MOX SERVICES, LLC

Mixed Oxide Fuel Fabrication Facility

License Application
(REDACTED)

January 2018

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LIST OF ACRONYMS AND ABBREVIATIONS

μ	micro
μCi	microCuries
μm	micrometer
°C	degrees Celsius
°F	degrees Fahrenheit
A	ampere
AAS	Active Automatic Sampling
AASHTO	American Association of State Highway and Transportation Officials
ac	acre
AC	alternating current
ACI	American Concrete Institute
ACL	access control list
ADCOH	Appalachian Ultadeep Core Hole
AEGL	Acute Exposure Guideline Level
AFS	Alternate Feedstock
AHJ	Authority Having Jurisdiction
AIHA	American Industrial Hygiene Association
AISC	American Institute of Steel Construction
ALARA	as low as reasonably achievable
ALI	annual limit on intake
ALOHA	Areal Locations of Hazardous Atmospheres
ALS	Active Local Sampling
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOA	area of applicability
AOV	Air Operated Valve
AP	aqueous polishing
APF	additional protective feature
API	American Petroleum Institute
APSF	Actinide Packaging and Storage Facility
ARF	airborne release fraction
ARM	area radiation monitor
ARR	airborne release rate
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
AWS	American Welding Society
BA	Bachelor of Arts degree
BAP	Aqueous Polishing Area
BAQ	Bureau of Air Quality
BET	Bruanuer, Emmet, and Teller
BLEVE	Boiling Liquid Expanding Vapor Explosion

LIST OF ACRONYMS AND ABBREVIATIONS (Continued)

BLWM	Bureau of Land and Waste Management
BMF	MOX Fuel Fabrication Building
BMP	MOX Fuel Fabrication Area (MOX Processing Area)
BN	Belgonucleaire
BPRA	Burnable Poison Rod Assembly
Bq	Becquerel
BR	breathing rate
BS	Bachelor of Science degree
Btu	British thermal unit
BWQ	Bureau of Water Quality
CAAS	criticality accident alarm system
CAB	Controlled Area Boundary
CAM	continuous air monitor
CAR	Construction Authorization Request
CAS	Central Alarm Station
CASRN	Central Abstract System Registry Number
CB&I	Chicago Bridge & Iron Company
cc	cubic centimeter
CCTV	closed circuit television
CDE	committed dose equivalent
CEC	cation exchange capacity
CECP	Construction Emissions Control Plan
CEDE	committed effective dose equivalent
cfm	cubic feet per minute
CFR	Code of Federal Regulations
cfs	cubic feet per second
CGA	Compressed Gas Association
Ci	curies
CIF	Consolidated Incineration Facility
cm	centimeter
CM	configuration management
cm^3	cubic centimeter
CNSI	Chem Nuclear Systems, Incorporated
CNWA	non-borated pourable plastic
COCORP	Consortium for Continental Reflection Profiling
COE	U.S. Army Corps of Engineers
CPS	chemical process safety
CPSG	CB&I Project Services Group, LLC.
CPT	cone penetrometer test
CPU	central processing unit
CRT	cargo restraint transporters
CRT	cargo restraint transporters
CS	conventional seismic
CSAS	Criticality Safety Analysis Sequence

LIST OF ACRONYMS AND ABBREVIATIONS (Continued)

CTF	Chemical Transfer Facility
DAC	derived air concentration
DACR	Digital Alarm Communications Receiver
DACT	Digital Alarm Communications Transmitter
DAS	Diagnostic Aide System
dB	decibel
DBE	design basis earthquake
DBP	dibutyl phosphate
DC	direct current
DCF	dose conversion factor
DCP	Design Change Package
DCS	Duke, Cogema, Stone & Webster
DDDS	Double Door Docking System
DDE	deep dose equivalent
DDT	deflagration to detonation transition
DE	dose equivalent
DE	design earthquake
DEAR	Department of Energy Acquisition Regulations
DER	dose equivalent rate
DETF	Dilute Effluent Treatment Facility
DOD	U.S. Department of Defense
DOE	U.S. Department of Energy
DOE-SR	U.S. Department of Energy Savannah River Operations Office
DOP	dioctyl phthalate
dpm	disintegrations per minute
DPSG	Duke Project Services Group, Inc
DR	damage ratio
DRB	Deep Rock Borings study
DUO_2	depleted uranium oxide
DWPF	Defense Waste Processing Facility
DX	direct expansion
EALF	energy of average lethargy causing fission
EC	effluent concentration
ECC	emergency control console
ECR	Engineering Change Request
EDMS	Electronic Data Management System
EDST	Eastern Daylight Savings Time
EIS	Environmental Impact Statement
EMMH	external man-made hazard
ENMC	epithermal neutron multiplicity counting
EOC	Emergency Operations Center
EPA	U.S. Environmental Protection Agency
EPRI	Electric Power Research Institute
ERDA	Energy Research and Development Administration

LIST OF ACRONYMS AND ABBREVIATIONS (Continued)

ERPG	Emergency Response Planning Guidelines
ES&H	Environment, Safety, and Health
ETF	Effluent Treatment Facility
eV	electron volt
FA	fuel assembly
FA	flame acceleration
FD	Fire Department
FEM	finite element model
FEMA	failure modes and effect analysis
FHA	Fire Hazard Analysis
FIC	final isotopic composition
FM	Factory Mutual
FMEA	Failure Modes and Effects Analysis
FOCI	foreign ownership, control, or influence
fpm	feet per minute
ft	foot/feet
FTS	Fluid Transport system
g	gram
g	acceleration due to gravity
gal	gallon
GSPNA	Gamma Spectroscopy/Passive Neutron Assay
GCC	Graphic Command Center
gpm	gallons per minute
GSA	General Separations Area
GSAR	Generic Safety Analysis Report
GSG	geological, seismological, geotechnical
ha	hectare
HAN	hydroxylamine nitrate
HAW	High Alpha Waste
HAZOP	hazards and operability study
HDE	High Depressurization Exhaust
HDPE	high density polyethylene
HEC-HMS	Hydrologic Engineering Center – Hydrologic Modeling System
HED	human engineering discrepancy
HEPA	high-efficiency particulate air
HFE	human factors engineering
HLW	high-level waste
HPIC	high-performance ion chromatography
HPLC	high performance liquid chromatography
HPT	hydrogenated polypropylene tetramer
hr	hour
HS&E	health, safety, and environment
HSI	human-system interface
HVAC	heating, ventilation, and air conditioning

LIST OF ACRONYMS AND ABBREVIATIONS (Continued)

Hz	hertz
I&C	instrumentation and control
I/O	input/output
IBC	International Building Code
ICBO	International Conference of Building Officials
ICN	Immediate Control Network
ICP-MS	inductive coupled plasma – mass spectroscopy
ID	identification
IDLH	Immediately Dangerous to Life and Health
IEEE	Institute of Electrical and Electronic Engineers
ILS	Inactive Local Sampling
in	inch
INES	International Nuclear Event Scale
IOC	individual outside of the controlled area
IROFS	items relied on for safety
ISA	integrated safety analysis
IT/SF	Interim Treatment/Storage Facility
ITP	In-Tank Precipitation Facility
ka	kilo annum or thousands of years
keV	kilo electron volt
kg	kilogram
kg/hour	kilogram per hour *
kip	kilopound
km	kilometer
kV	kilovolt
kW	kilowatt
L	liter
LA	License Application
lb	pound
lb/hour	pound per hour
LDE	lens of the eye dose equivalent
LETF	Liquid Effluent Treatment Facility
LFL	lower flammable limit
LIMS	Laboratory Information Management System
LIN	Local Industrial Network
LLC	Limited Liability Company
LLD	lower limit of detection
LLNL	Lawrence Livermore National Laboratory
LLW	low-level waste
LOC	level of severity or concern
LPF	leak path factor
LRD	Sample Receipt, Weighing and Dispatching Laboratory
LWR	light water reactor

LIST OF ACRONYMS AND ABBREVIATIONS (Continued)

m	meter
M	molar
M&O	Maintenance and Operations
m^3	cubic meter
Ma	mega annum or millions of years
MACCS2	MELCOR Accident Consequence Code System for the Calculation of the Health and Economic Consequences of Accidental Atmospheric Radiological Releases
MAR	material at risk
mb	body wave magnitude
mbar	millibar
MBP	monobutyl phosphate
MC&A	Material Control and Accountability
MCA	Multi-Channel Analyzer
MCC	motor control center
MCNP	Monte Carlo N-Particle
MD	duration magnitude
MDE	Medium Depressurization Exhaust
meq	milliequivalent
MeV	megavolt
MFFF	Mixed Oxide Fuel Fabrication Facility
MFFP	MOX Fresh Fuel Package
mg	milligram
mgd	million gallons per day
mi	mile
MIMAS	micronized master blend
min	minute
MJ	megajoule
ml	milliliter
mm	millimeter
MMI	Modified Mercalli
Mo	Molybdenum
MOI	maximally exposed offsite individual
MOX	mixed oxide
MP	MOX processing
mph	miles per hour
MPQAP	MOX Project Quality Assurance Plan
MPSSZ	Middleton Place-Summerville Seismic Zone
mrem	millirem
MSA	Metropolitan Statistical Area
MSDS	Material Safety Data Sheet
msl	mean sea level
MtHM	metric tons of heavy metal
MVA	megavolt-ampere

LIST OF ACRONYMS AND ABBREVIATIONS (Continued)

MW	megawatt
Mw	moment magnitude
N	normal (unit of chemical concentration)
NAC/AEGL	National Advisory Committee for Acute Exposure Guidelines
nCi	nanocurie
NCS	nuclear criticality safety
NCSE	nuclear criticality safety evaluation
NDA	nondestructive assay
NDE	nondestructive examination
NDU	Network Display Units
NEHRP	National Earthquake Hazards Reduction Program
NESHAP	National Emission Standard for Hazardous Air Pollutants
NFPA	National Fire Protection Association
ng	nanogram
NIM	nuclear incident monitoring
NIST	National Institute of Standards and Technology
NNSA	National Nuclear Security Administration
NOAA	National Oceanic and Atmospheric Administration
NOI	Notice of Intent
NO_X	nitrous fumes
NPDES	National Pollutant Discharge Elimination System
NPH	natural phenomena hazard
NPLC	normal programmable logic controller
NPU	Network Processing Units
NRC	U.S. Nuclear Regulatory Commission
NSE	Nuclear Safety Evaluation
O/M	oxygen-to-metal
OML	oxalic mother liquors
OSC	Operations Support Center
OSHA	Occupational Safety and Health Administration
Pa	Pascal
PA	paging/public address speaker
PA	Protected Area
PC	performance category
pCi	picocurie
pCi/g	picoCuries/gram
PCM	personnel contamination monitor
PCV	primary containment vessel
PDC	Pit Disassembly and Conversion
PEL	permissible exposure level
PFHA	Preliminary Fire Hazard Analysis
PGA	peak ground acceleration
PHA	Preliminary Hazards Analyses
PIDAS	Perimeter Intrusion Detection and Assessment System

LIST OF ACRONYMS AND ABBREVIATIONS (Continued)

PIP	Plutonium Immobilization Plant
PLC	programmable logic controller
PMF	probable maximum flood
PMI	Positive Material Identification
PMP	probable maximum precipitation
POE	Process Cell Exhaust System
ppb	parts per billion
PPB	porated polyethylene plaster
ppm	parts per million
PrHA	Process Hazards Analyses
psf	pounds per square foot
PSHA	Probabilistic Seismic Hazard Assessment
psi	pounds per square inch
psia	pounds per square inch, absolute
psig	pounds per square inch, gage
PSSC	principal systems, structures, and components
PSUP	Power Services Utilization Permit
Pu	plutonium
PUCR	Polishing and Utilities Control Room
PuO_2	plutonium oxide
PWR	pressurized water reactor
QA	quality assurance
QL	quality level
RAB	restricted area boundary
rad	radiation absorbed dose
RAIC	raffinates isotopic composition
RBOF	Receiving Basin for Offsite Fuels
RCRA	Resource Conservation and Recovery Act
RCZ	radiological control zone
rem	roentgen equivalent, man
RF	respirable fraction
RHC	high radiation controller
RIC	radiological isotopic composition
RM/RPR	Respiratory Protection and Radiological Protection Room
ROD	Record of Decision
RP	Radiological Protection
RPCA	Radiological Protection Control Area
RPM	Radiological Protection Functional Manager
rpm	revolutions per minute
RT	radiation detection devices
RTF	Replacement Tritium Facility
RVT	Random Vibration Theory
RWP	Radiation Work Permit
S&W	Stone & Webster, Inc.

LIST OF ACRONYMS AND ABBREVIATIONS (Continued)

SA	safety assessment
SAF	soil amplification function
SAM	seismic anchor movements
SAR	Safety Analysis Report
SC	seismic category
SCADA	System Control and Data Acquisition
SCAPA	DOE Subcommittee on Consequence Assessment and Protective Actions
SCB	Structural Consulting Board
SCDHEC	South Carolina Department of Health and Environmental Control
SCDNR	South Carolina Department of Natural Resources
SCE&G	South Carolina Electric and Gas Company
SCPTU	site-specific seismic piezocone penetration test soundings
SCR	South Carolina Route
SCV	secondary containment vessel
SCV	Soil Conservation Service
SDE	shallow dose equivalent
sec	second
SEUS	Southeastern United States
SEUSSN	Southeast U.S. Seismic Network
SGS	Site Geotechnical Services
SHPO	State Historic Preservation Officer
SIL	seismically induced liquefaction
SMA	strong motion accelerograph
SNM	special nuclear material
SOV	solenoid valves
SPCC	Spill Prevention Control and Countermeasures
SPLC	safety programmable logic controllers
SPS	Steam and Condensate system
SR	Shipping and Receiving
SREL	Savannah River Ecology Laboratory
SRFS	Savannah River Forest Station
SRP	Standard Review Plan
SRS	Savannah River Site
SRSS	square root of the sum of the squares
SRTC	Savannah River Technology Center
SS	stainless steel
SSCs	structures, systems, or components
SSI	soil-structure interaction
SSNM	strategic special nuclear material
SST	safe secure transport
ST	source term
STC	spinning tube conveyor
STE	secure telephone equipment
STEL	short-term exposure level

SUW	Stripped Uranium Waste
Sv	Sievert
SWDF	Solid Waste Disposal Facility
SWMF	Solid Waste Management Facility
SWPPP	Stormwater Pollution Prevention Plan
T	trace
TBD	to be determined
TBP	tributyl phosphate
TC	toxic concentration
TCP/IP	Ethernet communication protocol
TD	toxic dose
TEDE	total effective dose equivalent
TEEL	Temporary Emergency Exposure Limit
TIC	Today's Isotopic Composition
TID	tamper indicator
TIG	Tungsten Inert Gas
TIS	temperature indicating switch
TLD	Thermoluminescent Dosimeter
TLV	threshold limit value
TPH	hydrogenated tetrapropylene
TRU	transuranic
TWA	time-weighted average
UBC	Uniform Building Code
UCNI	Unclassified Controlled Nuclear Information
UCT	Universal Coordinated Time
UDP/IP	Universal Datagram Protocol/Internet Protocol
UFL	upper flammable limit
UHE	electric heater unit
UHS	Uniform Hazard Spectrum
UIC	Underground Injection Control
UL	Underwriters Laboratory
UO_2	uranium oxide
UPS	uninterruptible power supply
USDA	U.S. Department of Agriculture
USFS	United States Forest Service
USFWS	U.S. Fish and Wildlife Services
USGS	U.S. Geological Survey
USL	upper safety limit
USL	upper subcritical limit
USNRCS	U.S. Natural Resources Conservation Service
UST	underground storage tank
V	volt
V&V	verification and validation
VAC	volts alternating current

LIST OF ACRONYMS AND ABBREVIATIONS (Continued)

VDC	volts direct current
VEGP	Vogtle Electric Generating Plant
VFD	variable frequency drive
VHD	Very High Depressurization Exhaust System
vol %	volume percent
WAC	Waste Acceptance Criteria
WSI	Wackenhut Services Inc.
WSPRO	Water Surface Profile Computations
WSRC	Westinghouse Savannah River Company, LLC
wt %	weight percent
WTA	Work Task Agreement
WVA	Vehicle Access Portal
XTN	X-Terminal network
yd	yard
yr	year

LIST OF MFFF BUILDING AND SYSTEM DESIGNATIONS

ANN	Annunciator System
BAA	Structural Steel and Platforms
BAD	Administration Building
BAP	Aqueous Polishing Area
BAR	Plant Building Siding
BAS	Breathing Air
BBD	Barrier System Doors
BBJ	Civil Engineering – General Information
BBP	Barrier System, Personnel Access
BBS	Barrier System, Safe Haven
BBT	Barrier System, Truck Bay
BEG	Emergency Generator Building
BKA	Foundations and Concrete
BLA	Auxiliary concrete
BMF	MOX Fuel Fabrication Building

LIST OF MFFF BUILDING AND SYSTEM DESIGNATIONS (Continued)

BMP	MOX Processing Area
BRA	Penetrations and Seals
BRP	Reagents Processing Building
BRW	Receiving Warehouse Building
BSA	Finishing
BSH	Safe Haven Building
BSJ	Building Services, General
BSR	Shipping and Receiving Area
BSW	Secured Warehouse Building
BTP	Building Doors and Trap Doors
BTS	Technical Support Building
CHH	HVAC Chilled Water
CHP	Process Chilled Water
COL	Communication - Telephone (Leased Line)
COP	Communication - Paging (Public Address)
COR	Communication - Radio
COS	Communication - Sound Power
COT	Communication - Telephone (PBX)
DCE	PuO_2 Buffer Storage
DCM	PuO_2 3013 Storage
DCP	PuO_2 Receiving
DCS	Liquid Decontamination
DDP	UO_2 Drum Emptying
DMW	Demineralized Water
DWS	Domestic Water, Potable
EAA	13.8 KV Power
EAB	Normal 4.16 KV Power
EAC	Emergency 4.16 KV Power
EAS	480 V Power – Security System
EBA	125 VDC Battery – Normal
EBB	125 VDC Battery – Emergency
ECB	480 VAC Power – Normal
ECC	480 VAC Power – Emergency
EDB	Electrical Duct Bank
EDG	Emergency Diesel Generator
EEA	208/120 VAC – Essential
EEC	208/120 VAC – Vital
EED	480/277 V Egress Lighting
EEJ	Electrical – General
EGF	Fuel Oil Emergency DG
ELE	Lighting – Emergency
ELJ	Lighting – General
ELY	Lighting – Yard
EPA	480/277 V Lighting Distribution – Normal

LIST OF MFFF BUILDING AND SYSTEM DESIGNATIONS (Continued)

EPC	480/277 V Lighting Distribution – Emergency
EPG	240/120 VAC Power Distribution – Normal
EPH	240/120 VAC Power Distribution – Emergency
ESE	120 V UPS – Security System
ESL	Lighting – Security System
ESP	240/120 VAC Power Distribution – Security
FPA	Fire Detection – Alarm System
FPD	Fire Detection – Detection System
FPE	Fire Protection – Portable Extinguishers
FPG	Fire Protection – Clean Agent
FPM	Fire Protection – Supervision System
FPW	Fire Protection – Water
GAH	Argon/Hydrogen System – Gaseous
GDE	Rod Decladding
GHE	Helium System
GIS	Isolated Ground
GMA	P10 Gas (Methane/Argon)
GME	Rod Cladding And Decontamination
GMK	Rod Tray Loading
GNO	Nitrogen Oxide
GNS	Nitrogen System – Gaseous
GOX	Oxygen System
GPD	Small Rod Components Cleaning (In Warehouse)
GRS	Grounding System
HDE	High Depressurization Exhaust
HSA	HVAC, Supply Air
HSC	HVAC, Entry Control
HSH	HVAC, Safe Haven
HTS	Heat Tracing
HVC	HVAC, Emergency Control Room
HVD	Emergency Diesel Generator Building HVAC
HVR	HVAC, Reagents Building
HVT	HVAC, Truck Bay
HVV	HVAC, Shipping & Receiving Building
HWS	Process Hot Water
IAS	Instrument Air
KCA	Precipitation – Filtration – Oxidation
KCB	Homogenization – Sampling
KCC	PuO_2 Canning
KCD	Oxalic Mother Liquors Recovery
KDA	PuO_2 Decanning
KDB	Dissolution
KDD	Dissolution of Chlorinated Feed
KDM	Pre-Polishing Milling

LIST OF MFFF BUILDING AND SYSTEM DESIGNATIONS (Continued)

KDR	Recanning
KLA	High Pu Content Solution Analysis and Preparation (LAB)
KLB	Low Pu Content Solution Analysis and Preparation
KLC	Alpha and Gamma Spectrometry Preparation
KLD	Mass Spectrometry Preparation
KLE	PuO_2 Dissolution
KLF	Mass Spectrometry Preparation (AP Powder Samples)
KLH	Sampling Split
KLI	Dissolution of Impure PuO_2
KLJ	Metallic Impurities Determination
KLK	Chlorine and Fluorine Impurities Determination
KLL	Nitrogen Sulfur and Carbon Impurities Determination
KLN	Particle Size Determination (PDC-type/ARIES Powder)
KPA	Purification Cycle
KPB	Solvent Recovery
KPC	Acid Recovery
KPG	Sampling, Automatic
KWD	Aqueous Waste Reception
KWG	Process Offgas Treatment
KWS	Solvent Waste Reception
LAC	Fluorine, Chlorine, Oxygen/Heavy Metal Ratio Determination, Insolubility Test Preparation
LAU	Autoclaves (LAB)
LBT	Specific Surface Analysis and Grain Size Determination
LCP	ICP-MS Spectrometer
LCT	Test Line
LDS	MOX Dissolution
LET	Calibration
LFT	Thermal Stability Analysis
LFX	Gamma X-Ray Fluorescence
LGF	Liquid Waste Processing
LLI	Reagent Preparation
LLJ	Laboratory, General
LLP	Laboratory Pneumatic Transfer System (33 mm)
LME	Ceramographic and Metallographic Tests
LPG	Gas Analysis
LPO	Photo Laboratory
LPS	Gamma and Alpha Spectrometer
LRD	Sample Receipt, Weighing and Dispatching
LSG	Gas Storage
LSP	Chemicals Transfer Trolley
LSR	Mass Spectrometer
LTP	Sample Pneumatic Transfer System (76 mm)

LIST OF MFFF BUILDING AND SYSTEM DESIGNATIONS (Continued)

MDE	Medium Depressurization Exhaust & Tertiary Confinement
MMIS	Manufacturing Management Information System
NBX	Primary Blend Ball Milling
NBY	Scrap Ball Milling
NCP	Powder Containers
NCR	Scrap Processing
NCS	Normal Control System
NDD	PuO_2 Can Receiving and Emptying
NDP	Primary Dosing
NDS	Final Dosing
NIM	Nuclear Incident Monitoring
NPG	one of the two identical Homogenization and Pelletizing
NPH	one of the two identical Homogenization and Pelletizing
NPP	Additives Preparation
NTM	Jar Storage and Handling
NTP	Can Pneumatic Transfer System (133 mm)
NXR	Powder Auxiliary
PAD	Pellet Repackaging
PAR	Scrap Box Loading
PBS	Sanitary Sewerage Facilities - System
PCT	Pellet Containers
PFE	Sintering Furnace
PFF	Sintering Furnace
PML	Pellet Handling
POE	Process Cell Exhaust System (High Depressurization Exhaust & Secondary Confinement, Polishing Enclosure Areas)
PQE	Quality Control and Manual Sorting
PRE	Grinding
PRF	Grinding
PSE	Green Pellet Storage
PSF	Sintered Pellet Storage
PSI	Scrap Pellet Storage
PSJ	Ground and Sorted Pellet Storage
PTE	Pellet Inspection and Sorting
PTF	Pellet Inspection and Sorting
PWS	Plant Water
RAN	Aluminum Nitrate System
RDO	Dodecane
RHN	Hydroxylamine Nitrate
RHP	Hydrogen Peroxide
RMN	Manganese Nitrate
RMS	Radiation Monitoring System
RNA	Nitric Acid
ROA	Oxalic Acid

LIST OF MFFF BUILDING AND SYSTEM DESIGNATIONS (Continued)

RSC	Sodium Carbonate
RSH	Sodium Hydroxide
RSI	Sodium Nitrite
RSN	Silver Nitrate
RSS	Sodium Sulfite
RTP	Tributyl Phosphate
RUN	Reagent Uranyl Nitrate
RZN	Zirconium Nitrate Solution
SAS	Service Air (System Designation)
SAS	Secondary Alarm Station (Area Designation)
SCE	Rod Scanning
SDK	Rod Inspection and Sorting
SEK	Helium Leak Test
SMK	Rod Tray Handling
SMS	Stack Monitoring System
SMT	Seismic Monitoring and Trip System
SPC	Process Condensate
SPS	Process Steam and Condensate
SSG	Security – General
STK	Rod Storage
SXE	X Ray Inspection
TAS	Assembly Handling and Storage
TCK	Assembly Dry Cleaning
TCL	Assembly Final Inspection
TCP	Assembly Dimensional Inspection
TGJ	Reserve Pit
TGM	Fuel Assembly Rod Loading
TGV	Fuel Assembly Fabrication
TXE	Assembly Packaging
UEF	Emergency Fuel Storage Vault
UGS	Gas Storage Facility
VCX	HVAC Chiller Pad
VCY	Process Chiller Pad
VDQ	Waste Storage
VDT	Waste Nuclear Counting
VHD	Very High Depressurization Exhaust System
VRM	Radiation Monitoring System Vacuum
WAV	Roadways and Paving
WEP	Stormwater Drainage
WRS	Soils, Geotechnics
WRT	Excavation
WSB	Waste Solidification Building
WVA	Vehicle Access Portal
WWJ	General Information

LIST OF MFFF BUILDING AND SYSTEM DESIGNATIONS (Continued)

XGA	Building
XGP	Site Plan
XXJ	General Information, Buildings (BFF)
XXP	Temporary Construction
XXR	General Information, Buildings (BAP)
ZMS	General Mechanical Supports and Stress

1.0 GENERAL INFORMATION

1.1 FACILITY AND PROCESS OVERVIEW

1.1.1 Introduction

The consortium of CB&I Project Services Group, LLC. and AREVA, Inc., has formed a Limited Liability Company called CB&I AREVA MOX Services, LLC (MOX Services). MOX Services seeks authorization to possess and use by-product material, source material, and special nuclear material at the Mixed Oxide (MOX) Fuel Fabrication Facility (MFFF), which is owned by the U.S. Department of Energy (DOE), located on DOE's Savannah River Site (SRS) near Aiken, South Carolina. The MFFF is designed to convert surplus weapons-grade plutonium to MOX fuel that can be used to generate electricity at commercial nuclear power stations. The fabrication of the MOX fuel, which is a blend of uranium and plutonium oxides, is based on the proven European technology of AREVA NC.

This license application is written in the present tense. It describes the MFFF site, design features, processes, programs, commitments, etc., in effect in the time perspective of receipt of the U.S. Nuclear Regulatory Commission (NRC) approved license for possession and use of nuclear materials for operation of the facility.

1.1.2 General Facility Description

The MFFF is located in F Area of SRS as indicated in Figure 1.1.2-1. The arrangement of the buildings and facilities of the MFFF is shown in Figure 1.1.2-2.

1.1.2.1 Structural Systems

This section describes the facility civil/structural design. In addition to the structural systems design description and requirements, this section presents the seismic qualification of systems, structures, and components as well as the qualification of process equipment structural components.

This section presents the design of buildings, structures and facilities at the MFFF site. The design of structures, systems and components (SSCs) accommodates structural loading conditions and meets performance criteria for Items Relied on for Safety (IROFS) and for non-IROFS. Design features and administrative controls for IROFS satisfy the performance requirements of Title 10 of the Code of Federal Regulations (CFR) Part 70, "Domestic Licensing of Special Nuclear Material."

1.1.2.1.1 Function

Civil structural systems provide the following functions:

- Support IROFS and non-IROFS SSCs during normal, severe, and extreme loading conditions
- Provide confinement functions as part of secondary and tertiary confinement systems
- Protect IROFS from the effects of normal, severe, and extreme environmental loads

- Protect IROFS from the effects of design basis internal and external fires by providing fire barriers.

1.1.2.1.2 Description

The civil structural systems for the MFFF include the buildings, support structures, and facilities that house, support, protect, confine, or contain various plant systems, components, and equipment associated with licensed nuclear materials, or hazardous chemicals associated with the licensed nuclear materials.

See Figure 1.1.2-3 for the arrangement of buildings and structures of the MFFF. The buildings and structures provide for safe, secure, and efficient performance of MFFF functions. The site layout and facility features satisfy security criteria for safeguarding the special nuclear materials utilized at the MFFF.

The MFFF site comprises an area of approximately 41 acres. Approximately 17 acres of the site are developed with buildings, facilities, or paving. The remaining 24 acres are grass, gravel, or natural terrain. No public highways, railroads, or waterways traverse the MFFF site. The movement of material and personnel to and from the MFFF site takes place via the internal road system of the SRS. The public transportation right-of-way nearest to the MFFF site and F Area is South Carolina Route 125 to the west. Access to the MFFF site is via SRS Roads C and C-3.

A double perimeter intrusion detection and surveillance fence surrounds the protected area (PA) of the MFFF. The 14-acre PA is roughly square in shape. The MFFF Administration building and the Gas Storage facility are located outside the PA. The Receiving Warehouse building, which is located near the site access road, is part of the inner Perimeter Intrusion Detection and Assessment System (PIDAS) security barrier. Other buildings and facilities of the MFFF lie within the PA.

There are three categories of structural design requirements. The categories, loadings, and structures are defined in subsequent sections and are summarized as follows:

- **Seismic Category I (SC-I)** – Required to perform design functions with normal, severe, and extreme environmental loads, including the design earthquake (DE) and tornado, which are applied to IROFS SSCs.
- **Seismic Category II (SC-II)** – Qualified for normal, severe, and extreme loadings, with extreme loads limited to the DE, to preclude adverse interactions with required safety equipment. SC-II applies to SSCs whose failure could adversely impact IROFS SSCs (i.e., secondary seismic interaction).
- **Conventional Seismic (CS)** – SSC qualified for normal, severe, and extreme loads with extreme loads limited to the conventional seismic loads as specified by the Uniform Building Code (UBC) or International Building Code (IBC). CS addresses worker safety concerns, good engineering practice, and protection of capital investment.

See Table 1.1.2-1 for identification of the building structures located at the MFFF site and definitions of the seismic category classification of each. The following section describes the MFFF buildings and structures.

1.1.2.1.3 Major Components

1.1.2.1.3.1 MOX Fuel Fabrication Building

The MOX Fuel Fabrication Building (BMF) consists of the MOX processing area (BMP), the Aqueous Polishing area (BAP), and the Shipping and Receiving area (BSR). See Figure 1.1.2-4 through Figure 1.1.2-6 and Figure 1.1.2-21 through Figure 1.1.2-25 for general arrangement drawings of the BMF.

The BMF is a multi-story, hardened, reinforced concrete structure. The building meets the applicable requirements for processing special nuclear material. The structure is designed to withstand natural phenomena including floods, winds, tornadoes and the DE, as well as potential industrial accidents (e.g., load drops, fire) that could impact the fissile materials. The BMF includes features such as a heating, ventilation, and air conditioning (HVAC) exhaust vent stack and stair towers on the roof.

The BMF consists of reinforced concrete shear walls, floors, and roof slab. Interior partitions are constructed of reinforced concrete. The roof structure is a flat reinforced concrete slab (details discussed below) with a membrane top. Personnel doors are hollow metal in metal frames. There are a number of special doors and barriers for security or function as well as a number of removable wall panels for removal and/or replacement of processing equipment.

The base mat of the entire BMF is 6.5-foot (ft) thick reinforced concrete. The exterior walls are reinforced concrete with an additional outer reinforced concrete security wall. This additional wall, which is part of the outer security barrier, is 3 ft away from the exterior wall of the BMF and is attached to the exterior wall with tie-back beams. The space between the two walls is filled with gabion stone.

The HVAC main intake penetrations are located on the third level of the BMP. Four penetrations are located on the outside security wall which extends away from the BMP interior wall. To prevent access through penetrations, a security barrier is installed in series at each opening that is sized to prevent unauthorized access into the building. The four penetrations are offset to prevent tornado generated missiles from entering the building. The structural roof of the BMF is a reinforced concrete slab.

1.1.2.1.3.2 Emergency Generator Building

The Emergency Generator Building (BEG) is a single-story slab-on-grade reinforced concrete building. The BEG has two independent rooms that contain the diesel generators and another two independent rooms that contain the switchgear equipment. HVAC equipment for the BEG is located on an elevated structural steel framed platform located above the switchgear rooms. Air intake and exhaust vents for the diesel generators are configured to preclude entry of the design basis tornado missile. See Figure 1.1.2-39A for the general arrangement drawing of the BEG.

The BEG is evaluated for natural phenomena hazards, which includes floods, tornadoes, and the DE. The building contains the emergency generators and switchgear that provides power to the IROFS.

1.1.2.1.3.3 Emergency Fuel Storage Vault

The Emergency Fuel Storage Vault (UEF) is a single-story in-ground buried reinforced concrete building. The design of the walls, floor, and roof are of sufficient strength and thickness to protect against the effects of wind, design basis tornado, and associated generated missiles. The design also resists the DE.

The UEF provides support and protection for the two fuel oil storage tanks and associated equipment. A concrete wall provides separation between the fuel oil tanks. It is evaluated for natural phenomena hazards, which includes floods, wind, tornadoes, and the DE.

1.1.2.1.3.4 Safe Haven Buildings

The Safe Haven Buildings (BSH) are single-story, hardened, reinforced concrete structures. Each building is used for the emergency assembly of personnel during exit from the BMF. Each BSH contains its own HVAC and lighting system. The structure is classified as SC-II and is evaluated for natural phenomena, which includes floods, wind and the DE, as well as potential industrial accidents (e.g., fires). They are also designed to provide personnel protection against tornado and tornado missiles.

The BSHs are located at grade and at each emergency exit from the BMF. The safe havens are single-story buildings. See Figure 1.1.2-4 and Figure 1.1.2-22 for general arrangement drawings of the BSHs.

1.1.2.1.3.5 Reagent Processing Building

The Reagent Processing Building (BRP) is a single-story with a partial basement. The floor is reinforced concrete slab-on-grade with turn down edges along the perimeter and spread footing for the building foundation, which is designed for the building loads and soil conditions. The exterior walls are constructed of reinforced concrete. Interior walls are constructed of reinforced concrete or reinforced masonry. The roof is constructed of steel frame, metal decking and concrete.

The BRP provides storage and mixing facilities for the bulk chemical reagents used in the Aqueous Polishing (AP) process. Chemical reagents are transferred through piping in the BRP to piping in the BAP via an underground tunnel located between the buildings. The building is categorized as CS and is evaluated for normal, severe, and extreme loads. The extreme loads are limited to the conventional seismic loads as specified by the UBC or IBC.

1.1.2.1.3.6 Administration Building

The Administration Building (BAD) is located outside of the protected area of the MFFF complex. The BAD is accessed from the main facility personnel and public parking area. It is a three-story, steel-framed structure. The first-story is slab-on-grade, and the second and third stories are concrete on metal decking and bar joist framing. The exterior walls consist of modular metal panels with an integrated glazing system.

Functions included in the BAD are facility management, facility operations, material accountability administration, finance, administration, health and safety, quality assurance, and personnel management. The BAD contains offices, conference rooms, a lunchroom, computer simulation laboratory, document control and record storage, and rest room facilities. It is categorized as CS and is evaluated for normal, severe, and extreme loads. The extreme loads are limited to the conventional seismic loads as specified by the UBC or IBC.

1.1.2.1.3.7 Secured Warehouse Building

The Secured Warehouse Building (BSW) is a single-story, slab-on-grade foundation, steel framed building with insulated metal roofing and siding. The BSW receives and stores the non-nuclear materials, supplies, and equipment received in the PA that is stored onsite for future use. The BSW also provides space for storage of depleted uranium and empty MOX fresh fuel packages. In addition, a small parts washing area is provided for tools and components.

The building is categorized as CS and is evaluated for normal, severe, and extreme loads, with extreme loads limited to the conventional seismic loads as specified by the UBC or IBC.

1.1.2.1.3.8 Technical Support Building

The Technical Support Building (BTS) is a two-story, steel-framed building supported on spread footings. The building covers an area of approximately 49,300 ft^2 (4,580 m^2). The first floor is supported by a slab-on-grade. The BTS is located between the BAD and the BMF. The front wall of the BTS is integrated with the inner PIDAS fence/barrier. The security turnstiles and associated barriers located at the Protected Area (PA) Personnel Access Portal delineate the PA boundary. Persons passing through the turnstiles will have met the screening and identification/badging requirements.

The BTS contains offices for various departments such as Security, Material Control & Accounting, Operations, and Maintenance and Quality Control. The BTS provides the main support facilities for BMF personnel. It serves as the sole personnel access into and out of the PA access (except for vehicle drivers escorted in and out of the Vehicle Access Portal). Such activities as photo identification, search, and pass-through take place in the personnel access portal in the access control area. The BTS is not directly involved in the principal processing functions of the MFFF. Supporting activities and facilities located in this building include health physics facilities, an electronics maintenance laboratory, a mechanical maintenance shop, personnel locker rooms, and a first aid station.

1.1.2.1.3.9 Receiving Warehouse Building

The Receiving Warehouse Building (BRW) is a single-story, slab-on-grade, metal building with insulated metal roofing and siding. The building contains areas for receipt, unpacking, and inspection of material, supplies, and equipment prior to transfer through the PIDAS into the protected area or the BAD. The Vehicle Access Portal (WVA) is adjacent to the BRW and is the access and inspection area for vehicles passing through the PIDAS and entering the PA.

The building is categorized as CS and is evaluated for normal, severe, and extreme loads. The extreme loads are limited to the conventional seismic loads as specified by the UBC or IBC.

1.1.2.1.3.10 Miscellaneous Site Structures

The miscellaneous site structures include a bulk gas storage pad and process chiller water pad, diesel fuel filling station, electric transformer pads, and other minor structures.

The miscellaneous site structures are categorized as CS and are evaluated for normal, severe, and extreme loads, with extreme loads limited to the conventional seismic loads as specified by the UBC or IBC.

1.1.2.1.4 Control Concepts

This section is not applicable to buildings and structures.

1.1.2.1.5 System Interfaces

Civil structural systems interface with all site and plant systems to provide protection and support for IROFS.

1.1.2.1.6 Design Basis for IROFS

1.1.2.1.6.1 Functions of SC-I Structures

SC-I buildings and structures provide the following safety functions for IROFS:

- Support and protect IROFS from the effects of normal, severe, and extreme environmental loads
- Provide confinement functions as part of secondary and tertiary confinement systems
- Provide support from the effects of temperature extremes, including design basis internal and external fires
- Provide support from the effects of design basis man-induced events, including potential load drops.

1.1.2.1.6.2 Requirements for SC-I Structures

1.1.2.1.6.2.1 General Structural Analysis Requirements

SC-I structures are designed for the loads and loading combinations specified in Section 1.1.2.1.6.4. Appropriate consideration is given to the load distribution on the structure (e.g., point loads, uniformly distributed loads, or varying distribution of loads) and the end restraint conditions applicable for the structural component being considered.

Analyses are performed using equivalent static loads with appropriate consideration of impact effects for moving loads as specified for the particular loads (see Section 1.1.2.1.6.4). See Section 1.1.2.1.6.4.1.3 (under "Seismic Loads" and "Tornado Loads for SC-I Structures") for an outline of the special provisions that are used for performing analyses for seismic loads and tornado missile impact loads, respectively.

1.1.2.1.6.2.1.1 Seismic Analysis Requirements for SC-I Structures

The free-field DE acceleration is applied at grade elevation (see Table 1.1.2-2 and Section 1.1.2.1.6.4.1.3 under "Seismic Loads").

Analysis methods for converting the design earthquake acceleration into seismic loads on SC-I structures are as defined in Section 1.1.2.1.6.4.1.3 (under "Seismic Loads"). Seismic loads are applied simultaneously to structures in the three orthogonal directions, and the three-dimensional effects of each of these inputs are considered. Seismic load forces and moments may be combined using the "square root of the sum of the squares" (SRSS) method or the 100-40-40 Percent Rule, as described in American Society of Civil Engineers (ASCE) Standard 4, to determine the resultant design earthquake loads on structural components.

When designing structures for seismic loads, consideration is given to the additional seismic loads resulting from accidental torsion, indicated in U.S. Nuclear Regulatory Commission (NUREG)-0800, Section 3.7.2, Subsection II.11. This additional seismic loading accounts for variations in material densities, member sizes, architectural features, equipment loads, etc. At each level under consideration (floor levels or roof), the accidental torsion is equal to the applicable lateral seismic inertia force times 5 percent (%) of the maximum building dimension at the level being considered.

1.1.2.1.6.2.1.2 Tornado Missile Impact Analysis Requirements

The SC-I structures are analyzed for the effects of tornado-generated missiles. This analysis is performed in accordance with the guidance provided in NUREG-0800, Section 3.5.3, Subsection II, with the tornado-generated missile spectrum (see Table 1.1.2-2). The response of a structure to missile impact depends largely on the location of impact, the material properties of the structure, the dynamic properties of the missile, and the kinetic energy of the missile. Both the local and overall effects of missile impact are examined, with appropriate consideration given to impact effects of the loading. Some localized overstressing, deformation, and damage is permissible for structures subjected to missile impact. It is acceptable to allow inelastic or plastic structural response when examining the effects of missile impact.

The modified National Defense Research Committee (NDRC) formula, as specified in ASCE Manual and Report No. 58, is used to estimate missile penetration. The missile barrier thickness is selected to preclude perforation through the concrete barrier and to avoid generation of secondary missiles as a result of scabbing. Section 6.4.1.2.1, on page 336 of the ASCE Manual and Report No. 58, provides the modified NDRC formula for determining missile penetration.

These modified NDRC formulas are used to analyze for the local penetration, perforation, and scabbing effects of tornado-generated missiles on concrete barriers. It is assured that the concrete tornado missile barriers are of sufficient thickness to prevent complete perforation of missiles and generation of secondary missiles. In order to provide sufficient safety margin, the perforation thickness and the scabbing thickness is increased by a factor of 1.2 as recommended in ASCE Manual and Report No. 58.

Overall effects of missile impact on a structural system are investigated to ensure the structure retains its integrity and functionality subsequent to a missile strike. Overall missile impact effects were analyzed much the same as for other loads, with additional consideration given to increased loading due to the dynamics of the impacting load. Tornado pressure boundary structures are checked for missiles impacting at the worst locations possible for subsequent damage. The non-deformable penetrating (3-in diameter steel pipe) missile is determined to be the most applicable missile spectrum for examining challenges to the overall integrity of the structure. Only one missile impacting a structure at a given time is considered.

Overall effects of a non-deformable penetrating (3-in diameter steel pipe) missile on concrete barriers are investigated in accordance with the hard missile impact analysis specified in Section 6.4.2.2 of ASCE Manual and Report No. 58.

Overall effects of a deformable penetrating (2x4 timber plank) missile on concrete barriers are investigated in accordance with the soft missile impact analysis specified in Section 6.4.2.1.2 of ASCE Manual and Report No. 58.

Steel barriers shall be analyzed for both the local and overall effects of tornado-generated missiles. Local effects shall be evaluated using the more stringent results from the Stanford Equation and the Ballistic Research Laboratory Formula, as specified in ASCE Manual and Report No. 58. Overall effects shall be evaluated using the equivalent static load method for the case with no penetration as referenced in NUREG-0800, Section 3.5.3.

1.1.2.1.6.2.2 Structural Design Requirements for SC-I Structures

The structural design requirements for SC-I concrete structures and SC-I steel structures are described below:

1.1.2.1.6.2.2.1 Structural Design Requirements for SC-I Concrete Structures

The design of SC-I concrete structures uses the “ultimate strength design methods” in accordance with American Concrete Institute (ACI)-349-97.

Structural concrete used in construction of SC-I structures has a minimum compressive strength of 4,000 pounds per square inch (psi). Reinforcing steel used in SC-I structures has a minimum yield strength of 60,000 psi.

Design of concrete structures also follows the guidelines and recommendations provided in ACI-207.1R.

Concrete walls and roofs exceed the required thicknesses to accommodate tornado missile impact as determined in accordance with the guidance in NUREG-0800, Standard Review Plan 3.5.3 (see Table 1.1.2-4).

One exception is taken to ACI 349-97, Appendix B as follows: The design of SC-I embedded plates, cast-in place anchors, and post-installed concrete anchors are in accordance with the requirements of ACI 349-01, Appendix B. An exception is also taken to ACI 349-97 Section 21.5.4.1.

For wall to wall and slab to wall joints, where ACI 349-97 requires anchorage for fy (specified yield), it is acceptable to provide a reduced development length if it is demonstrated that there is adequate capacity for 1.67 times the Design Earthquake Load in design loading combinations.

Alternate methodology may be used to evaluate the capacity of standard hooks with reduced development for wall to wall and slab to wall joints provided potential concrete breakout is considered.

Splicing of reinforcing by lapping, mechanical means, or welding is permitted as long as the ductility and confinement requirements of ACI 349 and its appendices and ACI 439.3R are satisfied. Adequate reinforcing is provided at construction joints to develop shear-friction forces across the joints.

1.1.2.1.6.2.2.2 Structural Design Requirements for SC-I Steel Structures

SC-I steel structures are designed in accordance with American Institute of Steel Construction (AISC) N690. Elastic design methods are generally used for steel design. However, under the extreme loading conditions of seismic or missile impact loading, plastic design methods and use of ultimate steel strength may be used. The special requirements specified in AISC N690 are used when designing for moving or impact loads.

Structural steel connections are designed as either friction or bearing type bolted connections or welded connections. Bolted connections are designed in accordance with AISC N690, with guidance from AISC, Manual of Steel Construction, Volume II. Welded connections are designed in accordance with AISC N690, American Welding Society (AWS) D1.1, and AWS D1.6. Guidance for the design of connections and member properties for hollow structural sections (HSS) is given in AISC Connections Manual for Hollow Structural Sections or AISC Steel Design Guide 24 for Hollow Structural Section Connections.

The requirements of AISC N690 are supplemented by the following provisions recommended in Proposed NRC Staff Position on the Use of Industry Standard ANSI/AISC N690 in the Advanced Reactor Applications, dated 22 Feb 1993:

- In Section Q1.0.2, the definition of secondary stress applies to stresses developed by temperature loading only (i.e., other loads are considered to produce primary stresses).
- The following notes are added to Section Q1.3:
- “When any load reduces the effect of other loads, the corresponding coefficient for the load is taken as 0.9, if it can be demonstrated that the load is always present or occurs simultaneously with other loads. Otherwise, the coefficient for that load is taken as zero.”
- “Where the structural effects of differential settlement are present, they are included with the dead load ‘D’.”

- The stress limit coefficients for compression in Table Q1.5.7.1 are as follows:
  - 1.3 instead of 1.5, stated in footnote (C), in loading combinations 2, 5, and 6
  - 1.4 instead of 1.6 in loading combinations 7, 8, and 9
  - 1.6 instead of 1.7 in loading combination 11.
- The following note is added to Section Q1.5.8:

"For constrained (rotation and/or displacement) members supporting safety related structures, systems, or components, the stresses under loading combinations 9, 10, and 11 are limited to those allowed in Table Q1.5.7.1 as modified by provision above. Ductility factors of Table Q1.5.8.1 (or the provision below) are not used in these cases."

- For ductility factors 'micro (μ)' in Sections Q1.5.7.2 and Q1.5.8, the provisions of NUREG-0800, Section 3.5.3, "Barrier Design Procedures," Subsection II.2, Appendix A, are substituted in lieu of Table Q1.5.8.1.
- In loading combination 9 of Section Q2.1, the load factor applied to load (Pascal) P_a is 1.5/1.1 = 1.37, instead of 1.25.
- Sections Q1.24 and Q1.25.10 are supplemented with the following requirements regarding painting of structural steel:
  - Shop painting is in accordance with Section M3 of AISC ASD, 9th Edition.
  - Exposed areas after installation are field painted (or coated) in accordance with the applicable Section M3 of AISC ASD, 9th Edition.

Welding activities associated with SC-I structural steel components and their connections are accomplished in accordance with written procedures and meet the requirements of AWS D1.1 or AWS D1.6. The visual acceptance criteria for carbon and low alloy steel welds are as defined in AWS D1.1 or NCIG-01. The visual acceptance criteria for stainless steel welds and dissimilar welds are as defined in AWS D1.6.

Structural steel materials used in construction of SC-I buildings and structures consist of American Society for Testing and Materials) (ASTM A36, or A992 rolled shapes, ASTM A500 Gr. B tube shapes, and ASTM A36, A572 Gr. 50, A514 or A852 steel plates. ASTM A240 Type 304 or 304L is used for stainless steel plates, unless otherwise specified. Use of other materials is permissible as needed for specific designs. Bolts used for primary structural connections are either ASTM A325 or A490 for carbon steel and A193, GR B8 for stainless steel. A307 bolts are used for attaching ancillary components or equipment to structures, but they are not used for primary structural member connections. Welding electrodes in accordance with AWS D1.1 or D1.6 are selected to be compatible with the materials being joined.

1.1.2.1.6.2.2.3 Foundation Design Requirements for SC-I Structures

SC-I structures are supported by mat and spread foundation systems established on prepared natural soils or on engineered structural fill. The soil conditions at the site satisfy applicable static and seismic design codes and acceptance criteria specified in NUREG-0800, Section 3.8.5, "Foundations," II.5, "Structural Acceptance Criteria."

The minimum factor of safety against bearing capacity failure due to static loads (dead loads + normal live loads, such as equipment loads) is 3.0, based on typical geotechnical engineering practice (p. 271, Peck, Hanson and Thornburn, 1974). The minimum factor of safety against bearing capacity failure due to static loads + severe environmental loads, such as design wind, is 1.5, and for static loads + extreme environmental loads, such as seismic loads due to the design earthquake or wind loads due to the design tornado is 1.1. This is consistent with the acceptance criteria specified in NUREG-0800, Section 3.8.5, "Foundations," II.5, "Structural Acceptance Criteria" for the factor of safety against overturning.

Additionally, the SC-I structures are designed for the effect of differential and post earthquake induced settlements. The required structural element strengths are within the code allowed capabilities.

For evaluation of subsurface conditions to include liquefaction and dynamic settlements, bedrock motions based upon a 2,000-year recurrence frequency bedrock spectrum are used. They are scaled so that when amplified through the site soil profile, the resulting surface ground motion will have 0.20 acceleration due to gravity (g) peak ground acceleration. A settlement monitoring program is implemented for SC-I structures. Settlement monuments are provided to track total and differential settlement. The actual settlement versus the predicted settlement is evaluated.

1.1.2.1.6.3 Codes and Standards for SC-I Structures

Codes and standards applied to the SC-I structures at the MFFF include the following:

The effective date of these Codes and Standards is the MFFF contract date of 22 Mar 1999, unless otherwise noted.

American Concrete Institute (ACI)

- ACI-224R-90, *Control of Concrete Cracking in Concrete Structures*
- ACI-301-99, *Standard Specifications for Structural Concrete.*

  Section 5.3.7.5 of ACI 301-99 specifies a site-mixed cement repair mortar not to exceed (in cement concentration) a mixture greater than 1:2.5 cement to sand. Construction used a 1:1 ratio of cement to sand for minor repairs such as filling taper tie holes and abandoned drilled holes.
- ACI-315-99, *Details and Detailing of Concrete Reinforcement*
- ACI-336.2R-88, *Suggested Analysis and Design Procedures for Combined Footings and Mats*
- ACI-349-97, *Code Requirements for Nuclear Safety-Related Concrete Structures & Commentary*, (Note: excluding Anchoring to Concrete criteria, see ACI-349-01, Appendix B below)

Section 3.5.3.5 of ACI-349-97: For deformed wire used on embedment plates that are required to meet the minimum height of deformation indicated in Section 6.4 of ASTM A496, specifically Nelson Stud Welding (NSW) D2L Deformed Bar Anchors (DBA) meeting NSW material process specification MPS-102D Revision C, deviation to strict compliance with height of deformation is allowed provided the relative rib area is greater than 0.083 for the 5/8-inch NSW D2L DBA and 0.11 for the ¾-inch NSW D2L DBA. The Relative Rib area is defined in ACI 408-03, Chapter 1 Figure 1-3 as $0.8x(h_r/s_r)$, where h_r is the average height of deformation and s_r is the average spacing of the deformations.

- ACI-349.1R-91, *Reinforced Concrete Design for Thermal Effects on Nuclear Power Plant Structures*, Reapproved 1996
- ACI-351.1R-99, *Grouting for Support of Equipment & Machinery*
- ACI-352R-91, *Recommendations for Design of Beam-Column Joints in Monolithic Reinforced Concrete Structures*, Reapproved 1997
- ACI-352.1R-89, *Recommendations for Design of Slab-Column Connections in Monolithic Reinforced Concrete Structures*, Reapproved 1997
- ACI-349-01, Appendix B, Code Requirements for Nuclear Safety Related Concrete Structures and Commentary, *Anchoring to Concrete* (for anchoring to concrete only)
- ACI-360R-92, *Design of Slabs on Grade*, Reapproved 1997
- ACI-351.2R-94, *Foundations for Static Equipment*
- ACI-439.3R-91, *Mechanical Connections of Reinforcing Bars*
- ACI-SP-152-95, *Design and Performance of Mat Foundations*
- ACI-503R-93, *Use of Epoxy Compounds with Concret*e
- ACI-442-88, *Response of Concrete Buildings to Lateral Forces*
- ACI-207.1R-96, *Mass Concrete*
- ACI-207.2R-95, *Effect of Restraint, Volume Change, and Reinforcement on Cracking of Mass Concrete*
- ACI-207.4R-93, *Cooling and Insulating Systems for Mass Concrete*
- ACI-SP-175-98, Concrete and Blast Effects

American Institute of Steel Construction (AISC)

- AISC ASD, *Manual of Steel Construction, Allowable Stress Design*, 9th Edition, 1989 and Supplement #1, dated December 17, 2001
- ANSI/AISC N690-1994, *Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities* and Supplement 1 dated April 15, 2002 (with exceptions: Section CQ1.0.1 Scope and Table CQ1.0.1)

  Sway struts complying with Manufacturers Standardization Society of the Valve and Fittings Industry, Inc. (MSS) SP-58 are acceptable for use as tension/compression members in structural applications as long as any capacity increase due to load category comply with ASME, Section III, NF Code.

- AISC, *Seismic Provisions for Structural Steel Buildings*, April 1997
- AISC *Manual of Steel Construction, Volume II - Connections*, ASD 9th Edition, 1989 /LRFD 2nd Edition, 1998
- AISC *Hollow Structural Sections – Connections Manual*, 1997 or AISC Steel Design Guide 24 for Hollow Structural Section Connections, 2010

American Society of Civil Engineers (ASCE)

- ASCE Standard 4-98, *Seismic Analysis of Safety Related Nuclear Structures*
- ASCE Standard 7-98, *Minimum Design Loads for Buildings and Other Structures*
- ASCE Standard 8-02, *Specification for the Design of Cold-Formed Stainless Steel Structural Members*
- ASCE Manual & Report No. 58-80, *Structural Analysis and Design of Nuclear Plant Facilities*
- ASCE & SEI, 1999, *Structural Design for Physical Security, State of the Practice*

American Welding Society (AWS)

- AWS-D1.1-98, *Structural Welding Code – Steel*, 1998

  One end of the deformed bars connecting sandwich ledger plates for 12 inch thick walls supporting precast slabs were welded using the flux core arc welding process (FCAW). The FCAW fillet welds were found to be undersize according to Table 5.8 and Table 7.2 of AWS-D1.1-98. Calculations demonstrate that the deformed bar welds associated with the worst case (minimum effective size based on testing and examination) are adequate because their load capacity exceeds the design basis values for load carrying capability of the embed plates.
- NCIG-01, *Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants*, Revision 2, EPRI NP-5380.
- AWS-D1.6-99, *Structural Welding Code-Stainless Steel,* 1999
- AWS-D1.3-98, *Structural Welding Code-Sheet Steel, 1998*

American Association of State Highway and Transportation Officials (AASHTO)

- AASHTO HB-16, Standard Specifications for Highway Bridges, Sixteenth Edition, 1996

American Iron and Steel Institute (AISI)

- AISI, Specifications for the Design of Cold-Formed Steel Structural Members, 1996
- Research Council on Structural Connections of the Engineering Foundation (RCSC)
- Research Council on Structural Connections, Specification for Structural Joints Using ASTM A325 and A490 Bolts, June 23, 2000

Code of Federal Regulations (CFR)

- 10 CFR Part 70, *Domestic Licensing of Special Nuclear Material*
- 10 CFR Part 73, *Physical Protection of Plants and Materials*
- 10 CFR Part 75, Safeguards on Nuclear Material

Crane Manufacturers Association of America (CMAA)

- CMAA Spec. 70, *Specification for Top Running Bridge and Gantry Type Multiple Girder Electrical Overhead Traveling Cranes*, 1994
- CMAA Spec. 74, *Specification for Top Running and Under Running Types of Single Girder Electric Overhead Traveling Cranes*, 1994

Manufacturers Standardization Society of the Valve and Fittings Industry, Inc. (MSS)

- MSS SP-58 – 1993, *Pipe Hangers and Supports – Materials, Design and Manufacture*

SRS Engineering Standards Manual (WSRC-TM-95-1)

- Engineering Standard No. 01060, *Structural Design Criteria*, Revision 5, dated September 2001
- Engineering Standard No. 01110, *Civil Site Design Criteria*, Revision 4, dated July 9, 2002

U.S. Nuclear Regulatory Commission (NUREG)

- NUREG-0800, Standard Review Plan, Section 3.5.3, *Barrier Design Procedures*, July 1981
- NUREG-0800, Standard Review Plan, Section 3.8.4, *Other Seismic Category I Structures*, July 1981
- NUREG-0800, Standard Review Plan, Section 3.3.2, *Tornado Loading*, July 1981
- NUREG-0800, Standard Review Plan, Section 3.5.1.6, *Aircraft Hazards*, April 1996.
- NUREG-0800, Standard Review Plan, Section 3.7.1, *Seismic Design Parameters*, August 1989
- NUREG-0800, Standard Review Plan, Section 3.7.2, *Seismic System Analysis*, August 1989
- NUREG-0612, *Control of Heavy Loads at Nuclear Plants*, July 1980
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1.1.2.1.6.4 Design Values for SC-I Structures

1.1.2.1.6.4.1 Structural Design Loads for SC-I Structures

Design loads are based upon anticipated building loads (i.e., dead loads, live loads, operating and transient loads, and natural phenomena hazard loads). These loads are divided into three classifications (normal loads, severe environmental loads, and extreme environmental loads) consistent with the guidance provided in NUREG-0800, Section 3.8.4. See Table 1.1.2-2 for the MFFF site design criteria summary. See Table 1.1.2-1 for the structures located at the MFFF site, along with the seismic category classification.

1.1.2.1.6.4.1.1 Normal Loads

Normal loads are those loads associated with normal operation of the MFFF. Normal loads include the following: dead loads (D); live loads (L); hydrostatic fluid pressure loads (F); lateral soil pressure loads (H); thermal loads (T_o); and pipe, HVAC duct, conduit, and cable tray reaction loads (R_o). These loads are defined in the following subsections.

Dead Loads

Dead loads (D) are gravity loads and are defined as loads, which include related internal moments and forces that are constant in magnitude, orientation, and point of application. Dead loads include the mass of the structure, permanent equipment loads, and permanent hydrostatic loads that have constant fluid levels. The weight of permanent items (e.g., roofing materials, including insulation and engineered fill, wall materials, equipment, cable trays, mechanical piping, and HVAC equipment and ducts) is included in the dead load. When determining dead loads, the effects of differential settlement are considered.

Actual equipment loads are applied to the design of structural systems and components. In addition, unless specifically reviewed, a minimum uniform dead load is applied to elevated floors, platforms, roof areas, walls, and floor slabs to account for miscellaneous equipment loads, piping, cable trays, conduits, and HVAC ducts. A uniform dead load of 25 pounds per square foot (psf) is applied to wall surfaces, (i.e., a total of 50 psf on each wall panel). Also, 50 psf is applied to the underside of elevated floor slabs and roof slabs to account for miscellaneous attachments. A uniform dead load of 50 psf is applied to platforms (25 psf on top and 25 psf at bottom of platforms).

Live Loads

Live loads (L) are defined as normal load, which includes related internal moments and forces that may vary with intensity, orientation, and/or location of application. Movable equipment loads, loads caused by vibration, support movement effects, and operating loads are types of live loads. The following subsections provide design requirements for the various types of live loads.

<u>Floor Live Loads</u>

Minimum uniformly distributed live loads are in accordance with ASCE Standard 7 and are applied as follows:

Platform and Work area	125 psf	(Note 1)
Light Storage	125 psf	
Heavy Storage	250 psf	(Note 2)
Heavy Operation	250 psf	(Note 3)
Office	100 psf	
Computer room	150 psf	
Dining/Meeting rooms	100 psf	
Laboratory	200 psf	
Toilet areas	100 psf	
Mechanical (Utility) rooms	150 psf	
Electrical rooms	150 psf	
Stairs, Fire Escapes, and Corridors	100 psf	
Transportation Vehicle Loads	300 psf or forklift truck, 6 kip capacity (HS20-44 capacity in designated areas)	

Roof	50 psf, ASCE Standard 7, Table 4-1 (This load does not combine concurrently with the rain load, 50 psf, on the roof).

Note 1: Special use platforms provided by the equipment supplier, such as glovebox access platforms, may be designed to a different live load as defined in the appropriate equipment specification.

Note 2: Canister and other storage areas may require a greater live load.

Note 3: Includes rooms in BMP with gloveboxes and/or heavy equipment and rooms in BAP with process cells.

Concentrated Live Load

As described in ASCE Standard 7, Section 4, floors and other similar surfaces shall be designed to safely support the greatest load effects from Uniformly Distributed Live Loads or a Concentrated Live Load. Concentrated Live Loads shall be applied over a 2.5-ft square area or the actual loaded area, whichever is greater, and shall be located so as to produce the maximum load effects on structural members or area.

Rain Loads

Rain loads (R) are determined in accordance with the requirements of ASCE Standard 7, Section 8. The roof system for SC-I structures is designed for a minimum rain load of 50 psf. The design load of 50 psf is equal to more than 9.6 in equivalent weight of standing water and is adequate to account for effects that may result from ponding of rainwater due to deflection of the supporting roof or the blockage of primary roof drains. This design load of 50 psf is equivalent to 9.6 in accumulation of precipitation over a period of 1 hour and 39 minutes based on a linear interpolation of the data from Table 1.1.2-2. Parapets or other structures, which may potentially contribute to significant ponding, are not used on the roofs of SC-I structures. The rain load is not applied concurrently with the roof live load.

Snow and Ice Loads

Snow (S) and ice (I) loads are determined in accordance with Table 1.1.2-2. The minimum design live load due to snow and ice is 10 psf. This load is applied concurrently with the roof live load. An exposure factor of $C_e = 1.0$ is used to consider wind effects for analysis and design of roof structures resisting snow and ice loads. An importance factor of I = 1.2 is used for SC-I structures.

Transportation Vehicle Loads and Heavy Floor Loads

Loads caused by transportation vehicular truck traffic in designated building areas are in accordance with standard loadings defined by AASHTO HB-16. The minimum truck loading of HS 20-44 is used for wheel loading design. Special heavy-loading conditions resulting from transport of finished fuel assemblies and storage casks on trucks are considered. Heavy floor loading of 300 psf or forklift truck (6 kip capacity) in areas used for transportation, transfer, and storage of finished fuel assemblies is considered along with dynamic load factors for impact resulting from placing the moving loads on the floor or other area of a structure.

Crane, Monorail, Hoist, and Elevator Loads

These loads apply to structural members and components required to support permanently installed cranes, monorail, hoists, and elevators. Design loads for cranes, monorail, hoists, and elevators envelop, as a minimum, the full-rated capacity of the crane, monorail, hoist, and elevator. This includes impact loads as well as test load requirements. Seismic effects on fully loaded cranes, monorail, hoist and elevators are considered. The effects of crane load drop are also evaluated in accordance with guidance provided in NUREG-0612.

Hydrostatic Fluid Pressure Loads

Hydrostatic fluid pressure loads (F) are due to fluids held in internal building compartments (such as tanks). Fluid pressure loads are limited to containment curbs to limit postulated spills, as described in Section 1.1.2.1.6.5.2, SC-1 Steel Structures, Process Cell Drip Trays, and protection from postulated flooding of the BRP pipe tunnel.

Lateral Soil Pressure Loads

Lateral soil pressure loads (H) on structures and/or elements of structures retaining soil are based on the density of the soil and any surcharge load, plus the hydrostatic pressure caused by the groundwater or soil saturation. Lateral soil pressures and coefficients for exterior walls are documented within the MFFF Site Geotechnical Calculations. The minimum lateral soil pressure loads on structures and/or elements of structures retaining soil are as defined in ASCE Standard 7, Section 5. The soil pressure caused by earthquakes based on ASCE Standard 4 is included.

The groundwater at the MFFF site is approximately 70 ft below finished grade; therefore, no hydrostatic pressure caused by groundwater and flooding is anticipated on building structures.

