components and the requirement for online surveillance of channels used for safety. The system is presently designed as dual sensing for the ~~IROFS~~SR ~~sensing measurements~~functions. For the RICS, periodic surveillance will be performed to verify the safety functions can be performed.
36	10CFR 50, Appendix A, GDC 22 Protection system independence	1) RICS 2) ESFs 3) RICS display 4) RICS ~~IROFS~~SR end devices 5) CAAS 6) RAMS 7) CAMS	1) 7b.2.3 2) 7b.4 3) 7b.5 4) 7b.4 5) 7b.6 6) 7b.6 7) 7b.6	This criterion directly addresses the protection system independence as it relates to natural phenomena and influence from other systems during operation including loss of redundant devices. The RICS is modular redundant so the likelihood of complete loss of all monitored parameters is very low. There are redundant measurement devices for monitored parameters. The measurements are independent, thus limiting the possibility of failure.
37	10CFR 50, Appendix A, GDC 23 Protection system failure modes	1) RICS 2) ESFs	1) 7b.2.3 2) 7b.4	This criterion directly addresses the protection system failure mode. The RICS and ESFs are designed to fail-safe upon loss of electric power and loss of instrument air. The failed state is isolation for the hot cells and confinement zones.
38	10CFR 50, Appendix A, GDC 24 Separation of protection and control systems	N/A	N/A	This criterion directly addresses the separation of the protection system from the control system. This criterion was developed for nuclear reactors. The RPF has no reactor or reactor-like systems.
39	10CFR 50, Appendix A, GDC 25 Protection system requirements for reactivity control malfunctions	N/A	N/A	This criterion directly addresses the protection system ability to limit the control system for fuel insertion. This criterion does not apply to the system utilized by RPF.
40	10CFR 50, Appendix A, GDC 26 Reactivity control system redundancy and capability	N/A	N/A	This criterion directly addresses the control systems ability to modify reactivity by two separate means. This criterion does not apply to the system utilized by RPF.

Table 7b.2-2 RPF Verification Matrix Design Criteria, Bases, Description (Sheet 7 of 7)

Verification Matrix	Design Criterion	Design Basis Applicability	Detailed Section Reference	Functional Means
41	10CFR 50, Appendix A, GDC 27 Combined reactivity control systems capability	N/A	N/A	This criterion directly addresses the ability of the control system to handle a postulated stuck rod even after poison injection. This criterion does not apply to the RPF.
42	10CFR 50, Appendix A, GDC 28 Reactivity limits	N/A	N/A	This criterion directly addresses the control systems ability to limit the reactivity rate of change. This criterion does not apply to the RPF.
43	10CFR 50, Appendix A, GDC 29 Protection against anticipated operational occurrences	1) RICS 2) ESFs 3) RICS display 4) RICS ~~IROFS~~SR end devices 5) CAAS 6) RAMS 7) CAMS	1) 7b.2.3 2) 7b.4 3) 7b.5 4) 7b.4 5) 7b.6 6) 7b.6 7) 7b.6	This criterion directly addresses the protection system's and the control system's ability to function with high reliability for operational occurrences which require a safety action. The RICS/ESFs, CAMS, RAMS, and CAAS are designed as independent systems so that measured malfunctions that create an off-normal event either initiate a mitigative ESF action or alerts the operator of the abnormal event. The RICS is specifically chosen for its high reliability and fault tolerance and recognized as such by independent testing as a pre-qualified platform for safety-instrumented systems.

7b.4 ENGINEERED SAFETY FEATURE AND ALARMING

7b.4.1 SYSTEM DESCRIPTION

Process control ESFs within the RPF are activated by the RICS. An RPF ESF actuation system does not exist as a standalone system. The RICS performs the following ESF actuation functions:

- For hot cells, gloveboxes, or other cells (including the noble gas storage cell) that require isolation in the RPF, the RICS monitors parameters designated ~~IROFS~~SR and when appropriate, actuates the ESF for the hot cells, gloveboxes, or other cells. The ESFs that are actively controlled are the isolation inlet and outlet bubble-tight dampers and isolation valves for other penetrations into the enclosure that are determined to require isolation during the final safety analysis. Upon recognition of an off-normal ~~IROFS~~SR parameter, the RICS de-energizes the dampers and isolation valves in the system and the dampers and valves move to a closed safe-state for the affected hot cell, glovebox, or other cell. The ESF dampers and isolation valves within the RPF are designed as fail-closed dampers so that any loss of power results in closure and subsequent isolation of the hot cell, glovebox, or other cell.

