CHAPTER 3

DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

Table of Contents

Section	Title	Page
3.1	DESIGN CRITERIA	3-1
3.2	METEOROLOGICAL DAMAGE	3-19
3.2.1	WIND LOADING	3-19
3.2.2	TORNADO LOADING	3-19
3.2.3	SNOW, ICE, AND RAIN LOADING	3-21
3.3	WATER DAMAGE	3-22
3.3.1	FLOOD PROTECTION	3-22
3.3.2	STRUCTURAL DESIGN FOR FLOODING	3-24
3.4	SEISMIC DAMAGE	3-25
3.4.1	SEISMIC INPUT	3-25
3.4.2	SEISMIC ANALYSIS OF FACILITY STRUCTURES	3-26
3.4.3	SEISMIC QUALIFICATION OF SUBSYSTEMS AND EQUIPMENT	3-33
3.4.4	SEISMIC INSTRUMENTATION	3-34
3.4.5	SEISMIC ENVELOPE DESIGN FOR EXTERNAL HAZARDS	3-35
3.5	SYSTEMS AND COMPONENTS	3-45
3.5.1	CLASSIFICATION OF SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY	3-48
3.5.2	SEISMIC CLASSIFICATION	3-51
3.5a	IRRADIATION FACILITY	3-57
3.5a.1	IRRADIATION FACILITY SAFETY-RELATED SYSTEMS AND COMPONENTS	3-57
3.5a.2	IRRADIATION FACILITY CATEGORY I MECHANICAL SYSTEMS AND COMPONENTS	3-57
3.5a.3	IRRADIATION FACILITY CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT	3-58
3.5a.4	IRRADIATION FACILITY SAFETY-RELATED SYSTEMS AND COMPONENTS ENVIRONMENTAL DESIGN BASES CRITERIA	3-58
3.5a.5	SUBCRITICAL ASSEMBLY SYSTEM DESIGN BASIS	3-59
3.5a.6	PRIMARY CLOSED LOOP COOLING SYSTEM	3-60
3.5a.7	LIGHT WATER POOL SYSTEM	3-60
3.5a.8	TARGET SOLUTION VESSEL OFF-GAS SYSTEM	3-61
3.5a.9	NEUTRON DRIVER	3-62
3.5a.10	CONFINEMENT BARRIERS	3-62

Table of Contents

Section	Title	Page
3.5a.11	TRITIUM PURIFICATION SYSTEM	3-63
3.5a.12	INSTRUMENTATION AND CONTROL	3-63
3.5b	RADIOISOTOPE PRODUCTION FACILITY	3-94
3.5b.1	RADIOISOTOPE PRODUCTION FACILITY SYSTEMS AND COMPONENTS	3-94
3.6	REFERENCES	3-107

List of Tables

Number	Title
3.1-1	Systems List
3.1-2	Codes and Standards Used to Guide the Design of the SHINE Facility
3.1-3	NRC Guidance Used in the Design of the SHINE Facility
3.4-1	Results of Analysis for Representative Elements
3.4-2	Out-of-plane Shear Results of Analysis for Representative Elements
3.4-3	SSI Nodal Accelerations
3.4-4	Aircraft Impact Analysis Results
3.5-1	System and Classifications
3.5-2	Likelihood Index Limit Guidelines
3.5a-1	Appendix A to 10 CFR 50 General Design Criteria Which Have Been Interpreted As They Apply to the SHINE Irradiation Facility
3.5b-1	Baseline and General Design Criteria for Radioisotope Production Facility

List of Figures

Number	Title
3.4-1	View Looking Southwest of the Representative Concrete Sections
3.4-2	View Looking Northeast of the Representative Concrete Sections
3.4-3	Top View of the Representative Concrete Sections
3.4-4	SSI Node Locations Plan View
3.4-5	SSI Node Locations Elevation View, N-S Wall (Y = 0 Ft.)
3.4-6	SSI Node Locations Elevation View, N-S Wall (Y = 106.33 Ft.)
3.4-7	SSI Node Locations Elevation View, N-S Wall (Y = 163.33 Ft.)
3.4-8	SSI Node Locations Elevation View, E-W Wall (X = 0 Ft.)
3.4-9	SSI Node Locations Elevation View, E-W Wall (X = 148 Ft.)
3.4-10	SSI Node Locations Elevation View, E-W Wall (X = 198.33 Ft.)
3.4-11	SSI Node Locations Plan View, Elevation -10 Ft.
3.4-12	SSI Node Locations Plan View, Elevation 5 Ft.
3.4-13	SSI Node Locations Plan View, Elevation 58 Ft.
3.4-14	SSI Node Locations Elevation View, N-S Wall (Y = -29 Ft.)

Acronyms and Abbreviations

Acronym/Abbreviation	Definition
ºF	degrees Fahrenheit
ºC	degrees Celsius
10 CFR	Title 10 of the Code of Federal Regulations
ASCE	American Society of Civil Engineers
Btu/hr	british thermal units per hour
CAAS	criticality accident and alarm system
CAMS	continuous air monitoring system
cfm	cubic feet per minute
cm	centimeter
DBA	design basis accident
DBEQ	design basis earthquake
ESF	engineered safety features
ESFAS	engineered safety features actuation system
FFPS	facility fire detection and suppression system
FHA	fire hazards analysis
FIAS	facility instrument air system
FICS	facility integrated control system
ft.	foot
ft/sec	feet per second
ft^2	square foot
ft^3	cubic foot
gal.	gallons
GDC	general design criteria

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
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

Acronyms and Abbreviations (cont'd)

Acronym/Abbreviation	Definition
m	meter
m^2	square meter
m^3	cubic meter
m/s	meters per second
MEPS	Mo extraction and purification system
MeV	million electron volts
min.	minutes
MIPS	Moly isotope product packaging system
mph	miles per hour
mrem	millirem
MUPS	light water pool and primary closed loop cooling makeup system
n/cm^2s	neutron per centimeter-second
NDAS	neutron driver assembly system
NDT	nil-ductility transition
NFDS	neutron flux detection system
NGRS	noble gas removal system
NPSS	normal electrical power supply system
NRC	U.S. Nuclear Regulatory Commission
PCLS	primary closed loop cooling system
PFBS	production facility biological shield system
PGA	peak ground acceleration
PLC	programmable logic controller
PMF	probable maximum flood

Acronyms and Abbreviations (cont'd)

Acronym/Abbreviation	Definition
PMP	probable maximum precipitation
PPE	personal protective equipment
PSAR	Preliminary Safety Analysis Report
PSB	primary system boundary
psf	pounds per square foot
psi	pounds per square inch
PVVS	process vessel vent system
QAP	quality assurance plan
QAPD	quality assurance program document
RAMS	radiation area monitoring system
RCA	radiologically controlled area
RDS	radioactive drain system
RICS	radiological integrated control system
RPCS	radioisotope process facility cooling system
RPF	radioisotope production facility
RVZ1	RCA ventilation system Zone 1
RVZ2	RCA ventilation system Zone 2
RVZ3	RCA ventilation system Zone 3
SASS	subcritical assembly support structure
SCAS	subcritical assembly system
SHINE	SHINE Medical Technologies, Inc.
SNM	special nuclear material
SRM	stack release monitoring
SRP	Standard Review Plan for the Review of Safety Analysis for Nuclear Power Plants
SRSS	square root of the sum of the square

Acronyms and Abbreviations (cont'd)

Acronym/Abbreviation	Definition
SRWP	solid radioactive waste packaging
SSC	structure, system, or component
SSD	safe shutdown
SSE	safe shutdown earthquake
SSI	soil-structure interaction
Sv	sievert
SWRA	Southern Wisconsin Regional Airport
TDN	thermal denitration
TOGS	target solution vessel off-gas system
TPCS	target solution vessel process control system
TPS	tritium purification system
TRPS	target solution vessel reactivity protection system
TSPS	target solution preparation system
TSV	target solution vessel
UNCS	uranyl nitrate conversion system
UPSS	uninterruptible electrical power supply 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.

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.

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.

Various chapters throughout the PSAR describe design components that facilitate necessary inspection, testing, and maintenance for SSCs that are necessary for safe shut down of the facility and provide protection of the public, the facility, and the environment.

Table 3.1-2 describes the quality standards commensurate with the safety functions and potential risks that were used in the design of the SSCs.

Meteorological design bases describing the most severe weather extremes predicted to occur during the life of the facility are provided in Section 2.3. Design criteria for facility SSCs to withstand the most severe weather extremes predicted to occur during the life of the facility are provided in Section 3.2.

Hydrological design bases describing the most severe predicted hydrological events during the life of the facility are provided in Section 2.4. Design criteria for facility SSCs to withstand the most severe predicted hydrological events during the lifetime of the facility are provided in Section 3.3.

Seismic design bases for the facility are provided in Section 2.5. Seismic design criteria for the facility SSCs are provided in Section 3.4. Section 3.4.5 addresses aircraft crashes and external explosions.

Analyses concerning function, reliability, and maintainability of SSCs may be found throughout the PSAR, and may be provided at the FSAR stage for some SSCs.

General design criteria for systems and components are addressed in Section 3.5. Specific design criteria for systems and components are provided in other sections.

Potential conditions or other items that will be probable subjects of a technical specification associated with the facility structures and design features are provided in Chapter 14.

Table 3.1-1 Systems List
(Sheet 1 of 3)

System Name	System Code	Section Reference[(a)]
Facility Structure	FSTR	**3.4.2**
Subcritical Assembly System	SCAS	3.5a.5, 4a2.1, **4a2.2**
Light Water Pool System	LWPS	3.5a.7, 4a2.1, 5a2.1.1, **5a2.2**
TSV Off-gas System	TOGS	3.5a.8, 4a2.1, **4a2.8**, 11.2.2.2.3
Neutron Flux Detection System	NFDS	3.5a.12.3, 4a2.1, **7a2.4.3**
Noble Gas Removal System	NGRS	3.5b.1.6, 4b.1.3.4, **9b.6.2**, 11.2.2.2.11
Process Vessel Vent System	PVVS	3.5b.1.5, 4b.1.3.5, 9b.5.1.5, **9b.6.1**, 11.2.2.2.7
Criticality Accident and Alarm System	CAAS	3.5b.1.7, 6b.3, **7b.6**
Continuous Air Monitoring System	CAMS	3.5b.1.8, **7a2.7.4.1**, 11.1.4.1
Radiation Area Monitoring System	RAMS	3.5a.12.5.1, **7a2.7.4.2**
Irradiation Cell Biological Shielding	ICBS	3.5a.10.2, 4a2.1, **4a2.5**
Radiologically Controlled Area (RCA) Ventilation (Zones 1, 2, and 3)	RVZ1	3.5a.10.1, **9a2.1.1**
	RVZ2	3.5b.1.9.1, **9a2.1.1**
	RVZ3	**9a2.1.1**
Radioactive Drain System	RDS	3.5b.1.5, 4b.1.3.10, 9b.5.1.2, 9b.5.3.2, **9b.7.6**
TSV Reactivity Protection System	TRPS	3.5a.12.1, **7a2.4**
Engineered Safety Features Actuation System	ESFAS	3.5a.12.4, 7a2.1.4, **7a2.5**
Radiological Integrated Control System	RICS	3.5b.1.11, **7b.2.3**, 7b.3, 7b.4
Uninterruptible Electrical Power Supply System	UPSS	3.5a.12.6, **8a2.2**
Hot Cell Fire Detection and Suppression System	HCFD	3.5b.1.10, **9a2.3.4.4.2.4**
Molybdenum Extraction and Purification System	MEPS	1.5.2.1, 1.5.2.2, 3.5b.1.12, 4b.1.3.2, **4b.3**, 11.2.2.2.4, 11.2.2.2.5
Target Solution Preparation System	TSPS	3.5b.1.13, 4a2.2.1.8, 4b.1.3.1, 4b.4.2, 9a2.2.2, 9a2.5.1.2, **9b.2**, 9b.5.1.1, 11.2.2.2.1
Uranyl Nitrate Conversion System	UNCS	1.5.2.3, 1.5.2.4, 1.5.2.5, 3.5b.1.14, 4b.1.3.3, **4b.4.1**, 11.2.2.2.5

Table 3.1-1 Systems List
(Sheet 2 of 3)

System Name	System Code	Section Reference[(a)]
Production Facility Biological Shield	PFBS	3.5b.1.9.2, **4b.2**
Facility Fire Detection and Suppression System	FFPS	**9a2.3**
Material Control and Accountability System	MCAS	Table 3.5-1
Facility Instrument Air System	FIAS	**9b.7.9**
Facility Control Room	FCR	Table 3.5-1
Stack Release Monitoring	SRM	**11.1.4.1**
Tritium Purification System	TPS	1.5.2.7, 3.5a.11, 4a2.1, 9a2.5.1.1, **9a2.7.1**
Neutron Driver Assembly System	NDAS	4a2.1, **4a2.3**, 11.2.2.2.2
Primary Closed Loop Cooling System	PCLS	4a2.1, 5a2.1.1, **5a2.2**
Light Water Pool and Primary Closed Loop Cooling Makeup System	MUPS	5a2.1.4, **5a2.5**
Health Physics Monitors	HPM	Table 3.5-1
TSV Process Control System	TPCS	3.5a.12.2, 7a2.1.2, **7a2.3**, 7a2.6.8
Normal Electrical Power Supply System	NPSS	**8a2.1**
Inert Gas Control	IGS	Table 3.5-1
Material Handling System	MHS	**9b.7.7**
RCA Material Handling System	RMHS	**9b.7.2**
Solid Radioactive Waste Packaging	SRWP	9b.5.1.7, 9b.5.3.5, **9b.7.5**
Radioactive Liquid Waste Evaporation and Immobilization	RLWE	1.5.2.6, 3.5b.1.16, 4b.1.3.6, 9b.5.1.4, 9b.5.3.4, **9b.7.3**
Aqueous Radioactive Liquid Waste Storage	RLWS	1.5.2.6, 3.5b.1.17, 4b.1.3.7, 9b.5.1.3, 9b.5.3.3, **9b.7.4**
Organic Liquid Waste Storage and Export System	OLWS	4b.1.3.8, **9b.7.19**
Radioisotope Process Facility Cooling System	RPCS	4a2.7.2, 5a2.1.2, 5a2.2.9, **5a2.3**
Molybdenum Isotope Product Packaging System	MIPS	4b.1.3.9, 9b.5.1.6, **9b.7.1**

Table 3.1-1 Systems List
(Sheet 3 of 3)

System Name	System Code	Section Reference[a]
Facility Ventilation Zone 4	FVZ4	**9a2.1.2**
Facility Integrated Control System	FICS	**7b.2.3.2.1**
Facility Potable Water System	FPWS	**9b.7.8**
Facility Compressed Air System	FCAS	**9b.7.10**
Facility Breathing Air System	FBAS	**9b.7.11**
Facility Inert Gas System	FIGS	**9b.7.12**
Facility Welding System	FWS	Table 3.5-1
Facility Roof Drains System	FRDS	**9b.7.13**
Facility Sanitary Drains System	FSDS	**9b.7.14**
Facility Data and Communications System	FDCS	**9a2.4**
Facility Lighting Protections System	FLPS	**8a2.1.7**
Facility Demineralized Water System	FDWS	Table 3.5-1
Facility Chilled Water Supply and Distribution System	FCHS	**9a2.1.3**
Facility Heating Water System	FHWS	**9a2.1.4**
Facility Acid Reagent Storage and Distribution System	FARS	**9b.7.15**
Facility Alkaline Reagent Storage and Distribution System	FLRS	**9b.7.16**
Facility Salt Reagent Storage and Distribution System	FSRS	**9b.7.17**
Facility Organic Reagent Storage and Distribution System	FORS	**9b.7.18**
Cathodic Protection System	CPS	**8a2.1.8**
Emergency Lighting System	ELTG	Table 3.5-1
Facility Grounding System	FGND	**8a2.1.6**
Lighting System	LTG	Table 3.5-1
Standby Diesel Generator System	SDGS	**8a2.1.3**, 8a2.1.4

a) Bolded section numbers indicate the location of the most detailed system description.