Thermal Loads

Thermal loads (T_o) consist of thermally induced forces and moments resulting from operation and environmental conditions affecting the building structure. Thermal loads are based on the most critical transient or steady-state condition. Thermal expansion loads caused by axial restraint, as well as loads resulting from thermal gradients, are considered. Thermal loads considered include the ambient temperature gradient imposed on the structure by process equipment.

Pipe, HVAC Duct, Conduit, and Cable Tray Reaction Loads

Pipe, HVAC duct, conduit, and cable tray reaction loads (R_o) are those loads applied by the distribution system supports during normal operating conditions, based on the most critical transient or steady-state condition. See previously described, Dead Loads, for the design allowance to address these loads. Loads are tracked to ensure that the final design envelops actual loads.

1.1.2.1.6.4.1.2 Severe Environmental Loads

Severe environmental loads are those loads that are encountered infrequently during the life of the MFFF. They include severe wind loads (W) and flood loads (F'). These loads are defined in the following subsections.

Wind Loads

Wind loads (W) are those pressure loads generated by the design basis wind and are determined by procedures in ASCE Standard 7, Section 6. See Table 1.1.2-2 for the severe wind speed, which is used in the design of SC-I buildings, structures, and facilities. These loads do not incorporate loads associated with tornadoes (see the description of tornadoes below).

Flood Loads

Flood loads (F') are caused by exterior flood waters from the design basis flood exerting forces and moments on exterior building structures or entering a building and exerting loads on interior building structures. Flood loads on building, structures, and facilities are determined in accordance with ASCE Standard 7, Section 5.3. Guidance for determining the design basis flood is provided in Regulatory Guide 3.40. See Table 1.1.2-2 for the design basis flood and probable maximum flood elevations that are well below the MFFF site elevation. Thus, external flood loads are not applicable to the MFFF.

Disturbance HVAC Pressures

Disturbance HVAC pressure loads (L_{HVS}) are pressure loadings resulting from abnormal operations of the ventilation system. These loadings are treated as severe live loads in the areas where they occur. Loads are tracked to ensure that the final design envelops actual loads.

1.1.2.1.6.4.1.3 Extreme Environmental Loads

Extreme environmental loads are those loads that are credible but are not expected to occur during the life of the MFFF. They include design basis seismic loads (E'), tornado loads (W_t), explosive loads, and loads due to post-earthquake settlements. These loads are defined in the following subsections. As described in this subsection, the possibility of aircraft impact and range fires were determined not to be credible events.

Seismic Loads

<u>Design Earthquake Loads for SC-I Structures</u>

In accordance with the guidance of Regulatory Guide 3.14, and DOE-STD-1020-94, SC-I buildings, structures, and facilities at the MFFF are designed to accommodate a design earthquake. To evaluate seismic response in the stress analysis, design seismic loads (E') include inertia loads due to 25% of the live load along with structure and equipment dead weights.

The design earthquake for the MFFF is defined in Section 1.3.6 and summarized in Table 1.1.2-2. Design Earthquake Loads (E') for SC-I buildings, structures, and facilities are determined based upon a horizontal and vertical component. The horizontal component at the ground surface is characterized by a horizontal spectrum shape from Regulatory Guide 1.60, scaled to 0.20g peak ground acceleration (PGA) (see Figure 1.1.2-51). The vertical component is the vertical spectrum shape from Regulatory Guide 1.60 scaled to 0.20g peak ground acceleration (see Figure 1.1.2-51). Methods used for the soil structure interaction (SSI) determining seismic responses from these acceleration input criteria conform to NUREG-0800, Sections 3.7.1 and 3.7.2 and the requirements of ASCE Standard 4.

Total seismic loads affecting a structure are determined by simultaneously applying the design earthquake accelerations in the three orthogonal directions (two horizontal and one vertical). Appropriate consideration of SSI, torsional effects, structural frequency, stiffness, and displacement is factored into structure-specific seismic load analyses. The SSI analyses for the BMF and the BEG are performed on a simplified 3D finite element model and a 3D lumped-mass stick model, respectively, using the computer code SASSI (Framatome-ANP version). From the SSI analysis, the response spectra at the foundation and each floor elevation and roof level are obtained for the building and equipment design and acceleration profile for building design.

<u>Synthetic Time History of Free-Field Seismic Motion</u>

Three statistically independent synthetic time histories, H1, H2, and V, are generated to closely match the design earthquake spectra. Components H1 and H2 are in the EW and NS directions, respectively, and component V is in the vertical direction. The duration of the time history is 24 seconds. See Figure 1.1.2-52 for the plots of the synthetic time histories. For each component, the SIMQKE code is first used to generate a raw synthetic time history with response spectra closely straddled about the target design spectra. Then the raw synthetic time history is modified by running the MOTH code in iterations, until its response spectra for 2%, 5% and 7% damping envelop the corresponding target design spectra at the following 76 frequencies (Hertz [Hz]):

- 0.2, ... (in increments of 0.1) ..., 3.0, 3.15, 3.3, 3.45, 3.6, ... (in increments of 0.2), 5.0, (in increments of 0.25) ..., 8.0, ... (in increments of 0.5) ..., 15.0, ... (in increments of 1.0), 18.0, 20.0, 22.0, 25.0, 28.0, 31.0, 34.0, 50.0.

The above frequencies are based on the guidelines of the NRC NUREG-0800, Standard Review Plan (SRP) Section 3.7.1. Version 1.1A of RESPEC, a FANP code, is used to compute the response spectra of the synthetic time histories. According to the guidelines of SRP Section 3.7.1, each response spectrum of the synthetic time history does not fall below the corresponding design spectrum at more than 5 frequency points nor by more than 10%. In addition, in view of the anticipated revision to the SRP, the 5% damping spectrum of the synthetic time history does not exceed the corresponding design spectrum by more than 130%. See Figure 1.1.2-53 to Figure 1.1.2-55 for diagrams of the 2%, 5%, and 7% damping response spectra of the synthetic time histories, H1, H2, and V, respectively.

In addition to meeting NUREG-0800, Section 3.7.2 "Seismic System Analysis" guidance of spectrum enveloping, the average Power Spectral Density (PSD) function of each horizontal component of the time histories also envelops the minimum PSD function specified in SRP Section 3.7.1 at frequencies between 0.3 and 24 Hz (see Figure 1.1.2-56). The average PSD function of the time history at a given frequency is obtained by averaging the computed PSD function over a ±20% window centered about the given frequency. The PSD function is computed using the PSD code. The minimum PSD function is equal to 80% of the target PSD specified in Appendix A of SRP Section 3.7.1, scaled accordingly to correspond to a horizontal PGA of 0.20g. For the vertical time history, SRP Section 3.7.1 does not specify a minimum PSD. For informational purposes, however, the average PSD of the vertical time history is computed and compared with the 80% horizontal target PSD (see Figure 1.1.2-56).

The cross correlation coefficients between the three components of time histories are:

Components	Cross Correlation Coefficient (Absolute Value Shown)
H1 & H2	0.046
H1 & V	0.039
H2 & V	0.111

Statistic independence between the three components of synthetic time histories is thus established because the three cross correlation coefficients are smaller than the limit value of 0.14 suggested in SRP Section 3.7.1 (NUREG 0800).

Soil Model

In the SSI analysis, the soil model consists of a sufficient number of idealized soil layers from the ground surface to the bedrock. The thickness of each soil layer is small enough to allow vertical propagation of shear waves having frequencies up to the desired cutoff frequencies. The properties of the idealized soil layers are developed based upon the information provided by the soil exploration report and site response analysis. Variations in soil properties are considered. The exploration report of the soil provides best-estimate, upper-bound and lower-bound values of the soil layers shear modulus, G or shear velocity, V_s, compression wave velocity, V_p, soil damping value, β, Poisson's ratio, μ, and density. See Table 1.1.2-5 to Table 1.1.2-7 for the strain-dependent soil properties determined from the site response analysis, for use in the SSI analysis. The value of the Poisson's ratio for the soil layers is in the range of 0.25 to 0.47, with 0.47 being the Poisson's ratio for soil layers below elevation 211'. With the breadth of this range, the effect of the Poisson's ratio on the compression wave propagation is adequately accounted for in the SSI analysis.

Structure Models

A detailed 3D finite element model (FEM) using standard computer structural modeling codes (e.g., Analysis System [ANSYS]) is first generated based on the structural drawings. In addition to the applicable dead weights and equipment weights, the FEM includes appropriate parts of the live loads (25% of the applicable live loads, which are verified during design) in the mass properties of the model.

A simplified 3D finite element model is generated from the detailed 3D model based on coarser mesh. The simplified 3D finite element model is used for the SSI analysis of the BMF (e.g., integrated structures of BMP, BAP, and BSR). In the simplified model, the in-plane flexibility of the floor slabs in the BMF is sufficiently represented and the slabs are assumed uncracked. The embedment of BAP and BSR is shallow compared to the plan dimensions of BMF and is ignored in the SSI analysis. Thus for the purpose of the SSI analysis, the simplified 3D structural model of the BMF is surface founded. See Table 1.1.2-8 for the natural frequencies for the first 20 modes of the simplified finite element model. The frequencies of the first major horizontal mode are 8.78 Hz and 9.19 Hz in the NS and EW directions, respectively. They are about 10% higher than those of the detailed FEM. See Figure 1.1.2-57A and Figure 1.1.2-57B for the perspective view of the simplified finite element model and the plan view of the roof, respectively. See Figure 1.1.2-58 for the North and West elevations of the model.

For the BEG, a 3D lumped-mass stick model is generated for use in the SSI analysis. See Figure 1.1.2-59 for the 3D lumped-mass stick model of BEG with a grid of horizontal rigid beams representing the rigid slab at grade.

Damping Values for Structures

The following structural damping values, which are in accordance with Regulatory Guide 1.61 for a safe shutdown earthquake (design earthquake, E'), are used to determine seismic loading:

Structure Type	% of Critical Damping
Welded Steel	4
Bolted Steel	7
Reinforced Concrete	7

Both the BMF and BEG are reinforced concrete structures, and in the SSI analysis a 7% damping is used for the structural elements of both structures.

Soil-Structure Interaction (SSI) Model

The SSI model is developed based on the Framatome ANP version of the SASSI code by coupling the 3D finite element model of the building with the soil model at the grade (El. 273'0"). The soil-structure coupling takes place at the nodes of the slab at grade that are interconnected by the horizontal grid of rigid beam elements. This simulates the rigid slab for the purpose of the SSI analysis. See Figure 1.1.2-60 for a conceptual elevation view of the SSI model for the BMF (looking east) in which the structural model and soil model are coupled to each other at the structural nodes on the rigid slab located at grade. The SSI model for the BEG is similar to that for the BMF.

Soil-Structure Interaction (SSI) Analysis

The SSI analysis is by means of the Framatome ANP version of the SASSI code. It performs the analysis in the frequency domain using the complex response method. Briefly, the code first computes the impedances at the soil-structure interaction nodes. The impedances form the boundary conditions for the structural model at the interaction nodes. The code then solves for the response of the structure to each of the three components of seismic input motion specified at grade in the free-field. The analyses are done one at a time for each of the three components of seismic input motions, and are then repeated for the three soil conditions.

Maximum Structural Responses

The maximum response acceleration at a given nodal location on the finite element model of the structure is the maximum amplitude (i.e., the zero period acceleration [ZPA]) of the corresponding nodal response acceleration time history output from the SSI analysis. The nodal response acceleration time history properly includes the contributions from the three earthquake components, and is also the basis for the generation of floor response spectrum as discussed later.

The maximum vertical floor accelerations are valid only for the design of slab panels that are sufficiently rigid in the vertical direction (i.e., having a fundamental vertical frequency exceeding 33 Hz when the slab is assumed cracked). Slab panels that are identified to be "flexible" may be subjected to amplified vertical accelerations. To account for the amplification effect, additional time history analyses are performed in which each flexible slab panel is modeled simply as a vertical 1-degree-of-freedom (1-DOF) system having a damping of 7% and a frequency corresponding to the fundamental vertical frequency of the slab panel. In the BMF, the flexible slabs at and above grade are divided into six representative groups based on the vertical slab frequency, f_{cr}, for the cracked condition (see Table 1.1.2-9). The accompanying slab frequency, f_{uc}, for the uncracked condition is taken to be 1.4 x f_{cr}. See Table 1.1.2-10 for lists of those floor slab panels in the BMF that are identified to be flexible in accordance with the grouping criteria of Table 1.1.2-9. In the BEG, there are only two flexible slabs (see Table 1.1.2-11). For each of the flexible slab panels identified on a given floor elevation, a time history analysis of the corresponding vertical 1-DOF system is performed using the SAP2000 code. The input motion is the vertical response acceleration time history output from the SSI analysis at the applicable nodal point on the particular floor elevation of the building. Both the cracked and uncracked slab conditions are considered. The vertical response acceleration time history of the 1-DOF system is the basis for the generation of the flexible slab vertical floor response spectrum to be discussed later. Its ZPA represents the maximum vertical acceleration of the particular flexible slab panel. The worst vertical acceleration for both the cracked and uncracked slab conditions and the rigid slab is the design acceleration for the particular flexible slab panel.

Similarly, the horizontal floor accelerations obtained from the SSI analysis are valid only for the calculation of the out-of-plane seismic loads for sufficiently rigid walls and the subsystems mounted on such walls. The horizontal floor accelerations will be amplified by flexible walls in the out-of-plane directions. The analysis for the flexible wall is similar to that of the flexible slab. The wall is modeled as a 7% damped, 1-DOF system, and the input motion is conservatively taken to be the floor acceleration time history at the upper end of the wall. The horizontal response acceleration time history of the 1-DOF system is the basis for the generation of the flexible wall horizontal response spectrum to be discussed later. The flexible walls are not expected to crack under the amplified seismic loads, and only the uncracked condition need be considered. See Table 1.1.2-12 for the grouping of the flexible walls in the BMF based on the representative uncracked condition frequency, f_{uc}, of the wall. See Table 1.1.2-13 for identification of the flexible walls in the BMF. There is no flexible wall in the BEG.

The maximum relative displacements between floors are not available from the outputs of the SSI analysis with the SASSI code. They are conservatively computed from the static analysis of the FEM of the structure in which the maximum floor accelerations from the SSI analysis are applied as the seismic loads. Maximum relative displacements of floors with respect to slab on grade are provided in Table 1.1.2-15. Inter-floor relative horizontal displacements are provided in Table 1.1.2-16.

In-Structure Response Spectrum Envelope with Peak Broadening

In each of the three directions (north-south, east-west, and vertical) and at each given structural location, the in-structure response spectra from the 3D SSI analysis of the simplified finite element model of the BMF and the stick model of the BEG for the lower-bound, best-estimate, and upper-bound soil conditions, were broadened. For the BMF, a broadening of the in-structure response spectrum peak(s) by -25% and +15% was applied to the best estimate soil condition and by -20% and +10% to the lower-bound and upper-bound soil conditions for the horizontal spectra. The extra -10% broadening of the horizontal spectral peaks accounts for the effect of the 10% discrepancy mentioned previously in the first major horizontal mode frequency between the simplified and detailed FEM. The corresponding peak-broadening for the vertical spectra is -15% and +15% for the best estimate soil condition and by -10% and +10% to the lower-bound and upper-bound soil conditions. For the BEG, the in-structure response spectrum peaks were broadened by -15% and +15% for the three soil conditions and in the three directions. The peak-broadened spectra from the three soil conditions were then enveloped. At each floor, the response spectra at the center of mass, plus four corners of the building, were enveloped to conservatively account for the effect of torsion between the center of rigidity and the location of any equipment. Floor flexibility, where applicable, was accounted for in the generation of the vertical in-structure response spectra. Wall flexibility was also accounted for in the generation of the horizontal in-structure response spectra. The peak-broadened in-structure spectrum envelope was used for the seismic design of safety related equipment, components, and systems.

Acceleration Profile Envelope for Static Analysis of 3D FEM of Structure

An acceleration profile in each of the north-south, east-west, and vertical directions was developed from the 3D SSI analysis of the simplified finite element model. In each given direction, the acceleration profiles from the lower-bound, best-estimate, and upper-bound soil conditions are enveloped. The acceleration profile envelope was applied as a static load to the 3D FEM of the individual buildings for the design of structural elements.

Combination of Seismic Response Components

Two approaches were used to combine seismic loads in the three orthogonal directions for the building analysis and design. The first approach applies the equivalent accelerations for each level from the SSI analysis to the building FEM statically for each direction. This approach was used for the BMF and BEG. In this approach, the equivalent accelerations were determined to ensure that the resulting global structural forces (shear and axial forces) at each level matched those from the SSI analysis of the structure. The results from the equivalent static analyses, due to the equivalent accelerations applied in the three directions, were combined using the 100-40-40 Percent Rule (see Section 3.2.7.1.2 of ASCE Standard 4) to determine the resultant design earthquake loads on structural components. When combining forces and moments using the 100-40-40 Percent Rule, participation factors of 100% in the primary load direction and 40% in the other two directions were applied to the individual loads, as permitted by ASCE Standard 4.

The second approach applies the applicable seismic response spectrum to the base of the structural model. This approach was used for other structures. Each of the three directional components of the design earthquake produces responses in a structure in the three directions (i.e., three responses in the *x* direction, three in the *y* direction, and three in the *z* direction). Guidance provided by Regulatory Guide 1.92, and ASCE Standard 4 was used for combining modal responses and collinear responses from the three individual earthquake components. Modal responses due to each of the three individual earthquake components were combined using the SRSS method. Responses from modes that are clustered within a 10% frequency range were combined by the absolute sum method in accordance with Regulatory Guide 1.92. The remaining responses were then combined by the SRSS method. A sufficient number of modes were considered such that the accumulated modal mass exceeds 90% of the total mass in each direction. After modal responses were combined to obtain one set in each of the three orthogonal directions, the collinear responses due to contributions from the three earthquake components were combined by the SRSS method.

The following examples show formulas for determining the seismic load force in the *x* direction using the two methods:

100-40-40 Percent Rule
$\Sigma F_x = \Sigma 100\% \, F_{x \text{ due to } Ex'} + \Sigma 40\% \, F_{x \text{ due to } Ey'} + \Sigma 40\% \, F_{x \text{ due to } Ez}$

SRSS Method
$\Sigma F_x = (\Sigma F_{x \text{ due to } Ex'}{}^2 + \Sigma F_{x \text{ due to } Ey'}{}^2 + \Sigma F_{x \text{ due to } Ez'}{}^2)^{1/2}$

Seismic loads account for the mass inertia in the previously described seismic response spectra analysis using 100% of the dead load and a minimum of 25% of the live load of the structure. The amount of live load included in the seismic load contribution is based on the guidance provided in Section 3.1.4.2 of ASCE Standard 4 and the required functionality of the structure to support particular loading conditions. If a crane supports a load during a seismic event and remains functional, the full weight of the load is included in the live load that is used to determine the seismic load on the structure that supports the crane.

Dynamic Lateral Soil Pressure for Embedded Wall Design

The dynamic lateral soil pressures are determined based on the guidelines of ASCE Standard 4, Section 3.5.3.

Tornado Loads for SC-I Structures

Tornado loads (W_t) are those loads generated by the design basis tornado for the MFFF. They include tornado wind pressure loads (W_w), tornado-created differential pressure loads (W_p), and tornado-generated missile loads (W_m). See Section 1.3.3 for the definition of tornado loads and Table 1.1.2-1 for the tornado load summary. The three types of tornado loads on MFFF structures and facilities are defined in the following subsections.

Tornado Wind Pressure Loads

Tornado wind pressure loads (W_w) are those pressure loads generated by the tornado wind velocity, which is the combined translational and rotational wind speed (see Table 1.1.2-2). ASCE Standard 7, Section 6 was used to convert tornado wind velocity into effective structural pressure loads. DOE-STD-1020 is used to determine tornado loads for DOE facilities.

Tornado-Created Differential Pressure Loads

Tornado-created differential pressure loads (W_p) are those loads acting as an internal pressure loading on structures, caused by the negative pressure created by the tornado. See Table 1.1.2-2 for the definition of the design pressure drop and the rate of pressure drop. This internal pressure was applied to the interior surfaces of exterior building walls and roofs of structures requiring design against the effects of a tornado. Some reduction in this pressure differential was taken for structures that are vented, as permitted by NUREG-0800, Section 3.3.2, "Tornado Loadings."

Tornado-Generated Missile Loads

Tornado-generated missile loads (W_m) are impact loads applied to structures caused by strikes in accordance with the missile spectra criteria (see Table 1.1.2-2). The provisions of NUREG-0800, Section 3.5.3, Subsection II, ACI 349, Appendix C, and AISC N690 are used for determining the missile loading on various types of structures. ASCE Manual and Report No. 58 provide guidance on determining the tornado impact loads on buildings, structures and facilities.

Explosive Loads for SC-I Structures

Explosions could impact the BMF, BEG, UEF, tornado missile barriers, HVAC Intakes, or IROFS buried structures (i.e., buried conduit bank and buried pipes) by either inflicting structural damage or causing loss of control rooms habitability; either of which could potentially result in a release of hazardous material. Bounding external explosion hazard events and establishment of bounding overpressure consequences were determined. Also, results of SRS facility explosions, SRS transportation explosions, MFFF transportation hypothetical explosions, and BRP hypothetical explosions were used to determine the impact of the bounding explosion on the BMF, the BEG, the UEF, and other IROFS components.

Information and design requirements for impulsive-pressure waves associated with explosive loads are found in ASCE Publication 58-80, ASCE & SEI, ACI SP-175, and U.S. Department of the Army, the Navy, and the Air Force Manual TM 5-1300.

For the BMF structural steel and reinforced concrete building walls the ductility ratio beyond yield was derived in accordance with AISC N690 for structural steel and ACI-349, Appendix C for reinforced concrete.

Post-Earthquake Settlements

Settling of the foundation may occur from dissipation of excess soil pore pressure at depth due to the design earthquake. The surface manifestation of this settlement produces differential displacements within the building. The resulting forces within the structural members are referred to as post-earthquake settlement loads (S_{pe}). The post-earthquake settlements were calculated based on the method recommended by Ishihara & Yoshimine, *Evaluation of Settlements in Sand Deposits following Liquefaction during Earthquakes*.

Post-earthquake settlements calculated using the 1886 Charleston (50^{th} percentile) motion are greater than those calculated using PC-3+ motion (PC-3 motion scaled up by 1.25). Therefore, those calculated based on the Charleston motion were used for design of the SC-1 structures at the MFFF site.

The surface manifestation of these post-earthquake settlements was calculated using the FLAC Mohr-Coulomb Model. This method uses the FLAC Mohr-Coulomb model to evaluate the redistribution of settlements occurring at depth within the subsurface profile and the propagation of those displacements up through the soil profile to the ground surface.

Aircraft/Helicopter Impact for SC-I Structures

The aircraft screening analysis (including helicopters) was performed according to NUREG-0800, SRP 3.5.1.6, to determine the likelihood of an aircraft accident for the BMF, UEF, and BEG. The only aircraft event considered credible is helicopter impact. The effect of aircraft/helicopter event is determined in a two step process. First, it calculates the frequency of the building being hit by an aircraft and then analyzes the consequence of that hit. Even though the hit was credible, it is determined that the probability of accidents resulting in an unacceptable radiological consequence is less than 1.0E-07 meeting the acceptance criteria of NUREG-0800 Section 3.5.1.6 paragraph C. The possibility of an aircraft/helicopter impact resulting in BMF, BEG and UEF structure penetration is evaluated with the conclusion that the "probability of aircraft accidents resulting in an unacceptable radiological consequences is less than 1.0E-07." It is determined that the possibility of an aircraft/helicopter accident is highly unlikely during the life of the facility; thus, aircraft/helicopter impact is not evaluated as a design basis event. Potential acts of terrorism are covered by the MFFF Safeguards and Security program.

Range Fires for SC-I Structures

Protection of SC-I, IROFS buildings from exterior exposure fires is in accordance with industry standards. A compliance verification review in accordance with NFPA 80A concluded that the hazards presented by range fires to the MFFF are minimized by the site layout, and the design of SC-I structures is not adversely impacted by the effects of an external fire. Thus, range fires are not a credible design basis event for these structures.

1.1.2.1.6.4.2 Structural Design Loading Combinations for SC-I Structures

The following loads are addressed in the loading combinations used for the design of SC-I structures at the MFFF:

D	=	dead load
L	=	live load
F	=	hydrostatic fluid pressure load

H	=	lateral soil pressure load
T_o	=	thermal load
R_o	=	pipe, HVAC duct, conduit, and cable tray reaction load
W	=	wind load
E'	=	design earthquake seismic load
S_{PE}	=	post-earthquake settlement
W_t	=	tornado loads including:
		W_w = tornado wind pressure load
		W_p = tornado-created differential pressure load
		W_m = tornado-generated missile load
T_a	=	thermal load (due to postulated break and including T_o)
R_a	=	pipe reaction (due to postulated break under thermal condition and including R_o).

Loading combinations for the design of SC-I structures were determined using NUREG-0800, Section 3.8.4, as a guide. Since there are no operating basis earthquake loads (E), flood loads (F'), compartmental pressure loads (P_a), or high energy pipe break accident loads (Y_r, Y_j, or Y_m) applicable to the MFFF, these loads, although specified in NUREG-0800, Section 3.8.4, are not included in above list of loads or the loading combinations that follow. The following definitions apply to terms used in the loading combinations specified in this section:

- For concrete structures, "U" is the section strength required to resist design loads based upon the ultimate strength design methods described in ACI 349, (SC-I).
- For steel structures, "S" is the required section strength based on elastic design methods and the allowable stresses defined in Part 1 of AISC N690, (SC-I).
- For steel structures, "Y" is the section strength required to resist design loads based on the plastic design methods defined in Part 2 of AISC N690 (SC-I).

Acceptance criteria for the allowable limits of structural members being designed are factored into the loading combinations that follow. These acceptance criteria are in accordance with NUREG-0800, Sections 3.8.4, Subsection II.5a and b. The criteria have been applied to the section strength parameters (U, S, and Y) to which each loading combination has been defined as equivalent.

1.1.2.1.6.4.2.1 Loading Combinations for SC-I Concrete Structures

The following loading combinations are used for the design of SC-I concrete structures. These loading combinations are used in conjunction with the ultimate strength design method for concrete design. Two conditions of structural loading were considered: (1) service loading conditions, and (2) extreme loading conditions.

Service Loading Combinations for SC-I Concrete Structures

Service loading combinations represent the loading conditions that SC-I structures are expected to experience during normal facility operations and during severe environmental conditions. Loads included in the service loading combinations are dead loads, live loads, hydrostatic fluid pressure loads, lateral soil pressure loads, design wind loads, flood loads, thermal loads, and reaction loads (pipe, HVAC, and/or cable tray). No seismic loads are included in the MFFF service loading combinations.

SC-I concrete structures are designed for the following service loading combinations:

$U = 1.4D + 1.4F + 1.7L + 1.7H$
$U = 1.4D + 1.4F + 1.7L + 1.7H + 1.7W$
$U = 1.2D + 1.2F + 1.7W$

If thermal stresses caused by T_o and/or R_o are present on the structure, the following loading combinations are also considered:

$U = 1.05D + 1.05F + 1.275L + 1.275H + 1.275T_o + 1.275R_o$
$U = 1.05D + 1.05F + 1.275L + 1.275H + 1.275W + 1.275T_o + 1.275R_o$

Extreme Loading Combinations for SC-I Concrete Structures

Extreme loading combinations represent the loading conditions that SC-I structures could experience under extreme environmental conditions. Loads included in the extreme loading combinations are dead loads, live loads, thermal loads, reaction loads (pipe, HVAC, and cable tray), design earthquake seismic loads, tornado loads, and flood loads. Extreme environmental loads (i.e., seismic and tornado loadings) are not considered to act simultaneously.

SC-I concrete structures are designed for the following extreme loading combinations:

$U = D + F + L + H + T_o + R_o + E'$
$U = D + F + L + H + T_o + R_o + W_t$ (see Note 1 below)
$U = D + F + L + H + E' + T_a + R_a$ (see Note 2 below)
$U = D + F + L + H + T_o + R_o + S_{PE}$

Note 1: In accordance with NUREG-0800, Section 3.3.2, Subsection II.3.d, the following combinations of W_t are considered:

$W_t = W_w$
$W_t = W_p$
$W_t = W_m$
$W_t = W_w + 0.5W_p$
$W_t = W_w + W_m$
$W_t = W_w + 0.5W_p + W_m$
Note 2: $T_a = T_o$ and $R_a = R_o$, since pipe break accident loads are not applicable.

1.1.2.1.6.4.2.2 Loading Combinations for SC-I Steel Structures

The following loading combinations are used for the design of SC-I steel structures. Applicable combinations are given for designs that utilize either elastic working stress design methods or plastic design methods. In each case, loading combinations are provided for service loading conditions and for extreme loading conditions.

Service Loading Combinations for SC-I Steel Structures

Service loading combinations for SC-I steel structures encompass the same type loads as included for service loading combinations for SC-I concrete structures in Section 1.1.2.1.6.4.2.1.

Service Loading Combinations for Elastic Working Stress Design

If elastic working stress design methods are used, SC-I steel structures are designed for the following service loading combinations:

$S = D + F + L + H$
$S = D + F + L + H + W$

If stresses due to T_o and/or R_o are present on the structure, the following loading combinations are also considered:

$(1.5)S = D + F + L + H + T_o + R_o$ (tension members)
$(1.3)S = D + F + L + H + T_o + R_o$ (compression members)
$(1.5)S = D + F + L + H + T_o + R_o + W$ (tension members)
$(1.3)S = D + F + L + H + T_o + R_o + W$ (compression members)

Service Loading Combinations for Plastic Design

If plastic design methods are used, SC-I steel structures are designed for the following service loading combinations:

$Y = 1.7D + 1.7F + 1.7L + 1.7H$
$Y = 1.7D + 1.7F + 1.7L + 1.7H + 1.7W$

If stresses due to T_o and/or R_o are present on the structure, the following loading combinations are also considered:

$Y = 1.3D + 1.3F + 1.3L + 1.3\ H + 1.3T_o + 1.3R_o$
$Y = 1.3D + 1.3F + 1.3L + 1.3\ H + 1.3T_o + 1.3R_o + 1.3W$

Extreme Loading Combinations for SC-I Steel Structures

Extreme loading combinations for SC-I steel structures encompass the same type loads as included for extreme loading combinations for SC-I concrete structures in Section 1.1.2.1.6.4.2.1.

Extreme Loading Combinations for Elastic Working Stress Design

If elastic working stress design methods are used, SC-I steel structures are designed for the following extreme loading combinations:

$(1.6)S = D + F + L + H + T_o + R_o + E'$ (tension members)
$(1.4)S = D + F + L + H + T_o + R_o + E'$ (compression members)

$(1.6)S = D + F + L + H + T_o + R_o + W_t$ (tension members) (see note below)
$(1.4)S = D + F + L + H + T_o + R_o + W_t$ (compression members) (see note below)
$(1.7)S = D + F + L + H + E' + T_a + R_a$ ($T_a = T_o$ and $R_a = R_o$) (tension members)
$(1.6)S = D + F + L + H + E' + T_a + R_a$ ($T_a = T_o$ and $R_a = R_o$) (compression members)

Note: The six subloading combinations for tornado loads W_t that are specified in Section 1.1.2.1.6.4.2.1 are also considered in this loading combination.

Extreme Loading Combinations for Plastic Design

If plastic design methods are used, SC-I steel structures are designed for the following extreme loading combinations, as defined in NUREG-0800, Section 3.8.4, Subsection II.5:

$Y = D + F + L + H + T_o + R_o + E'$
$Y = D + F + L + H + T_o + R_o + W_t$ (consider the subloading combinations of W_t)
$Y = D + F + L + H + E' + T_a + R_a$ ($T_a = T_o$ and $R_a = R_o$)

1.1.2.1.6.4.3 Loading Combinations for SC-I Structures for Overturning, Sliding, and Flotation

Specific loading combinations were checked to ensure the overall stability of structures against the effects of overturning, sliding, and flotation. These loading combinations were determined using NUREG-0800, Section 3.8.5, Subsections II.3 and II.5, as guides. Minimum factors of safety have been satisfied for each stability condition considered (see Table 1.1.2-3 for results).

1.1.2.1.6.4.4 Applicability of Loads

The following requirements were considered when determining applicable loading combinations for the design of the MFFF structures:

- Live loads are applied fully, partially, totally removed from the members, or shifted in location and pattern as necessary to obtain the worst-case loading conditions for maximizing internal forces and moments. Impact forces caused by moving loads are applied where appropriate.
- Appropriate construction loads are considered in the service loading combinations. Construction methods and sequences are also considered and appropriate loading conditions applied to ensure the structural integrity of partially erected or open structures.
- Where a load reduces the overall loading on a structural member, a load coefficient of 0.9 was applied to that load component in the loading combination. The reducing coefficient was only used for loads that are always present or that always occur simultaneously with other loads. Otherwise, the coefficient for that load was taken conservatively as zero.
- Tornado loads are applied to roofs and the exterior walls of SC-I structures. The tornado differential pressure boundary is established by installation of tornado dampers at the ventilation openings, unless there are also other unprotected openings that require them. Where the exterior walls have unprotected openings without tornado dampers, appropriate interior walls are designed as the tornado pressure boundary.

1.1.2.1.6.5 Analysis Results for SC-I Structures

This section presents analysis description and the results of Seismic SC-I Concrete and Steel Structures.

1.1.2.1.6.5.1 SC-I Concrete Structures

MOX Fuel Fabrication Building (BMF)

Soil Structure Interaction (SSI)

Determination of Maximum Structural Responses:

Maximum structural responses such as the floor accelerations are extracted from the SSI analyses for the structural design of the building (see Section 1.1.2.1.6.4.1.3).

The worst case maximum structural responses including the floor accelerations and forces in the tie-back beams are extracted. See Table 1.1.2-14 for lists of the seismic accelerations at different elevations of the BMF structures.

The maximum floor accelerations are used as seismic load input to the finite element static analysis of the structure to determine the design seismic loads in the various structural elements. The maximum accelerations at several nodal locations on the same floor elevation are typically enveloped to account for the variability in the acceleration inputs to the static structural analysis.

Generation of Floor Response Spectra:

The in-plane flexibility of the slabs in BMP and BAP amplify the horizontal responses within certain zones on the slabs. See Figure 1.1.2-61 and Figure 1.1.2-62 for the boundary of each zone on the slab and the locations (nodes) at which the responses are considered to be representative of the given zone.

See Figure 1.1.2-63 to Figure 1.1.2-65 for the 5% damping response spectra at both the roof and slab at grade of BMF from the three soil cases in the X, Y and Z direction, respectively. Each spectrum envelops the motions at the four building corners and the building center. The results show that the upper bound soil case (see Figure 1.1.2-66 and Figure 1.1.2-67) compares the 5% damping response spectrum at the four building corners of the roof to that at the building center. The difference in spectrum is small in both horizontal directions, but substantial in the vertical direction. This indicates that the torsional response of the building is negligibly small while the rocking response of the building, due to SSI, is significant. See Figure 1.1.2-69 for a comparison of the 5% damping X-direction FRS at nodes F and K on the roof (representing zone BMPX-1) with that outside the zone. This confirms the amplification effect of the in-plane flexibility of the slab on the X-direction FRS. See Figure 1.1.2-70, which similarly confirms the amplification effect of the in-plane flexibility of the roof slab on the Y-direction FRS within zone BMPY-1 (represented by FRS at nodes M and G) and BAPY-1 (represented by FRS at node H), respectively. Similar amplification occurs at other floor elevations.

Three directional in-structure response spectra and the amplified in-structure response spectra are generated considering the three earthquake components (see Section 1.1.2.1.6.4.1.3 for the method description). The response spectra for 2%, 3%, 4%, 5%, 7%, and 10% damping are calculated from the nodal response acceleration time histories. For each soil case, the individual nodal response spectra at the same floor elevation are enveloped and broadened to account for the effect of the potential uncertainty in the material properties, SSI modeling and analysis techniques. The amount of spectrum broadening varies depending on the soil case. Separate horizontal response spectra are calculated at locations on the slab where the effect of in-plane flexibility of the slab is found significant. Corresponding broadened floor spectrum envelopes from the three soil cases are then enveloped for use in the seismic design of systems and components.

See Figure 1.1.2-72 through Figure 1.1.2-77 for a diagram of the 2%, 3%, 4%, 5%, 7%, and 10% damping typical FRS envelopes at roof Elevation (El). 73’0” for the three soil cases. The vertical floor spectrum envelopes are sufficient for use in the seismic design of systems and components that are supported on the structural walls and rigid slab panels. The vertical FRS envelopes are generated for each flexible slab group and peaks of spectra are broadened. In addition, for each given flexible slab group FRS envelopes are generated separately both near the center of the building and near the edges of the building. See Figure 1.1.2-78 to Figure 1.1.2-80 for a diagram of typical vertical FRS envelopes at the roof for the flexible slab group 2 and 4 (see Tables 1.1.2-9 and 1.1.2-10 for definition).

Horizontal Spectra for Flexible Walls:

Similar to the generation of the vertical spectra for flexible slabs, the generation of the horizontal spectra for flexible walls in the out-of-plane direction is based on the response acceleration time history from the analysis of the 1-DOF system representing the wall in uncracked condition. See Figure 1.1.2-82 through Figure 1.1.2-83 for a diagram of the 2%, 3%, 4%, 5%, 7%, and 10% damping typical FRS envelopes at the roof for the NS flexible wall group 3 and EW flexible wall group 1 (see Tables 1.1.2-12 and 1.1.2-13 for definition).

<u>Foundations</u>

Foundation Preparation:

The BMF and the adjacent BEG are monolithic, reinforced concrete structures. The BMF is founded on a 6.5 ft thick reinforced concrete mat, and the BEG is founded on a 3-ft thick reinforced concrete mat. Seven and one half (7.5) ft of the natural soils beneath the BMP floor and 2.5 ft of the natural soils beneath the BAP and BSR basement floors and the BEG floor are excavated and replaced with engineered select structural fill. The structural fill provides a firm, uniform foundation bearing material for the high static and dynamic loads applicable for these structures. This helps distribute concentrated static and earthquake edge pressures into the underlying sub grade and minimize effects of potential differential settlement.

Average gross static bearing pressures beneath the lower floors of the BMF structure are calculated to be approximately 6.11 ksf in the BMP area, 6.80 ksf in the BSR area, and 7.93 ksf in the BAP area. Bearing pressures beneath the outer security wall system are calculated to be approximately 8.64 ksf for the walls adjacent to the main building floors and approximately 11.22 ksf for the walls adjacent to the BAP and BSR basement floors. The foundation pressure beneath the BEG is calculated to be 2 ksf. The pressures are based on the weight of the structure and other long-term loads within the structure.

Bearing Capacity:

The ultimate bearing capacity of the soils underlying the BMF structure exceeds 70 ksf, based on conservative soil parameters and groundwater levels. This results in a factor of safety against bearing capacity failure of 11.7 for this structure.

The ultimate bearing capacity of the soils underlying the BEG structure exceeds 13 ksf. The factor of safety against bearing capacity failure of the BEG structure for the average foundation pressure of 2 ksf is 6.3.

Settlements of SC-I Structures

General:

The FLAC computer program was used to evaluate the potential effect of variations in structure properties (E and I), soft zone and soft material parameters (C_c, ϕ, overconsolidation ratio), and engineering unit parameters (preconsolidation pressures, compression indices, ϕ) on model results. It is also used to provide a detailed settlement analysis and deformation profile for the BMF, BEG and UEF. The settlements calculated using FLAC were shown to be consistent with results of conventional settlement analysis, considering the accuracy of the estimating techniques of each method. The results of the parametric evaluations showed the calculated settlements were not significantly affected by variations likely to occur in structure properties and material parameters.

Calculated Settlement and Pressure:

The FLAC Cam-Clay model is used to predict BMF and BEG settlements and bearing pressures due to static loads along each of 6 geotechnical sections. Normally consolidated soft zone and soft material layers are included in the analysis, and the resulting settlements varied from 2.0 - 2.8 in under the BMF and from 1.1 - 1.8 in under the BEG. Based on settlement measurements of comparable structures existing at the site, approximately 1/2 in of secondary compression settlement is expected to occur, resulting in total (primary plus secondary compression) calculated settlements of the BMF structure ranging from approximately 2.5 - 3.3 in.

The results of the FLAC analysis along each of the sections are used to develop contours of calculated settlement and pressures beneath the MFFF and BEG. The magnitudes and patterns of settlements and ground pressures indicated by FLAC are typical of those for stiff structures founded on subsurface soils such as those at the MFFF site. The initially applied pressures from the structures to the soils are redistributed through the stiff structures, resulting in lower pressures under interior floors and greater pressures under the exterior walls. Large differential settlements are not predicted, and the largest settlements occurred over areas underlain by the thicker layers of more compressible materials (soft zone and soft material layers).

Most of the calculated settlement of the BMF structure is expected to occur shortly after the foundation loads are applied. On the basis of the settlement records of existing structures at the site, it is expected that 60 to 80% of the calculated settlement occurs by the end of construction and, thereafter, primary consolidation continue for 6 to 18 months and secondary compression continue for approximately 5 to 10 years.

Coefficients of subgrade reaction (soil spring constants) are calculated for use in structural modeling (ANSYS) of the BMF, BEG and UEF to approximate the stress-strain response of the foundation soils to static structure loads. The estimates are made by dividing the calculated pressures by the calculated settlements. The coefficients of subgrade reaction, as defined here, include elastic compression and nonelastic consolidation of the entire soil mass beneath the structures at specific structure pressures. As such, the calculated coefficients represent elastic springs and the magnitude of settlement indicated at a particular location within the structural model is proportional to the pressure applied to the spring at that location. The coefficients of subgrade reaction are "best-estimate" values based on the combinations of applied static pressures, structure properties, and subsoil properties used in the FLAC models.

Tornado Missile Barrier Analysis and Design

See Table 1.1.2-2 for lists of the dimensions, mass, maximum height and design velocity of the postulated tornado generated missiles.

Design Results:

The minimum thickness of concrete for the roof and walls required to protect against tornado generated penetration, perforation and scabbing are calculated. The thickness of concrete for the roof and walls provided in the design exceed the minimum requirement. See Table 1.1.2-4 for lists of the minimum thicknesses required to prevent penetration, perforation and scabbing for the BMF.

The walls and the roof slab of the BMF are reinforced such that overall failure of the wall and roof panels, as well as local penetration due to the postulated tornado generated missiles is precluded. The outer security wall and roof of BMF, including gabion stone material (see Figure 1.1.2-112 and Figure 1.1.2-113), are conservatively not considered in the analysis of the BMF. Much of the impact energy would be absorbed by these features.

External Explosion Analysis and Design

The BMF structure is analyzed for the effect of an accidental external explosion (see Section 1.1.2.1.6.4.1.3 for description).

Design Results:

The BMF analysis and design determines the adequacy of the elements of various thicknesses with predetermined amounts of reinforcement to withstand a design explosive charge. The structure is analyzed and designed for both local (panel moments and shears) and global (sliding and over-turning) effects.

The walls and roof are checked for local effects of blast pressure. The pressure due to explosion is checked against a slab collapse pressure (capacity). Reaction shear at slab edges is also checked. The pressure due to explosion and reaction shear is found to be acceptable.

Sliding and overturning are not an issue with the BMF due to the magnitude of the explosion forces and the fact that it is embedded below grade.

The walls and the roof slab of the BMF are reinforced such that there is no overall failure of the panels due to an external explosion.

<u>Structural Analysis</u>

The structural analysis of the BMF is performed using the finite element computer program ANSYS, R5.6. BMF concrete outline drawings are used as design input data to ANSYS model of the geometry of the BMF building.

The detailed analytical model of the BMF building is developed using shell elements to model the walls and slabs. Beam elements are used to model floor beams, tie-back beams and columns. The model reflects the principal structural features of the building, consistent with the objective of determining the forces on the building elements.

See Figure 1.1.2-94 for an illustration of the ANSYS Finite Element Model used in the overall BMF analysis.

ANSYS "SHELL63", linear elastic shell elements are used to model the slab-on-grade foundation using their elastic foundation stiffness capability. ANSYS "SHELL43" elements, a plastic large strain shell option, are used to model the walls and slabs to obtain the out of plane shear forces. "BEAM4" elements are used to model concrete beams, columns and the steel tie-back beams. "BEAM44" elements are used to model rigid links, and "MASS21" elements are used for the key points with mass.

The major openings in elevated slab areas are modeled in the ANSYS analysis. The major openings or cutout areas in the walls (for HVAC, piping, electrical, cable trays, etc.) are not explicitly modeled in the analysis. To account for the cutout areas' effect on the element forces, equivalent stiffness and area for walls with opening are calculated. Young's modulus and mass for each panel are then adjusted. In addition to the reduction of Young's modulus and mass, the effective stress intensification for in-plane shear force is tabulated for use during the detail wall design.

Five ANSYS models (three seismic and two static) are created to perform the analysis for the loading conditions and combinations. These five models have essentially the same geometry except for the material property input for density. In the model for the seismic loading condition, the density is adjusted to include the 25% live load for mass calculation for the slabs. A uniform seismic acceleration input is used. The masses are multiplied by the ratio, based on the Soil Structure Interaction Analysis, to account for the variable seismic accelerations profile.

This analysis is intended to generate the results for the reinforced concrete and steel tie-back beam design. The wind and tornado loads are included in this analysis along with static and seismic loads. The lateral gabion stone pressure loads, static soil pressure, and seismic induced soil pressure loads are considered in the ANSYS analysis.

Loads:

See Section 1.1.2.1.6.4.1 for a description of the design loads associated with the BMF.

Loading Combinations:

See Section 1.1.2.1.6.4.2 for the loading combinations used for SC-I Concrete Structures.

Results:

ANSYS analysis results provide out-of-plane shear forces Nx & Ny, axial forces Tx & Ty, in-plane shear force Txy, orthogonal bending moments Mx & My, and twisting moment Mxy for each shell element of the model for the load cases. ANSYS results for individual load combinations are obtained from the Post-Process files. For design of structural elements to determine the reinforcement for foundation the mat, walls, elevated slabs, roof slab etc., appropriate post-process files are evaluated using three Visual Basic/Excel spreadsheets.

Maximum and minimum (maximum negative) forces in each category (N_x, N_y, M_x, M_y, T_x, T_y, T_{xy}, and M_{xy}) are determined for each ANSYS area and load combination by post processing the structural analysis output. This data is used for designing the individual structural elements.

The structural elements are designed for the maximum/minimum loadings using the ultimate strength methodology and building code requirement delineated in ACI 349-97. The reinforcing provided satisfies the strength requirements for all loadings and load combinations.

The SC-I concrete structures are designed utilizing 4000 psi compressive strength of concrete and 60 ksi yield strength for the reinforcing steel. See Figure 1.1.2-96 through Figure 1.1.2-99 and Figure 1.1.2-103 through Figure 1.1.2-110 for representative concrete and reinforcing details.

Emergency Generator Building (BEG)

Soil Structure Interaction (SSI)

The BEG is an above ground structure founded on soils. The SSI analysis is similar to that for the BMF (see Section.1.1.2.1.6.5.1.2).

Foundations

The BEG foundation is considered as a monolithic slab with a uniform soil spring of 13.5 pounds per cubic inch (pci)for dead loads and 27 pci for seismic, wind, tornado and live loads.

Tornado Missile Barrier Analysis and Design

See BMF Section 1.1.2.1.6.5.1 for a description of the tornado missile impact analysis. The walls and the roof slab of the BEG are reinforced such that overall failure of the wall and roof panels, as well as local penetration due to the postulated tornado generated missiles is precluded.

The BEG includes removable wall panels to allow access to the generator rooms and the subsequent removal of large pieces of equipment. The design of these removable panels is controlled by the tornado missile impact forces of an automobile.

Results:

The structural analysis is based on criteria specified in the Basis of Design for Structures and consists of finite element analysis performed using ANSYS structural analysis software.

The ANSYS computer analysis results are used to design the primary structural reinforced concrete elements (foundation, walls and roof slab) of the BEG. The design and evaluation of the concrete structure conforms to the requirements of ACI 349-97. The reinforcing provided satisfies the strength requirements for all loadings and load combinations.

External Explosion Analysis and Design:

See BMF Section 1.1.2.1.6.5.1 for a description of the explosion analysis and design. The design and analysis determine the adequacy of the elements of various thicknesses to withstand a design explosive charge. The BEG structure is designed and analyzed for both local (walls and roof) and global (sliding and over-turning) effects. The thicknesses and reinforcement of the structural elements, (walls and slabs) are adequate to withstand the design explosive charge. The over pressures of the blast are insufficient to cause either sliding or over-turning of the structure.

See Figure 1.1.2-114 through Figure 1.1.2-115 for representative concrete and reinforcing details.

Emergency Fuel Storage Vault (UEF)

Soil Structure Interaction (SSI)

The UEF is primarily a below ground structure founded on soils. The effect of SSI is negligible and the structural acceleration is taken to be the same as that of the free field ground motion.

Foundations

The UEF foundation is considered as a monolithic slab.

Results:

The structural analysis is based on criteria specified in the Basis of Design for Structures and consists of finite element analysis performed using ANSYS structural analysis software.

The ANSYS computer analysis results are used to design the primary structural reinforced concrete elements (foundation, walls and roof slab) of the UEF. The design and evaluation of the concrete structure conforms to the requirements of ACI 349-97. The reinforcing provided satisfies the strength requirements for all loadings and load combinations.

Tornado Missile Barrier Analysis and Design:

See BMF Section 1.1.2.1.6.5.1 for a description of the tornado missile impact analysis. The walls and the roof slab of the UEF are reinforced such that overall failure of the wall and roof panels, as well as local penetration due to the postulated tornado generated missiles, is precluded.

External Explosion Analysis and Design

See BMF Section 1.1.2.1.6.5.1 for a description of the explosion analysis and design. The analysis and design determines the adequacy of the elements of various thicknesses to withstand a design explosive charge. The UEF structure is designed and analyzed for both local, (walls and roof) and global, (sliding and over-turning) affects. The thicknesses and reinforcement of the structural elements (walls and slabs) are adequate to withstand the design explosive charge.

See Figure 1.1.2-118 through Figure 1.1.2-121 for representative concrete and reinforcing details.

1.1.2.1.6.5.2 SC-I Steel Structures

This section provides analysis results of other miscellaneous SC-I steel structures located within and exterior to the main concrete structures.

HVAC Penetrations (BMP)

The HVAC main intake and discharge penetrations are located on the third level of BMP, and extend away from the outside wall to the security wall. To prevent access through them, a security barrier is provided.

Missile barriers are provided to prevent design tornado generated missiles from entering the facility. The penetrations have tornado dampers.

HVAC Penetrations (BSR)

The intake and discharge penetrations are located in the exterior and security walls of the BSR. The penetrations are welded to embedded plates located in the building and the outside wall of the BSR and they have studs that are cast in place into the structural walls. They have missile barriers, security bars and tornado dampers.

HVAC Penetrations (BSH)

The intake and discharge penetrations are similar to those indicated in the BSR described above.

HVAC Penetrations (BEG & UEF)

The intake and discharge penetrations are located in the walls of the BEG & UEF. They have square and rectangular shapes that vary in size. The penetrations are protected from tornado generated missiles. The switchgear rooms in the BEG building have been provided with tornado dampers. The HVAC penetrations consist simply of openings in the walls. The openings are framed by embedded plates that serve to reinforce the opening and to allow for the attachment of equipment.

Embedded Plate in BMF Ceiling

The embedded plates are installed in the ceiling at elevation 70'0" in the BMF. The embedded plates are 3/4-in thick and are installed wall to wall with 7-1/2 ft wide and maximum 40-ft long steel plate panels to optimize shipping and installation requirements. See Figure 1.1.2-92 for diagram of the ceiling embedded plates.

Nelson deformed bar anchors are welded at 9-in spacing in accordance with manufacturer instructions. The plate sections are shop welded to suit room dimensions where possible. The entire ceiling area enclosed by the walls is covered by the ceiling plates, which are to be used as form work for pouring the concrete. Temporary supports are provided to avoid initial stress in the plates until the concrete hardens.

The embedded plates are used for attaching supports for HVAC ducts, cable trays and piping. A typical plate section 3/4 in x 45 in x 90 in is chosen as representative of the ceiling plate to study the behavior of the ceiling plate due to loads of the supported systems. Based on a review of the typical layouts of the HVAC ducts, cable trays and piping, various support configurations are postulated. The critical loads along global X, Y and Z directions are evaluated based on standardized support loads developed specifically for the BMF area. The support reaction loads of the suspended systems are not to exceed the design load carrying capacity of the embedded plates without special consideration or additional analysis. The design of the anchors (size and embedment depth) and embedment geometry (anchor spacing and edge distance) is in accordance with ACI 349-01, Appendix B. The design of the plates is in accordance with ACI 349-01.

Strip Plates (BAP, BMP, BSR, BEG, and UEF)

Three different sizes of carbon and stainless steel strip plates are embedded in the concrete slabs, walls, columns, and beams of SC-I and SC-II structures to facilitate supporting suspended systems such as pipe hangers, conduits, ducts and cable trays. The embedded plates are the primary means of attachment to the concrete members. The ANSYS computer code is used to qualify the plates and anchors for the loads. The plates and studs are analyzed for various calculated loads of the supported commodities with different forces and moments considered in the three orthogonal directions. Loads are applied by placing various attachment types/sizes at different plate locations to generate a 'worst-case' scenario to qualify the plates for several loading conditions, or vectors (i.e., high tension, high tension and shear, high biaxial moment, etc.).

The design of the anchors (size and embedment depth) and embedment geometry (anchor spacing and edge distance) is in accordance with ACI 349-01, Appendix B. The design of the plates is in accordance with ACI 349-01. The allowable attachment design loads and minimum attachment spacing on the plates are tabulated for use by other disciplines to attach supports to these plates. The support reaction loads of the suspended systems are not to exceed the design load carrying capacity of the embedded plates without special consideration or additional analysis. See Figure 1.1.2-90 for diagram of the strip plates.

Dedicated Plates (BAP, BMP, BSR, BEG, and UEF)

Various sizes of carbon and stainless steel 'dedicated plates' (i.e., plates that are placed as-needed and generally have one attachment) are embedded in the concrete slabs, walls, columns, and beams of SC-I and SC-II structures to facilitate supporting structures and equipment throughout the plant. Embedded plates are the primary means of equipment attachment to concrete members.

A set of standard dedicated embedded plates with welded headed studs are qualified for standard (anticipated) load sets with different forces and moments considered in the three orthogonal directions. Various attachment types/sizes are analyzed at different locations on the plates to generate a 'worst-case' scenario to qualify the embedment for several loading conditions, or vectors (i.e., high tension, high tension and shear, high biaxial moment, etc.). The load vectors produced are intended to maximize the embedment for the given conditions. The ANSYS computer code is used to qualify the plates and the anchors for these loads and conditions.

The design of the plate and anchors (size and embedment depth) and embedment geometry (anchor spacing and edge distance) is in accordance with ACI 349-01, Appendix B. The design of the plates is in accordance with ACI 349-01. The allowable (attachment) design loads and minimum plate-to-plate spacing (or spacing to other types of embedment) are tabulated for use by other disciplines. The support reaction loads of the structures or equipment are not to exceed the design load carrying capacity of the embedded plates without special consideration or additional analysis. See Figure 1.1.2-90 for diagram of the dedicated plates.

Deformed Bar Embedded Plates (BAP, BMP, BSR, BEG, and UEF)

Various sizes of carbon steel 'deformed bar plates' are embedded in the concrete slabs, walls, columns, and beams of SC-I and SC-II structures to facilitate supporting structures and equipment (especially gloveboxes) throughout the plant.

A set of standard embedded plates with welded deformed bars are qualified for anticipated load sets with different forces and moments considered in the three orthogonal directions. Selected (single) attachment types/sizes are analyzed to qualify the embedment for a set of loads that were enveloped from available glovebox reaction loads with the analysis considering close proximity to other types of embedment. The plates and the bars are qualified by manual methods for these loads and conditions.

The design of the plates and deformed bars (size and development length) and embedment geometry (bar spacing and edge distance) is in accordance with ACI 349-97 (main body). The allowable attachment design loads and minimum plate-to-plate spacing (or spacing to other types of embedment) are tabulated for use by other disciplines. The support reaction loads of the structures or equipment are not to exceed the design load carrying capacity of the embedded plates without special consideration or additional analysis. See Figure 1.1.2-90 for diagram of the deformed bar embedded plates.

Structural Steel Platforms

Structural steel platforms are located in various areas of the buildings and provide access to gloveboxes, mechanical equipment, electrical equipment and other equipment. The platforms are considered IROFS in the event that equipment, conduits or utilities need to be supported by the platform structure. The platform members are analyzed and designed by performing a static analysis and a 3D dynamic (response spectra) analysis using the GTStrudl program. In addition to dead and live loads, additional loads are input in the analysis in order to account for equipment and supports that may be attached to the platform. The design of members and connections is per ANSI/AISC N690.

Structural Steel for Process Handling Devices

Process cranes and monorails are designed to handle glovebox material. Some of the glovebox processes that are used include plutonium oxide (PuO_2) storage, PuO_2 receiving, rod handling and fuel assembly storage, and fuel assembly packaging areas.

The cranes and monorails are electrically operated and have load cycles ranging from 5 to 200 per day. The duty service class ranges from A to D.

The process cranes and monorails will be used during the normal operation of the plant. Therefore, in order to provide an additional margin of safety, the lift load was added to the self weight of the cranes and monorails for the seismic analysis and design. The design of members and connections is per ANSI/AISC N690.

Laboratory Ceilings (BMP)

Based on the “Fire Hazards Analysis for the Mixed Oxide Fuel Fabrication Facility” and “Fire Damper Reduction Study,” the laboratory ceilings in BMF are designed to confine fire. The fire resistance rating of these rooms is identified and the adequacy of the structural systems is substantiated for 2-hour or 3-hour fire confinement. The 2-hour and 3-hour Fire Rated systems are similar to the Veterans Administration approved design documented in The National Institute of Standards and Technology Report NBSIR 85-3158. The design is based on the 2-hour rated UL Design P676 Alternate Construction. For the 3-hour rated system, thicker planks are used. The gypsum plank and bulb tee deck is supported from the beam’s bottom flange and the 5/8-in thick layer of gypsum board is moved below the beam as a membrane similar to 3-hour rated UL Design G512. The ceiling and its support steel framing is supported from the steel plates embedded in the concrete walls of the rooms. The ceiling steel structural support system is analyzed per Seismic Category SC-I. See Figure 1.1.2-86 for diagram of typical detail of the 2-hour and 3-hour system. The design of members and connections is per ANSI/AISC N690.

Process Cell Drip Trays

See Figure 1.1.2-87 for diagram of a typical process cell drip tray system. It is made of a stainless steel liner, a drain channel and a sump. The liner and the drain channel are sloped so that the leak can flow to the sump equipped with detection. Emptying pipes are also installed in the sump. Grout or borated concrete (for the subcritical drip trays) is installed between the liner and the room walls and floor.

Subcritical drip trays are designed to receive fissile solutions. The risk of criticality is made highly unlikely due to:

- The use of borated concrete below the drip tray liners which reduces reflection
- The installation of narrow drain channels
- The reduction of the slope of liner if necessary.

Borated concrete is made of cement, borated aggregates and water. Borated aggregates contain more than 12 weight percent (wt %) boron, and more than 18 wt % bonded water. Boron and hydrogen act as neutron absorbers.

Narrow drain channels make the geometry less reactive. A reduced slope decreases the fissile liquid height in case of leakage.

Concrete containment or drip containment is provided in some of the rooms in the BAP to contain the volume of the spilled solution due to failure of the fluid containers. Passive design features such as curbs, doorsills and drain system are provided in many rooms to contain the volume of the spilled solution. These passive design features (IROFS) are not credible to fail under the conditions for which they are designed to function. The floor of these containments is covered with a chemical resistant coating system. A sump is provided to collect and drain the spilled liquid.

Similarly, doorsills are provided at the entry doors to process rooms to prevent water from spills or fire suppression discharge spray from entering the process room. Such spills are collected in drains outside the process rooms and collected in a holding tank inside the BMF.

1.1.2.1.7 Design Basis for Non IROFS

1.1.2.1.7.1 Functions of SC-II Structures

SC-II buildings, structures and components have to maintain structural integrity during the design earthquake to avoid adverse impact on IROFS.

1.1.2.1.7.2 Requirements for SC-II Structures

SC-II structures are analyzed for the loads and loading combinations (see Section 1.1.2.1.6.4). Appropriate consideration is given to the load distribution on the structure (e.g., point loads, uniformly distributed loads, or varying distribution of loads) and the end restraint conditions applicable for the structural component being considered.

Analyses were performed using equivalent static loads with appropriate consideration of impact effects for moving loads as specified for the particular loads (see Section 1.1.2.1.6.4.1 for description).

1.1.2.1.7.3 Seismic Analysis Requirements for SC-II Structures

The seismic analysis requirements for SC-II structures are the same as specified in Section 1.1.2.1.6.4.1.3 (under "Seismic Loads") for SC-I structures.

1.1.2.1.7.4 Structural Design of SC-II Concrete Structures

The design of SC-II concrete structures uses the ultimate strength design methods in accordance with the requirements of ACI-318.

Structural concrete used in construction of SC-II structures has a minimum compressive strength of 4,000 psi. Reinforcing steel used in SC-II structures has minimum yield strength of 60,000 psi.

Design of concrete structures considers the recommendations in ACI-352R, ACI-352.1R, and ACI-442, as appropriate.

Splicing of reinforcing by lapping, mechanical means, or welding is permitted with demonstrated compliance with the ductility and confinement requirements of ACI-318 and associated appendices. Reinforcing is provided at construction joints to develop shear-friction forces across the joints.

The design of post-installed concrete anchors and cast-in-place anchors for SC-II structural applications are in accordance with the requirements of ACI 349-01, Appendix B. Alternatively, SC-II post-installed mechanical concrete anchors used for mounting light components may be designed in accordance with the International Building Code and applicable International Code Council Evaluation Service Evaluation Report in lieu of ACI 349-01, Appendix B. In addition, SC-II post-installed adhesive anchors used for anchoring reinforcing bars may be designed in accordance with the International Building Code and International Code Council Evaluation Service Evaluation Report in lieu of ACI 349-01, Appendix B. Test data must be provided to demonstrate acceptable behavior when subjected to applicable thermal and radiation exposure.

1.1.2.1.7.5 Structural Design of SC-II Steel Structures

SC-II steel structures are designed in accordance with AISC ASD with guidance from AISC, *Manual of Steel Construction, Volume II, Connections*. Elastic design methods are generally used for steel design. However, under extreme loading conditions, plastic design methods are used.

Structural steel connections are designed as either friction or bearing type bolted or welded connections. Bolted connections are designed in accordance with AISC ASD and the Research Council on Structural Connections, *Specification for Structural Joints Using ASTM A325 or A490 Bolts*. Welded connections are designed in accordance with AISC ASD and AWS D1.1, *Structural Welding Code – Steel* or AWS D1.6, *Structural Welding Code – Stainless Steel*.

Guidance for the design of connections and member properties for HSS is given in AISC Connections Manual for Hollow Structural Sections or AISC Steel Design Guide 24 for Hollow Structural Section Connections.

Welding activities associated with SC-II structural steel components and their connections are accomplished in accordance with written procedures and meet the requirements of AWS D1.1 or AWS D1.6. The visual acceptance criteria for carbon and low alloy steel welds are as defined in AWS D1.1 or NCIG-01. The visual acceptance criteria for stainless steel welds and dissimilar welds are as defined in AWS D1.6.

Structural steel materials used in construction of SC-II buildings and structures consist of ASTM A36 or A992 rolled shapes, ASTM A500 Gr. B tube shapes, and ASTM A36 or A572 Gr. 50 carbon steel plate. ASTM A240 Type 304 or 304L is used for stainless steel plates, unless otherwise specified. Use of other qualified materials is permissible as required for specific applications. Bolts used for primary structural connections are A325, A490 or A193 Gr B8 Class 2 for stainless steel. A307 bolts are used for attaching ancillary components or equipment to structures. Welding electrodes are selected to be compatible with the materials being joined.

1.1.2.1.7.6 Foundation Design Requirements for SC-II Structures

The allowable soil bearing capacity for SC-II foundation design is based on SC-I allowable soil bearing pressure (see Section 1.1.2.1.6.2.2.3). Foundations for SC-II structures are designed in accordance with the appropriate requirements of Section 1.1.2.1.7.7.

1.1.2.1.7.7 Codes and Standards for SC-II Structures

Codes and standards listed in this section are limited to those applied to the SC-II structures at the MFFF:

American Concrete Institute (ACI)

- ACI-224R-90, Control of Concrete Cracking in Concrete Structures
- ACI-301-99, Standard Specifications for Structural Concrete.