- For the RCA ventilation system in the RPF, the RICS monitors parameters designated as ~~IROFS~~SR and when appropriate, actuates the ESF for the specific RCA ventilation system zone. For the RCA ventilation system zones, the ESFs that are actively controlled are the inlet and outlet bubble-tight dampers for each zone. Upon recognition of a~~n~~ ~~IROFS~~SR parameter exceeding acceptable limits for isolation, the RICS de-energizes the dampers in the system and the bubble-tight dampers move to a closed safe-state for the affected ventilation zone. The bubble-tight dampers within the RCA zone ventilation system are designed as fail-closed dampers so that loss of power results in closure of the damper and subsequent isolation of the RCA ventilation system zone.

- The internal logic of the RICS monitors the ESF and provides assurance that the ESF activation goes to completion. The ESF is reset by the operator from the RICS HMI display. The RICS is described in Subsection 7b.2.3.

7b.4.1.1 RICS Trips Description (Functional Performance)

This section identifies the monitored parameters and describes the events for initiating an ESF. The monitoring and control functions are described on a parameter by parameter basis in the following.

The RICS performs two automated initiations of ESFs. One is for mitigation of fire and the other for mitigation of radiation contamination. In the event of an activation of the HCFD or the FFPS, the RICS activates dampers to isolate affected areas. The FFPS isolation function is not a~~n~~ ~~IROFS~~SR function. The other automated response occurs when an active radiation monitored parameter within the isolable cell exceeds a trip level setting. In the case of an individual hot cell, glovebox, or other cell the RICS activates the ESF for bubble-tight damper isolation of the affected hot cell, glovebox, or other cell.

7b.5.1.4 Hot Cell Operator Interface Terminals

Each hot cell system has an OIT. This terminal is an extension of the information collected by the RICS. Each terminal displays the ~~IROFS~~ status for the individual system. The terminal shows current alarm status and ESF state for each hot cell. Pump controls for specific hot cells are initiated, interlocked, and controlled from a touchscreen-type OIT.

Acronyms and Abbreviations

Acronym/Abbreviation	**Definition**
A	ampere
AC	alternating current
ANSI	American National Standards Institute
CAAS	criticality accident and alarm system
CAMS	continuous air monitoring system
DC	direct current
ESF	engineered safety feature
FFPS	facility fire detection and suppression
FSAR	Final Safety Analysis Report
HCFD	hot cell fire detection and suppression system
hp	horsepower
HVAC	heating, ventilation, and air conditioning
Hz	hertz
IEEE	Institute of Electrical and Electronics Engineers
IF	irradiation facility
~~IROFS~~	~~items relied on for safety~~
kV	kilovolt
kVA	kilovoltampere
kW	kilowatt
LEU	low enriched uranium
LOEP	loss of electric power
LOOP	loss of off-site power
LWPS	light water pool system
MCC	motor control center
NACE	National Association of Corrosion Engineering
NDAS	neutron driver assembly system
NEC	National Electrical Code
NEMA	National Electrical Manufacturers Association

8a2.1.12 SHINE FACILITY IRRADIATION UNITS

Each irradiation unit (IU) requires two separate power supplies, one at 480 VAC, 3-phase, 60 Hz and one at 208 VAC, 3-phase, 60 Hz. The anticipated loads for each IU are as follows:

- 480 VAC load is approximately 50 kVA
- 208 VAC load is approximately 11 kVA

The fission process is monitored and controlled for conditions from source range through high operating ranges. The source and high ranges of the flux monitoring system detect neutron flux during startup and irradiation modes and provide a signal to the TSV reactivity protection system (TRPS). Flux monitors, located in the light water pool, are used for neutron detection. The monitors are located to provide optimum monitoring in the source and high ranges. This information is input to the TRPS for appropriate automatic response. The TRPS protects the TSV integrity by monitoring IU parameters and causing an IU shutdown when predetermined setpoints are exceeded. Separation of the TRPS functions and normal TSV process control function prevents failures in the TSV process control system (TPCS) control circuit from affecting the TRPS circuitry. ~~All i~~Items that are safety-related~~ or items relied on for safety (IROFS)~~ are supported by the safety-related UPSS and are not dependent on utility power. Therefore, any LOOP, either for a short or long duration, does not affect the safe operation of the IUs.