Table 3.1-2 Codes and Standards Used to Guide the Design of the SHINE Facility (Sheet 1 of 11)

Item	Title / Description	Version
ACI 349-06	Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary	2007
ACI 349.1R-07	Reinforced Concrete Design for Thermal Effects on Nuclear Power Plant Structures	2007
AHRI Standard 410	Forced-Circulation Air-Cooling and Air-Heating Coils	2001
AISC 303	Code of Standard Practice for Steel Buildings and Bridges	
AMCA Publication 99	Standards Handbook	
AMCA Publication 201	Fans and Systems	
AMCA Publication 203	Field Performance Measurement of Fan Systems	
AMCA 301-06	Methods for Calculating Fan Sound Ratings from Laboratory Test Data	2006
American Conference on Governmental Industrial Hygienists (ACGIH)	Industrial Ventilation Manual of Recommended Practice	
ANSI C84.1	American National Standard for Electrical Power Systems and Equipment - Voltage Ratings (60 Hertz)	2011
ANSI N13.1	Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities	1999
ANSI N323D	American National Standard for Installed Radiation Protection Instrumentation	2002

Table 3.1-2 Codes and Standards Used to Guide the Design of the SHINE Facility (Sheet 2 of 11)

Item	Title / Description	Version
ANSI N323-1978	Radiation Protection Instrumentation Test and Calibration	1978
American National Standards Institute (ANSI) Z21.13	Gas Fired Low Pressure Steam & Hot Water Boilers	2010
ANSI/AHRI Standard 365	Performance Rating of Commercial and Industrial Unitary Air-Conditioning Condensing Units	2009
ANSI/AHRI Standard 430	Performance Rating of Central Station Air Handling Units	2009
ANSI/AHRI Standard 850	Performance Rating of Commercial and Industrial Air Filter Equipment	2004
ANSI/AHRI Standard 551/591	Performance Rating of Water-Chilling and Heat Pump Water Heating Packages Using the Vapor Compression Cycle	2011
ANSI/AIHA Z9.5	Laboratory Ventilation	2003
ANSI/AISC N690	Specification for Safety-Related Steel Structures for Nuclear Facilities	2012
ANSI/AMCA Standard 204-05	Balance Quality and Vibration Levels for Fans	R2012
ANSI/AMCA Standard 210-07	Laboratory Methods of Testing Fans for Certified Aerodynamic Performance Rating	2007
ANSI/AMCA Standard 500-D-12	Laboratory Methods of Testing Dampers for Rating	2012
ANSI/AMCA Standard 500-L-12	Laboratory Methods of Testing Louvers for Rating	2012
ANSI/AMCA Standard 510-09	Methods of Testing Heavy Duty Dampers for Rating	2009

Table 3.1-2 Codes and Standards Used to Guide the Design of the SHINE Facility (Sheet 3 of 11)

Item	Title / Description	Version
ANSI/ANS 6.3.1	Program for Testing Radiation Shields in Light Water Reactors	2007
ANSI/ANS 6.4	Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants	2006
ANSI/ANS 6.4.2	Specification for Radiation Shielding Materials	2006
ANSI/ANS 8.1	Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors	1998 (R2007)
ANSI/ANS 8.3	Criticality Accident Alarm System	1997 (R2012)
ANSI/ANS 8.7	Guide for Nuclear Criticality Safety in Storage of Nuclear Materials	1998 (R2007)
ANSI/ANS 8.10	Criteria for Nuclear Criticality Control in Operations with Shielding and Confinement	1983 (R2005)
ANSI/ANS 8.15	Nuclear Criticality Control of Special Actinide Elements	1981 (R2005)
ANSI/ANS 8.19	Administrative Practices for Nuclear Criticality Safety	2005
ANSI/ANS 8.20	Nuclear Criticality Safety Training	1991 (R2005)
ANSI/ANS 8.21	Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors	1995 (R2011)
ANSI/ANS 8.23	Nuclear Criticality Accident Emergency Planning and Response	2007 (R2012)
ANSI/ANS 8.24	Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations	2007 (R2012)
ANSI/ANS 8.26	Criticality Safety Engineer Training and Qualification Program	2007 (R2012)

Table 3.1-2 Codes and Standards Used to Guide the Design of the SHINE Facility (Sheet 4 of 11)

Item	Title / Description	Version
ANSI/ANS 15.1	The Development of Technical Specifications for Research Reactors	2007
ANSI/ANS 15.4	American National Standard for the Selection and Training of Personnel for Research Reactors	2007
ANSI/ANS 15.8	Quality Assurance Program Requirements for Research Reactors	1995 (R2005)
ANSI/ANS 15.16	Emergency Planning for Research Reactors	2008
ANSI/ANS 55.4	Gaseous Radioactive Waste Processing Systems for Light Water Reactor Plants	1993 (R2007)
ANSI/API 617	Axial and Centrifugal Compressors and Expander-Compressors for Petroleum, Chemical and Gas Industry Services	2009
ANSI/ASHRAE Standard 15	Safety Standard for Refrigeration Systems	2010
ANSI/ASHRAE Standard 34	Designation and Safety Classification of Refrigerants	2010
ANSI/ASHRAE Standard 52.2	Method of Testing General Ventilation Air-Cleaning Devices for Removal Efficiency by Particle Size	2007
ANSI/ASHRAE Standard 55	Thermal Environmental Conditions for Human Occupancy	2010
ANSI/ASHRAE Standard 62.1	Ventilation for Acceptable Indoor Air Quality	2010
ANSI/ASHRAE Standard 90.1	Energy Standard for Buildings Except Low-Rise Residential Buildings	2004
ANSI/ASHRAE/IESNA Standard 90.1	Energy Standard for Buildings Except Low-Rise Residential Buildings, I-P Edition (ANSI Approved; IESNA Co-sponsored)	2010

Table 3.1-2 Codes and Standards Used to Guide the Design of the SHINE Facility (Sheet 5 of 11)

Item	Title / Description	Version
ANSI/ASME N509	Nuclear Power Plant Air-Cleaning Units and Components	2002 (R2008)
ANSI/ASME N510	Testing of Nuclear Air Treatment Systems	2007
ANSI/HI 3.1-3.5	Rotary Pumps (A109),	2008
ANSI/ISA 84.00.01	Functional Safety: Safety Instrumented Systems for the Process Industry Sector	2004
API 617	Axial and Centrifugal Compressors and Expander-Compressors for Petroleum, Chemical and Gas Industry Services	2009
ASCE 4-98	Seismic Analysis of Safety-Related Nuclear Structures	1998 (R2000)
ASCE 7-05	Minimum Design Loads for Buildings and Other Structures	2006
ASCE/SEI 43-05	Standard, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities	2005
ASHRAE Standard 111	Measurement, Testing, Adjusting, and Balancing of Building HVAC Systems	2008
ASME AG-1	Code on Nuclear Air and Gas Treatment	2009, Addenda 1a, Addenda 1-b-2011
ASME B&PVC Section III	Rules for construction of nuclear facility components	

Table 3.1-2 Codes and Standards Used to Guide the Design of the SHINE Facility (Sheet 6 of 11)

Item	Title / Description	Version
ASME B&PVC Section VIII	Division 1 - Rules for Construction of Pressure Vessels	
ASME B30.17	Overhead and Gantry Cranes (Top Running Bridge, Single Girder, Underhung Hoist)	2003
ASME B30.20	Below-the-Hook Lifting Devices	2010
ASME B31.3	Process Piping	2008
ASME B31.3	Process Piping (code used for analysis of process systems)	2010
ASME B40.100-2005	Pressure Gauges and Gauge Attachments	2005
ASME B40.200-2008	Thermometers, Direct Reading and Remote Reading	2008
ASTM E84-09	Standard Test Method for Surface Burning Characteristics of Building Materials	
ASTM E1168-95	Radiological Protection Training for Nuclear Facility Workers	2008
City of Janesville City Ordinances	Water and Sewer	
CMAA 70	Specification for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes	2010
EPRI TR-106439	Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications	1996
IAEA-TECDOC-1347	Consideration of external events in the design of nuclear facilities other than nuclear power plants, with emphasis on earthquakes	2003
IBC	International Code Council (ICC) International Building Code (IBC)	2012

Table 3.1-2 Codes and Standards Used to Guide the Design of the SHINE Facility (Sheet 7 of 11)

Item	Title / Description	Version
IEC 61508	Functional safety of electrical/electronic/programmable electronic safety-related systems	2010
IEC 61511 MOD	Functional Safety - Safety instrumented systems for the process industry sector	2004
IEEE 7-4.3.2	IEEE Standard Criteria for Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations	2003
IEEE 308	IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations	2012
IEEE 323	IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations	2003
IEEE 336	IEEE Guide for Installation, Inspection, and Testing for Class 1E Power, Instrumentation, and Control Equipment at Nuclear Facilities	2010
IEEE 338	IEEE Standard for Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems	2012
IEEE 344	IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations	2004
IEEE 379	IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems	2000
IEEE 384	Standard Criteria for Independence of Class 1E Equipment and Circuits	2008
IEEE 450	IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications	2010

Table 3.1-2 Codes and Standards Used to Guide the Design of the SHINE Facility (Sheet 8 of 11)

Item	Title / Description	Version
IEEE 485	IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications	2010
IEEE 497	IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations	2010
IEEE 535	IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations	2006
IEEE 577	IEEE Standard Requirements for Reliability Analysis in the Design and Operation of Safety Systems for Nuclear Facilities	2012
IEEE 603	IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations	2009
IEEE 650	IEEE Standard for Qualification of Class 1E Static Battery Chargers and Inverters for Nuclear Power Generating Stations	2006
IEEE 828	IEEE Standard for Configuration Management in System and Software Engineering	2012
IEEE 829	IEEE Standard for Software and System Test Documentation	2008
IEEE 946	IEEE Recommended Practice for the Design of DC Auxiliary Power Systems for Generating Stations	2004
IEEE 1012	IEEE Standard Criteria for Software Verification and Validation	2004
IEEE 1028	IEEE Standard for Software Reviews and Audits	2008

Table 3.1-2 Codes and Standards Used to Guide the Design of the SHINE Facility (Sheet 9 of 11)

Item	Title / Description	Version
IEEE 1050	IEEE Guide for Instrumentation and Control Equipment Grounding in Generating Stations	2004
IEEE 15939	Standard adoption of ISO/IEC 15939:2007 systems and software engineering measurement	2008
IESNA	Illuminating Engineering Society of North America Handbook, Ninth Edition	2009
ISO/IEC/IEEE 12207	Software Life Cycle Processes	2008
ISO/IEC/IEEE 15288	System Life Cycle Processes	2008
ISO/IEC/IEEE 24765	System and Software Engineers- Vocabulary	2010
NEMA MG-1	Motors and Generators	2011
NFPA 10	Standard for Portable Fire Extinguishers	2010
NFPA 13	Standard for the Installation of Sprinkler Systems	2013
NFPA 14	Standard for the Installation of Standpipes and Hose Systems	2010
NFPA 15	Standard for Water Spray Fixed Systems for Fire Protection	2012
NFPA 20	Standard for the Installation of Stationary Pumps for Fire Protection	2013
NFPA 22	Standard for Water Tanks for Private Fire Protection	2008
NFPA 24	Standard for the Installation of Private Fire Service Mains and Their Appurtenances	2013

Table 3.1-2 Codes and Standards Used to Guide the Design of the SHINE Facility (Sheet 10 of 11)

Item	Title / Description	Version
NFPA 25	Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems	2011
NFPA 30	Flammable and Combustible Liquids Code	2010
NFPA 45	Standard on Fire Protection for Laboratories Using Chemicals	2011
NFPA 70	National Electrical Code	2011
NFPA 72	National Fire Alarm and Signaling Code	2013
NFPA 80	Standard for Fire Doors and Other Opening Protectives	2013
NFPA 90A	Standard for the Installation of Air-Conditioning and Ventilating Systems	2012
NFPA 92A	Standard for Smoke-Control Systems Utilizing Barriers and Pressure Differences	2009
NFPA 101	Life Safety Code®	2012
NFPA 220	Standard on Types of Building Construction	2009
NFPA 221	Standard for High Challenge Fire Walls, Fire Walls, and Fire Barrier Walls	2009
NFPA 430	Code for the Storage of Liquid and Solid Oxidizers	2004
NFPA 484	Standard for Combustible Metals	2012g
NFPA 780	Standard for the Installation of Lightning Protection Systems	2011
NFPA 801	Standard for Fire Protection for Facilities Handling Radioactive Materials	2008
NFPA 2001	Standard on Clean Agent Fire Extinguishing Systems	2012

Table 3.1-2 Codes and Standards Used to Guide the Design of the SHINE Facility (Sheet 11 of 11)

Item	Title / Description	Version
SMACNA	HVAC Systems Duct Design	2006
SMACNA	Seismic Restraint Manual: Guideline for Mechanical Systems, 3rd edition	2008
SMACNA 1143	HVAC Air Duct Leakage Test Manual	2012
SMACNA 1520	Round Industrial Duct Construction Standards	1999
SMACNA 1922	Rectangular Industrial Duct Construction Standards	2009
SMACNA 1966	HVAC Duct Construction Standards – Metal and Flexible	2005
UL 555	Standard for Fire Dampers - Seventh Edition	2012
UL 555S	Standard for Smoke Dampers - Fourth Edition	2009
UL 900	Standard for Air Filter Units - Seventh Edition	2012
UL 1995	Heating and Cooling equipment - Fourth Edition	2011
UL 1996	Standard for Electric Duct Heaters - Fourth Edition	2011
Wisconsin Administrative Code Chapter SPS 363	Energy Conservation	
Wisconsin Administrative Code Chapter SPS 364	Heating, Ventilating, and Air Conditioning	
Wisconsin Commercial Building Code SPS 360-366	Applicable in entirety, this amends the International Building Code (IBC)	
Wisconsin Administrative Code Chapter SPS 345	Mechanical Refrigeration	

Table 3.1-3 NRC Guidance Used in the Design of the SHINE Facility
(Sheet 1 of 2)

Document No. (Code/Standard)	Title / Description	Version
NRC Regulatory Guide 1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants	Rev. 4, 3/2007
NRC Regulatory Guide 1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems	Rev. 2, 11/2003
NRC Regulatory Guide 1.59	Design Basis Floods for Nuclear Power Plants	Rev. 2, 8/1977
NRC Regulatory Guide 1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants	Rev. 1, 12/1973
NRC Regulatory Guide 1.61	Damping Values for Seismic Design of Nuclear Power Plants	Rev. 1., 2007
NRC Regulatory Guide 1.69	Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants	Rev. 1, 2009
NRC Regulatory Guide 1.75	Criteria for Independence of Electrical Safety Systems	Rev. 3, 2/2005
NRC Regulatory Guide 1.76	Design Basis Tornado and Tornado Missiles for Nuclear Power Plants	Rev. 1, 2007
NRC Regulatory Guide 1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis	Rev. 3, 9/2012
NRC Regulatory Guide 1.97	Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants	Rev. 4, 6/2006
NRC Regulatory Guide 1.12	Nuclear Power Plant Instrumentation for Earthquakes	Rev. 2, 3/1997
NRC Regulatory Guide 1.122	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components	Rev. 1, 2/1978
NRC Regulatory Guide 1.128	Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants	Rev. 2, 2/2007
NRC Regulatory Guide 1.129	Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants	Rev. 2, 2/2007

Table 3.1-3 NRC Guidance Used in the Design of to the SHINE Facility (Sheet 2 of 2)

Document No. (Code/Standard)	Title / Description	Version
NRC Regulatory Guide 1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants	Rev. 2, 11/2001
NRC Regulatory Guide 1.152	Criteria for Use of Computers in Safety Systems of Nuclear Power Plants	Rev. 3, 7/2011
NRC Regulatory Guide 1.172	Software Requirements Specifications for Digital Computer Software Used In Safety Systems of Nuclear Power Plants	Rev. 0, 9/1997
NRC Regulatory Guide 1.201	Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance	Rev. 1, 5/2006
NRC Regulatory Guide 3.71	Nuclear Criticality Safety Standards for Fuels and Materials	Rev. 2, 12/2010
NRC Regulatory Guide 4.1	Radiological Environmental Monitoring for Nuclear Power Plants	Rev. 2, 6/2009
NRC Regulatory Guide 4.20	Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees Other than Power Reactors	Rev. 1, 4/2012
NRC Regulatory Guide 5.59	Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance	Rev. 1, 2/1983
NRC Regulatory Guide 5.71	Cyber Security Programs for Nuclear Facilities	Rev. 0, 01/2010
NRC Regulatory Guide 8.2	Administrative Practices in Radiation Surveys and Monitoring	Rev. 1, 5/2011
NRC Regulatory Guide 8.8	Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable	Rev. 3, 6/1978
NRC Regulatory Guide 8.10	Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable	Rev. 1R, 5/1977

3.2 METEOROLOGICAL DAMAGE

3.2.1 WIND LOADING

This subsection discusses the criteria used to design the SHINE facility for protection from wind loading conditions.

3.2.1.1 Design Wind Velocity

The SHINE facility structure is designed to withstand a basic wind velocity of 90 miles per hour (mph) (145 kilometers per hour [kph]) as described by Figure 6-1 of ASCE 7-05, Minimum Design Loads for Buildings and Other Structures (ASCE, 2006), for Wisconsin.

3.2.1.2 Determination of Applied Forces

The design wind velocity is converted to velocity pressure in accordance with Equation 6-15 of American Society of Civil Engineers (ASCE) 7-05 (ASCE, 2006):

$q_z = 0.00256 K_z K_{zt} K_d V^2 I$ (lb/ft^2) (Equation 3.2-1)

Where,

K_z = velocity pressure exposure coefficient evaluated at height (z) in Table 6-3 of ASCE 7-05

K_{zt} = topographic factor as defined in Section 6.5.7 of ASCE 7-05

K_d = wind directionality factor in Table 6-4 of ASCE 7-05

V = basic wind speed (3-second gust) obtained from Figure 6-1 of ASCE 7-05 for Wisconsin

I = importance factor obtained from ASCE 7-05

The design wind pressures and forces for the building at various heights above ground are obtained in accordance with Section 6.5.12.2.1 of ASCE 7-05 (ASCE, 2006) by multiplying the velocity pressure by the appropriate pressure coefficients, gust factors, and angles for sloped surfaces (i.e., the roof of the building). The building is categorized as an enclosed building according to Section 6.2 of ASCE 7-05 (ASCE, 2006) and, as a result, both external and internal pressures are applied to the structure. A positive and negative internal pressure is applied to the internal surfaces of the exterior walls as well as the roof.

3.2.2 TORNADO LOADING

This subsection discusses the criteria used to design of the SHINE facility to withstand the effects of a design-basis tornado phenomenon.

3.2.2.1 Applicable Design Parameters

The design-basis tornado characteristics are described in Regulatory Guide 1.76, Design Basis Tornado for Nuclear Power Plants:

a. Design-basis tornado characteristics are listed in the provisions of Table 1 for Region I.
b. The design-basis tornado missile spectrum and maximum horizontal speeds are given in Table 2.

3.2.2.2 Determination of Applied Forces

The maximum tornado wind speed is converted to velocity pressure in accordance with Equation 6-15 of ASCE 7-05 (ASCE, 2006):

$$q_z = 0.00256 K_z K_{zt} K_d V^2 I \text{ (lb/ft}^2\text{)} \quad \text{(Equation 3.2-2)}$$

Where,

K_z = velocity pressure exposure coefficient equal to 0.87

K_{zt} = topographic factor equal to 1.0

K_d = wind directionality factor equal to 1.0

V = maximum tornado wind speed equal to 230 mph (370 kph) for Region I

I = importance factor equal to 1.15

The tornado differential pressure is defined in Regulatory Guide 1.76, Table 1 as 1.2 pounds per square inch (psi) (8.3 kilopascals [kPa]) for Region I. The tornado differential pressure is applied as an outward pressure to the exterior walls of the building, as well as the roof because the structure is categorized as an enclosed building in accordance with Section 6.2 of ASCE 7-05 (ASCE, 2006).