  Section 5.3.7.5 of ACI 301-99 specifies a site-mixed cement repair mortar not to exceed (in cement concentration) a mixture greater than 1:2.5 cement to sand. Construction used a 1:1 ratio of cement to sand for minor repairs such as filling taper tie holes and abandoned drilled holes.
- ACI-315-99, Details and Detailing of Concrete Reinforcement
- ACI-318-99, Building Code Requirements for Reinforced Concrete Structures & Commentary
- ACI-336.2R-88, Suggested Analysis and Design Procedures for Combined Footings and Mats
- ACI-351.1R-99, Grouting for Support of Equipment & Machinery

- ACI-352R-91, Recommendations for Design of Beam-Column Joints in Monolithic Reinforced Concrete Structures, Reapproved 1997
- ACI-352.1R-89, Recommendations for Design of Slab-Column Connections in Monolithic Reinforced Concrete Structures, Reapproved 1997
- ACI-349-01, Appendix B, Anchoring to Concrete (for anchoring to concrete only)
- ACI-360R-92, Design of Slabs on Grade, Reapproved 1997
- ACI-351.2R-94, Foundations for Static Equipment
- ACI-439.3R-91, Mechanical Connections of Reinforcing Bars
- ACI-SP-152-95, Design and Performance of Mat Foundations
- ACI-503R-93, Use of Epoxy Compounds with Concrete
- ACI-442-88, Response of Concrete Buildings to Lateral Forces
- ACI-207.1R-96, Mass Concrete
- ACI-207.2R-95, Effect of Restraint, Volume Change, and Reinforcement on Cracking of Mass Concrete
- ACI-207.4R-93, Cooling and Insulating Systems for Mass Concrete

American Institute of Steel Construction (AISC)

- AISC ASD, *Manual of Steel Construction, Allowable Stress Design*, 9th Edition, 1989 and Supplement #1, dated December 17, 2001

  With respect to the QL-2 tie-back beams connecting the gabion walls to the MFFF, the standard hole size for 1" diameter bolts and larger is 1/8" over the bolt diameter to accommodate manufacturing process tolerances and to provide fit and rotation capacity proportional to the size of connections typically using large bolt diameters.
- AISC, *Seismic Provisions for Structural Steel Buildings*, April 1997
- AISC Manual of Steel Construction, Volume II - Connections, ASD 9th Edition, 1989 / 2nd Edition, 1998

American Society of Civil Engineers (ASCE)

- ASCE & SEI, 1999, Structural Design for Physical Security, State of the Practice
- ASCE Standard 4-98, Seismic Analysis of Safety Related Nuclear Structures
- ASCE Standard 7-98, Minimum Design Loads for Buildings and Other Structures
- ASCE Standard 8-02, Specification for the Design of Cold-Formed Stainless Steel Structural Members

American Welding Society (AWS)

- AWS-D1.1-98, *Structural Welding Code – Steel*, 1998

- NCIG-01, *Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants*, Revision 2, EPRI NP-5380
- AWS-D1.6-99, *Structural Welding Code-Stainless Steel*, 1999
- AWS-D1.3-98, *Structural Welding Code-Sheet Steel*, 1998

American Association of State Highway and Transportation Officials (AASHTO)

- AASHTO HB-16, Standard Specifications for Highway Bridges, Sixteenth Edition, 1996

American Iron and Steel Institute (AISI)

- AISI, Specifications for the Design of Cold-Formed Steel Structural Members, 1986

Research Council on Structural Connections of the Engineering Foundation (RCSC)

- Research Council on Structural Connections, Specification for Structural Joints Using ASTM A325 and A490 Bolts, June 23, 2000

Code of Federal Regulations (CFR)

- 10 CFR Part 70, Domestic Licensing of Special Nuclear Material
- 10 CFR Part 73, Physical Protection of Plants and Materials
- 10 CFR Part 75, Safeguards on Nuclear Material (effective as amended in the U.S. Federal Register)

Crane Manufactures Association of America (CMAA)

- CMAA Spec. 70, Specification for Top Running Bridge and Gantry Type Multiple Girder Electrical Overhead Traveling Cranes, 1994
- CMAA Spec. 74, Specification for Top Running and Under Running Types of Single Girder Electric Overhead Traveling Cranes, 1994

SRS Engineering Standards Manual (WSRC-TM-95-1)

- Engineering Standard No. 01060, *Structural Design Criteria*, Revision 5, dated September 2001
- Engineering Standard No. 01110, Civil Site Design Criteria, Revision 4, dated July 9, 2002

U.S. Nuclear Regulatory Commission (NUREG)

- NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports Nuclear Power Plants 3.5.3, July 1981
- NUREG-0800, Standard Review Plan 3.8.4, Other Seismic Category I Structures, July 1981
- NUREG-0800, Standard Review Plan 3.5.1.6, Aircraft Hazards, April 1996

1.1.2.1.7.8 Design Values for SC-II Structures

1.1.2.1.7.8.1 Structural Design Loads for SC-II Structures

Design loads are based upon postulated building loads (i.e., dead loads, live loads, operating and transient loads, and natural phenomena hazard loads). These loads are divided into three classifications (normal loads, severe environmental loads, and extreme environmental loads) specified in Section 1.1.2.1.6.4, except the only extreme environmental load considered is the design earthquake (E'). See Table 1.1.2-2 for the MFFF site design criteria. See Table 1.1.2-1 for the structures located at the MFFF site, along with their seismic category classification.

1.1.2.1.7.8.2 Loading Combinations for SC-II Structures

Loading combinations for the design of SC-II structures and facilities are the same as the loading combinations for SC-I structures (see Section 1.1.2.1.6.4), except the only extreme environmental load considered is the design earthquake (E'). Also, for concrete structures, "U" is the section strength required to resist design loads based upon the ultimate strength design methods described in ACI 318. For steel structures, "S" is the required section strength based on elastic design methods and the allowable stresses defined in AISC ASD Manual of Steel Construction, Allowable Stress Design.

Applicability of Loads for SC-II Structures

The following criteria are considered when determining applicable loading combinations for the design of the MFFF SC-II structures:

- Live loads are applied fully, partially, totally removed from the members, or shifted in location and pattern as necessary to obtain the worst-case loading condition for maximizing internal forces and moments for the loading combinations. Impact forces caused by moving loads are applied where appropriate.
- Appropriate construction loads are considered in the service loading combinations. Construction methods and sequence are considered, and appropriate loading conditions are applied to ensure the structural integrity of partially erected or open structures.
- Where a load reduces the overall loading on a structural member, a load coefficient of 0.9 is applied to that load component in the loading combination. The reducing coefficient was only used for loads that are always present or that always occur simultaneously with other loads. Otherwise, the coefficient for that load was taken conservatively as zero.

1.1.2.1.7.9 Analysis Results for SC-II Structures

Safe Haven Building (BSH)

Safe Haven Buildings are Seismic Category II (SC-II) concrete structures. They are evaluated for natural phenomena using ANSYS code. The ANSYS results are evaluated and concrete design for the various structural components is performed using Excel spreadsheets (see Section 1.1.2.1.6.5). The design is performed for the composite worst case forces and moments occurring at an element from the analyzed loading combinations, using ACI-318 Building Code. The structure is also evaluated for tornado loads and missile impact.

Soil Structure Interaction (SSI)

The BSH is a relatively low, above grade structure founded on soils and immediately adjacent to the BMF. The maximum floor acceleration of the BMF slab at grade is used for the seismic input to the structural analysis of BSH.

Foundations

The BSH foundation is considered as a monolithic slab with a uniform soil spring of 14.0 pci for dead loads and 28 pci for seismic, wind, tornado and live loads.

Tornado Missile Barrier Analysis and Design

See BMF Section 1.1.2.1.6.5.1 for a description of the tornado missile impact analysis and design.

Results:

The ANSYS computer analysis results are used to design the primary structural reinforced concrete elements, (foundation, walls and roof slab). The structural elements are designed for the maximum/minimum loading using the ultimate strength methodology and building code requirements designated in ACI 318-99. The reinforcing provided satisfies the strength requirements for all loading combinations.

See Figure 1.1.2-116 and Figure 1.1.2-117 for representative concrete and reinforcing details of BSH.

BMF Stair Towers

The two BMF stair towers are located and supported from the outside of each berm wall. They provide access from the ground elevation of the MFFF site to the roof of the BMF.

The stair towers are SC-II structures because of their proximity with other Seismic Category I structures. They are fabricated from structural steel and have their own concrete foundation. The element thickness, reinforcing, and detailing provide satisfactory strength for all loadings and load combinations.

Secondary Alarm Station (SAS)

The SAS is a vital security area and is designed as a hardened (bullet resistant) area, with its own support systems. Walls and ceiling are reinforced concrete. The SAS contains controls and monitoring equipment required for site security and safety. The SAS also includes telecommunications, offices, restroom facilities, and drinking water.

The SAS is a single-story SC-II structure. The element thickness, reinforcing, and detailing provide satisfactory strength for all loadings and load combinations.

HVAC Exhaust Vent Stack (BMP)

The HVAC exhaust vent stack is installed on a concrete pedestal located on the roof of the BMP. Its base elevation is 79’7” and its top elevation is 119’6”. The vent stack is 8-1/2-ft diameter, approximately 40-ft tall and has an outer wall thickness of 3/4 in. See Figure 1.1.2-91 for diagram of the exhaust vent and the platform around the vent stack. The purpose of the exhaust vent stack is to provide a path for the exhaust air and to mix and blend gases. It is designed to resist normal, severe environmental and extreme environmental loads. The design loads considered are dead, live, wind, seismic, tornado, and impact of a tornado generated missile.

In the analysis, the HVAC vent stack is treated as a cantilevered beam with a uniform distribution of wind loads and seismic loads.

Vent Stack Barriers (BMP)

The HVAC exhaust vent stack is located 8 ft west of column line V and 8 ft south of column line 6 at roof El. 73'0" of the BMF. The security barriers are located below the base of the vent stack.

The security barriers are designed for seismic loads, tornado missile and security loads. An elastic stability analysis is made by calculating the buckling capacity.

The barriers are also designed for a pressure load applied normal to the plane of the barriers to simulate potential explosion loads.

1.1.2.1.8 Conventional Seismic (CS) Structures

1.1.2.1.8.1 Functions of CS Structures

CS buildings and structures protect and support QL-3, and QL-4 SSCs. QL-3 and QL-4 SSCs are non-IROFS and consequently are not required to fulfill a safety function or maintain structural integrity following postulated design basis events. Qualification is based on the conventional seismic requirements of the UBC or IBC with consideration given to worker/operator safety, sound engineering practice, and protection of investment.

1.1.2.1.8.2 Requirements for CS Structures

Structural analysis and design of CS structures is in accordance with the UBC or IBC.

1.1.2.1.8.3 Codes and Standards for CS Structures

The following codes are used for determining applicable loading combinations for the design of CS structures:

- UBC, 1997
- IBC, 2003
- ASCE Standard 7-98, *Minimum Design Loads for Buildings and Other Structures*
- ACI 318-99, *Building Code Requirements for Reinforced Concrete*

- AISC Manual of Steel Construction, Allowable Stress Design, *Specification for Structural Steel Buildings 9th Edition and Supplement #1*
- AISC LRFD – Manual of Steel Construction, *Load Resistance Factor Design*, 2nd Edition, 1998
- ACI 349-01, Appendix B, Code Requirements for Nuclear Safety Related Concrete Structures and Commentary, *Anchoring to Concrete*

1.1.2.1.8.4 Design Values for CS Structures

Design loads for CS structures are normal, severe, and extreme loads with extreme loads limited to conventional seismic loads as specified by the UBC or IBC. Normal loads are those loads encountered during normal operation of CS Buildings, Structures and Facilities at the MFFF site. They include dead loads (D), live loads (L), hydrostatic fluid pressure loads (F), soil pressure loads (H), thermal loads (T_o), and pipe, cable tray, conduit, and HVAC duct reactions (R_o), (see Section 1.1.2.1.6.4.1.1). Also, rain loads (R), snow (S) and ice (I) loads, transportation vehicle loads and heavy floor loads, crane, monorail, hoist and elevator loads, and lateral soil pressure loads (H) are considered.

Loading combinations for the design of CS buildings, structures and facilities are in accordance with conventional structural design codes and standards. CS structures are designed for the conventional seismic load (E_c) as defined in ASCE Standard 7 and the UBC or IBC.

1.1.2.1.8.5 Analysis Results for CS Structures

Design results for CS structures are in conformance with the codes and standards for CS structures. CS structures are not IROFS and the analysis results are not presented.

1.1.2.1.9 Evaluation of Post Development Intense Precipitation Storm Runoff

See Section 1.3.4.2.3 for description of the effects of locally intense precipitation on the Savannah River Site and F Area. MOX Services has evaluated these effects for the "as-developed" configuration of the MFFF site. The objective of this study was to determine flood or flow depths that could potentially develop on drainage areas on the MFFF site from a "Local Intense Precipitation Design Storm" (LIPDS) event. The drainage areas focused on were those believed to be the most vulnerable to flooding. The criteria used for the evaluation defined the LIPDS as a 100,000 year frequency, 24-hour duration storm having a 22.7-in rainfall accumulation.

A conservative approach was applied by making the following assumptions.

- The drainage areas selected for this study within the MFFF building complex are more confined. Under these conditions runoff will tend to concentrate and flow deep. Opposed to this are the drainage conditions around or outside the periphery of the complex that were not selected. The perimeter drainage conditions here are considered to be less constrictive by having open graded areas designed to let runoff flow shallow and away from structures.

- The drainage subareas selected, delineated and used in the study are the most vulnerable to flooding.
- The surface depressions, truck docks, and under ground storm drains and inlets are full of water and debris (the runoff is flowing on the ground surface).
- Grassed and unpaved drainage areas addressed in the study are saturated (no infiltration is occurring and runoff coefficients = 98).
- The roof areas adjacent to a drainage areas addressed in the study are fully contributing to the runoff within that drainage area (gutters are overflowing).
- Flow velocities used in Manning's formula are subcritical where applicable to obtain maximum flow depth (Froude Number <1).

The results of this MOX Services evaluation conclude that the current MFFF site grading design does maintain flood levels sufficiently below the building floor elevations when the runoff generated by a LIPDS is at the peak flow rate determined under the above listed conditions.

The anticipated maximum flood levels that could occur under the different possible scenarios depicted for the Savannah River Basin are well below the plateau elevations established for the MFFF project site (see Section 1.3.4.2). The design basis flood determines the MFFF site to be a dry site with the design basis flood level safely below the MFFF site by approximately 64 ft (see Section 1.3.4.2.4.1).

Table 1.1.2-1. Building Seismic Classifications

Buildings	Abbreviation	Seismic Category
MOX Fuel Fabrication building MOX Processing area[a] Aqueous Polishing area[a] Shipping and Receiving area[a]	BMF BMP BAP BSR	SC-I SC-I SC-I SC-I
Emergency Generator building	BEG	SC-I
Safe Haven buildings	BSH	SC-II
Reagent Processing building	BRP	CS
Administration building	BAD	CS
Secured Warehouse building	BSW	CS
Receiving Warehouse building	BRW	CS
Technical Support building	BTS	CS

Facilities, Structures, and Areas	Abbreviation	Seismic Category
Emergency Fuel Storage Vault	UEF	SC-I
Gas Storage Facility	UGS	CS
Process Chiller Pads	VCY	CS
Main 13.8 kV/4.16 kV Transformers	EAB-XS	CS
Vehicle Access Portal	WVA	CS
HVAC Chiller Pads	VCX	CS

[a] These areas are part of the MOX Fuel Fabrication building.

Table 1.1.2-2. Summary of MFFF Site Design Criteria

Criterion	Value
Severe Wind (SC-I and SC-II)	Three-second wind speed: 130 mph Missile criteria: 2-in x 4-in timber plank, 15 lb at 50 mph (horizontal); max. height 50 ft above grade(see Note 1)
Extreme Wind/Tornado (Wind Loads) (SC-I)	Three-second wind speed: 240 mph Atmospheric pressure change: 150 psf at 55 psf/sec Rate of pressure drop: 55 psf/sec
Tornado Missile Spectrum (SC-I)	2-in x 4-in timber plank Mass: 15 lb Horizontal Impact Speed: 150 mph Vertical Impact Speed: 100 mph Maximum Height: 200 ft above grade 3-in diameter standard steel pipe Mass: 75 lb Horizontal impact speed: 75 mph Vertical impact speed: 50 mph Maximum height: 100 ft above grade 3000-lb automobile Horizontal impact speed: 25 mph, rolls and tumbles
Basic Wind (CS)	Three second wind speed: 107 mph (Note 1)
Floods	Design flood level above MSL: 207.9 ft Probable maximum flood level above MSL: 224.5 ft Site grade level ≈ 272 ft above mean sea level (MSL)
Precipitation	Accumulative precipitation: Fifteen minutes: 3.9 in One hour: 7.4 in Three hours: 14.1 in Six hours: 16.7 in Twenty-four hours: 22.7 in
Snow and Ice Loads	Snow/ice loading: 10 lb/ft² Exposure factor: 1.0 Snow drift is considered in the design For SC-I and SC-II , Importance factor = 1.2 For CS, Importance factor = 1.0
Seismic (Ground Motion) (SC-I and SC-II) (CS)	Regulatory Guide 1.60 scaled to 0.20g peak ground acceleration In accordance with requirements of ASCE Standard 7, Section 9, and UBC or IBC, Sections 1626 to 1634
Foundation Design Bearing Pressure	(see Section 1.1.2.1.6.2.2.3)
Groundwater	Maximum groundwater elevation above MSL: 210.0 ft.
Explosions	(see Section 1.1.2.1.6.4.1.3)
Aircraft Impact	(Not a design basis event, see Section 1.1.2.1.6.4.1.3)
Range Fires	(Not a design basis event, see Section 1.1.2.1.6.4.1.3)

Note 1: For determining wind loads using the ASCE 7-98 procedure, the following definitions apply: I = 1.15 (SC-I and SC-II), I = 1.0 (CS); Exposure Category = C; K_{zt} = 1.0; and K_d = 1.0.

Table 1.1.2-3. Minimum Factors of Safety for Overturning, Sliding and Flotation

Loading Combination	Overturning	Sliding	Flotation
D + F + H + W	1.5	1.5	-
D + F + H + E'	1.1	1.1	-
D + F + H + W_t (Note 1)	1.1	1.1	-
D + F + F'	-	-	1.1

Note 1: When checking for overall structural stability, only the tornado wind load case of $W_t = W_w$ was considered. Neither tornado internal pressure loadings, W_p, nor tornado missile impact loadings, W_m, appreciably affect structure overturning, sliding, or flotation.

Table 1.1.2-4. Minimum Required Concrete Thickness or Soil Cover

Missile Description	Target	Minimum Thickness Requirement to Protect Against		
		Penetration	Perforation (e)*	Scabbing (s)*
3-in Dia. Steel Pipe	BMF/BEG/UEF/BSH Roof	x=2.46”	6.34”	8.30”
	BMF/BEG/BSH Wall	x=3.48”	7.84”	9.96”
2-in x 4-in Timber Plank	BMF/BEG/UEF/BSH Roof	x=1.73”	5.63”	9.40”
	BMF/BEG/BSH Wall	x=2.44”	7.36”	10.56”
3-in Dia Steel Pipe and 2-in x 4-in Timber Plank (Horizontal panels envelop BMF/BEG Roof, Vertical panels envelop BMF/BEG wall)	BMF HVAC Intake	x=3.48”	7.84”	10.56”
	UEF/BEG/BSH	x=2.46”	6.34”	9.40”
	Electrical Conduit Bank	Required Soil Depth = 3 ft.		
	Buried IROFS Piping	Required Soil Depth = 3 ft.		

*20% additional margin included.

Table 1.1.2-5. Strain-Dependent Soil Properties in SSI Model – Best Estimate Case

Layer Number	Layer Top Elevation (ft)	Shear Wave Propagation		Compression Wave Propagation		Unit Weight γ (pcf)
		Velocity Vs (ft/sec)	Damping βs (%)	Velocity Vp (ft/sec)	Damping βp (%)	
1	270	1429.8	0.74	2904.0	0.25	114
2	265	1360.8	1.16	2593.0	0.39	116
3	257	1275.7	1.41	2210.0	0.47	120
4	253.5	1241.0	2.01	2149.0	0.67	120
5	245	1214.1	2.47	2103.0	0.82	120
6	236	1088.3	3.52	2074.0	1.17	118
7	223.5	1045.9	4.42	1993.0	1.47	118
8	211	912.8	3.01	3837.0	1.00	114
9	204	976.5	3.82	4104.0	1.27	119
10	193	970.7	3.94	4080.0	1.31	119
11	182	896.6	4.36	3769.0	1.45	110
12	172	977.7	4.02	4109.0	1.34	121
13	162	975.8	4.06	4101.0	1.35	121
14	152	855.0	5.07	3594.0	1.69	113
15	146	849.4	5.23	3570.0	1.74	113
16	140	1060.3	3.90	4457.0	1.30	121
17	135	1241.1	3.33	5217.0	1.11	125
18	122.5	1241.1	3.33	5217.0	1.11	125
19	110	1241.1	3.33	5217.0	1.11	125
20	97.5	1241.1	3.33	5217.0	1.11	125
21	85	1299.7	3.78	5463.0	1.26	125
22	72	1294.1	3.87	5439.0	1.29	125
23	59	1268.2	4.28	5330.0	1.43	125
24	29	1268.2	4.28	5330.0	1.43	125
25	-2	1804.8	1.56	7586.0	0.52	125
26	-50	2127.8	1.30	8944.0	0.43	130
27	-85	2127.8	1.30	8944.0	0.43	130
28	-120	2234.0	1.36	9390.0	0.45	125

Table 1.1.2-5 Strain-Dependent Soil Properties in SSI Model – Best Estimate Case (continued)

Layer Number	Layer Top Elevation (ft)	Shear Wave Propagation		Compression Wave Propagation		Unit Weight γ (pcf)
		Velocity Vs (ft/sec)	Damping βs (%)	Velocity Vp (ft/sec)	Damping βp (%)	
29	-160	2234.0	1.36	9390.0	0.45	125
30	-200	2234.0	1.36	9390.0	0.45	125
31	-240	2234.0	1.36	9390.0	0.45	125
32	-280	2624.0	1.24	11029.0	0.41	130
33	-330	2459.8	1.32	10339.0	0.44	130
34	-380	2565.2	1.33	10782.0	0.44	135
35	-438	2813.6	1.19	11826.0	0.40	135
36	-496	3067.7	1.13	12894.0	0.38	135
37	-534	2809.3	1.24	11808.0	0.41	135
Bedrock	-595	11000.0	0.01	19315.0	0.01	150

Table 1.1.2-6. Strain-Dependent Soil Properties in SSI Model – Lower Bound Case

Layer Number	Layer Top Elevation(ft)	Shear Wave Propagation		Compression Wave Propagation		Unit Weight γ (pcf)
		Velocity Vs (ft/sec)	Damping βs (%)	Velocity Vp (ft/sec)	Damping βp (%)	
1	270	1165.4	0.77	2367.0	0.26	114
2	265	1101.6	1.35	2099.0	0.45	116
3	257	1026.1	1.74	1777.0	0.58	120
4	253.5	997.8	2.34	1657.0	0.78	120
5	245	972.3	2.90	1684.0	0.97	120
6	236	851.7	4.47	1623.0	1.49	118
7	223.5	821.9	5.20	1566.0	1.73	118
8	211	737.2	3.26	3099.0	1.09	114
9	204	775.8	4.38	3261.0	1.46	119
10	193	762.7	4.81	3206.0	1.60	119
11	182	685.9	5.99	2883.0	2.00	110
12	172	755.2	5.35	3174.0	1.78	121
13	162	749.0	5.55	3148.0	1.85	121
14	152	647.4	6.83	2721.0	2.28	113
15	146	648.6	6.79	2726.0	2.26	113
16	140	840.8	4.53	3534.0	1.51	121
17	135	994.0	3.74	4178.0	1.25	125
18	122.5	994.0	3.74	4178.0	1.25	125
19	110	994.0	3.74	4178.0	1.25	125
20	97.5	994.0	3.74	4178.0	1.25	125
21	85	1038.0	4.23	4363.0	1.41	125
22	72	1036.4	4.26	4356.0	1.42	125
23	59	1044.9	4.10	4392.0	1.37	125
24	29	1044.9	4.10	4392.0	1.37	125
25	-2	1475.2	1.55	6201.0	0.52	125
26	-50	1735.1	1.33	7293.0	0.44	130
27	-85	1735.1	1.33	7293.0	0.44	130
28	-120	1823.1	1.38	7663.0	0.46	125

Table 1.1.2-6 Strain-Dependent Soil Properties in SSI Model – Lower Bound Case (continued)

Layer Number	Layer Top Elevation (ft)	Shear Wave Propagation		Compression Wave Propagation		Unit Weight γ (pcf)
		Velocity Vs (ft/sec)	Damping βs (%)	Velocity Vp (ft/sec)	Damping βp (%)	
29	-160	1823.1	1.38	7663.0	0.46	125
30	-200	1823.1	1.38	7663.0	0.46	125
31	-240	1823.1	1.38	7663.0	0.46	125
32	-280	2147.0	1.19	9024.0	0.40	130
33	-330	2013.1	1.27	8461.0	0.42	130
34	-380	2305.8	1.31	9692.0	0.44	135
35	-438	2295.7	1.20	9649.0	0.40	135
36	-496	2504.4	1.13	10526.0	0.38	135
37	-534	2293.6	1.25	9640.0	0.42	135
Bedrock	-595	11000.0	0.01	19315.0	0.01	150

Table 1.1.2-7. Strain-Dependent Soil Properties in SSI Model – Upper Bound Case

Layer Number	Layer Top Elevation (ft)	Shear Wave Propagation		Compression Wave Propagation		Unit Weight γ (pcf)
		Velocity Vs (ft/sec)	Damping βs (%)	Velocity Vp (ft/sec)	Damping βp (%)	
1	270	1754.9	0.69	3564.0	0.23	114
2	265	1679.0	0.99	3200.0	0.33	116
3	257	1576.7	1.20	2731.0	0.40	120
4	253.5	1549.1	1.59	2683.0	0.53	120
5	245	1516.7	2.05	2627.0	0.68	120
6	236	1387.1	2.60	2643.0	0.87	118
7	223.5	1361.7	3.01	2595.0	1.00	118
8	211	1140.3	2.59	4793.0	0.86	114
9	204	1239.9	3.06	5212.0	1.02	119
10	193	1229.9	3.23	5169.0	1.08	119
11	182	1134.5	3.70	4769.0	1.23	110
12	172	1233.2	3.42	5183.0	1.14	121
13	162	1226.9	3.53	5157.0	1.18	121
14	152	1092.1	4.07	4590.0	1.36	113
15	146	1090.4	4.10	4583.0	1.37	113
16	140	1337.5	3.27	5622.0	1.09	121
17	135	1566.6	2.66	6585.0	0.89	125
18	122.5	1566.6	2.66	6585.0	0.89	125
19	110	1566.6	2.66	6585.0	0.89	125
20	97.5	1566.6	2.66	6585.0	0.89	125
21	85	1701.9	2.27	7153.0	0.76	125
22	72	1701.8	2.27	7153.0	0.76	125
23	59	1665.8	2.77	7002.0	0.92	125
24	29	1665.8	2.77	7002.0	0.92	125
25	-2	2228.1	1.38	9365.0	0.46	125
26	-50	2612.9	1.25	10982.0	0.42	130
27	-85	2612.9	1.25	10982.0	0.42	130
28	-120	2745.9	1.29	11541.0	0.43	125

Table 1.1.2-7 Strain-Dependent Soil Properties in SSI Model – Upper Bound Case (continued)

Layer Number	Layer Top Elevation (ft)	Shear Wave Propagation		Compression Wave Propagation		Unit Weight γ (pcf)
		Velocity Vs (ft/sec)	Damping βs (%)	Velocity Vp (ft/sec)	Damping βp (%)	
29	-160	2745.9	1.29	11541.0	0.43	125
30	-200	2745.9	1.29	11541.0	0.43	125
31	-240	2745.9	1.29	11541.0	0.43	125
32	-280	3249.2	1.00	13657.0	0.33	130
33	-330	3045.9	1.08	12802.0	0.36	130
34	-380	3153.0	1.25	13253.0	0.42	135
35	-438	3472.3	1.21	14595.0	0.40	135
36	-496	3783.4	0.97	15902.0	0.32	135
37	-534	3464.4	1.09	14561.0	0.36	135
Bedrock	-595	11000.0	0.01	19315.0	0.01	150

Table 1.1.2-8. Frequencies and Modal Mass Ratios of First 20 Modes of the Simplified Finite Element Structural Model Fixed at Grade

Mode Number	Frequency (Hz)	Modal Mass Ratio* X Input	Modal Mass Ratio* Y Input	Modal Mass Ratio* Z Input
1	8.78	0.01	0.64	-
2	9.19	0.61	0.01	-
3	9.95	-	0.09	-
4	11.46	0.01	0.01	-
5	11.90	0.02	-	-
6	12.06	0.06	-	-
7	13.36	0.02	0.01	-
8	13.42	-	-	-
9	13.67	-	-	-
10	14.01	-	-	0.07
11	14.41	-	-	-
12	15.21	-	-	-
13	15.52	-	-	0.01
14	15.67	-	-	0.02
15	16.36	-	-	-
16	16.64	-	-	-
17	17.12	-	-	0.09
18	17.27	-	-	0.01
19	17.32	-	-	-
20	17.49	-	-	0.01

* Modal Mass Ratio = effective modal mass / total horizontal mass of FEM structural model excluding lumped mass at grade = effective modal mass / (16,488x1,000/12) lb-sec^2/in

Table 1.1.2-9. Flexible Slab Groups in BMF

Slab Group	Cracked Slab Freq., fcr (Hz)	Uncracked Slab Freq. fuc = 1.414xfcr (Hz)	Applicable Range of Cracked Slab Frequency (Hz)
1	10.6	15.1	$10.6 \leq f_{cr} < 12.3$
2	12.3	17.4	$12.3 \leq f_{cr} < 14.3$
3	14.3	20.3	$14.3 \leq f_{cr} < 16.3$
4	16.3	23.1	$16.3 \leq f_{cr} < 19.5$
5	19.5	27.6	$19.5 \leq f_{cr} < 22.8$
6	22.8	32.2	$22.8 \leq f_{cr} < 33.0$

Table 1.1.2-10. Flexible Slab Groups at Different Floor Elevations of BMF

TOC Floor Elev. (ft)	Applicable Slab Groups BMP Area	Applicable Slab Groups BAP Area	Applicable Slab Groups BSR Area
73.00	2, 4, 5, 6		4, 6
52.50		6	
46.83	1, 2, 3, 4, 5, 6		2, 5, 6
35.00		6	
23.33	1, 2, 3, 4, 5, 6		6
17.50		4, 6	
Grade		6	2

Table 1.1.2-11. Flexible Slabs in BEG

Slab Group	Cracked Slab Frequency, fcr (Hz)	Associated Uncracked Slab Frequency, fuc* (Hz)
1	13.5	19.2
2	23.2	32.8

*NS walls run in the Y-direction, and EW walls run in the X-direction.

Table 1.1.2-12. Flexible Wall Groups in BMP and BSR Areas of BMF

Slab Group	Lower Bound Uncracked Wall Frequency, fuc (Hz)	Applicable Range of Uncracked Wall Frequency (Hz)
1	16.4	$16.4 \le f_{uc} < 18.9$
2	18.9	$18.9 \le f_{uc} < 22.0$
3	22.0	$22.0 \le f_{uc} < 25.3$
4	25.3	$25.3 \le f_{uc} < 30.8$

Table 1.1.2-13. Applicable Flexible Wall Groups at Different Elevations

TOC Floor Elevation (ft)	Applicable Wall Groups - BMP		Applicable Wall Groups - BSR	
	NS Walls	EW Walls	NS Walls	EW Walls
46.83 to 73.00	4	1, 2, 3, 4	3	3, 4
23.33 to 46.83	4	2, 3, 4	3, 4	3, 4
Grade to 23.33	3, 4	4	-	3, 4

Table 1.1.2-14. Maximum Floor Accelerations (SSI Analysis)

TOC Floor Elevation (ft)	X (EW) Accel. (g)			Y (NS) Accel. (g)				Z (Vert.) Accel. (g)
	BMP/BSR		BAP	BMP/BSR		BAP		BMP/BSR/ BAP
	Zone BMPX-1	Outside BMPX-1		Zone BMPY-1	Outside BMPY-1	Zone BAPY-1	Outside BAPY-1	
73.00	0.34	0.30	0.30	0.31	0.31	0.31	0.31	0.38
52.50	-	-	0.27	-	-	0.30	0.29	0.38
46.83	0.31	0.27	-	0.30	0.28	-	-	0.37
35.00	-	-	0.27	-	-	0.27	0.27	0.36
23.33	0.31	0.27	-	0.28	0.27	-	-	0.35
17.50	-	-	0.26	-	-	0.26	0.26	0.36
Grade	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.36
BSR Basement	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.36
BAP Basement	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.36

Note: (1) Maximum floor accelerations are to be input seismic loads in static finite element analysis of structure.
(2) Accelerations at BSR and BAP basements are taken to be the same as those at grade.

Table 1.1.2-15. Maximum Relative Displacements of Floors with Respect to Slab on Grade[1,5]

TOC Floor Elevation (ft)	X (EW) Rel. Disp. (in)			Y (NS) Rel. Disp. (in)			Vert. Rel. Disp. (in)2
	BMP/BSR		BAP	BMP/BSR		BAP3	BMP/BSR/BAP
	Zone BMPX-1	Outside BMPX-1		Zone BMPY-1	Outside BMPY-1		
73.00	0.15	0..12	0.12	0.17	0.16	0.19	0.06
52.50	-	-	0.10	-	-	0.16	0.06
46.83	0.40	0.08	-	0.13	0.13	-	0.06
35.00	-	-	0.08	-	-	0.13	0.06
23.33	0.06	0.05	-	0.08	0.07	-	0.06
17.50	-	-	0.06	-	-	0.10	0.06
0.00 (Grade)	0	0	0.03	0	0	0.05	0
BSR Basement[4]	0	0	0	0	0	0	0
BAP Basement[4]	0	0	0	0	0	0	0

Notes:

1. The displacement values should be considered both positive and negative.
2. The vertical relative displacements with respect to foundation are conservatively taken to be the same as the worst one in the entire building. The relative vertical displacement of foundation mat with respect to grade considered to be zero.
3. Relative Y-direction displacements at BAP floors are worse values both inside and outside zone BAPY-1.
4. The three buildings (BAP, BMP, & BSR) foundations are assumed to have no relative displacements with respect to grade in X and Y directions.
5. These relative displacements are applicable only for design of systems within the building. For design of systems extending outside the building, see Section 1.1.2.2.

Table 1.1.2-16. Inter-Floor Relative Horizontal Displacements*

BMP/BSR

FLOOR ELEVATIONS (FT)	X (EW) DISP. (IN)	Y (NS) DISP. (IN)
46.83 TO 73.00	0.04	0.04
23.33 TO 46.83	0.05	0.06
GRADE TO 23.33	0.06	0.08

BAP

FLOOR ELEVATIONS (FT)	X (EW) DISP. (IN)	Y (NS) DISP. (IN)
52.50 TO 73.00	0.02	0.03
35.00 TO 52.50	0.02	0.03
17.50 TO 35.00	0.02	0.03
GRADE TO 17.50	0.03	0.05

*Between consecutive floors above Elev. 17.50 ft, difference in the maximum inter-floor displacements both inside and outside the amplification zones is typically small, and for simplicity, only the larger value from both zones is shown.

Figure 1.1.2-1. Location of Savannah River Site and F-Area

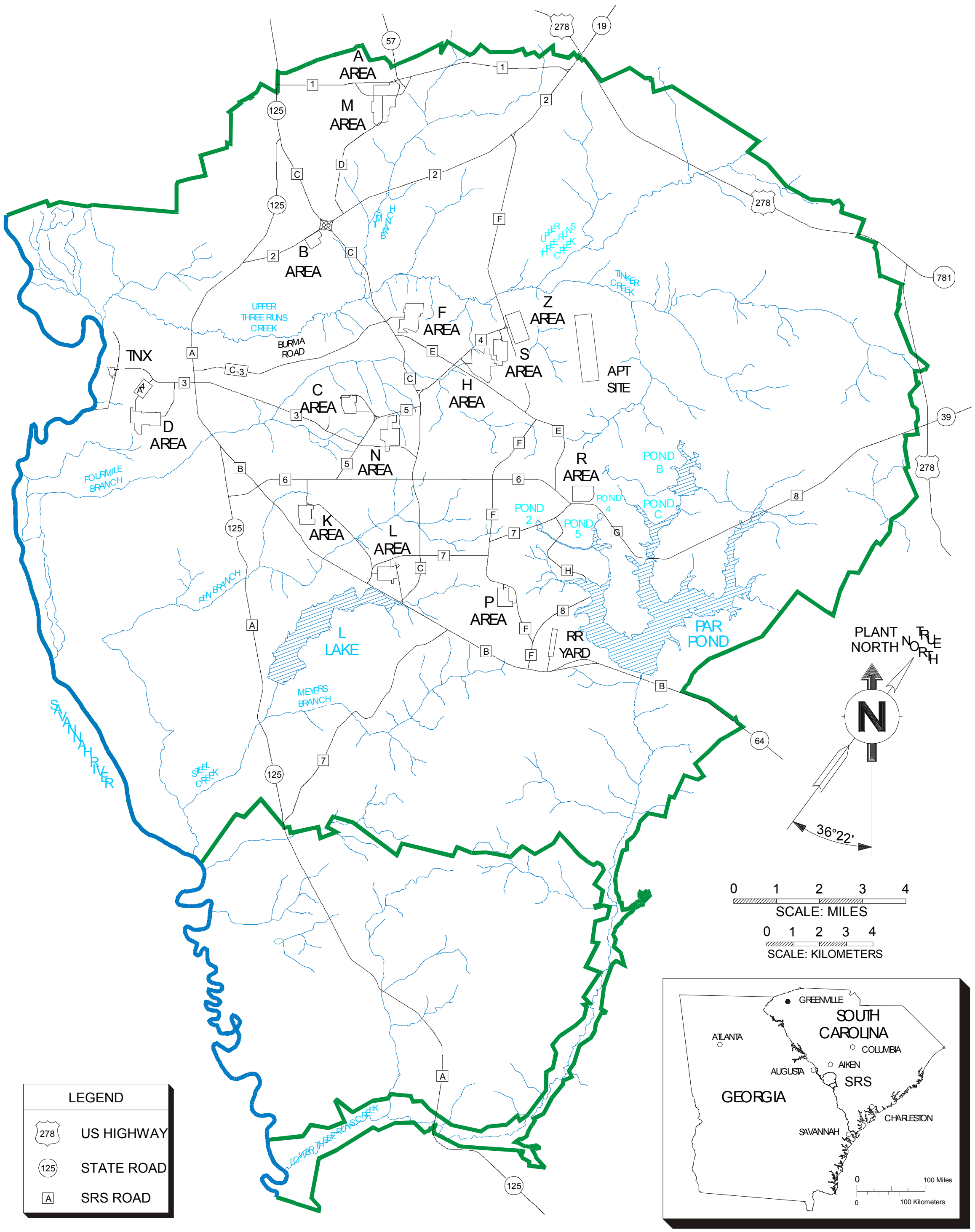


Figure 1.1.2-2. through Figure 1.1.2-121. Withheld from Public Disclosure Under 10CFR2.390

1.1.2.2 Seismic Qualification Requirements for Systems, Structures, and Components

The integrated safety analysis (ISA) identifies the items relied on for safety (IROFS) required to provide mitigation or prevention of Design Basis Events (DBEs). This section identifies the corresponding design bases and functional requirements that are incorporated into the design of facilities systems and components to assure safety in accordance with 10 CFR §70.61. The DBE specifically addressed in this section is the seismic hazard from the set of natural phenomena hazards addressed in the ISA.

Qualification requirements for building structures are addressed in Section 1.1.2.1. Additional qualification requirements applicable to equipment and components are addressed in Section 1.1.2.3.

1.1.2.2.1 Seismic Classification of Systems, Structures, and Components

SC and seismic performance requirements (SPR) are used to classify SSCs. The seismic category identifies the role of the SSC during the DBE. The role of SSCs is either safety, directly preventing adverse interaction with IROFS, or indirectly maintaining the integrity and effectiveness of the IROFS SSCs. The SPRs specify the special requirements that are satisfied by the SSC in the given seismic environment. Seismic qualification requirements specified herein include the seismic loading, analytical approach, design codes and acceptance criteria, using a graded classification program that considers the relative importance of the safety function performed by the SSC. The application of seismic classifications is discussed below.

1.1.2.2.1.1 Seismic Category

The SC-I classification is applicable to the MFFF SSCs, as well as the supporting SSCs that are required to withstand the effects of the DE to prevent or mitigate adverse consequences of the earthquake. The DE is described in Section 1.3.6. The SC-I classification applies to the principal equipment, systems and components that must perform safety functions during and/or after the DE to comply with the MFFF safety analysis.

SSCs with no assigned safety function in a seismic environment, but whose failure after a DE could adversely impact the ability of an IROFS equipment, system or component to perform its safety function, are designated as SC-II. SSCs designated as SC-II are designed such that its failure in a DE event does not adversely affect the performance of an IROFS.

Components that form the boundaries between SC-I and SC-II portions of SSCs are designed to SC-I requirements. The SC-I requirements are extended to either the first anchor point in the SC-II SSC or a sufficient distance into the SC-II SSC so that the SC-I analysis remains valid.

SSCs that are neither SC-I nor SC-II are not classified with respect to seismic category, and are categorized as CS.

1.1.2.2.1.2 Seismic Performance Requirement

The SPR is a functional classification which further categorizes SC-I and SC-II items by the type of safety function they must perform during and/or after the DE. Definitions for each SPR classification are provided below. SPR types C and B apply to SC-I SSCs. SPR type A is identical to SC-II. Unless noted otherwise, the safety functions must be performed both, during and after the DE.

- C2: Item must remain active during and after the DE
- C1: Item must remain active after the DE
- B3: Item must maintain pressure boundary integrity
- B2: Item must maintain structural integrity
- B1: Item must maintain general configuration of nuclear material
- A: Item must not fail in a way that compromises an IROFS by interaction (SC-II), including pressure boundary integrity for piping systems.

1.1.2.2.2 Seismic Analysis

SSCs classified as SC-I and SC-II are generally qualified to maintain structural integrity under seismic loading by analysis. Analysis methods are discussed below.

1.1.2.2.2.1 Seismic Design Parameters

SC-I and SC-II SSCs are generally qualified to maintain structural integrity under seismic loading based upon ground motion that meets the DE for the facility. The qualification is performed using in-structure acceleration response spectra corresponding to the point of structural attachment. In-structure response spectra are generated in accordance with one of the methods cited in Section 3.4 of the American Society of Civil Engineers' "Standard Seismic Analysis of Safety-Related Nuclear Structures" (ASCE 4-98). The in-structure response spectra are peak-broadened to account for uncertainties in the specification of the input parameters used to calculate the spectra.

1.1.2.2.2.1.1 Seismic Inertia Analysis

The inertial response of SC-I and SC-II SSCs to seismic loading is determined using either a response spectrum dynamic or equivalent static load method. General requirements for these methods are described below.

1.1.2.2.2.1.1.1 Response Spectrum Method

Response spectrum dynamic analyses used to qualify SC-I and SC-II items for seismic loading are performed in accordance with ASCE 4-98, Section 3.2.3. Elements of this method are as follows:

- **Mass/stiffness representation** – Analytical models incorporate the mass and stiffness characteristics that significantly affect dynamic response. Support stiffness that significantly affects dynamic response is considered in the SC-I analysis. Masses are lumped or distributed as appropriate. Modeling considerations given in ASCE 4-98 apply.

- **Dynamic loading** – Loading in the form of acceleration response spectra corresponding to structural support points is used to determine the equipment seismic inertial response. For SSCs supported at multiple points, the envelope of the acceleration response spectra corresponding to each support point is used as one optional approach to determine the inertial response.
- **Damping** – The effects of structural/system damping are incorporated in the analysis by selecting acceleration response spectrum curves that reflect an appropriate level of damping for the equipment response. Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," October 1973 is used for determining the structural damping for SSCs except for piping systems and supports. A constant damping value of 5% is used for piping systems in accordance with ASME Section III, Appendix N, DOE-STD-1020-2002 and ASCE/SEI 43-05.
- **Modal participation** – Vibratory modes below a rigid cutoff frequency of 33 Hz (or zero-period-acceleration frequency) are included in calculating the system dynamic response. In the event that the mass participating in the response in any of the three global directions is less than 90% of the total system mass, corrections are made to account for the response of the non-participating mass.
- **Modal combinations** – Modes with frequencies less than 33 Hz (or zero-period-acceleration frequency) are included. The residual rigid response, calculated using Equation 3.2.8 of ASCE 4-98, is combined with the total combined response of the modes with frequencies less than 33 Hz (or zero-period-acceleration frequency) using the square root of the sum of the squares (SRSS) method. The total combined response (due to loading in a given direction) of the modes with frequencies less than 33 Hz (or zero-period-acceleration frequency) is obtained by any of the applicable methods provided in Regulatory Guide 1.92. Alternately, the modal combination rule in Section 3.2.7.1-1(a) of ASCE 4-98 may be used.
- **Spatial combinations** – The seismic analysis considers the combined effects of seismic loading in both principal horizontal directions acting concurrently with loading in the vertical direction. The responses in the two horizontal and one vertical direction are calculated independently and then combined. Spatial responses are combined using the square root of the SRSS method.

1.1.2.2.2.1.1.2 Equivalent Static Method

SC-I and SC-II elements that are adequately represented by a single-degree-of-freedom or simple multiple-degree-of-freedom model may be analyzed using the equivalent static analysis method in accordance with ASCE 4-98, Section 3.2.5. Elements of this procedure are as follows:

- **Design load** – The seismic design load applied to an element in a given direction is determined by multiplying the mass of the element by the peak acceleration from the applicable floor response spectrum, and amplified by a factor taken from ASCE 4-98, Section 3.2.5.
- **Spatial combinations** – The seismic analysis considers the combined effects of seismic loading in both principal horizontal directions acting concurrently with loading in the

vertical direction. The responses in the two horizontal and one vertical direction are calculated independently and then combined using the SRSS method.

1.1.2.2.2.1.2 Seismic Anchor Movement Analysis

Where SSCs are supported at multiple points, the effects of the relative displacements of these support points, under the design basis seismic condition, are incorporated in the overall seismic response calculation. The effect of seismic anchor movements is calculated by applying the worst-case combination of peak support point displacements. The results of the seismic anchor movement analysis are combined with the results of the seismic inertia analysis, using the SRSS method, to determine the overall seismic response.

1.1.2.2.2.1.3 Seismic Qualification Requirements

Requirements for the seismic qualification of SC-I, SC-II, and CS SSCs, including methods and acceptance criteria, are presented below.

1.1.2.2.2.1.3.1 Qualification of C2 and C1 SSCs for Operability

Qualification of active SC-I items to perform required functions during and/or after the DE (i.e., C2 classification) or after the DE only (i.e., C1 classification) is typically demonstrated by analysis or by shake table testing, or by a combination of these methods. Shake table testing is required for cases where analysis alone is insufficient to ensure operability after the seismic event (e.g., for electrical components). Qualification by analysis, in general, uses the appropriate design code allowable stress as acceptance criteria. Qualification by test, in general, uses a demonstration of operability during and after exposure to the specified level of shaking on a shake table.

1.1.2.2.2.1.3.2 Qualification of B3 SSCs for Pressure Boundary Integrity

Qualification of SC-I/B3 items to maintain pressure boundary integrity during and/or after the DE is typically demonstrated by analysis. Qualification by analysis ensures that stresses in system pressure boundary elements and their supports meet allowable stresses, based on the stress limits specified in the applicable design codes for the stipulated seismic loading combinations.

1.1.2.2.2.1.3.3 Qualification of B2 SSCs for Structural Integrity

Qualification of SC-I/B2 items to maintain structural integrity during and/or after a DE is typically demonstrated by analysis. Qualification by analysis ensures that stresses in the structural elements in the load path calculated under DE loading conditions meet allowable stresses for the seismic loading combination from the applicable design code.

1.1.2.2.2.1.3.4 Qualification of B1 SSCs to Maintain Nuclear Material Configuration

Qualification of SC-I/B1 items to maintain nuclear material configuration during and/or after the DE is demonstrated by analysis. Qualification by analysis ensures that deflections in the elements in the load path, calculated under DE seismic loading conditions, do not exceed the specified critical dimensions for the system or component.

1.1.2.2.2.1.3.5 Qualification of Category A SSCs to Prevent Adverse Interactions

Qualification of SC-II /A items to prevent adverse interactions with IROFS components during and/or after the DE is demonstrated by analysis. Qualification by analysis ensures that stresses in the elements in the load path calculated under DE seismic loading conditions do not exceed allowable stresses of the system or component specified in the applicable design code, and no adverse physical interaction with an IROFS system or component occurs.

1.1.2.2.2.1.3.6 Qualification of Conventional Seismic SSCs

SSCs classified as CS are designed to resist seismic loading in accordance with the applicable consensus design code or standard. Qualification of conventional seismic items is for worker life safety concerns, good engineering practice, and protection of the capital investment. Seismic loads are calculated in accordance with the guidance in the UBC or IBC. Loading combinations and stress criteria are taken from the design code or standard applicable to the item, or alternatively, from the UBC or IBC.

1.1.2.2.3 Seismic Qualification of Suspended Systems

There are three types of suspended systems that are seismically qualified: process piping systems, HVAC duct systems, and electrical cable trays and conduit systems. The qualification of each of these systems is addressed in the following sections.

1.1.2.2.3.1 Piping Systems

Seismic qualification requirements for piping systems or fluid transport systems (FTS) are based on the nature of the fluid contained, quality level, and safety function as described in Section 11.7. Piping that is IROFS is seismically categorized as SC-I with an SPR of B3. Piping internal to the MFFF and that is non-IROFS is seismically categorized as SC-II with an SPR of A.

Piping systems are designed for pressure, deadweight, thermal, occasional anticipated loads, seismic inertia and seismic anchor motion loads, per ASME B31.3 Process Piping code. To maintain the nuclear material configuration, the deflections caused by temperature differential and seismic inertia are maintained within specified critical dimensions.

The deadweight analysis considers the effects of the dead weight of piping and inline components, insulation, and fluid contents, or other permanent sustained loads imposed on the system. The pressure analysis considers the effects of the internal design pressure acting on the system, which includes longitudinal pressure stresses in the pipe walls and unbalanced thrust loads at the joints. The effects of thermal expansion or contraction of the piping system, including thermal displacements at equipment anchor points or other support points, caused by temperature change from ambient to the maximum operating temperature of the system, are evaluated for flexibility.

The inertial effects of the DE on SC-I and SC-II piping systems are evaluated using floor and wall acceleration response spectra corresponding to the attached support locations with proper damping values, previously described. The ASME B31.3 code includes the effects of seismic loads in the stress analysis of the piping system, and hence confirms that the piping system remains within the limit of the allowable stresses prescribed by the code and meets the safety requirements.

Outdoor piping systems are subject to occasional loads per ASME B31.3 and evaluated per required criteria. The relative seismic anchor movement of support points located in different structures is considered out of phase and accounted for in stress evaluation.

Double contained piping is constituted of inner and outer pipe and is analyzed in a single computer model to ensure that the effects of both pipes together are considered for qualification. The buried jacket piping subjected to seismic ground motion, seismic wave passage, soil settlements, permanent and transient seismic anchor motion, and the effects of differential thermal expansion between the inner piping and jacket piping are designed and qualified within the limits of allowable stresses per the ASME B31.1 and B31.3 codes.

In-line piping components such as flanges, valves, and expansion joints, are subject to the same seismic qualification requirements as the piping systems in which they are installed. In addition, they meet specific requirements where necessary. In-line piping flange connections are evaluated in accordance with ASME B31.3. In-line valves are qualified for pressure boundary per ASME B31.1 or B31.3 and verified for allowable valve accelerations and valve nozzle loads provided by the respective vendors. Piping reactions at terminal equipment nozzle connections are identified and qualified per ASME Section VIII, Divisions 1 and 2, WRC Bulletin 107, and/or vendor provided allowable nozzle loads information. Local stresses arising from integral welded attachments such as trunnions or lugs to piping are evaluated per ASME Code Case N 318 and N 392, WRC Bulletin 107, and Finite Element Analysis (FEA). The piping displacements and penetration seals for fire protection, air confinement, and radiation shielding, are kept within the allowable movement at the seal provided by the vendor.

1.1.2.2.3.2 HVAC Duct Systems

HVAC systems and components within the facility provide the required operating environment and minimize the release of contamination to the atmosphere. IROFS HVAC duct systems, identified as SC-I, and non-IROFS duct systems, identified as SC-II, are seismically qualified to withstand the DE.

HVAC systems and components identified as SC-I, such as Very High Depressurization Exhaust System exhaust system piping, are qualified in accordance to the ASME B31.3 Process Piping code. Structural supports for these SC-I piping are qualified in accordance with the ANSI/AISC N-690 code.

Round and rectangular SC-I and SC-II HVAC ductwork are evaluated for structural integrity by a visual inspection to confirm that duct material, stiffeners, and joints in accordance with Sheet Metal and Air Conditioning Contractors' National Association (SMACNA)-1980, Rectangular Industrial Duct Construction Standards and SMACNA-1999, Round Industrial Duct Construction Standards.

Square groove butt joints used to close inspection slots will not be visually inspected for complete joint penetration as required by AWS D9.1-2006, section 8.2, due to the ideal joint preparation necessary for the use of these inspection slots and successful use of GTAW welds to develop complete joint penetration.

Structural supports for SC-I and SC-II ductwork are qualified in accordance with ANSI/AISC N-690 code.

HVAC ductwork and their supports whose seismic category is identified as CS are qualified for structural integrity following methodology in DOE Seismic Evaluation Procedure, DOE EH-0545 for seismic loading as per the UBC or IBC.

SC-I and SC-II ductwork is analyzed for seismic loading using the equivalent static method or response spectrum dynamic analysis procedure. Dynamic analysis of ductwork systems uses a 4% damping ratio in calculating system seismic response in accordance with RG-1.61.

To ensure adequate performance, duct pressure boundary integrity (when applicable), and structural integrity of ductwork and supports during normal and seismic conditions, the ducts will be constructed to the applicable requirements from SMACNA standards.

1.1.2.2.3.3 Electrical Raceway System

The electrical power system provides power to designated equipment during normal operation, abnormal operation, design basis accident conditions, and during the loss of offsite power event. Electrical raceway systems include electrical cable trays and conduits.

Electrical raceways and their structural support frames are components of the electrical system that provide structural support and mechanical protection from damage for the cables contained within them and as such their structural designs are based on seismic design categories and SPR designation. The seismic design categories of SC-I, SC-II and CS and the SPR are determined from electrical raceway drawings. A single or combination of analytical and/or test methods per design codes referenced below may be used to ensure the structural adequacy of the electrical raceway system to withstand the seismic design loads commensurate with their seismic category and performance requirements. SC-I underground duct bank is evaluated for the effects of gross deflection due to DE seismic wave passage on conductor integrity, where judged to be significant.

Cable trays satisfy as a minimum the manufacturing requirements in accordance with NEMA VE 1. NEMA VE 1 does not address the effects due to seismic load for design of cable trays. AISI code for the design of cold-formed carbon steel or ASCE 8-02 for the design of cold-formed stainless steel is used to ensure the structural adequacy of cable trays and wireways due to seismic effects from DE. In lieu of analysis, cable trays and wireways can be qualified by testing in three directions to determine the allowable load for each direction.

Conduits are designed for various loads such as deadloads, liveloads, windloads, earthquake, tornado, etc., applicable load combinations, and stress limit coefficients in accordance with AISC N690.

Seismic analyses of SC-I and SC-II raceways utilize the methodology prescribed in Section 1.1.2.2. The seismic loads are computed using the response spectra corresponding to the modal damping percentages of 7% and 10% for conduits and cable tray systems, respectively.

The seismic responses determined from the analyses are combined with responses from other loads to determine the most critical load condition. Stresses under the normal and extreme loading conditions are determined using the load combinations of ANSI/AISC N690-1994 Supplement 2.

In general, enveloped peak acceleration values from the applicable in-structure response spectra and supported span masses are used to determine the DE loads for structural design of the raceway, in addition to the dead weight attributed to the support structure.

Combined weights of cable, cable trays or conduits, fittings, splices, side rail extensions, and covers as applicable, are included as dead loads for the supported system. For cable trays and support designs, a concentrated load of 200 pounds applied at the mid-span, where it produces the most severe stress condition in the cable tray and supports, is considered. This load is not combined with the DE effects.

1.1.2.2.4 Seismic Qualification of Suspended System Supports

Supports for suspended systems and equipment, including piping, HVAC ducts, electrical raceways, instrumentation, and electrical equipment are classified according to the seismic performance requirement of the system and components. Supports for SC-I and SC-II are qualified for seismic loads by analysis.

The seismic loading for SC-I and SC-II supports is determined using the in-structure response spectra that correspond to the elevation of the structural attachment point in the facility, and uses envelope spectral values for generic support designs. The inertial response of the supports is determined using Equivalent Static Analysis method previously described. The Response Spectrum Dynamic Analysis method is also utilized for specific designs. In addition to the dead load and seismic loads, other piping loads such as thermal and occasional loads are considered in the support design. Supports are designed to ANSI/AISC N690-1994 S2 load combinations and design criteria. The load combinations considered in the design and analysis are Normal and Extreme load combinations. The supports are attached to the building via surface mounted or embedded plates. The supports in most areas are made of carbon steel, but supports in process cell areas are made of stainless steel to provide corrosion resistance. Welded connections are designed with AISC N690 and AWS D1.1 for carbon steel and AWS D1.6 for stainless steel.

Local stresses at integral attachments to pipe are evaluated as part of the pipe stress analysis. Integral welded attachments per ASME B31.3 are evaluated using AISC N690.

1.1.2.2.4.1 Pipe Support Loads and Attachments

Manufactured pipe support components are designed and fabricated in accordance with MSS SP-58 to ensure that the supports meet seismic performance requirements. Integral and non-integral attachments to piping are designed in accordance with ASME B31.3. Design loads applied to support components are maintained less than manufacturer specified allowable loads for a given temperature. If no manufacturer allowable load is available, components are evaluated for the maximum calculated capacity in accordance with AISC N690.

For pipe support components representing engineered items, seismic calculations are prepared to determine the allowable load capacity, based on maximum piping design temperature. Design loads are ensured to be maintained less than allowable load capacity. Specifications are prepared to implement fabrication and installation requirements.

Guidance for the design of connections and member properties for Hollow Structural Sections (HSS) given in AISC Connections Manual for Hollow Structural Sections is used.

1.1.2.2.4.2 HVAC and Raceway Support Loads

The seismic design requirements for SC-I and SC-II supports of the cable tray and HVAC duct systems require the consideration of seismic load from the DE in addition to dead load and maintenance live load as specified.

1.1.2.3 General Qualification of Systems and Components

The ISA identifies the IROFS required to provide mitigation or prevention of DBE. This section identifies general requirements for qualifying systems and components, otherwise referred to as items, to perform required IROFS functions during evaluated event sequences. These qualifications support the demonstration that the consequences of event sequences satisfy the performance requirements of 10 CFR §70.61.

Seismic qualification requirements applicable to systems and components are discussed in Section 1.1.2.2. Qualification requirements for building structures are addressed in Section 1.1.2.1.

The qualification requirements that generally apply to equipment, systems, and components are based upon the general safety functions that items must be qualified to perform, the classification systems used to determine qualification requirements applied to a particular item, the general event sequences that items are qualified for, and the design code based loading combinations and acceptance criteria used to establish acceptability. A discussion of the analysis methods used to determine the seismic response of equipment, systems, and components is also provided. Descriptions, specific safety and non-safety functions, and the particular codes and standards applied to the various equipment, systems, and components are identified in other sections [e.g., Section 11.3 (HVAC and confinement systems), Section 11.6 (material handling equipment), Section 11.7 (fluid transport systems), etc.].

1.1.2.3.1 Qualification Process

The qualification process involves identifying the required functions that must be performed, the conditions under which the functions must be performed, and the analysis or test method and acceptance criteria that establishes the capability to perform the function. The applied loads, methods, and acceptance criteria used to qualify items vary, with more stringent requirements applied to items that perform functions of relatively higher importance to safety. The general safety functions from which specific structural performance requirements are derived for each item are provided in Section 1.1.2.3.1.1. These functions are classified with respect to Quality Levels defined in the MOX Project Quality Assurance Plan. Additional classifications used to establish the qualification method (i.e., analysis or test), and acceptance criteria used to evaluate items under seismic loading are provided in Section 1.1.2.3.1.2. The general event sequences or scenarios under which equipment must be qualified are identified in Section 1.1.2.3.1.3.

1.1.2.3.1.1 Safety Functions

Equipment, systems, and components classified as IROFS or QL-2 are designed and qualified to perform the general safety functions listed below:

- Confine radioactive or toxic materials by maintaining the integrity of the pressure boundary used to confine the material in the elastic range. (IROFS)
- Confine radioactive or toxic materials by maintaining structural integrity of the confinement boundary elements in the elastic range. (IROFS) (Structural integrity implying the load carrying capability of the structural elements and connections, and the retention of the geometric configuration of the joints, including seal compression as applicable, but not necessarily to mean pressure boundary integrity).
- Prevent criticality by maintaining integrity of nuclear material handling equipment structural elements, nuclear material container supports, and neutron absorbers in the elastic range (IROFS)
- Prevent interactions between confinement boundaries or criticality prevention elements, and non-safety systems and equipment that could damage them (QL-2).

Criticality prevention functions and confinement boundary integrity functions must be performed under normal operating conditions, credible accident scenarios, and design basis natural phenomena conditions to the extent credited in individual event sequences.

1.1.2.3.1.2 Classification Systems

The classification systems used to determine the qualification requirements applied to individual items include the Quality Level system identified above, as well as SC, SPR, and hoisting equipment type. A discussion of the SC and SPR classifications is provided in Section 1.1.2.2. A discussion of hoisting equipment type classifications and their effects on qualification requirements are presented below.

1.1.2.3.1.2.1 Hoisting Equipment Type

Cranes and hoists described in Section 11.9 are classified as one of the following different types based on the safety function that has to be performed:

- Type I – cranes and hoists that must retain payload during and after credible accident and design basis natural phenomena events in order to meet the performance requirements of 10 CFR §70.61

- Type II – cranes and hoists that need not retain payload, but have to maintain structural integrity during and after credible accident and design basis natural phenomena events in order to meet the performance requirements of 10 CFR §70.61
- Type III – cranes and hoists that need not retain payload, nor maintain structural integrity during and after credible accident and design basis natural phenomena events in order to meet the performance requirements of 10 CFR §70.61.

Hoisting equipment type is used to define the scope, loading combinations, and acceptance criteria used to qualify hoisting equipment components in accordance with the applicable design codes.

1.1.2.3.1.3 Loading Conditions

1.1.2.3.1.3.1 Normal Operating Conditions

Normal operations include the range of plant conditions when the process equipment is either producing fuel, is down for maintenance, or is operational but idle. Normal operating loads on items include dead loads, live loads including dynamic mechanical drive loads, design pressure loads, and normal operating thermal expansion and reactions from interfacing equipment or systems.

1.1.2.3.1.3.2 Accident Conditions

Accidents include the range of off-normal events that challenge the performance of required safety or process functions. General descriptions of the events are provided below. The particular events and associated process parameter magnitudes that IROFS are qualified for are identified in the specific event sequence descriptions and supporting analyses as described below.

1.1.2.3.1.3.2.1 Pressure Excursions

Overpressurization events are postulated due to failure of control devices in pressurized gas supplies to gloveboxes. Passive flow control devices installed to prevent the overpressurization events will be qualified to reduce excess gas supply flows to acceptable levels.

1.1.2.3.1.3.2.2 Temperature Excursions

Temperature excursion events which induce thermal expansion stresses in gloveboxes and other equipment required for confinement and criticality prevention are postulated due to failure of process equipment or ventilation systems. Items required to maintain structural integrity under such conditions will be qualified by structural analysis.

1.1.2.3.1.3.2.3 Equipment Impacts

Impact events are postulated where material handling equipment, drive mechanisms, or controls fail, causing an interaction between the equipment and confinement boundary or criticality prevention elements. Either the equipment, mechanical stops or barriers, or the confinement boundary or criticality prevention elements will be qualified to withstand the impact in order to ensure performance of the safety function.

1.1.2.3.1.3.2.4 Load Drops

Load drop events are postulated where material handling equipment fails during normal operations, releasing its payload, or components inside of gloveboxes fall as a result of an earthquake, impacting confinement boundary or criticality prevention elements. The resulting interactions may breach the confinement boundary or compromise the criticality prevention element. Either the material handling equipment or component supports must be qualified to retain their loads, or the confinement boundary or criticality prevention elements will be qualified to withstand the impact.

1.1.2.3.1.3.3 Design Basis Natural Phenomena Conditions

Safety functions must be performed under all design basis natural phenomena conditions to the extent credited in individual event sequences. With the exception of the DE, the MFFF building structures provide protection of IROFS against the affects of natural phenomena events. The DE has the potential to affect the ability of process equipment, systems, and components to perform required safety functions. The DE event is described in Section 1.3.6. Seismic qualification analyses are performed using in-structure acceleration response spectra, which characterize the response of vibratory systems with varying amounts of structural damping to seismic motions calculated throughout the building structure. The in-structure acceleration response spectra are catalogued with corresponding room references for use in qualifying specific systems and components.

1.1.2.3.2 Qualification by Analysis

1.1.2.3.2.1 Static Analysis

Static analysis is generally used to demonstrate that items are capable of performing required structural integrity or pressure boundary integrity functions under normal operating conditions. Loading combinations and acceptance criteria are derived from applicable design codes.