8a2.1.13 DESIGN BASES

The design function is to provide sufficient, and reliable electrical power to all SHINE facility systems and components requiring electrical power for normal operations and abnormal operations. The normal electrical power supply system (NPSS) is nonsafety-related but may support safety-related systems or components during normal operations. In the event of the loss of normal AC electrical power, the UPSS automatically provides power to the safety-related~~/IROFS~~ systems and components. Systems powered by the NPSS are described in Chapters 4, 5, 9, and 11. Further information on the design bases is provided in Chapter 3.

8a2.1.14 TECHNICAL SPECIFICATIONS

There are no potential variables, conditions, or other items that will be probable subjects of a technical specification associated with the normal electrical power system.

8a2.2-1 UPSS Load List[a]

Load Description	Nominal Connected Load (kW)	Nominal Demand Load (kW)
TSV Reactivity Protection System (TRPS)	7.20	7.20
Hot Cell Fire Detection and Suppression System (HCFD)[b]	1.2	1.2
Neutron Flux Detection System (NFDS)	1.2	1.2
Continuous Air Monitoring System (CAMS)	2.40	2.40
Radiation Area Monitoring System (RAMS)	2.40	2.40
Criticality Accident and Alarm System (CAAS)	2.40	2.40
Radiological Integrated Control System (RICS)	3.60	3.60
Engineered Safety Features Actuation System (ESFAS)	7.20	7.20
TSV Off-Gas System (TOGS) Recirculating Blower	5.57	13.92
Human Machine Interface (HMI)/ESFAS	2.40	2.40
Process Vessel Vent System (PVVS) Blower	5.57	13.92
HMI/TRPS	3.60	3.60
	UPSS Total Nominal Demand Load	**61.44 kW**

a) Load information above is for a single train. The same loads apply to the redundant UPSS train.

b) The hot cell fire detection system (HCFD) is defined as ~~an IROFS~~SR. At final design a determination will be made whether additional components of the FFPS/HCFD systems will be safety-related.

Acronyms and Abbreviations (cont'd)

Acronym/Abbreviation	Definition
g	gram
g U/L	gram of uranium per liter
GTCC	greater than Class C
gpm	gallons per minute
HCFD	hot cell fire detection and suppression system
HEPA	high efficiency particulate air
HMI	human machine interface
hp	horsepower
HVAC	Heating, ventilation, and air conditioning
IBC	International Building Code
IEEE	Institute of Electrical and Electronic Engineers
IESNA	Illuminating Engineering Society of North America
IFC	International Fire Code
IGS	inert gas control
IMC	International Mechanical Code
~~IROFS~~	~~item relied on for safety~~
ISA	Integrated Safety Analysis
IU	irradiation unit
J/g	joules per gram
L	liter
lb U/gal	pound of uranium per gallon
lbs	pounds

The RVZ1 exhausts air from hot cells (controlled-environment work enclosures providing radioactive shielding and primary confinement of hazardous material), the noble gas storage cell, TSV off-gas system (TOGS) shielded cells, and irradiation unit cells in the RCA. The system also captures exhaust from the process vent vessel system (PVVS). The PVVS system includes a dedicated filtration and exhaust blower subsystem. The RVZ1 hot cell and irradiation unit cell enclosures draw ventilation air from the surrounding RVZ2 spaces through high efficiency particulate air (HEPA) filters. The exhaust air from each cell filters through local HEPA filters. The cells include automatic isolation dampers on the air inlet and exhaust outlet to enable confinement at the irradiation unit, noble gas storage cell, TOGS shielded cell or hot cell boundaries. These automatic dampers are safety-related (SR)~~ or items relied on for safety (IROFS)~~ and isolate the cells upon a signal from the engineered safety features actuation system (ESFAS) or RICS and reduce the exhaust of released airborne material prior to decaying to safe levels.

Negative space pressure in RVZ1 is controlled through modulation of local exhaust air flow control valves for each cell. The exhaust from the cells collects in an RVZ1 system duct header and then draws through final, testable, HEPA filters and carbon adsorbers prior to discharge into the exhaust stack. The speed of the RVZ1 exhaust fans is controlled to maintain a negative pressure set point in the RVZ1 exhaust duct header. The exhaust fans are fully redundant. If the operating fan fails, the standby fan will start automatically.