The procedure used for transforming the tornado-generated missile impact into an effective or equivalent static load on the structure is consistent with NUREG-0800 Standard Review Plan for the Review of Safety Analysis for Nuclear Power Plants (SRP) Section 3.5.3, Subsection II.

The loading combinations of the individual tornado loading components and the load factors are in accordance with SRP Section 3.3.2.

3.2.2.3 Effect of Failure of Structures, Systems, or Components Not Designed for Tornado Loads

SSCs whose failure during a tornado event could affect the safety-related portions of the facility are either designed to resist the tornado loading or the effect on the safety-related structures from the failure of these SSCs or portions thereof are shown to be bounded by the tornado missile or aircraft impact evaluations.

The portion of the SHINE facility inside the Seismic Category I boundary is not vented during a tornado event. Tornado dampers are provided at the Seismic Category I boundary of the facility to ensure venting does not occur through the heating, ventilation, and air conditioning (HVAC) system. Pressure doors, or other means, are used to prevent significant pressure fluctuation within the facility due to a tornado event.

3.2.3 SNOW, ICE, AND RAIN LOADING

This subsection discusses the criteria used to design of the SHINE facility to withstand conditions due to snow, ice, and rain loading. Rain loading is not considered in the structural design of the building as the sloped roofs do not result in accumulation. As a result of the lack of rain accumulation, load due to ice is anticipated to be minimal and is enveloped by the design snow load.

3.2.3.1 Applicable Design Parameters

Snow load design parameters pertinent to the SHINE facility are provided in Chapter 7 of ASCE 7-05 (ASCE, 2006).

3.2.3.2 Determination of Applied Forces

The sloped roof snow load is calculated in accordance with Section 7.3 and 7.4 of ASCE 7-05 (ASCE, 2006). The combined equation utilized to calculate the sloped roof load is:

$$p_s = 0.7 C_s C_e C_t I p_g \qquad \text{(Equation 3.2-3)}$$

Where,

C_s = roof slope factor as determined by Sections 7.4.1 through 7.4.4 of ASCE 7-05

C_e = exposure factor as determined by Table 7-2 of ASCE 7-05

C_t = thermal factor as determined by Table 7-3 of ASCE 7-05

I = importance factor as determined by Table 7-4 of ASCE 7-05

p_g = ground snow load as set forth in Figure. 7-1 of ASCE 7-05

Unbalanced roof snow loads are computed in accordance with Section 7.6 of ASCE 7-05 (ASCE, 2006). The design snow drift surcharge loads are computed in accordance with Section 7.7.1 of ASCE 7-05 (ASCE, 2006).

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 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 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 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 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 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 SEISMIC DAMAGE

Seismic analysis criteria for the SHINE facility conform to IAEA-TECDOC-1347, “Consideration of external events in the design of nuclear facilities other than nuclear power plants, with emphasis on earthquakes”, which provides generic requirements and guidance for the seismic design of nuclear facilities other than nuclear power plants. Additional criteria provided in the Regulatory Guides and NUREG-0800 provide more detailed guidance in the seismic analysis of the SHINE facility.

Potential conditions or other items that will be probable subjects of technical specifications associated with the seismic design of structures are provided in Chapter 14.

3.4.1 SEISMIC INPUT

3.4.1.1 Design Response Spectra

The safe shutdown earthquake (SSE) ground motion is defined with a maximum ground acceleration of 0.2 g and design response spectra in accordance with Regulatory Guide 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants”.

Consistent with SRP Section 3.7.2, the location of the ground motion should be at the ground surface. The competent material (material with a minimum shear wave velocity of 1,000 feet per second [ft/sec] [305 meters per second {m/s}]) is 7.5 ft. (2.3 m) below the ground surface for the site. Hence, the SSE response spectra are defined as an outcrop at a depth of 7.5 ft. (2.3 m) below grade.

3.4.1.2 Design Time Histories

For soil-structure interaction (SSI) analysis and for generating in-structure response spectra, design acceleration time histories are required. Synthetic acceleration time histories are generated to envelop the design response spectra. Three mutually orthogonal synthetic acceleration time histories are generated for each horizontal direction and one for the vertical direction. Each of these time histories meet the design response spectra enveloping requirements consistent with Approach 2, Option 1 of SRP Section 3.7.1. The specifics of each of these time histories are:

- Each synthetic time history has been generated starting with seed recorded earthquake time histories.
- The strong motion durations (arias intensity to rise from 5 percent to 75 percent) of synthetic time histories are greater than a minimum of 6 seconds.
- The time history has a sufficiently small increment and sufficiently long duration. Records shall have a Nyquist frequency of at least 50 hertz (Hz) and a total duration of at least 20 seconds. The time step increment will be 0.005 seconds, which meets the Nyquist requirement for frequencies up to 100 Hz.
- Spectral acceleration at 5 percent damping is computed at a minimum of 100 points per frequency decade, uniformly spaced over the log frequency scale from 0.1 Hz to 50 Hz or the Nyquist frequency.
- Comparison of the response spectrum obtained from the synthetic time history with the target response spectrum shall be made at each frequency computed in the frequency range of interest.

- The computed 5 percent damped response spectrum of the acceleration time history shall not fall more than 10 percent below the target response spectrum at any one frequency and no more than 9 adjacent frequency points falling below the target response spectrum.
- The computed 5 percent damped response spectrum of the artificial time history shall not exceed the target spectrum at any frequency by more than 30 percent in the frequency range of interest.

3.4.1.3 Critical Damping Values

Structural damping values for various structural elements used in the seismic analyses are provided in Section 1.1 of Regulatory Guide 1.61, “Damping Values for Seismic Design of Nuclear Power Plants”. In the modal analysis, for structures composed of different materials (having different damping values) the composite modal damping is calculated using either the stiffness-weighted method or mass-weighted method, based on SRP Section 3.7.2. This applies to either the response spectrum method or the time history method.

3.4.2 SEISMIC ANALYSIS OF FACILITY STRUCTURES

3.4.2.1 Seismic Analysis Methods

The general equation of motion (as seen below) is used regardless of the method selected for the seismic analysis.

$$[M]\{\ddot{x}\} + [C]\{\dot{x}\} + [K]\{x\} = -[M]\{\ddot{u}_g\} \qquad \text{(Equation 3.4-1)}$$

Where:

$[M] =$ mass matrix
$[C] =$ damping matrix
$[K] =$ stiffness matrix
$\{\ddot{x}\} =$ column vector of relative accelerations
$\{\dot{x}\} =$ column vector of relative velocitie’s
$\{x\} =$ column vector of relative displacements
$\{\ddot{u}_g\} =$ ground acceleration

Analytical models are represented by finite element models. Consistent with SRP Section 3.7.2, SRP Acceptance Criterion 3.C, finite element models are acceptable if the following guidelines are met:

- The type of finite element used for modeling a structural system should depend on structural details, the purpose of analysis, and the theoretical formulation upon which the element is based. The mathematical discretization of the structure should consider the effect of element size, shape, and aspect ratio on solution accuracy.
- In developing a finite element model for dynamic response, it is necessary to consider that local regions of the structure, such as individual floor slabs or walls, may have fundamental vibration modes that can be excited by the dynamic seismic loading. These local vibration modes should be adequately represented in the dynamic response model,

in order to ensure that the in-structure response spectra include the additional amplification.

The finite element model consists of plate/shell and beam elements.

3.4.2.2 Soil-Structure Interaction Analysis

The SSI model provides structural responses for design basis level seismic loading of the SHINE facility, including transfer functions at selected locations, maximum seismic acceleration (zero point acceleration [ZPA]) values at selected locations, and in-structure response spectra (ISRS) (horizontal and vertical directions) for various damping values at selected locations. The SSI model is developed using the computer program Structural Analysis Software System Interface (SASSI).

Major structural elements of the SHINE facility, including walls, slabs, beams and columns, are modeled with appropriate mass and stiffness properties. Major openings within walls and slabs are included in the SSI model. The model uses shell elements to represent concrete slabs and walls, and frame elements to represent steel members, mostly comprising the truss components in the facility. Elements are modeled at the geometric centerline of the structural member they represent with the following exceptions:

- The basemat centerline is modeled at the top-of-slab elevation to ensure that the heights of the walls and columns are not modeled shorter than they actually are.
- The below grade mat sections are modeled at their bottom of slab elevation to maintain a uniform below grade mat mesh elevation.

In addition to self-weight of the structure, floor loads and equipment loads are converted to mass and included in the model. A portion of the loads are considered mass sources in the following manner according to SRP Section 3.7.2:

- Dead Load ………………100 percent
- Live Load…………………25 percent
- Snow Load……………….75 percent

Additional joint masses are applied to the model to represent the following:

- Door masses
- Crane masses
- Hydrodynamic masses

The SSI analyses are performed separately for mean, upper bound, and lower bound soil properties to represent potential variations in in-situ and backfill soil conditions around the building. The model for the concrete cracked case is identical to the uncracked case except that the stiffness of the concrete sections is reduced by 50 percent. Mean in-situ soil profile is used for this additional model. The response results from each of the four models are enveloped.

3.4.2.3 Combination of Earthquake Components

In order to account for the responses of the structures subjected to the three directional (two horizontal and the vertical) excitations, the maximum co-directional responses are combined using either the square root of the sum of the squares (SRSS) method or the 100-40-40 rule as described in Section 2.1 of Regulatory Guide 1.92.

3.4.2.4 Seismic Analysis Results

Figures 3.4-4 through 3.4-14 illustrate the location of the selected elements for which the averaged square root sum of the squares nodal accelerations are provided in Table 3.4-3.

3.4.2.5 Assessment of Structural Seismic Stability

The stability of the SHINE facility is evaluated for sliding and overturning considering the following load combinations and factors of safety in accordance with Section 7.2 of ASCE 43-05 (ASCE, 2005) and SRP 3.8.5:

Load Combination	**Minimum Factor of Safety**	
	Sliding	**Overturning**
D + H + E’	1.1	1.1
D + H + W_t	1.1	1.1
D + H + W	1.5	1.5

Where:
D = Dead Load
H = Lateral Earth Pressures
E’ = Earthquake Load
W_t = Tornado Load
W = Wind Load

The base reactions due to seismic forces envelope the reactions due to wind and tornado loading; therefore, a stability analysis for wind and tornado is not required. Seismic excitation in each direction is considered using the 100-40-40 percent combination rule as specified in Subsection 3.4.2.3 above.

The lateral driving forces applicable to the seismic stability evaluation of the SHINE facility include static lateral soil force, static surcharge lateral soil force, dynamic surcharge lateral soil, dynamic lateral soil force, and seismic lateral inertial force. The resistance for sliding is due to the static friction at the soil-basemat interface for sliding evaluation. The self-weight of the structure is considered in the resistance to overturning effects.

3.4.2.6 Structural Analysis of Facility

3.4.2.6.1 Description of the Structures

The SHINE facility is a box type shear wall system of reinforced concrete with reinforced concrete floor slabs. The major structural elements in the SHINE facility include the shear walls, the floor and roof slabs, and the foundation mat.

3.4.2.6.2 Applicable Codes and Standards

- ACI 349: Code Requirements for Nuclear Safety-Related Concrete Structures (ACI, 2007).
- ANSI/AISC-N690: Specification for the Design, Fabrication and Erection of Steel Structures for Nuclear Facilities, American Institute of Steel Construction (ANSI/AISC, 2012).

3.4.2.6.3 Site Design Parameters

The following subsections provide the site-specific parameters for the design of the SHINE facility.

3.4.2.6.3.1 Soil Parameters

The soil parameters for the SHINE facility are provided below.

- Net allowable static bearing capacity for 2-ft. (0.61-m) wide strip footings: 4000 pounds per foot square (psf) (191.5 kPa).
- Net allowable static bearing capacity for 6-ft. (1.8-m) wide spread footings: 6000 psf (287.3 kPa).
- Net allowable static bearing capacity for footings 22 ft. (6.7 m) below grade: 8000 psf (383.0 kPa).
- Minimum shear wave velocity: 459 ft/sec (140 m/s).
- Poisson's Ratio: 0.4.
- Unit Weight: 121 pounds per cubic foot (lb/ft^3) (1938 kilograms per cubic meters [kg/m^3]).

3.4.2.6.3.2 Maximum Ground Water Level

- 50 ft. (15.2 m) below grade level.

3.4.2.6.3.3 Maximum Flood Level

- Section 2.4 describes the PMP.
- Section 2.4 describes the probable maximum flood (PMF).

3.4.2.6.3.4 Snow Load

- Snow load: 30 psf (1.44 kPa) (50-year recurrence interval).
- A factor of 1.22 is used to account for the 100-year recurrence interval required.

3.4.2.6.3.5 Design Temperatures

- The winter dry-bulb temperature (-7 degrees Fahrenheit) is consistent with the 0.4 percent and 1 percent minimum dry-bulb temperatures (-9.1 degrees Fahrenheit and -2.9 degrees Fahrenheit respectively).
- The summer dry bulb temperature (88 degrees Fahrenheit) is consistent with the 0.4 percent and 2 percent maximum dry-bulb temperatures (91.5 degrees Fahrenheit and 85.8 degrees Fahrenheit respectively).

3.4.2.6.3.6 Seismology

- SSE Peak Ground Acceleration (PGA): 0.20 g (for both horizontal and vertical directions).
- SSE Response Spectra: Per Regulatory Guide 1.60.
- SSE Time History: Envelope SSE response spectra in accordance with SRP Section 3.7.1.

3.4.2.6.3.7 Extreme Wind

- Basic wind speed for Wisconsin: 90 mph (145 kph) (50-year recurrence interval).
- A factor of 1.07 is used to account for the 100-year recurrence interval required.
- Exposure Category C.

3.4.2.6.3.8 Tornado

- Maximum tornado wind speed (Region 1): 230 mph (370 kph).
- Radius of maximum rotational speed: 150 ft. (45.7 m).
- Tornado differential pressure: 1.2 psi (8.3 kPa).
- Missile Spectrum: See Table 2 of Regulatory Guide 1.76.

3.4.2.6.3.9 Rainfall

- The SHINE facility's sloped roof and building configuration preclude accumulation of rainwater; therefore, rain loads are not considered in this evaluation.

3.4.2.6.4 Design Loads and Loading Combinations

3.4.2.6.4.1 Dead Load

Dead loads consist of the weight of all materials of construction incorporated into the building, as well as the following:

- Concrete cover blocks for below grade tanks and trenches.
- Fixed equipment (includes tanks and hot cells).
- A floor area load of 0.35 kilopounds per square foot (ksf) (16.6 kPa) to account for the partition walls of the SHINE facility.
- Confinement doors.
- Crane dead loads as described in Subsection 3.4.2.6.4.6.

3.4.2.6.4.2 Live Load

The following four categories encompass the live loads for the SHINE facility:

- A distributed live load of 125 psf (5.99 kPa) is used for areas designated as light manufacturing.
- A distributed live load of 250 psf (12.0 kPa) is used for areas designated as heavy manufacturing.
- A uniform live load of 765 psf (36.6 kPa) is applied to the cover block laydown area in the IF and 630 psf (30.1 kPa) is applied to the cover block laydown area in the tank farm area to account for the movement of cover blocks.
- A uniform live load of 375 psf (18.0 kPa) is associated with the movement of a shipping container throughout the RCA.

3.4.2.6.4.3 Snow Load

The snow load is based on a ground snow load of 30 psf (1.44 kPa) with an importance factor of 1.2 and a mean recurrence interval of 100 years.

3.4.2.6.4.4 Wind Load

The wind load is based on a basic wind speed of 90 mph (145 kph) with an importance factor of 1.15 and a mean recurrence interval of 100 years.

3.4.2.6.4.5 Earthquake Load

A time history analysis is conducted on a fixed-base model with a portion of the loads considered as mass sources in the following manner according to SRP Section 3.7.2:

- Dead Load ………………100 percent.
- Live Load…………………25 percent.
- Snow Load……………….75 percent.
- Parked Crane Load…….100 percent.

Directional masses and accelerations for each joint are extracted from the fixed-base model. The directional masses for each joint are multiplied by the corresponding accelerations in order to obtain nine seismic force terms. Seismic design forces are increased by a factor derived by a ratio of SSI accelerations to design accelerations in order to envelope the SSI seismic design forces. Factors are applied to all terms including the accidental eccentricity term.

3.4.2.6.4.6 Crane Load

Loads associated with the three overhead bridge cranes are based on the required load rated capacities due to the anticipated maximum load in each of their respective areas.

3.4.2.6.4.7 Soil Pressure

Sub-grade walls of the SHINE facility are designed to resist static lateral earth pressure loads, compaction loads, static and dynamic surcharge loads, and elastic dynamic soil pressure loads. Application of active and passive earth pressures is not required because the building is stable against sliding and overturning. Therefore, the static lateral soil pressure is based on the at-rest earth pressure.

3.4.2.6.4.8 Fluid Load

The hydrostatic loading is calculated based on the actual dimensions of the IU cells and applied in the model as lateral hydrostatic pressure on the walls and vertical hydrostatic pressure on the bottom slabs.

- Maximum water depth: 12 ft. (3.7 m).
- Weight of water: 62.4 lb/ft^3 (1000 kg/m^3).

3.4.2.6.4.9 Tornado Load

The tornado load is based on a tornado wind speed of 230 mph (370 kph) and a tornado missile spectrum as described in Table 2 of Regulatory Guide 1.76. The tornado load, W_t is further defined by the following combinations:

$W_t = W_p$ (Equation 3.4-2)

$W_t = W_w + 0.5W_p$ (Equation 3.4-3)

$W_t = W_w + 0.5W_p + W_m$ (Equation 3.4-4)

Where:

W_p = load from tornado atmospheric pressure change.

W_w = load from tornado wind.

W_m = load from tornado missile impact.