1.1.2.3.2.2 Impact Analysis

Items required to withstand the effects of impacts are generally qualified by analysis, using quasi-static or dynamic methods. Quasi-static methods are used in cases where the loading duration is much greater than the natural period of vibration of the impacted structure, or the mass of the impacted structure is negligible. Under these conditions, structural response is determined using an energy balance approach, where kinetic energy from the impactor is converted to strain energy in the affected components.

Impact qualifications are performed based on structural design code loading combinations and acceptance criteria presented in the form of ductility ratios.

1.1.2.3.2.3 Seismic Analysis

Items classified as SC-I and SC-II are generally qualified to maintain structural integrity under seismic loading by analysis. Analysis methods and design parameters are discussed in Section 1.1.2.2.

1.1.2.3.3 Qualification by Testing

Testing is also used to qualify items to perform required functions under specified conditions. In general, the test conditions must bound the conditions under which the items must perform the required functions. Acceptance criteria for qualifications by testing are performance based.

1.1.2.3.3.1 Leak Testing

Leak testing is performed to ensure that items required to maintain pressure boundary integrity have been designed and fabricated to meet that objective. Test conditions, methods, and acceptance criteria are provided in the applicable pressure system design code.

1.1.2.3.3.2 Shake Table Testing

Shake table testing is typically performed to ensure that components which must perform active functions or remain operable during and/or after earthquakes will do so. Testing is typically conducted in accordance with Institute of Electrical and Electronics Engineers (IEEE)-344, based on input required acceleration response spectra defined for the seismic environment in which the component will be installed. The acceptance criterion normally consists of a demonstration of the functional capability of the component after being subjected to the test motions.

1.1.2.3.3.3 Drop Testing

Drop testing is generally performed to demonstrate either the structural capability of a material container or of a drop target under a specified load drop scenario. Test parameters typically include drop height, load mass, target configuration and performance based acceptance criteria related to container leakage or target structural integrity.

1.1.2.3.4 Qualification of Process Equipment

Process equipment includes the gloveboxes, material handling equipment, hoppers, mixers, mills, presses, furnaces, grinders, and other equipment used to process nuclear materials in a dry powder, pellet or rod form in the MOX PA.

1.1.2.3.4.1 Process Gloveboxes and Structural Elements

1.1.2.3.4.1.1 Design Codes for IROFS and QL-2

Process gloveboxes and structural elements, including plates, beam members, and structural connections, which are classified as IROFS or QL-2, are designed and qualified to maintain structural integrity in accordance with general design codes and standards.

Stresses in IROFS and QL-2 structural elements are evaluated by calculating maximum axial, bending, and shear stress components and comparing them to allowable stress values and interaction equations for the applicable loading combination from the ANSI/AISC N690 design code.

IROFS and QL-2 structural elements fabricated from materials not specified in the ANSI/AISC N690 design code are qualified using allowable stress values calculated based on material certified yield and ultimate strength properties.

IROFS and QL-2 structural elements are qualified to maintain structural integrity under normal operating conditions. This includes thermal expansion at normal operating temperatures and loading due to maintenance activities, accidental loading due to impacts or environmental transient conditions, and design basis natural phenomena, by stress analysis in accordance with the ANSI/AISC N690 design code. Stresses due to thermal expansion during normal operations are considered secondary stresses and are evaluated using design code provisions for secondary stresses. Thermal expansion under accidental environmental transient conditions is evaluated using either the allowable stress criteria or deformation criteria applicable to the abnormal load combination documented in section Q1.5.8 of the ANSI/AISC N690 design code. IROFS structural elements are evaluated for seismic loading in accordance with the Extreme loading combination. Accidental thermal expansion effects are evaluated using the Abnormal Extreme loading combination. QL-2 structural elements are evaluated for seismic loading using allowable stresses from the Abnormal Extreme loading combination.

For the heat treatable materials with tensile and yield strengths that are not specified in the ASTM specifications, values are specified in the design documents per agreement between the licensee and vendor.

1.1.2.3.4.1.2 Design Codes for QL-3 and QL-4

Process equipment structures and structural elements classified as QL-3 or QL-4 are designed and qualified to meet applicable structural integrity requirements provided in general design codes and standards.

1.1.2.3.4.1.3 Supplemental Qualification Requirements

1.1.2.3.4.1.3.1 Structural Plate Elements

Structural plates subject to non-cyclical loads shall either be qualified using the design code stress equations or by calculating the maximum design stress intensity (S_{INT}) and comparing it to the applicable allowable stress value from the table below. Stress intensity is defined as the difference between the maximum and minimum principal stresses at a given point.

Design Code	Loading Combination	Plate Surface	Allowable Stress[1]
AISC N690	Normal / Primary	Mid	$S_{INT} < 0.6\ F_Y$
		Top & Bottom	$S_{INT} < 0.75\ F_Y$
	Normal / Primary + secondary	Mid	$S_{INT} < 0.9\ F_Y$
		Top & Bottom	$S_{INT} < 1.12\ F_Y$

	Extreme	Mid	$S_{INT} < 0.96\ F_Y$
		Top & Bottom	$S_{INT} < 1.20\ F_Y$
	Abnormal Extreme	Mid	$S_{INT} < 1.02\ F_Y$
		Top & Bottom	$S_{INT} < 1.28\ F_Y$
AISC-ASD	Normal	Mid	$S_{INT} < 0.6\ F_Y$
		Top & Bottom	$S_{INT} < 0.75\ F_Y$
	Seismic	Mid	$S_{INT} < 0.8\ F_Y$
		Top & Bottom	$S_{INT} < 1.0\ F_Y$

[1] F_Y is the plate material yield strength at temperature.

1.1.2.3.4.1.3.2 Threaded Fasteners

Thread shear stresses in tapped holes and fasteners installed into tapped holes are evaluated using a suitable thread shear area and effective engagement length, and an allowable shear stress given by:

$$F_V = F_T / \sqrt{3}$$

$$F_T = 0.6\ F_Y \quad \text{(for tapped hole threads)}$$

$$F_T = \text{the lesser of } 0.33\ F_U \text{ (for fastener threads) or } 0.346\ F_y.$$

where:

F_V = allowable thread shear stress

F_T = allowable stress in tension

F_Y = material yield stress in tension

F_U = fastener ultimate stress in tension.

The process equipment designs include threaded fasteners that deviate from the AISC N690 or AISC-ASD structural design code specifications as follows:

- Fasteners fabricated from materials not approved for use under the code
- Fasteners fabricated to standard metric dimensions
- Fasteners with diameters less than the smallest diameter for which data is tabulated.

These threaded fasteners are evaluated in accordance with the design code requirements that would otherwise apply to fasteners that do not deviate from the code requirements.

1.1.2.3.4.1.3.3 Structural Fastener Minimum Edge Distance

Structural fasteners with minimum edge distances less than the allowable values specified in the structural design code, those with metric fasteners, or those with fastener sizes not specifically addressed in the code, are qualified for loads applied toward the edge of the plate using the following equation:

$$L_e/D \geq [0.5 + 1.733(f_p/F_u)].$$

where:

L_e = minimum edge distance

D = fastener diameter

f_p = bearing stress applied to the projected area of the plate

F_u = ultimate tensile stress of the plate.

The limitations on applying this equation are listed below:

- The ratio of minimum edge distance to fastener nominal diameter meets or exceeds a value of 1.0.
- The ratio between the ultimate strength of the plate material in shear, and the ultimate strength of the material in tension meets or exceeds a value of 0.577.
- Only standard hole diameters are permitted. For bolts with U.S. dimensions, standard hole diameters are as designated in the RCSC Specification for Structural Joints Using ASTM A325 or ASTM A490 Bolts. For bolts with metric dimensions, standard hole diameters are the nominal clearance holes specified in Machinery's Handbook, 26th Edition.
- A minimum of two fasteners along the edge, spaced in accordance with design code requirements, are used.
- The fastener head (bolt or nut) does not extend beyond the edge of the plate.
- The criterion is not applied to connections subject to vibration unless a specific fatigue evaluation is performed to establish the suitability of the material for the applied stress.
- Additional shear stress on the critical shear planes, such as that caused by prying action on the fastener, is combined with the tear-out stress to obtain new criteria.

The minimum edge distance defined above applies to plate edges that are rolled or laser, water jet, plasma, or gas cut. For shear edges, an additional allowance must be added to the minimum edge distance caused by shearing.

1.1.2.3.4.1.3.4 Concrete Anchorage

The anchorage of process equipment to concrete is evaluated in accordance with the criteria provided in Section 1.1.2.1.

1.1.2.3.4.1.3.5 Glovebox Shell and Frame Elements

Gloveboxes designed such that the shell constitutes the principal load bearing mechanism are evaluated to ensure that buckling does not occur under the worst case combination of normal operating, accident, and natural phenomena loading. In lieu of more definitive procedures, shell glovebox corners and sections between windows are evaluated for buckling under bounding loads using a column buckling approximation. The analysis considers applied loads that induce compressive stresses in those glovebox sections subject to buckling.

Localized stresses in the vicinity of nozzles or other structural discontinuities in glovebox shells are evaluated in accordance with stress criteria developed to ensure that local strains remain below allowable ductility ratios specified in ANSI/AISC-N690 design code.

1.1.2.3.4.1.3.6 Glovebox Window Panel Assemblies

Glovebox window panel assemblies, which include window panels, gaskets, clamps, and fasteners, are qualified to maintain confinement boundary integrity under normal operating and design basis natural phenomena loading by stress analysis. The analysis considers applicable loads from the Normal and Extreme loading combinations defined in the ANSI/AISC N690 design code. Maximum calculated tension, shear, and bending stresses in the window panels are compared to allowable stress values. Allowable values are based on the window panel certified yield strength in tension multiplied by the allowable stress factors applicable to polycarbonate used in normal industrial practice. Stresses in clamps and fasteners are compared to allowable stress values calculated in accordance with the ANSI/AISC N690 design code.

Glovebox windows consist of polycarbonate panels which clamp into frames and are sealed using channel shaped or flat elastomer gaskets. Gloveports and bagports mount in perforations cut into the window panels. The panels are fabricated from 9.5 millimeters (mm) thick (nominal) sheets of Lexan MR-10 polycarbonate, furnished to certified values of yield strength.

Glovebox window panel assemblies have been qualified to maintain confinement boundary integrity under impact loads from missiles with total kinetic energies up to 500 Joules by dynamic test. The test program was designed to measure the energy absorption capacity of bounding glovebox window designs under impact loads at failure under panel fracture or edge disengagement failure mechanisms. Key test parameters controlled include impactor energy, impactor velocity, impactor orientation, impactor point radius of curvature, window panel geometry, perforation geometry, window assembly materials, and window fastener installation procedure. Generation of missiles internal or external to the glovebox is precluded by design.

1.1.2.3.4.1.3.7 Glovebox Penetrations

Machine components in glovebox mechanical rotary or linear penetrations are qualified to maintain structural integrity in accordance with the requirements in the integrated safety analysis.

Pressure bearing components in glovebox fluid system penetrations are qualified to maintain pressure boundary integrity in accordance with the requirements in the integrated safety analysis.

1.1.2.3.4.1.3.8 Glovebox Expansion Joints

Glovebox expansion joints used in IROFS confinement boundary applications are qualified for use in confinement boundary applications in accordance with the performance criteria provided in the *Standards of the Expansion Joint Manufacturers Association, Inc., Sixth Edition.*

1.1.2.3.4.1.4 Fatigue Qualification Requirements

Structural elements subject to cyclical loading, as defined in the structural design codes during normal operations, are evaluated for the effects of fatigue in accordance with the following provisions and supplemental criteria.

1.1.2.3.4.1.4.1 General Structural Elements

IROFS and QL-2 carbon steel structural beam and plate elements are evaluated for the effects of fatigue in accordance with Appendix QB of the ANSI/AISC N690 design code. IROFS and QL-2 stainless steel structural beam and plate elements are also evaluated for the effects of fatigue in accordance with Appendix QB of the ANSI/AISC N690 design code.

QL-3 and QL-4 carbon steel structural beam elements and plate elements are evaluated for the effects of fatigue in accordance with Appendix K4 of the AISC ASD design code.

1.1.2.3.4.1.4.2 Structural Plate Elements

Structural plate elements evaluated based on the design stress intensities are evaluated for the effects of fatigue loading by ensuring the amplitude of the alternating stress remains below the allowable values given in Appendix 5 of the ASME Boiler and Pressure Vessel Code (B&PV) Code, Section VIII, Division 2, 1998 Edition.

1.1.2.3.4.1.4.3 Structural Welds

IROFS and QL-2 structural welds on carbon steel and metals other than stainless steel are evaluated for the effects of fatigue loading in accordance with the AWS D1.1-98 structural welding code and the factored loading combination provisions of the ANSI/AISC N690 structural design code.

QL-3 and QL-4 structural welds on carbon steel and metals other than stainless steel are evaluated for the effects of fatigue loading in accordance with the AWS D1.1-98 structural welding code and the factored loading combination provisions of the AISC-ASD structural design code.

IROFS and QL-2 structural welds stainless steel are evaluated for the effects of fatigue loading in accordance with the AWS D1.6-1999 structural welding code and the factored loading combination provisions of the ANSI/AISC N690 structural design code. Maximum resultant shear stresses, calculated in fillet and partial penetration welds, are compared to allowable stress values equal to the lesser of the code allowable stress for non-cyclically load welds, or 0.5 times the stress intensity value obtained from the Design Fatigue Curves given in the ASME B&PV Code, Section VIII, Division 2, 1998 Edition.

QL-3 and QL-4 structural welds stainless steel are evaluated for the effects of fatigue loading in accordance with the AWS D1.6-1999 structural welding code and the factored loading combination provisions of the AISC-ASD structural design code. Maximum resultant shear stresses, calculated in fillet and partial penetration welds, are compared to allowable stress values equal to the lesser of the code allowable stress for non-cyclically load welds, or 0.5 times the stress intensity value obtained from the Design Fatigue Curves given in the ASME B&PV Code, Section VIII, Division 2, 1998 Edition.

1.1.2.3.4.1.4.4 Structural Fasteners

IROFS and QL-2 structural fasteners, used in fatigue loading applications, are subject to design and evaluation provisions given in Appendix QB of the ANSI/AISC N690 structural design code.

QL-3 and QL-4 structural fasteners, used in fatigue loading applications, are subject to design and evaluation provisions given in Appendix K4 of the AISC-ASD structural design code.

1.1.2.3.4.2 Process Equipment Platforms

Structural platforms designed as integral parts of process gloveboxes or equipment are designed for loading for walkways and elevated platforms other than exit ways from ASCE-7-1998, *Minimum Design Loads for Buildings and Other Structures*.

IROFS and QL-2 platform structural elements are evaluated by stress analysis in accordance with the ANSI/AISC N690 design code. QL-3 and QL-4 platform structural elements are evaluated by stress analysis in accordance with the AISC-ASD design code.

1.1.2.3.4.3 Process Fixed Geometry Material Handling Equipment

Fixed geometry material handling equipment includes process equipment designed to transfer payloads where the position of the payload is controlled by fixed mechanical elements. Fixed geometry equipment classified as IROFS or QL-2 is required to retain payloads, and/or maintain structural integrity of the equipment under loading conditions including normal operations, accidental equipment impacts, load drops, and the Design Earthquake. The loading conditions applicable to a particular piece of equipment are designated in the event sequences. Impact loads caused by the accidental failure of IROFS and QL-2 fixed geometry material handling equipment or controls are calculated based on credible equipment velocities. Seismic inertia loading for IROFS and QL-2 fixed geometry equipment is evaluated using on modal dynamic or equivalent static analysis as described in Section 1.1.2.2.

Structural members, plates, and fasteners in IROFS and QL-2 fixed geometry material handling equipment are designed and qualified to maintain structural integrity in accordance with the AISC N690 design code. Welded connections on this equipment are qualified in accordance with either AWS D1.1 (carbon steel) or AWS D1.6 (stainless steel), as modified by ANSI/AISC N690.

Sizing criteria for structural members, plates, and fasteners in QL-3 and QL-4 fixed geometry material handling equipment are provided in the AISC-ASD design code. Sizing criteria for welded connections are provided in AWS D1.1, as modified by the AISC-ASD design code.

Mechanical drive components in IROFS and QL-2 fixed geometry material handling equipment are designed and qualified for strength and fatigue loading during normal operations in accordance with CMAA-70. Mechanical drive components in fixed geometry material handling equipment are designed and qualified to retain payloads during and after design basis natural phenomena events in accordance with the Extreme Environmental loading combination and allowable stress criteria provided in ASME NOG-1.

Structural plate elements on process material handling equipment evaluated in accordance with AISC N690 may be qualified on the basis of calculated design stress intensities. Plates subject to cyclical loads are also qualified for the effects of fatigue loading.

1.1.2.3.4.4 Process Hoisting Equipment

Hoisting equipment is designed and qualified in accordance with those portions of conventional design codes which assure performance of required Type I, II or III functions. In cases where conventional codes lack criteria for qualifying the equipment under seismic conditions, supplemental design criteria from nuclear hoisting equipment design codes is applied.

Double girder bridge cranes are designed and qualified in accordance with CMAA-70. IROFS and QL-2 double girder bridge cranes are qualified to perform their Type I or II function under seismic loading in accordance with ASME NOG-1.

Single girder underhung bridge cranes are designed and qualified in accordance with CMAA-74. IROFS and QL-2 single girder bridge cranes are qualified to perform their Type I or II function under seismic loading in accordance with ASME NUM-1.

Patented track underhung bridge cranes and monorails are designed and qualified in accordance with ANSI MH27.1. IROFS and QL-2 patented track underhung bridge cranes and monorails are qualified to perform their Type I or II function under seismic loading in accordance with ASME NUM-1.

Monorails, jib cranes, trolleys, and hoists of varying quality levels are qualified to perform required functions in accordance with ASME NUM-1.

Hoisting equipment is qualified for the loading conditions specified in the applicable design code. Normal operating conditions include rated loads and drive inertias. Structural components are evaluated for maximum stress and stability, and mechanical components are evaluated for fatigue stresses using the acceptance criteria from the code. Accidental impact loads are also calculated and evaluated in accordance with the conventional design code. IROFS and QL-2 hoisting equipment is also qualified to perform its required Type I or II functions under seismic loading in accordance with the applicable ASME NOG-1 or ASME NUM-1 code. The seismic evaluation is conducted in accordance with the dynamic response spectrum analysis procedures described in Section 1.1.2.2. Seismic loading for the analysis is provided by acceleration response spectra corresponding to the crane rail attachment point to the building structure. Hoisting equipment is evaluated based on a maximum percent of critical damping ratio of 5%.

1.1.2.3.5 Qualification of Facilities Equipment and Components

The MFFF program identifies the components of a system as individual pieces of equipment. Facilities equipment includes welded equipment, ventilation fans and air handling units, compressors, chillers, diesel generators, and general facilities hoisting equipment not directly associated with the process. For the purpose of bounding the seismic qualification process, equipment has been grouped as identified in the following sections.

1.1.2.3.5.1 Qualification Requirements

1.1.2.3.5.1.1 Operating (Active) Equipment

Operating (active) equipment that have moving parts and are expected to perform certain functions in the seismic environment to which it is exposed. Typical operating (active) equipment includes fans, pumps, compressors, blowers, etc. Operating (active) equipment is procured to equipment procurement specifications which cite the design code requirements for the equipment. The equipment provided by the supplier is required to meet the seismic performance requirements in accordance with a MOX Services approved qualification program consistent with the MOX Services equipment qualification specification. The specification has adopted the IEEE 344 methodology and allows for the use of experience data for qualification, as long as the caveats and criteria set by the standards and guides associated with each experience dataset are carefully addressed. For operating (active) equipment with a performance requirement that it “remain active” during and after the design basis seismic event, refer to Section 1.1.2.2.2.1.3.1.

1.1.2.3.5.1.2 Welded Equipment

Welded equipment is process vessels of welded construction. Welded equipment includes slab tanks, annular tanks, drip pots, demisters, process columns, etc. The welded equipment is manufactured to meet ASME Section VIII requirements, which include seismic loads in their design. Welded components are qualified by appropriate analyses as per IEEE 344. This includes demonstrating that critical dimensions are not violated during the DE.

1.1.2.3.5.1.3 Conventional Vessels

Conventional vessels are process vessels of standard construction. Typical conventional vessels are water storage tanks, fuel storage tanks, feeding tanks, etc. Conventional vessels are procured to procurement specifications which cite the design code requirements for the equipment. Conventional vessels are qualified by analysis as per IEEE 344 or by comparable industry standards.

1.1.2.3.5.1.4 Miscellaneous Equipment

Miscellaneous equipment is procured to procurement specifications, which cite the design code requirements for the equipment. Typical components in this group are evaporators, condensers, heaters and coolers, etc. Miscellaneous equipment is qualified by analysis as per IEEE 344, or testing.

1.1.2.3.5.2 Qualification of Equipment and Components

The MFFF design includes a large variety of process and utility equipment necessary to support fuel fabrication and nuclear and personnel safety objectives. Equipment supports are designed to be rigid with a natural frequency of at least 33 Hz. Alternatively, equipment seismic response is determined by considering the effect of the support stiffness on the overall seismic response of the equipment/support system. The design codes and standards, seismic qualification requirements, and design and procurement quality assurance requirements applied to this equipment are determined based on the Quality Level, Seismic Category, and Seismic Performance Requirement classification. General analysis requirements specified based on seismic category and seismic performance classifications are provided in Section 1.1.2.2. Additional qualification requirements for equipment are given below.

Seismic qualification of SC-1 equipment is performed in accordance with IEEE-344. Acceptable qualification methods can be analytical, testing or a combination of both. The seismic analysis flowchart in IEEE-344 is used as guidance to determine the most suitable qualification method.

Qualification by analysis shall ensure that stresses calculated for required loading combinations are less than allowable stress limits given in the applicable design codes listed. Structural integrity requirements are satisfied by analyzing equipment and its anchorage to ensure that calculated stresses are within material elastic stress limits or the allowable stress limits given in the applicable design code and excessive deflections will not permit detrimental interaction with SC-I component/system.

MFFF equipment meets seismic design provisions of the following design codes:

- Diesel fuel oil storage tanks are qualified in accordance with the ASME Code, Section VIII, Division 1.
- Process Vessels (including compressed air receivers) are qualified in accordance with the ASME VIII, Divisions 1 and 2.

Equipment is analyzed for the following loading conditions specified in the governing design code, or as otherwise applicable:

- The analysis considers the effects of the design pressure of the system.
- The analysis considers the effects of the deadweight of the equipment, insulation or shielding, fluid contents or internal equipment, appurtenances, and pressure due to the static head of liquids.
- The analysis considers significant effects on pressure boundary or structural integrity of process materials contained or supported by the equipment, conveying equipment hook loads, static and/or sloshing fluid effects, or operating loads.

- The analysis considers the effects of temperature gradients and differential thermal expansion of equipment and equipment supports caused by the temperature change from an assumed ambient temperature of 70°F to the design temperature of the system.
- The analysis considers the effects of reactions from attached piping and equipment.
- SC-I and SC-II equipment is qualified for DE seismic loading characterized by the acceleration response spectra corresponding to its point of attachment to the building structure. Equipment qualified by test is subject to seismic excitation, which envelops the in-structure spectra in accordance with the requirements of IEEE-344. Equipment qualified by analysis is evaluated using the in-structure response spectra and either the modal dynamic analysis method or the equivalent lateral force method if the equipment is adequately represented using a simple model. Analytical models address the amplifying effects of flexible mountings or equipment frames in determining the dynamic responses of equipment or components. Seismic inertia loading for Conventional Seismic equipment are determined in accordance with the UBC or IBC.

Seismic Category I equipment is qualified for pressure boundary integrity, structural integrity, or operability in accordance with the loading combinations and allowable stress limits of the applicable design code. The UBC or IBC is used to qualify equipment for structural integrity where a conventional seismic categorization has been applied and where no specific design code otherwise applies. In lieu of a specific loading combination from an applicable design code, the worst case combination of deadweight, live load, thermal expansion, operating load, and seismic inertia and seismic anchor movement loadings is considered for equipment qualification.

1.1.3 Controlled Area Boundary

The MFFF has established access control measures in order to provide positive control for ingress/egress of personnel, vehicles and materials for the MFFF operational areas during normal and emergency conditions. These measures include a system of barriers and controls which include protective force members that enforce and control access based on operational/emergency needs and security threat levels. Boundaries are established to provide geographical locations or lines of demarcation. These boundaries include the Savannah River Site Boundary (Controlled Access), the MFFF Site Controlled Area Boundary, the Protected Area (Restricted Area) Boundary. Each of these areas requires specific access authorization in order to be granted access or remove material based on operational conditions.

The MFFF Site Controlled Area has two primary ingress/egress roadways which will be restricted during security/emergency events and controlled by MFFF Protective Force personnel. These areas, as depicted in Figure 1.1.2-2, are in accordance with 10 CFR §20.1003 and 10 CFR §70.61(f).

A Memorandum of Agreement between the SRS Protective Force contractor, MOX Services, DOE-EM, and NNSA establishes Command and Control requirements concerning the MFFF for all phases of operations.

1.1.4 Material Flow

1.1.4.1 Plutonium Oxide Feed Material

PuO_2 feed material, transported in approved shipping containers, is received in the Shipping and Receiving Area of the MOX Fuel Fabrication Building. The feed material is offloaded in the PuO_2 truck bay where the outer packaging is removed. The feed material is then moved to the MOX Processing (MP) Area for sampling and storage for process use. Material control and accounting and radiation protection functions are performed.

1.1.4.2 Depleted Uranium Oxide Feed Material

Depleted uranium oxide (DUO_2) feed material, which is packaged in drums and shipped by truck, is received and stored in the DUO_2 storage area of the Secured Warehouse Building. Onsite vehicles transfer DUO_2 to the truck bay in the Shipping and Receiving Area, as needed in the MP Area.

1.1.4.3 Completed Fuel Assembly Handling

Completed fuel assemblies are stored in the assembly storage vault in the MP Area. For shipment offsite, the assemblies are loaded into a MOX fresh fuel shipping cask and conveyed into the Shipping and Receiving Area for loading onto a transport vehicle.

1.1.4.4 Conventional Materials

Other conventional materials and supplies are received at the Receiving Warehouse Building. Packing materials are removed, and the materials, supplies, or equipment are verified and inspected. The materials, supplies, or equipment are sorted and moved to storage in the Secured Warehouse Building, or delivered via onsite vehicles to other areas where needed.

1.1.4.5 Personnel Movement

The Administration Building contains offices for management, administration, production, health and safety, and quality assurance personnel. Personnel enter the PA through the personnel access portal in the Technical Support Building. Workspaces for security and production support personnel are located in this building.

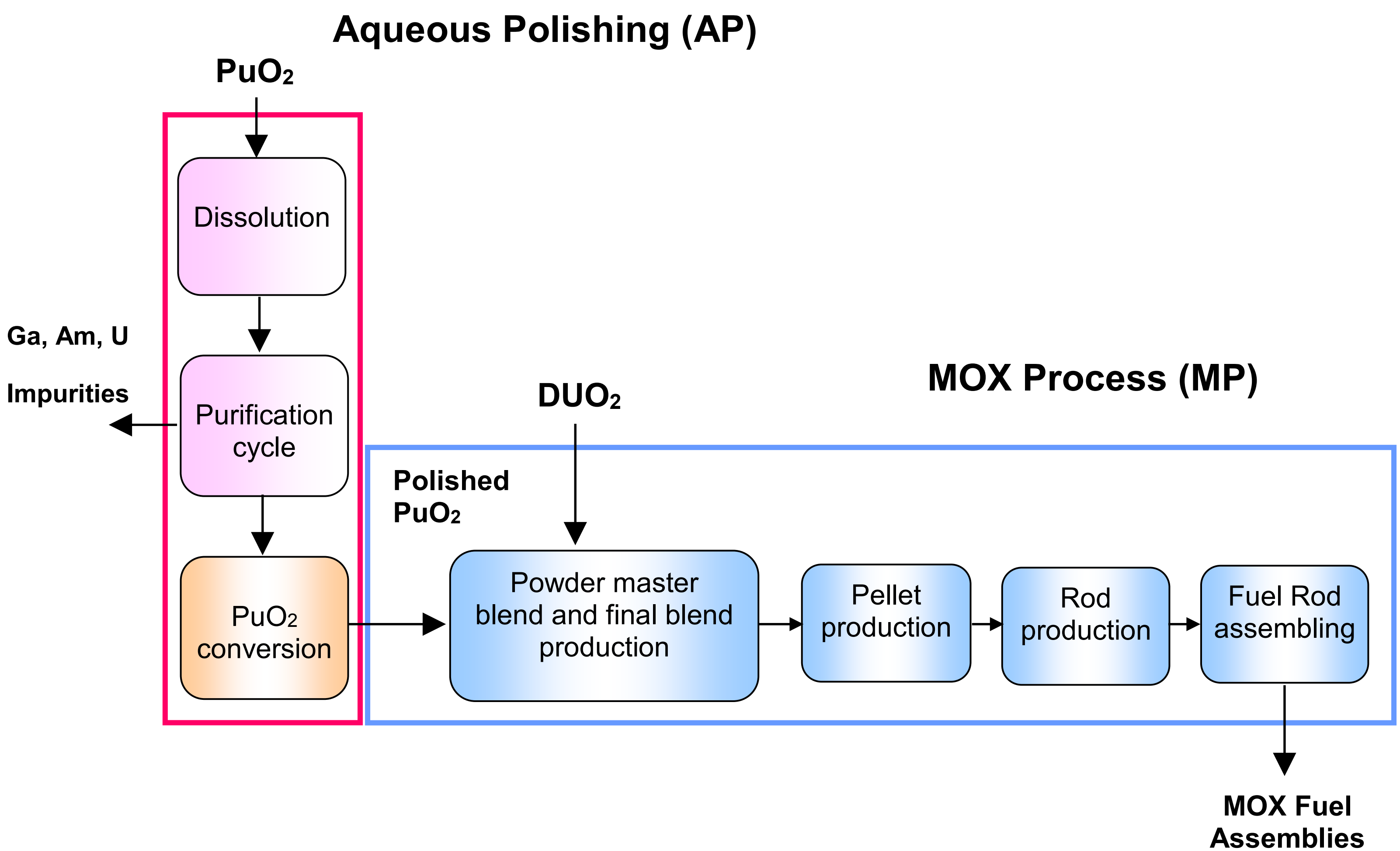
1.1.5 Radioactive Effluents and Waste Disposition

Radioactive effluents and waste disposition are described in Chapter 10.

1.1.6 Process Overview

The MFFF is designed to purify PuO_2 and then blend it with DUO_2 to produce completed MOX fuel assemblies for use in nuclear power reactors. The MFFF has two major process operations: (1) an aqueous polishing process, which serves primarily to remove americium, gallium, and other impurities from the plutonium, and (2) the MOX fuel fabrication process, which processes the oxides into pellets and manufactures the MOX fuel assemblies. These processes are designed and integrated so that waste and discarded powder/pellet material streams are recycled to the extent practical. The major steps in the aqueous polishing and MOX fuel fabrication processes are shown in Figure 1.1.6-1.

Figure 1.1.6-1. AP and MP Process Flow Diagram



1.1.6.1 Aqueous Polishing Process

A Pit Disassembly and Conversion (PDC) – type facility or the Advanced Recovery and Integrated Extraction System (ARIES) facility, or the HB-Line Facility, disassembles plutonium pits from weapons and converts the plutonium to PuO_2 for use as MFFF feedstock. A smaller amount of PuO_2 from other DOE sources is also utilized as MFFF feedstock (alternate feedstock).

The PuO_2 received at the MFFF contains small amounts of impurities that must be removed before the MOX fuel is used in reactors. The aqueous polishing process is used to remove these impurities through a wet extraction process. Impurities in the PDC-type/ARIES feeds are primarily gallium, americium, and highly enriched uranium. Alternate feedstock may contain those and other impurities at higher contaminant levels and may also contain chlorides and other salt contaminants. The aqueous polishing process involves the following three major steps: dissolution, purification, and conversion.

The dissolution step consists of the electrolytic dissolution of PuO_2 powder in nitric acid, and subsequent filtration of the plutonium nitrate solution. Hydrogen peroxide is added to the aqueous nitrate stream to reduce plutonium from the +6 to the +4 valence state so that it can be extracted during the purification step. For PuO_2 containing significant quantities of chlorides, a dechlorination step is utilized prior to dissolution. Chloride ions are electrolytically oxidized and removed from the process stream as chlorine gas. The gas stream is scrubbed of chlorine and then treated in the offgas system.

The purification step includes plutonium extraction with an organic solvent. This step also includes auxiliary processes for recovery of solvent and acid. Plutonium is extracted from the nitrate solution in pulsed columns by contact with a 30% tri-butyl phosphate (TBP) hydrogenated tetrapropylene solution. The plutonium and uranium are extracted into the organic phase and the impurities (americium, gallium, silver, etc.) remain in the aqueous phase as raffinates. The plutonium is then separated from the uranium in the solvent by reducing the plutonium from the +4 to the +3 valence state with the addition of hydroxylamine nitrate and acid stripping, during which the plutonium is removed from the organic stream into the aqueous stream. In the aqueous purified nitrate stream, the plutonium valence is oxidized back to the +4 state by passing nitrous fumes (NO_X) through the plutonium solution in a packed column. Downstream of the plutonium separation process, the solvent solution with the plutonium removed is stripped of uranium with a nitric acid solution. The unloaded solvent solution is sent to the solvent recovery unit, while the uranium stream is sent to the aqueous liquid waste system.

The organic waste streams are collected and sent to the solvent recovery unit where they are scrubbed in a multistage mixer-settler unit to remove the degradation products. The composition of the solvent mixture is adjusted to 30% TBP in the multistage mixer-settler before being recycled to the purification step.

Various aqueous waste streams are collected and sent to the acid recovery unit where the raffinates are concentrated and the nitric acid is recovered in a two-step concentration process that is followed by rectification. The recovered acid is then reused in the process while excess acid and concentrated raffinates are sent to the aqueous waste stream.

The conversion step converts the purified plutonium nitrate stream to PuO_2 powder by the processes of precipitation and calcination. The plutonium nitrate stream is reacted with oxalic acid to form a plutonium oxalate slurry that is collected by a filter and dried in a rotary calciner where the oxalate is converted into oxide at high temperature. The PuO_2 powder is then homogenized, sampled, and stored in cans for use in the fuel fabrication process. The filtered oxalic liquor stream is treated with manganese to facilitate the decomposition of the oxalates, concentrated, and then recycled to the beginning of the extraction cycle to maximize plutonium recovery. Offgas from the rotary calciner is routed through HEPA filters prior to discharge to the atmosphere through the plant vent stack.

1.1.6.2 MOX Fuel Fabrication Process

The MOX fuel fabrication process consists of four major steps: (1) powder master blend and final blend production, (2) pellet production, (3) rod production, and (4) fuel rod assembly.

The first operation is the production of the powder master blend. Polished PuO_2 is mixed with DUO_2 and recycled powder/pellet material to produce an initial mixture that is approximately 20% plutonium. This mixture is subjected to micronization in a ball mill and mixed with additional DUO_2 and recycled material to produce a final blend with the required plutonium content (typically from 2% to 6%). This final blend is further homogenized to meet plutonium distribution requirements. During the final homogenizing steps, a lubricant and poreformer are added to control density.

The final homogenized powder blend is pressed to form "green" pellets, which are then sintered to obtain the required ceramic qualities. The sintering step removes organic products dispersed in the pellets and removes the previously introduced poreformer. The sintered pellets are ground to a specified diameter in centerless grinding machines and sorted. Powder recovered from grinding and discarded pellets are recycled through a ball mill and reused in the powder processing.

Fuel rods are loaded to an adjusted pellet column length, pressurized with helium, welded, and then decontaminated. The decontaminated rods are removed from the gloveboxes and placed on racks for inspection and assembly. Fuel rods are inserted into the fuel assembly skeleton, and the fuel assembly construction is completed. Each fuel assembly is subjected to a final inspection prior to shipment in a DOE fresh fuel shipping cask.

1.2 INSTITUTIONAL INFORMATION

1.2.1 Corporate Identity

MOX Services is the applicant for the license to possess and use by-product material, source material, and special nuclear material (SNM). MOX Services is formed in the State of South Carolina as an LLC owned by CB&I Project Services Group, LLC. (CPSG), and AREVA Nuclear Materials, LLC (AREVA). These two companies are the equity owners of the LLC (CPSG 70% and AREVA 30%). MOX Services was formed to provide MOX fuel fabrication and other services to support the mission of DOE for the disposition of U.S.-owned surplus weapons-usable plutonium. The applicant's mailing address is:

CB&I AREVA MOX Services, LLC
Savannah River Site
P. O. Box 7097
Aiken, SC 29804-7097

The applicant's shipping address is:

CB&I AREVA MOX Services, LLC
Savannah River Site, F-Area, Building 706-1F
Aiken, SC 29808

DOE is the owner of the MFFF, which is located at SRS in Aiken, South Carolina. MOX Services is a South Carolina LLC whose direct owners are all U.S. legal entities. Specifically, CPSG is a proxy entity vested with representing, without interference from its parent company, all of CB&I's interest in MOX Services. This interest represents the CPSG combined 70% ownership and voting power of MOX Services. AREVA, operating under a Special Security Agreement with the National Nuclear Security Administration (NNSA) limiting its parent company involvement, represents AREVA's 30% ownership and voting power. As a result, there is no direct foreign ownership, no foreign control, and no significant foreign interest in MOX Services. Furthermore, in awarding the contract to MOX Services to design, construct, and operate the MFFF, DOE engaged in a foreign ownership, control, or influence (FOCI) review in accordance with DOE Order 470.1, "Safeguards and Security Program." Based upon that review, DOE rendered a favorable FOCI determination on 9 July 1999, based on a Security Control Agreement between CB&I AREVA MOX Services, LLC and DOE, mitigating Foreign Ownership, Control, or Influence.

The MOX Services corporate officers are shown in Table 1.2.1-1.

Table 1.2.1-1. Corporate Officers

Officer	Office	Citizenship
Bobby Wilson	Chairman, Board of Governors	USA
David Del Vecchio	President, and Project Manager	USA
Gilles Rousseau	Executive Vice President, Deputy Project Manager and Chief Operating Officer	FR
Ray Keeler	Vice President, Project Controls	USA
Hank B. Chavous	Vice President, Project Support	USA
Mark Gober	Vice President, Engineering	USA
Rex Norton	Vice President, Contracts and Supply Chain Management	USA
Lauren Wylie	General Counsel and Secretary	USA
Kirk Saunders	Chief Financial Officer	USA

The common address for all the officers listed above is:

CB&I AREVA MOX Service, LLC
Savannah River Site
P. O. Box 7097
Aiken, SC 29804-7097

MOX Services is solely responsible for the design, construction management, and operation of the MFFF. In addition to the CPSG engineering expertise, and AREVA operations expertise, the following companies provide technical support:

- SGN, a wholly owned subsidiary of AREVA NC, for facility and process design experience
- MELOX, a wholly owned subsidiary of AREVA NC, for operations experience
- AREVA, Inc. (formerly Framatome ANP) for operations and engineering experience
- AREVA Federal Services, LLC (wholly owned subsidiary of AREVA)

1.2.2 Type and Period of License and Type, Quantity, and Form of Licensed Material

MOX Services requests a license to receive, acquire, possess, use, store, and transfer by-product material, source material, and SNM. The requested period of the license is 20 years.

Authorization is requested for the types, maximum quantities, and forms of by-product material, source material, and SNM provided in Table 1.2.2-1.

Table 1.2.2-1 . Withheld from Public Disclosure Under 10CFR2.390

1.2.3 Proposed Authorized Uses

Authorized activities at the MFFF include receipt, handling, storage, and shipment of plutonium- and uranium-bearing materials for the following uses:

Aqueous Polishing

- Mechanical powder pretreatment (feed material dependent)
- Dissolution and chloride removal (feed material dependent)
- PuO_2 dissolution by electrolytic dissolution
- Plutonium purification by solvent extraction
- Conversion into PuO_2 by precipitation and calcination.

MOX Processing

- Blending and milling of plutonium, uranium, and mixed oxides
- Pelletizing
- Fuel rod and assembly manufacturing, inspection, and repair/rework
- Laboratory operations
- Discarded powder/pellet material and waste processing.

1.3 SITE DESCRIPTION

This section provides an overall description of the Mixed Oxide (MOX) Fuel Fabrication Facility (MFFF) site and its environment, including regional and local geography, demography, meteorology, hydrology, geology, seismology, and stability of subsurface materials. Significant portions of the information presented in this section were derived from WSRC-TR-2000-00454, *Natural Phenomena Hazards Design Criteria and Other Characterization Information for the Mixed Oxide (MOX) Fuel Fabrication Facility at Savannah River Site (U)* (Westinghouse Savannah River Company [WSRC] 2000b).

1.3.1 Site Geography

1.3.1.1 Site Location

The MFFF site is located adjacent to the separations area (existing F Area) at the SRS in South Carolina. SRS is an approximately circular tract of land occupying 310 mi^2 (803 km^2), or 198,400 acres (80,292 hectare (ha) within Aiken, Barnwell, and Allendale Counties in southwestern South Carolina.

F Area and the MFFF site are located in Aiken County near the center of SRS, east of SRS road C and north of SRS road E. The existing F Area comprises approximately 364 acres (160 ha) of SRS. The nearest SRS boundary to F Area is approximately 5.8 miles (9.3 kilometer [km]) to the west. The center of F Area is approximately 25 miles (40 km) southeast of the city limits of Augusta, Georgia; 100 miles (161 km) from the Atlantic Coast; 6 miles (9.7 km) east of the Georgia border; and about 110 miles (177 km) south-southwest of the North Carolina border. The MFFF site is located adjacent to the north-northwest corner of F Area.

See Table 1.3.1-1 for the location of SRS relative to towns, cities, and other political subdivisions within a 50-mile (80-km) radius. The largest nearby population centers are Aiken, South Carolina and Augusta, Georgia. See Figure 1.3.1-2 for a diagram of the towns near SRS. The only towns within 15 miles (24 km) of the center of SRS are New Ellenton, Jackson, Barnwell, Snelling, and Williston, South Carolina, which Figure 1.3.1-2 shows.

1.3.1.2 Public Roads and Transportation

No highways, railroads, or waterways traverse the MFFF site. The movement of material and personnel to and from the MFFF site takes place via the SRS internal road system. There is no public transportation to SRS.

1.3.1.2.1 Public Roads

Public roads include U.S. Route 278, South Carolina Route (SCR) 125, SCR 64, SCR 19, SCR 78, and SCR 57 via SRS road 1. Of these public roads, only SCR 125 passes through the 5-mile (8-km) radius of F area. These roads are public access corridors and are not routinely controlled. SRS is not open to the general public, but the public can traverse portions of SRS along the established transportation corridors. See Figure 1.3.1-2 for a diagram of the roadways and SRS barricades.

1.3.1.2.2 Railroads

Close to SRS, the Norfolk/Southern Railway owns two tracks that traverse the 5-mile (8-km) area outside the SRS boundary (greater than 10 miles [16.1 km] from the MFFF). One track extends east from Augusta, Georgia to Charleston, South Carolina. The other track extends south from Augusta turning eastward at the Burke County line to a point approximately 3 miles (4.8 km) from SRS and continues south to Savannah, Georgia. A CSX line traverses the site outside (West) of and approximately parallel to SCR 125.

SRS operates and maintains its own railroad system for providing direct rail service to various areas within SRS. The onsite rail system is interfaced with commercial railroads near D Area. Service to D Area is provided by the CSX tracks onto a short section of track owned by the U.S. Department of Energy (DOE). The bulk of rail traffic consists of coal and cask car movements. Other cargo, such as tank cars of bulk chemicals, helium, and various other goods, is moved from the Dunbarton and Ellenton interchanges to areas on SRS.

1.3.1.2.3 Water Transportation

The major river near SRS is the Savannah River, which bounds SRS for 17 miles (27.4 km) on the southwest side of SRS. During the early operation of the Thurmond and Hartwell Lakes (1953 to 1972), there was navigational traffic on the Savannah River from Augusta to Savannah, Georgia. By the late 1970s, waterborne commerce was limited to the transportation of oil to Augusta by the Koch Oil Company until the company discontinued shipping operations in 1979. Since that time, except for limited movements of construction-related items, no commercial shippers have used the river. Maintenance dredging of the river was discontinued in 1979. SRS has no commercial docking facilities, but it has a boat ramp that has accepted large transport barge shipments. Currently, the Savannah River is used primarily for recreation. Recreational uses of the Savannah River include boating, sport fishing, and a limited amount of contact activities such as swimming and water skiing.

1.3.1.2.4 Air Transportation

Bush Field in Augusta, Georgia and the Columbia Metropolitan Airport in Lexington County, South Carolina are the only two airports within 60 miles (96.6 km) of SRS that provide scheduled air passenger services. Bush Field is located approximately 20 miles (32.2 km) from the MFFF site. Columbia is the nearest air traffic hub and is approximately 60 miles (96.6 km) from SRS. Barnwell County Airport, a small general aviation facility, is nearly 16 miles (25.7 km) away from the MFFF and is the closest airport to the SRS boundary. Private aircraft, including corporate jets, use the Barnwell County Airport.

Other small nearby airports include Aiken Municipal Airport (25 miles [40 km] away), Allendale County Airport (27 miles [43 km] away), Bamberg County Airport (30 miles [48 km] away), Burke County Airport in Waynesboro (26 miles [41.8 km] away), and Daniel Field (28 miles [45 km] away) in Augusta. Wackenhut Services Inc. (WSI) operates a heliport at SRS in B Area about 3 miles (4.8 km) from the MFFF. WSI operates two lightweight multipurpose helicopters providing support to the security services at SRS. The U.S. Forest Service (USFS) conducts regular helicopter operations across SRS for purposes of wildfire detection/response, prescribed fire operations, and wildlife/forest health surveillance. USFS operations originate from the heliport adjacent to the USFS facility in G Area. In addition, South Carolina Electric and Gas (SCE&G) conducts limited helicopter operations across SRS for purposes of right-of-way inspection and clearance. Operations originate offsite with site access accomplished via electrical line pathways only.

1.3.1.3 Nearby Bodies of Water

Nearby bodies of water within 50 miles (80 km) of SRS are Thurmond Lake (formerly called Clarks Hill Reservoir) and the Savannah River. Thurmond Lake, operated by the U.S. Army Corps of Engineers, is the largest nearby public recreational area. This lake is an impoundment of the Savannah River about 40 miles (64 km) northwest of the center of SRS.

The principal surface-water body associated with SRS is the Savannah River, which flows along the site's southwest border for 17 miles (27.4 km). Six principal tributaries to the Savannah River are located on SRS: Upper Three Runs, Beaver Dam Creek, Fourmile Branch, Pen Branch, Steel Creek, and Lower Three Runs. F Area is drained by several tributaries of Upper Three Runs and by Fourmile Branch. The elevation of the MFFF site is 272 feet (82.9 meters [m]) above msl. See Figure 1.3.1-3 for a diagram of the topography of F Area and the MFFF site.

1.3.1.4 Other Significant Geographic Features

Two distinct physiographic subregions are represented at SRS. They are the Pleistocene Coastal Terraces, which are below 270 feet (82.3 m) in elevation, and the Aiken Plateau, which is above 270 feet (82.3 m) in elevation.

Table 1.3.1-1 Cities and Towns within 50 Miles of SRS

Population Center	County	State	Distance (Miles)	Sector	Population*
Augusta	Richmond	GA	25.0	WNW	43,459
Aiken	Aiken	SC	19.5	NNW	24,929
North Augusta	Aiken/Edgefield	SC	23.4	NW	17,618
Orangeburg	Orangeburg	SC	47.5	ENE	13,762
Evans	Columbia	GA	33.0	NW	13,713
Belvedere	Aiken	SC	25.9	NW	6,133
Red Bank	Lexington	SC	50.3	NE	5,950
Waynesboro	Burke	GA	25.8	WSW	6,712
Barnwell	Barnwell	SC	16.4	ESE	5,600
Clearwater	Aiken	SC	19.3	NE	4,731
Allendale	Allendale	SC	27.3	SE	4,316
Batesburg	Lexington/Saluda	SC	43.3	N	4,380
Bamberg	Bamberg	SC	35.2	E	3,596
Millen	Jenkins	GA	31.6	SW	3,977
Denmark	Bamburg	SC	28.9	E	3,640
Grovetown	Columbia	GA	34.2	WNW	4,427
Williston	Barnwell	SC	15.0	ENE	3,445
Hampton	Hampton	SC	41.3	SE	3,146
Sylvania	Screven	GA	37.0	S	3,109
Saluda	Saluda	SC	49.7	N	2,957
Gloverville	Aiken	SC	24.5	NW	2,753
Blackville	Barnwell	SC	22.2	ENE	2,640
Johnston	Edgefield	SC	38.9	NNW	2,670
New Ellenton	Aiken	SC	9.4	NNW	2,494
Edgefield	Edgefield	SC	38.8	NNW	2,644
Hephzibah	Richmond	GA	26.6	W	2,925
Louisville	Jefferson	GA	48.6	WSW	2,542
Wrens	Jefferson	GA	43.8	W	2,577
South Congaree	Lexington	SC	49.3	NE	2,736
Estill	Hampton	SC	43.6	SSE	2,513
Fairfax	Allendale	SC	32.8	SE	2,397
Harlem	Columbia	GA	40.0	WNW	2,592

Table 1.3.1-1 Cities and Towns within 50 Miles of SRS (continued)

Population Center	County	State	Distance (Miles)	Sector	Population*
Leesville	Lexington	SC	44.8	N	2,235
Varnville	Hampton	SC	44.8	SE	2,140
Pineridge	Lexington	SC	49.5	NE	1,927
Jackson	Aiken	SC	9.4	WNW	1,876
McCormick	McCormick	SC	48.8	NW	1,701
Sardis	Burke	GA	22.7	SSW	1,217
Branchville	Orangeburg	SC	47.7	E	1,243
Gaston	Lexington	SC	48.4	NE	1,140
Ridge Spring	Saluda	SC	38.8	N	992
North	Orangeburg	SC	38.8	NE	827
Wagener	Aiken	SC	30.0	NNE	1,236
Midville	Burke	GA	47.2	SW	642
Brunson	Hampton	SC	36.4	SE	619
Dearing	McDuffie	GA	44.1	WNW	650
Swansea	Lexington	SC	44.5	NE	572
Springfield	Orangeburg	SC	25.8	NE	546
Burnettown	Aiken	SC	25.0	NNW	521
Salley	Aiken	SC	27.5	NE	515
Ehrhardt	Bamberg	SC	38.8	ESE	577
Neeses	Orangeburg	SC	34.5	ENE	474
Hilltonia	Screven	GA	27.7	S	414
Norway	Orangeburg	SC	31.7	ENE	411
Olar	Bamberg	SC	31.5	E	352
Hilda	Barnwell	SC	23.0	E	253
Pelion	Lexington	SC	40.3	NE	349
Stapleton	Jefferson	GA	48.3	W	330
Gilbert	Lexington	SC	46.4	NNE	356
Rowesville	Orangeburg	SC	47.2	E	350
Trenton	Edgefield	SC	33.6	NNW	315
Furman	Hampton	SC	49.5	SSE	267
Summit	Lexington	SC	45.9	NNE	273

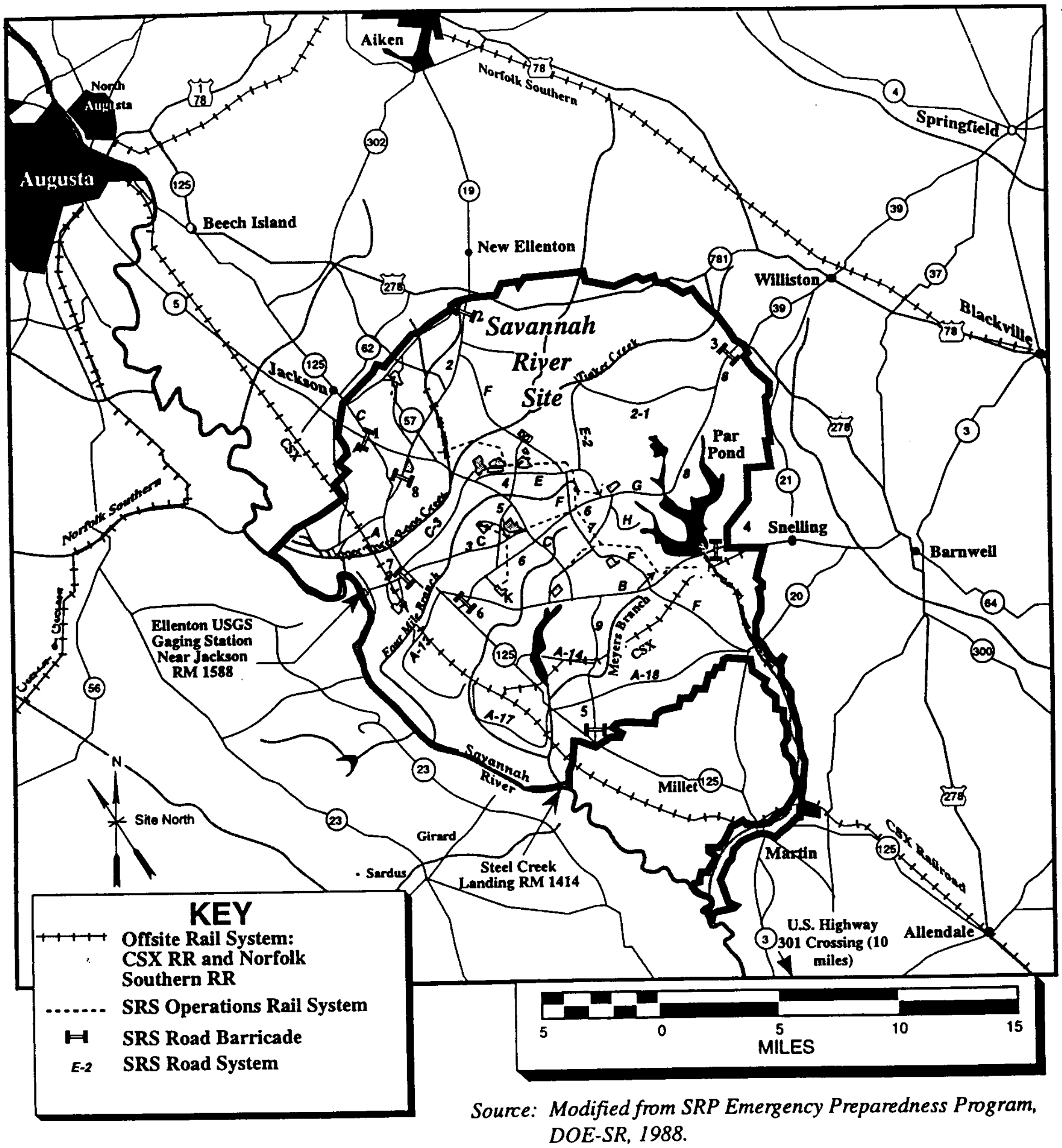
Table 1.3.1-1 Cities and Towns within 50 Miles of SRS (continued)

Population Center	County	State	Distance (Miles)	Sector	Population*
Perry	Aiken	SC	30.3	NE	230
Elko	Barnwell	SC	16.4	ENE	207
Sycamore	Allendale	SC	32.3	SE	203
Woodford	Orangeburg	SC	40.6	NE	215
Rocky Ford	Screven	GA	43.9	SSW	223
Girard	Burke	GA	17.5	SSW	222
Parksville	McCormick	SC	48.1	NE	199
Williams	Colleton	SC	49.5	ESE	175
Scotia	Hampton	SC	48.0	SSE	189
Livingston	Orangeburg	SC	47.7	ENE	178
Lodge	Colleton	SC	42.7	ESE	198
Smoaks	Colleton	SC	50.0	ESE	147
Cordova	Orangeburg	SC	43.1	ENE	139
Ward	Saluda	SC	25.6	N	141
Snelling	Barnwell	SC	11.3	ESE	133
Cope	Orangeburg	SC	37.3	E	130
Windsor	Aiken	SC	15.3	NNE	130
Luray	Hampton	SC	40.3	SE	71
Plum Branch	McCormick	SC	50.0	NW	104
Govan	Bamberg	SC	27.3	E	80
Ulmer	Allendale	SC	35.5	SE	67

* as of July 1, 1994; Data from WSRC 2000b.

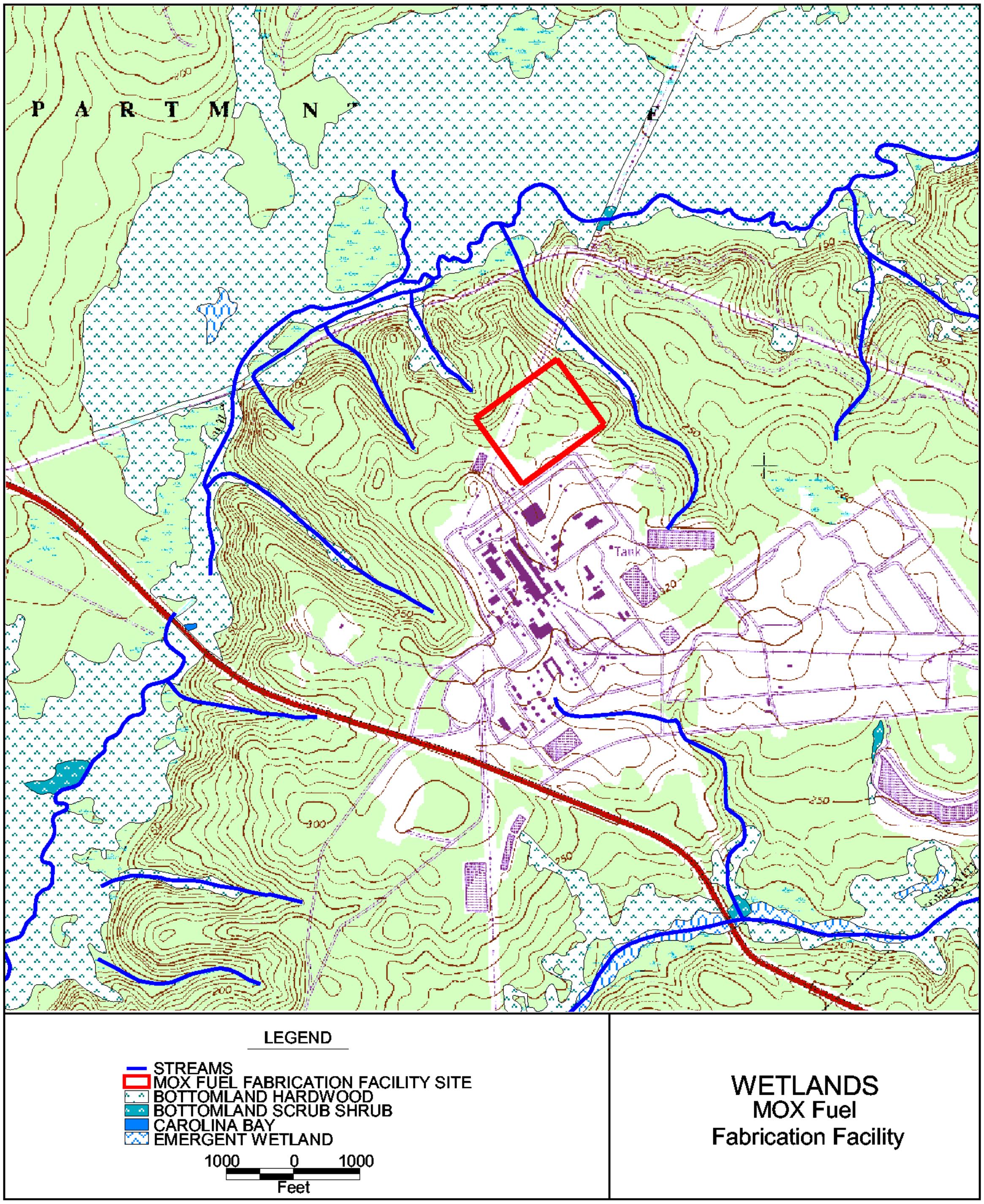
Figure 1.3.1-1. Figure Deleted

Figure 1.3.1-2. Towns and Roads Near SRS



Data from WSRC 2000b.

Figure 1.3.1-3. Topography in the Vicinity of the MFFF Site



Data from WSRC 2000b.

1.3.2 Demographics and Land Use

1.3.2.1 Population Information

1.3.2.1.1 Permanent Population and Distribution

Approximately 621,527 people resided within a 50-mile radius of the MOX MFFF site in 1990. That population is projected to grow to approximately 1,042,483 by the year 2030. See Table 1.3.2-1 through Table 1.3.2-5 for the population distributions for 1990, 2000, 2010, 2020, and 2030, respectively. The 1990 numbers are based on 1990 U.S. Census counts, while years 2000 through 2030 are projections compiled for the SRS Generic Safety Analysis Report (GSAR) and are based on growth projections provided by the University of Georgia. The analysis included spatial distribution of the population based on a circular grid comprised of 22½º azimuth sectors centered on the 16 cardinal compass point directions and six radial distances of 0 to 5, 5 to 10, 10 to 20, 20 to 30, 30 to 40, and 40 to 50 miles. Since the land within a 5-mile radius of the MFFF site is within SRS and contains no residential population, the usual 1 mile increment analysis for the area within 5 miles of the site is not shown.

The area within the 50-mile radius of the MFFF site includes all, or portions of, two major metropolitan areas where large concentrations of people may be found. See Figure 1.3.2-1 for a map of the 50-mile radius from the MFFF. The largest population centers are Aiken, South Carolina and Augusta, Georgia. The only towns within 15 miles of the center of F Area are New Ellenton, Jackson, Barnwell, Snelling, and Williston, South Carolina. The Augusta-Aiken Metropolitan Statistical Area (MSA), which includes Columbia, Richmond, and McDuffie Counties in Georgia and Edgefield and Aiken Counties in South Carolina, is anchored by the city of Augusta, which is over 20 miles west-northwest of the MFFF site. The Augusta-Aiken MSA contained 415,220 people in 1990 and an estimated 458,271 people in 1998, primarily in the cities of Augusta, Aiken, and North Augusta. The closest boundary of the Columbia City MSA, which includes Lexington and Richland Counties (South Carolina), is located over 30 miles northeast of the MFFF site. Columbia City, the core of this MSA, is located outside of the 50-mile radius. The Columbia City MSA contained 453,932 people in 1990 and an estimated 512,316 people in 1998. Greater than 50% of the population in the Columbia City MSA lives over 50 miles from the MFFF site.

The local area within a 10-mile radius around the MFFF site is comprised of portions of three counties: Aiken and Barnwell Counties in South Carolina and Burke County in Georgia. The MFFF is located on SRS in Aiken County. Only SRS facilities (no residential population) are located within 5 miles of the MFFF site.

The area between 5 and 10 miles from the MFFF site contained about 6,528 people in 1990. That population is projected to grow to a total of approximately 10,876 by the year 2030. A majority of this local population resides to the north and northwest of the site in the towns of New Ellenton and Jackson, which contained estimated populations of 7,197 and 2,843 people in 1998, respectively. See Table 1.3.2-1 through Table 1.3.2-5 for the existing (year 1990) and projected (years 2000, 2010, 2020, and 2030) populations between 5 and 10 miles of the MFFF site are included.

See Table 1.3.2-6 for the racial and ethnic mix of the local area population. Racially, the population is predominantly white, with 34% black and less than 2% Asian, Pacific Islander, American Indian, Eskimo, or Hispanic. Of the combined population of counties that are partially or entirely within the 50-mile radius of the MFFF site, about 48% is male and 52% is female.

See Table 1.3.2-7 for the economic and unemployment data for the counties within 50 miles of the MFFF. Over 20% of the population of a majority (i.e., 14 out of 21) of the counties in the 50-mile radius had income levels below the federal poverty threshold; only Aiken and Lexington Counties in South Carolina and Columbia and Glascock Counties in Georgia had lower percentages of population below the poverty threshold than their respective state averages. Only Aiken and Lexington Counties exceeded state averages for per-capita income in 1994.

Within the three Counties that make up the local 10-mile area, Burke County, Georgia contains the least affluent population, with a 1990 per-capita income of $11,172 and about 30.3% of its population living below the poverty level in 1989. In the same years, the per-capita income for the state of Georgia was $17,123 with approximately 14.7% of its population living below the poverty level. Within South Carolina, Aiken County had per-capita income and poverty levels superior to the state average, but Barnwell County was considerably below in income (i.e., about 20% below the state average) and contained a higher percentage of individuals below the poverty level. Income levels have grown slightly since 1989. However, the percentage of the population with incomes below the poverty level in each of the three local counties has remained consistent. See Table 1.3.2-8 for the income and poverty data for the three-county local area. Unemployment in the local area ranged from a high of 16% in Burke County to a low of 7% in Aiken County in 1996.

See Table 1.3.1-1 for the population and geographic locations of cities and towns within the 50-mile radius of SRS.

1.3.2.1.2 Transient Population Variations

A 5-mile radius for the MFFF site is considered when discussing the transient population variations for ISA purposes. The transient population components investigated are industrial, school, recreational, health care, and casual. The 5-mile radius for the MFFF site falls entirely within the SRS boundary. See Figure 1.3.2-2 for a diagram of the 5-mile radius from the MFFF. There are no facilities or populations within 5 miles of the MFFF site that are not part of the SRS complex. Therefore, the transient population consists only of employees, badged visitors, vendors making deliveries at SRS site locations within the area, and persons traveling on public highways on the SRS site.