The RVZ1 exhaust HVAC control components operate through the FICS and are nonsafety-related, except for the cell isolation dampers noted above, and the automatic isolation dampers located in the RVZ1 exhaust ductwork downstream of the final filters. These dampers perform a safety function and close when required to provide confinement at the RCA boundary.

RVZ1 exhaust discharges to the nominally 56 inch (142 centimeter) diameter exhaust stack with a radiation monitoring system. The discharge point of the stack is nominally 7.6 feet (ft.) (2.3 meters [m]) above the nominal 58 ft. (17.1 m) roof line.

- RVZ2 Exhaust

RVZ2 exhausts air from RVZ2 confinement zones and associated process systems.

The RVZ2 exhausts air from the operating areas, workrooms, and fume hoods to maintain confinement in radiologically controlled areas. This confinement protects workers in the facility from radiological and hazardous chemical releases in the RCA. The exhaust air from these spaces collects in an RVZ2 exhaust header and then draws through final, testable, HEPA filters and carbon adsorbers prior to discharge into the exhaust stack. The exhaust fan speed is controlled to maintain the desired negative pressure in the RVZ2 exhaust header. The exhaust fans are fully redundant. If the operating fan fails, the standby fan will start automatically.

Air flow control valves in the RVZ2 room exhaust duct system operate in conjunction with the zone supply valves to produce an offset between exhaust and supply flow rates. The flow offset enables a differential pressure.

Flow control valves in fume hood exhaust ducts maintain a constant volume through each fume hood. The control valves automatically modulate to compensate for changes in pressure drop due to loading of local filters.

The RVZ2 controls operate through the FICS and are nonsafety-related, except for the automatic isolation dampers in the RVZ2 exhaust duct system located downstream of the final filters. These perform a safety function and close when required to provide confinement at the RCA boundary.

RVZ2 exhaust system discharges to the nominally 56 inch (142 centimeter) diameter exhaust stack along with RVZ1. The discharge point of the stack is nominally 7.6 ft. (2.3 m) above the nominal 58 ft. (17.1 m) roof line.

- RVZ3

  RVZ3 is the tertiary confinement zone that includes process support areas where contamination is not expected to occur under normal operating conditions. RVZ3 areas are maintained at an elevated pressure relative to RVZ2 areas.

  RVZ3 is supplied with air from the RVZ2SA, and air is exhausted to RVZ2. RVZ3 does not contain separate AHUs for exhaust or supply air.

9a2.1.1.1 Design Bases

The RV is designed to provide environmental conditions suitable for personnel and equipment. The functions of the system include conditioning the RCA environment for workers and equipment, and confinement of hazardous chemical fumes and airborne radiological materials. The ventilation system includes functions designated as nonsafety-related~~,~~ and safety-related~~, and IROFS~~. System safety functions are achieved through maintaining negative pressure gradients between confinement zones and to the outside atmosphere, air-exchange rates, exhaust stream air filtration, and isolation (closure) of ventilation duct systems at designated boundaries.

The RV is designed such that the FICS monitors and controls the RCA ventilation system equipment, flow rates, pressures, and temperatures. Instrumentation monitors the ventilation systems for off-normal conditions and signal alarms as required. The FICS starts, shuts down, and operates the RV in normal operating modes, which prevent positive pressurization of contaminated areas and creates flow patterns that direct air toward areas of increasing contamination potential.

The primary operational design functions of the RCA ventilation system are summarized below:

- Provide ventilation air and condition the RCA environment for workers.
- Provide makeup air and condition the RCA environment for process equipment.
- Confine airborne radiological materials.
- Limit the spread of airborne contamination.
- Maintain dose uptake through ingestion to levels as low as reasonable achievable (ALARA) per 10 CFR 20 – Standards for Protection against Radiation.
- Confine hazardous chemical fumes.

Table 9a2.1-2 HVAC Damper Design Codes and Standards

Component	Codes and Standards
Safety-Related ~~or IROFS~~ Dampers	Code on Nuclear Air and Gas Treatment ASME AG-1, including 2009 Addenda 1a and 1b (2011), Section DA, Dampers and Louvers.
Nonsafety-Related Dampers	SMACNA 1966 (Version 2005), HVAC Duct Construction Standards, Metal and Flexible (Use for Low Pressure Duct Sections). SMACNA 1520 (Version 1999), Round Industrial Duct Construction Standards (Use for High Pressure Duct Sections). SMACNA 1922 (Version 2009), Rectangular Industrial Duct Construction Standards (Use for High Pressure Duct Sections). AMCA 500-D-12 (Version 2012), Laboratory Methods for Testing Dampers for Rating.