3.4.2.6.4.10 Accidental Eccentricity

The accidental eccentricity analysis calculates the torsional moments on the RCA resulting from accidental eccentricity with respect to the center of rigidity of the structure and the effects of incoherent wave motion as required by Section 3.1.1(e) of ASCE 4-98 (ASCE, 2009). These scenarios are accounted for by applying a torsional moment resulting from an accidental eccentricity of 5 percent of the plan dimension between the center of mass and center of rigidity. Due to the directional nature of the load (clockwise or counterclockwise), the accidental eccentricity load is applied to the SAP2000 design model as a set of directional equivalent static loads. The calculations of the directional inplane loads to account for the torsional moment produced by accidental eccentricity are presented in this attachment. The loads simulate the torsional load due to the accidental eccentricity and are applied to the RCA in the clockwise direction and counterclockwise direction in separate load combinations.

3.4.2.6.4.11 SSI Comparison Increase Factor

SSI Model accelerations are compared with the design model accelerations in order to develop increase factors to apply to the Design Model so that it envelops the SSI results.

3.4.2.6.5 Structural Analysis Model

A three-dimensional finite element model of the SHINE Facility structure was created using the computer program SAP2000. The model utilizes shell elements to represent slabs and walls, and frame elements to represent columns and beams. Elements are modeled at the geometric centerline of the structural member they represent with the following exceptions:

- The basemat centerline is modeled at its actual top-of-slab elevation.
- The below grade mat sections are modeled at their bottom of slab elevation to maintain a uniform below grade mat mesh elevation.
- Minor adjustments are made to the dimensions and locations of the storage cells and wall openings to maximize mesh regularity in the model.
- Roof truss locations are adjusted to align with the roof shell element mesh.
- Some of the sub-grade trenches running throughout the facility are not explicitly modeled; however, the SAP2000 Design Model does incorporate concrete trench cover loads. The modeled trenches are modeled at the same location as the IU and storage cells to maintain a uniform below grade mat mesh elevation.

3.4.2.6.6 Structural Analysis Results

Concrete walls and slabs in the SHINE facility are designed for axial, flexural, and shear loads per provisions of ACI 349-06 (ACI, 2007). Walls and slabs are modeled in SAP2000 using groups of shell elements. To determine the longitudinal and transverse reinforcement required within a wall or slab, the design is performed on an element basis. Using resultant loads obtained from SAP2000 model data, the element is designed as a reinforced concrete section per ACI 349-06 (ACI, 2007). The required area of steel is determined for combined axial and flexural loads, in-plane shear loads, and out-of-plane shear loads. Using these results, reinforcement size and spacing is specified. Figures 3.4-1 through 3.4-3 illustrate location of the selected elements for which the required reinforcement is provided in Table 3.4-1 and additional shear forces are provided in Table 3.4-2.

3.4.3 SEISMIC QUALIFICATION OF SUBSYSTEMS AND EQUIPMENT

This subsection discusses the methods by which the SHINE facility subsystems and components are qualified to ensure functional integrity.

In general, one of the following five methods of seismically qualifying the equipment is chosen based upon the characteristics and complexities of the subsystem:

- Dynamic analysis.
- Testing procedures.
- Equivalent static load method of analysis.
- A combination of the dynamic analysis and the testing procedures.
- A combination of the testing procedures and the equivalent static load method.

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 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 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).

Because engine diameter and engine weight are both critical for the local evaluation, the local impact evaluation was performed for the Hawker 400 as well as the Challenger 605 aircraft. The Challenger 605 and Hawker 400 are evaluated as design basis aircraft impacts.

In regions of exterior walls and slabs where the amount of longitudinal reinforcement specified exceeds that specified in the accidental aircraft impact evaluation, additional calculations are performed to ensure requirements of Section F.3.6 of ACI 349-06 (ACI, 2007) are satisfied.

To evaluate the capability of the structure to withstand impact from the Challenger 605 aircraft, 25 impact locations are considered.

Cases 1, 2, and 3

These are three different impacts normal to the surface of the north wall panel bounded by the interior north-south wall, the roof, the west wall, and the basemat.

- Case 1: impact at wall center.
- Case 2: impact near wall edge at mid-height.
- Case 3: impact near wall corner.

Cases 4, 5, and 6

These are three different impacts normal to the surface of the east wall panel bounded by the Interior east-west wall, roof, north wall, and basemat.

- Case 4: impact at wall center.
- Case 5: impact near wall edge at mid-height.
- Case 6: impact near wall corner.

Cases 7, 8, and 9

These are three different impacts normal to the surface of the east wall panel bounded by the south wall, roof, interior east-west wall, and basemat.

- Case 7: impact at wall center.
- Case 8: impact near wall edge at mid-height.
- Case 9: impact near wall corner.

Case 10

This is a single impact normal to the surface of the south wall panel of the safety-related area outside the RCA.

- Case 10: impact at wall center.

Cases 11 and 12

These are two different impacts normal to the surface of the safety-related area outside the RCA west wall.

- Case 11: impact at wall center.
- Case 12: impact near wall corner.

Cases 13 and 14

These are two different impacts normal to the roof at named nodes of one of the long span roof trusses. The truss spans from the interior north-south wall to the west wall of the facility.

- Case 13: impact at first truss node from interior north-south wall.
- Case 14: impact at center node of truss.

Cases 15 and 16

These are two different impacts normal to the roof at named nodes of one of the short span roof trusses. The truss spans from the interior north-south wall to the east wall.

- Case 15: impact at first truss node from interior wall.
- Case 16: impact at center node of truss.

Case 17

This is a vertical (in-plane) impact on the interior north-south wall. Two patch loads are considered: the first is centered on the wall, and the second is offset 8 ft (2.4 m) from the vertical edge of the wall. Energy is absorbed via elastic shortening of concrete and soil deformation.

Cases 18 and 19

These are two different impacts normal to the roof at the location of the plate girder supporting the roof and spanning between the interior east-west wall and the south wall.

- Case 18: impact at quarter point from support on interior wall.
- Case 19: impact center of girder span.

Cases 20 and 21

These are two different impacts normal to an interior roof panel between two of the long-span roof trusses.

- Case 20: impact at center of interior roof panel between long-span trusses.
- Case 21: impact near edge of interior roof panel between long-span trusses.

Cases 22 and 23

These are two different impacts normal to an interior roof panel between two of the short-span roof trusses.

- Case 22: impact at center of interior roof panel between short-span trusses.
- Case 23: impact near edge of interior roof panel between short-span trusses.

Cases 24 and 25

These are two different impacts normal to the surface of the roof of the safety-related area outside the Radiological Control Area.

- Case 24: impact at wall center.
- Case 25: impact near wall edge.

The results of the analysis for the impacts described above are provided in Table 3.4-4. If the minimum reinforcement pattern and steel member sizes stated in the table are provided, the acceptance criteria of ACI 349 for concrete, and AISC N690 for steel for aircraft impact are met.

3.4.5.2 EXTERNAL EXPLOSIONS

Because the SHINE facility is not licensed as an operating nuclear reactor, explosions postulated as a result of the design basis threat as defined in Regulatory Guide 5.69 are not considered. However, accidental explosions due to transportation or storage of hazardous materials outside the facility and accidental explosions due to chemical reactions inside the facility are assessed.

The maximum overpressure at any safety-related area of the facility from any credible external source is less than 1 psi (6.9 kPa) (see Subsection 2.2.3.1.1). The seismic area is protected by outer walls and roofs consisting of reinforced concrete robust enough to withstand credible external explosions as defined in Regulatory Guide 1.91.

Table 3.4-1 Results of Analysis for Representative Elements (Sheet 1 of 2)

Loc. #	Location	Concrete Thickness (in)	Reinforcing Direction	Based on P-M Results: Axial Force (kip/ft)	Based on P-M Results: Moment (kip-ft/ft)	Based on P-M Results: Long. Reinforcement[(a)] (in^2/ft)	Based on In-plane Shear Results: In-Plane Shear (kip)	Based on In-plane Shear Results: Corresp. In-Plane Axial Force (kip)	Based on In-plane Shear Results: Long. Reinforcement (in^2/ft)	Required Steel per face (in^2/ft)	Min. Provided Steel/face (in^2/ft)	Notes
1	North Wall	24	Horizontal	16.1	30.2	0.53	6653	-2117	0.36	0.89	1.27	(b), (c), (d)
			Vertical	4.81	67.6	0.79	3115	187.6	0.36	1.15	1.27	(b), (c)
2	East Wall	24	Horizontal	30	33.7	0.79	3614	-2973	0.36	1.15	1.27	(b), (c)
			Vertical	-27.5	95.3	1.00	1507	-452	0.36	1.36	1.69	
3	South Wall	24	Horizontal	38.9	-46.8	1.00	7045	-2170	0.36	1.36	1.69	(c), (d)
			Vertical	-37.5	-112	1.27	2316	628	0.36	1.63	1.69	(c), (d)
4	West Wall	24	Horizontal	71.5	-14.9	0.79	4868	-5621	0.36	1.15	1.27	
			Vertical	-43.3	-108	1.00	1820	36.5	0.36	1.36	1.69	
5	Safety-related area outside the Radiologically Controlled Area, West Wall	24	Horizontal	70.6	-10.3	0.79	1595	-243	0.36	1.15	1.27	(b), (d)
			Vertical	-34	-55.8	0.79	397	120	0.36	1.15	1.27	(b), (d)
6	Interior North-South Wall	24	Horizontal	33.7	16.5	0.53	5441	-7307	0.36	0.89	1.27	(b)
			Vertical	-57.8	85.5	0.67	1498	248	0.36	1.03	1.27	
7	Interior East-West Wall	24	Horizontal	25.7	-42.1	0.79	1918	-2479	0.36	1.15	1.27	
			Vertical	-61.5	-49.3	0.53	1181	-147	0.36	0.89	1.00	
8	Roof	24	Horizontal	19.6	66.6	1.00	1674	-4841	0.00	1.00	1.27	(b), (c), (d)
			Vertical	-53.8	37.6	0.53	3902	-1454	0.14	0.67	1.27	(b), (c), (d)
9	Basemat, Location 1	24	Horizontal	7.7	-128.2	1.69	2278	-3606	0.00	1.69	1.69	
			Vertical	5.5	-81.6	1.00	3783	-1490	0.00	1.00	1.00	
10	Basemat, Location 2	24	Horizontal	26	-63.8	1.00	759	3356	0.00	1.00	1.00	
			Vertical	30.3	-82.9	1.27	1342	465	0.00	1.27	1.27	
11	Irradiation Cell Wall	72	Horizontal	109	160	1.56	594	-67.8	1.08	2.64	3.12	
			Vertical	6.9	-14.7	1.56	1715	-726	1.08	2.64	3.12	
12	Off-Gas Cell Wall	48	Horizontal	43.9	-101	1.04	141	-99.1	0.72	1.76	2.00	
			Vertical	-4.78	-6.59	1.04	163	47.9	0.72	1.76	2.00	
13	Irradiation Cell Slab	72	Horizontal	34.6	-32	1.56	2806	3374	0.00	1.56	1.56	
			Vertical	73.5	194	1.56	800	9.72	0.00	1.56	1.56	
14	Below Grade Tank Room Wall, Location 1	36	Horizontal	38.6	44.7	1.00	672	-635	0.54	1.54	1.56	
			Vertical	36.7	-66.1	1.00	408	33.9	0.54	1.54	1.56	
15	Below Grade Tank Room Wall, Location 2	36	Horizontal	76.8	-24.7	1.00	655	-29.3	0.54	1.54	1.56	
			Vertical	31.4	35.2	1.00	984	584	0.54	1.54	1.56	
16	Below Grade Tank Room Slab	36	Horizontal	65.8	15.9	1.00	787	-1142	0.00	1.00	1.00	
			Vertical	34.9	78.7	1.00	1358	-222	0.00	1.00	1.00	

Table 3.4-1 Results of Analysis for Representative Elements
(Sheet 2 of 2)

a) Calculated based on actual bar sizes, which inherently includes margin based on location in the P-M diagram.
b) Minimum provided steel is governed by aircraft impact analysis.
c) Results of aircraft impact indicate that higher longitudinal reinforcement may be required in isolated areas of this slab or wall.
d) Results of aircraft impact indicate that out of plane shear reinforcement may be required in isolated areas of this slab or wall.

Table 3.4-2 Out-of-Plane Shear Results of Analysis for Representative Elements

Loc. #	Location	Concrete Thickness (in.)	Reinforcing Direction	Vu (F3) (kip/ft)	Nu (F1 or F2) (kip/ft)
1	North Wall	24	Horizontal	1.15	-17.8
			Vertical	0.46	5.92
2	East Wall	24	Horizontal	7.15	-29.50
			Vertical	3.65	29.90
3	South Wall	24	Horizontal	8.45	-32.30
			Vertical	1.47	26.70
4	West Wall	24	Horizontal	8.38	-42.40
			Vertical	0.35	22.80
5	Safety-related area outside the Radiologically Controlled Area, West Wall	24	Horizontal	-2.81	31.84
			Vertical	0.05	-36.89
6	Interior North-South Wall	24	Horizontal	6.24	-43.80
			Vertical	4.82	26.70
7	Interior East-West Wall	24	Horizontal	2.39	-57.40
			Vertical	3.77	15.60
8	Roof	24	Horizontal	5.18	-53.70
			Vertical	4.31	14.50
9	Basemat, Location 1	24	Horizontal	6.99	2.09
			Vertical	2.88	6.97
10	Basemat, Location 2	24	Horizontal	3.96	30.90
			Vertical	5.41	27.60
11	Irradiation Cell Wall	72	Horizontal	8.14	1.00
			Vertical	56.30	82.00
12	Off-Gas Cell Wall	48	Horizontal	1.44	-2.77
			Vertical	20.90	44.00
13	Irradiation Cell Slab	72	Horizontal	43.30	72.30
			Vertical	14.00	16.50
14	Below Grade Tank Room Wall, Location 1	36	Horizontal	24.10	36.70
			Vertical	21.00	38.60
15	Below Grade Tank Room Wall, Location 2	36	Horizontal	15.90	19.60
			Vertical	18.20	76.80
16	Below Grade Tank Room Slab	36	Horizontal	17.40	24.20
			Vertical	6.65	66.20

Table 3.4-3 SSI Nodal Accelerations
(Sheet 1 of 2)

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Table 3.4-3 SSI Nodal Accelerations
(Sheet 2 of 2)

SSI Nodes	X-SRSS AVG (%g)	Y-SRSS AVG (%g)	Z-SRSS AVG (%g)
15835			
15859	0.247	0.250	0.229
15888			
16527	0.347	0.291	0.436
17131			
17878	0.253	0.292	0.308
17958			
17919	0.250	0.361	0.412
20920			
20948			
20973	0.252	0.394	0.257
21050			
21077			
21102			
18669			
18697			
18722	0.249	0.289	0.235
18840			
18868			
18893			
19676			
19677			
19698	0.491	0.265	0.276
19721			
19722			
20078			
28748			
26440			
28789	0.269	0.290	0.256
22444			
28828			
27898	0.274	0.308	0.899

Table 3.4-4 Aircraft Impact Analysis Results (Sheet 1 of 2)

Case No.	Ductility Demand	Ductility Limit	Condition of Acceptability[(a), (d)]
1	2.4	10	**North Wall Panel** #10 @ 12 in. each face, each direction
2	7.2	10	
3	9.5	10	
4	2.0	10	**East Wall Panel** #10 @ 12 in. each face, each direction
5	6.6	10	
6	9.1	10	
7	3.4	10	**East Wall Panel - Center and Edge** #10 @ 12 in. each face, each direction
8	9.1	10	
9	7.6	10	**East Wall Panel - Corner** #10 @ 9 in. each face, each direction
10	5.5	10	**Safety-related area outside the RCA, South Wall** #10 @ 6 in. each face, each direction #3 shear ties @ 12 in. or equivalent
11	7.4	10	**Safety-related area outside the RCA, West Wall - Center** #10 @ 12 in. each face, each direction
12	9.4	10	**Safety-related area outside the RCA, West Wall - Edge** #10 @ 12 in. each face, each direction #3 shear ties @ 12 in. or equivalent
13	10.0	25 (Tension)	**Long Truss[(b)]** Vertical Web Members: 2L8 x 8 x 1-1/8 (ASTM A 572, GR50) Diagonal Web Members: 2L5 x 5 x 7/8 (ASTM A 572, GR50) Top Chord Members: W14 x 68 (ASTM A992) Bottom Chord Members: W14 x 68 (ASTM A992)
14	4.8	25 (Tension)	
15	9.0	25 (Tension)	**Short Truss[(b)]** Vertical Web Members: 2L8 x 8 x 1-1/8 (ASTM A 572, GR50) Diagonal Web Members: 2L8 x 8 x 1-1/8 (ASTM A 572, GR50) Top Chord Members: 2L8 x 8 x 1-1/8 (ASTM A 572, GR50) Bottom Chord Members: 2L8 x 8 x 1-1/8 (ASTM A 572, GR50)
16	5.4	25 (Tension)	
17	1.0	1.3	**Interior North-South Wall** #10 @ 12 in. each face, each direction
18	3.5	5 (Shear) 10 (Flexure)	**Long Plate Girder[(c)]** Section Modulus of Girder: $S_x = 4781\ in^3$ Moment of Inertia of Girder: $I_x = 235427\ in^4$ Material: ASTM A 572, GR50 Fy = 50ksi Flange Width: 18 in. Flange Thickness: 1.25 in. Web Depth: 96 in. Web Thickness: 1.75 in.
19	4.3	5 (Shear) 10 (Flexure)	
20	2.0	10	**Long-Span Roof Panel - Center** #10 @ 12 in. each face, each direction
21	7.6	10	**Long-Span Roof Panel - Edge** #10 @ 12 in. each face, each direction #4 shear ties @ 12 in. or equivalent
22	6.2	10	**Short-Span Roof Panel - Center** #10 @ 12 in. each face, each direction
23	9.5	10	**Short-Span Roof Panel - Edge** #10 @ 9 in. each face, each direction #4 shear ties @ 9 in. or equivalent
24	3.7	10	**Safety-related area outside the Radiologically Controlled Area, Roof - Center** #10 @ 12 in. each face, each direction
25	5.8	10	**Safety-related area outside the Radiologically Controlled Area, Roof - Edge** #10 @ 12 in. each face, each direction #3 shear ties @ 12 in. or equivalent

Table 3.4-4 Aircraft Impact Analysis Results
(Sheet 2 of 2)

a) Where shear ties are specified, the extent of slab or wall needing shear reinforcement will be determined.
b) The energy absorption capacity of truss as evaluated here shall be met.
c) The energy absorption capacity of plate girder as evaluated here shall be met.
d) All steel member sizes provided are a minimum required size.