There are no military reservations or correctional institutions located within the 5-mile radius of the MFFF site boundary.

1.3.2.1.3 Industrial Population

The industrial population within a 5-mile radius of F Area consists entirely of SRS employees at A/M, B, C, N, E, F, H, K, S, and Z Areas.

In 2002, SRS employed approximately 13,590 persons, including 12,051 employed by Westinghouse Savannah River Company, LLC (WSRC) (Management and Operations [M&O] Contractor); 823 employed by Wackenhut Services Inc. (WSI); 459 employees under U.S. Department of Energy Savannah River Operations Office (DOE-SR); and 257 other SRS contract employees. Approximately 90% of that workforce resides within five counties: Aiken, Barnwell, and Edgefield Counties in South Carolina and Columbia and Richmond Counties in Georgia. See Table 1.3.2-9 for the approximate number of SRS employees by county of residence.

1.3.2.2 Population Centers

The MFFF site within SRS is extremely rural, is entirely within the boundaries of the SRS property, and contains no communities, neighborhoods, or other areas that may be impacted by MFFF operations. The nearest population is located more than 5 miles from the MFFF site.

A majority of the population within a 10-mile radius of the MFFF site resides within Aiken County. See Section 1.3.2.1 for additional population information.

1.3.2.3 Public Facilities

1.3.2.3.1 School Population

A minimal number of facilities, mostly schools, containing transient populations are located within a 10-mile radius surrounding the MFFF site. Five public schools are located within the area to the northwest and west, with the closest being over 6 miles away from the MFFF site. See Table 1.3.2-10 for a list of the local public schools within the 10-mile radius of the MFFF site and recent enrollments (1998 to 1999). The schools operate from late August through late May. There are no private schools or colleges located in the 10-mile radius of the MFFF site. The students in these schools are assumed to be part of the resident population within the 50-mile radius of the MFFF site.

1.3.2.3.2 Health Care Populations

The U.S. Census Bureau estimated that 1,765 people resided in group quarters in Aiken County, 297 in Barnwell County, and 216 in Burke County in 1997. The only residential institutions classified as group quarters within 10 miles of the site are three residential care facilities located in New Ellenton: the New Ellenton Nursing Center (26 beds), Coleman's Residential Care (10 beds), and Parker's Residential Care Home (nine beds). The closest of these three facilities, Parker's Residential Care Home, is outside of the 6-mile radius northwest of the MFFF site. There are no hospitals located within a 6-mile radius of the MFFF site.

1.3.2.3.3 Recreational Population

The primary recreational activity within a 5-mile radius of the MFFF site is controlled sport hunting. Hunts at SRS, supervised by DOE, are conducted annually with the benefit of controlling deer and feral hog populations.

Hunting also takes place at Crackerneck, an area of 4,780 acres west of SRS in Aiken County. The South Carolina Department of Natural Resources manages hunts at Crackerneck for deer, hogs, small game, and waterfowl, although permits are issued by DOE. Another sporting area within 5 miles of SRS is a private commercial area of 4,000 acres about 15 miles east of Waynesboro, Georgia. Hunting and/or fishing, as well as available lodging, are available to the public year-round for a fee. No records of usage are available.

Additional recreational usage near the vicinity is available at three state parks located outside of the 5-mile radius of SRS production areas but within the 12-mile radius of the SRS site boundary. These areas include Redcliffe State Park, a historic site located off SCR 278 at Beech Island; Aiken State Park located off U.S. Route 78, 16 miles east of Aiken; and Barnwell State Park located off SCR 3 near Blackville. During fiscal year 1994/1995, total park usage was approximately 116,000 visitor-days. The parks are available to the public year-round.

Other recreational activities within the 5-mile radius of SRS production areas include fishing and boating. Numerous boat landings are located on the west bank of the Savannah River, which borders the southwestern portion of the site. In addition, a 95 acre man-made lake, Lake Edgar Brown, is located within the city limits of Barnwell. No records of usage at these areas are available.

1.3.2.4 Industrial Areas

1.3.2.4.1 Savannah River Site Operations Activities

SRS consists of six major operating areas: reactor areas (C, K, L, P, and R Areas); separations areas (F and H Areas); waste management areas (E, S, and Z Areas); heavy water reprocessing area (D Area); reactor materials area (M Area); and administration area (A Area).

1.3.2.4.1.1 Reactor Areas (C, K, L, P, and R Areas)

The five nuclear production reactor facilities (C, K, L, P, and R reactors) occupy 934 acres of SRS. The five reactors are located within a 10-mile radius of the MFFF site and have been placed in cold shutdown with no plans for restart. See Figure 1.3.2-2 for the approximate locations of the reactor areas. Although the reactor areas are being used for moderator, plutonium, and fuel storage, no effort is being expended to maintain the production capability of these reactors.

1.3.2.4.1.2 F Area

F Area is located in Aiken County, South Carolina near the center of SRS, east of SRS road C and north of SRS road E. See Figure 1.3.2-2 for the location of F Area. The existing F Area occupies 364 acres. The nearest SRS site boundary to F Area is less than 6 miles to the west.

The main processing facility in F Area is F Canyon, which is composed of two chemical separation plants and associated waste storage facilities. In the past, F Canyon was used to chemically separate uranium, plutonium, and fission products from irradiated fuel and target assemblies. The separated uranium and plutonium were transferred to other DOE facilities for further processing and final use. F Canyon has been deactivated with all radioactive materials removed. There are no current plans to restart F Canyon. The waste from F Canyon was transferred to high-level waste (HLW) tanks in the area for storage. The F-Area Tank Farm stores aqueous radioactive HLW and evaporated saltcake in underground storage tanks.

FB line previously converted plutonium solution produced in F Canyon to ^{239}Pu metal to support defense programs. FB line has been deactivated with all radioactive materials removed. There are no current plans to restart FB line.

Analytical laboratories in F Area (buildings 772-F, 772-1F, and 772-4F) principally support reprocessing and waste activities for the F and H Areas.

1.3.2.4.1.3 H Area

H Area is located 2 miles east of F Area in Aiken and Barnwell Counties in South Carolina near the center of SRS. See Figure 1.3.2-2 for the location of H Area. The nearest SRS site boundary to H Area is approximately 7.2 miles to the west.

H Area covers approximately 395 acres. The H-Canyon facility in H Area is used to convert highly enriched weapons-grade uranium to a low enriched form not usable for weapons production and to stabilize ^{242}Pu solutions. In the past, H Canyon, which is a large, shielded chemical separations plant, processed irradiated fuel and target assemblies by utilizing solvent extraction and ion exchange to separate uranium, plutonium, and fission products from waste. The separated uranium and plutonium were transferred to other H Area facilities for processing into a solid form. The waste was transferred to HLW tanks in the area for storage, and some of the nuclear materials were shipped to other DOE sites for final use.

HB line was constructed to support the production of ^{238}Pu. HB line is also used to stabilize ^{242}Pu solutions.

The tritium facilities in H Area consist of four main process buildings, designed for and operated to process tritium. In July 2000, work commenced on the Replacement Tritium Facility (RTF), which will extract tritium from irradiated fuel rods from the Tennessee Valley Authority Sequoyah and Watts Bar Nuclear Plants. The main mission of the tritium facilities is to purify and maintain existing inventories of tritium for defense purposes.

The Receiving Basin for Offsite Fuels is also located in H Area. Offsite fuels that will be processed in H Canyon were stored and packaged at the RBOF. Radioactive waste generated by the RBOF is stored in the HLW tanks in H Area. RBOF has been deactivated with all radioactive materials removed. There are no current plans to restart RBOF.

The Effluent Treatment Facility (ETF) is located on the south side of H Area. The ETF treats low-level radioactive wastewater. The ETF removes radioactive and non-radioactive contaminants, except tritium, from process effluents and allows the water to discharge to Upper Three Runs.

The H Area Tank Farm consists of 29 large (up to 1.3 million gallon capacity) underground storage tanks that store aqueous radioactive HLW and evaporated saltcake. Seven of these tanks are now dedicated as salt processing tanks.

The Consolidated Incineration Facility (CIF) is located on the east side of H Area. The CIF incinerates SRS hazardous, mixed, and low-level radioactive wastes. The CIF is not currently operated.

1.3.2.4.1.4 E Area

The E-Area Solid Waste Management Facility (SWMF) is located in Aiken County, South Carolina, near the approximate center of SRS between H and F Areas. See Figure 1.3.2-2 for the location of E Area. The SWMF occupies 195 acres. The nearest SRS site boundary to E Area is approximately 6.5 miles to the west.

The SWMF is used for disposal and/or storage of radioactive, hazardous, and mixed solid wastes generated at SRS, as well as occasional special shipments from offsite. It also provides interim storage for transuranic waste. Other facilities receive hazardous, low-level, and mixed wastes for incineration and non-radioactive and hazardous wastes for storage.

1.3.2.4.1.5 S Area

S Area is located in Aiken County, South Carolina north of H Area. See Figure 1.3.2-2 for the location of S Area. The nearest SRS site boundary to S Area is approximately 6.8 miles to the north.

S Area is the site of the Defense Waste Processing Facility (DWPF) Vitrification Plant. The DWPF immobilizes radioactive HLW sludge and precipitate by “vitrifying” it into a solid glass waste form.

1.3.2.4.1.6 Z Area

Facilities in Z Area are located about 2.5 miles from F Area in Aiken County, South Carolina near the center of SRS. See Figure 1.3.2-2 for the location of Z Area. These facilities are used to process and dispose of decontaminated salt solution supernatants from waste tanks. The nearest SRS site boundary to Z Area is approximately 6.2 miles to the north.

Z Area, which contains the DWPF Saltstone Facility, is located north of the intersection of SRS road F and SRS road 4. The Saltstone Facility treats and disposes of the filtrate created by the salt removal process by stabilizing it in a solid, cement-based waste form.

1.3.2.4.1.7 D Area

The 400-D Area occupies 445 acres at SRS. See Figure 1.3.2-2 for the location of D Area. A coal-fired power plant is located in D Area. This facility is the site's largest coal-fired powerhouse; it can produce approximately 70 MW of electricity and 420,000 pound (lb)/hour (hr) of process steam.

1.3.2.4.1.8 M Area

The 300-M Area occupies approximately 114 acres. See Figure 1.3.2-2 for the location of M Area. M Area previously provided support to the reactor facilities, heavy water facilities, and fuel fabrication facilities. The operations of these facilities have been discontinued and the facilities have been dismantled.

1.3.2.4.1.9 A Area

General site administrative functions are centered in A Area, which occupies 348 acres (141 ha). The main DOE and WSRC headquarters are in A Area. Other organizations in A Area provide scientific and logistical support for SRS operations. The Savannah River Technology Center (SRTC) supports the missions of SRS through applied research and development.

1.3.2.4.2 Other Nonproduction SRS Facilities

Activities conducted within SRS that are not related to production are performed by the following organizations: General Services Administration, WSI, Savannah River Forest Station (SRFS), Savannah River Ecology Laboratory (SREL), University of South Carolina Institute of Archaeology and Anthropology, Soil Conservation Service (SCS), and U.S. Department of Agriculture (USDA).

1.3.2.4.3 Other Industrial Populations (Non-SRS)

This section identifies nuclear and industrial facilities within a 50-mile radius of the SRS center.

1.3.2.4.3.1 Chem Nuclear Systems, Incorporated

Chem Nuclear Systems, Incorporated (CNSI), located in Barnwell County, South Carolina near the eastern SRS boundary, is a commercial facility for the disposal of low-level wastes and hazardous chemicals. The CNSI facility includes a burial site, transportation unit, maintenance unit, and facilities for waste solidification and decontamination.

1.3.2.4.3.2 Transnuclear, Incorporated

Transnuclear, Incorporated, located in Aiken County, South Carolina transports high- and low-level radioactive wastes and maintains temporary onsite storage of materials to be transported. The materials are transported from various industrial and military facilities nationwide; U.S. Department of Defense (DOD) waste is sent to SRS, and low-level waste is sent to CNSI. No commercial wastes are sent to SRS. The company also manufactures transport casks and provides cask decontamination services.

1.3.2.4.3.3 Carolina Metals, Incorporated

Carolina Metals, Incorporated, located in Barnwell County, South Carolina processes depleted uranium hexafluoride into uranium metal for DOD and commercial uses.

1.3.2.4.3.4 Vogtle Electric Generating Plant

The Vogtle Electric Generating Plant (VEGP) is a two-unit commercial nuclear power plant operated by Georgia Power. VEGP is located across the Savannah River from SRS in Burke County, Georgia, about 4.5 miles south-southeast of D Area. Unit 1 was licensed for full-power operation in May 1987. Unit 2 began operation in May 1989. An emergency plan and a communications protocol are in place between VEGP and SRS. Details of protective actions, with regard to an accident at VEGP, are contained in the SRS Emergency Plan.

1.3.2.4.3.5 Urquhart Station

Urquhart Station is a three-unit, 250 MW, coal and natural gas-fired steam electric plant in Beech Island, South Carolina. It is owned by SCE&G and is located on the Savannah River about 20 river miles north of SRS.

1.3.2.4.3.6 Military Facilities

Fort Gordon is the nearest military facility, located approximately 9 miles southwest of Augusta, Georgia and more than 20 miles from SRS. Approximately 50,000 individuals are involved in activities at Fort Gordon. North Air Base, located approximately 39 miles northeast of SRS, is closed and no military personnel are stationed there.

1.3.2.5 Land Use

The total area investigated within the SRS boundary area is approximately 800 mi^2. Of these 800 mi^2, 310 mi^2 are used for industrial purposes associated with the operation of SRS and for commercial and noncommercial timber management. Land use at SRS can be classified into three major categories: forest/undeveloped, water/wetlands, and developed facilities.

Approximately 226 mi^2 of SRS (i.e., 73% of the area) is undeveloped. Wetlands, streams, and lakes account for 70 mi^2 (22%) of the site, while developed facilities including production and support areas, roads, and utility corridors make up approximately 15 mi^2 (5%) of SRS. DOE manages the land that forms a buffer zone around the production facilities.

Land within F Area and the MFFF site is completely within SRS and is used for industrial purposes associated with SRS. See Table 1.3.2-11 for a listing of the land use at SRS.

Forested areas are managed by the SRFS, an administrative unit of the U.S. Forest Service (USFS). Through an interagency agreement between DOE and the USDA, the USFS maintains the SRFS to provide timber management, research support, soil and water protection, wildlife management, secondary road management, and fire management. The land in the affected area is primarily used for timbering. Small tracts of land are clear-cut on a rotating basis.

1.3.2.6 Water Use

1.3.2.6.1 General Uses of the Savannah River

The Savannah River forms the boundary between Georgia and South Carolina. Downstream from Augusta, Georgia, the Savannah River has been classified as Class B waters suitable for domestic supply after treatment, propagation of fish, and industrial and agricultural uses. The river supplies water for Augusta, Georgia; North Augusta, South Carolina; and Beaufort and Jasper Counties, South Carolina; and supplements the water supply of Savannah, Georgia. It also receives domestic and industrial wastes from Augusta, Georgia; North Augusta, South Carolina; and Horse Creek Valley (Aiken County, South Carolina).

At SRS, the coal-fired power plants are cooled with water pumped from the river. Effluents and wastewater from SRS are discharged into the Savannah River tributaries that flow across SRS.

Recreational uses of the Savannah River include boating and sport fishing, and a limited amount of contact activities such as swimming and water skiing.

1.3.2.6.2 Navigation

During the early operation of the Thurmond and Hartwell Lakes (1953 to 1972), there was navigational traffic on the river from Augusta to Savannah. By the late 1970s, waterborne commerce was limited to the transportation of oil to Augusta. In 1979, this shipping was discontinued. Since that time, except for limited movements of construction-related items, no commercial shippers have used the river. Maintenance dredging of the river was discontinued in 1979.

1.3.2.6.3 Fisheries

Three types of fisheries are found along the Savannah River. Freshwater trout are in the cold waters flowing from the mountains of North Carolina, South Carolina, and Georgia. Other freshwater fish species are found in the warmer waters in the Piedmont and Coastal Plain; saltwater species are found downstream in the brackish waters near the mouth and in the estuary.

Warm-water fishing constitutes most of the sport fishing in the Savannah River.

1.3.2.6.4 Recreation

Over 95% of South Carolina's impounded waters are contained in the large reservoirs and have multipurpose recreational uses such as swimming, water skiing, boating, and fishing. Par Pond and L Lake, both previously used for reactor cooling water, are completely within the boundary of SRS and are not accessible to the public. Thurmond Lake (Clarks Hill Reservoir), Hartwell Reservoir, and Russell Dam are located northwest of Augusta approximately 65 to 133 river miles from the center of the site. They are used for hydroelectric power generation, flood control, and water supply, as well as for recreation.

1.3.2.6.5 Agricultural Water Use

Water for agricultural use in Aiken, Barnwell, and Allendale Counties is obtained primarily from lakes and ponds. No uses of the Savannah River for crop irrigation were identified in Richmond and Burke Counties, Georgia or for Aiken, Barnwell, and Allendale Counties, South Carolina.

1.3.2.6.6 Municipal Use of Local Surface Water

The Savannah River and its reservoirs are the sources of water for 64 domestic and industrial users. Total withdrawals amount to approximately 1 billion gallons per day. The largest water users are SRS and VEGP. At the lower end of the river, freshwater intakes and canals are maintained by the Beaufort-Jasper Water Supply Authority, the City of Savannah Municipal and Industrial Plant, and the Savannah National Wildlife Refuge.

The larger communities in Aiken, Richmond, and Burke Counties use surface water supplies as well as groundwater. None of these surface water supplies are impacted by liquid discharges from operations at SRS. These intakes are either on the Savannah River upstream from SRS or on tributaries of the Savannah River that do not cross or drain at SRS.

1.3.2.6.7 Municipal and Industrial Use of Savannah River Water Downstream from Savannah River Site

Two water treatment plants about 100 miles downriver from SRS supply Savannah River water to customers in Beaufort and Jasper Counties, South Carolina and Chatham County, Georgia.

The City of Savannah Industrial and Domestic Water Supply (Chatham County, Georgia) is the largest of the two water treatment plants.

The Beaufort-Jasper Water/Sewer Authority near Hardeeville, South Carolina has been in operation since 1965. It serves a consumer population of about 50,000 people who live in Beaufort and Jasper Counties. The plant is located about 18 miles from the Savannah River. A canal transports water from the river to the plant. The plant processes an average of 6 million gallons per day (mgd), varying from about 5 mgd in the winter to 10 to 12 mgd in the summer. Increased use in the summer is associated with watering lawns, filling swimming pools, and uses in the home. The City of Savannah Industrial and Domestic Water Supply at Port Wentworth has been treating water during the entire period of operation of SRS. Treated water from this plant is used primarily for industrial and manufacturing purposes in an industrial complex near Savannah, Georgia. The complex serves a non-community/non-transient population of 6,000 people, primarily adults working in industrial facilities; it also serves as a backup for the City of Savannah's domestic groundwater system. The plant processes about 40 to 50 mgd. Usage of this water for the City of Savannah does not show a strong summer demand, since the water is primarily used for industrial purposes.

1.3.2.6.8 Groundwater Use

The coastal plain sediments that underlie SRS are an important hydrologic resource, since the formations are sources for drinking water, industrial processes, cooling water, and water used for agricultural purposes. Fifty-six municipalities and industries identified near the site use this groundwater. Total pumpage by these users in 1985 was approximately 35 mgd. In addition, several small communities, mobile home parks, schools, and small commercial interests draw from this groundwater resource.

Table 1.3.2-1. Population Distribution from the MFFF Site – 1990

Direction	5 to 10 mi	10 to 20 mi	20 to 30 mi	30 to 40 mi	40 to 50 mi	Total
N	2,072	21,439	9,195	6,687	10,462	49,855
NNE	235	1,782	2,081	4,100	17,085	25,283
NE	8	1,545	2,730	5,240	11,442	20,965
ENE	0	3,277	4,657	5,189	31,845	44,968
E	1	4,773	5,086	10,908	5,512	26,280
ESE	8	2,166	2,577	2,839	2,891	10,481
SE	0	563	4,543	6,387	10,432	21,925
SSE	0	364	683	1,046	2,507	4,600
S	0	545	1,596	6,730	3,560	12,431
SSW	99	780	2,186	4,805	2,591	10,461
SW	110	1,171	4,578	2,093	2,711	10,663
WSW	101	1,523	4,472	2,586	6,149	14,831
W	241	6,031	10,519	8,946	6,959	32,696
WNW	1,380	5,066	129,791	32,475	14,790	183,502
NW	1,102	15,212	81,259	9,385	3,296	110,254
NNW	1,171	19,728	11,205	6,884	3,344	42,332
Total	6,528	85,965	277,158	116,300	135,576	621,527

Data from WSRC 2000b.

Table 1.3.2-2. Projected Population Distribution from the MFFF Site – 2000

Direction	5 to 10 mi	10 to 20 mi	20 to 30 mi	30 to 40 mi	40 to 50 mi	Total
N	2,362	24,440	10,482	7,623	11,927	56,834
NNE	268	2,031	2,372	4,674	19,477	28,822
NE	9	1,761	3,112	5,974	13,044	23,900
ENE	0	3,736	5,309	5,915	36,303	51,263
E	1	5,441	5,798	12,435	6,284	29,959
ESE	9	2,469	2,938	3,236	3,296	11,948
SE	0	642	5,179	7,281	11,892	24,994
SSE	0	415	779	1,192	2,858	5,244
S	0	621	1,819	7,672	4,058	14,170
SSW	10	889	2,492	5,478	2,954	11,823
SW	125	1,335	5,219	2,386	3,091	12,156
WSW	115	1,736	5,098	2,948	7,010	16,907
W	275	6,875	11,992	10,198	7,933	37,273
WNW	1,573	5,775	147,962	37,022	16,861	209,193
NW	1,256	17,342	92,635	10,699	3,757	125,689
NNW	1,335	22,490	12,774	7,848	3,812	48,259
Total	7,338	97,998	315,960	132,581	154,557	708,434

Note: The figures above use WSRC 2000b for the basis for population projections. This predicts a 14% increase in population within 50 miles of the MFFF for the year 2000 compared to 1990. After reviewing the actual increases from the 2000 census data, MOX Services has determined that the County populations within 50 miles actually increased by 16%. Therefore, the figures above underestimate population increase by 2%. The ISA Summary does not use these populations in any calculations. Accordingly, MOX Services does not believe that the difference in population data is significant enough to warrant updating to the 2000 census.

Data from WSRC 2000b.

Table 1.3.2-3. Projected Population Distribution from the MFFF Site – 2010

Direction	5 to 10 mi	10 to 20 mi	20 to 30 mi	30 to 40 mi	40 to 50 mi	Total
N	2,693	27,862	11,950	8,690	13,596	64,791
NNE	305	2,316	2,704	5,328	22,204	32,857
NE	10	2,008	3,548	6,810	14,870	27,246
ENE	0	4,259	6,052	6,744	41,386	58,441
E	1	6,203	6,610	14,176	7,163	34,153
ESE	10	2,815	3,349	3,690	3,757	13,621
SE	0	732	5,904	8,301	13,557	28,494
SSE	0	473	888	1,359	3,258	5,978
S	0	708	2,074	8,746	4,627	16,155
SSW	12	1,014	2,841	6,245	3,367	13,479
SW	143	1,522	5,950	2,720	3,523	13,858
WSW	131	1,979	5,812	3,361	7,991	19,274
W	313	7,838	13,670	11,626	9,044	42,491
WNW	1,793	6,584	168,676	42,205	19,221	238,479
NW	1,432	19,770	105,604	12,197	4,283	143,286
NNW	1,522	25,639	14,562	8,946	4,346	55,015
Total	8,365	111,722	360,194	151,144	176,193	807,618

Data from WSRC 2000b.

Table 1.3.2-4. Projected Population Distribution from the MFFF Site – 2020

Direction	5 to 10 mi	10 to 20 mi	20 to 30 mi	30 to 40 mi	40 to 50 mi	Total
N	3,070	31,763	13,623	9,907	15,500	73,863
NNE	348	3,640	3,083	6,074	25,312	38,457
NE	12	2,289	4,045	7,763	16,952	31,061
ENE	0	4,855	6,900	7,688	47,180	66,623
E	1	7,071	7,535	16,161	8,166	38,934
ESE	12	3,209	3,818	4,206	4,283	15,528
SE	0	834	6,731	9,463	15,455	32,483
SSE	0	539	1,012	1,550	3,714	6,815
S	0	807	2,365	9,971	5,274	18,417
SSW	13	1,156	3,239	7,119	3,839	15,366
SW	163	1,735	6,783	3,101	4,016	15,798
WSW	150	2,256	6,625	3,831	9,110	21,972
W	357	8,935	15,584	13,254	10,310	48,440
WNW	2,045	7,506	192,291	48,113	21,912	271,867
NW	1,633	22,537	120,389	13,904	4,883	163,346
NNW	1,735	29,228	16,601	10,199	4,954	62,717
Total	9,539	128,360	410,624	172,304	200,860	921,687

Data from WSRC 2000b.

Table 1.3.2-5. Projected Population Distribution from the MFFF Site – 2030

Direction	5 to 10 mi	10 to 20 mi	20 to 30 mi	30 to 40 mi	40 to 50 mi	Total
N	3,500	36,210	15,530	11,294	17,670	84,204
NNE	397	3,010	3,515	6,925	28,857	42,704
NE	14	2,609	4,611	8,850	19,325	35,409
ENE	0	5,535	7,865	8,764	53,785	75,949
E	2	8,061	8,590	18,423	9,310	44,386
ESE	14	3,658	4,352	5,466	488	13,978
SE	0	951	7,673	7,409	17,619	33,652
SSE	0	615	1,154	1,767	4,234	7,770
S	0	920	2,696	11,367	6,013	20,996
SSW	15	1,317	3,692	8,115	4,376	17,515
SW	186	1,978	7,732	3,535	4,579	18,010
WSW	171	2,572	7,553	4,368	10,385	25,049
W	407	10,186	17,766	15,109	11,753	55,221
WNW	2,331	8,556	219,212	54,849	24,980	309,928
NW	1,861	25,692	137,243	15,851	5,567	186,214
NNW	1,978	33,320	18,925	11,627	5,648	71,498
Total	10,876	145,190	468,109	193,719	224,589	1,042,483

Data from WSRC 2000b.

Table 1.3.2-6. Racial and Ethnic Mix of Local Area Population – 1997 (Estimated)

Population Group	Aiken County, SC	Barnwell County, SC	Burke County, GA	Georgia	South Carolina
Total Population	133,980	21,830	22,725	6,478,216	3,486,703
White	74.3%	56.0%	43.8%	71.0%	69.0%
Black	24.9%	43.7%	56.0%	26.9%	29.8%
American Indian, Eskimo, or Aleut	0.2%	0.2%	0.1%	0.2%	0.3%
Asian or Pacific Islander	0.6%	0.1%	0.2%	1.1%	0.6%
Hispanic (any race)	1.0%	0.8%	0.5%	0.6%	0.3%

Data from WSRC 2000b.

Table 1.3.2-7. Economic and Unemployment Data for Counties within 50 Miles of the MFFF

County	1994 Per-Capita Income	1993 Percent of Pop. Below Poverty	Unemployment Rate – 1996 (%)
South Carolina	**$17,710**	**16.6**	**6.0**
Aiken	$19,468	13.8	7.0
Allendale	$12,175	34.3	9.1
Bamberg	$13,253	27.9	9.9
Barnwell	$16,736	21.9	10.9
Colleton	$13,988	24.1	6.8
Edgefield	$15,076	17.4	7.4
Hampton	$14,595	24.4	7.3
Lexington	$20,111	9.8	3.3
McCormick	$12,500	21.1	10.2
Orangeburg	$14,932	25.6	10.4
Saluda	$15,316	17.7	6.6
Georgia	**$20,212**	**16.8**	**4.6**
Bulloch	$14,319	22.4	3.1
Burke	$14,270	29.2	16.0
Columbia	$17,810	7.7	4.1
Glascock	$16,417	16.1	9.0
Jefferson	$15,303	27.7	13.4
Jenkins	$14,098	25.2	4.7
Lincoln	$15,358	17.5	6.4
McDuffie	$16,422	20.7	9.3
Richmond	$19,251	21.9	7.3
Warren	$13,747	27.1	9.8

Data from WSRC 2000b.

Table 1.3.2-8. Income and Poverty Data for the Three-County Local Area

Area	1990 Population	1990 Per-Capita Income	1989 % Population Below Poverty	1994 Per-Capita Income	1993 % Below Poverty
Aiken, SC	120,940	$17,156	14.0	$19,468	13.8
Barnwell, SC	20,293	$13,397	21.8	$16,736	21.9
Burke, GA	20,579	$11,172	30.3	$14,270	29.2
Georgia	6,478,216	$17,123	14.7	$20,212	16.8
South Carolina	3,487,714	$15,106	15.4	$17,710	16.6

Data from WSRC 2000b.

Table 1.3.2-9. Year 2002 SRS Employees (Approximate) by County of Residence

County	WSRC/ M&O	DOE-SR Operations	WSI	Other Employers	Total	Percent
Aiken, SC	6,380	296	360	180	7,216	53.1
Columbia, GA	1,868	66	72	6	2,012	14.8
Richmond, GA	1,577	66	231	25	1,899	14.0
Barnwell, SC	863	11	64	9	947	7.0
Edgefield, SC	224	3	8	1	236	1.7
Other Counties	1,139	17	88	36	1,280	9.4
Total	12,051	459	823	257	13,590	100

Source: Personal Communication (Bozzone 2002).

Table 1.3.2-10. Public School Population within 10 Miles of the MFFF

School	Location	Grades	1998 - 1999 Enrollment
Greendale Elementary	New Ellenton, SC	Pre-K through 5	426
Jackson Middle	Jackson, SC	6 through 8	517
New Ellenton Middle	New Ellenton, SC	6 through 8	263
Redcliff Elementary	Jackson, SC	Pre-K through 5	967
Silver Bluff High	Aiken, SC	9 through 12	914

Data from WSRC 2000b.

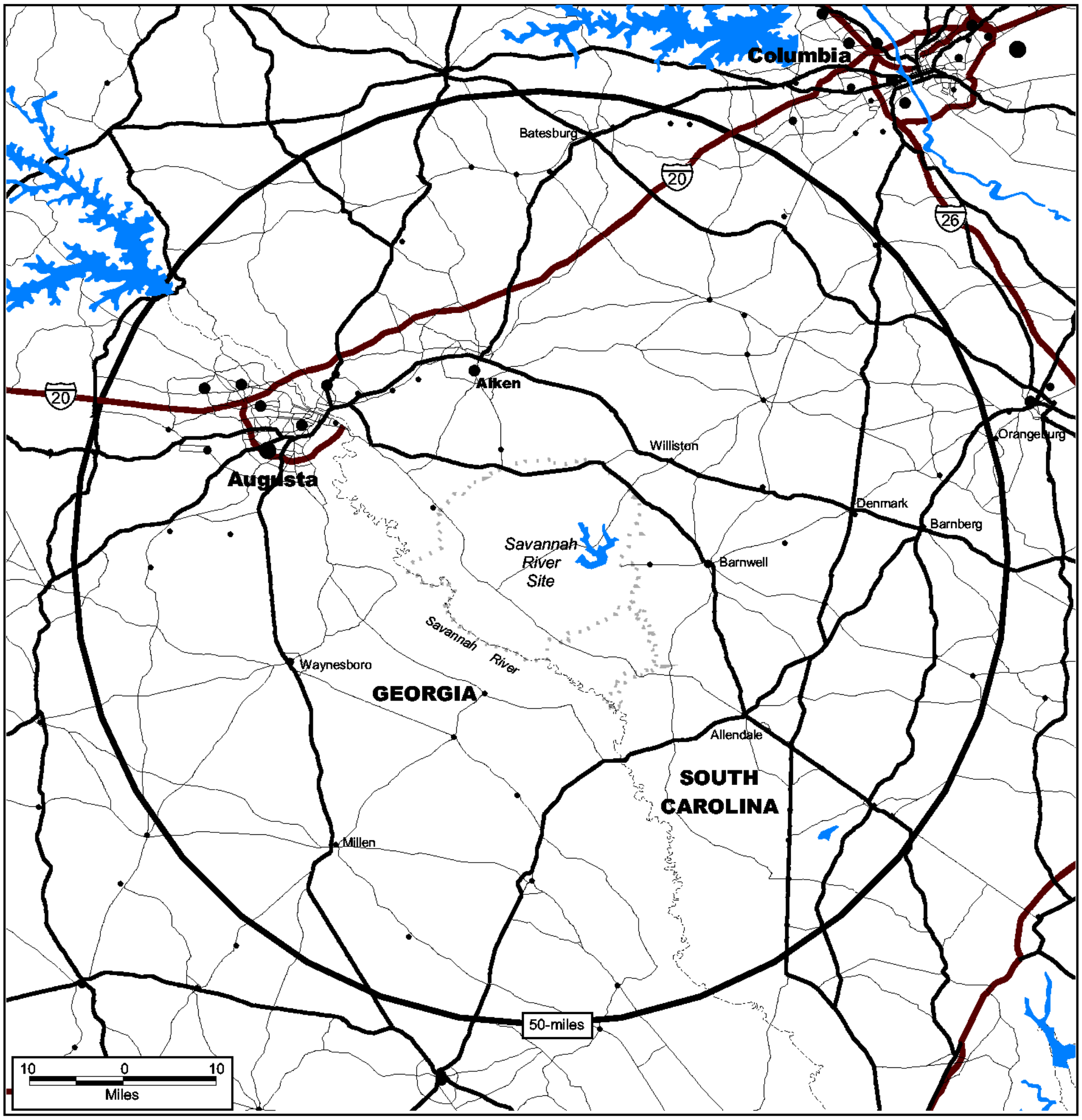
Table 1.3.2-11. Land Use at SRS

Use	Acres
Vegetation Types	
Bottomland Hardwoods	28,492
Upland Hardwoods	6,459
Mixed Hardwood/Pine	10,425
Swamp Species	9,158
Undrained Flatwoods	551
Longleaf Pine	40,804
Loblolly Pine	63,952
Slash Pine	21,616
Other Pine	265
Permanent Grass Openings	4,419
Non-Forest	12,377
	198,518 (site GIS acres)
Water/Wetlands	
Savannah River Swamp	9,894
Par Pond	2,640
L Lake	1,184
	13,718
Production and Support Areas	
100-C	182
100-K	247
100-L	183
100-P	185
100-R	137
200-E & F	1,058
200-S & H	580
200-Z	182
300-M & 700-A	330
400-D	422
600-B	114
N Area (Central Shops)	375
	3,995

Total	216,231

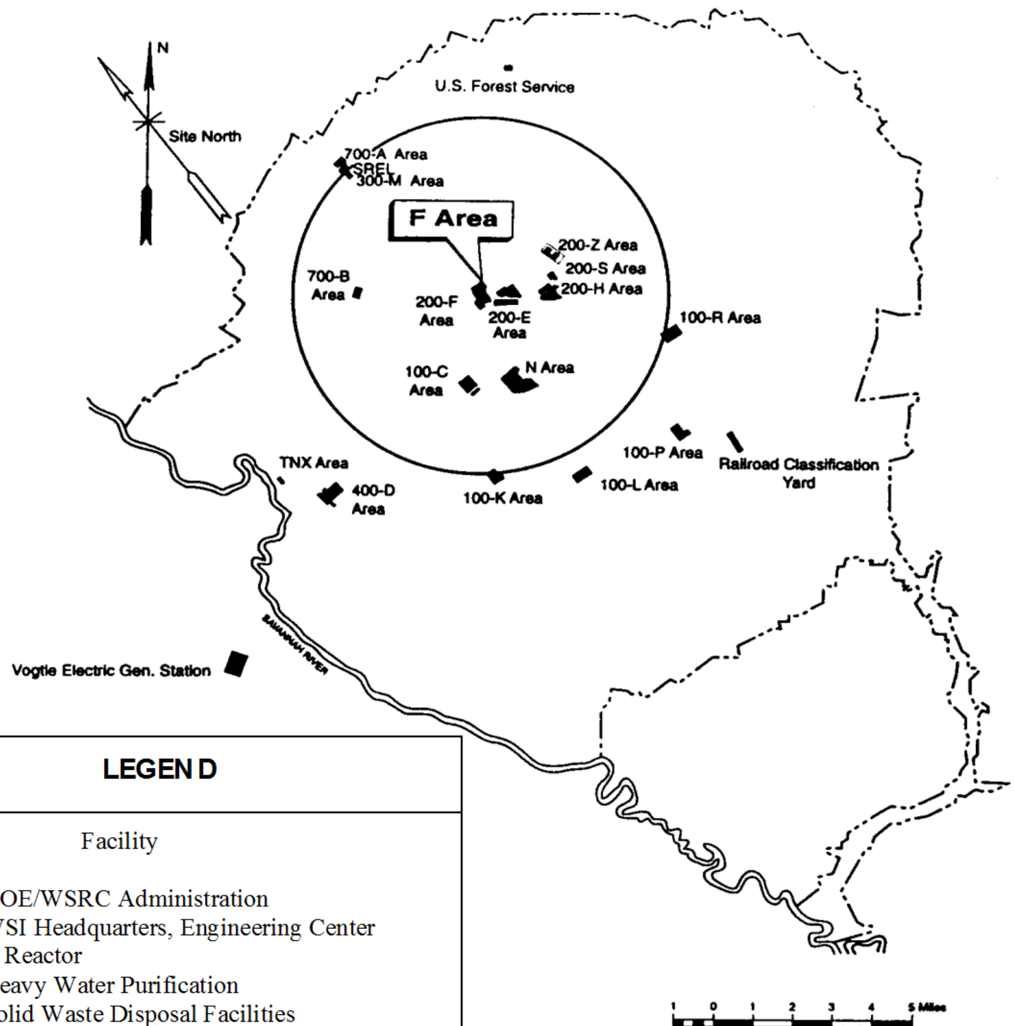
Data from WSRC 2000b.

Figure 1.3.2-1. Map Showing the 50-Mile Radius from the MFFF



Data from WSRC 2000b.

Figure 1.3.2-2. Map Showing the 5-Mile Radius from the MFFF



LEGEND

Area	Facility
A Area	DOE/WSRC Administration
B Area	WSI Headquarters, Engineering Center
C Area	C Reactor
D Area	Heavy Water Purification
E Area	Solid Waste Disposal Facilities
F Area	F Canyon, FB Line, 235-F
H Area	H Canyon, HB Line, Tritium Facility
K Area	K Reactor
L Area	L Reactor
M Area	Reactor Materials Facilities
N Area	Central Shops
P Area	P Reactor
R Area	R Reactor
S Area	DWPF Ops.
Z Area	Saltstone Ops.
Offsite	Vogtle Electric Generating Station

FG 94343150.1

Data from WSRC 2000b.

1.3.3 Meteorology

Climatology information is based on *Climatography of the United States No. 60, Climate of South Carolina* (DOC [U.S. Department of Commerce] 1977) published by the National Climatic Data Center and Section 1.4.1 of the Savannah River Site (SRS) Generic Safety Analysis Report (GSAR) (Washington Savannah River Company, LLC [WSRC] 1999c). It is also based on long-term meteorological data collected by the National Weather Service at Bush Field in Augusta, Georgia as summarized by the National Climatic Data Center. Bush Field is located approximately 12 miles (19.3 km) northwest of SRS. Normals, means, and extremes of temperature, precipitation, and wind speed are taken from the National Oceanic and Atmospheric Administration. Data on tornado occurrences and hurricanes are derived from *Significant Tornadoes 1680 – 1991, Tornado Project of Environmental Films* (Grazulis 1993) and *Natural Phenomena Hazards Design Criteria and Other Characterization Information for the Mixed Oxide (MOX) Fuel Fabrication Facility at Savannah River Site (U)* (WSRC 2000b).

1.3.3.1 Local Wind Patterns and Average and Maximum Wind Speeds

Winds in the SRS region are generally light to moderate with the highest speeds occurring during spring, with an average of approximately 7 mph (11.3 km/hr) for those months at Bush Field. The lightest winds occur in the summer and fall, with the lowest monthly average wind speed of 5.1 mph (8.2 km/hr) occurring in August. The highest monthly average wind speed of 7.7 mph (12.4 km/hr) occurs in March, and the long-term average wind speed for the year is 6.2 mph (10 km/hr) measured at Bush Field. The prevailing wind direction at Augusta is generally from the northwest during the winter months, from the southeast during the late spring and early autumn, and from the southwest in the summer. There is no overall prevailing wind direction because it is variable throughout the year.

A meteorological database, comprised of data from the eight SRS meteorological towers at SRS, for the 10 year period 1987 to 1996 is currently used for the safety analysis. As indicated by this database, there is no strongly prevailing wind direction at SRS. Northeasterly winds occurred approximately 10% of the time, and west to southwest winds occurred about 8% of the time. Annual average wind speeds at each of the towers ranged from 9.4 mph (15.1 km/hr) to 8.0 mph (12.9 km/hr). The maximum one minute wind since 1950 was 83 mph (134 km/hr) measured on May 28, 1950. See Table 1.3.1-1 for a listing of the observed annual fastest one minute wind speeds for SRS.

The peak wind gust at Augusta is 60 mph (96.5 km/hr) from the northwest based on 10 years of observations.

1.3.3.2 Annual Amounts and Forms of Precipitation

Annual average precipitation for SRS over the 30-year period 1967 to 1996 is 49.5 inches (126 centimeters (cms)), and the average precipitation for Augusta is slightly less with 44.7 inches (114 cm). See Table 1.3.3-2 for the average precipitation at SRS. See Table 1.3.3-3 for Augusta Climatological Summary.

Monthly precipitation extremes for SRS range from a maximum of 19.62 inches (50 cm), recorded in October 1990, to a trace observed in October 1963. The greatest observed rainfall for a 24-hour period was 7.5 inches (19 cm) in October 1990. Hourly observations at Augusta indicate that rainfall rates are usually less than 0.5 inches/hour (1.3 cm/hr), although rainfall rates of up to 2 inches/hour (5.1 cm/hr) can occur during summer thunderstorms.

See Table 1.3.3-4 for a summary of the snowfall statistics for Augusta (1951 to 1995). The average annual snowfall for the SRS area (Augusta) for the period 1951 to 1995 was 1.1 inches/year (2.8 cm/yr), and the average number of days per year with snow was 0.6 day.

An average of about 54 thunderstorm days per year was observed in the SRS area during the period 1951 to 1995. See Table 1.3.3-5 for a listing of the average thunderstorm days per month.

The occurrence of hail with thunderstorms is infrequent. Based on observations in a 1^{o} square of latitude and longitude that includes SRS, hail occurs an average of once every two years.

1.3.3.3 Snow and Ice

Snow and ice storms in the region occur very infrequently. Snowfalls of 1 inch (2.5 cm) or greater occur once every three years on the average. Furthermore, any accumulation of snow rarely lasts for more than three days.

The greatest single snowfall recorded in the SRS area (Augusta) during the period 1951 to 1995 occurred in February 1973. This storm produced a total of 14.0 inches (35.6 cm) of accumulation, including 13.7 inches (34.8 cm) in a 24 hour period. See Table 1.3.3-5 for a summary of the maximum total snowfalls for 24 hour and monthly periods, observed at the National Weather Service office at Augusta, Georgia. The maximum ground snow load for the SRS area for a 100 year recurrence period is estimated to be 6 psf.

Ice accumulates on exposed surfaces in the SRS area an average of about once every two years. See Table 1.3.3-6 for the average ice accumulations for various recurrence intervals for a region that includes SRS and consists of the Gulf Coast states. The 100 year recurrence ice storm is estimated to produce an accumulation of approximately 0.67 inches (1.7 cm), which is equivalent to 3 psf.

Based on the values above, the combined snow and ice design basis load for the SRS area for a 100 year recurrence period is 10 psf. This load is considered as a normal design live load in the design of buildings and structures. The ice and snow load is bounded by the allowance that is specified in Section 1.1.2.1 for general live loading effects; therefore, snow and ice do not control the design of MFFF structures, systems, or components (SSCs).

It is also possible to estimate the magnitude of snow and ice loads with a larger interval. See Table 1.3.3-6 for the estimated ice accumulation values. The values listed in Table 1.3.3-6 can be extrapolated to higher recurrence intervals. Using the return period conversions shown in ASCE-7-98, snow loading at higher recurrence intervals can also be extrapolated. With this method, it is estimated that the design basis snow or ice load for a recurrence period of 10,000 years would be approximately twice that for 100 years. Even if the design basis snow and ice loading were increased by this factor to represent a highly unlikely (extreme) snow and ice loading, its magnitude would still be bounded by the allowance for general live loadings and would not control the design of MFFF SSCs. Such highly unlikely snow and ice roof loads are not combined with roof live loads from other sources in the structural evaluations as described in Section 1.1.2.1.

1.3.3.4 Type, Frequency, and Magnitude of Severe Weather

The SRS region occasionally experiences severe weather in the form of violent thunderstorms, tornadoes, and hurricanes. Although thunderstorms are common in the summer months, the more violent storms are commonly associated with squall lines and active cold fronts in the spring. Augusta averages 54 thunderstorm days per year with the highest number of days (9 to 12 days per month) occurring in June, July, and August. The occurrence of hail with thunderstorms is infrequent.

1.3.3.4.1 Tornadoes

A total of 165 tornadoes occurring within a 2º square of latitude and longitude centered on SRS over a 30 year period from 1967 have been identified. See Table 1.3.3-7 for a summary of the tornado occurrences by month and Fujita (F) scale intensity category since 1951.

Nine tornadoes have occurred on or in close proximity to SRS since operations began in the 1950s. A tornado that occurred on October 1, 1989, knocked down thousands of trees over a 16-mile (26-km) path across the southern and eastern portions of the site. Wind speeds produced by this F2 tornado were estimated to be as high as 150 mph (241 km/hr). Four F2 tornadoes struck forested areas of SRS on three separate days during March 1991. Considerable damage to trees was observed in the affected area. The other four confirmed tornadoes were classified as F1 and produced relative minor damage. None of the nine tornadoes caused damage to buildings.

See Table 1.3.3-8 for estimates of the expected tornado wind speeds that are exceeded at SRS for various return frequencies. These estimates were determined from a tornado wind hazard model developed by Lawrence Livermore National Laboratory (LLNL). See Table 1.3.3-9 for a summary of the estimated wind speed. In this table, each of the return intervals represents a mean of the resulting set of wind speed values.

See Table 1.3.3-9 for the design basis tornado speeds for the DOE moderate hazard (performance category or PC-3) and high hazard (PC-4) facilities. In this table, the PC-3 and PC-4 design basis atmospheric pressure change and the rate thereof, are taken as the rounded values corresponding to the tornado speeds of 180 and 240 mph (290 and 386 km/hr), respectively. MFFF IROFS structures are evaluated for a recurrence interval of 2×10^{-6} for a design basis tornado with a three second tornado speed of 240 mph (386 km/hr).

1.3.3.4.2 Extreme Winds

Extreme winds in the SRS area, excluding tornado winds, are associated with tropical weather systems, thunderstorms, or strong winter storms. See Table 1.3.3-1 for a summary of the extreme fastest one minute wind speeds for the 30 year period 1967 to 1996.

See Table 1.3.3-8 for a summary of the estimates of an expected maximum "straight-line" (nontornadic) wind speed (three second gust) for any point on the site for return periods from 100 to 100,000 years. These estimates were generated from a Fisher-Tippet Type I extreme value distribution function using historical wind speed (gust) data from the SRS meteorological database and from nearby National Weather Service stations (Columbia in South Carolina and Augusta, Macon, and Athens in Georgia).

See Table 1.3.3-8 for the design basis wind speeds for DOE moderate hazard (PC-3) and high hazard (PC-4) facilities. MFFF IROFS structures are evaluated for a recurrence interval of 1×10^{-4} for design basis wind with a three second wind speed of 130 mph (209 km/hr).

1.3.3.4.3 Hurricanes

A total of 36 hurricanes have caused damage in South Carolina over the 293 year period from 1700 to 1992. The average frequency of occurrence of a hurricane in the state is once every eight years; however, the observed interval between hurricane occurrences has ranged from two months to 27 years. See Table 1.3.3-10 for the percentages of hurricane occurrences by month in South Carolina.

Because SRS is approximately 100 miles (160 km) inland, winds associated with tropical weather systems usually diminish below hurricane force, which is sustained speeds of 75 mph (120 km/hr) or greater, before reaching SRS. However, winds associated with Hurricane Gracie, which passed to the north of SRS on September 29, 1959, were measured as high as 75 mph (120 km/hr) on an anemometer located in F Area. No other hurricane-force wind has been measured on the site. On September 22, 1989, the center of Hurricane Hugo passed about 100 miles (160 km) northeast of SRS. The maximum 15 minute average wind speed observed onsite during this hurricane was 38 mph (61 km/hr). The highest observed instantaneous wind speed was 62 mph (100 km/hr).

1.3.3.4.4 Extreme Precipitation

See Table 1.3.3-11 for a summary of the maximum observed rainfall recorded at Augusta's Bush Field and the Columbia, South Carolina airport for various accumulation periods. This data were based on a 48 year period of record (1948 to 1995).

See Table 1.3.3-12 for the estimates of expected maximum rainfall at SRS for rainfall durations of 15 minutes to 24 hours and return periods from 10 years to 100,000 years. These estimates were based on a statistical analysis of hourly rainfall from eight National Weather Service first-order and cooperative stations (Augusta, Macon, Athens, Sylvania, and Louisville in Georgia and Columbia, Wagener, and Clark Hill in South Carolina), 15 minute rainfall from three of the cooperative stations (Sylvania, Louisville, and Wagener), and daily rainfall from four rain gages at SRS. Stations were selected based on proximity to and geographic similarity with SRS. For each station (as appropriate to the data set), the annual maximum observed rainfall for each of the six duration intervals of interest over the available period of record was determined. The period of record ranged from 25 to 47 years.

Significant rainfall events occurred at SRS in the summer and fall of 1990. See Table 1.3.3-12 for the observed rainfall totals from those storms that exceeded the predicted extreme rainfall values. Short duration extreme rainfalls are generally produced by spring and summer thunderstorms. Longer duration extreme rains are usually produced by the remnants of tropical weather systems. MFFF IROFS structures are evaluated for an annual recurrence frequency of 1×10^{-5} for extreme precipitation values. See Table 1.3.3-12 for the extreme precipitation recurrence estimates by accumulation period.

1.3.3.4.5 Lightning

The frequency of cloud-to-ground lightning strikes has been estimated using conservative input values. The number of flashes to earth per square kilometer was estimated to be 10 per year. Measurements of cloud-to-ground lightning strikes recorded from the National Lightning Detection Network over the five year period 1989 to 1993 show an average of four strikes per square kilometer per year in the SRS area.

1.3.3.5 Temperature

See Table 1.3.3-13 for the monthly and annual average temperatures for SRS for the 30 year period 1967 to 1996. At SRS, the annual average temperature is 64.7°F. July is the warmest month with an average daily high temperature of 92.1°F and an average daily low temperature of 71.5°F. January is the coldest month with an average daily high temperature of 55.9°F and an average daily low temperature of 36.0°F. Observed temperature extremes for SRS over the period 1961 to 1996 ranged from 107°F to -3°F. Data for Augusta, Georgia indicate that prolonged periods of cold weather seldom occur. Daytime high temperatures during the winter months are rarely below 32°F. Conversely, high temperatures in the summer months are above 90°F on more than half of the days. The average dates of the first and last freeze are November 12 and March 16, respectively.

Table 1.3.3-1. Observed Annual Fastest One Minute Wind Speeds for SRS [a, b, c]

Year	Wind Speed (mph) [d]	Direction	Date
1967	52	W	5/8
1968	43	NW	7/16
1969	43	NE	7/8
1970	52	NW	7/16
1971	34	SW	7/11
1972	56	SW	3/2
1973	37	NW	11/21
1974	49	W	3/21
1975	37	W	7/6
1976	32	NW	3/9
1977	43	S	10/2
1978	39	SW	1/26
1979	30	W	5/12
1980	32	S	7/9
1981	33	NW	3/16
1982	40	NW	2/16
1983	32	NW	12/31
1984	32	SW	3/28
1985	35	W	2/11
1986	32	NW	7/2
1987	35	NNW	7/24
1988	32	WNW	5/24
1989	39	NW	6/22
1990	28	WSW	1/29
1991	29	NW	2/15
1992	29	SW	7/1
1993	33	W	3/13
1994	34	SE	7/10
1995	38	W	11/11
1996	35	W	2/12

[a] Maximum one minute wind since 1950: 83 mph on 5/28/50

[b] Data for 1967-1994 from National Weather Service Office, Bush Field, Augusta, Georgia.
Source: *Local Climatological Data, Annual Summary with Comparative Data, 1995, Augusta, Georgia* (DOC 1996)

[c] Data for 1995-1996 from SRS Central Climatology Facility.
Source: "Updated Meteorological Data for Revision 4 of the SRS Generic Safety Analysis Report" (Hunter 1999)

[d] Values interpolated to a 10 m anemometer height.

Data from WSRC 2000b.

Table 1.3.3-2. Average and Extreme Precipitation at SRS (Water Equivalent),in Inches

Month	Average[a]	Maximum (Year)[b]	Minimum (Year)[b]
January	4.44	10.02 (1978)	0.89 (1981)
February	4.25	7.97 (1995)	0.94 (1968)
March	4.83	10.96 (1980)	0.91 (1995)
April	3.02	8.20 (1961)	0.57 (1972)
May	3.86	10.90 (1976)	1.33 (1965)
June	4.53	10.98 (1973)	0.89 (1990)
July	5.57	11.48 (1982)	0.90 (1980)
August	5.44	12.34 (1964)	1.04 (1963)
September	3.63	8.71 (1959)	0.49 (1985)
October	3.40	19.62 (1990)	0.00 (1963)
November	2.89	7.78 (1992)	0.21 (1958)
December	3.59	9.55 (1981)	0.46 (1955)
Year	49.46	73.47 (1964)	28.82 (1954)

[a] Period of record: 1967-1996.
[b] Period of record: 1952-1996.

Table 1.3.3-3. Augusta[a] Climatological Summary-Precipitation (Inches)

Month	Normal Monthly	Maximum Monthly	Year Occurred	Minimum Monthly	Year Occurred	24 Hour Maximum	Year Occurred
January	4.05	8.91	1987	0.75	1981	3.61	1960
February	4.27	7.67	1961	0.69	1968	3.69	1985
March	4.65	11.92	1980	0.88	1968	5.31	1967
April	3.31	8.43	1961	0.60	1970	3.96	1955
May	3.77	9.61	1979	0.48	1951	4.44	1981
June	4.13	8.84	1989	0.68	1984	5.08	1981
July	4.24	11.43	1967	1.02	1987	3.71	1979
August	4.50	11.34	1986	0.65	1980	5.98	1964
September	3.02	9.51	1975	0.31	1984	7.30	1998
October	2.84	14.82	1990	T	1953	8.57	1990
November	2.48	7.76	1985	0.09	1960	3.82	1985
December	3.40	8.65	1981	0.32	1955	3.12	1970
Year	44.66	14.82	1990	T	1953	8.57	1990

Source: *Local Climatological Data, Annual Summary with Comparative Data, 1991* (DOC 1991)

T – Trace

[a] Taken at Bush Field, Augusta, Georgia national weather station

Data from WSRC 2000b.

Table 1.3.3-4. Maximum Snow/Ice Pellets Augusta, Georgia (Inches)

Month	Average	Maximum (Year)	24 Hour Maximum (Year)
January	0.3	2.6 (1992)	2.6 (1992)
February	0.7	14.0 (1973)	13.7 (1973)
March	<0.1	1.1 (1980)	1.1 (1980)
April	0.0	0.0	0.0
May	0.0	0.0	0.0
June	0.0	0.0	0.0
July	0.0	0.0	0.0
August	0.0	0.0	0.0
September	0.0	0.0	0.0
October	0.0	0.0	0.0
November	<0.1	Trace (1968)	Trace (1968)
December	0.1	1.0 (1993)	1.0 (1993)
Year	1.1	14.0 (1973)	13.7 (1973)

Period of record: 1951-1995

Data from WSRC 2000b.

Table 1.3.3-5. Average Number of Thunderstorm Days - Augusta, Georgia (1951-1995)

Month	Thunderstorm Days
January	0.8
February	1.7
March	2.6
April	3.9
May	6.3
June	9.7
July	13.1
August	10.0
September	3.5
October	1.3
November	0.8
December	0.7
Annual	54.4

Data from WSRC 2000b.

Table 1.3.3-6. Estimated Ice Accumulation for Various Recurrence Intervals for the Gulf Coast States

RECURRENCE INTERVAL (YEAR)	ACCUMULATION (INCHES)
2	0
5	0.24
10	0.39
25	0.51
50	0.59
100	0.66

Data from WSRC 2000b.

Table 1.3.3-7. Number of Tornadoes Reported Between 1951 and 1996 by Month and F Scale in a 2º Square Centered at SRS

MONTH	F0	F1	F2	F3	F4	F5	TOTAL	PERCENT
January	3	8	2	1	0	0	14	7.0
February	4	12	1	0	0	0	17	8.5
March	1	10	9	0	1	0	21	10.5
April	4	17	4	1	0	0	26	13.0
May	3	18	6	0	0	0	27	13.5
June	4	10	0	0	0	0	14	7.0
July	2	8	3	0	0	0	13	6.5
August	4	7	5	2	0	0	18	9.0
September	0	5	3	0	0	0	8	4.0
October	1	2	4	0	0	0	7	3.5
November	10	8	7	2	0	0	27	13.5
December	1	2	2	2	1	0	8	4.0
Total	37	107	46	8	2	0	200	100.0

Data from WSRC 2000b.

Table 1.3.3-8. Estimated Maximum Three Second Wind Speeds for Tornadoes and "Straight-Line" Winds

Recurrence Interval (Years)	Probability Events/Year	Estimated Maximum 3-Second Wind Speed (mph)	
		Tornadoes	"Straight-Line" Winds
100	1×10^{-2}	---	88
200	5×10^{-3}	---	94
500	2×10^{-3}	---	102
1,000	1×10^{-3}	70	107
5,000	2×10^{-4}	120	120
10,000	1×10^{-4}	135	126
50,000	2×10^{-5}	180	140
100,000	1×10^{-5}	200	145
500,000	2×10^{-6}	240	---
1,000,000	1×10^{-6}	251	---

Data from WSRC 2000b.

Table 1.3.3-9. Wind and Tornado Design Criteria for SRS

	Item	PC-3	PC-4
Wind	Annual Hazard Exceedance Probability	$1x10^{-3}$	$1x10^{-4}$
	3-Sec Wind Speed, mph	110 rounded up value	130 rounded up value
	Missile Criteria	2” x 4” timber plank 15 lb @50 mph (horizontal); max height 30 ft	2” x 4” timber plank 15 lb @50 mph (horizontal); max height 50 ft
	ASCE 7-98 [a]		
Tornado	Annual Hazard Exceedance Probability	$2x10^{-5}$	$2x10^{-6}$
	3- Sec Tornado Speed, mph	180	240
	Atmospheric Pressure Change (APC), psf, at the rate of psf/sec	70 psf at 31 psf/sec	150 psf at 55 psf/sec
	Missile Criteria	2” x 4” timber plank 15 lb @100 mph (horizontal); max height 150 ft; 70 mph (vertical) 3“diameter standard steel pipe, 75 lb @50 mph (horizontal); max height 75 ft; 35 mph (vertical) 3000 lb automobile @19 mph rolls and tumbles	2” x 4” timber plank 15 lb @150 mph (horizontal); max height 200 ft; 100 mph (vertical) 3” diameter standard steel pipe, 75 lb @75 mph (horizontal); max height 100 ft; 50 mph (vertical) 3000 lb automobile @25 mph rolls and tumbles
	ASCE 7-98 [a]		

[a] For determining wind and tornado loads using the ASCE 7-98 procedure, the following definitions apply: I = 1.0, Exposure Category = C, K_{zt} = 1.0, and K_d = 1.0.

Data from WSRC 2000b.

Table 1.3.3-10. Total Occurrences of Hurricanes in South Carolina by Month (1700-1992)

Month	Number	Percent of Total
June	1	2.8
July	2	5.6
August	11	30.5
September	18	50.0
October	4	11.1

Data from WSRC 2000b.

Table 1.3.3-11. Extreme Total Rainfall for SRS Region (August 1948-December 1995)

Hours	Days	Period	Time	Date
Augusta Bush Field				
1		3.14	1300	7/24/86
3		4.25	1900	9/20/75
6		4.50	1900	9/20/75
12		7.62	2100	10/11/90
24		8.57	1300	10/11/90
	3	12.24		10/10/90
	7	12.24		10/10/90
	10	12.24		10/10/90
	14	14.56		10/10/90
	30	15.47		9/30/90
	60	19.84		7/15/64
	90	25.88		7/18/64
Columbia Airport				
1		3.80	2000	8/18/65
3		5.03	1900	8/18/65
6		5.29	1700	6/15/73
12		7.03	2200	8/16/49
24		7.66	1600	8/16/49
	3	8.41		8/14/90
	7	10.22		6/15/73
	10	10.29		6/13/73
	14	14.71		8/14/49
	30	19.30		7/29/49
	60	25.64		6/18/71
	90	33.69		7/18/64

Data from WSRC 2000b.

Table 1.3.3-12. Extreme Precipitation Recurrence Estimates by Accumulation Period

Recurrence Interval (Yrs)	15 min	1 hr	3 hr	6 hr	24 hr	48 hr
10	1.5	2.7	3.3	3.6	5.0	6.5
						7.39[B]
25	1.8	3.2	4.0	4.4	6.1	7.9
50	2.0	3.5	4.6	5.0	6.9	8.6
					(7.39)[B]	
100	2.1	3.9	5.1	5.7	7.8	9.4
			(5.2)[A]	(5.8)[B]		(10.2)[C]
						(11.15)[D]
1,000	2.7	5.0	7.4	8.3	11.5	N/A
10,000	3.3	6.2	10.3	11.8	16.3	N/A
100,000	3.9	7.4	14.1	16.7	22.7	N/A

[a] July 25 rainfall at the 700 Area.
[b] August 22 rainfall at the Climatology Site.
[c] October 11-12 rainfall at the 773-A Area.
[d] October 11-12 rainfall at Bush Field.

Data from WSRC 2000b.

Table 1.3.3-13. Monthly Average and Extreme Temperatures for SRS

Average Daily Temperature (°F [a])				Extreme Temperature (°F [b])	
Month	**Maximum**	**Minimum**	**Month**	**Max(yr)**	**Min (yr)**
January	55.9	36.0	45.8	86 (1975)	-3 (1985)
February	60.0	38.3	49.1	86 (1989)	10 (1996)
March	68.6	45.4	57.0	91 (1974)	11 (1980)
April	77.1	52.5	64.8	99 (1986)	29 (1983)
May	83.5	60.7	72.1	102 (1963)	38 (1989)
June	89.6	68.0	78.8	105 (1985)	48 (1984)
July	92.1	71.5	81.7	107 (1986)	56 (1963)
August	90.1	69.6	80.3	107 (1983)	56 (1986)
September	85.4	65.6	75.4	104 (1990)	41 (1967)
October	76.6	54.6	65.6	96 (1986)	28 (1976)
November	67.0	45.2	56.2	89 (1974)	18 (1970)
December	59.3	39.1	49.1	82 (1984)	5 (1962)
Annual	75.5	54.0	64.7	107 (1986)	-3 (1985)

[a] Period of record: 1967-1996.

[b] Period of record: 1961-1996.

Data from WSRC 2000b.

1.3.4 Hydrology

1.3.4.1 Surface Hydrology

1.3.4.1.1 Hydrologic Description

Much of SRS is located on the Aiken Plateau. The plateau slopes to the southeast approximately 5 feet/miles (1 m/km). The plateau is dissected by streams that drain into the Savannah River. The major tributaries that occur on SRS are Upper Three Runs, Fourmile Branch, Pen Branch, Steel Creek, and Lower Three Runs. Beaver Dam Creek, the smallest of the six SRS tributaries of the Savannah River, is located north of Fourmile Branch, primarily in the floodplain of the Savannah River. Tinker Creek and Tims Branch are tributaries of Upper Three Runs; Indian Grave Branch is a tributary of Pen Branch. Each creek originates on the Aiken Plateau and descends 49 to 200 ft (15 to 61 m) before discharging to the Savannah River. The interstream upland area is flat to gently rolling and is characterized by gently dipping units of sand, sandy clay, and clayey sand.

The Savannah River is the principal surface water system near SRS. The river adjoins the site along its southwestern boundary for a distance of about 17 miles (27.4 km) and is 140 river miles (225 river km) from the Atlantic Ocean.

The Savannah River cuts a broad valley approximately 250 ft (76.2 m) deep through the Aiken Plateau. Pleistocene coastal terraces lie between the Savannah River and the Aiken Plateau. The lowest terrace is the Savannah River floodplain, which is covered with a dense swamp forest. Higher terraces rise successively from the river floodplain to the Aiken Plateau and have a level to gently rolling topography.