9a2.3 FIRE PROTECTION SYSTEMS AND PROGRAMS

9a2.3.1 INTRODUCTION

The fire protection system at the SHINE site is designed to protect the SHINE facility from damage by fire and to provide means to safely shut down the IUs in case of a fire. The fire protection system detects and suppresses fires, and is an integral part of the fire protection program.

The sections that follow include the fire protection system design basis and the associated fire hazards analysis for the SHINE facility. The procedures for operation, testing, and surveillance of the fire protection systems, including relationships between fire protection plans, operating procedures, and the emergency plan will be developed and maintained under the fire protection program.

9a2.3.2 DESIGN BASES

The fire protection program will comply with American National Standards Institute (ANSI)/American Nuclear Society (ANS) Standard for Fire Protection Program Criteria for Research Reactors (ANSI/ANS, 1981). The fire protection system at the SHINE site meets the design criteria of National Fire Protection Association (NFPA) Standard for Standard for Fire Protection for Facilities Handling Radioactive Materials (NFPA, 2008c). Other nationally-recognized codes and standards, listed at the bottom of this subsection, are also used in the design of the fire protection system as applicable in order to achieve reasonable assurance of fire safety. The facility fire protection system (FFPS) is designed to:

- Prevent fire initiation by controlling, separating, and limiting the quantities of combustibles and sources of ignition.
- Isolate combustible materials and limit the spread of fire by subdividing the SHINE facility into fire areas separated by fire barriers.
- Separate redundant safe shutdown components and associated electrical divisions to preserve the capability to safely shut down the IUs following a fire.
- Separate redundant trains of safety-related~~ and IROFS~~ equipment used to mitigate the consequences of a design basis accident (but not required for safe shutdown following a fire) so that a fire within one train will not damage the redundant train.
- Provide confidence that failure or inadvertent operation of the fire protection system cannot prevent SHINE facility safety functions from being performed.
- Preclude the loss of structural support, due to warping or distortion of SHINE facility structural members caused by the heat from a fire, to the extent that such a failure could adversely affect safe shutdown capabilities.
- Provide firefighting personnel access and life safety escape routes for each fire area.
- Minimize exposure to personnel and releases to the environment of radioactivity or hazardous chemicals as a result of a fire.
- The fire protection system is classified as a nonsafety-related, non-seismic system. The fire protection system is not required to remain functional following an accident or the most severe natural phenomena.
- Maintain fire pump design capacity with two 100 percent capacity fire pumps.

Noncombustible Material: Materials having the characteristics listed below:

a) Material which, in the form in which it is used and under the conditions anticipated, will not ignite, burn, support combustion, or release flammable vapors when subjected to fire or heat.
b) Material having a structural base of noncombustible material, as defined in (a) with a surfacing not over 3.2 millimeter (0.13 inches) thick which has a flame spread rating not higher than 50 when measured in accordance with American Society for Testing and Materials (ASTM) Standard Test Method for Surface Burning Characteristics of Building Materials (ASTM, 2006).
c) There is an exception to this definition that allows the use of combustible interior finishes when listed by a nationally recognized testing laboratory, such as Factory Mutual (FM) or Underwriters Laboratories (UL) Incorporated, for a flame spread, smoke, and fuel contribution of 25 or less in its use configuration.

Mezzanine: An intermediate level or levels between the floor and ceiling of any story that meet IBC (IBC, 2009) floor area and openness requirements.

Non-combustible Construction: A construction type in which the structural elements are entirely of noncombustible or limited-combustible materials. Type IIB noncombustible construction has no fire resistant rating.