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 safety-related (SR) 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. 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 parameter limits for these events are detailed in Chapter 13. The SR 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 administrative controls for surveillance, maintenance, and testing.

SR 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 component and system demonstrate that the SR 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 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 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 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 boundary and are designed such that no credible failure at the output of the isolation device prevents the associated SR 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 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 components and systems. SR 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 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 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.

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 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;
c. The capability to prevent or mitigate the consequences of accidents which could result in potential exposures comparable to the applicable guideline exposures set forth in 10 CFR 20;
d. That all nuclear processes are subcritical, including use of an approved margin of subcriticality;
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.

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 safety classification.

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 measure of the QAPD and shall be applied to safety-related SSCs and to safety-related activities.

3.5.1.2.2 QL-2

This quality level is applied to selected SSCs and activities intended to support or protect the safety function of safety-related equipment. Quality Assurance Program elements are applied to an extent that is commensurate with the item's importance to safety. Implementing documents establish program element applicability.

3.5.1.2.3 QL-3

This quality level is applied to nonsafety-related SSCs and activities and does not support or protect the safety function of safety-related SSCs or activities. However, the performance of QL-3 SSCs and activities may be important to ensuring customer requirements are met, or operational or mission-related goals such as throughput, cost, or schedule are achieved. Controls, appropriate for the application, are applied to SSCs and activities using the SHINE Quality Assurance Program for efficiency to avoid the creation and use of a separate or redundant management system. The controls on these SSCs and activities do not impact QL-1 or QL-2 SSCs and activities or the regulatory basis of the facility.

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 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 which could result in potential exposures comparable to the applicable guideline exposures set forth in 10 CFR 20;
4. That all nuclear processes are subcritical, including use of an approved margin of subcriticality;
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
7. They do not degrade the function and performance of any SR 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 (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)				
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	SR	Category I	QL-1
Process Vessel Vent System	PVVS	SR	Category I	QL-1
Criticality Accident and Alarm System	CAAS	SR	Category I	QL-1
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	SR	Category I	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	SR	Category I	QL-1
Mo Extraction and Purification System	MEPS	SR	Category I	QL-1
Target Solution Preparation System	TSPS	SR	Category I	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	SR	Category I	QL-1
Target Solution Cleanup (UREX)				
Thermal Denitration				
Production Facility Biological Shield System	PFBS	SR	Category I	QL-1
Radioactive Drain System	RDS	SR	Category I	QL-1
Radioactive Liquid Waste Evaporation and Immobilization	RLWE	SR	Category I	QL-1
Aqueous Radioactive Liquid Waste Storage	RLWS	SR	Category I	QL-1
RCA Material Handling Systems	RMHS	SR	Category I	QL-1
Facility Control Room	FCR	SR	Category I	QL-1
Other Facility Systems and Components				
Hot Cell Fire Detection and Suppression System	HCFD	NSR	Category II	QL-2
Facility Instrument Air System	FIAS	NSR	Category III	QL-2
Stack Release Monitoring	SRM	NSR	Category III	QL-2
Facility Fire Detection and Suppression	FFPS	NSR	Category III	QL-2
Neutron Driver Assembly System	NDAS	NSR	Category III	QL-2
Primary Closed Loop Cooling System	PCLS	NSR	Category II	QL-2
Primary Closed Loop Cooling and Light Water Pool Makeup System	MUPS	NSR	Category III	QL-3
Health Physics Monitors	HPM	NSR	Category III	QL-3
TSV Process Control System	TPCS	NSR	Category II	QL-2
Normal Electrical Power Supply System	NPSS	NSR	Category II	QL-2
Inert Gas Control	IGS	NSR	Category III	QL-2
Material Handling	MHS	NSR	Category II	QL-2
Solid Radioactive Waste Packaging	SRWP	NSR	Category II	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
Organic Liquid Waste Storage and Export	OLWS	NSR	Category II	QL-2
Radioisotope Process Facility Cooling System	RPCS	NSR	Category III	QL-2
Moly Isotope Product Packaging System	MIPS	NSR	Category III	QL-2
Standby Diesel Generator System	SDGS	NSR	Category II	QL-2
Radiologically Controlled Area (RCA) Ventilation Zone 3	RVZ3	NSR	Category III	QL-2
Facility Ventilation Zone 4	FVZ4	NSR	Category III	QL-2
Facility Integrated Control System	FICS	NSR	Category III	QL-2
Facility Potable Water System	FPWS	NSR	Category III	QL-3
Facility Compressed Air System	FCAS	NSR	Category III	QL-3
Facility Breathing Air System	FBAS	NSR	Category III	QL-2
Facility Inert Gas System	FIGS	NSR	Category III	QL-2
Facility Welding System	FWS	NSR	Category III	QL-3
Facility Roof Drains System	FRDS	NSR	Category III	QL-3
Facility Sanitary Drains System	FSDS	NSR	Category III	QL-3
Facility Data and Communications System	FDCS	NSR	Category III	QL-2
Facility Lightning Protection System	FLPS	NSR	Category III	QL-3
Facility Demineralized Water System	FDWS	NSR	Category III	QL-3
Facility Chilled Water Supply and Distribution System	FCHS	NSR	Category III	QL-2
Facility Acid Reagent Storage and Distribution System	FARS	NSR	Category II	QL-2
Facility Alkaline Reagent Storage and Distribution System	FLRS	NSR	Category II	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-2
Facility Organic Reagent Storage and Distribution System	FORS	NSR	Category II	QL-2
Cathodic Protection System	CPS	NSR	Category III	QL-3
Emergency Lighting System	ELTG	NSR	Category II	QL-2
Facility Grounding System	FGND	NSR	Category III	QL-2
Lighting System	LTG	NSR	Category III	QL-3
Process Facility Wet Vacuum System	PFWV	NSR	Category III	QL-2
Process Facility Sampling System	PFSS	NSR	Category III	QL-3
Continuous Air Monitoring System	CAMS	NSR	Category III	QL-3

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 are nonsafety-related components, such as fans, 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.

Table 3.5-2 Likelihood Index Limit Guidelines

	Likelihood Category	Event Frequency Limits	Risk Index Limits
Likely Normal Facility Process Condition (FPC)	4	Multiple events per year	> or = 0
Not unlikely (Frequent FPC)	3	more than 10^{-4} per event, per year	>-4 <0
Unlikely (Infrequent FPC)	2	between 10^{-4} and 10^{-5} per event, per year	-4 to -5
Highly Unlikely (Limiting FPC)	1	less than 10^{-5} per event, per year	≤-5

3.5a IRRADIATION FACILITY

This section contains an evaluation of the design bases of the SHINE facility IF systems and components as measured against Appendix A to 10 CFR 50. This evaluation is used to develop the principal design criteria for the SHINE facility.

The general design criteria (GDC), which are divided into six groups and total sixty-four in number, are intended to establish minimum requirements for the design of nuclear power plants. The GDC are not applicable to the SHINE facility per NUREG 1537, Appendix A. However, they provide a proven basis with which to develop an initial assessment of the safety of the design of the SHINE facility.

Under the provisions of 10 CFR 50.34, an application for a Construction Permit must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for SSCs important to safety; that is, SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

For these reasons, the design basis of the IF has been compared to the GDC, as a means of good design practice, and the results of that comparison are presented herein. There are some cases where the application of a particular criterion is not directly measurable. In these cases, the comparison of the IF design to the interpretation of the criterion is discussed. For each of the sixty-four criteria, a general assessment of the IF design is made.

The IF design basis evaluation against the GDC is summarized in Table 3.5a-1.

3.5a.1 IRRADIATION FACILITY SAFETY-RELATED SYSTEMS AND COMPONENTS

This section addresses the design bases of safety-related systems and components. Systems that are common to the RPF and the IF are discussed in the section appropriate to their limiting safety classifications.

3.5a.2 IRRADIATION FACILITY CATEGORY I MECHANICAL SYSTEMS AND COMPONENTS

Seismic Category I mechanical equipment and components (as defined in Section 3.5) are qualified for operation under the DBEQ seismic conditions by, in order of preference, prototype testing, operating experience, or appropriate analysis.

All Seismic Category I mechanical equipment is designed to withstand loadings due to the DBEQ, vibrational loadings transmitted through piping, and operational vibratory loading, such as floor vibration due to other operating equipment, without loss of function or fluid boundary. This analysis considers the natural frequency of the operating equipment, the floor response spectra at the equipment location, and loadings transmitted to the equipment and the equipment anchorage.

These qualification documents and supporting analysis and test reports are maintained as part of the permanent plant record in accordance with the requirements of the SHINE QAPD.

3.5a.3 IRRADIATION FACILITY CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

Facility Seismic Category I instrumentation and electrical equipment (as defined by Section 3.5) 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 and 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.

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

3.5a.4 IRRADIATION FACILITY SAFETY-RELATED SYSTEMS AND COMPONENTS ENVIRONMENTAL DESIGN BASES CRITERIA

The safety-related equipment and components within the IU cells that are required to function post-accident are qualified to function in the post-accident environment. Certain subsystems and components used in the ESF systems are located in a mild environment.

3.5a.4.1 Qualification Methods

3.5a.4.1.1 Instrumentation and Electrical Systems and Components

Environmental qualification of 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.

3.5a.4.1.2 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.

3.5a.5 SUBCRITICAL ASSEMBLY SYSTEM DESIGN BASIS

The SCAS is comprised of several subsystems and components including:

- TSV and TSV dump tank (including dump tank valves).
- Neutron multiplier.
- Subcritical assembly support structure (SASS).
- Subcritical multiplication source.

Only the systems and components that contribute to the safe operation and shutdown of the IU are discussed in detail here.

3.5a.5.1 TSV and Dump Tank

3.5a.5.1.1 TSV and Dump Tank Design Basis Functions

a) The TSV holds the target solution in a subcritical state during irradiation.
b) The TSV and dump tank are part of the primary system boundary for the target solution and related fission products during irradiation and other facility normal operations and shutdown.
c) The TSV and dump tank are pressure vessels designed to conditions of service.
d) The TSV and dump tank are normally operated at sub-atmospheric conditions.
e) The dump tank holds the discharged target solution in a subcritical state after irradiation in normal operations and TSV emergency drainage in off-normal situations.
f) The dump tank is the primary isolation barrier between the target solution and the light water pool system for all post-accident design basis events.
g) During all normal operations and shutdown conditions, the TSV discharges heat primarily through forced convection by the PCLS. Some heat is removed by the TSV off-gas system by recombination and subsequent condensation of radiolysis products. A small amount of heat is also transported to the LWPS by long-range ionizing radiation.
h) During loss of forced cooling, the TSV discharges heat primarily through natural convection and conduction paths to the LWPS.
i) During all normal operations and shutdown conditions, as well as post-accident conditions, the dump tank discharges waste heat by natural convection and conduction paths to the LWPS.

3.5a.5.1.2 TSV and Dump Tank Design Basis Values

a) The TSV and associated dump tank have a 30-year design life.
b) The TSV and associated dump tank are of welded construction.
c) The TSV and associated dump tank design pressures ensure a robust vessel design that can withstand credible hydrogen deflagrations with no vessel damage.
d) The TSV is exposed to total neutron fluxes of up to [Proprietary Information] neutron per centimeter-second (n/cm^2s).
e) In post-accident conditions such as a DBEQ, the TSV and associated dump tank maintain their primary system boundary and contain the target solution.
f) For normal and limiting environmental operating envelopes the TSV and associated dump tank maintain their primary system boundary and contain the target solution.

3.5a.5.2 Neutron Multiplier

Refer to Subsection 4a2.2.6 for a detailed discussion. This subsystem does not perform a safety-related function.

3.5a.5.3 Subcritical Assembly Support Structure

Refer to Subsection 4a2.2.5 for a detailed discussion.

3.5a.5.3.1 SASS Design Basis Functions

a) In post-accident conditions such as deflagration or detonation within the primary system boundary, the SASS maintains its integrity and passively confines target solution and fission products in the event of a loss of TSV integrity.
b) The SASS functions as the shell side of PCLS during normal operations and shutdown of the TSV.
c) The SASS maintains structural integrity and full function for normal operations and shutdown as well as post-accident scenarios.
d) The SASS provides structural support and geometric configuration of the subcritical assembly components.

3.5a.5.3.2 SASS Design Basis Values

a) The SASS has a 30-year design life.
b) The SASS maintains structural and geometric integrity post-DBEQ with limited localized plastic deformation. Loading includes a consideration of seismic slosh events.
c) The SASS design pressure is at least as high as the TSV design pressure so that it can safely accommodate a breach of the TSV.
d) The SASS is of welded construction with bolted access covers.
e) The SASS is designed for external loading conditions of 10 psi (69 kPa).

3.5a.5.4 Subcritical Multiplication Source

Refer to Subsection 4a2.2.4 for a detailed discussion. This subsystem does not perform a safety-related function.

3.5a.6 PRIMARY CLOSED LOOP COOLING SYSTEM

Refer to Sections 4a2 and 5a2 for a detailed discussion. This system does not perform a safety-related function.

3.5a.7 LIGHT WATER POOL SYSTEM

3.5a.7.1 LWPS Cooling Loop

Refer to Section 5a2.2 for a detailed discussion. This system does not perform a safety-related function.

3.5a.7.2 LWPS Pool Structure

3.5a.7.2.1 LWPS Pool Structure Design Basis Functions

a) Provide passive heat sink for a portion of TSV heat loads during normal operation and shutdown.
b) Provide passive heat sink for TSV and dump tank heat loads and residual/decay heat removal under post-accident conditions.
c) Withstand the effects of a DBEQ and maintain structural integrity.
d) Provide additional neutron and gamma shielding of the target solution during normal operations and shutdown and in post-accident conditions.
e) Provide passive heat removal whenever PCLS is unavailable.
f) Systems interfacing with the LWPS are designed so that any abnormal operations or post-accident failures in those interfacing systems do not cause the water level to fall below safe limits.
g) The design of the pool excludes installation of drains, permanently connected systems, or other features that could, by abnormal operation or failure, cause a significant loss of water.
h) Pool water is maintained within acceptable quality limits.

3.5a.7.2.2 LWPS Pool Structure Design Basis Values

a) LWPS has a 30-year design life.
b) Pool liner and structural system remains structurally and functionally intact through DBAs.
c) Pool dimensions are approximately [Proprietary Information] (approximately [Proprietary Information] gallons without installed equipment).
d) Pool normal operating temperatures are between 68°F (20°C) and 75°F (24°C).
e) Pool removes [Proprietary Information] Btu/hr [Proprietary Information] in all normal operational conditions at licensed power limit.
f) Pool removes approximately [Proprietary Information] in 90 days following a DBA where the IU is shutdown and PCLS is unavailable.
g) Pool depth is maintained within technical specification limits.

3.5a.8 TARGET SOLUTION VESSEL OFF-GAS SYSTEM

3.5a.8.1 TOGS Design Basis Functions

a) Manage hydrogen and oxygen production through recombination during normal operations and DBAs, and in post-accident environments.
b) Provide continuous data on hydrogen concentrations.
c) Provide a sealed blower system to create differential pressures over the headspace of the TSV to create constant gas sweep within the TOGS closed loop.
d) Provide a portion of the primary system boundary to contain fission product off-gas generated during target solution irradiation and maintain primary system boundary during normal operations and credible accidents.
e) Provide a TRPS trip signal if acceptable hydrogen concentrations are violated.
f) Condense and return water vapor in the off-gas to the TSV in order to minimize water loss in the target solution.
g) Remove iodine from the off-gas.

3.5a.8.2 TOGS Design Basis Values

a) Maintain bulk hydrogen concentrations in the TSV headspace and the TSV dump tank below the lower flammability limit.
b) Operate at slightly sub-atmospheric pressures during normal conditions.

3.5a.9 NEUTRON DRIVER

Refer to Section 5a2 for a detailed discussion. This system does not perform a safety-related function.

3.5a.10 CONFINEMENT BARRIERS

Confinement is provided by a combination of the IU cell structures, the supporting ventilation systems, and automatic isolation valves or bubble-tight dampers on all IU cell penetrations.

3.5a.10.1 RCA Ventilation Zone 1

Refer to Subsection 9a2.1.1 for a detailed description.

3.5a.10.1.1 RCA Ventilation Zone 1 Design Basis Functions

a) Maintain pressure gradients throughout the Zone 1 areas to ensure the proper flow of air from the least potentially contaminated areas to the most potentially contaminated areas, thereby limiting the spread of airborne radioactive materials.
b) Provide confinement of airborne radioactive materials by providing for the rapid, automatic closure of isolation dampers at the ventilation penetrations to the IU cells, hot cells, other process enclosures requiring isolation, and at the RCA boundary for various accident conditions.
c) Process exhaust flow from the PVVS.
d) The isolation dampers remain functional for DBA.
e) The system has sufficient redundancy to perform its safety function in the event of single failure.
f) The components are environmentally qualified to perform their safety function following a DBA.
g) Provide sufficient air flow to the IU cell during normal operation to maintain Ar-41 concentrations at acceptable levels.

3.5a.10.1.2 RZV1 Design Basis Values

a) The RVZ1 has a 30-year design life.
b) Provide an integrated leak rate for the confinement boundaries that meets the requirements of the accident analyses described in Chapter 13 and the dose limits of 10 CFR 20 for DBAs.
c) Maintain its leakage rate performance for at least 40 days following a DBA (Subsection 11.1.1.1).
d) Maintain its confinement capabilities for at least 40 days following the DBA (Subsection 11.1.1.1)

3.5a.10.2 Irradiation Cell Biological Shielding

Refer to Section 4a2.5 for a detailed discussion.

3.5a.10.2.1 Irradiation Cell Biological Shielding Design Basis Functions

a) Provide biological shielding from radiation sources in the IU cells for workers in the occupied areas of the facility.
b) Limit physical access to the IU cells.
c) Survive DBEQ effects, without loss of structural integrity.