The Savannah River Swamp lies in the floodplain along the Savannah River for a distance of about 10 miles (16.1 km) and averages about 1.5 miles (2.4 km) wide. A small embankment or natural levee has built up along the north side of the river from sediments deposited during periods of flooding. The top of the natural levee is approximately 3 to 6 ft (0.9 to 1.8 m) above the river during normal flow (river stage 85 ft [25.9 m]) at the SRS boat dock. Three breaches in this levee (at the confluences with Beaver Dam Creek, Fourmile Branch, and Steel Creek) allow discharge of stream water to the river. During periods of high river level (above 88 ft [26.8 m]), river water overflows the levee and stream mouths and floods the entire swamp area. The water from these streams mixes with river water and then flows through the swamp parallel to the river and combines with the Pen Branch flow. The flows of Steel Creek and Pen Branch converge 0.5 miles (0.8 km) above the Steel Creek mouth. However, when the river level is high, the flows are diverted parallel to the river across the offsite Creek Plantation Swamp; ultimately they join the Savannah River flow near Little Hell Landing.

Surface water is held in artificial impoundments and natural wetlands on the Aiken Plateau. Par Pond, the largest impoundment on SRS, is an artificial lake located in the eastern part of the site that covers approximately 2,700 ac (1,093 ha). A second large artificial impoundment, L Lake, lies in the southern portion of SRS and covers approximately 1,000 ac (405 ha). Water from both Par Pond (200 ft [61 m]) and L Lake (190 ft [57.9 m]) drains to the south via Lower Three Runs and Steel Creek, respectively, into the Savannah River. Water is also retained intermittently in natural lowland and upland marshes and natural basins, some of which are Carolina bay depressions.

The source of most of the surface water on SRS is either natural rainfall, which averages 48 inches (122 cm) annually, water pumped from the Savannah River and used for cooling site facilities, or groundwater discharging to the surface streams. Cooling water is discharged to streams that flow back to the Savannah River, L Lake, or Par Pond. Small volumes of water are also discharged from other SRS facilities to the streams.

The flow data used for computing statistics for the Savannah River and SRS streams were obtained from U.S. Geological Survey (USGS) stream measurement data. The data set consisted of daily average flows with varying periods of record (from 2 to 81 years) for SRS streams and the Savannah River.

Flow statistics were derived from this data set over the period of record: daily minimums, maximums, and means; average flow; seven-day low flow, and the seven-day flow with a 10-year recurrence interval (7Q10) flow. Emphasis was placed on low flow statistics because disposal of wastes and maintenance of conditions for aquatic life are usually based on some type of low flow statistic. The seven-day low flow is widely used and is less likely to be influenced by minor disturbances upstream than is the minimum daily flow. The 7Q10 flow is a measure of the dependability of flow. The 7Q10 flow is derived from the frequency curve of the yearly seven-day low flow statistics over the period of record at that stream or river location. The Log Pearson Type III distribution statistics are normally used for computation of low flows in natural streams. Other distributions may be more appropriate in streams that are not naturally driven (such as those where cooling water may be the dominant component of flow).

The Log Pearson Type III distribution was applied to SRS stream locations where a 7Q10 flow was computed (a program equivalent to the USGS A193 for computing Log Pearson Type III distributions was used). The climatic year, April 1 to March 31, is used for calculation of low flow statistics. In the United States, this period contains the low flow period for each year. See Table 1.3.4-1 for a summary of the flow statistics (average flow, standard deviation, 7Q10 flow, and seven-day low flow).

The Savannah River drainage basin has a total area of 10,600 mi^2 (27,454 km^2) and forms the boundary between Georgia and South Carolina. The total drainage area of the river encompasses all or part of 41 counties in Georgia, South Carolina, and North Carolina. The Savannah River Basin is located in three physiographic regions or provinces: the Mountain, the Piedmont, and the Coastal Plain.

The Mountain Province contains most of the major tributaries of the Savannah River, including the Seneca, Tugaloo, and Chattooga Rivers. The region is characterized by a relatively steep gradient ranging in elevation from about 5,497 to 1,000 ft (1,675 to 305 m) and includes 2,042 mi^2 (5,289 km^2) (19%) of the total drainage basin. The Mountain Province lies in the Blue

Ridge Mountains and has bedrock composed of gneisses, granites, schists, and quartzites; the subsoil is composed of brown and red sandy clays. In this region, the Savannah River and its tributaries have the character of mountain streams with shallow riffles, clear creeks, and a fairly steep gradient. The streambed is mainly sand and rubble.

The Piedmont Region has an intermediate gradient with elevations ranging from 1,000 to 200 ft (305 to 61 m). This region includes 5,234 mi^2 (13,556 km^2)(50%) of the total drainage basin. Soils in the Piedmont are primarily red, sandy, or silty clays with weathered bedrock consisting of ancient sediments containing granitic intrusions. The Piedmont is bordered by the Fall Line, an area where the sandy soils of the Coastal Plain meet the rocky terrane of the Piedmont foothills. The city of Augusta, Georgia is located near this line. The Savannah River picks up the majority of its silt load in the Piedmont Region, and most of this silt load is deposited in the large reservoirs located in the Piedmont Region.

The Coastal Plain has a negligible gradient ranging from an elevation of 200 ft (61 m) to sea level. The soils of this region are primarily stratified sand, silts, and clays. The Coastal Plain contains 3,366 mi^2 (8,718 km^2) (31%) of the total Savannah River drainage area (10,681 mi^2 [27,664 km^2]) and includes the city of Savannah, Georgia. In the Coastal Plain, the Savannah River is slow moving. Tidal effects may be observed up to 40 miles (65 km) upriver, and a salt front extends upstream along the bottom of the riverbed for about 20 miles (32 km).

Dredging operations on the Savannah River have been conducted by the U.S. Army Corps of Engineers between the cities of Savannah and Augusta, Georgia. This program, initiated in October 1958, was designed to dredge and maintain a 9-ft (2.7-m) navigation channel in the Savannah River from Savannah to Augusta, Georgia. Sixty-one sets of pile dikes were placed to constrict the river flow, thereby increasing flow velocities, and 38,000 linear feet of wood and stone revetment was laid to reduce erosion on banks opposite from the dikes. In addition, the channel was dredged and 31 cutoffs were made, reducing the total river distance from Augusta to Savannah by about 15 river miles (24.1 river km). The project was completed in July 1965; periodic dredging was continued to maintain the channel until 1985.

SRS is located in the Coastal Plain Province of the Savannah River, about 25 miles (40.2 km) downstream of Augusta, Georgia. Construction of upriver reservoirs (Strom Thurmond, Richard B. Russell, Hartwell, Keowee, and Jocassee) and the New Savannah River Bluff Lock and Dam have reduced the variability of the river flow. Low flows in the Savannah River typically occur during the autumn months while higher flows occur in late winter and early spring.

Upstream of SRS at Augusta, Georgia, the average flow for the 81-year period of record is 10,027 cfs. The average flow at Augusta since the filling of Thurmond Lake (Clarks Hill) has been 9,571 cfs (Table 1.3.4-1). Flows increase below Augusta to about 12,009 cfs near Clyo, Georgia, about 100 miles (161 km) downriver (Table 1.3.4-1). The 7Q10 flow at Augusta is 3,746 cfs.

The peak historic flow for the 81-year period of record was 350,021 cfs in 1929. Since the construction of the upstream reservoirs, the maximum average monthly flow has been 43,867 cfs for the month of April.

Natural discharge patterns on the Savannah River are cyclic: the highest river levels are recorded in the winter and spring, and lowest levels are recorded in the summer and fall. Stream flow on the Savannah River near the site is regulated by a series of three upstream reservoirs: Thurmond, Russell, and Hartwell. These reservoirs have stabilized average, annual stream flow to 10,200 cfs near Augusta and 10,419 cfs at SRS.

The river overflows its channel and floods the swamps bordering the site when its elevation rises higher than 88.5 ft (27 m) above mean sea level (msl) (which corresponds to flows equal to or greater than 15,470 cfs). River elevation measurements made at the SRS Boat Dock indicate that the swamp was flooded approximately 20% of the time (74 days per year on the average) from 1958 through 1967.

The Savannah River forms the boundary between Georgia and South Carolina. Upstream of SRS, the river supplies domestic and industrial water needs for Augusta, Georgia, and North Augusta, South Carolina. The river receives treated wastewater from these municipalities and from Horse Creek Valley (Aiken, South Carolina). The Savannah River Class B waterway is used for commercial and sport fishing and pleasure boating downstream from SRS.

Water withdrawn from the river is used for various SRS activities. The Savannah River downstream from Augusta, Georgia, is classified by the State of South Carolina as a Class B waterway, which is suitable for agricultural and industrial use, the propagation of fish, and after treatment, domestic use. The river upstream from the site supplies municipal water for Augusta, Georgia (river mile 187 [river km 301]), and North Augusta, South Carolina (river mile 201 [river km 323]). Downstream, the Beaufort-Jasper Water Authority in South Carolina (river mile 39.2 [river km 63]) withdraws water to supply a population of about 51,000. The Cherokee Hill Water Treatment Plant at Port Wentworth, Georgia (river mile 29.0 [river km 46.7]) withdraws water to supply a business-industrial complex near Savannah, Georgia, that has an estimated consumer population of about 20,000. It is estimated that each individual served by the two water treatment plants consumes an average of 0.34 gal (1.3 L) of water per day. Site expansions for both systems are planned for the future.

SRS was once a major user of water from the Savannah River and withdrew a maximum of 920 cfs from the river. Currently, SRS reactors are shut down, and river water withdrawals are minimal. Past operations typically removed about 9% of the average annual Savannah River flow, but river water usage averaged 0.133 cfs during the second quarter of 1995.

In 1995, DOE decided to discharge a minimum flow of 10 ft^3 (0.28 m^3) per second to Lower Three Runs and to allow the water level in Par Pond to fluctuate naturally near its operating level (200 ft [61 m] above msl) but not allowing the water level to fall below 195 ft (59.4 m).

Additionally it was decided to reduce the flow to L Lake so long as the normal operating level of 190 ft (57.9 m) was maintained and the flow in Steel Creek (downstream of L Lake) was greater than 10 cfs.

Currently, only one of the pumps at pumphouse 3G is operated; it supplies 23,000 gpm (1.5 m^3/sec), which is more than is needed for system uses. The excess water is discharged from reactor areas to Fourmile Branch, Pen Branch, L Lake, and the headwaters of Steel Creek.

The river also receives sewage treatment plant effluents from Augusta, Georgia; North Augusta, Aiken, and Horse Creek Valley, South Carolina; and other waste discharges along with the heated SRS cooling water via its tributaries. VEGP withdraws an average of 92 cfs from the river for cooling and returns an average of 25 cfs. The Urquhart Steam Generating Station at Beech Island withdraws approximately 261 cfs of once-through cooling water. Upstream, recreational use of impoundments on the Savannah River, including water contact recreation, is more extensive than it is near SRS and downstream. No uses of the Savannah River for irrigation have been identified in either South Carolina or Georgia.

The Beaufort-Jasper Water Authority in South Carolina (river miles 39.2 [river km 63]) withdraws about 8 cfs to supply domestic water for a population of about 51,000. The Cherokee Hill Water Treatment Plant at Port Wentworth, Georgia (river miles 29.0 [river km 46.7]) withdraws about 50 cfs from the river to supply a business-industrial complex near Savannah, which has an estimated consumer population of about 20,000.

Based on available information, the following sections describe surface hydrology in reference to specific local facilities.

1.3.4.1.1.1 F Area and MFFF Site

Surface drainage for F Area and surrounding land drains into Upper Three Runs and Fourmile Branch. The MFFF site is adjacent to F Area. See Figure 1.3.4-5 for a diagram of the MFFF in F Area. The MFFF site is at an elevation of approximately 270 ft (82.3 m) above msl. The nearest significant stream to F Area and the MFFF site is Upper Three Runs, which is located about 0.7 miles (1.1 km) north and west of F Area. Upper Three Runs flows at elevations below 150 ft (45.7 m) above msl and has a mean annual flow at a gauging station approximately 3 miles (4.8 km) from F Area of 215 cfs and approximately 245 cfs at its mouth. The measured maximum flow for the period 1974 to 1986 was about 950 cfs. Runoff due to precipitation from F Area and the MFFF site is diverted into storm sewers and then discharged into an unnamed tributary of Upper Three Runs, which empties into the Savannah River.

1.3.4.1.2 Hydrosphere – Surface Waters

Surface water includes marine or freshwater bodies that occur above the ground surface, including rivers, streams, lakes, ponds, rainwater catchments, embayments, and oceans.

1.3.4.1.2.1 Savannah River

SRS is bounded on the southwest for approximately 17 miles (27 km) by the Savannah River. Six streams flow through SRS and discharge into the Savannah River: Upper Three Runs, Beaver Dam Creek, Fourmile Branch, Pen Branch, Steel Creek, and Lower Three Runs. Upper Three Runs has two tributaries (Tims Branch and Tinker Creek); Pen Branch has one tributary (Indian Grave Branch); and Steel Creek has one tributary (Meyers Branch).

The Savannah River Basin is one of the major river basins in the southeastern United States. It has a drainage area of 10,577 mi^2 (27,394 km^2) of which 8,160 mi^2 (21,134 km^2) are upstream of SRS. The headwaters of the Savannah River are in the Blue Ridge Mountains of North Carolina, South Carolina, and Georgia. The river forms at the junction of the Tugaloo and Seneca Rivers approximately 100 miles (161 km) northwest of SRS, now the site of Hartwell Reservoir, and empties into the Atlantic Ocean near Savannah, Georgia, approximately 95 miles (153 km) southeast of SRS. From the Hartwell Reservoir Dam to the Savannah Harbor, the river runs a course of 289 river miles (465 river km).

Three large reservoirs on the Savannah River upstream of SRS provide hydroelectric power, flood control, and recreation. Strom Thurmond Reservoir (2.51 million acre-feet), completed in 1952 (Table 1.3.4-2), is approximately 35 miles (56.3 km) upstream of SRS. The Richard B. Russell Reservoir (1.026 million acre-feet), completed in 1984, is approximately 72 miles (116 km) upstream of SRS. Hartwell Reservoir (2.549 million acre-feet), completed in 1961, is approximately 90 miles (145 km) upstream of SRS. These three dams are owned by the U.S. Army Corps of Engineers. The Stevens Creek Dam, also on the Savannah River, is owned by SCE&G.

Additional dams lie upstream of Hartwell Reservoir and are used primarily for hydroelectric power generation. The Yonah, Tugaloo, Tallulah Falls, Mathis, Nacoochee, and Burton Dams are owned by Georgia Power Company, and the Keowee, Little River, and Jocassee Dams are owned by Duke Power Company. Although many of these dams impound water to depths in excess of 100 ft (30.5 m), only the Jocassee Dam and the combined Little River-Keowee Dams impound significant quantities (approximately 1 million acre-feet each).

See Section 1.3.4.1.1 for information about the dredging operations on the Savannah River. The Savannah River is gauged above SRS near Augusta, Georgia (station 02197000), 0.5 miles (0.8 km) downstream from Upper Three Runs at Ellenton Landing (station 02197320), at Steel Creek (station 02197357), and below SRS at Burtons Ferry Bridge (station 02197500) and 3 miles (4.8 km) north of Clyo, Georgia (station 02198500). Since upstream stabilization, the yearly average flow of the Savannah River near SRS has been approximately 10,419 cfs. See Section 1.3.4.2 for information about the flow extremes. The elevation of the river at SRS pumphouses is 80.4 ft (24.5 m) above msl at a flow of 5,800 cfs. The Savannah River has a flow of 5,800 cfs and has an average velocity of approximately 2 mph at VEGP, which is across the river from SRS. The river is about 340 ft (1.4 m) wide and from 9 to 16 ft (2.7 to 4.9 m) deep. The minimum flow that is required for navigation downstream from Strom Thurmond Dam is 5,800 cfs. From SRS, river water usually reaches the coast in approximately five to six days but may take as few as three days.

Three locations below the mouth of Upper Three Runs pump raw water from the Savannah River for drinking water supplies. The Cherokee Hill Water Plant at Port Wentworth, Georgia, can withdraw about 70 cfs for an effective consumer population of about 20,000. The Beaufort-Jasper Water Treatment Plant at Hardeeville, South Carolina, can withdraw about 12 cfs for a consumer population of approximately 51,000. The SRS D Area, downstream of the mouth of Upper Three Runs, removes approximately 0.1 cfs from the river.

Savannah River water is also used for industrial water cooling purposes by several facilities. SRS is a major user, with intake points downstream of the confluence of Upper Three Runs with the Savannah River. SRS could remove 1,450 cfs with the pumps in three pumphouses concurrently in use, but it usually withdraws a maximum of 1,320 cfs from the river. The coal-fired power plant in D Area receives about 100 cfs, and Par Pond receives about 20 cfs to compensate for seepage and evaporation. VEGP uses 100 cfs and SCE&G's Urquhart Steam Station, located between Augusta and SRS, uses 260 cfs.

1.3.4.1.2.2 Upper Three Runs

Upper Three Runs is the longest of the onsite streams. It drains an area of over 195 mi^2 (505 km^2) and differs from the other five onsite streams in two respects; it is the only stream with headwaters arising outside the site and it is the only stream that has never received heated discharges of cooling water from the production reactors. Upper Three Runs and its tributaries receive contaminants migrating from industrial and nuclear facilities in F, E, and H Areas. Tims Branch receives primarily treated industrial waste waters from M Area, SRTC, a small coal-fired plant, and treated sanitary wastewater and remediated groundwater from A and M Areas.

See Section 1.3.4.2 for information about the minimum and maximum flow history for Upper Three Runs. The Upper Three Runs stream channel has a low gradient and is meandering, especially in the lower reaches. Its floodplain ranges in width from 0.25 to 1 miles (0.4 to 1.6 km) and contains extensive stands (about 98% coverage) of bottomland hardwood forest. Within SRS, the Upper Three Runs valley is asymmetrical, having a steep southeastern side and a gently sloping northwestern side.

Upper Three Runs is gauged near Highway 278 (station 02197300 relocated downstream), at SRS Road C (station 02197310), and at SRS Road A about 3 miles (4.8 km) above the confluence of Upper Three Runs with the Savannah River (station 02197315). The Highway 278 station is a National Hydrologic Benchmark Station. Benchmark streams are measured monthly for water flow, temperature, and quality to provide hydrologic data on river basins governed by natural conditions.

The average Upper Three Runs flow at Highway 278 from 1966 to 1986 was 106 cfs, which represents a water yield of about 1.0 ft^3/mi^2 or 16.55 in/yr from the drainage basin. The average annual precipitation at SRS is 48.3 inches (123 cm). Thus, in the upper reaches of Upper Three Runs, about 35% of the rainfall appears as stream discharge. Flow rates are also measured downstream of the Highway 278 site at SRS Road C and at SRS Road A. Average daily flows were calculated to be 102, 203, and 251 cfs, respectively. The minimum daily flow rates recorded at these sites during this period were 45, 117, and 124 cfs, respectively.

1.3.4.1.2.3 Fourmile Branch

Fourmile Branch drains about 23 mi^2 (59.6 km^2) within SRS, including much of F, H, and C Areas. The creek flows to the southwest into the Savannah River Swamp and then into the Savannah River. The valley is V-shaped, with the sides varying from steep to gently sloping. The floodplain is up to 1,000 ft (305 m) wide. No human population resides in the Fourmile Branch drainage.

Fourmile Branch receives effluents from F, H, and C Areas; and a groundwater plume from a radioactive waste burial ground (southeast of F Area), F-Area Seepage Basin (southwest of F Area), and H-Area Seepage Basin (use discontinued in November 1988). Until June 1985, it received large volumes of cooling water from the production reactor in C Area. The creek valley has been modified by the cooling water discharge, which has created a delta into the Savannah River Swamp. Downstream of this delta area, it re-forms into one main channel, and most of the flow discharges into the Savannah River at river miles 152.1 (river km 245). When the Savannah River floods, water from Fourmile Branch flows along the northern boundary of the floodplain and joins with other site streams to exit the swamp via Steel Creek instead of flowing directly into the Savannah River. Fourmile Branch also receives contaminants migrating from the F- and H-Area Seepage Basins and the Solid Waste Management Facility.

Water flow measurements have been made on Fourmile Branch near SRS Road A-12.2 at SRS (station 02197344) since November 1976. Mean monthly flows for water years 1986 and 1987, after C-Reactor shutdown, ranged from 88 cfs in January 1986 to 17 cfs in August 1987. Extreme flows for this period were 436 cfs (gage height 3.14 ft [0.96 m]) on March 1, 1987, to 13 cfs on August 24-25 and 28-29, 1987. The maximum and minimum discharges for the period of record are 903 cfs (gage height 3.93 ft [1.2 m]) on March 13, 1980, and 13 cfs on August 24-25 and 28-29, 1987, respectively.

1.3.4.1.3 Environmental Acceptance of Effluents

The National Pollutant Discharge Elimination System permitted outfalls within SRS are identified in the annual SRS Environmental Report (WSRC 1997b).

1.3.4.1.4 Chemical and Biological Composition of Adjacent Watercourses

1.3.4.1.4.1 Upper Three Runs

The Upper Three Runs valley is swampy with a meandering and braided channel, especially in the lower reaches. In SRS, the stream has a gradient of approximately 5.3 ft/miles (l m/km). Upper Three Runs stream channel sediments have sand as the dominant fraction, with silt plus clay fractions increasing to about 40% at SRS Road A.

Upper Three Runs is a slightly dystrophic, large, cool, blackwater stream that flows into the Savannah River. The stream is neutral to somewhat acidic and carries a relatively low load of suspended and dissolved organics compared to other streams of the southeastern Atlantic Coastal Plain. Suspended solid loads are heaviest during periods of highest stream flow, normally late winter to early spring when vegetative cover is reduced. From the upper to lower reaches, the suspended load increases substantially. Although inorganic sediments are preferentially deposited in the floodplains, there is a concurrent input of organics from the floodplains, which causes an increase in total suspended solids (mostly organic matter). This increase is more pronounced in periods when the stream overflows its banks and floods the surrounding swamps. Water quality samples for Upper Three Runs are collected monthly, and the data are presented in the annual SRS Environmental Report (WSRC 1997b).

The water of Upper Three Runs is soft, usually clear, and low in nutrients. The temperature ranges from approximately 41°F to 78°F (5°C to 26°C), with lows occurring from December through February. The highest temperature and lowest flow are normally observed in July. Temperature, pH, and dissolved oxygen levels in the stream meet South Carolina Water

Classification Standards for Class B streams. Conductivity, suspended solids, and alkalinity concentrations increase in the downstream direction, but the concentrations are low at the stations. Nutrient levels are also low, although phosphorus and nitrate levels are highest during the spring and summer, possibly due to offsite agricultural activities.

The effluents include process wastes, cooling water, surface runoff, and ash basin effluent. The F/H Area Effluent Treatment Facility (ETF) discharges into Upper Three Runs near SRS road C. Tims Branch, a tributary to Upper Three Runs, has received trace amounts of radioactivity and heavy metals contamination and is currently receiving elevated levels of nitrates from M Area. Total discharges to Upper Three Runs range from approximately 10 gpm to over 1,000 gpm. By comparison, the minimum recorded flow at the Highway 278 gage about 10 miles (16.1 km) upstream on Upper Three Runs is 66 cfs, which is approximately 30,000 gpm.

The cation exchange capacity (CEC) ranged from 0.1 to 12.4 milliequivalent per 100 grams (meq/100 g) in the Upper Three Runs and tributary sediments, indicating low CEC values throughout the Upper Three Runs watershed. Elevated levels of nickel were found in Tims Branch sediments, probably originating from the nickel-plating operations in M Area. Sediments from the Upper Three Runs watershed exhibited background levels of ^{137}Cs ($\leq$ 2 pCi/g) and naturally occurring radionuclides (^{40}K, radium, ^{208}Tl, and natural uranium).

The swamp forest of the Upper Three Runs floodplain consists primarily of bald cypress (*Taxodium distichum*) and tupelo gum (*Nyssa aquatica*), while the bottomland hardwoods associated with the stream are primarily sweet gum (*Liquidambar styraciflua*), red oak (*Quercus rubra*), and beech (*Fagus grandifolia*). The stream is well shaded in most reaches.

Leaf litter input is high, and the leaves are rapidly broken down by macroinvertebrate shredders. The relatively complete canopy results in low periphyton and macrophyte biomass, especially in summer when the creek is most shaded. The periphytons that do occur are largely green algae and diatoms.

Sampling conducted in 1984 and 1985 found ichthyoplankton densities to be low, with spotted suckers the dominant taxon. Crappie and darters also composed a large portion of the overall ichthyoplankton population. The dominant fish species found were redbreast sunfish, spotted suckers, channel catfish, and flat bullhead. Species numbers tend to peak in the spring and drop in the summer.

1.3.4.1.4.2 Fourmile Branch

Fourmile Branch originates on SRS and flows southwest across the site toward the Savannah River. In the Savannah River Swamp, when C Reactor operated, part of Fourmile Branch flowed to Beaver Dam Creek, which flows directly into the Savannah River through a breach in the natural levees. With C Reactor shut down, Fourmile Branch flows parallel to the river behind the natural levees and enters the river through a breach downriver from Beaver Dam Creek.

Fourmile Branch receives nonradioactive effluents from C, F, and H Areas, which increase the hardness, nutrient content, and trace metal concentrations in the water. From March 1955 to June 1985, Fourmile Branch also received 180,000 gpm of cooling water from the production reactor in C Area. During this period, water quality in the thermal reaches of the creek generally reflected the waters of the Savannah River, which served as source water for C Reactor.

Fourmile Branch has been greatly influenced by the temperature and volume of cooling water it once received from the C-Area production reactor. The native swamp forest has been eliminated, and the stream is mostly unshaded. Above its thermal reach, the water quality of Fourmile Branch resembles that of other nonthermal streams on the site. Samples taken from 1983 to 1985 showed this portion of Fourmile Branch to have higher conductivity, nitrate (as N), calcium, and sodium levels than Upper Three Runs. Levels of copper, cadmium, mercury, nickel, lead, chromium, and zinc were at or near detection limits. Water temperatures in the nonthermal reaches of Fourmile Branch averaged approximately 62.6°F (17°C), with highs usually less than 86°F (30°C).

Water quality samples for Fourmile Branch are collected monthly, and the data are presented in the annual SRS Environmental Report (WSRC 1997b). The mean temperature, the pH range, and the mean dissolved oxygen concentration were similar to those for Upper Three Runs at SRS road C during the same period. The mean concentrations of most other parameters measured were higher or approximately equal to those for Upper Three Runs. Turbidity, volatile solids, chemical oxygen demand, nitrites, nitrates, and manganese were lower in Fourmile Branch than in the lower reaches of Upper Three Runs (measured at SRS road C).

When C Reactor was in operation, only the thermophilic blue-green algae (i.e., *Phormidium* and *Oscillatoria* spp.) survived regularly in waters exceeding 122°F (50ºC). Leaf decomposition was low due to the absence of macroinvertebrate shredders. The macroinvertebrate populations exhibited low biomass and low densities except for some oligochaetes, nematodes, and chironomids that were more tolerant to heat. Upstream from this zone, diatoms were the predominant and most diverse primary producers. Blue-green algae of the genera Microcoleus, Schizothrix, and Oscillatoria were found in decaying organic surfaces such as submerged logs and leaf litter. Besides the thermophilic blue-green algae, the mosquito fish was the only other survivor during periods of thermal stress. Following reactor shutdown in 1985, the macroinvertebrate density and biomass increased. Many fish species have readily reinvaded during this period, and fish catch rates have increased markedly. It is expected that the current biology of Fourmile Branch will more closely resemble that of other site streams.

1.3.4.2 Floods

This section describes the flood history and potential types of flooding at SRS. The section also discusses design considerations, frequency and effects of locally intense precipitation at the MFFF site, and flood protection requirements at the MFFF site. The probable maximum flood (PMF) at the site is characterized by a water level of 224.5 ft (68.4 m) above msl (see Section 1.3.4.2.3). The design basis flood for the MFFF site is based on an annual recurrence frequency of 1 x 10^{-5} and is associated with a water level of 207.9 ft (63.4 m) above msl (see Section 1.3.4.2.4).

1.3.4.2.1 Flood History

The floods represented by the data in this section were the result of excess precipitation runoff and the associated creek or stream flooding. No floods have been caused by surge, seiche, dam failure, or ice jams.

1.3.4.2.1.1 Flood History of the Savannah River

See Table 1.3.4-3 for the annual maximum daily flows of the Savannah River. Historical records span from 1796 to1999. The earliest historical data were determined primarily from high-water marks; flow gauging by the USGS began in 1882. The record historical flood at Augusta, Georgia, occurred in 1796, with an estimated discharge of 360,000 cfs; the peak flow recorded by the USGS (350,000 cfs) occurred on October 3, 1929. Since Strom Thurmond Dam was constructed, no major flood has occurred at Augusta, Georgia. The U.S. Army Corps of Engineers simulated the October 3, 1929, storm event using current control requirements. The unregulated peak flow of 350,000 cfs resulted in a regulated peak flood flow of 252,000 cfs at Augusta, Georgia.

A statistical analysis of Savannah River annual maximum flows downstream at Augusta, Georgia, was conducted using the Log Pearson Type III distribution. For the 30-year period from 1921 to 1950, before construction of Strom Thurmond Dam, the mean annual maximum flow was 92,600 cfs, the 10-year maximum flow was 211,000 cfs, and the estimated 50-year maximum flow was 362,000 cfs. After construction of the Strom Thurmond Dam, the Savannah River flows were controlled to meet various demands: hydroelectric power, water supply allocations, flood control, water qualities, habitat, recreation, and aquatic plant control. For the 44-year period from 1956 to 1999, after construction of Strom Thurmond Dam, the mean annual maximum flow, based on mean daily flow rates, was 36,300 cfs, the 10-year maximum flow was 55,400 cfs, and the estimated 50-year maximum flow was 74,600 cfs.

1.3.4.2.1.2 Flood History of Upper Three Runs

See Table 1.3.4-4 for a listing of the instantaneous annual maximum flows for Upper Three Runs gauging stations at Highway 278 near SRS road C and at SRS road A. The station at Highway 278 has the longest historical record.

For Upper Three Runs at Highway 278, the maximum flood recorded was 820 cfs on October 23, 1990, and the corresponding flood stage elevation was 183.5 ft (55.9 m) above msl. Similarly, the maximum flow at SRS road C was 2,040 cfs (129.4 ft [39.4 m] above msl) on October 12, 1990, and at SRS Road A was more than 2,580 cfs (97.9 ft [29.8 m] above msl) on October 12, 1990. No dams are located in the Upper Three Runs watershed.

1.3.4.2.1.3 Flood History of Tims Branch

See Table 1.3.4-5 for a listing of the annual maximum daily flows for station 02197309 on Tims Branch at SRS road C. Data for water years 1974, 1975, and 1977 to 1984 were not available.

The maximum flood discharge recorded for Tims Branch was 129 cfs on October 12, 1990, with a corresponding gage height of approximately 145.7 ft (44.4 m) above msl. The highest flood stage level recorded was approximately 146.7 ft (44.7 m) above msl on May 29, 1976.

1.3.4.2.1.4 Flood History of Fourmile Branch

See Table 1.3.4-6 for a listing of the annual instantaneous maximum flows for Fourmile Branch gage stations at SRS road C, SRS road A-7, and SRS road A-12.2. The maximum floods occurred on August 2, 1991. The flood elevation at SRS road C was 194.2 ft (59.2 m) above msl, at SRS road A-7 was 161.9 ft (49.3 m) above msl, and at SRS road A-12.2 was 116.7 ft (35.6 m) above msl.

1.3.4.2.2 Flood Design Considerations

The MFFF site is located on topographic high points and is well inland from the coast. The only significant impoundments, Par Pond and L Lake, are relatively small and sufficiently lower than the MFFF site. There is no safety threat to the MFFF site from high water.

The calculated PMF water level for the Savannah River at the VEGP site is 118 ft (36 m) above msl without wave run-up. With wave run-up, the water may reach as high as 165 ft (50.3 m) above msl. Because the minimum plant grade near the MFFF site is approximately 270 ft (82.3 m) above msl, it is well above the flood stage. If the valley storage effect between Strom Thurmond Dam and VEGP is taken into account, this results in a lower flood peak and lower flood stage.

Flood levels due to precipitation, as a function of return period (annual probability of exceedance), for the Upper Three Runs, Tims Branch, Fourmile Branch, and Pen Branch basins have been calculated. Results indicate that the probabilities of flooding at F Area and the MFFF site are significantly less than 1.0E-05 per year. The basin hydrologic routing method was used to calculate the flood level as a function of the annual probability of exceedance (see Section 1.3.4.2.4.)

1.3.4.2.3 Effects of Local Intense Precipitation

This section describes flood design considerations for F Area and the MFFF site. Unusually intense local rainfalls occurred on SRS on July 25, 1990; August 22, 1990; October 10-12, 1990; and October 22-23, 1990. Although over 6 inches (15.2 cm) of rain fell in a 10-mi^2 (25.9 km^2) area during the August 22 storm, this amount is just 20% of the six-hour PMP of 31.0 inches (78.7 cm).

1.3.4.2.3.1 F Area and MFFF Site

The six-hour, 10-mi^2 (25.9 km^2) PMP is 31 inches (78.7 cm), with a maximum intensity of 15.1 inches (38.4 cm) in one hour. This rainfall was adjusted to a point PMP of 19 inches (48.3 cm) in one hour and used to generate the PMF for the small watershed of the unnamed tributary on Upper Three Runs. The unnamed tributary on Upper Three Runs is near F Area and is located about 0.40 miles (0.6 km) northwest of F Canyon. Incremental rainfall for one-hour periods adjacent to the PMP was also determined as shown in Table 1.3.4-7. A synthetic hydrograph was used to determine peak flow. The peak stage corresponding to the PMF is 224.5 ft (68.4 m) above msl or approximately 45 ft (13.7 m) below the F-Area and MFFF site grade. Because F Area and the MFFF site lie near a watershed divide, incident rainfall naturally drains away from the facilities.

Unusual short-duration heavy rainfall occurred in F Area in August 1990 and October 1990. Total rainfall measured in F Area was as follows:

- On August 22, 1990, 6.1 inches (15.5 cm) of rainwater was collected
- Between October 11 and 12, 1990, about 10 inches (25.4 cm) was collected.

1.3.4.2.4 Flood Hazard Recurrence Frequencies

Flood levels have been calculated due to precipitation as a function of annual probability of exceedance for Upper Three Runs, Tims Branch, Fourmile Branch, Pen Branch, and Steel Creek upstream from L-Lake basins. A basin hydrologic routing method was employed to calculate the flood level as a function of the annual probability of exceedance. The procedures used for the method are presented next.

Step 1. Hyetographs (rainfall depth or intensity as a function of time) for various return periods were synthesized based on rainfall intensity-duration-frequency data.

Step 2. The Hydrologic Modeling System computer code was used to calculate basin peak flow based on the hyetograph for a given return period and basin properties.

Step 3. The peak flow calculated by HEC-HMS (Step 2) was then used in the Computer Model for Water Surface Profile Computations (WSPRO) to calculate the flood water elevations.

WSPRO was developed by the USGS for the Federal Highway Administration. WSPRO uses a step-backwater analysis method to calculate water surface elevations for one-dimensional, gradually varied, steady flow through bridges and overtopping embankments.

Step 4. Steps 2 and 3 were repeated for each return period.

Steps 1 through 4 were applied to both the Upper Three Runs and Fourmile Branch basins.

1.3.4.2.4.1 Design Basis Flood

Flood flows and elevations for the Upper Three Runs, Tim Branch, Fourmile Branch, and Pen Branch basins were calculated by the steps described above. See Table 1.3.4-8 for the synthesized 24-hour storm hyetographs for various annual probabilities of exceedance. See Table 1.3.4-9 for the calculated flood elevations at A, C, E, F, H, K, S, Y, and Z Areas. See Table 1.3.3-10 for the MFFF site design basis flood as a function of performance category, respectively. The design basis flood for DOE moderate hazard (PC-3) and high hazard (PC-4) facilities is given in Table 1.3.4-10. MFFF IROFS structures are evaluated for an annual recurrence frequency of 1 x 10^{-5} for a design basis flood with an elevation of 207.9 ft (63.4 m) above msl. The elevation of the MFFF site is 272 ft (82.9 m) above msl. The MFFF site is a dry site for consideration of the design basis flood.

1.3.4.2.5 Potential Dam Failures (Seismically Induced)

1.3.4.2.5.1 Reservoir Description

The only significant dams or impoundment structures that could affect the safety of SRS are large dams on the Savannah River and its tributaries upstream of Augusta, Georgia. See Section 1.3.4.1 for information on these structures. The Stephens Creek Dam is owned by SCE&G. The other dams on the Savannah River are owned by the U.S. Army Corps of Engineers. The dams on the Tugaloo and Tallulah rivers are owned by Georgia Power Company. The dams on the Keowee and Little Rivers are owned by Duke Power Company.

1.3.4.2.5.2 Dam Failure Permutations

A domino failure of the dams on the Savannah River and its tributaries upstream of VEGP was analyzed in the VEGP Final SAR (Georgia Power Company 1987). The worst possible case resulted from Jocassee Dam failing during a combined standard project flood and earthquake, with the resulting chain reaction.

Using conservative assumptions, this worst dam failure would yield a peak flow of 2,400,000 cfs at Strom Thurmond Dam. This rate, undiminished in magnitude, was transferred to below Augusta, Georgia. However, because of the great width of the floodplain, routing of the dam failure surge to the VEGP site (river miles 151 [river km 243]) resulted in a peak discharge of 980,000 cfs, with a corresponding stage of 141 ft (43 m) above msl.

1.3.4.2.5.3 Unsteady Flow Analysis of Potential Dam Failures

No dams are located near SRS areas. Therefore, this section does not apply.

1.3.4.2.5.4 Water Level at Facility Site

See Section 1.3.4.2.2 for information about the peak water surface elevation of the Savannah River that corresponds to wave run-up of a wind-induced wave, superimposed upon the passage of a flood wave resulting from a sequence of dam failures.

1.3.4.2.6 Probable Maximum Surge and Seiche Flooding

No large water bodies exist near the site; therefore, this section does not apply. Run-up of flood waters from the worst combination of wind and waves on the Savannah River is not a hazard at the site because the peak flood elevation is well below minimum plant grade, and the maximum wave under the worst circumstances is less than 3 ft (0.9 m).

1.3.4.2.7 Ice Flooding

Because of regional climatic conditions, the formation of significant amounts of ice on streams and rivers rarely occurs. The Hartwell, Richard B. Russell, and Strom Thurmond dams moderate water temperature extremes, making ice formation on the Savannah River at SRS unlikely.

No historical ice flooding has been noted, although ice has been observed in the Savannah River on occasion. Because the sites are so much higher than the nearest streams and rivers, it is not considered credible that they could be affected by ice flooding, even if the climatic conditions were conducive to ice formation.

1.3.4.2.8 Flooding Protection Requirements

Because the site is located on a local topographic high, there is no threat to SRS from flooding, as described in previous sections. Special flooding protection requirements are not necessary to ensure the safety of F Area and the MFFF site because they are located at elevations well above the maximum flood.

1.3.4.3 Regional Hydrogeology (Within 75-Mile Radius)

1.3.4.3.1 Regional Hydrogeological Setting

Two hydrogeologic provinces are recognized in the subsurface beneath the SRS region. The uppermost province, which consists of the wedge of unconsolidated Coastal Plain sediments of Late Cretaceous and Tertiary ages, is referred to as the Southeastern Coastal Plain hydrogeologic province. It is further subdivided into aquifer or confining systems, units, and zones. The underlying province, referred to as the Piedmont hydrogeologic province, includes Paleozoic metamorphic and igneous basement rocks and Upper Triassic lithified mudstone, sandstone, and conglomerate in the Dunbarton basin (see Section 2.5.1).

The following hydrogeological characteristics are of particular interest.

- The layered structure of the coastal plain sediments effectively controls migration of contaminants in the subsurface, limiting vertical migration to deeper aquifers.
- Between the ground surface and the primary drinking water aquifer(s) are several low permeability zones, which restrict vertical migration from a given point source.
- The abundance of clay size material and clay minerals in the aquifer and aquitard zones affects groundwater composition and vertical migration. The concentration of some potential contaminants, especially metals and radionuclides, may be attenuated by exchange and fixation of dissolved constituents on clay surfaces.
- The recharge area(s) for the deeper drinking water aquifers used are updip of SRS, near the Fall Line. Some recharge areas are located at the northernmost fringe of the site.
- Recharge for the water table aquifers, namely the Upper Three Runs and Gordon aquifers, is primarily from local precipitation.

- Discharge of groundwater from the Upper Three Runs and Gordon aquifers is typically to the local streams on SRS.
- Groundwater at SRS is typically of low ionic, low dissolved solids and moderate pH (typically ranging from 4.4 to 6.0). Other constituents such as dissolved oxygen and alkalinity are more closely associated with recharge and aquifer material. Dissolved oxygen is typically higher in the updip and near-surface recharge areas, and alkalinity, pH and dissolved solids are typically higher in those portions of the aquifer regions containing significant carbonate materials.
- The presence of an upward vertical gradient or "head reversal" between the Upper Three Runs and Gordon aquifers and the Crouch Branch aquifer is significant in that it prevents downward vertical migration of contaminants into deeper aquifers over much of central SRS.

1.3.4.3.2 Hydrostratigraphic Classification and Nomenclature of Coastal Plain Sediments

The method for establishing a nomenclature for the hydrogeologic units in the following discussion generally follows the guidelines set forth by the South Carolina Hydrostratigraphic Subcommittee.

A hydrogeologic unit is defined by its hydraulic properties (i.e., hydraulic conductivity, hydraulic head relationships, porosity, leakance coefficients, vertical flow velocity, and transmissivity) relative to those properties measured in the overlying and underlying units. The properties are measured at a type well or type well cluster location. Aquifers and confining units are mapped on the basis of the hydrogeologic continuity, potentiometric conditions, and leakance-coefficient estimates for the units. These properties are largely dependent on the thickness, areal distribution, and continuity of the lithology of the particular unit. However, a hydrogeologic unit may traverse lithologic unit boundaries if there is not a significant change in hydrogeologic properties corresponding to the change in lithology.

1.3.4.3.2.1 Delineation and Classification of Units

The hydrostratigraphic classification is based on aquifer and confining units ranked at four levels (I through IV).

1.3.4.3.2.1.1.1 Level I - Hydrogeologic Province

A hydrogeologic province is a major regional rock and/or sediment package that behaves as a single unified hydrologic unit. The names, areal extent, and underlying geological context of the regional hydrogeologic provinces used in this report are the same as those defined by Miller and Renken (1988) as regional hydrologic systems. For example, the "Southeastern Coastal Plain *hydrologic* system" of Miller and Renken reads "Southeastern Coastal Plain *hydrogeologic* province" in this report.

1.3.4.3.2.1.1.2 Level II - Aquifer and Confining Systems

These define the primary or regional units of the hydrogeologic province. The aquifer system may be composed of a single aquifer or two or more coalescing aquifers that transmit groundwater on a regional basis. Aquifer systems may be locally divided by confining units that impede groundwater movement but do not greatly affect the regional hydraulic continuity of the system. A confining system may be composed of a single confining unit or two or more confining units that serve as an impediment to regional groundwater flow.

SRS is located near the updip limit of the aquifer and confining systems comprising the Coastal Plain sediments in the region. Here, the lateral continuity and thickness of the clay and clayey sand beds that constitute the confining systems decrease, and the beds become increasingly discontinuous. Where the clay beds no longer separate the overlying and underlying aquifers, the updip limit of a confining system is defined. Updip from this line, the overlying and underlying aquifer systems coalesce into a single unified aquifer system. Where aquifer systems have combined, some of the individual aquifer and confining units may persist in the updip-combined system.

1.3.4.3.2.1.1.3 Level III - Aquifer and Confining Units

These are the fundamental units of the classification. An aquifer is a mappable (~400 mi^2 [> 1036 km^2]) body of rock or sediments that is sufficiently permeable to conduct groundwater and yield significant quantities of water to wells and springs. A confining unit, on the other hand, is a mappable (~ 400 mi^2 [> 1036 km^2]) body of rock or sediments of significantly lower hydraulic conductivity than an adjacent aquifer that serves as an impediment to groundwater flow into or out of an aquifer. A confining unit's hydraulic conductivity may range from nearly zero to some value distinctly lower than that of the nearby aquifer. The assignment of a unit level and name to a hydrostratigraphic unit does not imply a quantitative ranking of hydraulic continuity but is intended to distinguish relative differences in hydraulic properties between adjacent units. Where the confining unit that separates one aquifer from another thins and becomes laterally discontinuous and/or is breached by faults and fractures, the overlying and underlying aquifers coalesce and a single unified aquifer may be defined.

Aquifers and confining units may be informally subdivided into zones that are characterized by properties significantly different from the rest of the unit, such as hydraulic conductivity, water chemistry, lithology, and/or color. For example, an aquifer may contain a “confining zone” such as the “tan clay” confining zone of the Upper Three Runs aquifer. Conversely, a confining unit may contain an “aquifer zone” such as the “middle sand” aquifer zone of the Crouch Branch confining unit. The “Fernandina permeable zone” is a zone in the Lower Floridan aquifer in coastal areas of Georgia, where the permeability greatly exceeds that of the rest of the aquifer.

In the study area, zonal differentiation is undertaken on a local site-specific scale where useful and necessary distinctions are made in the hydraulic characteristics of specific aquifer or confining units. Thus, the intermittent but persistent clay beds in the Dry Branch Formation, informally referred to as the “tan clay” in previous SRS reports, is designated the “tan clay” confining zone of the Upper Three Runs aquifer. The Dry Branch clay in the Separations area of SRS is a confining zone and is commonly known as “tan clay.”

1.3.4.3.3 Southeastern Coastal Plain Hydrogeologic Province

The Southeastern Coastal Plain hydrogeologic province underlies 120,000 mi^2 (310,799 km^2) of the Coastal Plain of South Carolina, Georgia, Alabama, Mississippi, and Florida and a small contiguous area of southeastern North Carolina. This hydrogeologic province grades laterally to the northeast into the Northern Atlantic Coastal Plain aquifer system and to the west into the Mississippi embayment and Coastal Lowlands aquifer systems. In South Carolina, the northern and northwestern limits of the province are its contact with crystalline rocks at the Fall Line, which marks the updip limit of Coastal Plain sediments.

The Southeastern Coastal Plain hydrogeologic province comprises a multilayered hydraulic complex in which retarding beds composed of clay and marl are interspersed with beds of sand and limestone that transmit water more readily. Groundwater flow paths and flow velocity for each of these units are governed by the unit's hydraulic properties, the geometry of the particular unit, and the distribution of recharge and discharge areas. The Southeastern Coastal Plain hydrogeologic province can be divided into seven regional hydrologic units. These are four regional aquifer units separated by three regional confining units. Six of the seven hydrologic units are recognized in the SRS area and are referred to as hydrogeologic systems. These systems have been grouped into three aquifer systems divided by two confining systems, which are underlain by the Appleton confining system. The Appleton separates the Southeastern Coastal Plain hydrogeologic province from the underlying Piedmont hydrogeologic province.

In descending order, the aquifer systems beneath SRS are the Floridan aquifer system, the Dublin aquifer system, and the Midville aquifer system. In descending order, the confining systems are the Meyers Branch confining system, the Allendale confining system, and the Appleton confining system.Beneath SRS, the Midville and Dublin aquifer systems each consists of a single aquifer, the McQueen Branch aquifer and Crouch Branch aquifer, respectively. Downdip, beyond SRS, these aquifer systems are subdivided into several aquifers and confining units.

The Floridan aquifer system consists of two aquifers in the study area: the Upper Three Runs aquifer and the underlying Gordon aquifer, which are separated by the Gordon confining unit. Northward, the Gordon and Upper Three Runs aquifers coalesce to form the Steed Pond aquifer.

The Allendale and Meyers Branch confining systems each consists of a single confining unit in the study area: the McQueen Branch and Crouch Branch confining units, respectively. The basal Appleton confining system is thought to consist of a single confining unit in the study area. The confining unit ("Appleton"), however, has not been formally defined owing to insufficient data. Downdip, each confining system may be subdivided into several confining units and aquifer units.

Where the confining beds of the Allendale confining system no longer regionally separate the Dublin and Midville aquifer systems hydrologically, the Dublin-Midville aquifer system is defined. Similarly, where the Meyers Branch confining system no longer regionally separates the Floridan aquifer system from the underlying Dublin-Midville aquifer system, the entire sedimentary sequence from the top of the Appleton confining system to the water table is hydraulically connected and the Floridan-Midville aquifer system is defined.

In general, the number of aquifer systems present beneath SRS decreases updip due to pinching out of confining units in the updip direction. Thus, in the southern site area, three aquifer systems are designated. As the confining systems become ineffective flow barriers updip, the number of aquifer systems decreases to one (Floridan-Midville) in the northern site region. The stratigraphic position of the two aquifer system areas is dependent on which confining system (Allendale or Meyers Branch) pinches out.

The following discussion treats each of the hydrogeologic units in greater detail. It presents the units in descending order, from the water table to the Piedmont hydrogeologic province. Within each unit, the discussion traces the unit updip. In general, confining layers pinch out and aquifers coalesce in an updip direction.

1.3.4.3.3.1 Floridan Aquifer System

The Floridan aquifer system is defined as a vertically continuous sequence of carbonate rocks of generally high permeability that are mostly of middle and late Tertiary age. The rocks are hydraulically connected in varying degrees, and their permeability is generally an order to several orders of magnitude greater than that of the rocks that bound the system above and below. Thus, the definition of the Floridan aquifer system is partly lithologic and partly hydraulic. The system is sometimes referred to as the principal artesian aquifer in South Carolina, Georgia, and Alabama. The rocks that characterize the main body of the Floridan aquifer system are mostly platform carbonates.

The Floridan aquifer system includes the platform carbonates, as well as the updip equivalent clastics that are in hydrogeologic communication with the carbonates. The updip clastic facies equivalents of the Floridan carbonate rocks are not considered to be part of the Floridan aquifer system. However, they are hydraulically connected with it and are part of its regional flow system. Thus, the updip clastic facies equivalent of the Floridan aquifer system and the carbonate phase of the Floridan aquifer system are treated as a single hydrologic unit (the Floridan aquifer system). The updip clastic facies equivalents represent the recharge areas for the downdip Floridan. The downdip carbonate phase of the Floridan aquifer system is used extensively in the southeastern part of the South Carolina Coastal Plain as an aquifer.

The transition zone between the carbonate rocks of the Floridan and the updip clastic facies equivalents of the system is the approximate northern extent of the thick carbonate platform that extended from the Florida peninsula through the coastal area of Georgia to southwestern South Carolina during early Tertiary time. The transition zone extended toward the north to a line approximated by the updip limit of the Santee Limestone platform carbonate beds. At SRS, which lies mostly north of the line established for the updip limit of the carbonate phase of the Floridan aquifer system, there are thin beds and lenses of limestone that may be either connected to the main limestone body or isolated from it, owing in part to depositional isolation or to postdepositional erosion or diagenetic alteration. They are considered part of the updip clastic phase of the Floridan.

1.3.4.3.3.2 Carbonate Phase of the Floridan Aquifer System

The carbonate phase of the Floridan aquifer system that develops in the southernmost fringe of SRS, just south of well C-10, is divided into the Upper and the Lower Floridan aquifer units, separated by the "middle confining unit." The hydraulic characteristics of the carbonate phase of the Floridan aquifer system vary considerably in the South Carolina-Georgia region. This variation results from several different processes, the most important being the dissolution of calcium carbonate by groundwater. The variability in the amount of dissolution is strongly influenced by the chemical composition of the water and the local differences in geology and lithology that affect the rate of groundwater movement.

See Table 1.3.4-11 for the hydraulic parameters data of the Floridan aquifer system.

1.3.4.3.3.3 Clastic Phase of the Floridan Aquifer System

The updip clastic phase of the Floridan aquifer system dominates in the SRS region and consists of a thick sequence of Paleocene to late Eocene sand with minor amounts of gravel and clay and a few limestone beds. At the southern fringe of SRS, the clastic sediments of the aquifer system grade directly into the platform limestone that forms the carbonate phase of the Floridan. The lithologic transition between the clastic phase and the carbonate phase of the aquifer system does not represent a hydrologic boundary, and the two lithofacies are in direct hydrogeologic communication. The Floridan aquifer system overlies the Meyers Branch confining system throughout the lower two-thirds of the study area. Toward the north, the confining beds of the Meyers Branch confining system thin and become intermittent, and the entire Floridan aquifer system coalesces with the Dublin-Midville aquifer system to form the Floridan-Midville aquifer system.

In the central portion of SRS, clay to sandy clay beds in the Warley Hill Formation support a substantial head difference between overlying and underlying units. These fine-grained sediments constitute the Gordon confining unit, which divides the system into two aquifers: the Gordon aquifer and the overlying Upper Three Runs aquifer. The former of the two is between the lower surface of the Gordon confining unit and the upper surface of the Crouch Branch confining unit. Updip, the Warley Hill sediments do not support a substantial head difference; thus, there is only one aquifer (the Steed Pond aquifer).

The sedimentary sequence that corresponds to the updip clastic phase of the Floridan aquifer system is penetrated in the P-27 reference well near the center of SRS. The system at P-27 is 216 ft (65.8 m) thick; the base is at 48 ft (14.6 m) above msl, and the top occurs at the water table, which is at 264 ft (80.5 m) above msl, or 10 ft (3.1 m) below land surface. Groundwater in F Area varies in elevation from 190 to 220 ft (58 to 67 m) above msl and is found at a depth of over 50 ft (15 m) below existing ground level. The system includes 22 ft (6.7 m) of clay in five beds, and the remainder consists of sand and clayey sand beds. The stratigraphic units that constitute the clastic phase of the Floridan aquifer system include the Fourmile Formation and the locally sandy parts of the Snapp Formation of the Black Mingo Group, the Orangeburg and Barnwell Groups, and the overlying Miocene/Oligocene "Upland unit".

Recharge of the Floridan aquifer system occurs generally in the northwestern part of the study area, where rainfall percolates into the outcrop of the Gordon and Upper Three Runs aquifers. The Savannah River has the greatest area wide influence on water levels, followed by the South Fork Edisto River and, to a much lesser degree, the Salkehatchie River. In the updip portion of the study area, Upper Three Runs controls the direction of groundwater movement. Here, the Gordon confining unit has been breached by the stream, creating a groundwater sink that induces flow out of the Gordon toward the stream. Using an average transmissivity value of 300 ft^2/day (28 m^2/day) and an average hydraulic gradient of 25 ft/miles (4.7 m/km) near Upper Three Runs, an estimated 112,000 gal/day (423,920 L/day) is being discharged through each 1 mile (1.6-km) strip of the aquifer along the creek, for a total of 1.4 million gal/day (5.3 million L/day).

The transmissivity of the clastic and carbonate phases of the Floridan aquifer system is lowest near their updip limits because of the reduced aquifer thickness there. Transmissivity increases rapidly from the northwest to the southeast along the Savannah River through the clastic facies and across the limestone facies change of the Floridan aquifer system.

Upper Three Runs Aquifer

The Upper Three Runs aquifer occurs between the water table and the Gordon confining unit and includes the strata above the Warley Hill Formation (in updip areas) and the Blue Bluff Member of the Santee Limestone (in downdip areas). It includes the sandy and sometimes calcareous sediments of the Tinker/Santee Formation and the heterogeneous sediments in the overlying Barnwell Group. The Upper Three Runs aquifer is the updip clastic facies equivalent of the Upper Floridan aquifer in the carbonate phase of the Floridan aquifer system.

The Upper Three Runs aquifer is defined by the hydrogeologic properties of the sediments penetrated in well P-27 located near Upper Three Runs in the center of SRS. Here, the aquifer is 132 ft (40.2 m) thick and consists mainly of quartz sand and clayey sand of the Tinker/Santee Formation; sand with interbedded tan to gray clay of the Dry Branch Formation; and sand, pebbly sand, and minor clay beds of the Tobacco Road Formation. Calcareous sand, clay, and limestone, although not observed in the P-27 well, are present in the Tinker/Santee Formation throughout the General Separations Area near well P-27.

Downdip, at the C-10 reference well, the Upper Three Runs aquifer is 380 ft (116 m) thick and consists of clayey sand and sand of the upper Cooper Group; sandy, shelly limestone, and calcareous sand of the lower Cooper Group/Barnwell Group; and sandy, shelly, limestone and micritic limestone of the Santee Limestone.

Water-level data are sparse for the Upper Three Runs aquifer except within SRS. The hydraulic-head distribution of the aquifer is controlled by the location and depth of incisement of creeks that dissect the area. The incisement of these streams and their tributaries has divided the interstream areas of the water table aquifer into “groundwater islands.” Each “groundwater island” behaves as an independent hydrogeologic subset of the water table aquifer with unique recharge and discharge areas. The stream acts as the groundwater discharge boundary for the interstream area. The head distribution pattern in these groundwater islands tends to follow topography and is characterized by higher heads in the interstream area with gradually declining heads toward the bounding streams. Groundwater divides are present near the center of the interstream areas. Water table elevations reach a maximum of 250 ft (76 m) above msl in the northwest corner of the study area and decline to approximately 100 ft (30 m) above msl near the Savannah River.

Porosity and permeability of the Upper Three Runs aquifer are variable across the study area. In the northern and central regions, the aquifer yields only small quantities of water, owing to the presence of interstitial silt and clay and poorly sorted sediments that combine to significantly reduce permeability. Local lenses of relatively clean, permeable sand, however, may yield sufficient quantities for domestic use. Such high-permeability zones have been observed in the General Separations Area (defined as F, E, S, H, and Z Areas) near the center of the study area and may locally influence the movement of groundwater.

Porosity and permeability were determined for Upper Three Runs aquifer sand samples containing less than 25% mud. Porosity averages 35.3%; the distribution is approximately normal, but skewed slightly toward higher values. Geometric mean permeability is 31.5 Darcies (23 m/day [76.7 ft/day]) with about 60% of the values between 16 and 64 Darcies (12 and 48 m/day [39 and 156 ft/day]).

Pumping-test and slug-test results in the General Separations Area indicate that hydraulic conductivity is variable, ranging from less than 1.0 ft/day (0.3 m/day) to 32.8 ft/day (10 m/day).

Hydraulic conductivity values derived from long-duration, multiple-well aquifer tests are in the range of 10 ft/day (3 m/day), which may be a more reliable estimation of average hydraulic conductivity. At the south end of the study area, near well C-10, sediments in the aquifer become increasingly calcareous, the amount of silt and clay tends to decline, and permeability and yields generally increase. Here, hydraulic conductivity values are in the 59 ft/day (18 m/day) range.

The majority of hydrogeologic data available on the Upper Three Runs aquifer is from wells in the General Separations Area at SRS. Thus, the discussion that follows is largely focused on that area. The Upper Three Runs aquifer is divided into two aquifer zones divided by the “tan clay” confining zone. In the General Separations Area, the “upper” aquifer zone consists of the saturated strata in the upper parts of the Dry Branch Formation and the Tobacco Road Formation that lie between the water table and the “tan clay” confining zone. The aquifer zone has a general downward hydraulic potential into the underlying aquifer unit. The confining beds of the “tan clay” located near the base of the Dry Branch Formation impede the vertical movement of water and often support a local hydraulic head difference. The “lower aquifer” zone of the Upper Three Runs aquifer occurs between the “tan clay” confining zone and the Gordon confining unit and consists of sand, clayey sand, and calcareous sand of the Tinker/Santee Formation and sand and clayey sand of the lower part of the Dry Branch Formation.

Slug tests, minipermeameter tests, pumping tests, and sieve analyses have been used to calculate hydraulic conductivity values for the “upper” aquifer zone near the General Separations Area. Hydraulic conductivity values derived from 103 slug tests range from a high of 45.4 ft/day (14 m/day) to a low of 0.07 ft/day (0.02 m/day) and average (arithmetic mean) 5.1 ft/day (1.5 m/day).

As stated previously, the “tan clay” confining zone at the General Separations Area separates the “upper” aquifer zone from the “lower” aquifer zone in the Upper Three Runs aquifer. This zone is a leaky confining zone. Total thickness of the “tan clay” confining zone, based on measurements at 46 wells distributed throughout the General Separations Area, ranges from 0 to 32.8 ft (0 to 10 m) and averages 11 ft (3.4 m). The sandy clay to clay beds range from 0 to 18 ft (0 to 5.5 m) in thickness and average 7 ft (2.1 m). The clayey sand beds range from 0 to 12 ft (0 to 3.7 m) and average 3 ft (1 m).

Laboratory analyses, including horizontal and vertical hydraulic conductivity, were run on 28 selected clayey sand samples and 55 sandy, often silty clay, and clay samples from the various confining units and “low-permeability” beds in the aquifers. The results are presented in Table 1.3.4-11. The generally accepted value of effective porosity used to determine vertical-flow velocities is 5% for the clay to sandy clay beds and 12% for the clayey sand beds.

Recharge to the Upper Three Runs aquifer occurs at the water table by infiltration downward from the land surface. In the “upper” aquifer zone, part of this groundwater moves laterally toward the bounding streams while part moves vertically downward. The generally low vertical hydraulic conductivities of the “upper” aquifer zone and the intermittent occurrence of the “tan clay” confining zone retard the downward flow of water, producing vertical hydraulic-head gradients in the “upper” aquifer zone and across the “tan clay” confining zone.

Downward hydraulic-head differences in the “upper” aquifer zone vary from 4.5 to 5.4 ft (1.4 to 1.64 m), and differences across the “tan clay” are as much as 15.8 ft (6.5 m) in H Area. At other locations in the General Separations Area, the head difference across the “tan clay” confining zone is only 0.1 to 3.2 ft (0.3 to 1 m), essentially what might be expected due simply to low vertical flow in a clayey sand aquifer. Therefore, the ability of the “tan clay” confining zone to impede water flow varies greatly over the General Separations Area.

Groundwater leaking downward across the “tan clay” confining zone recharges the “lower” aquifer zone of the Upper Three Runs aquifer. Most of this water moves laterally toward the bounding streams; the remainder flows vertically downward across the Gordon confining unit into the Gordon aquifer. Groundwater moving toward Upper Three Runs leaks through the Gordon confining unit or enters small streams. Vertical hydraulic-head differences in the “lower” aquifer zone range from 1.5 to 3.2 ft (0.5 to 1 m) in H Area and indicate some vertical resistance to flow.

Gordon Confining Unit

Clayey sand and clay of the Warley Hill Formation and clayey, micritic limestone of the Blue Bluff Member of the Santee Limestone constitute the Gordon confining unit. The Gordon confining unit separates the Gordon aquifer from the overlying Upper Three Runs aquifer. The unit has been informally termed the “green clay” in previous SRS reports.

In the study area, the thickness of the Gordon confining unit ranges from about 5 to 85 ft (1.5 to 26 m). The unit thickens to the southeast. From Upper Three Runs to the vicinity of L Lake and Par Pond, the confining unit generally consists of one or more thin clay beds, sandy mud beds, and sandy clay beds intercalated with subordinate layers and lenses of quartz sand, gravelly sand, gravelly muddy sand, and calcareous mud. Southward from L Lake and Par Pond, however, the unit undergoes a stratigraphic facies change to clayey micritic limestone and limey clay typical of the Blue Bluff Member. The fine-grained carbonates and carbonate-rich muds constitute the farthest updip extent of the "middle confining unit" of the Floridan aquifer system (the hydrostratigraphic equivalent of the Gordon confining unit), which dominates in coastal areas of South Carolina and Georgia.

North of the updip limit of the Gordon confining unit, the fine-grained clastics of the Warley Hill Formation are thin, intermittent, and no longer effective in regionally separating groundwater flow. Here, the Steed Pond aquifer is defined. Although thin and intermittent, the clay, sandy clay, and clayey sand beds of the Warley Hill Formation can be significant at the site-specific level and often divide the Steed Pond aquifer into aquifer zones.

The values for hydraulic conductivity obtained from the Gordon confining unit are comparable to the average vertical hydraulic conductivity values of clayey sand (8.9×10^{-3} ft/day [2.71×10^{-3} m/day]) and sandy clay to clay (1.7×10^{-4} ft/day [5.09×10^{-5} m/day]) calculated for 83 samples analyzed in the Tertiary/Cretaceous section. Selected parameters determined for the unit are listed in Table 1.3.4-11 .

Gordon Aquifer

The Gordon aquifer consists of the saturated strata that occur between the Gordon confining unit and the Crouch Branch confining unit in both the Floridan-Midville aquifer system and the Meyers Branch confining system. The aquifer is semiconfined, with a downward potential from the overlying Upper Three Runs aquifer observed in interfluvial areas, and an upward potential observed along the tributaries of the Savannah River where the Upper Three Runs aquifer is incised. The thickness of the Gordon aquifer ranges from 38 ft (12 m) at well P-4A to 185 ft (56 m) at well C-6 and generally thickens to the east and southeast. Thickness variations in the confining lithologies near the Pen Branch Fault suggest depositional effects owing to movements on the fault in early Eocene time. The Gordon aquifer is partially eroded near the Savannah River and Upper Three Runs. The regional potentiometric map of the Gordon aquifer indicates that major deviations in the flow direction are present where the aquifer is deeply incised by streams that drain water from the aquifers.

The Gordon aquifer is characterized by the hydraulic properties of the sediments penetrated in reference well P-27 located near the center of SRS. The unit is 75.5 ft (23 m) thick in well P-27 and occurs from 125 to 48 ft (38 to 15 m) above msl. The aquifer consists of the sandy parts of the Snapp Formation and the overlying Fourmile and Congaree Formations. Clay beds and stringers are present in the aquifer, but they are too thin and discontinuous to be more than local confining beds. The aquifer in wells P-21 and P-22 includes a clay bed that separates the Congaree and Fourmile Formations. The clay bed appears sufficiently thick and continuous to justify splitting the Gordon aquifer into zones in the southeastern quadrant of SRS.