~~**Nuclear Safety-Related and IROFS Structures, Systems, and Components:** SHINE facility features necessary to ensure safe shutdown, maintain a safe shutdown condition, and to prevent or mitigate the consequences of accidents. See section 3.5 for a detailed description. IROFS and safety-related SSCs are collectively referred to as safety SSCs.~~

9a2.3.4.3.2 General

The FHA provides descriptions of construction, operations, and fire hazards associated with each fire area in the SHINE facility. The following are considered and assessed in the analysis:

- The applicable NRC fire protection requirements and guidance.
- Amounts, types, configurations, and locations of cable insulation and other combustible materials.
- In-situ fire hazards.
- Automatic fire detection and suppression capability and other fire protection features
- Reliance on and qualifications of fire barriers.
- Location and type of manual firefighting equipment and accessibility for manual fire fighting.
- Potential for a toxic, biological, or radiation incident due to a fire.
- Damage potential: maximum possible fire loss (MPFL).
- Protection of SSCs.
- Life safety considerations.

The collected data is used to perform the FHA review for every room or area, on a floor-by-floor basis. The collected data contain the following items:

- Identification of the safety and nonsafety-related systems, and associated cabling within each fire area that could provide support for ensuring a safe shutdown condition.

9a2.3.4.4.2.3 Manual Suppression

Manual fire suppression is conducted by a fully staffed, completely equipped, and adequately trained off-site fire department, capable and committed to respond to fires and related emergencies in a timely and effective manner. Personnel trained in the use of portable fire extinguishers are assumed to voluntarily conduct firefighting efforts on fires in the incipient stage.

Class I standpipes are provided for manual fire suppression capability in the SHINE facility. Class I systems include 2½ inch hose connections. The 2½ inch connection is not provided with a hose and is there for “trained fire-fighter” use only. The Class I standpipes are designed and installed in accordance with NFPA 14, Standard for the Installation of Standpipe and Hose Systems (NFPA, 2010b).

Means for supporting manual fire suppression efforts in the SHINE facility consist of portable multipurpose dry-chemical and clean-agent extinguishers. These extinguishers are selected, installed, inspected, tested, and maintained in accordance with NFPA 10, Standard for Portable Fire Extinguishers (NFPA, 2010a).

9a2.3.4.4.2.4 Fire Detection and Alarm Systems

Fire alarm and detection systems are provided throughout the SHINE facility and are designed, installed, located, inspected, tested, and maintained in accordance with NFPA 72, National Fire Alarm and Signaling Code (NFPA, 2013e).

Fire detection is provided as part of the facility fire detection and suppression system (FFPS) and the hot cell fire detection and suppression system (HCFD). The HCFD provides fire detection and suppression capabilities for the supercells and the hot cells in the RPF. Fire detectors in the HCFD send a signal to isolate the fire-rated dampers in the supercells and the hot cells in the event of a fire in one of these cells. These dampers reduce the potential release of radioactive materials from the hot cell or supercell due to a fire (see Subsection 13b.2.6). The fire detection in the HCFD is classified as ~~IROFS~~SR. The suppression subsystem of the HCFD is classified as nonsafety-related.

The fire detection in the rest of the SHINE facility is part of the FFPS. The FFPS is classified as nonsafety-related.

9a2.3.4.4.3 Fire Barriers and Protection of Penetrations

The SHINE facility is generally of reinforced concrete construction. The walls, floors, and ceilings have a 3-hour fire resistive rating where required by a high combustible loading in the room or where adjacent room contains equipment or systems from a different safety train. Stair towers which do not communicate between areas of different divisions may have walls and doors with a 2-hour fire rating for personnel protection during egress from the areas. Non-concrete interior walls are constructed of metal studs and gypsum wallboard to the required fire resistive rating.

9b.6 COVER GAS CONTROL IN CLOSED PRIMARY COOLANT SYSTEMS

There are no primary coolant systems in the RPF. This section will discuss systems that handle radioactive gases from process vessels.

9b.6.1 PROCESS VESSEL VENT SYSTEM

9b.6.1.1 Process Description

9b.6.1.1.1 Process Functions

The PVVS collects and treats the off-gases from process vessels in the SHINE facility. The PVVS collects off-gases from each vented vessel containing a significant quantity of radioactive material in the RPF, and receives noble gases from the NGRS after a period of decay. The PVVS consists of an acid gas scrubber loop and a blower to vent treated gases out of the RCA.

The process functions for the PVVS are the following:

- Receive off-gas from the process systems within the RPF.
- Treat off-gas to remove excess acids.
- Transfer treated off-gas to the RVZ1 exhaust system.
- Maintain process vessels at a negative pressure for confinement.
- Prevent detonations or deflagrations in process vessels from potential hydrogen accumulation.