3.5a.10.2.2 Biological Shielding Design Basis Values

a) Provide dose rates at 12 in. (30.48 cm) from surface of shielding of 0.25 mrem/hr or less for normally-occupied areas. Localized dose rates at penetrations and during some planned operations, may be higher and are posted appropriately.

3.5a.11 TRITIUM PURIFICATION SYSTEM

Refer to Subsection 9a2.7.1 for a detailed discussion.

3.5a.11.1 TPS Design Basis Functions

a) Provide tritium gas to the NDAS.
b) Receive tritium gas from a bottle periodically.
c) Receive tritium/deuterium gas from the NDAS.
d) Remove water, organic impurities, and deuterium from the tritium gas.
e) Receive flush gas from the NDAS and remove tritium from the flush gas.
f) Monitor TPS waste streams that are discharged to the exhaust systems.
g) Scrub the glovebox atmosphere for tritium that may escape from the purification process.
h) Minimize chronic or acute tritium release.

3.5a.11.2 TPS Design Basis Values

a) Provide approximately [Proprietary Information][Security-Related Information] tritium gas [Proprietary Information] to the NDAS.
b) Receive approximately [Proprietary Information][Security-Related Information] tritium/ deuterium gas from the NDAS.

3.5a.12 INSTRUMENTATION AND CONTROL

3.5a.12.1 TSV Reactivity Protection System

Refer to Section 7a2.4 for a detailed discussion.

3.5a.12.1.1 TRPS Design Basis Functions

a) Provide and maintain automatic safe shutdown of IUs.
b) Function during and after DBAs.

c) Meet single failure criteria through the use of redundant independent channels.
d) Operate independently of process control systems.

3.5a.12.1.2 TRPS Design Basis Values

a) TRPS has a 30-year design life.

3.5a.12.2 TSV Process Control System

Refer to Section 4a for further discussion. This system does not provide a safety-related function.

3.5a.12.3 Neutron Flux Detection System

Refer to Subsection 7a2.4.3 for a detailed discussion.

3.5a.12.3.1 NFDS Design Basis Functions

a) Provide flux measurement in TSV operating ranges.
b) Provide signals to TRPS.
c) Output signals are isolated.
d) Meet single failure criteria with independent channels.

3.5a.12.3.2 NFDS Design Basis Values

a) NFDS has a 30-year design life.

3.5a.12.4 Engineered Safety Features Actuation System

Refer to Section 7a2.5 for a detailed description.

3.5a.12.4.1 Engineered Safety Features Actuation System Design Basis Functions

a) Activate isolation functions of necessary confinement areas upon measured parameters exceeding setpoints.
b) Provide actuation and isolation of affected confinement areas to ensure doses during DBAs are maintained below 10 CFR 20 limits.
c) Functioning during and after normal operations, shutdown conditions, and DBAs.

3.5a.12.4.2 ESFAS Design Basis Values

a) ESFAS has a 30-year design life.
b) ESFAS actuates the necessary isolation functions within the time required by the safety analysis with acceptable margin.

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 SR SSCs 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 1 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 1—*Quality standards and records.* Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.	<u>As Applied</u> 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. <u>Means of Compliance</u> 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.
Criterion 2—*Design bases for protection against natural phenomena.* Structures, systems, and components important to safety shall be 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 functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the 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 in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.	<u>As Applied</u> 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 functions. The design bases for these SSCs reflect: 1) 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. 2) Appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena. 3) The importance of the safety functions to be performed. <u>Means of Compliance</u> The design basis and criteria are discussed in this chapter.

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 2 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 3—*Fire protection*. Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.	<u>As Applied</u> 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 IU cell 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 ensure that their rupture or inadvertent operation does not significantly impair the safety capability of these SSCs. <u>Means of Compliance</u> The facility design has been subject to a Fire Hazards Analysis (FHA). This analysis has led to the establishment of designated fire zones and areas within the facility. Where necessary within confinement areas that contain fissile materials or where criticality and access are an issue, required systems are manually initiated by an operator after review of a detection signal. The facility fire protection system is designed such that a failure of any component of the system will not significantly impair the safety capability of SSCs. See Chapter 9 for more information.
Criterion 4—*Environmental and dynamic effects design bases*. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.	<u>As Applied and Means of Compliance</u> 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, including loss of primary coolant. These structures, systems, and components are appropriately protected against dynamic effects, including the effects of missiles that may result from equipment failures and from external events and conditions outside the SHINE facility. 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. The application of the design criteria are described in detail in this chapter.

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 3 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 5—*Sharing of structures, systems, and components*. Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.	Criterion 5 does not apply to the SHINE facility.
Criterion 10—*Reactor design*. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.	<u>As Applied and Means of Compliance</u> The SHINE facility does not include a reactor. However, this criterion has been assessed against the IU design. The TSV, target solution, and associated primary coolant, TOGS, TPCS, and TRPS are designed with appropriate margin to assure that specified acceptable system design limits (such as criticality limits) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Refer to Chapter 4 for further discussion.

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 4 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 11—*Reactor inherent protection*. The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.	<u>As Applied and Means of Compliance</u> The SHINE facility does not include a reactor. However, this criterion has been assessed against the IU inherent feedback characteristics. The TSV, target solution, and associated PCLS are designed so that the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity. The IU is designed to have a reactivity response that regulates or damps changes in power level to a level consistent with safe and efficient operation. This is achieved through the use of an aqueous target solution. The inherent dynamic behavior of the target solution is characterized in terms of: a. Target solution temperature coefficient of reactivity. b. Target solution negative void coefficient. c. Target solution thermal inertia. d. Low reactivity worth per unit volume. The combined effect of these coefficients is termed the power coefficient. Temperature reactivity feedback occurs simultaneously with a change in target solution temperature and opposes the power change that caused it, therefore contributing to system stability. The target solution void coefficient is of importance during irradiation. Nuclear design requires the void coefficient inside the TSV to be negative. The negative void reactivity coefficient provides an inherent negative feedback during power transients. Because of the large negative coefficient of reactivity, the IU has a number of inherent advantages, such as: a. Ease of control due to minimal power oscillation. b. Sustained and stable operations in a subcritical state. Therefore the IU, target solution, and associated PCLS are designed so that prompt inherent dynamic behavior tends to compensate for any rapid increase in reactivity. Refer to Section 4a2.6 for a detailed discussion.

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 5 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 12—*Suppression of reactor power oscillations*. The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.	<u>As Applied and Means of Compliance</u> The SHINE facility does not include a reactor. However this criterion has been assessed against the suppression of IU power oscillations. The TSV, target solution, and associated PCLS, TOGS, TPCS, and TRPS are designed to ensure that power oscillations that can result in conditions exceeding specified acceptable target solution design limits can be reliably and readily detected and suppressed. The IU is designed to ensure that no power oscillation will cause primary system design limits to be exceeded. The power reactivity coefficient is the composite simultaneous effect of the target solution temperature or Doppler coefficient, moderator void coefficient, and target solution thermal inertia to the change in power level. It is negative and well within the range required for adequate damping of power. Analytical studies indicate that for aqueous fuel solutions (similar to the target solution), under-damped, unacceptable power distribution behavior could only be expected to occur with extremely large power coefficients. Operating experience has shown aqueous fuel solutions to be inherently stable against xenon-induced power instability. The large negative operating coefficients provide: a) Stable power levels with a well-damped behavior and little undershoot or overshoot in the heat transfer response. b) Strong damping of spatial power disturbances. The TRPS design provides protection from excessive target solution temperatures and protects the primary system boundary from pressures that threaten the integrity of the system. Local abnormalities are sensed, and, if protection system limits are reached, corrective action is initiated through an automatic discharge of the target solution to a criticality safe holding vessel. High integrity of the protection system is achieved through the combination of logic arrangement, trip channel redundancy, power supply redundancy, and physical separation. The TSV and associated PCLS, TOGS, TPCS, and TRPS are designed to suppress any power oscillations that could result in exceeding target solution design limits. Refer to Sections 4a and 5.5a for a detailed discussion.

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 6 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 13—*Instrumentation and control*. Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.	<u>As Applied</u> Instrumentation is provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to ensure adequate safety, including those variables and systems that can affect the fission process, the integrity of the primary system boundary, the PCLS pressure boundary, and the confinement and its associated systems. Appropriate controls are provided to maintain these variables and systems within prescribed operating ranges. <u>Means of Compliance</u> The fission process is monitored and controlled from source range through high range (irradiation mode). The high range of the NFDS detect target solution conditions that threaten the overall integrity of the TSV due to excess power generation and provide a signal to the TRPS. Flux monitors, located in the light water pool, are used for neutron detection. The monitors are located to provide optimum monitoring in the high range. Additional instrumentation is provided to monitor temperature, pressure, and hydrogen gas levels. This information is input to the TRPS for appropriate automatic response. The TRPS is the protection system for initiating protective functions associated with the IU trip. The ESFAS is the controlling system for initiating the active engineered safety features for isolation of the IU cell within the IF. The TRPS protects the primary system boundary integrity by monitoring TSV and associated parameters and causing neutron driver shutdown and TSV dump when predetermined setpoints are exceeded. Additionally, in order to provide protection against the consequences of accidents involving the release of radioactive materials from the primary system boundary (e.g., a rupture or leakage of the TSV or associated piping), the ESFAS initiates automatic isolation of the necessary ventilation and piping that penetrate the IU cell and pose a potential leakage path, whenever monitored variables exceed preselected limits. The ESFAS monitors parameters determined necessary by the safety analysis to evaluate if release of radioactive material has occurred within the IF that exceeds normal operational levels. Radiation monitors monitor radiation levels of the IU Cell and of various processes and provide actuation signals to the ESFAS whenever pre-established limits are exceeded. Upon actuation, ESFAS adjust ventilation alignment to isolate the affected area and isolates any other systems penetrating the confinement barrier that are determined could result in excessive leakage. As noted above, adequate instrumentation has been provided to monitor system variables in the IU, associated subsystems and the IU cell confinement. Appropriate controls have been provided to maintain the variables in the operating range and to initiate the necessary corrective action in the event of abnormal operational occurrence or accident. Refer to Section 7a for a further discussion.

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 7 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 14—*Reactor coolant pressure boundary*. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.	As Applied and Means of Compliance The SHINE facility does not employ a reactor vessel. However, the primary system boundary (PSB) provides a similar comprehensive system. The PSB is designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. The PSB normally operates near atmospheric pressure, and only experiences significant pressures during certain accident conditions. The PSB is designed to withstand the calculated pressures with sufficient margin without gross rupture. The primary cooling pressure boundary fully encloses the TSV. The LWPS provides cooling to the multiplier and target chamber, and increases the depth of defense to the release of fission products. The pressure boundaries for the PCLS and LWPS primary coolant systems have been designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. Metal surfaces in contact with primary coolant are either zircaloy, aluminum, or stainless steel, except where otherwise noted in the design specifications. The system components outside of the light water pool have a low probability of serious leakage or gross failure. Rupture of the light water pool liner is virtually impossible since it is supported on the bottom and sides by reinforced concrete. In addition, the primary and pool coolant systems operate at low pressures and temperatures.
Criterion 15—*Reactor coolant system design*. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.	As Applied and Means of Compliance The SHINE facility does not employ a reactor vessel, however, the PSB and the PCLS pressure boundary provide a similar comprehensive system. The PSB and PCLS and associated auxiliary, control, and protection systems are designed with sufficient margin to ensure that the design conditions of the PSB and PCLS pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. These systems are designed, fabricated, erected, and tested to stringent quality requirements and appropriate codes and standards, which ensure high integrity of the PSB and PCLS pressure boundary throughout the plant lifetime. The auxiliary, control, and protection systems associated with the PSB and PCLS (refer to Chapter 7 for details) act to provide sufficient margin to assure that the design conditions of the PSB and PCLS pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. As described in the evaluation of Criterion 13, instrumentation is provided to monitor essential variables to ensure that they are within prescribed operating limits. If the monitored variables exceed their predetermined settings, the auxiliary, control, and protection systems automatically respond to maintain the variables and systems within allowable design limits.

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 8 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 16—*Containment design*. Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.	<u>As Applied and Means of Compliance</u> The SHINE facility does not have a containment but has a confinement per the NUREG-1537 definition. As opposed to containment, the IU confinement provided by the IU cell (ICBS) and associated systems is provided to establish a barrier against the uncontrolled release of radioactivity to the environment and to ensure that the confinement design conditions important to safety are not exceeded for as long as postulated accident conditions require. The confinement system that provides a leakage barrier consists of the confinement and supporting confinement isolation systems. Additional systems provided to control the release of radioactivity from the confinement to the environment are the RCA ventilation system, and its associated filtration systems. These systems are designed to protect the public from the consequences of a breach of the PSB.

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 9 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 17—*Electric power systems.* An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents. The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure. Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained. Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.	<u>As Applied and Means of Compliance</u> An on-site electric power system (defined to be the SHINE facility provided power supply) and an off-site (defined to be the public utility transmission network supplied substations) electric power system are provided to permit functioning of structures, systems, and components important to safety. The functions for each system (assuming the other system is not functioning) are to provide sufficient capacity and capability to ensure that: 1) Specified acceptable target solution design limits and design conditions of the primary system boundary are not exceeded as a result of anticipated operational occurrences and 2) The target solution and TSV are cooled and confinement integrity and other vital functions are maintained in the event of postulated accidents. The on-site electric power supply systems include a safety-related UPSS, a commercial grade diesel generator system, and the on-site electric distribution system. The UPSS has sufficient independence, redundancy, and testability to perform its safety functions assuming a single failure. Each of the redundant UPSS circuits are designed to be available within a few seconds following a loss of all off-site electric power, to ensure that specified acceptable target solution design limits and design conditions of the primary system boundary are not exceeded. It performs this function by supplying power to safety-related SSCs, such as TOGS blowers and related instrumentation. The diesel generator system performs no safety functions but provides an additional degree of defense-in-depth to the electric power system. Portions of the on-site electric distribution, those circuits which operate in conjunction with the UPSS, are safety-related. The SHINE facility receives a single physically off-site independent power circuit. This off-site power circuit consists of two power feeds connected to two local outdoor transformers. Provisions are included to minimize the probability of losing electric power from the UPSS as a result of or coincident with, the loss of power from the transmission network. Refer to Section 6a and Chapter 8 for a detailed discussion.

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.	<u>As Applied and Means of Compliance</u> The SHINE facility electric power system important to safety, as identified by the 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.	<u>As Applied</u> 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. <u>Means of Compliance</u> 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.

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 11 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 20—*Protection system functions.* The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.	As Applied The protection system is designed to: a) Initiate automatically the operation of appropriate systems including the reactivity control systems, to ensure that specified acceptable target solution design limits are not exceeded as a result of anticipated operational occurrences; and, b) To sense accident conditions and to initiate the operation of systems and components important to safety. Means of Compliance The protection system for the SHINE subcritical assembly is the TRPS. The TRPS is designed to provide timely protection and control against the onset and consequences of conditions that threaten the integrity of the target solution and the PSB. Damage is prevented by initiation of an automatic IU shutdown if monitored system variables exceed pre-established limits of normal operation. Trip settings are selected and verified to be far enough above or below operating levels to provide proper protection but reduce the probability of spurious shutdowns. Specifically, these process parameters initiate a shutdown in time to prevent the TSV from exceeding thermal-hydraulic safety limits during abnormal operational transients. Response by the TRPS is prompt and the total shutdown time is short. In addition to the TRPS protective functions and TPCS control, the ESFAS senses accident conditions and automatically initiates the operation of active ESFs to ensure doses from accidents are within acceptable limits. Components such as bubble-tight dampers which provide for the isolation of the IU cell following a breach of the TSV to prevent the release of significant amounts of radioactive materials from the IU cell are actuated by ESFAS. The design of the protection system satisfies the functional requirements as specified in Criterion 20.

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 12 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 21—*Protection system reliability and testability.* The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.	<u>As Applied</u> The protection system is designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system are sufficient to ensure that: a. No single failure results in loss of the protection function; and, b. Removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system is designed to permit periodic testing, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred. <u>Means of Compliance</u> The protection system for the SHINE subcritical assembly is the TRPS. The TRPS design provides assurance that, through redundancy, it has sufficient reliability to fulfill the single-failure criterion. No single component failure, intentional bypass maintenance operation, calibration operation, or test to verify operational availability impairs the ability of the system to perform its intended safety function. Additionally, the system design ensures that when a shut-down trip-point is exceeded there is a high shutdown probability. There is sufficient electrical and physical separation between channels and between trip logics monitoring the same variable to prevent environmental factors, electrical transients, and physical events from impairing the ability of the system to respond correctly. The TRPS includes design features that permit in-service testing. This ensures the functional reliability of the system should the variable important to safety exceed the corrective action setpoint. The TRPS initiates an automatic IU shutdown if the monitored plant variables exceed pre-established limits. Manual trip testing is performed by operating one of the two manual shutdown controls. The total test verifies the ability to de-energize the TSV dump tank valves and de-energize the high voltage power supply to the neutron driver. Refer to Subsection 7a2.4.1 for a detailed discussion.

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 13 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 22—*Protection system independence*. The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.	As Applied The protection system is designed to ensure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or are demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, are used to the extent practical to prevent loss of the protection function. Means of Compliance The protection system for the SHINE subcritical assembly is the TRPS. This is achieved through the use of multiple independent input sensors within the protective system architecture and independent TRPS controllers. Through the use of diverse component design, there are no identified common mode failures that could compromise safety. Signal lines from the process sensors, transmitters, and controllers that provide redundant protective functions are separated and clearly identified. Therefore, the protection system satisfies the intent of IEEE-603 with regard to channel independence. Refer to Subsection 7a2.4.1 for a detailed discussion.
Criterion 23—*Protection system failure modes*. The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.	As Applied The protection system is designed to fail into a safe state if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced. Means of Compliance The protection system is designed to fail into a safe state. Loss of electric power of the TRPS causes a fail to a safe state through de-energizing the neutron driver high voltage power supply and opening the TSV dump valves. Intentional bypass, maintenance operation, calibration operation, or test results in a single channel trip. A failure of a TRPS input or subsystem component will produce a trip signal in the respective logic scheme so that receipt of any one additional trip signal will result in an IU trip. Furthermore, the TSV dump valves fail open on a loss of electric power, and the neutron driver high voltage power supply de-energizes on a loss of electric power. The environmental conditions in which the instrumentation and equipment of the protection system must operate were considered in establishing the component specifications. Instrumentation specifications are based on the worst expected ambient conditions in which the instruments must operate. The failure modes of the protection system are such that it fails into a safe state as required by Criterion 23. Refer to Section 7a2.3 for a detailed discussion.