Downdip, the quartz sand of the Gordon aquifer grades into quartz-rich, fossiliferous lime grainstone, packstone, and wackestone, which contain considerably more glauconite than the updip equivalents. Porosity of the limestone as measured in thin-section ranges from 5% to 30% and is mostly moldic and vuggy.

South of SRS, near well ALL-324, the Gordon aquifer consists of interbedded glauconitic sand and shale, grading to sandy limestone. Farther south, beyond well C-10, the aquifer grades into platform limestone of the Lower Floridan aquifer of the carbonate phase of the Floridan aquifer system.

The Gordon aquifer is recharged directly by precipitation in the outcrop area and in interstream drainage divides in and near the outcrop area. South of the outcrop area, the Gordon aquifer is recharged by leakage from overlying and underlying aquifers. Because streams such as the Savannah River and Upper Three Runs cut through the aquifers of the Floridan aquifer system, they represent no-flow boundaries. As such, water availability or flow patterns on one side of the boundary (stream) will not change appreciably due to water on the other side. In the central part of SRS, where the Gordon confining unit is breached by faulting, recharge to the Gordon aquifer is locally increased.

Most of the Gordon aquifer is under confined conditions, except along the fringes of Upper Three Runs (i.e., near the updip limit of the Gordon confining unit) and the Savannah River. The potentiometric-surface map of the aquifer shows that the natural discharge areas of the Gordon aquifer at SRS are the swamps and marshes along Upper Three Runs and the Savannah River. These streams dissect the Floridan aquifer system, resulting in unconfined conditions in the stream valleys and probably in semiconfined (leaky) conditions near the valley walls. Reduced head near Upper Three Runs induces upward flow from the Crouch Branch aquifer and develops the "head reversal" that is an important aspect of the SRS hydrogeological system. The northeast-southwest oriented hydraulic gradient across SRS is consistent and averages 4.8 ft/miles (0.9 m/km). The northeastward deflection of the contours along Upper Three Runs indicates incisement of the sediments that constitute the aquifer by the creek.

Hydraulic characteristics of the Gordon aquifer are less variable than those noted in the Upper Three Runs aquifer. See Table 1.3.4-12 for the selected parameters. Hydraulic conductivity decreases downdip near well C-10 owing to poor sorting, finer grain size, and an increase in clay content.

1.3.4.3.3.4 Floridan-Dublin Aquifer System

Over most of the study area, the Meyers Branch confining system extends north of the Allendale confining system, hydraulically isolating the Floridan from the underlying Dublin and Dublin-Midville systems. However, in a small region in the eastern part of the study area near well C-5, clay beds of the Meyers Branch confining system thin dramatically, leakance values increase, and the Floridan and Dublin aquifer systems are in overall hydraulic communication. In this region, the Floridan and Dublin aquifer systems coalesce to form the Floridan-Dublin aquifer system. Thick, continuous clay beds in the underlying Allendale confining system continue to hydrogeologically isolate the Midville and Floridan-Dublin aquifer systems.

The Floridan-Dublin aquifer system is divided into three aquifers in the study area. In descending order, these include the Upper Three Runs, Gordon, and Crouch Branch aquifers separated by the Gordon and Crouch Branch confining units. The Upper Three Runs and Gordon aquifers coalesce updip forming the Steed Pond aquifer. The Crouch Branch aquifer is continuous across the entire study area.

The Floridan-Dublin aquifer system is defined by the hydrogeologic properties of sediments penetrated in well C-5 located north of the town of Barnwell. Here, the system is 560 ft (171 m) thick and includes sediments from the water table to the top of the McQueen Branch confining unit. The Upper Three Runs aquifer is 144 ft (44 m) thick and consists entirely of sand. The Gordon aquifer is 108 ft (33 m) thick and consists of two sand beds that total 105 ft (32 m). The Crouch Branch aquifer is 244 ft (74 m) thick and consists of two sand beds that total 230 ft (70 m).

1.3.4.3.3.5 Floridan-Midville Aquifer System

Northwest of Upper Three Runs, the permeable beds that correspond to the Floridan and Dublin-Midville aquifer systems are often in hydrologic communication owing to the thin and laterally discontinuous character of the intervening clay and silty clay beds, to faulting that breaches the confining beds, and to erosion by the local stream systems that dissect the interval. Here, the Floridan and Dublin-Midville aquifer systems coalesce to form the Floridan-Midville aquifer system.

The Floridan-Midville aquifer system is divided into three aquifers: in descending order, the Steed Pond aquifer, the Crouch Branch aquifer, and the McQueen Branch aquifer, separated by the Crouch Branch and McQueen Branch confining units. Both the Crouch Branch and the McQueen Branch aquifers extend northwestward from the southern part of SRS. The Steed Pond aquifer is the updip hydrostratigraphic equivalent of the Gordon and Upper Three Runs aquifers. At the northern fringe of the study area, the Steed Pond and underlying Crouch Branch aquifers coalesce and a single, yet unnamed, aquifer unit is present.

The Floridan-Midville aquifer system is defined by the hydrogeologic properties of the sediments penetrated in the GCB-1 type well located in the A/M Area in the northwest corner of SRS. Near GCB-1, the system is 557 ft (170 m) thick and includes sediments from the water table to the top of the Appleton confining system. The Steed Pond aquifer is 97 ft (30 m) thick at the GCB-1 well and consists of 86 ft (26 m) of sand in four beds. The Crouch Branch aquifer is 167 ft (51 m) thick and consists of 139 ft (42 m) of sand in four beds. It is overlain by the Crouch Branch confining unit, which is 81 ft (25 m) thick and consists of 31 ft (9 m) of clay in four beds. The McQueen Branch aquifer is 169 ft (52 m) thick and consists of 147 ft (45 m) of sand in three beds. The McQueen Branch confining unit is 43 ft (13 m) thick and consists of 28 ft (9 m) of clay in two beds.

Steed Pond Aquifer

North of Upper Three Runs where the Floridan-Midville aquifer system is defined, the permeable beds that correspond to the Gordon and Upper Three Runs aquifers of the Floridan aquifer system are only locally separated, owing to the thin and intermittent character of the intervening clay beds of the Gordon confining unit (Warley Hill Formation) and to erosion by the local stream systems that dissect the interval. Here, the aquifers coalesce to form the Steed Pond aquifer of the Floridan-Midville aquifer system.

The Steed Pond aquifer is defined by hydrogeologic characteristics of sediments penetrated in well MSB-42 located in A/M Area in the northwest corner of SRS. The aquifer is 97 ft (29.6 m) thick. Permeable beds consist mainly of subangular, coarse- and medium-grained, slightly gravelly, submature quartz sand and clayey sand. Locally, the Steed Pond aquifer can be divided into zones. In A/M Area three zones are delineated, the “Lost Lake” zone, and the overlying “M Area” aquifer zones, separated by clay and clayey sand beds of the “green clay” confining zone.

In A/M Area, water enters the subsurface through precipitation, and recharge into the “M-Area” aquifer zone occurs at the water table by infiltration downward from the land surface. A groundwater divide exists in the A/M Area in which lateral groundwater flow is to the southeast towards Tims Branch and southwest towards Upper Three Runs and the Savannah River floodplain. Groundwater also migrates downward and leaks through the “green clay” confining zone into the “Lost Lake” aquifer. The “green clay” confining zone that underlies the “M-Area” aquifer zone is correlative with the Gordon confining unit south of Upper Three Runs.

1.3.4.3.3.6 Meyers Branch Confining System

The Meyers Branch confining system separates the Floridan aquifer system from the underlying Dublin and Dublin-Midville aquifer systems. North of the updip limit of the confining system, the Floridan and Dublin-Midville aquifer systems are in hydraulic communication and the aquifer systems coalesce to form the Floridan-Midville aquifer system.

Sediments of the Meyers Branch confining system correspond to clay and interbedded sand of the uppermost Steel Creek Formation, and to clay and laminated shale of the Sawdust Landing/Lang Syne and Snapp Formations. In the northwestern part of the study area, the sediments that form the Meyers Branch confining system are better sorted and less silty, with thinner clay interbeds. This is the updip limit of the Meyers Branch confining system.

Crouch Branch Confining Unit

In the SRS area, the Meyers Branch confining system consists of a single hydrostratigraphic unit, the Crouch Branch confining unit, which includes several thick and relatively continuous (over several kilometers) clay beds. The Crouch Branch confining unit extends north of the updip limit of the Meyers Branch confining system where the clay thins and is locally absent and faulting observed in the region locally breaches the unit. Here, the Crouch Branch confining unit separates the Steed Pond aquifer from the underlying Crouch Branch aquifer. Downdip, generally south of the study area, the Meyers Branch confining system could be further subdivided into aquifers and confining units if this should prove useful for hydrogeologic characterization.

As indicated earlier, a hydraulic-head difference persists across the Crouch Branch confining unit near SRS. Owing to deep incisement by the Savannah River and Upper Three Runs into the sediments of the overlying Gordon aquifer, an upward hydraulic gradient (vertical-head reversal) persists across the Crouch Branch confining unit over a large area adjacent to the Savannah River floodplain and the Upper Three Runs drainage system. This “head reversal” is an important aspect of the groundwater flow system near SRS and provides a natural means of protection from contamination of the lower aquifers.

The total thickness of the Crouch Branch confining unit where it constitutes the Meyers Branch confining system ranges from 57 to 184 ft (17.4 to 56.1 m). Updip, the thickness of the Crouch Branch confining unit ranges from 3.3 to 104 ft (< 1 to 31.7 m). The confining unit dips approximately 16 ft/miles (5 m/km) to the southeast. The confining unit is comprised of the "upper" and "lower" confining zones, which are separated by a "middle sand" zone.

In general, the Crouch Branch confining unit contains two to seven clay to sandy clay beds separated by clayey sand and sand beds that are relatively continuous over distances of several kilometers. The clay beds in the confining unit are anomalously thin and fewer in number along a line that parallels the southwest-northeast trend of the Pen Branch and Steel Creek Faults and the northeast-southwest trending Crackerneck Fault. The reduced clay content near the faults suggests shoaling due to uplift along the faults during deposition of the Paleocene Black Mingo Group sediments.

In A/M Area, the Crouch Branch confining unit can often be divided into three zones: an "upper clay" confining zone is separated from the underlying "lower clay" confining zone by the "middle sand" aquifer zone. The "middle sand" aquifer zone consists of very poorly sorted sand and clayey silt of the Lang Syne/Sawdust Landing Formations. The "middle sand" aquifer zone has a flow direction that is predominantly south/southwest toward Upper Three Runs.

In places, especially in the northern part of A/M Area, the "upper clay" confining zone is very thin or absent. Here, only the "lower clay" confining zone is capable of acting as a confining unit and the "middle sand" zone is considered part of the Steed Pond aquifer. Similarly, when the clay beds of the "lower clay" confining zone are very thin or absent, the "middle sand" aquifer zone is considered part of the Crouch Branch aquifer. This is the case in the far northeastern part of the study area.

The "lower clay" confining zone has been referred to as the lower Ellenton clay, the Ellenton clay, the Peedee clay, and the Ellenton/Peedee clay in previous SRS reports. It consists of the massive clay bed that caps the Steel Creek Formation. The zone is variable in total thickness and, based on 31 wells that penetrate the unit, ranges from 5 to 62 ft (1.5 to 19 m) and averages 24 ft (7.3 m) thick.

1.3.4.3.3.7 Dublin Aquifer System

The Dublin aquifer system is present in the southeastern half of SRS and consists of one aquifer, the Crouch Branch aquifer. It is underlain by the Allendale confining system and overlain by the Meyers Branch confining system. The updip limit of the Dublin aquifer system in the study area corresponds to the updip limit of the Allendale confining system. North of this line, the Dublin-Midville aquifer system is defined.

The thickness of the Dublin aquifer system generally increases toward the south and ranges from approximately 175 to 290 ft (53 to 88 m). The top of the unit dips 3.79 m/km (20 ft/mi) to the southeast. The unit thins to the east toward the Salkehatchie River and to the west toward Georgia. Near the updip limit of the system, thicknesses are variable and probably reflect the effects of movement along the Pen Branch Fault during deposition of the middle Black Creek clay.The Dublin aquifer system was defined and named for sediments penetrated by well 21-U4 drilled near the town of Dublin in Laurens County, Georgia. The upper part of the Dublin aquifer system consists of fine to coarse sand and limestone of the lower Huber-Ellenton unit. Comparable stratigraphic units serve as confining beds in the SRS area and are considered part of the overlying Meyers Branch confining system. To the east near the Savannah River, clay in the upper part of the lower Huber-Ellenton unit forms a confining unit that separates an upper aquifer of Paleocene age from a lower aquifer of Late Cretaceous age. The upper aquifer is the Gordon aquifer, and their confining unit constitutes the Meyers Branch confining system of the SRS region. The lower part of the Dublin aquifer system consists of alternating layers of clayey sand and clay of the Peedee-Providence unit.

Sediments typical of the Dublin aquifer system are penetrated in the reference well P-22. The system consists of the well-sorted sand and clayey sand of the Black Creek Formation and the moderately sorted sand and interbedded sand and clay of the Steel Creek Formation. The aquifer is overlain by the clay beds that cap the Steel Creek Formation. These clay beds constitute the base of the Meyers Branch confining system.

The Dublin aquifer system is 213 ft (65 m) thick in well P-22; the top is at an elevation of -223 ft (-68 m) above msl and the bottom at -436 ft (-133 m) above msl. The Dublin aquifer system includes five clay beds in this well.

In the southern part of the study area and farther south and east, the Dublin aquifer system shows much lower values for hydraulic conductivity and transmissivity, probably due to the increase of fine-grained sediments toward the coast.

1.3.4.3.3.8 Dublin-Midville Aquifer System

The Dublin-Midville aquifer system underlies the central part of SRS. The system includes the sediments in the Cretaceous Lumbee Group from the Middendorf Formation up to the sand beds in the lower part of the Steel Creek Formation. The system is overlain by the Meyers Branch confining system and underlain by the indurated clayey silty sand and silty clay of the Appleton confining system. The updip limit of the system is established at the updip pinchout of the overlying Meyers Branch confining system. The downdip limit of the Dublin-Midville aquifer system is where the Allendale becomes an effective confining system. The Dublin-Midville and the updip Floridan-Midville aquifer systems are referred to as the Tuscaloosa aquifer.

The thickness of the Dublin-Midville aquifer system ranges from approximately 250 to 550 ft (76 to 168 m). The dip of the upper surface of the system is about 20 ft/miles (3.8 m/km) to the southeast. Near the downdip limit of the system, thicknesses are variable and probably reflect the effects of movement along the Pen Branch Fault. Shoaling along the fault trace resulted in a relative increase in the thickness of the aquifers at the expense of the intervening confining unit.

The Dublin-Midville aquifer system includes two aquifers: the McQueen Branch aquifer and the Crouch Branch aquifer, separated by the McQueen Branch confining unit. The two aquifers can be traced northward, where they continue to be an integral part of the Floridan-Midville aquifer system and southward where they constitute the aquifers of the Midville and Dublin aquifer systems, respectively.

The Dublin-Midville aquifer system is defined at the type well P-27. Here, the system is 505 ft (153 m) thick and occurs from -82 ft (-25 m) above msl to -587 ft (-179 m) above msl. It consists of medium- to very coarse-grained, silty sand of the Middendorf Formation and clayey, fine to medium sand and silty clay beds of the Black Creek Formation. The system includes a thick clay bed, occurring from -329 ft (-100 m) above msl to -384 ft (-117 m) above msl, which constitutes the McQueen Branch confining unit.

A regional potentiometric surface map prepared for the Tuscaloosa aquifer indicates that the Savannah River has breached the Cretaceous sediments and is a regional discharge area for the Floridan-Midville aquifer system, the Dublin-Midville aquifer system, and the updip part of both the Dublin and Midville aquifer systems. The Savannah River, therefore, represents a no-flow boundary preventing the groundwater in these aquifer systems from flowing southward into Georgia.

Crouch Branch Aquifer

The Crouch Branch aquifer constitutes the Dublin aquifer system in the southern part of the study area. Farther south, the Dublin aquifer system can be subdivided into several aquifers and confining units. In the central part of the study area, the Crouch Branch aquifer is the uppermost of the two aquifers that constitute the Dublin-Midville aquifer system. Farther north in the northwestern part of SRS and north of the site, the Crouch Branch aquifer is the middle aquifer of the three aquifers that constitute the Floridan-Midville aquifer system.

The Crouch Branch aquifer is overlain by the Crouch Branch confining unit and is underlain by the McQueen Branch confining unit. It persists throughout the northern part of the study area, but near the updip limit of the Coastal Plain sedimentary clastic wedge, the Crouch Branch confining unit ceases to be effective and the Crouch Branch aquifer coalesces with the Steed Pond aquifer.

The Crouch Branch aquifer ranges in thickness from about 100 to 350 ft (30 to 107 m). Thickness of the unit is variable near the updip limit of the Dublin aquifer system where sedimentation was affected by movement along the Pen Branch Fault. The reduced clay content in this vicinity suggests shoaling due to uplift along the fault during Late Cretaceous and Paleocene time, resulting in the deposition of increased quantities of shallow-water, coarse-grained clastics along the crest of the fault trace. The sandy beds act hydrogeologically as part of the Crouch Branch aquifer, resulting in fewer and thinner, less persistent clay beds in the overlying and underlying confining units.

The Crouch Branch aquifer thins dramatically in the eastern part of the study area at the same general location where the underlying McQueen Branch confining unit and the overlying Crouch Branch confining unit thicken at the expense of Crouch Branch sand. Clay beds in the Crouch Branch aquifer generally thicken in the same area and constitute as much as 33% of the unit at well C-6.

Sediments of the Crouch Branch aquifer are chiefly sand, muddy sand, and slightly gravelly sand intercalated with thin, discontinuous layers of sandy clay and sandy mud. Hydraulic conductivity of the Crouch Branch aquifer ranges from 28 to 228 ft/day (8.5 to 69 m/day).

Comparatively high hydraulic conductivity occurs in a northeast-southwest trending region connecting D Area, Central Shops, and R Area and defines a “high permeability” zone in the aquifer. Here, hydraulic conductivities range from 117 to 227 ft/day (36 to 69 m/day). The “high permeability” zone parallels the trace of the Pen Branch Fault and reflects changing depositional environments in response to movement along the fault as described above. South of the trace of the Pen Branch Fault, hydraulic conductivity for the aquifer reflects the return to a deeper water shelf/deltaic depositional regime.

1.3.4.3.3.9 Allendale Confining System

The Allendale confining system is present in the southeastern half of the study area and separates the Midville aquifer system from the overlying Dublin aquifer system. In the study area, the Allendale confining system consists of a single unit, the McQueen Branch confining unit. The confining system is correlative with the unnamed confining unit that separates the Middendorf and Black Creek aquifers and with the Black Creek-Cusseta confining unit. The system dips approximately 27 ft/miles (6.7 m/km) to the southeast and thickens uniformly from about 50 ft (15.2 m) at the updip limit to about 200 ft (61 m) near the eastern boundary of the study area. The rate of thickening is greater in the east than in the west. The updip limit of this confining system is established where pronounced thinning occurs parallel to the Pen Branch Fault.

Sediments of the Allendale confining system are fine grained and consist of clayey, silty sand, clay, and silty clay and micritic clay beds that constitute the middle third of the Black Creek Formation. North of the updip limit of the confining system, where the McQueen Branch confining unit is part of the Dublin-Midville aquifer system, the section consists of coarser-grained, clayey, silty sand and clay beds.

McQueen Branch Confining Unit

The McQueen Branch confining unit is defined by the hydrogeologic properties of the sediments penetrated in well P-27. At its type-well location, the McQueen Branch confining unit is 55 ft (17 m) thick and is present from -329 to -384 ft (-100 to -117 m) above msl. Total clay thickness is 45 ft (14 m) in three beds, which is 82% of the total thickness of the unit, with a leakance coefficient of 1.03×10^{-5} ft/day (3.14×10^{-6} m/day). The confining unit in well P-27 consists of the interbedded, silty, often sandy clay and sand beds that constitute the middle third of the Black Creek Formation.

The clay beds tend to be anomalously thin along a line that parallels the southwest-northeast trend of the Pen Branch Fault and the north-south trend of the Atta Fault. The reduced clay content in these areas suggests shoaling due to uplift along the faults during Upper Black Creek-Steel Creek time.

1.3.4.3.3.10 Midville Aquifer System

The Midville aquifer system is present in the southern half of the study area; it overlies the Appleton confining system and is succeeded by the Allendale confining system. In the study area, the Midville aquifer system consists of one aquifer, the McQueen Branch aquifer. South of well C-10, the system may warrant further subdivision into several aquifers and confining units.

Thickness of the unit ranges from 232 ft (71 m) at well P-21 to 339 ft (103 m) at well C-10. Variation in the thickness of the unit, as well as the updip limit of the system, results from variation in the thickness and persistence of clay beds in the overlying Allendale confining system. Near the Pen Branch Fault, contemporaneous movement on the fault may have resulted in shoaling in the depositional environment, which is manifested in a thickening of the sands associated with the Midville aquifer system. The upper surface of the aquifer system dips approximately 25 ft/miles (4.73 m/km) to the southeast across the study area.

The Midville aquifer system was defined and named for the hydrogeologic properties of the sediments penetrated in well 28-X1, near the town of Midville in Burke County, Georgia. Here, the upper part of the aquifer system consists of fine to medium sand of the lower part of the Black Creek-Cusseta unit. The Midville is comparable to the lower portion of the Chattahoochee River aquifer and correlative with the Middendorf aquifer.

McQueen Branch Aquifer

The McQueen Branch aquifer occurs beneath the entire study area. It thickens from the northwest to the southeast and ranges from 118 ft (36 m) at well AIK-858 to 339 ft (103 m) at well C-10 to the south. Locally, thicknesses are greater along the trace of the Pen Branch Fault because of the absence and/or thinning of clay beds that compose the overlying McQueen Branch confining unit. The upper surface of the McQueen Branch dips approximately 25 ft/miles (4.7 m/km) to the southeast.

The McQueen Branch aquifer is defined for the hydrogeologic properties of sediments penetrated by well P-27 near the center of the study area. Here, it is 203 ft (62 m) thick and occurs from -384 to -587 ft (-117 to -180 m) above msl. It contains 183 ft (56 m) of sand in four beds, (which is 90% of the total thickness of the unit). The aquifer consists of silty sand of the Middendorf Formation and clayey sand and silty clay of the lower one-third of the Black Creek Formation. Typically, a clay bed or several clay beds that cap the Middendorf Formation are present in the aquifer. These clay beds locally divide the aquifer into two aquifer zones.

Eight pumping tests of the McQueen Branch aquifer were made in F and H Areas, in the central part of the study area. Hydraulic conductivity values ranged from 53 to 210 ft/day (16 to 64 m/day) and averaged 18 ft/day (36 m/day). Three pumping tests of the aquifer were performed: two in F Area and one in L Area. Hydraulic conductivity ranged from 41 to 290 ft/day (13 to 88 m/day) in F Area and was 93 ft/day (28 m/day) in L Area.

1.3.4.3.3.11 Appleton Confining System

The Appleton confining system is the lowermost confining system of the Southeastern Coastal Plain hydrogeologic province and separates the province from the underlying Piedmont hydrogeologic province. It is equivalent to the Black Warrior River aquifer and to the basal unnamed confining unit. The confining system is essentially saprolite of the Paleozoic and Mesozoic basement rocks and indurated, silty and sandy clay beds, silty clayey sand and sand beds of the Cretaceous Cape Fear Formation. Thickness of saprolite ranges from 6 to 47 ft (2 to 14 m), reflecting the degree of weathering on the basement unconformity prior to deposition of the Cape Fear terrigenous clastics. Thickness of saprolite determined from the Deep Rock

Borings study (DRB, described in Section 1.3.5.1.6) ranges from 30 to 97 ft (9 to 30 m) and averages 40 ft (12 m) in wells DRB-1 to DRB-7. In the northern part of the study area, the Cape Fear Formation pinches out and the Appleton confining system consists solely of saprolite.

Some variability in thickness is noted along the trace of the Pen Branch Fault. It dips at about 31 ft/miles (5.9 m/km) to the southeast and thickens from 15 ft (4.6 m) in well C-2 near the north end of the study area to 72.2 ft (22 m) in well C-10 in the south. Sediments of the confining system do not crop out in the study area. Thinning of the Appleton confining system in well PBF-2 is probably a result of truncation of the section by the Pen Branch Fault.

The confining system consists of a single confining unit throughout the study area. Toward the coast, however, the Appleton confining system thickens considerably and includes several aquifers. The aquifers included in the confining system in the downdip region are poorly defined because few wells penetrate them. They are potentially water-producing, but the depth and generally poor quality of water in the aquifers probably precludes their utilization in the foreseeable future. The Appleton confining system includes no aquifer units or zones in the northern and central parts of the study area.

Fine- to coarse-grained sand beds, often very silty and clayey, occur in the upper part of the Cape Fear Formation in the southern part of the study area. The sand appears to be in communication with sand of the overlying McQueen Branch aquifer system and is included with that unit.

1.3.4.3.4 Hydrogeology of the Piedmont Province

The basement complex, designated the Piedmont hydrogeologic province in this report, consists of Paleozoic crystalline rocks, and consolidated to semiconsolidated Upper Triassic sedimentary rocks of the Dunbarton basin. These rocks that make up this basement complex have low permeability. The hydrogeology of the province was studied intensively at SRS to assess the safety and feasibility of storing radioactive waste in these rocks. The upper surface of the province dips approximately 36 ft/miles (11 m/km) to the southeast. Origins of the crystalline and sedimentary basement rocks are different, but their hydraulic properties are similar. The rocks are massive, dense, and practically impermeable except where fracture openings are encountered. Water quality in these units is also similar. Both contain water with high alkalinity and high levels of calcium, sodium, sulfate, and chloride. The low aquifer permeability and poor water quality in the Paleozoic and Triassic rocks render them undesirable for water supply in the study area.

1.3.4.4 Area Hydrogeological Characteristics – General Separations Areas

Field activities at the various site facilities are reported in the annual SRS Environmental Report (WSRC 1997b). In the past, the focus of F-Area facilities has been on chemical separations; changes in the site mission have impacted operations in the General Separations Area, including the construction and startup of the Tritium Facilities and various waste management facilities (Defense Waste Processing Facility, E Area Vaults, and the Consolidated Incineration Facility).

1.3.4.4.1 Water Usage

Water usage at the F-Area/E-Area facilities varies from year to year and is a function of increased or decreased site activities. Current data for pumping in F/E Area is reported in the SRS Annual Environmental Report (WSRC 1997b). Operation of production water wells has not caused subsidence of the F-Canyon foundation nor influenced potential contaminant flow paths in the post-Cretaceous aquifers.

1.3.4.4.2 Hydrogeologic Setting

The General Separations Area sits above a water table ridge, defined on the south by Fourmile Branch and on the north and west by Upper Three Runs. The ridge is dissected on the northern flank by Crouch Branch (between E Area and H Area) and McQueen Branch (east of Z Area). Thus, the facilities lie above minor groundwater divides; flow at the water table is generally away from the facilities and toward the nearest surface water. The majority of water that reaches the water table beneath the General Separations Area is discharged into either Upper Three Runs (or its tributaries) or Fourmile Branch.

In general, there is very limited downward migration of groundwater across the Meyers Branch confining system beneath the General Separations Area. Therefore, the hydrostratigraphic units linked to General Separations Area operations are the Upper Three Runs aquifer (the water table aquifer), the Gordon confining unit, and the Gordon aquifer. See Section 1.3.4.3.3 for information about the hydraulic properties and hydraulic gradients for these units. Hydraulic conductivity values for the Upper Three Runs, Gordon, and Steed Pond aquifers are presented in Table 1.3.4-12.

1.3.4.5 Groundwater Chemistry

1.3.4.5.1 Regional Groundwater Chemistry

SRS groundwater quality samples are collected quarterly, and the data, as well as interpreted results, are presented in the Annual SRS Environmental Report (WSRC 1997b). See Table 1.3.4-13 for a set of water analyses from sources within SRS and the vicinity. See Table 1.3.4-14 for a tabulation of the pumpages.

An investigation of the geochemistry of the water residing in the principal aquifer units at SRS was undertaken as part of the Baseline Hydrologic Investigation (Bledsoe, Aadland, and Sargent 1990). This study investigated the effects of the mineralogy of the aquifer materials, source of the water, and the effect of biological activity on the evolution and chemistry of the groundwater. Groundwater chemistry and geologic data utilized for this study were obtained from monitoring wells and core samples collected during drilling activities. The majority of the ensuing discussions were adapted directly from this report.

The primary source of groundwater at SRS is precipitation. As the water migrates away from the source or recharge area, it experiences a decrease of pH and an increase in total dissolved solids. In addition, the overall chemistry changes as it encounters different aquifer material. The primary recharge areas for the deeper aquifers in the SRS vicinity are located near the Fall Line or Coastal Plain onlap. From there, the groundwater migrates in a general southwesterly direction. The extent to which the local discharge and recharge areas impact the groundwater chemistry is dependent upon the depth of a particular aquifer system below ground surface and the overall aquifer material. Recharge for the water table aquifers is derived from local, recent precipitation at the site as evidenced by elevated amounts of short-lived isotopes, such as tritium, and the ionic composition of the groundwater.

Tritium levels in local precipitation are in excess of the normal background levels for the Northern Hemisphere. Washout from the atmosphere during periods of precipitation has elevated the concentration of rainfall tritium to where pre- and post-1954 rainfall-derived water can clearly be distinguished in groundwater. The year 1954 is significant in that it represents the beginning of SRS facility operations. The impact of rainfall-derived tritium on the groundwater is observed in groundwater resident at depths of less than 200 ft (61 m).

The ionic composition of the groundwater also clearly reflects a meteoric origin of the water. Chemical data from rainwater collected near SRS exhibit approximately the same ratio of sodium to chlorine as that in seawater, which is a principal source of atmospheric salts, but higher levels of sulfate and calcium. These latter constituents are commonly contributed to the atmosphere over landmasses by natural biological processes and industrial emissions.

1.3.4.5.1.1 Aquifer Materials

Groundwater principally resides in the pore spaces of the sandy aquifers. In these aquifers, quartz is the dominant mineral. Despite its abundance, its effect on overall water chemistry is negligible due to the low reactivity of this quartz (except in cases of extremely basic pH). The minerals that potentially impact the chemistry of the groundwater are less abundant. Minerals identified by x-ray diffraction and x-ray fluorescence data include feldspars and a host of phyllosilicates (i.e., clays and micas). Other non-silicate minerals such as pyrite, gypsum, barite, calcite, and hematite were also identified, but these are relatively sparse and have little impact on the overall groundwater chemistry. Clay minerals present include kaolinite, smectites and in minor amounts, illite.

1.3.4.5.1.2 Groundwater Chemistry (Hydrochemical Facies)

The evolution of groundwater in the Coastal Plain sediments can be defined from the source or recharge areas down the hydraulic gradient within the aquifer. Although groundwaters at SRS are very dilute, they show significant changes in the levels of dissolved oxygen, redox potential, dissolved trace constituents, and in the major cations and anions present. The variations in these major constituents are useful in delineating the chemical reactions, which occur during the chemical development of the groundwater.

On the northern edge of the site where there is a single aquifer system, the waters are of very low total dissolved solids (less than 20 mg/L). They contain high concentrations of dissolved oxygen, have pH lower than 6.0, and are classified as mixed water types (i.e., there are no predominant cations or anions in the water). The confining units that separate the aquifers are only of local extent and the hydraulic gradient is downward from the Tertiary formations into the underlying Cretaceous formations over much of this portion of the site. The Cretaceous aquifer receives recharge from Tertiary units where the confining units are thin or absent.

South of this region, where two or more aquifer systems are present, the waters become geochemically distinctive because of bio-geochemical and geochemical interaction with the water and the sediments and buried organic materials. Water samples in both of the aquifers are shown to have a predominance of calcium-bicarbonate. The presence of calcium-bicarbonate is most frequently attributed to the dissolution of $CaCO_3$. Several reaction mechanisms are known to exist for the dissolution reactions. The dissolution by weak carbonic acid

$$CaCO_3 + H_2CO_3 \dashrightarrow Ca^{2+} + 2HCO_3^-$$

produces two bicarbonate ions per calcium ion whereas the hydrolysis reaction produces a single bicarbonate plus a hydroxyl ion.

$$CaCO_3 + H_2O \dashrightarrow Ca^{2+} + HCO_3^- + OH^-$$

In either case, equal amounts of alkalinity are produced by the reaction so that the bicarbonate concentration calculated from alkalinity data in this study are not useful indicators to distinguish the reaction mechanisms. It is probable that both reactions contribute in the Tertiary aquifers. There have not been sufficient ^{13}C isotopic data obtained on these aquifer units or direct measurement of dissolved inorganic carbon to generalize at the present time.

The samples from monitoring wells screened in the Tertiary section at the P-19 well site cluster are anomalous in their water chemistry because they are low in total dissolved solids and show no evidence of having had opportunity to react with carbonates (low alkalinity and moderate pH). This is true of the P-19 wells screened in the Upper Three Runs aquifer and Gordon aquifer. In addition, limestones, marls, and clay units are conspicuously absent from the Tertiary section at this locality and, therefore, high vertical permeabilities are expected.

The Cretaceous or deeper aquifers (Midville, Dublin, and or Dublin-Midville) south of Upper Three Runs have a somewhat more complex chemistry. Examination of Piper diagrams for these units shows a marked evolution from sulfate-rich waters at low total dissolved solids toward bicarbonate-rich waters at higher total dissolved solids. The evolution toward calcium-rich waters is not as pronounced as in the Tertiary units. Alkalis (Na+K) are major contributors to the cation compositions, and the waters would be classified as mixed water types or Na+K-HCO_3 waters. The reaction pathways toward these compositions are complex and not well understood at present.

The calcium in these waters may be derived from several sources, including dissolution of gypsum from confining beds such as the Rhems (Ellenton) Formation, which is the downdip equivalent of the Lang Syne/Sawdust Landing Formation, the dissolution of calcite or calcium plagioclase, or displacement of calcium by potassium in cation exchange reactions. The alkalis in the Cretaceous aquifer waters are primarily derived from the breakdown of silicate minerals including feldspars, mica, and various clay minerals including illite.

There is no consistent trend in the proportion of potassium to sodium in the waters as total dissolved solids increases. Because potassium is usually the most tightly bound ion in cation exchange reactions, its relative abundance in the samples from the McQueen Branch and Crouch Branch aquifers suggests that cation exchange has not played a dominant role in the evolution of these waters. The exceptions are the samples from well C-10, where sodium is clearly the dominant cation. In this downgradient locality south of SRS, cation exchange processes have led to water conditions comparable to those formed by exchange processes observed in other regions of the South Carolina Coastal Plain.

Increases in the HCO_3^- concentration are apparently largely through the microbial oxidation of lignite within the aquifers. The ^{13}C signatures of the water are typically light (in the range of -0% to -25%). Usually, these light values indicate an organic source of carbon rather than the dissolution of limestone or other inorganic ion source.

Dissolved oxygen is less than 0.1 mg/L for most of the samples from the Dublin-Midville aquifer system. From Upper Three Runs southward, the aquifers in this system are anaerobic and contain abundant dissolved iron. The iron content in these aquifers is undesirably high, usually between 1 and 5 mg/L. The anaerobic conditions allow the dissolved iron to remain in the ferrous form but have not become reducing to the extent that sulfate has been reduced to the sulfide form.

A high-iron groundwater zone in the Middendorf aquifer (comparable to McQueen Branch), approximately 124 miles (40 km) wide, extends across South Carolina from SRS to North Carolina approximately paralleling the Fall Line. This high-iron zone is inferred to result from the reduction of iron oxyhydroxide grain coatings by bacteria during the oxidation of organic matter within the confined zones of the aquifer. The activity of the iron-reducing bacteria may inhibit the activity of sulfate-reducing bacteria. Sulfate reduction begins further downgradient after the more easily oxidized organics have been consumed.

1.3.4.5.1.3 Water Chemistry - F Area and MFFF Site

A monitoring well network consisting of over 100 wells has been installed to monitor groundwater quality in F Area. Well construction information, including maps showing well location, is provided in the SRS quarterly well inventory. The most recent sampling information is presented in periodic SRS groundwater monitoring reports and the SRS Environmental Report for 2004 (WSRC 2005).

The potential local groundwater recharge zone in F Area is the upland area with downward vertical gradients just to the southeast of F Area. Recharge areas for the Cretaceous aquifers are located outside of the SRS boundary. Construction of the F-Area facilities has had no effect on groundwater recharge areas and, groundwater has had no effect on construction activities. Construction of the MFFF site will have no adverse effect on groundwater recharge areas. As indicated in the MFFF Site Geotechnical Report (DCS 2005), the design groundwater elevation is 210 ft above msl. The bottom of the deepest excavation planned for construction of the BMF is Elevation 250 ft above msl; therefore, the MFFF will not penetrate into the ground water and groundwater will have no effect on construction activities.

1.3.4.6 Groundwater Hydrology at the MFFF Site

Section 1.4.2 of *Natural Phenomena Hazards and Design Criteria and Other Characterization Information for the Mixed Oxide (MOX) Fuel Fabrication Facility at Savannah River Site (U)*, (WSRC 2000b) and the *MOX Fuel Fabrication Facility Site Geotechnical Report* (DCS 2000) discuss the groundwater hydrology at SRS and the MFFF site. This section presents a summary of groundwater hydrology for the MFFF site.

The groundwater conditions at the MFFF site have the same characteristics as F Area. Groundwater in the shallow, intermediate, and deep aquifers at the MFFF site flows in different directions, depending on the depths of the streams that cut the aquifers. The Upper Three Runs aquifer is the shallow aquifer beneath the MFFF site that flows to the north and discharges into Upper Three Runs. The Gordon aquifer underlies the Upper Three Runs aquifer at the MFFF site and flows horizontally toward the Savannah River. Groundwater in the intermediate and deep aquifers flows horizontally towards the Savannah River southeast toward the coast.

Groundwater at the MFFF site also moves vertically. In the Upper Three Runs and Gordon aquifers, flow moves downward until its movement is obstructed by impermeable material. Operating under a different set of physical conditions, groundwater in the intermediate and deep aquifers flows mostly horizontally. At the MFFF site, flow from deeper aquifers moves upward due to higher water pressure below the confining unit between the upper and lower aquifer systems. This upward movement helps to protect the lower aquifers from contaminants found in the shallow Upper Three Runs aquifer. Groundwater in F Area varies in elevation from 190 to 220 ft (58 to 67 m) above msl and is found at a depth of over 50 ft (15 m) below existing ground level.

Groundwater quality in F Area and the MFFF site is not significantly different from that for SRS as a whole. It is abundant, usually soft, slightly acidic, and low in dissolved solids. High dissolved iron concentrations occur in some aquifers. Where needed, groundwater is treated to raise the pH and remove iron.

The *2000 RCRA Part B Permit Renewal Application* (WSRC 2000a) provides a comprehensive discussion of groundwater contamination plumes in F Area and covers the MFFF site. Also, the *Application for a Hazardous Waste Part B, Post-Closure Care Permit, Mixed Waste Management Facility, Hydrogeological Characterization (U)* (WSRC 1992) defines the soil and groundwater contamination from past wastewater discharge into the seepage basin. The Old F-Area Seepage Basin is located just west of the MFFF site as shown on Figure 1.2-3. The *Groundwater Mixing Zone Application for the Old F-Area Seepage Basin (U)* (WSRC 1997a) describes the justification and basis for the groundwater mixing zone application in support of the Record of Decision (ROD) for closure of the seepage basin. However, subsequent to the original ROD, groundwater monitoring data strongly suggested that contamination observed within the mixing zone was derived from multiple upgradient sources, including but not limited to the Old F-Area Seepage Basin (OFASB). This circumstance, the commingling of contaminant plumes from multiple sources, effectively invalidates the mixing zone.

The Department of Energy (DOE), SC Department of Health and Environmental Control (SCDHEC), and US Environmental Protection Agency (US EPA) have agreed to decouple and manage separately the groundwater component and the surface unit at OFASB. To support this decision, DOE, SCDHEC, and US EPA have signed an ROD Amendment for the OFASB (WSRC-RP-2003-4136, Rev.1, April 2004) that eliminates the mixing zone. SRS has established the GSA Western Groundwater Operable Unit that includes groundwater associated with OFASB and other F Area facilities. This new groundwater operable unit is being characterized (WSRC-RP-2003-4147, RFI/RI Phase I Work Plan for GSA Western Groundwater OU, Rev. 1.1, July 2004) and any long term monitoring strategies or remedial actions will be established by DOE, SCDHEC, and US EPA.

The selected remedial action for closure of OFASB consisted of stabilizing the contaminated soil within the basin by an in-situ cement slurry mixing method, filling the basin with clean soil, and capping the basin. The contaminated soil zone within OFSAB was remediated in 2000.

These reports indicated that, at that time, there was no known soil or groundwater contamination beneath critical structures located on the MFFF site. This was confirmed with the recent comprehensive geotechnical investigations conducted in 2000 at the MFFF site. During the site exploration program, radiological testing was performed for drill cuttings and samples. Radioactive contamination was not detected in samples obtained from the Upper Three Runs aquifer or Gordon aquifer, which are the upper aquifers at the MFFF site.

These reports also indicate that the groundwater contamination plumes in F Area are well defined as to direction of movement. The identified contamination plumes in F Area have well-defined monitoring and testing programs that are in compliance with State and EPA requirements. The *2000 RCRA Part B Permit Renewal Application* (WSRC 2000a) indicates that the northwest plume from the Old Radioactive Waste Burial Ground, located southeast of F Area, will not pass beneath the MFFF site. The Old F-Area Seepage Basin reports indicate that the groundwater contamination plume is located at a depth of over 70 ft (21.3 m) and extends beneath a portion of the northwest corner of the MFFF site boundary. The plume migration is to the north towards Upper Three Runs and away from the MFFF site. Therefore, further migration of the contaminated plume onto the MFFF site is not anticipated. Continued dilution of this contamination plume in the northwest portion of the MFFF site is expected to occur from northward-flowing groundwater beneath the MFFF site.

During the 2000 geotechnical investigations, radiological testing was performed for drill cuttings and samples to ensure worker protection and acceptability of samples for transport over public highways. The scans consisted of local reading with a G-M meter from each location for which materials were removed for geotechnical testing. The nominal sensitivity for worker protection and transportation measurements is 0.1 mrem/hr. Following field measurements, select samples were analyzed in the laboratory for gross alpha and gross beta with minimum detectable activities of about 200 nCi/gm of gross alpha and about 100 pCi/gm of gross beta. Radioactive contamination was not detected in samples obtained at the MFFF site.

Subsequent to the 2000 geotechnical investigations, DOE reported exceedances of drinking water maximum contaminant levels in the Old F-Area Seepage Basin monitoring wells. As a consequence of the exceedances in wells FNB-13, FNB-14, and FNB-15, MOX Services performed a groundwater survey on the MFFF site before beginning additional geotechnical work. Results of that sampling indicate that there was no groundwater located above the “tan clay” confining zone of the Dry Branch Formation. The Upper Three Runs aquifer below the “tan clay” confining zone of the Dry Branch Formation, which is at least 70 ft (21.3 m) below the MFFF site, is apparently contaminated from up gradient sources in F Area and not solely from the Old F-Area Seepage Basin. Concentrations of gross alpha and beta activity, tritium, uranium, and trichloroethyene exceeded maximum contaminant limits for drinking water. The source of groundwater contamination is from various heavy industrial and nuclear operations over the past 50 years in the F Area. The contaminant plume appears to originate inside F Area and extends beneath the MFFF site with movement in a fan-like direction of groundwater flow under the MFFF site. Contamination is most pronounced under the western edge of the site. Contamination was confined to the groundwater below the “tan clay” confining zone of the Dry Branch Formation. The deepest MFFF construction activities are anticipated to occur at least 30 ft (9.1 m) above the zone of contamination (WSRC 2002a).

The planned site construction, preparation, and development for the MFFF facilities will be confined within the near-surface soils that comprise the Upper Three Runs aquifer. Only surface grading and shallow excavation are anticipated to level the northwest area of the MFFF site for construction of parking lots, roads, and shallow spread foundations to support the Technical Support Building and Administration Building. Excavations will not extend at depth to the groundwater level. The planned construction activities will not have any adverse effects on the existing aquifer systems beneath the MFFF site.

Table 1.3.4-1. Flow Summary for the Savannah River and Savannah River Site Streams (ft^3/sec)

	Mean	STD Dev.	7Q10	7-Day Low Flow
Savannah River				
at Augusta, GA	9493	2611	4332	3746
at SRS Boat Dock	----	----	4293	3773
at Hwy 301 [a]	10397	2830	4411	3991
at Clyo	12019	3687	5211	4513
Upper Three Runs				
at Hwy 278	105	8	56	55
at SRS Road C	211	30	100	86
at SRS Road A	245	41	100	84
Beaver Dam Creek				
at 400D	81.5	8.7	0.01	18
Fourmile Branch				
at SRS Site 7	17.8	5.4	0.58	3.2
Pen Branch				
at SRS Road B	7.5	8.2	0.27	0.22
at SRS Road A-13	210	45	5.5	8.8
Steel Creek				
at Hattiesville Bridge	160	12.3	12.9	12.0
Lower Three Runs				
below Par Pond	38.4	10.4	1.2	0.9
near Snelling, SC	85.8	27.9	16	15

[A] Eleven years are missing between 1971 and 1982.

Note: The flow data used for computing statistics for the Savannah River and SRS streams were based on U.S. Geological Survey stream gage measurements after construction of Thurmond Dam. Values listed for seven-day low-flow, 10-year recurrence (7Q10) are based on adjusted "natural" flows (i.e., without the effects of cooling water discharges from SRS reactors).

Data from WSRC 2000.

Table 1.3.4-2. Water Quality of the Savannah River above SRS for 1983 to 1987

Analyte	Units	No. of Analyses	Min	Max	Mean
Alkalinity	mg/L	36	13	23	18.28
Aluminum	mg/L	36	0.08	0.95	0.38
Ammonia	mg/L	36	0.04	0.27	0.11
Cadmium	mg/L	36	0	0	0
Calcium	mg/L	36	3.1	4.24	3.62
Chloride	mg/L	36	4	13	7.73
Chromium	mg/L	36	0	0.01	0.01
Conductivity	μS/cm	36	54	107	80.42
Copper	mg/L	36	0	0	0
DO	mg/L	72	6.4	24	9.42
Fixed residue	mg/L	36	1	17	7.69
Iron	mg/L	36	0.27	1.39	0.62
Lead	mg/L	36	0	0	0
Magnesium	mg/L	36	0.98	1.55	1.31
Manganese	mg/L	36	0.06	0.1	0.08
Mercury	mg/L	36	0	0	0
Nickel	mg/L	36	0	0.03	0.02
Nitrate + Nitrite	mg/L	36	0.02	0.63	0.27
Phosphate	mg/L	36	0.03	0.09	0.06
Sodium	mg/L	36	4.67	11.6	8.93
Sulfate	mg/L	36	4	9	6.82
Suspended Solids	mg/L	36	3	18	9.69
Temperature	C	36	8.9	24.8	17.48
Total Dissolved Solids	mg/L	36	48	85	63.89
Total Solids	mg/L	36	54	96	73.58
Turbidity	NTU	36	2.22	3.3	9.66
Volatile Solids	mg/L	36	1	7	2.34
Water Volume	L	36	1.08E+11	2.31E+12	8.4E+11
Zinc	mg/L	36	0	0.02	0.01
pH	pH	36	5.7	7.8	6.44

Data from WSRC 2000b.

Table 1.3.4-3. Annual Maximum Instantaneous Discharges of the Savannah River at Augusta, Georgia for Water Years 1921 Through 1999 (USGS Flow Data, 1922-1999)

Year	Discharge (cfs)	Year	Discharge (cfs)
1921	129,000	1961	34,800
1922	92,000	1962	32,500
1923	59,700	1963	31,300
1924	56,400	1964	87,100
1925	150,000	1965	34,600
1926	55,300	1966	39,300
1927	39,000	1967	35,900
1928	226,000	1968	35,900
1929	191,000	1969	45,600
1930	350,000	1970	25,200
1931	26,100	1971	63,900
1932	93,800	1972	33,700
1933	48,200	1973	40,200
1934	73,200	1974	32,900
1935	63,700	1975	45,600
1936	258,000	1976	33,300
1937	90,200	1977	34,200
1938	65,300	1978	43,100
1939	82,400	1979	37,300
1940	252,000	1980	47,200
1941	52,200	1981	17,300
1942	115,000	1982	30,700
1943	132,000	1983	66,100
1944	141,000	1984	34,000
1945	62,100	1985	25,700

Table 1.3.4-3 Annual Maximum Instantaneous Discharges of the Savannah River at Augusta, Georgia for Water Years 1921 Through 1999 (USGS Flow Data, 1922-1999) (continued)

Year	Discharge (cfs)	Year	Discharge (cfs)
1946	109,000	1986	21,000
1947	90,200	1987	29,200
1948	76,100	1988	13,600
1949	172,000	1989	20,200
1950	32,500	1990	35,300
1951	41,400	1991	59,200
1952	39,300	1992	22,100
1953	35,200	1993	45,100
1954	25,500	1994	40,700
1955	23,900	1995	33,600
1956	18,600	1996	34,400
1957	18,000	1997	26,300
1958	66,300	1998	43,000
1959	28,500	1999	19,000
1960	34,900		

Note: Station 02197000; drainage area 7,508 square miles (including Butler Creek drainage area). The maximum instantaneous discharge since gaging by the USGS began in 1882 is 350,000 cfs on October 3, 1929. The maximum historical flow is 360,000 cfs in 1796.

Data from WSRC 2000b.

Table 1.3.4-4. Annual Maximum Instantaneous Discharges of Upper Three Runs for Water Years 1967 Through 1999

Water Year	Discharge at Highway 278[a] (cfs)	Discharge at SRS Road C[b] (cfs)	Discharge at SRS Road A[c] (cfs)
1967	320	- [d]	
1968	237	-	-
1969	301	-	-
1970	303	-	-
1971	420	-	-
1972	382	-	-
1973	472	-	-
1974	260	-	-
1975	341	586	-
1976	429	732	1230
1977	304	540	717
1978	344	646	Not gauged
1979	341	680	996
1980	420	880	951
1981	308	582	620
1982	364	696	793
1983	472	880	1010
1984	466	840	861
1985	400	962	893
1986	360	802	780
1987	370	819	869
1988	278	460	428
1989	304	613	592
1990	202	869	572
1991	820	2040	2580

Table 1.3.4-4 Annual Maximum Instantaneous Discharges of Upper Three Runs for Water Years 1967 Through 1999 (continued)

Water Year	Discharge at Highway 278[a] (cfs)	Discharge at SRS Road C[b] (cfs)	Discharge at SRS Road A[c] (cfs)
1992	742	1010	926
1993	421	1280	1100
1994	302	826	667
1995	412	1240	1010
1996	240	691	638
1997	242	840	709
1998	596	-	1200
1999	252	-	717

[a] Station 02197300; drainage area 87 square miles.

[b] Station 02197310; drainage area 176 square miles.

[c] Station 02197315; drainage area 203 square miles.

[d] Indicates discharge point that was not monitored.

Data from WSRC 2000b.

Table 1.3.4-5. Annual Maximum Instantaneous Discharges of Tims Branch for Water Years 1974 Through 1995, Station 02197309

Water Year	Discharge at SRS Road C (ft^3/s)[a]	Gage Height (ft msl)
1974	N/A	N/A
1975	N/A	N/A
1976	61	6.17
1977	N/A	N/A
1978	N/A	N/A
1979	N/A	N/A
1980	N/A	N/A
1981	N/A	N/A
1982	N/A	N/A
1983	NM	NM
1984	N/A	N/A
1985	41	144.76
1986	42	144.88
1987	63	145.16
1988	38	144.28
1989	38	144.26
1990	91	145.27
1991	129	145.69
1992	61	144.77
1993	107	145.47
1994	77	145.07
1995	107	145.47

[a] Drainage area 17.5 square miles.

N/A = data not available.

NM = discharge point not monitored.

Data from WSRC 2000b.

Table 1.3.4-6. Annual Maximum Daily Discharges of Fourmile Branch for Water Years 1980 Through 1999

Water Year	Discharge at SRS Road C[a] (cfs)	Discharge at SRS Road A-7[b] (cfs)	Discharge at SRS Road A-12.2[c] (cfs)
1980	288	204	903
1981	123	-[d]	585
1982	262	177	745
1983	136	163	678
1984	267	189	692
1985	149	121	621
1986	211	181	415
1987	161	163	436
1988	89	74	102
1989	-	157	392
1990	-	1230	1060
1991	-	-	-
1992	135	465	493
1993	126	500	477
1994	90	176	-
1995	179	610	595
1996	89	156	200
1997	-	254	299
1998	-	773	837
1999	-	194	264

[a] Station 02197340; drainage area 7.53 square miles.
[b] Station 02197342; drainage area 12.5 square miles.
[c] Station 02197344; drainage area 22.0 square miles.
[d] Indicates discharge unknown.
Data from WSRC 2000b.

Table 1.3.4-7. Probable Maximum Precipitation for F Area

Time (hr)	Incremental Rainfall (in)	Total Rainfall (in)
0	–	0
1	2.2	2.2
2	2.8	5
3	3.1	8.1
4	15.1	23.2
5	4.9	28.1
6	2.7	30.8

Data from WSRC 2000b.

Table 1.3.4-8. Hour Storm Rainfall Distributions as a Function of Annual Probability of Exceedance (Rainfall/Inches)

Annual Probability of Exceedance	2E-02	1E-02	2E-03	1E-03	2E-04	1E-04	2E-05	1E-05
Hour 1	0.035	0.039	0.052	0.058	0.074	0.082	0.103	0.114
Hour 2	0.062	0.070	0.093	0.104	0.132	0.147	0.185	0.204
Hour 3	0.083	0.094	0.124	0.138	0.176	0.196	0.247	0.272
Hour 4	0.242	0.273	0.361	0.403	0.515	0.571	0.721	0.795
Hour 5	0.393	0.445	0.587	0.656	0.838	0.929	1.174	1.294
Hour 6	0.524	0.593	0.783	0.874	1.117	1.239	1.566	1.725
Hour 7	0.725	0.819	1.082	1.208	1.544	1.712	2.163	2.384
Hour 8	1.863	2.106	2.781	3.105	3.969	4.401	5.562	6.129
Hour 9	1.139	1.287	1.700	1.898	2.426	2.690	3.399	3.746
Hour 10	0.628	0.710	0.937	1.047	1.338	1.483	1.875	2.066
Hour 11	0.414	0.468	0.618	0.690	0.882	0.978	1.236	1.362
Hour 12	0.338	0.382	0.505	0.564	0.720	0.799	1.009	1.112
Hour 13	0.117	0.133	0.175	0.196	0.250	0.277	0.350	0.386
Hour 14	0.076	0.086	0.113	0.127	0.162	0.179	0.227	0.250
Hour 15	0.048	0.055	0.072	0.081	0.103	0.114	0.144	0.159
Hour 16	0.035	0.039	0.052	0.058	0.074	0.082	0.103	0.114
Hour 17	0.035	0.039	0.052	0.058	0.074	0.082	0.103	0.114
Hour 18	0.028	0.031	0.041	0.046	0.059	0.065	0.082	0.091
Hour 19	0.028	0.031	0.041	0.046	0.059	0.065	0.082	0.091
Hour 20	0.021	0.023	0.031	0.035	0.044	0.049	0.062	0.068
Hour 21	0.021	0.023	0.031	0.035	0.044	0.049	0.062	0.068
Hour 22	0.021	0.023	0.031	0.035	0.044	0.049	0.062	0.068
Hour 23	0.014	0.016	0.021	0.023	0.029	0.033	0.041	0.045
Hour 24	0.014	0.016	0.021	0.023	0.029	0.033	0.041	0.045
Accumulation	6.900	7.800	10.300	11.500	14.700	16.300	20.600	22.700

Data from WSRC 2000b.

Table 1.3.4-9. Design Basis Flood for SRS Areas

Performance Category Annual Exceedance Probability	1 2E-03	2 5E-04	3 1E-04	4 1E-05
Tims Branch Basin (A Area)				
Flood (CFS)	2399	3568	5154	8233
Flood Elevation (Feet above msl)	247.1	247.4	247.6	248.2
Fourmile Branch Basin (C Area)				
Flood (cfs)	2072	3040	4413	7102
Flood Elevation (Feet above msl)	189.3	190.3	191.5	193.6
Fourmile Branch Basin (E Area)				
Flood (cfs)	1440	2155	3189	5246
Flood Elevation (Feet above msl)	202.0	203.0	204.4	207.9
Upper Three Runs Basin (F Area)				
Flood (cfs)	11966	17396	25022	39576
Flood Elevation (Feet above msl)	144.4	146.6	148.6	150.9
Fourmile Branch Basin (F Area)				
Flood (cfs)	1683	2507	3700	6058
Flood Elevation (Feet above msl)	193.2	194.2	195.5	197.7
Fourmile Branch Basin (H Area)				
Flood (cfs)	1404	2103	3113	5126
Flood Elevation (Feet above msl)	236.1	236.8	237.1	239.2
Pen Branch Basin (K Area)				
Flood (cfs)	4430	6224	8638	13185
Flood Elevation (Feet above msl)	176.3	177.7	179.7	182.5
Indian Grave Branch Basin (K Area)				
Flood (cfs)	781	1087	1524	2326
Flood Elevation (Feet above msl)	180.5	181.1	181.8	182.9
Upper Three Runs Basin (S Area)				
Flood (CFS)	11966	17396	25022	39576
Flood Elevation (Feet above msl)	151.8	153.4	155.3	158.2
Upper Three Runs Basin (Z and Y Areas)				
Flood (cfs)	11966	17396	25022	39576
Flood Elevation (Feet above msl)	158.5	160.4	161.7	163.8

Data from WSRC 2000b.

Table 1.3.4-10. Design Basis Flood for MFFF Site

Performance Category Annual Exceedance Probability	1 2E-03	2 5E-04	3 1E-04	4 1E-05
Upper Three Runs Basin				
Flood (cfs)	11966	17532	25022	39576
Flood Elevation (feet above msl)	146.4	148.4	150.5	153.1
Fourmile Branch Basin				
Flood (cfs)	1440	2155	3189	5246
Flood Elevation (feet above msl)	202.0	203.0	204.4	207.9

Data from WSRC 2000b.

Table 1.3.4-11. Hydraulic Parameters of the Carbonate Phase of the Floridan Aquifer

Parameter	Value [Mean] (Average)	Maximum	Minimum	Comments
Transmissivity	[1,486 m^2/day]	9,290 m^2/day	30 m^2/day	Floridan undifferentiated, South Carolina
		46,450	929	Upper Floridan, various areas, Georgia
		3,066	2,601	Upper Floridan, Savannah, Georgia
	(929 to 4,645)			Upper Floridan, Coastal South Carolina
		20,066	186	Lower Floridan
		465	46	Lower Floridan
		929	65	Updip clastic phase
Hydraulic Conductivity	(53 to 122 m/day)			Upper Floridan, Beaufort County
		31 m/day	23 m/day	Lower Floridan, Coastal South Carolina

Data from WSRC 2000b.

Table 1.3.4-12. Parameters Determined for the Upper Three Runs Aquifer

Parameter	Value [Mean] (Average)	Maximum	Minimum	Comments
Hydraulic conductivity (vertical)	[2.71×10^{-3} m/d]	1.55×10^{-1} m/d	8.2×10^{-3} m/d	Clayey sand samples
Hydraulic conductivity (horizontal)	[3.38×10^{-3} m/d]	7.3×10^{-1}	9.66×10^{-4}	Clayey sand samples
Porosity	[40%]	55%	10%	Clayey sand samples
Effective porosity	12%			Clayey sand samples
Hydraulic conductivity (vertical)	5.09×10^{-3} m/d	6.4×10^{-3} m/d	1.04×10^{-3} m/d	Sandy clay samples
Hydraulic conductivity (horizontal)	1.24×10^{-4} m/d	9.85×10^{-2}	7.77×10^{-4}	Sandy clay samples
Porosity	41%	71%	23%	Sandy clay samples
Effective porosity	5%			Sandy clay samples
Leakance coefficient		2.58×10^{-4} m/d	4.11×10^{-4} m/d	

Data from WSRC 2000b.

Table 1.3.4-13. Water Quality of the Savannah River Below SRS (River Mile 120) for 1992-1994

Analyte	Units	Number of Analyses	Min	Max	Mean
Alkalinity	mg/L	48	13	26	19.24
Aluminum	mg/L	36	0.08	0.64	0.4
Ammonia	mg/L	48	00.02	0.44	0.13
BOD 5 Day	mg/L	12	0.7	1.8	1.29
Cadmium	mg/L	36	0	0	0
Calcium	mg/L	38	3.26	5.02	4.18
Chloride	mg/L	36	4	12	6.27
Chromium	mg/L	36	0	0.01	0.01
Conductivity	μS/cm[a]	48	51	114	83.93
Copper	mg/L	36	0	0	0
DO	mg/L	84	5.8	21	8.77
Fecal Colloms	MPNECMED[b]	12	430	9300	3749.17
Fixed Residue	mg/L	36	1	42	8.81
Iron	mg/L	36	0.40	1.32	0.79
Lead	mg/L	36	0	0	0
Magnesium	mg/L	36	0.92	1.52	1.3
Manganese	mg/L	36	0.03	0.1	0.07
Mercury	mg/L	36	0	0.92	0.23
Nickel	mg/L	36	0	0.03	0.02
Nitrate + Nitrite	mg/L	48	0.11	0.47	0.29
pH	pH	1	6.7	6.7	6.7
Phosphate	mg/L	36	0.03	0.01	0.06
Sodium	mg/L	36	5.28	13	9.29
Sulfate	mg/L	36	4	11	7.64
Suspended Solids	mg/L	36	3	48	11.31

Table 1.3.4-13 Water Quality of the Savannah River Below SRS (River Mile 120) for 1992-1994 (continued)

Analyte	Units	Number of Analyses	Min	Max	Mean
TOC	mg/L	12	1.5	14	5.08
Temperature	C	60	1	30	17.83
Total Dissolved Solids	mg/L	36	49	105	65.94
Total Phosphate	mg/L	12	0.07	0.13	0.1
Total Solids	mg/L	36	54	120	77.26
Turbidity	JTU[c]	48	2.66	32.4	10.77
Volatile Solids	mg/L	36	1	9	2.72
Water Volume	L	36	4E+11	2.68E+12	9.58E+11
Zinc	mg/L	36	0	0.01	0.01
pH	pH	36	5.9	7.2	6.34
pH (lab)	pH	12	6.7	7	6.86

[a] microsiemens per centimeter.

[b] Maximum probable number per 100 mL.

[c] Jackson turbidity units.

Data from WSRC 2000b.

Table 1.3.4-14. Pumpage for Municipal Supplies

Location	User	Distance From SRS Center (miles)	Average Number Served	Water - Daily Use (gpd x 10^6)	Bearing Formation[a]	Type Source
Aiken County, SC						
1	City of Aiken	22	28,000	2.0	"Tuscaloosa"[b]	Springs
2	Town of Jackson	10	3,152	0.175	"Tuscaloosa"	2 Wells
3	Town of New Ellenton	11	4,000	0.300	"Tuscaloosa"	2 Wells
4	Town of Langley	19	1,330	0.130	"Tuscaloosa"	2 Wells
5	College Acres	15	1,264	0.065	"Tuscaloosa"	3 Wells
6	Bath Water Dist.	19	1,239	0.325	"Tuscaloosa"	2 Wells
7	Beech Island	18	4,500	0.300	"Tuscaloosa"	3 Wells
8	Talatha	10	1,260	0.040	"Tuscaloosa"	2 Wells
9	Breezy Hill	22	4,500	0.233	"Tuscaloosa"	4 Wells
10	Burnettown	20	1,200	0.150	"Tuscaloosa"	2 Wells
11	Montmorenci	17	4,232	0.423	"Tuscaloosa"	2 Wells
12	Warrenville	19	788	0.300	"Tuscaloosa"	4 Wells
13	Johnstown	18	1,560	0.144		
	Nowlandville	18	1,232	0.100	"Tuscaloosa"	1 Well
	Gloverville	18	1,440	0.144	Gloverville	
14	Belvedere	24	6,300	0.362	"Tuscaloosa"	5 Wells
15	Barnwell	15	6,500	4.0	Congaree	11 Wells
16	Williston	15	3,800	0.700	Santee	4 Wells
					"Tuscaloosa"	
Barnwell County, SC						
17	Blackville	22	2,975	0.300	"Tuscaloosa"	3 Wells
18	Hilda	22	315	0.009	"Tuscaloosa"	1 Well
19	Elko	17	315	0.010	Santee	1 Well
20	Girard	16	210	0.020	"Tuscaloosa"	3 Wells

[a] Many of these wells are gravel-packed from the bottom of the well to the free water table; thus, the water-bearing formation may not be clearly defined.

[b] "Tuscaloosa" refers to undifferentiated Cretaceous formations of the Lumbee Group.