9b.6.1.1.2 Safety Functions

PVVS preliminary safety functions and classifications:

- Prevent detonation or deflagration of radiolytic hydrogen gas by maintaining the hydrogen concentration below the LFL. Classification: ~~IROFS~~SR.
- Capture noble gases from process vessel vents. Classification: ~~Defense-in-Depth~~SR

9b.6.1.1.3 Primary System Interfaces

Vented process vessels within the RPF that contain significant quantities of radioactive material are connected to the PVVS, which exhausts the off-gas from the tank ullage. For a list of system interfaces, refer to Table 9b.6-1. The stream numbers referred to in the interface descriptions can be found in Figure 9b.6-1.

9b.6.1.1.4 Process Sequence

The PVVS receives off-gas from vented process vessels in the RPF. The majority of process vessels contain sulfuric acid or nitric acid solutions, and the off-gas is assumed to entrain some of these acids. A blower maintains the tank ullage at a slight negative pressure and draws all the off-gas through an acid-gas scrubber. The scrubber is a packed column with countercurrent flow: the off-gas in up-flow and a 1 M sodium hydroxide caustic scrub solution in gravity down-flow. The caustic scrub solution neutralizes the acids present in the off-gas, and is recycled through the scrubber until it is spent and requires replacing. A heat exchanger, cooled with process chilled water, removes the acid-base heat of reaction, heat introduced by the pump, and the excess heat from any high-temperature off-gases.

Off-gas is provided to the PVVS intermittently in varying quantities depending on what processes are occurring in the RCA at any given moment. The caustic scrub solution flow rate is set to process the maximum quantity of off-gas that may enter the scrubber at any one time. The caustic scrub solution is periodically purged from the acid-gas scrubber loop when it no longer neutralizes the acid off-gas, as measured by a pH meter located in the acid-gas scrubber loop.

Radioactive particles potentially entrained in the process off-gas are either absorbed in caustic scrub solution in the acid-gas scrubber, or are vented out of the PVVS. Treated off-gas is vented to the RVZ1 where, following HEPA and charcoal filtration, the gas is exhausted out the facility stack. Spent caustic scrub solution is pumped to the RLWS system for waste treatment. Fresh caustic scrub solution is supplied by the facility alkaline reagent storage and distribution system (FLRS), located outside the RCA.

See Figure 9b.6-1 for the PFD of the PVVS. Refer to Table 9b.6-2 for a description of the process equipment.

9b.6.1.1.5 Hydrogen Mitigation

Hydrogen may be generated by radiolysis in the target solution or uranyl nitrate solution contained in various process vessels downstream of the IU. The hydrogen may cause detonations or deflagrations if the concentration reaches the lower flammability limit (LFL) of 4 volume percent. PVVS is a~~n~~ ~~IROFS~~SR system to prevent hydrogen detonation or deflagration in the process vessels.

PVVS maintains low hydrogen concentrations in the ullage of process tanks by diluting the evolved gases. Vents located above the overflow lines of tanks provide an inlet for air to sweep the tank ullage and dilute any hydrogen that is present. The required sweep rate for each process tank will be determined in detailed design based on hydrogen generation rate calculations and, if necessary, hydrogen monitors will be installed.

In the case of loss of off-site power (LOOP), the PVVS blower is connected to a UPS. There are redundant blowers connected to separate UPS systems. A differential pressure switch across the blower indicates whether or not the blower is operating properly. The PVVS blower continues to operate even if the RVZ1 exhaust system is no longer functioning.

The current design of the UPSS provides for a two hour duration of Class 1E power following LOOP. The effects of post-shutdown hydrogen generation have not been fully determined. Final hydrogen generation rates and associated mission time for the emergency power system will be determined as part of final design. Design studies are on-going to establish design features that could be used for maintaining stable long-term post-accident conditions assuming that the off-site power is not available. These potential design features include:

- Passive hydrogen recombiners
- On-site safety-related emergency diesel generators
- Robust piping systems
- Deflagration flame arrestors

9b.6.2.1.1 Process Functions

The NGRS monitors, stores, and releases the radioactive isotopes of iodine and noble gases (primarily krypton and xenon) that are received from the TSV off-gas system (TOGS) following an irradiation cycle. Radioactive gases are generated and released from the target solution in the TSV during irradiation. The NGRS collects the off-gas from the TSV after irradiation is complete, and holds the gases in order to allow the short-lived noble gas radioisotopes to decay prior to release of the off-gas to the PVVS. The radiation level in the decayed gases is verified with radiation monitors prior to release.