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 14 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 24—*Separation of protection and control systems*. The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.	<u>As Applied and Means of Compliance</u> The TRPS provides the safety-related trip functions for the TSV to protect the PSB. Only limited control system functionality is required for the subcritical assembly due its subcritical nature and absence of control rods through the use of signal isolation devices. There is separation between the TRPS and the TPCS through the use of signal isolation devices. Interconnections between the control systems and the protection systems are minimized to the extent possible. Therefore, failure in the controls and instrumentation of process systems (e.g., TPCS or facility integrated control system (FICS)) cannot induce failure of any portion of the TRPS. High shutdown reliability is designed into the TRPS. The shutdown signal and mode of operation overrides all other signals. The TRPS system is separated from TPCS and other control systems as required in Criterion 24. Refer to Chapter 7 for a detailed discussion.
Criterion 25—*Protection system requirements for reactivity control malfunctions*. The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.	<u>As Applied and Means of Compliance</u> The protection system is designed to ensure that specified acceptable target solution design limits are not exceeded for any single malfunction of the reactivity control systems, such as insertion of excess reactivity through excess target solution addition, deflagration, detonation, or other means. The protection system provides protection against the onset and consequences of conditions that threaten the integrity of the target solution, PSB, and the PCLS pressure boundary. The design of the protection system ensures that specified acceptable target solution design limits are not exceeded for any single malfunction of the reactivity control systems as specified in Criterion 25. Refer to Section 7a2.2 for a detailed discussion.

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 15 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 26—*Reactivity control system redundancy and capability*. Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.	<u>As Applied and Means of Compliance</u> The reactivity control system for the SHINE subcritical assembly consists of two redundant dump valves and associated operators, either of which can safely shutdown the subcritical assembly and bring the target solution to a shutdown state in the TSV dump tank. Each dump valve and associated flow path provides drain capability for the TSV that exceeds the fill capacity by significant margin. Each dump valve is highly reliable and meets stringent quality assurance standards. Further, concurrent failure of both dump valves would not result in a critical system or accident scenario unless other malfunctions occurred concurrently, which is highly unlikely. While not a direct reactivity control system, the neutron driver is also de-energized by the protection system when an IU shutdown is required. This shutdown of the neutron driver decreases the fission power of the subcritical assembly to essentially zero in normal and accident conditions. Refer to Subsection 7a2.1.2 for a detailed discussion.
Criterion 27—*Combined reactivity control systems capability*. The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.	<u>As Applied and Means of Compliance</u> This criterion is not applicable to the SHINE system. Sufficient capacity to shut down the subcritical assembly exists as the TSV dump tank is designed to be subcritical by geometry for the most reactive uranium concentration in solution. Refer to Section 7a2.3 for a detailed discussion.
Criterion 28—*Reactivity limits.* The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.	<u>As Applied and Means of Compliance</u> The TSV dump valves cannot add reactivity to the system. The primary cooling systems and target solution fill subsystem are designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither (1) result in damage to the PSB greater than limited local yielding, nor, (2) sufficiently disturb the TSV, its support structures or other TSV internals to impair significantly the capability to drain the TSV. These postulated reactivity accidents include consideration of, excess target solution addition, changes in primary cooling temperature and pressure, deflagration or detonation in the TOGS, and cooling water addition. Refer to Chapter 13 for a detailed discussion.

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 16 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 29—Protection against anticipated operational occurrences. The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.	As Applied The TRPS and TPCS systems are designed to ensure an extremely high probability of accomplishing their control and intended safety functions in the event of anticipated operational occurrences. Means of Compliance High functional reliability of the TRPS and TPCS is achieved through the combination of logic arrangement, redundancy, physical and electrical independence, functional separation, fail-safe design, and inservice testability. An extremely high reliability of the protection and reactivity control systems in response to anticipated operational occurrences is maintained by a thorough program of inservice testing and surveillance of control components and systems. No anticipated operational occurrences result in the failure of the TRPS or TPCS to perform their intended functions. The TRPS and TPCS are powered through a highly reliable UPS system, and no emergency electric power is required to safely shut down the subcritical assembly. No anticipated operational occurrences result in the failure of the TSV dump valves to perform their safety-related function to shutdown the subcritical assembly. The TSV dump valves do not require electrical power to perform their safety-related function. The capabilities of the protection and reactivity control systems to perform their safety functions in the event of anticipated operational occurrences are satisfied in agreement with the requirements of Criterion 29. Refer to Subsections 7a2.3.1 and 7a2.4.1 for a detailed discussion.

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 17 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 30—*Quality of reactor coolant pressure boundary*. Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.	As Applied Components which are part of the PSB, PCLS, and LWPS are designed, fabricated, erected, and tested to the highest quality standards practical. Means are provided for detecting and, to the extent practical, identifying the location of the source of PSB or primary coolant leakage. Means of Compliance The pressure retaining components of the PSB, PCLS, and LWPS are designed and fabricated to retain their integrity during normal and postulated accident conditions. Accordingly, components which comprise the PSB, PCLS, and LWPS pressure boundaries are designed, fabricated, erected, and tested in accordance with the recognized industry codes and standards listed in Chapter 5. Further, product and process quality assurance is provided as described in Chapter 12 to ensure conformance with the applicable codes and standards, and to retain appropriate documented evidence verifying compliance. Surfaces in contact with primary coolant or target solution are stainless steel, aluminum, or zirconium alloys, except where noted in design specifications. Means are provided for detecting PSB, PCLS, and LWPS pressure boundary leakage. Very low levels of leakage out of the PSB is detected through sampling. The location of the increased radiation level assists in identifying the source of the leakage, and the magnitude of the reading assists in determining the magnitude of the leakage. Leakage into the PSB is detected through changes in TSV level, TSV dump tank level, and/or changes in the neutron flux levels. Leakage into the PSB is analyzed as part of the accident analyses and does not challenge the subcritical nature of the subcritical assembly. Leakage in the PCLS is readily detected through abnormal changes in head tank level or increases in pool level. Leakage in the LWPS is detected through changes in pool level and through the leak chase system behind the pool liner. Furthermore, the PSB, PCLS, and LWPS operate at low pressures and temperatures, greatly limiting the potential for leakage. Refer to Chapters 4 and 5 for a detailed discussion.

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 18 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 31—*Fracture prevention of reactor coolant pressure boundary*. The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.	As Applied and Means of Compliance The PSB and PCLS are designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions the probability of rapidly propagating fracture is minimized. The PSB and PCLS operate at relatively low temperatures and pressures during normal operation and accident conditions. Minimal stresses are incurred from temperature and pressure transients during normal operational cycles. The TSV and SASS are exposed to neutron fluxes similar to research reactors due to their locations near the target solution during the irradiation process. The effect of irradiation on the material properties over the lifetime of the components is in incorporated into the design to ensure adequate margin exists throughout system and component life. The TSV is part of the PSB, and provides the primary barrier to fission product release. The SHINE design has also included a closed support structure design (SASS), which encapsulates the TSV to provide an additional degree of defense-in-depth should the TSV suffer a loss of integrity. This SASS design greatly reduces the consequences in the unlikely event of a fracture of the TSV. Refer to Section 4a for further discussion.

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 19 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 32—*Inspection of reactor coolant pressure boundary*. Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.	<u>As Applied and Means of Compliance</u> The PSB and PCLS pressure boundary designs include provisions for inservice inspection to ensure structural and leak tight integrity. Access to the IU cells and working platforms provide access for examination of the SASS, PSB and PCLS piping, and limited inspection of the TSV via the light water pool. Also, where insulation is necessary, removable insulation is provided on the TOGS and the PCLS piping systems to allow inspection.
Criterion 33—*Reactor coolant makeup.* A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.	<u>As Applied and Means of Compliance</u> Primary cooling makeup is provided to the PCLS and the LWPS by the MUPS. In the event a minor leak is detected, the manual addition of makeup water can be provided to the PCLS. MUPS has no safety function as the forced convection of water by the PCLS is not safety-related. MUPS and forced convection by the PCLS are not available upon a loss of off-site power. Loss of primary coolant from the PCLS has no potential to prevent the decay heat removal capability of the TSV via natural convection due to the layout and arrangement of the PCLS. Even if decay heat removal capability for the TSV was lost due to complete evacuation of all PCLS water, decay heat would still be adequately removed by draining the TSV to the TSV dump tank, which is contained within the light water pool. Refer to Section 5a2.5 for a detailed discussion.
Criterion 34—*Residual heat removal*. A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.	<u>As Applied and Means of Compliance</u> A means to remove residual heat is provided by the LWPS. The system safety function is to transfer fission product decay heat and other residual heat from the TSV and PCLS at a rate such that specified acceptable design limits and the design conditions of the TSV and PCLS pressure boundary are not exceeded. The LWPS is capable of removing decay heat from target solution contained within the TSV or within the TSV dump tank. This safety-related decay heat removal function is passive in nature. Refer to Section 5a2.2 for a detailed discussion.

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 20 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 35—*Emergency core cooling*. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts. Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure	No emergency cooling system is required for the SHINE subcritical assembly. As discussed in Chapter 13, the light water pool, PCLS, and subcritical assembly design allows for adequate cooling of the target solution in accident scenarios. No active emergency cooling system is required for loss of primary cooling scenarios. Due to the robust design of the IU cell walls (approximately 6 feet thick reinforced concrete) and the design of the stainless steel pool liner, a loss of pool water causing loss of heat removal capability is not a credible event.
Criterion 36—*Inspection of emergency core cooling system*. The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.	Criterion 36 does not apply to the SHINE IF.
Criterion 37—*Testing of emergency core cooling system*. The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.	Criterion 37 does not apply to the SHINE IF.

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 21 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 38—*Containment heat removal*. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels. Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.	<u>As Applied and Means of Compliance</u> The SHINE facility does not have a containment but has a confinement per the NUREG-1537 definition. A system to remove heat from the IU cell confinement, as opposed to containment, is provided for normal operation and shutdown. This is described in Section 9a2.1. An engineered safety system is not required for post-accident cooling due to the low operating powers involved with the IF and correspondingly low residual heat. Refer to Subsection 6a2.2.1 for further confinement discussion.
Criterion 39—*Inspection of containment heat removal system*. The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.	Criterion 39 does not apply to the SHINE IF.
Criterion 40—*Testing of containment heat removal system*. The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.	Criterion 40 does not apply to the SHINE IF.
Criterion 41—*Containment atmosphere cleanup*. Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained. Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.	<u>As Applied and Means of Compliance</u> The SHINE facility does not have a containment but has a confinement per the NUREG-1537 definition. Post-accident functioning of equipment following release of material into the IU cell does not require confinement atmosphere cleanup to reduce fission products, hydrogen, oxygen, or other substances that may be released into the IU cell. Due to the low power and low temperature operation of the subcritical assembly, hydrogen production through zirconium-water reaction is not credible. Upon ESFAS actuation of IU cell isolation, the ventilation dampers and other necessary penetrations to the IU cell are isolated. No active cooling or ventilation is required to maintain isolation or to maintain the target solution in a shutdown state. Refer to Section 9a2.1 for a detailed discussion.

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 22 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 42—*Inspection of containment atmosphere cleanup systems*. The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.	Criterion 42 does not apply to the SHINE IF.
Criterion 43—*Testing of containment atmosphere cleanup systems.* The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.	Criterion 43 does not apply to the SHINE IF.
Criterion 44—*Cooling water.* A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.	Heat is removed from the PCLS and LWPS during normal operation via the RPCS. In addition to removing heat from the subcritical assembly, the RPCS removes heat from various processes throughout the RPF and IF. The RPCS discharges heat removed from the facility to the FCHS, where it is subsequently discharged to the environment. Given the defense-in-depth nature of the cooling system design, multiple pressure boundaries must fail to create a path to the outside environment from systems containing target solution, making fission product release to the outside environment via this pathway incredible. The heat removal provided by the RPCS and FCHS is not safety-related. Upon loss of off-site power, the irradiation process ceases, and decay heat removal is passively and safely handled by the LWPS. No operator action is required to manage decay heat for 90 days after loss of off-site power.
Criterion 45—*Inspection of cooling water system*. The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.	The RPCS and FCHS are not safety-related components. However, they will be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to ensure the integrity and capability of the systems.

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 23 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 46—*Testing of cooling water system*. The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak tight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.	Criterion 46 does not apply to the SHINE IF.
Criterion 50—*Containment design basis*. The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.	<u>As Applied and Means of Compliance</u> The SHINE IF does not contain containment structures. The SHINE IU cell serves as a confinement boundary surrounding the subcritical assembly. During normal operation, the IU cell operates at negative pressures relative to the surrounding IF atmosphere, ensuring in-leakage of radioactive gases that could be present. The confinement boundary contains limited penetrations and those penetrations are sealed to reduce leakage. Penetrations that could present an unacceptable source of leakage during accident conditions (such as line rupture due to the accident condition) contain isolation valves that are controlled by the ESFAS. During accident conditions requiring IU cell isolation, ESFAS initiates closure of the necessary penetrations. Potential radioactive material release rates from accidents and the resulting consequences are evaluated in Chapter 13 and shown to be within 10 CFR 20 limits for off-site exposure. Given the low temperature and low power operation of the SHINE IF systems, very limited energy is available for release during accident conditions. The SHINE IU cell confinement barrier is designed to withstand the conditions generated during accident conditions. See Subsection 6a2.2.1 for a detailed discussion.
Criterion 51—*Fracture prevention of containment pressure boundary.* The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.	<u>As Applied and Means of Compliance</u> The SHINE IF does not have a containment but has a confinement per the NUREG-1537 definition. The pressure differentials across the IU cell walls during operation and accident conditions are minimal, precluding fracture of the confinement boundary.

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 24 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 52—*Capability for containment leakage rate testing.* The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.	As Applied and Means of Compliance The SHINE IF does not contain a containment. However, the IU cell confinement design does contain the capability to test the integrated leakage rate of the confinement boundary for anticipated differential pressures. See Subsection 6a2.2.1 for a detailed discussion.
Criterion 53—*Provisions for containment testing and inspection.* The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.	As Applied and Means of Compliance The SHINE facility does not have a containment but has a confinement per the NUREG-1537 definition. The IU cell confinement is designed to permit: 1) Appropriate periodic inspection of all important areas, such as penetrations; 2) An appropriate surveillance program; and, 3) Periodic testing at confinement design pressure of the leak tightness of penetrations which have resilient seals (e.g., bubble-tight isolation dampers) at the anticipated differential pressures. See Subsection 6a2.2.1 for a detailed discussion.
Criterion 54—*Piping systems penetrating containment.* Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.	As Applied and Means of Compliance The SHINE facility does not have a containment but has a confinement per NUREG 1537 definition. Piping and HVAC systems penetrating the IU cell confinement boundary that have been shown in the accident analyses to have the potential for excessive leakage are provided with isolation capabilities. For example, systems that penetrate the IU cell, are connected directly to the IU cell atmosphere, and are normally maintained open during TSV irradiation, have redundant automatic isolation capabilities. Systems that are connected directly to the PSB also have redundant automatic isolation capabilities. Such systems are designed with a capability to test periodically the operability of the isolation valves and associated apparatus to ensure proper functioning. With the use of double barriers/isolation valves no single active failure can result in lack of isolation. Systems that have no effect on the safety of operation, such as closed cooling water loops, do not require redundant isolation capabilities. A detailed description of the confinement and the penetrations through the confinement are contained in Chapter 6.

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 25 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 55—*Reactor coolant pressure boundary penetrating containment*. Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment. Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety. Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.	<u>As Applied and Means of Compliance</u> The SHINE IF does not contain a containment structure. The fluid lines of the PSB that penetrate the IU cell confinement barrier contain redundant isolation valves. The isolation valves prevent the inadvertent filling of the TSV, inadvertent transfer of solution to the RPF prior to required decay, and isolation of the target solution within IU cell during accident conditions. Due to the low operating pressures and temperatures of the PCLS, accidental rupture of these lines is not considered in design, and isolation valves are not required.

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 26 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 56—*Primary containment isolation.* Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment. Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.	As Applied and Means of Compliance The SHINE IF does not contain a containment structure. The IU cell provides confinement of radioactive material in the event of the release of target solution or gaseous fission products into the IU cell. Each line that connects directly to the IU cell atmosphere and penetrates the IU cell is provided with redundant isolation valves to prevent releases of gaseous or other airborne radioactive material. For all isolation valves that are not automatic, the valves are locked closed whenever the IU contains a target solution batch. Isolation valves outside confinement are located as close to the confinement as practical and upon loss of actuating power, automatic isolation valves are designed to take the position that provides greater safety. All electrical connections from equipment external to the IU cell, as well as electrical power supplies that must enter the IU cell, pass through steel penetration plates located in the IU cell walls. Each penetration plate contains sealed connectors which allow electrical lines to enter or exit the IU cell with minimal air leakage. There are no unsealed through-the-confinement-wall electrical conduits. A detailed description of the IU cell and the penetrations through the confinement structure are contained in Chapter 6.
Criterion 57—*Closed system isolation valves*. Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.	As Applied and Means of Compliance Each line that penetrates IU cell and is neither part of the PSB or PCLS pressure boundary nor connected directly to the confinement atmosphere has no effect on the safety of the IU operation. Therefore, isolation of these lines is not required. Potential sources of excessive leakage from the IU cell confinement boundary during an accident will be identified in the FSAR and appropriate isolation will be provided. Refer to Chapter 6 for a detailed discussion.

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 27 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 60—Control of releases of radioactive materials to the environment. The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.	As Applied The IF design includes means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal operation, including anticipated operational occurrences. Sufficient holdup capacity is provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment. Means of Compliance Gaseous effluents: The IF employs a means to collect gaseous effluents by using a purge from the TOGS at the end of each irradiation cycle. The purge of gaseous effluents is routed to a storage system, the NGRS, which stores gaseous effluents for a suitable period of decay. The collected gases are then subject to controlled, filtered release via the HVAC Zone 1 system to the facility discharge stack. The stack is subject to radiological monitoring via the stack release monitoring (SRM) system. Liquid and solid radioactive wastes are properly characterized and packaged before being transferred off-site through appropriate disposal paths. Refer to Sections 3.5b, 4b, 9b and 11.2 for a detailed discussion.
Criterion 61—*Fuel storage and handling and radioactivity contro*l. 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 In the SHINE IF, target solution is stored within the TS hold tank when not contained within the PSB. Due to the low decay heat load of the target solution, no active heat removal systems are required. Decay heat is managed through small temperature increases within the target solution and natural convection heat removal to the surrounding environment. Substantial radiation shielding surrounds the TS hold tank to ensure adequate shielding for normal and accident conditions. Appropriate periodic inspection capability is included in the design. For discussion of radioactive waste, see Table 3.5b-1.
Criterion 62—*Prevention of criticality in fuel storage and handling*. Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.	As Applied and Means of Compliance The TS dump tank and TS hold tank are designed to be geometrically-safe, with a k_{eff} below 0.95 for the most reactive uranium concentration.

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 28 of 28)

General Design Criteria as Stated	As Applied to SHINE
Criterion 63—*Monitoring fuel and waste storage*. Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.	As Applied and Means of Compliance Due to the low decay heat loads of the target solution, passive decay heat removal is adequate for the target solution in normal and accident conditions. Radiation monitoring systems are present in the IF to detect excessive radiation levels, alert the operators, and initiate appropriate safety actions. For discussion of radioactive waste, see Table 3.5b-1.
Criterion 64—*Monitoring radioactivity releases*. Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.	As Applied and Means of Compliance Means are provided for monitoring the IF confinement atmosphere to detect potential leakage of gaseous or other airborne radioactive material. Potential effluent discharge paths and the plant environs are monitored for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents. Refer to Sections 3.5b, 11.2, and Chapter 6 for a detailed discussion.

3.5b RADIOISOTOPE PRODUCTION FACILITY

This section contains a general evaluation of the design bases of the SHINE facility RPF against the baseline design criteria specified under 10 CFR 70.64. It has also been compared to GDC 61, 62, 63, and 64 of Appendix A to 10 CFR 50 (see Table 3.5b-1).

The criteria are generic in nature and subject to a variety of interpretations. However, they also provide a proven basis from which to provide for and assess the safety of the SHINE facility and develop the principal design criteria.

The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for SSCs important to safety; that is, SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public, operators, and the environment.

For these reasons the conformance of the RPF has been evaluated against the baseline design criteria specified under 10 CFR 70.64 as well as the specific GDC noted above. The results of that evaluation are presented herein. There are some cases where conformance to a particular criterion is not directly measurable. In these cases, the conformance of plant design to the interpretation of the criterion is discussed. For each of the criteria, a specific assessment of the radioisotope production facility design is made and a complete list of references is included to identify where detailed design information pertinent to each criterion is treated.

The radioisotope production facility design basis evaluation against the baseline and GDC is summarized in Table 3.5b-1.

Details of the radioisotope production facility and its various SSCs important to safe operation and shutdown are provided in Sections 4b, 5b, 6b, 7b, 8a2, and 9b, as appropriate.

3.5b.1 RADIOISOTOPE PRODUCTION FACILITY SYSTEMS AND COMPONENTS

This section addresses the design bases for systems and components required for safe operation and shutdown of the radioisotope production facility.

3.5b.1.1 Mechanical Systems and Components

Seismic Category I mechanical equipment and components (as identified in Table 3.5-1) are qualified for operation under the DBEQ seismic conditions by, in order of preference, prototype testing, operating experience or appropriate analysis.

All Seismic Category I mechanical equipment is designed to withstand loadings due to the DBEQ, vibrational loadings transmitted through piping, and operational vibratory loading, such as floor vibration due to other operating equipment, without loss of function or fluid boundary. This analysis considers the natural frequency of the operating equipment, the floor response spectra at the equipment location, and loadings transmitted to the equipment and the equipment anchorage.

These qualification documents and all supporting analysis and test reports are maintained as part of the permanent plant record in accordance with the requirements of the SHINE QAPD.

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 Safety-Related Mechanical Systems and Components

The 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 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.

3.5b.1.5 Process Vessel Vent System

Refer to Subsections 4b.1.3.5 and 9b.6.1 for a detailed description.

3.5b.1.5.1 PVVS Design Basis Functions

a) Maintain hydrogen concentration process tanks and piping below the LFL by diluting the evolved gases.
b) Confine off-gases from process vessels and systems within the RPF and route to RVZ1 for release to a monitored stack.
c) Operates at sub-atmospheric conditions.
d) Treat off-gas to remove excess acids.

3.5b.1.5.2 PVVS Design Basis Values

a) Provide ventilation for process vessels to maintain hydrogen concentration below the LFL.
b) The PVVS has a 30-year design life.
c) The PVVS is welded construction.

3.5b.1.6 Noble Gas Removal System

Refer to Subsections 4b.1.3.4 and 9b.6.2 for a detailed description.

3.5b.1.6.1 NGRS Design Basis Functions

a) Collect and store radiolytic noble gases for radioactive decay.
b) Release decayed off-gas to PVVS for discharge to the environment.
c) Monitor off-gas releases to ensure radioactivity levels are below regulatory limits for discharge to the environment.
d) Remain functional through DBAs.

3.5b.1.6.2 NGRS Design Basis Values

a) The NGRS has a 30-year design life.
b) The NGRS is welded construction.
c) Contain and store noble gases generated in the TSV for at least 40 days.

3.5b.1.7 Criticality Accident and Alarm System

Refer to 7b.6 for a detailed description.

3.5b.1.7.1 CAAS Design Basis Functions

a) Provide local and remote annunciation of a criticality excursion.
b) Remain functional through DBAs.

3.5b.1.7.2 CAAS Design Basis Values

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

b) The CAAS is capable of detecting a criticality accident that produces and absorbed dose in soft tissue of 20 rads of combined neutron or gamma radiation at an unshielded distance of 2 meters from the reacting material within one minute, except for events occurring in areas not normally accessed by personnel and where shielding provides protection against a criticality.

3.5b.1.8 Continuous Air Monitoring System

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

3.5b.1.8.1 CAMS Design Basis Functions

a) CAMS provide real time local and remote annunciation of airborne contamination in excess of preset limits.

3.5b.1.8.2 CAMS Design Basis Values

To be provided in the FSAR.

3.5b.1.9 Confinement Barriers

Confinement is provided by a combination of the hot cell structures, the supporting ventilation systems, and isolation valves or bubble-tight dampers on all hot cell penetrations.

3.5b.1.9.1 RCA Ventilation System Zone 2

3.5b.1.9.1.1 RVZ2 Design Basis Functions

a) Maintain pressure gradients throughout the Zone 2 areas to ensure the proper flow of air from the least potentially contaminated areas to the most potentially contaminated areas, thereby limiting the spread of airborne radioactive materials.
b) Provide confinement of airborne radioactive materials by providing for the rapid, automatic closure of isolation dampers at the at the RCA boundary for various accident conditions.
c) Provide confinement of hazardous chemical fumes.
d) The isolation dampers remain functional for DBA.
e) Provide conditioned air to ensure suitable environmental conditions for personnel and equipment in the RCA.
f) The system has sufficient redundancy to perform its safety function in the event of a single failure.

3.5b.1.9.1.2 RVZ2 Design Basis Values

a) The RVZ2 has a 30-year design life.
b) Maintain air quality with the occupied RVZ2 areas that complies with the dose limits of 10 CFR 20 for normal operations and shutdown.
c) Maintain air quality with the occupied RVZ2 areas that complies with the dose limits of 10 CFR 20 for DBAs.
d) Maintain its leakage rate performance for 40 days post-accident (Subsection 11.1.1.1).

3.5b.1.9.2 Production Facility Biological Shielding

Refer to Section 4b.2 for a detailed description.

3.5b.1.9.2.1 PFBS Design Basis Functions

a) Provide biological shielding from radiation sources in the hot cells for workers in the occupied areas of the facility.
b) Limit physical access to the hot cells.
c) Design to survive DBEQ effects, without loss of structural integrity.
d) Remain functional through DBAs.

3.5b.1.9.2.2 PFBS Design Basis Values

a) The PFBS has a 30-year design life.
b) Provide dose rates at 12 in. (30.48 cm) from surface of shielding of 0.25 mrem/hr or less for normally-occupied areas. Localized dose rates at penetrations and during some planned operations, may be higher and are posted appropriately.

3.5b.1.10 Hot Cell Fire Detection and Suppression System

Refer to Subsection 9a2.3.4.4.2.4 for a description of the HCFD.

3.5b.1.10.1 HCFD Design Basis Functions

a) Provide fire detection in hot cells and enclosures and initiate fire-rated damper closure.

3.5b.1.10.2 HCFD Design Basis Values

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

3.5b.1.11 Radiological Integrated Control System

Refer to Subsection 7b.2.3 for a description of the RICS.

3.5b.1.11.1 RICS Design Basis Functions

a) Monitors valve positions for inter-equipment process fluid transfers.
b) Monitors and controls inter-equipment process fluid transfers in the RPF.
c) Controls the RPF components for transfer of target solution from the TSV dump tank in an IU cell to one of the MEPS.
d) Controls the transfer of prepared target solution from the TSPS in the RPF to the TSV hold tank.
e) Controls the transfer of recycled target solution from the target solution recycle holding tank in the RPF to the TSV hold tank.
f) Initiate ESF actuation of isolation dampers and valves for RPF hot cells, glove boxes or other cells that require isolation upon measured parameters exceeding setpoints.
g) Initiate ESF actuation of isolation dampers for the RCA ventilation system in the RPF, upon measured parameters exceeding setpoints.
h) Monitor the ESF and ensure that ESF actuations go to completion.

i) The portion of RICS that monitors and controls 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.

3.5b.1.14 Uranyl Nitrate Conversion System

3.5b.1.14.1 UNCS Design Basis Functions

a) Convert uranyl nitrate to uranium oxide.
b) Convert uranyl sulfate in the target solution to uranyl nitrate.
c) Separate uranium from fission products and transuranic isotopes.
d) Provide storage capacity for target solution prior to recycle to the TSV.
e) Convert uranyl nitrate to uranium oxide.
f) Prevent inadvertent criticality through inherent design of equipment.

3.5b.1.14.2 UNCS Design Basis Values

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

3.5b.1.15 Radioactive Drain System

3.5b.1.15.1 RDS Design Basis Functions

a) Receive and hold radioactive liquids from the sumps in the processing areas.
b) Detect and alarm when radioactive liquids are received.
c) Perform pH adjustment.
d) Route collected liquids to UNCS.
e) Prevent inadvertent criticality through inherent design of equipment.

3.5b.1.15.2 RDS Design Basis Values

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

3.5b.1.16 Radioactive Liquid Waste Evaporation and Immobilization

3.5b.1.16.1 RLWE Design Basis Functions

a) Reduce the volume of liquid wastes.
b) Convert liquid waste to a form suitable for shipping and disposal as radioactive waste.

3.5b.1.16.2 RLWE Design Basis Values

a) Maintain primary fission product boundary during and after normal operations, shutdown conditions, and DBAs.
b) RLWE has a 30-year design life.

3.5b.1.17 Aqueous Radioactive Liquid Waste Storage

3.5b.1.17.1 RLWS Design Basis Functions

a) Receive and hold radioactive liquid wastes for decay.

3.5b.1.17.2 RLWS Design Basis Values

a) Maintain primary fission product boundary during and after normal operations, shutdown conditions, and DBAs.
b) RLWS has 30-year design life.

3.5b.1.18 RCA Material Handling

Refer to Subsection 9b.7.2 for a detailed discussion.

3.5b.1.18.1 RMHS Design Basis Functions

a) The Overhead Crane OC-0001 provides crane coverage over the IU cells and the TPS.
b) Overhead Crane OC-0002 provides crane coverage over the tank farm and UREX hot cell, SRWP hot cell, RLWE evaporation hot cell, RLWE immobilization hot cell.
c) Overhead Crane OC-0003 provides crane coverage over the receipt area supercells
d) The in-cell overhead hoists are controlled remotely from outside the cells.
e) The hot cells and supercells are fitted with master-slave manipulators on the front of the cell.

3.5b.1.18.2 RMHS Design Basis Values

a) Prevent inadvertent criticality in the uranium metal/trioxide material transfer cart.
b) Overhead Crane OC-0001 has a 75-ton capacity.
c) Overhead Crane OC-0002 has a 40-ton capacity.
d) Overhead Crane OC-0003 has a 40-ton capacity.

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. 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 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 An on-site electric power system (defined to be the SHINE facility provided power supply) and an off-site (defined to be the public utility transmission network supplied substations) electric power system are provided to permit functioning of structures, systems, and components important to safety. The functions for each system (assuming the other system is not functioning) are to provide sufficient capacity and capability to ensure that the RPF processes remain in a safe condition and confinement integrity and other vital functions are maintained in the event of postulated accidents. The on-site electric power supply systems include a safety-related UPSS, a commercial grade diesel generator system, and the on-site electric distribution system. The UPSS has sufficient independence, redundancy, and testability to perform its safety functions assuming a single failure. Each of the redundant UPSS circuits are designed to be available within a few seconds following a loss of all off-site electric power, to ensure that safety systems requiring electrical power (such as RAMS and CAAS) can perform their safety functions. The diesel generator system performs no safety functions but provides an additional degree of defense-in-depth to the electric power system. Portions of the on-site electric distribution, those circuits which operate in conjunction with the UPSS, are safety-related. The SHINE facility receives a single physically off-site independent power circuit. This off-site power circuit consists of two power feeds connected to two local outdoor transformers. Provisions are included to minimize the probability of losing electric power from the UPSS as a result of or coincident with, the loss of power from the transmission network. Refer to Section 6b and Chapter 8 for a detailed discussion.

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

BASELINE DESIGN CRITERIA 10 CFR 70.64	As Applied to SHINE
(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 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.	<u>As Applied and Means of Compliance</u> 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 components and systems. System descriptions identify their associated 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 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. 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.	<u>As Applied and Means of Compliance</u> 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.

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

Appendix A to 10 CFR 50 General Design Criteria as Stated	As Applied to SHINE
Criterion 62—*Prevention of criticality in fuel storage and handling*. Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.	As Applied Criticality in target solution storage and handling systems is prevented by physical separation design and criteria and geometrically safe component and system configurations. Means of Compliance Compliance is achieved through the use of geometrically safe vessels, components, piping, and storage configurations. Refer to Section 6b.3 for details of the Criticality Safety Program.
Criterion 63—*Monitoring fuel and waste storage*. Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.	As Applied Appropriate systems are provided in target solution storage and radioactive waste systems and associated handling areas: 1) To detect conditions that may result in loss of residual heat removal capability and excessive radiation levels. 2) To initiate appropriate safety actions. Means of Compliance Temperature, airborne radiation, background radiation and criticality alarms are monitored continuously and provide real time displays and annunciation locally and remotely in the control room. Refer to Section 4b.4, Chapter 6, and Section 9b for additional information.
Criterion 64—*Monitoring radioactivity releases*. Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.	As Applied and Means of Compliance Means are provided for monitoring the RPF hot cell and glovebox atmospheres and general areas in the RCA to detect potential leakage of gaseous or other airborne radioactive material. Potential effluent discharge paths and the plant environs are monitored for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents. Refer to Sections 3.5b, 11.2, and Chapter 6 for a detailed discussion.

3.6 REFERENCES

ACI, 2007. Code Requirements for Nuclear Safety-Related Concrete Structures & Commentary, ACI 349-06, American Concrete Institute, September 1, 2007.

ANSI/AISC, 2012. Specification for the Design, Fabrication and Erection of Steel Structures for Nuclear Facilities, ANSI/AISC-N690, American National Standards Institute/American Institute of Steel Construction, January 31, 2012.

ANSI/ANS, 2005. Quality Assurance Program Requirements for Research Reactors, ANSI/ANS 15.8-1995 (R2005), American National Standards Institute/American Nuclear Society, 2005.

ASCE, 2005. Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities, ASCE 43-05, January 1, 2005

ASCE, 2006. Minimum Design Loads for Buildings and Other Structures, ASCE/SEI 7-05, American Society of Civil Engineers, 2006.

ASCE, 2009. Seismic Analysis of Safety-Related Nuclear Structures and Commentary, ASCE 4-98, American Society of Civil Engineers, 2009.

DOE, 2006. Accident Analysis for Aircraft Crash into Hazard Facilities, DOE-STD-3014-2006, US Department of Energy, Washington, DC, 2006.

IEEE, 1971a. Criteria for Protection Systems for Nuclear Power Generating Stations, IEEE 279-1971, Institute of Electrical and Electronic Engineers, 1971.

IEEE, 1971b. IEEE Trial-Use Guide for Seismic Qualification of Class I Electrical Equipment for Nuclear Power Generating Stations, IEEE 344-1971, Institute of Electrical and Electronic Engineers, 1971.

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

IEEE, 2009. Standard Criteria for Safety Systems for Nuclear Power Generating Stations, IEEE 603-2009, Institute of Electrical and Electronic Engineers, 2009.

NFPA, 2008. Fire Protection for Facilities Handling Radioactive Materials, NFPA 801, National Fire Protection Association, December 31, 2007.

UCRL, 2007. Hossain, Q.A., R.P. Kennedy, R.C. Murray, K. Mutreja, and B.P. Tripathi, Structures, Systems, and Components Evaluation Technical Support Documents, DOE Standard, Accident Analysis for Aircraft Crash into Hazardous Facilities, UCRL-ID-123577 Lawrence Livermore National Laboratory, Livermore, CA, January 1997.