The process functions for the NGRS are the following:

- Store TSV off-gas to allow for noble gas radioactive decay.
- Release decayed off-gas to PVVS.
- Monitor off-gas releases to ensure radioactivity levels are below regulatory limits for discharge to the environment.

9b.6.2.1.2 Safety Functions

NGRS preliminary safety functions and classifications:

- Prevent the release of radioactive material to the environment beyond the applicable regulatory limits. Classification: ~~IROFS~~SR.
- Limit the exposure of the worker, public, and environment to radioactive material. Classification: ~~IROFS~~SR.
- Contain and store noble gases (primarily xenon and krypton) generated in the TSV for at least 40 days. Classification: ~~IROFS~~SR

9b.6.2.1.3 Primary System Interfaces

Refer to Table 9b.6-3 for a list of system interfaces. The stream numbers referred to in the interface descriptions can be found in Figure 9b.6-2.

9b.6.2.1.4 Process Sequence

The NGRS consists of two gas compressors, five noble gas decay tanks, a condensate knock-out tank, associated piping components, and monitoring instrumentation. After a TSV completes an irradiation cycle, a mixture of gases is present in the TOGS loop, mainly air sweep gas with small amounts of radioactive gaseous isotopes including iodine, krypton, and xenon. Hydrogen is also present, and the TOGS catalytic recombiner maintains the hydrogen concentration below the LFL. Refer to Subsection 4a2.8 for more detail regarding the TOGS.

The TOGS transfers the TSV off-gas to the NGRS, where a compressor stores the off-gas in decay tanks. Prior to transfer to the NGRS, hydrogen sensors in the TOGS ensure hydrogen concentration is below the acceptable limit. The NGRS uses five noble gas decay tanks to provide temporary storage of TSV off-gas containing radioactive noble gases. The tanks allow for the decay of short-lived noble gas radioisotopes, including xenon-133, which is an isotope of concern for nuclear non-proliferation monitoring. The tanks are sized to store the TSV off-gas for

Acronyms and Abbreviations

Acronym/Abbreviation	Definition
hr	hour
I	iodine
I-131	iodine-131
ICRP	International Commission on Radiological Protection
IF	irradiation facility
in.	inch
~~ISA~~	~~Integrated Safety Analysis~~
IU	irradiation unit
km	kilometers
Kr	krypton
kW	kilowatt
LEM	liquid effluent monitor
LEU	low-enriched uranium
LLW	low level waste
LSA	low specific activity
MCNP	Monte Carlo N-Particle
MEI	maximum exposed individual
MEPS	molybdenum extraction and purification system
mi.	miles
MLLW	mixed low level waste
Mo	molybdenum
Mo-99	molybdenum-99
mrem	millirem
mrem/hr	millirem per hour
mrem/yr	millirem per year
mSv	millisievert

12.1.5 RADIATION SAFETY

The RP program meets the requirements of 10 CFR 20, Standards for Protection Against Radiation and is consistent with the guidance provided in Regulatory Guide 8.2, Guide for Administrative Practice in Radiation Surveys and Monitoring. The facility develops, documents, and implements the RP program commensurate with the risks posed by a medical isotope production facility. The facility uses, to the extent practicable, procedures and engineering controls, based upon sound RP principles, to achieve occupational doses to facility personnel and doses to members of the public that are ALARA. The RP staff reports to the RP Manager, who in turn implements the RP program by supporting ongoing activities in the SHINE facility. The RP program content and implementation are reviewed at least annually, as required by 10 CFR 20.1101(c). Sufficient resources in terms of staffing and equipment are provided to implement an effective RP program. Further details related to the authority of the radiation safety staff with respect to facility operations will be provided in the FSAR.

The RP program is described in greater detail in Subsection 11.1.2.

12.1.6 PRODUCTION FACILITY SAFETY PROGRAM

The production facility safety program is developed and integrated with the radiological safety program and additional facility safety programs and utilizes the methods described in ~~10 CFR 70.61 and 10 CFR 70.62, per~~ 10 CFR 50. Further details of the facility safety program ~~and the Integrated Safety Analysis~~ will be provided in the FSAR.