Data from WSRC 2000b.

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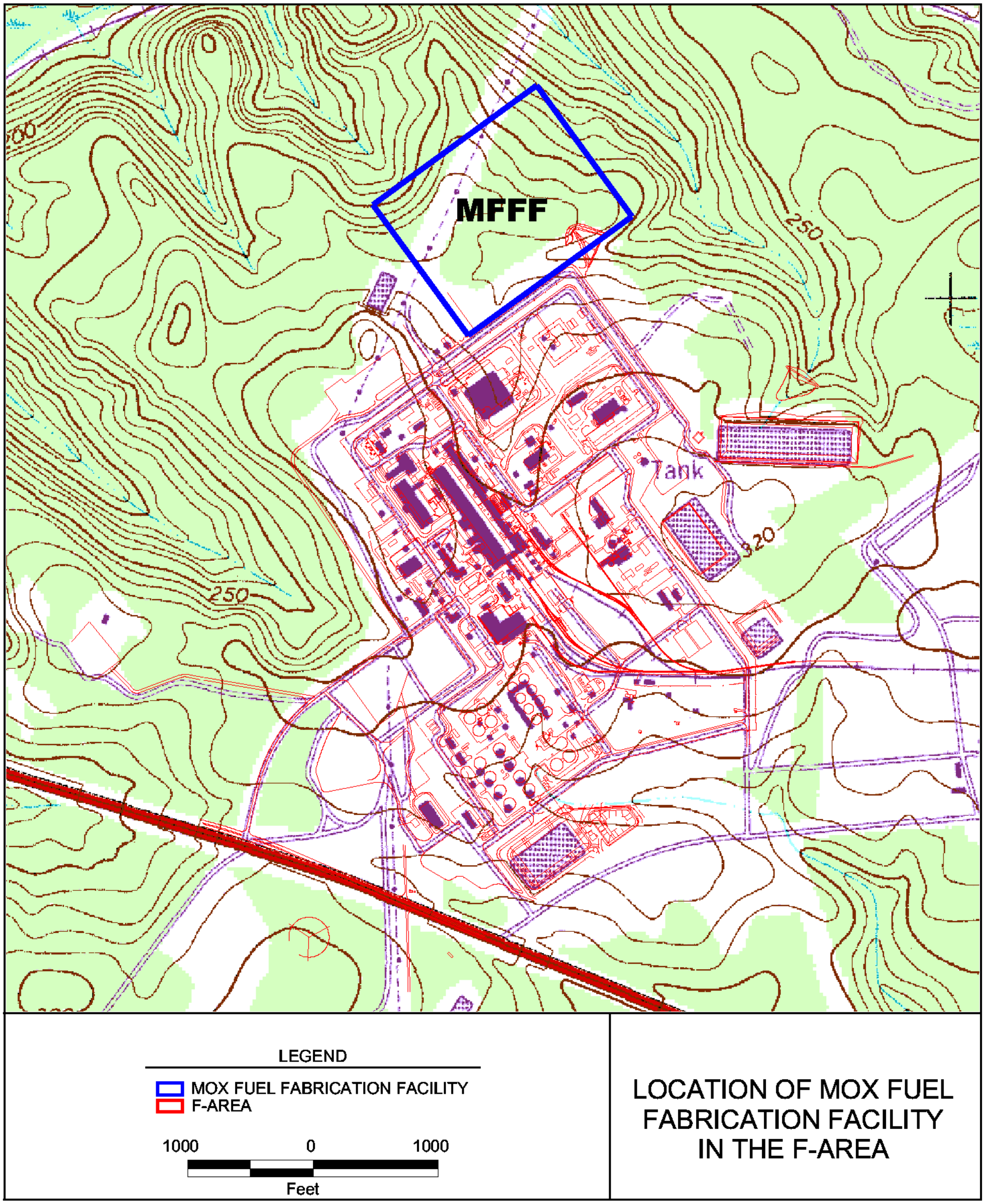
Figure 1.3.4-5. Location of the MFFF in F Area



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1.3.5 Geology

1.3.5.1 Regional Geology

The following discussion on the regional and MOX MFFF site geology is based on detailed discussions presented in Section 1.4.3 of the *Natural Phenomena Hazards (NPH) Design Criteria and Other Characterization Information for the Mixed Oxide (MOX) Fuel Fabrication Facility at Savannah River Site (U)* (Westinghouse Savannah River Company, LLC [WSRC] 2000b) and in the *MOX Fuel Fabrication Facility Site Geotechnical Report* (DCS 2005). The area of interest evaluated includes a radius of about 200 miles (322 km) from SRS and the MFFF site. The information also provides the basis for understanding the regional and SRS geology as applied to the subsurface encountered at the MFFF site.

Many SRS investigations and an extensive literature review have been used to reach the conclusion that there are no known capable or active faults within the 320-km (200-mile) radius of the site that influence the seismicity of the region, with the exception of the blind, poorly constrained faults associated with the Charleston seismic zone (WSRC 2000b).

The southeastern continental margin, within a 200 mile (mi) (322 km) radius of SRS, contains portions of all the major divisions of the Appalachian orogen (mountain belt) in addition to the elements that represent the evolution to a passive margin.

Within the Appalachian orogen, several lithotectonic terranes that have been extensively documented include the foreland fold belt (Valley and Ridge) and western Blue Ridge Precambrian-Paleozoic continental margin; the eastern Blue Ridge-Chauga Belt-Inner Piedmont terrane; the volcanic-plutonic Carolina terrane; and the geophysically defined basement terrane beneath the Atlantic Coastal Plain. These geological divisions record a series of compressional and extensional events that span the Paleozoic. The modern continental margin includes the Triassic-Jurassic rift basins that record the beginning of extension and continental rifting during the early to middle Mesozoic. The offshore Jurassic-Cretaceous clastic-carbonate bank sequence covered by younger Cretaceous and Tertiary marine sediments, and onshore Cenozoic sediments represent a prograding shelf-slope and the final evolution to a passive margin. Other offshore continental margin elements include the Florida-Hatteras shelf and slope and the unusual Blake Plateau basin and escarpment.

From the Cumberland Plateau and the Valley and Ridge provinces to the offshore Blake Plateau basin, the regional geology records the complete cycles of opening and closing of Paleozoic oceans and the opening of a new ocean (Atlantic). Late Proterozoic rifting is recorded in rift-related sediments at the edge of the frontal Blue Ridge province and the Ocoee and Tallulah Falls basins in the western and eastern Blue Ridge, respectively. Passive margin conditions began in the middle Cambrian and persisted through early Ordovician. The Cambro-Ordovician sedimentary section in the Valley and Ridge reflects this condition. The collision-accretionary phase of the Appalachians began in the middle Ordovician and persisted with pulses through the early Permian. Mesozoic rifting of the continents led to the creation of Triassic rift basins on the modern eastern continental margin and ultimately to the creation of the Atlantic Ocean basin. The evolution to a passive margin is recorded in the Cretaceous through Holocene Coastal Plain sediments and offshore carbonate bank and shelf sequences.

The two predominant processes sculpting the landscape during this tectonically quiet period included erosion of the newly formed highlands and subsequent deposition of the sediments on the coastal plain to the east. The passive margin region consists of a wedge of Cretaceous and Cenozoic sediments that thicken from near zero at the Fall Line to about 1,100 ft (335 m) in the center of SRS, and to approximately 4,000 ft (1,220 m) at the South Carolina coast. The fluvial to marine sedimentary wedge consists of alternating sand and clay with tidal and shelf carbonates common in the downdip Tertiary section.

1.3.5.1.1 Valley and Ridge Province

The Valley and Ridge Province includes Paleozoic sedimentary rocks consisting of conglomerate, sandstone, shale, and limestone. The shelf sequence was extensively folded and thrust faulted during the Alleghanian collisional event. The physiography is expressed as a series of parallel ridges and valleys that are a result of the erosion of breached anticlines with the oldest layers exposed in the valleys and the younger layers forming the ridges. The topographic expression of the folds is best expressed in the central and southern Appalachians. In the central and northern Appalachians the folded structure is dominant and thrust faults are not as numerous or expressed at the surface. The eastern boundary with the Blue Ridge province is formed by the Blue Ridge-Piedmont thrust. This boundary is distinct in most places along the strike of the Appalachians and marks the change from folded rocks that are not penetratively deformed to rocks that are penetratively deformed.

1.3.5.1.2 Blue Ridge Province

The Blue Ridge geologic province is bounded on the southeast by the Brevard fault zone and on the northwest by the Blue Ridge-Piedmont fault system. The province is a metamorphosed basement/cover sequence that has been complexly folded, faulted, penetratively deformed, and intruded. These rocks record multiple late Proterozoic to late Paleozoic deformation (extension and compression) associated with the formation of the Iapetos Ocean and the Appalachian orogen. The province consists of a series of westward-vergent thrust sheets, each with different tectonic histories and different lithologies (including gneisses, plutons, metavolcanic, and metasedimentary rift sequences), as well as continental and platform deposits. The Blue Ridge-Piedmont fault system thrust the entire Blue Ridge province northwest over Paleozoic sedimentary rock of the Valley and Ridge province during the Alleghanian orogeny. The Blue Ridge geologic province reaches its greatest width in the southern Appalachians.

The Blue Ridge is divided into a western and an eastern belt separated by the Hayesville-Gossan Lead fault. Thrust sheets in the western Blue Ridge consist of a rift-facies sequence of clastic sedimentary rocks deposited on continental basement, whereas thrust sheets in the eastern Blue Ridge consist of slope and rise sequences deposited in part on continental basement and in part on oceanic crust. Western Blue Ridge stratigraphy consists of basement gneisses, metasedimentary, metaplutonic, and metavolcanic rocks, whereas Eastern Blue Ridge stratigraphy consists of fewer lithologies, more abundant mafic rocks, and minor amounts of continental basement. These divisions of the Blue Ridge are discussed in more detail below.

1.3.5.1.2.1 Western Blue Ridge

The western Blue Ridge consists of an assemblage of Middle Proterozoic continental (Greenville) basement nonconformably overlain by Late Proterozoic to Early Paleozoic rift and drift facies sedimentary rock. The basement consists of various types of gneisses, amphibolite, and gabbroic and volcanic rock and metasedimentary rock. All basement is metamorphosed to granulite or uppermost amphibolite facies. The calculated ages of these rocks generally range from 1,000 to 1,200 mega annum or millions of years (Ma).

The rifting event during the Late Proterozoic through Early Paleozoic that formed the Iapetos Ocean is recorded in the rift-drift sequence of the Ocoee Supergroup and Chillhowie Group. These rocks, basement and sedimentary cover, were all later affected by Taconic and possibly Acadian deformation and metamorphism. The entire composite thrust sheet was transported west as an intact package during the Alleghanian collision event on the Blue Ridge-Piedmont thrust.

1.3.5.1.2.2 Eastern Blue Ridge

The eastern Blue Ridge is located southeast of the western Blue Ridge and is separated from that province by the Hayesville-Gossan Lead fault. The Brevard fault zone forms the southeastern boundary with the Inner Piedmont. Lithologically, the eastern Blue Ridge is composed of continental slope, rise, and ocean floor metasedimentary rocks in association with oceanic or transitional to oceanic crust. This contrasts with the western Blue Ridge, which contains metasedimentary rocks suggesting continental rift-drift facies of a paleomargin setting. The eastern Blue Ridge is structurally complex with several major thrust faults, multiple fold generations, and two high-grade metamorphic episodes. Metamorphism took place during the Taconic and possibly Acadian orogenies.

1.3.5.1.3 Inner Piedmont Province

The Inner Piedmont province in northwestern South Carolina consists of variably deformed and metamorphosed igneous and sedimentary rocks ranging in age from Middle Proterozoic to Permian (1,100 to 265 Ma). The province consists of the Western Piedmont and the Carolina terrane. This designation is made because of different tectonic origins for the western and eastern parts of the province. The province can also be subdivided into seven distinctive tectonostratigraphic belts, separated by major faults (e.g., Towaliga fault), contrasts in metamorphic grade, or both. From northwest to southeast, these are the Chauga, Inner Piedmont, Kings Mountain, Charlotte, Carolina Slate, Kiokee, and Belair belts. The metamorphic grade of these belts alternates between low grade (Chauga, Kings Mountain, Carolina Slate, and Belair) and medium to high grade (Inner Piedmont, Charlotte, and Kiokee). The Charlotte and Carolina Slate belts are combined and discussed as the Carolina terrane. The rocks of the Inner Piedmont province have been deformed into isoclinal recumbent and upright folds, which have been refolded and are contained in several thrust sheets or nappes. These metamorphic rocks extend beneath the Coastal Plain sediments in central and eastern South Carolina. The southeastern extent of the Inner Piedmont province underneath the Coastal Plain is unknown.

1.3.5.1.3.1 Western Piedmont

The Western Piedmont encompasses the Inner Piedmont block, the Smith River Allochthon, and the Sauratown Mountains Anticlinorium. It is separated from the Blue Ridge province on the northwest by the Brevard fault zone. It is separated from the Carolina terrane on the southeast by a complex series of fault zones approximately coincident with the Central Piedmont suture. These faults include Lowndesville, Kings Mountain, Eufola, Shacktown, and Chatham fault zones. The province is a composite stack of thrust sheets containing a variety of gneisses, schists, amphibolite, sparse ultramafic bodies and intrusive granitoids. The protoliths are immature quartzo-feldspathic sandstone, pelitic sediments, and mafic lavas.

1.3.5.1.3.2 Carolina Terrane

The Carolina terrane is part of a late Precambrian-Cambrian composite arc terrane, exotic to North America, and accreted sometime during the Ordovician to Devonian. It consists of felsic to mafic volcanic rock and associated volcaniclastic rock. Middle Cambrian fossil fauna indicate a European or African affinity.

The northeastern boundary of the Carolina terrane is formed by a complex of faults that comprise the Central Piedmont suture and separate the terrane from rocks of North American affinity. This structure was reactivated during the later Alleghanian collisional events as a dextral shear fault system. Subsequent investigators have further established understanding of the complicated structure that suggested the Central Piedmont suture is a low-angle normal fault. The Carolina terrane is bounded on the southeast by the Modoc fault zone and the Kiokee belt.

The Carolina terrane is the combination of the earlier Charlotte and Carolina slate belts. The belts were initially distinguished by metamorphic grade and were later recognized as the same protolith and thus were combined. Metamorphic grade increases to the northwest from lower greenschist facies to upper amphibolite facies. Pre-Alleghanian structure is dominated by large northeast trending folds with steeply dipping axial surfaces. Country rock of the Carolina terrane has been penetratively deformed, thereby producing axial plane cleavage and foliation.

1.3.5.1.3.3 Kiokee Belt

The Kiokee belt is located between the Carolina terrane and the Belair belt in Georgia and South Carolina, and is referred to as the Savannah River terrane. The Kiokee belt is bounded on the northwest by the Modoc fault zone and on the southeast by the Augusta fault. It is a medium- to high-grade metamorphic belt with associated plutonism, and is recognized as the Alleghanian metamorphic core. The faults are mylonite zones that overprint the amphibolite facies infrastructure of the core of the belt. The core was deformed and metamorphosed prior to the development of the plastic shear zones bounding it.

The Kiokee belt is an antiformal structure that strikes northeast. The interior is a migmatitic complex of biotite amphibole paragneiss, leucocratic paragneiss, sillimanite schist, amphibolite, ultramafic schist, serpentinite, feldspathic metaquartzite, and granitic intrusions of Late Paleozoic age. Some of the lithologic units found in the Carolina slate belt may occur at higher metamorphic grade in the Kiokee belt.

1.3.5.1.3.4 Belair Belt

The Belair belt (also Augusta terrane) is locally exposed in the Savannah River valley, near Augusta, Georgia. It is largely concealed beneath the Atlantic Coastal Plain with several small erosional windows through the Coastal Plain sediments in eastern Georgia. The Belair belt consists of intermediate to felsic volcanic tuffs and related volcaniclastic sediments penetratively deformed and metamorphosed to greenschist facies. The Belair belt contains similar characteristics to the Carolina terrane. Geophysical and well data indicate that the Belair belt extends beneath the Atlantic Coastal Plain.

1.3.5.1.4 Mesozoic Rift Basins

Mesozoic age rift basins are found along the entire eastern continental margin of North America from the Gulf Coast through Nova Scotia. The basins formed in response to the continental rifting episode that broke up the super continent, Pangea, and led to the formation of the Atlantic Ocean basin. Rift basins are exposed in the Piedmont province as well as buried beneath Cretaceous and younger Coastal Plain sediments. Many underlie offshore regions. Structurally, the basins are grabens or half grabens, elongated in a northeast direction and are bounded by normal faults on one or both sides. Several basins were localized along reactivated Paleozoic ductile or brittle fault zones.

There are two belts of basins that trend northeastward along the continental margin from the Carolinas to Pennsylvania. In North and South Carolina, the Deep River, Elberbe and Crowburg basins are included in the eastern belt, and the Dan River and Davie County basins are in the western belt. The Dunbarton, Florence, Riddleville, and South Georgia basins are buried beneath coastal plain sediments in the eastern belt. The basins are generally filled with lacustrine sedimentary and igneous rock.

Strata within the basins consist mainly of non-marine sandstone, conglomerate, siltstone, and shale. Carbonate rocks and coal are found locally in several basins. Igneous rocks of basaltic composition occur as flows, sills, and stocks within the basins and as extensive dike swarms within and outside the basins. These basin fill strata have been described and named the Newark Supergroup. In general, the stratigraphy can be broken out into three sections. The lower section is characteristically fluvial and contains reddish-brown, arkosic coarse-grained sandstone, and conglomerate. The middle section mainly includes sediments of lacustrine origin. These sediments include grey-black fossiliferous siltstone, carbonaceous shale, and thin coal beds. The upper section is a complex of deltaic, fluvial, and lacustrine environments. These sediments include red-brown siltstone, arkosic sandstone, pebbly sandstone, and red and grey mudstone and conglomerate.

The Dunbarton basin beneath SRS has a master border fault dipping to the southeast, and so does the Riddleville basin in Georgia. The Dunbarton basin is not known to contain any basalt sills. The South Georgia Rift, in Georgia and South Carolina, is a much larger, deeper, and more complex basin than either the Riddleville or Dunbarton basins. The basin is as wide as 62.1 mi (100 km) and as deep as 4.3 mi (7 km). It is not a single basin but is a complex of isolated synrift grabens with limited to major crustal extension. The major border fault dips northward as opposed to southeastward for the master faults bounding Riddleville and Dunbarton basins.

1.3.5.1.5 Atlantic Coastal Plain

The sediments of the Atlantic Coastal Plain in South Carolina are stratified sand, clay, limestone, and gravel that dip gently seaward and range in age from Late Cretaceous to Recent. The sedimentary sequence thickens from essentially zero at the Fall Line to more than 4,000 ft (1,219 m) at the coast. Regional dip is to the southeast, although beds dip and thicken locally in other directions because of locally variable depositional regimes and differential subsidence of basement features such as the Cape Fear Arch and the South Georgia Embayment.

The Coastal Plain sedimentary sequence near the center of the region (i.e., SRS) consists of about 700 ft (213 m) of Late Cretaceous quartz sand, pebbly sand, and kaolinitic clay, overlain by about 60 ft (18 m) of Paleocene clayey and silty quartz sand, glauconitic sand, and silt. The Paleocene beds are in turn overlain by about 350 ft (107 m) of Eocene quartz sand, glauconitic quartz sand, clay, and limestone grading into calcareous sand, silt, and clay. The calcareous strata are common in the upper part of the Eocene section in downdip parts of the study area. In places, especially at higher elevations, the sequence is capped by deposits of pebbly, clayey sand, conglomerate, and clay of Miocene or Oligocene age. Lateral and vertical facies changes are characteristic of most of the Coastal Plain sequence, and the lithologic descriptions below are therefore generalized. The stratigraphic section, which delineates the coastal plain lithology, is divided into several formations and groups based principally on age and lithology.

1.3.5.1.5.1 Geology of the Coastal Plain Sediments - General

The following sections describe regional stratigraphy and lithologies, with emphasis on variations near SRS. The data presented are based upon direct observations of surface outcrops; geologic cores obtained during drilling of boreholes; microfossil age dating; and borehole geophysical logs. Several key boring locations within the SRS boundaries and in the adjacent regions are referenced throughout the following discussions.

Rocks of Paleozoic and Triassic ages have been leveled by erosion and are unconformably overlain by unconsolidated to poorly consolidated Coastal Plain. This erosional surface dips approximately 37 ft/mi (7 m/km) toward the southeast. The Atlantic Coastal Plain sediments in South Carolina are stratified sand, clay, limestone, and gravel that dip gently seaward and range in age from Late Cretaceous to Recent. Near the coast, the wedge is approximately 4,000 ft (1,219 m) thick.

1.3.5.1.5.2 Upper Cretaceous Sediments

Upper Cretaceous sediments overlie Paleozoic crystalline rocks or lower Mesozoic sedimentary rocks throughout most of the study area. The Upper Cretaceous sequence includes the basal Cape Fear Formation and the overlying Lumbee Group, which is divided into three formations. The sediments in this region consist predominantly of poorly consolidated, clay-rich, fine- to medium-grained, micaceous sand, sandy clay, and gravel, and is about 700 ft (213 m) thick near the center of the study area. Thin clay layers are common. In parts of the section, clay beds and lenses up to 70 ft (21 m) thick are present. Depositional environments were fluvial to prodeltaic.

Cape Fear Formation

The Cape Fear Formation rests directly on a thin veneer of saprolitic bedrock and is the basal unit of the Coastal Plain stratigraphic section at SRS. The saprolite ranges from less than 10 ft (3 m) to more than 40 ft (12 m) in thickness and defines the surface of the crystalline basement rocks and sedimentary rocks of the Newark Supergroup (Middle to Upper Triassic age). The thickness of the saprolite reflects the degree of weathering of the basement prior to deposition of the Cape Fear Formation. The Cape Fear Formation is encountered at about 200 ft (61 m) above msl just south of well C-3 in the north and at about 1,200 ft (366 m) above msl at well C-10 in the south. The Cape Fear Formation does not crop out in the study area, and its northern limit is north of the C-1 and P-16 wells and south of wells C-2 and C-3. The unit thickens to more than 230 ft (70 m) at well C-10 and has a maximum known thickness of about 700 ft (213 m) in Georgia. The top of the Cape Fear Formation dips approximately 30 ft/mi (5.7 m/km) to the southeast across the study area. The Cape Fear Formation was erosionally truncated prior to deposition of the overlying Middendorf Formation, resulting in a disconformity between the two formations.

Lumbee Group

Three formations of the Late Cretaceous Lumbee Group are present in the study area. These are, from oldest to youngest, the Middendorf, Black Creek, and Steel Creek Formations.

The Lumbee Group consists of fluvial and deltaic quartz sand, pebbly sand, and clay in the study area. The sedimentary sequence is more clayey and fine-grained downdip from the study area, reflecting shallow to deep marine shelf sedimentary environments. Thickness ranges from about 400 ft (122 m) at well C-3 in the north, to about 780 ft (238 m) near well C-10 in the south. At least part of the group crops out in the northern part of the study area but it is difficult to distinguish the individual formations. Consequently, the Lumbee Group was mapped as undifferentiated Upper Cretaceous. The dip of the upper surface of the Lumbee Group is to the southeast at approximately 20 ft/mi (3.8 m/km) across the study area.

The Middendorf Formation unconformably overlies the Cape Fear Formation with a distinct contact. The contact is marked by an abrupt change from the moderately indurated clay and clayey sand of the underlying Cape Fear to the slightly indurated sand and lesser clayey sand of the Middendorf. The basal zone is often pebbly. The contact is unconformable and is marked by a sudden increase in electrical resistivity on geophysical logs. Thickness of the formation ranges from approximately 120 ft (37 m) in well C-2 in the north, to 240 ft (73 m) in well C-10 in the south. It has a maximum known thickness of about 520 ft (158 m) in Georgia. The top of the formation dips to the southeast at about 26 ft/mi (4.9 m/km) across the study area. Fossil data for the Middendorf are sparse and the formation is not well dated in the study area.

Paleontological control for the Black Creek is poor updip in South Carolina and Georgia. A Late Cretaceous age has been suggested for the Black Creek Formation, as indicated by various paleontological data from the unit. Sediments assigned to the Black Creek Formation in the vicinity of SRS yield Late Cretaceous paleontological ages and unconformably overlie the Middendorf Formation.

The Black Creek Formation is penetrated at virtually all well-cluster sites in the study area. The unit ranges in thickness from approximately 150 ft (46 m) at well C-2 in the north to 300 ft

(91 m) near the center of the study area in well Pen Branch Formation-3 and to 370 ft (113 m) at well C-10 in the south. The unit dips approximately 22 ft/mi (4 m/km) to the southeast.

The Black Creek is distinguished from the overlying and underlying Cretaceous units by its better sorted sand, fine-grained texture, and relatively high clay content. It is generally darker, more lignitic, and more micaceous, especially in the updip part of the section, than the other Cretaceous units. In much of the study area, the lower one-third of the formation is mostly sand that is separated from the upper two-thirds of the unit by clay beds. These beds are 20 ft (6 m) to 40 ft (12 m) thick in the northern part of the region and more than 150 ft (46 m) at well C-10 in the south. In general, the top of the Black Creek Formation is picked at the top of a clay bed that ranges from 10 ft (3 m) to 25 ft (8 m) in thickness. The clay bed is exceptionally thick but not laterally extensive. For example, it is essentially absent in wells P-21, CPC-1, P-26, and P-29. This suggests lagoonal back barrier bay deposition associated with nearby shorelines. Often the thick clay beds flank the areas where shoaling is suggested owing to uplift along the Pen Branch and Steel Creek faults, which was contemporaneous with deposition. Overall, the Black Creek consists of two thick, fining-upward sequences each capped by thick clay beds. The lower sequence is predominantly silty, micaceous sand in the area of SRS, while the upper sequence is mostly clay and silt.

The Peedee Formation was previously considered by some investigators to be absent in the study area; however, recent paleontological evidence provides dates of Peedee age from sediment samples in the southern part of SRS. Because there is a considerable difference in lithology between the type Peedee and the sediments in the SRS region, Peedee-equivalent sediments in the vicinity of SRS were referred to as the "Steel Creek Member" of the Peedee Formation. The type well for the Steel Creek Formation is P-21, located near Steel Creek. The top of the Steel Creek is picked at the top of a massive clay bed that ranges from 3 ft (1 m) to more than 30 ft (9 m) in thickness. The formation dips approximately 20 ft/mi (3.8 m/km) to the southeast.

The unit ranges in thickness from approximately 60 ft (18 m) at well P-30 to 175 ft (53 m) at well C-10 in the south. It has a maximum known thickness of 380 ft (116 m) in Georgia. The Steel Creek section thins dramatically between the ALL-324 and the P-22 wells due to truncation by erosion at the Cretaceous-Tertiary unconformity. The Steel Creek Formation overlies the Black Creek Formation and is distinguished from it by a higher percentage of sand, which is represented on geophysical logs by a generally higher electrical resistivity and lower natural gamma radiation count.

1.3.5.1.5.3 Tertiary Sediments

Tertiary sediments range in age from Early Paleocene to Miocene and were deposited in fluvial to marine shelf environments. The Tertiary sequence of sand, silt, and clay generally grades into highly permeable platform carbonates in the southern part of the study area and these continue southward to the coast. The Tertiary sequence is divided into three groups, the Black Mingo Group, Orangeburg Group, and Barnwell Group, which are further subdivided into formations and members. These groups are overlain by the ubiquitous "Upland unit."

The Tertiary sedimentary sequence deposited in west-central South Carolina has been punctuated by numerous sea level low stands and/or affected by subsidence in the source areas (which reduced or eliminated sediment availability) resulting in a series of regional unconformities. Four such regionally significant unconformities are defined in the Tertiary stratigraphic section in A/M Area. From base upwards, they include the “Cretaceous-Tertiary” unconformity, the “Lang Syne/Sawdust Landing” unconformity, the “Santee” unconformity and the “Upland unit” unconformity. Based on these unconformities, four sequence stratigraphic units (unconformity bounded sedimentary units) have been delineated.

Sequence stratigraphic unit I includes the sediments deposited between the “Cretaceous-Tertiary” unconformity and the “Lang Syne/Sawdust Landing” unconformity, and includes the Lang Syne/Sawdust Landing Formations undifferentiated of the Black Mingo Group. Sequence stragraphic unit II lies between the Lang Syne/Sawdust Landing unconformity and the Santee unconformity, and includes from oldest to youngest the Fourmile/Congaree Formations undifferentiated, the Warley Hill Formation, the Tinker/Santee Formation of the Orangeburg Group, and the carbonates (Utley Member) of the Clinchfield Formation. The Santee unconformity that caps the sequence is a major erosional event in the SRS region. Sequence stratigraphic unit III lies between the Santee unconformity and the “Upland unit” unconformity, and includes the Dry Branch and Tobacco Road Formations of the Barnwell Group. Sequence stratigraphic unit IV includes the fluvial sediments overlying the “Upland unit” unconformity.

Black Mingo Group

The Black Mingo Group consists of quartz sand, silty clay, and clay that suggest upper and lower delta plain environments of deposition generally under marine influences. In the southern part of the study area, massive clay beds, often more than 50 ft (15 m) thick, predominate. Downdip from the study area, thin red to brown sandy clay beds, gray to black clay beds and laminated shale dominate the Black Mingo Group and suggest deposition in clastic shelf environments. At the South Carolina coast, carbonate platform facies-equivalents of the updip Black Mingo clastic sediments first appear. The carbonate units are referred to as “unnamed limestones.” These are equivalent to the thick beds of anhydrite and dolomite of the Paleocene Cedar Keys Formation and the lower Eocene glauconitic limestone and dolomite of the Oldsmar Formation. Both carbonate units are delineated and mapped in coastal Georgia and northeastern Florida.

Basal Black Mingo Group sediments were deposited on the regional “Cretaceous-Tertiary” unconformity that defines the base of sequence stratigraphic unit I. There is no apparent structural control of this unconformity. Above the unconformity, the clay and clayey sand beds of the Black Mingo Group thin and often pinch out along the traces of the Pen Branch and Crackerneck faults. This suggests that coarser-grained materials were deposited preferentially along the fault traces, perhaps due to shoaling of the depositional surface. This, in turn, suggests movement (reactivation) along the faults. This reactivation would have occurred during Black Mingo deposition, that is, in Paleocene and lower Eocene time.

The upper surface of the Black Mingo Group dips to the southeast at 16 ft/mi (3 m/km), and the group thickens from 60 ft (18 m) at well C-2 in the north, to about 170 ft (52 m) near well C-10 in the south. The group is about 700 ft (213 m) thick at the South Carolina coast. Throughout the downdip part of the South Carolina Coastal Plain, the Black Mingo Group consists of the Rhems Formation and the overlying Williamsburg Formation.

Lang Syne/Sawdust Landing Formations

The name of the Ellenton Formation was proposed for a subsurface lithologic unit in the SRS area consisting of beds of dark, lignitic clay and coarse sand, which are equivalent to the Sawdust Landing and Lang Syne Members of the Rhems Formation. It has been suggested that the Sawdust Landing Member and the overlying Lang Syne Member of the Rhems Formation be raised to formational status and replace the term Ellenton in the study area.

In the absence of detailed paleontological control, the Sawdust Landing Formation and the overlying Lang Syne Formation could not be systematically separated for mapping in this region. Thus, they are treated as a single unit; the Lang Syne/Sawdust Landing undifferentiated, on all sections and maps. The sediments of the unit generally consist of two fining-upward sand-to-clay sequences, which range from about 40 ft (12 m) in thickness at the northwestern boundary of SRS to about 100 ft (30 m), near the southeastern boundary. The unit is mostly dark gray to black, moderately to poorly sorted, fine to coarse-grained, micaceous, lignitic, silty and clayey quartz sand interbedded with dark gray clay and clayey silt. Pebbly zones, muscovite, feldspar, and iron sulfide are common. Individual clay beds up to 20 ft (6 m) thick are present in the unit. Clay and silt beds make up approximately one-third of the unit in the study area. The dark, fine-grained sediments represent lower delta plain, bay-dominated environments. Tan, light gray, yellow, brown, purple, and orange sand, pebbly sand, and clay represent upper delta plain, channel-dominated environments.

Snapp Formation (Williamsburg Formation)

Sediments in the study area that are time equivalent to the Williamsburg Formation differ from the type Williamsburg and have been designated the "Snapp Member of the Williamsburg Formation." It has been suggested that the "Snapp Member" of the Williamsburg be raised to formational status. The Snapp Formation is used in this report. The unit is encountered in well P-22 in the southeastern part of SRS near Snapp Station. The basal contact with the underlying Lang Syne/Sawdust Landing undifferentiated is probably unconformable. The Snapp Formation appears to pinch out in the northwestern part of SRS and thickens to about 50 ft (15 m) near the southeastern boundary of the site.

The Snapp Formation (Williamsburg Formation) crops out in Calhoun County. The sediments in the upper part of the unit consist of low-density, fissile, dark-gray to black siltstone, and thin layers of black clay interbedded with sand in the lower part. These and similar sediments in Aiken and Orangeburg Counties were probably deposited in lagoonal or estuarine environments. Within and near SRS, the Snapp Formation sediments typically are silty, medium- to coarse-grained quartz sand interbedded with clay. Dark, micaceous, lignitic sand also occurs, and all are suggestive of lower delta plain environments. In Georgia, the unit consists of thinly laminated, silty clay locally containing layers of medium- to dark-gray carbonaceous clay. This lithology is indicative of marginal marine (lagoonal to shallow shelf) depositional environments. Clayey parts of the unit are characterized on geophysical logs as zones of low electrical resistivity and a relatively high-gamma ray response. In the southernmost part of the study area, the Snapp consists of gray-green, fine to medium, well-rounded, calcareous quartz sand and interbedded micritic limestone and limey clay that is highly fossiliferous and glauconitic. This lithology suggests deposition in shallow shelf environments somewhat removed from clastic sediment sources.

The upper surface of the Snapp Formation is defined by the “Lang Syne/Sawdust Landing” unconformity and defines the upper boundary of sequence stratigraphic unit I. The surface has been offset by normal faulting as noted in A/M Area.

Fourmile Formation

Early Eocene ages, derived from paleontological assemblages, indicate that the sand immediately overlying the Snapp Formation in the study area is equivalent to the Fishburne. These sediments were deposited on the “Lang Syne/Sawdust Landing” unconformity and constitute the basal unit of sequence stratigraphic unit II. The Fishburne is a calcareous unit that occurs downdip near the coast. The sand was initially designated the Fourmile Member of the Fishburne Formation. Owing to the distinctive difference in lithology between the type, Fishburne Formation and the time-equivalent sediments observed in the study area, it has been recommended that the Fourmile Member of the Fishburne be raised to formational rank. The term Fourmile Formation is used in this report.

The Fourmile Formation averages 30 ft (9 m) in thickness, is mostly tan, yellow-orange, brown, and white, moderately to well-sorted sand, with clay beds a few feet thick near the middle and at the top of the unit. The sand is very coarse to fine grained, with pebbly zones common, especially near the base. Glauconite, up to about 5%, is present in places, as is weathered feldspar. In the center and southeastern parts of SRS, the unit can be distinguished from the underlying Paleocene strata by its lighter color and lower content of silt and clay. Glauconite and microfossil assemblages indicate that the Fourmile is a shallow marine deposit.

Overlying the Fourmile Formation in the study area is 30 ft (9 m) or less of sand similar to the Fourmile. This sand is better sorted, contains fewer pebbly zones, less muscovite and glauconite, and in many wells is lighter in color. Microfossil assemblages indicate that the sand is correlative with the early middle Eocene Congaree Formation. In some wells a thin clay occurs at the top of the Fourmile, separating the two units; however, the difficulty in distinguishing the Fourmile Formation from the overlying Congaree Formation has led many to include the entire 960 ft (293 m) section in the Congaree Formation.

Orangeburg Group

The Orangeburg Group consists of the lower middle Eocene Congaree Formation (Tallahatta equivalent) and the upper middle Eocene Warley Hill Formation and Santee Limestone (Lisbon equivalent). Over most of the study area, these post-Paleocene units are more marine in character than the underlying Cretaceous and Paleocene units; they consist of alternating layers of sand, limestone, marl, and clay.

The group crops out at lower elevations in many places within and near SRS. The sediments thicken from about 85 ft (26 m) at well P-30 near the northwestern SRS boundary to 200 ft (61 m) at well C-10 in the south. Dip of the upper surface is 12 ft/mi (2 m/km) to the southeast. Downdip at the coast, the Orangeburg Group is about 325 ft (99 m) thick and is composed of shallow carbonate platform deposits of the Santee Limestone.

In the extreme northern part of the study area, the entire middle Eocene Orangeburg Group is mapped as the Huber Formation. The micaceous, poorly sorted sand, abundant channel fill deposits and cross bedding, and carbonaceous kaolin clay in the Huber is indicative of fluvial, upper delta plain environments.

In the central part of the study area, the group includes, in ascending order, the Congaree, Warley Hill, and Tinker/Santee Formations. The units consist of alternating layers of sand, limestone, marl, and clay that are indicative of deposition in shoreline to shallow shelf environments. From the base upward, the Orangeburg Group passes from clean shoreline sand characteristic of the Congaree Formation to shelf marl, clay, sand, and limestone typical of the Warley Hill and Santee Limestone. Near the center of the study area, the Santee sediments consist of up to 30 vol% carbonate. The sequence is transgressive, with the middle Eocene Sea reaching its most northerly position during Tinker/Santee deposition.

Toward the south, near wells P-21, ALL-324, and C-10, the carbonate content of the three formations increases dramatically. The shoreline sand of the Congaree undergoes a facies change to interbedded glauconitic sand and shale, grading to glauconitic argillaceous, fossiliferous, sandy limestone. Downdip, the fine-grained, glauconitic sand, and clay of the Warley Hill become increasingly calcareous and grades imperceptibly into carbonate-rich facies comparable to both the overlying and underlying units. Carbonate content in the glauconitic marl, calcareous sand, and sandy limestone of the Santee increases towards the south. Carbonate sediments constitute the vast majority of the Santee from well P-21 southward.

Congaree Formation

The early middle Eocene Congaree Formation has been traced from the Congaree valley in east central South Carolina into the study area. It has been paleontologically correlated with the early and middle Eocene Tallahatta Formation in neighboring southeastern Georgia.

The Congaree is about 30 ft (9 m) thick near the center of the study area and consists of yellow, orange, tan, gray, green, and greenish gray, well-sorted, fine to coarse quartz sand, with granule and small pebble zones common. Thin clay laminae occur throughout the section. The quartz grains tend to be better rounded than those in the rest of the stratigraphic column are. The sand is glauconitic in places suggesting deposition in shoreline or shallow shelf environments. To the south, near well ALL-324, the Congaree Formation consists of interbedded glauconitic sand and shale, grading to glauconitic, argillaceous, fossiliferous sandy limestone suggestive of shallow to deeper shelf environments of deposition. Farther south, beyond well C-10, the Congaree grades into platform carbonate facies of the lower Santee Limestone.

The equivalent of the Congaree northwest of SRS has been mapped as the Huber Formation. At these locations it becomes more micaceous and poorly sorted, indicating deposition in fluvial and upper delta plain environments. On geophysical logs, the Congaree has a distinctive low gamma ray count and high electrical resistivity.

Warley Hill Formation

Unconformably overlying the Congaree Formation are 10 ft (3 m) to 20 ft (6 m) of fine-grained, often glauconitic sand and green clay beds that have been referred to respectively as the Warley

Hill and Caw Caw Members of the Santee Limestone. The green sand and clay beds are referred to informally as the “green clay” in previous SRS reports. Both the glauconitic sand and the clay at the top of the Congaree are assigned to the Warley Hill Formation. In the updip parts of the study area, the Warley Hill apparently is missing or very thin, and the overlying Tinker/Santee Formation rests unconformably on the Congaree Formation.

The Warley Hill sediments indicate shallow to deeper clastic shelf environments of deposition in the study area, representing deeper water than the underlying Congaree Formation. This suggests a continuation of a transgressive pulse during upper middle Eocene time. To the south, beyond well P-21, the green silty sand, and clay of the Warley Hill undergo a facies change to the clayey micritic limestone and limey clay typical of the overlying Santee Limestone. The Warley Hill blends imperceptibly into a thick clayey micritic limestone that divides the Floridan Aquifer System south of the study area. The Warley Hill is correlative with the lower part of the Avon Park Limestone in southern Georgia and the lower part of the Lisbon Formation in western Georgia. In the study area, the thickness of the Warley Hill Formation is generally less than 20 ft (6 m). In a part of Bamberg County, South Carolina, the Congaree Formation is not present, and the Warley Hill rests directly on the Williamsburg Formation.

Tinker/Santee Formation

The late middle Eocene deposits overlying the Warley Hill Formation consist of moderately sorted yellow and tan sand, calcareous sand and clay, limestone, and marl. Calcareous sediments dominate downdip, are sporadic in the middle of the study area, and are missing in the northwest. The limestone represents the farthest advance to the northwest of the transgressing carbonate platform first developed in early Paleocene time near the South Carolina and Georgia coasts.

The Santee is divided into three members in the study area: the McBean, Blue Bluff, and Tims Branch Members. The McBean Member consists of tan to white, calcilutite, calcarenite, shelly limestone, and calcareous sand and clay. It dominates the Santee in the central part of the study area and represents the transitional lithologies between clastics in the north and northwest (Tims Branch Member), and fine-grained carbonates in the south (Blue Bluff Member).

The carbonates and carbonate-rich clastics are restricted essentially to three horizons in the central part of the Griffins Landing Member of the Dry Branch Formation, the McBean Member of the Tinker/Santee Formation, and the Utley Limestone member of the Clinchfield Formation. The uppermost horizon includes the carbonates of the Griffins Landing Member of the Dry Branch Formation found below the “tan clay” interval that occurs near the middle of the Dry Branch. The isolated carbonate patches of the Griffins Landing are the oyster banks that formed in the back barrier marsh zone behind the barrier island system. Underlying the Dry Branch, directly below the regionally significant Santee Unconformity, is the Utley Limestone Member of the Clinchfield Formation. Without the benefit of detailed petrographic and paleontological analysis, the Utley carbonates cannot be systematically distinguished from the carbonates of the underlying Tinker/Santee Formation. Thus, the carbonate-rich sediments between the Santee Unconformity and the Warley Hill Formation are referred to as the Tinker/Santee (Utley) sequence in this report.

Approximately 40 to 50% of the wells that drilled through the Tinker/Santee (Utley) interval in the General Separation Area (GSA) penetrated quantities of carbonate ranging from 5 to 78% of the sediment sampled. The calcareous sediment in the GSA consists of calcareous sand, calcareous mud, sandy limestone, muddy limestone, and sandy muddy limestone. Viewing the Tinker/Santee (Utley) sedimentary package parallel to the shoreline, the carbonate-rich sediments would be concentrated in the areas furthest removed from the tidal inlets at the shore face where clastic sediments supplied by riverine input is concentrated. The clastic-rich on the other hand would concentrate opposite the tidal inlet areas where clastic sediment is more readily available. The lateral facies transition of the sediments in the subtidal shelf environment from carbonate-rich to clastic-rich lithologies is therefore gradual and measures in the thousands of feet. Shifting locations of the tidal inlets at the shoreline has resulted in a complex sedimentary package where facies gradually transition from one lithology to another both laterally and vertically.

The GSA is in that part of the mixed clastics/carbonate zone where the clastic sediments generally constitute a greater percentage of the section than the carbonates. In northern SRS, the Tinker/Santee (Utley) sediments are mostly sands and muddy sands (Tims Branch Member) deposited in shoreline to lesser lagoonal and tidal marsh environments. In the central SRS, the sequence was deposited in middle marine shelf environments resulting in a varied mix of lithologies from carbonate-rich sands and muds to sandy and muddy limestones. In southern SRS, the Tinker/Santee (Utley) sediments were deposited further offshore, further removed from riverine clastic input into the shelf environment resulting in deposition of carbonate muds (Blue Bluff Member).

The Blue Bluff Member consists of gray to green, laminated micritic limestone. The unit includes gray, fissile, calcareous clay and clayey micritic limestone and very thinly layered to laminated, clayey, calcareous, silty, fine sand, with shells and hard, calcareous nodules, lenses, and layers. Cores of Blue Bluff sediments are glauconitic, up to 30% in places. The Blue Bluff lithology suggests deposition in offshore shelf environments. Blue Bluff sediments tend to dominate the formation in the southern part of the study area and constitute the major part of the "middle confining unit" that separates the Upper and Lower Floridan aquifers south of the study area.

The Tims Branch Member of the Santee is described as the siliciclastic part of the unit, consisting of fine- and medium-grained, tan, orange, and yellow, poorly to well sorted, and slightly to moderately indurated sand. The clastic lithologies of the Tims Branch Member dominate the Santee in the northern part of the study area. Because the clastic lithologies differ so markedly from the type Santee, the Tims Branch Member of the Santee has been raised to formational rank, namely the Tinker Formation. Because the clastic and carbonate lithologies that constitute the Tinker/Santee sequence in the upper and middle parts of the study area are hydrologically undifferentiated, the units are not systematically separated, and they are designated Tinker/Santee Formation on maps and sections. The thickness of the Tinker/Santee Formation is variable due in part to displacement of the sediments, but more commonly to dissolution of the carbonate resulting in consolidation of the interval and slumping of the overlying sediments of the Tobacco Road and Dry Branch Formations into the resulting lows.

The Tinker/Santee (Utley) interval is about 70 ft (21 m) thick near the center of SRS, and the sediments indicate deposition in shallow marine environments. The top of the unit is picked on geophysical logs where Tinker/Santee (Utley) sediments with lower electrical resistivity are overlain by the more resistive sediments of the Dry Branch Formation. In general, the gamma-ray count is higher than in surrounding stratigraphic units.

Often found within the Tinker/Santee (Utley) sediments, particularly in the upper third of the interval, are weak zones interspersed in stronger carbonate-rich matrix materials, referred to as "soft zones," which are described in Section 1.3.5.1.5.5.

Barnwell Group

Upper Eocene sediments of the Barnwell Group represent the Upper Coastal Plain of western South Carolina and eastern Georgia. Sediments of the Barnwell Group are chronostratigraphically equivalent to the lower Cooper Group (late Eocene).

The Cooper Group includes sediments of both late Eocene and early Oligocene age and appears downdip in the Lower Coastal Plain of eastern South Carolina.

Sediments of the Barnwell Group overlie the Tinker/Santee Formation and consist mostly of shallow marine quartz sand containing sporadic clay layers. The upper Eocene stratigraphy of the Georgia Coastal Plain has recently been revised and extended into South Carolina. The Eocene "Barnwell Formation" has been elevated to the "Barnwell Group." In Burke County, Georgia, the group includes (from oldest to youngest) the Clinchfield Formation, Dry Branch Formation, and the Tobacco Road Formation. The group is about 70 ft (21 m) thick near the northwestern boundary of SRS and 170 ft (52 m) near its southeastern boundary. The regionally significant Santee Unconformity that defines of boundary between Sequence Stratigraphic units II and III separates the Clinchfield Formation from the overlying Dry Branch Formation. The Santee Unconformity is a pronounced erosional surface observable throughout the SRS region.

In the northern part of the study area, the Barnwell Group consists of red or brown, fine to coarse-grained, well-sorted, massive sandy clay and clayey sand, calcareous sand and clay, as well as scattered thin layers of silicified fossiliferous limestone. All are suggestive of lower delta plain and/or shallow shelf environments. Downdip, the Barnwell undergoes a facies change to the phosphatic clayey limestone that constitutes the lower Cooper Group. The lower Cooper Group limestone beds indicate deeper shelf environments.

<u>Clinchfield Formation</u>

The basal late Eocene Clinchfield Formation consists of light colored quartz sand and glauconitic, biomoldic limestone, calcareous sand, and clay. Sand beds of the formation constitute the Riggins Mill Member of the Clinchfield Formation and are composed of medium to coarse, poorly to well sorted, loose and slightly indurated, tan, clay, and green quartz. The sand is difficult to identify unless it occurs between the overlying carbonate layers of the Griffins Landing Member and the underlying carbonate layers of the Santee Limestone. The Clinchfield is about 25 ft (8 m) thick in the southeastern part of SRS and pinches out or becomes unrecognizable at the center of the site.

The carbonate sequence of the Clinchfield Formation is designated the Utley Limestone Member. It is composed of sandy, glauconitic limestone and calcareous sand, with an indurated, biomoldic facies developed in places. In cores, the sediments are tan and white and slightly to well indurated. Without the benefit of detailed petrographic and paleontological analysis, the Utley carbonates cannot be systematically distinguished from the carbonates of the underlying Tinker/Santee Formation. Thus, the carbonate-rich sediments between the Santee Unconformity and the Warley Hill Formation are referred to as the Tinker/Santee (Utley) sequence in this report.

Dry Branch Formation

The late Eocene Dry Branch Formation is divided into the Irwinton Sand Member, the Twiggs Clay Member, and the Griffins Landing Member. The unit is about 60 ft (18 m) thick near the center of the study area. The top of the Dry Branch is picked on geophysical logs where a low gamma-ray count in the relatively clean Dry Branch sand increases sharply in the more argillaceous sediments of the overlying Tobacco Road Sand.

The Dry Branch sediments overlying the Tinker/Santee (Utley) interval in the central portion of SRS were deposited in shoreline/lagoonal/tidal marsh environments. The shoreline retreated from its position in northern SRS during Tinker/Santee (Utley) time to the central part of SRS in Dry Branch time. Progradation of the shoreline environments to the south resulted in the sands and muddy sands of the Dry Branch being deposited over the shelf carbonates and clastics of the Tinker/Santee (Utley) sequence.

The Twiggs Clay Member does not seem to be mappable in the study area. Lithologically similar clay is present at various stratigraphic levels in the Dry Branch Formation. The tan, light-gray, and brown clay is as thick as 12 ft (4 m) in SRS wells but is not continuous over long distances. This has been referred to in the past as the “tan clay” in SRS reports. The Twiggs Clay Member, which predominates west of the Ocmulgee River in Georgia, is not observed as a separate unit in the study area.

The Griffins Landing Member is composed mostly of tan or green, slightly to well indurated, quartzose calcareous micrite and sparite, calcareous quartz sand and slightly calcareous clay. Oyster beds are common in the sparry carbonate facies. The unit seems to be widespread in the southeastern part of SRS, where it is about 50 ft (15 m) thick, but becomes sporadic in the center and pinches out. Carbonate content is highly variable. In places, the unit lies unconformably on the Utley Limestone Member, which contains much more indurated, moldic limestone. In other areas, it lies on the noncalcareous quartz sand of the Clinchfield. Updip, the underlying Clinchfield is difficult to identify or is missing, and the unit may lie unconformably on the sand and clay facies of the Tinker/Santee Formation. The Griffins Landing Member appears to have formed in lagoonal/marsh environments.

The Irwinton Sand Member is composed of tan, yellow and orange, moderately sorted quartz sand, with interlaminated and interbedded clay abundant in places. Pebbly layers are present, as are clay clast-rich zones (Twiggs Clay lithology). Clay beds, which are not continuous over long distances, are tan, light gray, and brown in color, and can be several feet thick in places. These are the “tan clay” beds of various SRS reports. Irwinton Sand beds have the characteristics of shoreline to shallow marine sediments. The Irwinton Sand crops out in SRS. Thickness is variable, but is about 40 ft (12 m) near the northwestern site boundary and 70 ft (21 m) near the southeastern boundary.

Tobacco Road Formation

The Late Eocene Tobacco Road Formation consists of moderately to poorly sorted, red, brown, tan, purple, and orange, fine to coarse, clayey quartz sand. Pebble layers are common, as are clay laminae and beds. Ophiomorpha burrows are abundant in parts of the formation. Sediments have the characteristics of lower Delta plain to shallow marine deposits. The top of the Tobacco Road is characterized by the change from a comparatively well-sorted sand to the more poorly sorted sand, pebbly sand, and clay of the "Upland unit." Contact between the units constitutes the "Upland" unconformity. The unconformity is very irregular due to fluvial incision that accompanied deposition of the overlying "Upland unit" and later erosion. As stated previously, the lower part of the Cooper Group (upper Eocene) is the probable downdip equivalent of the Tobacco Road Formation.

"Upland Unit"/Hawthorn/Chandler Bridge Formations

Deposits of poorly sorted silty, clayey sand, pebbly sand, and conglomerate of the "Upland unit" cap many of the hills at higher elevations over much of the study area. Weathered feldspar is abundant in places. The color is variable, and facies changes are abrupt. These sediments are assigned to the Hawthorn Formation. It has been mapped as the "Upland unit," with evidence for a Miocene age. The unit is up to 60 ft (18 m) thick. The environment of deposition appears to be fluvial, and the thickness changes abruptly owing to channeling of the underlying Tobacco Road Formation during "Upland" deposition and subsequent erosion of the "Upland" unit itself. This erosion formed the "Upland" unconformity. The unit is up to 60 ft (18 m) thick.

Lithologic types comparable to the "Upland" unit but assigned to the Hawthorn Formation overlie the Barnwell Group and the Cooper Group in the southern part of the study area. In this area, the Hawthorn Formation consists of very poorly sorted, sandy clay, and clayey sand, with lenses of gravel and thin beds of sand very similar to the "Upland unit." Farther downdip, the Hawthorn overlies the equivalent of the Suwanee Limestone and acts as the confining layer overlying the Floridan Aquifer System. It consists of phosphatic, sandy clay and phosphatic, clayey sand and sandy, dolomitic limestone interbedded with layers of hard, brittle clay resembling stratified fuller's earth.

It has been suggested that the "Upland unit," Tobacco Road Formation, and Dry Branch Formation are similar in granularity and composition, indicating that they might be part of the same transgressive/regressive depositional cycle. The "Upland unit" represents the most continental end member (lithofacies) and the Dry Branch Formation represents the most marine end member. Thus, the "Upland unit" is the result of a major regressive pulse that closed out deposition of the Barnwell Group/Cooper Group depositional cycle. It has also been suggested that the "Upland unit" is correlative with the Chandler Bridge Formation downdip toward the coast. This hypothesis is significant because it implies that there was no major hiatus between the "Upland unit" and the underlying Tobacco Road and Dry Branch Formations. The existence of a hiatus between the units has been reported by numerous studies of the South Carolina Coastal Plain.

1.3.5.1.5.4 Quaternary Surfaces and Deposits

Determining fault capability requires assessing the potential for Quaternary (1.6 - 0.01 Ma) deformation. The Quaternary and neotectonic studies conducted at SRS during 1991-1992 were designed to span the geologic record between deposition of the "Upland unit" and the present, and to determine if deformation has affected Quaternary-age deposits or surfaces. The Quaternary record in the SRS area is preserved primarily in fluvial terraces along the Savannah River and its major tributaries and in deposits of colluvium, alluvium, and eolian sediments on upland interfluve areas.

SRS lies within the interfluve area between the Savannah and the Salkahatchie Rivers. The drainage systems within the site consist entirely of streams that are tributary to the Savannah River. A series of nested fluvial terraces are preserved along the river and major tributaries. Fluvial terraces are the primary geomorphic surface that can be used to evaluate Quaternary deformation within SRS. However, there is limited data available for the estimation of ages of river terraces in both the Atlantic and Gulf Coastal Plains.

Major stream terraces form by sequential erosional and depositional events in response to tectonism, isostasy, and climate variation. Streams respond to uplift by cutting down into the underlying substrate in order to achieve a smooth longitudinal profile that grades to the regional base level. Aggradation or deposition occurs when down-cutting is reversed by a rise in base level. The stream channel is elevated and isolated from the underlying marine strata by layers of newly deposited fluvial sediments. Down-cutting may resume and the aggraded surface is abandoned. The result is a landform referred to as a fill terrace.

At the SRS there are two prominent terraces above the modern floodplain. These designations are based on morphology and relative height above local base level. Local base level is the present elevation of the Savannah River channel. In addition, there are other minor terraces: one lower and several higher, older terrace remnants.

The terraces of Upper Three Runs and Steel Creek were mapped on false color, infrared aerial photography, and field checked. Although exposures of fluvial deposits are extremely limited, these terraces are laterally continuous. Upper Three Runs terraces are of interest to SRS because of their position over the Atta and Upper Three Runs faults. The terraces along Steel Creek represent a family of seven sets of well-defined fluvial terraces, one of the best sequences of terraces at SRS. These terraces range from less than 3 to 100 ft (1 to 30 m) above local base level. The lower terraces appear to be fill terraces whereas the higher terraces appear to be strath terraces that cut into Tertiary strata. The Steel Creek drainage parallels the trace of the subsurface Steel Creek fault.

Estimated ages of the terraces are based on several techniques including radiometric carbon-14 dates, soil chronosequences, relative position above base level and correlation to other dated river or marine terraces. The modern floodplain is as old as the latest Pleistocene to Holocene. Others have indicated a much younger age of 4,000 years. Based on soil chronosequences, it is at least 400 ka to perhaps 1 Ma. An early to middle Holocene age (less than 10 ka) has been concluded based on geoarchaeological studies. The terraces on Upper Three Runs range from 11 ka for the lower (1.6 to 14.8 ft [0.5 to 4.5 m]) terrace to 38 to 47 ka for the higher (greater than 30 ft [6 m]) terrace. Overall, the terraces at SRS represent ages from middle Holocene (less than 10 ka) to late Pleistocene (1 Ma).

1.3.5.1.5.5 Carolina Bays

Carolina bays are shallow, elliptical depressions with associated sand rims that are found on the surface of the Coastal Plain sediments. They are found from southern New Jersey to northern Florida with the greatest occurrence in the Carolinas. One hundred ninety-seven confirmed or suspected Carolina bays have been identified at SRS. The long axes of the bays are oriented S50°E and the sand rims are observed on the east and southeast flanks. Several hypotheses have been provided for the timing and mode of origin for these bays. Theories regarding the origin of bays include meteorite impact, sinks, wind, and water currents. The origin of these features continues to be studied.

The most likely explanation of formation suggests that the bays were formed by action of strong unidirectional wind on water ponded in surface depressions. The resulting waves caused the formation of the sand rims as shoreline features, and the sand rims formed perpendicular to the wind direction. Therefore, the wind that formed the bays observed today was a southwesterly wind.

The Carolina bays are surficial features that have no effect on the subsurface sediments. Based on subsurface core data, it has been demonstrated that a clay layer mapped beneath the bays and beyond had no greater relief beneath the bays than beyond them. Certain identified strata can be mapped and found continuous and undeformed beneath bay and interbay areas. In Horry and Marion Counties, South Carolina, there is no evidence of solution-related subsidence of the Carolina bays, in spite of the presence of carbonate-rich strata in the subsurface and some localized sink holes of irregular shape with depths on the order of 20 ft (6 m).

The minimum age of the bays is set at middle to late Wisconsinian based on radiocarbon dating. The maximum age can be relatively determined by examination of the formations on which the bays rest. If one assumes a single generation of formation for all bays, then the bays formed after deposition of the Socastee Formation and before the Wando Formation. This places bay formation between 100 and 200 ka. If there is more than one generation, then the bays could be as old as the formations on which they rest.

Carbonate and Soft Zones

Often found within the Tinker/Santee (Utley) sediments, particularly in the upper third of this section, are weak zones interspersed in stronger carbonate-rich matrix materials. These weak zones, which vary in apparent thickness and lateral extent, were recorded where rod drops and/or lost circulation occurred during drilling, low blow counts occurred during standard penetration tests, and so on. They have variously been termed as “soft zones,” “the critical layer,” “underconsolidated zones,” “bad ground,” and “void.” The preferred term used to describe these zones is “soft zones.”

The initial U.S. Corps of Engineer (COE) characterization in 1952 (COE 1952) identified soft zones as being the major concern for foundation design. This initial study made many important observations concerning the formation, geometry, distribution, and physical attributes of soft zones (and potential associated voids) within the Santee Formation. Some of the soft zone observations and hypotheses set forth by the COE report have remained unchanged to this day. However, several important aspects of early soft zone analyses run counter to current thinking on this subject.

Historically, the soft zones were grouted as an expedient way of resolving any potential foundation stability issues. This method continued through the restart of K Reactor where the project chose to grout the Santee Formation beneath the cooling water lines to resolve a potential foundation stability issue. The results of that effort were carefully studied and it was found that the grout was not having the desired effect on the subsurface soft zones.

More recently, technology improvements have allowed sampling and testing which have resulted in additional insight to the properties of the soft zone soils. With these properties, advanced analytical techniques have been used to resolve the foundation stability issues without requiring soil remediation. The information provided herein allows for a clearer understanding of the geologic underpinnings that established the carbonates and the attendant soft zones.

In general, where carbonates are found soft zones are likely to be found as well. This conclusion is based on a significant study of soil samples from borings, boring logs, geophysical logs, and CPT soundings throughout the General Separation Area (GSA). This review was instrumental in delineating the extent of both carbonates and soft zones. The data were studied in many different ways but resulted in the simple conclusion that although carbonates and soft zones are not found in every drill hole or cone penetrometer test (CPT), they are generally found in every area that was investigated in the GSA.

Isopach maps reveal that carbonate thickness and concentration is directly related to the isopach thickness of the Tinker/Santee (Utley) interval. Where the Santee-Utley interval is thick, carbonate is more concentrated, where the interval is thin, carbonate thickness and concentration is reduced. It is further observed that where carbonate is concentrated in the Santee-Utley section the overlying "Upland unit," Tobacco Road/Dry Branch section is generally structurally high, and where the carbonate content is reduced or absent the overlying "Upland unit," Tobacco Road/Dry Branch section is generally structurally low. This indicates that the removal (dissolution) of carbonate and the thinning of the Santee-Utley interval occurred in post Tobacco Road time. Since the thickness and distribution of soft zones is closely linked to the thickness and distribution of carbonate, those areas where clastic sediments were initially concentrated and in structurally low areas where a great deal of carbonate has been removed would be areas where soft zones may not be present.

<u>Origin of Carbonates and Soft Zones</u>

The origin of the carbonates in the Tinker/Santee (Utley) interval is fairly clear. The carbonate content ranges from zero to approximately 90%. The presence of glauconite along with a normal marine fauna including foraminifers, molluscs, bryozoans, and echinoderms, indicates that the limestones and limy sandstones were deposited in clear, open-marine water of normal salinity on the inner to middle shelf. The abundance of carbonate mud (micrite) in the limestones suggests deposition in quiet water below normal marine wave base. The presence of abraded and well-worn skeletal grains indicates that bottom transport by currents or storm-generated waves alternated with quiet-water conditions in which the sediments accumulated.

Viewing the Santee sedimentary package parallel to the shoreline, the carbonate-rich sediments would be concentrated in the areas furthest removed from the tidal inlets at the shore face where clastic sediments supplied by riverine input is concentrated. The clastic-rich sediments on the other hand would concentrate opposite the tidal inlet areas where clastic sediment is more readily available. The lateral facies transition of the sediments in the subtidal shelf environment from carbonate-rich to clastic-rich lithologies is therefore gradual and measures in the thousands of feet. Shifting locations of the tidal inlets at the shoreline has resulted in a complex sedimentary package where facies gradually transition from one lithology to another both laterally and vertically. Therefore, both vertical and lateral lithologic variability in the Tinker/Santee (Utley) sequence is the rule rather than the exception. Locally the contact between carbonate sediments and laterally comparable clastic sediments is often sharply drawn, occurring over distances of only a few feet.

The original thoughts were that the soft zones were the result of the dissolution of the shell debris concentrated in bioherms (oyster banks). This premise has since been proven to be false. Significant study of the deposition of the Tinker/Santee (Utley) sediments precludes the formation of bioherms. Several hypotheses exist concerning the origin of the soft zones: one being that these zones consisted of varying amounts of carbonate material that has undergone dissolution over geologic time leaving sediments that are now subjected to low vertical effective stresses due to arching of more competent soils above the soft zone intervals.

A second hypothesis is based on recent studies that indicate that soft zones occur where silica replacement/cementation of the carbonate occurred. The silicification (by amorphous opaline silica) of the enclosing carbonate sediment would follow and spread along bedding planes, along microfractures of varied orientations and along corridors of locally enhanced permeability. The resulting “soft zone” could be in the form of irregular isolated pods, extended thin ribbons or stacked thin ribbons separated by intervening unsilicified parent sediment. Careful observations of the grouting programs conducted by the COE in the early 1950s, and more recently for the restart of K Reactor, corroborate these recent findings.

Soft zones encountered in one CPT sounding could be absent in the neighboring CPT only a few feet away. Only where silicification has spread far enough away from the bedding planes and/or fractures along which the silica replacement has taken place, where all the intervening sediment is replaced, would the soft zones be large enough and coherent enough to pose a question for the siting of new facilities. In all likelihood, this would be a most uncommon event.

1.3.5.1.6 Regional Physiography

The site region, defined as the area within a 200 mi (322 km) radius of the center of SRS, includes parts of the Atlantic Coastal Plain, Piedmont, and Blue Ridge physiographic provinces. SRS is located on the upper Atlantic Coastal Plain, about 30 mi (50 km) southeast of the Fall Line.

The Atlantic Coastal Plain extends southward from Cape Cod, Massachusetts, to south central Georgia where it merges with the Gulf Coastal Plain. The surface of the Coastal Plain slopes gently seaward. The South Carolina Coastal Plain can be divided into three physiographic belts: Upper, Middle, and Lower Coastal Plain. The Upper Coastal Plain slopes from a maximum elevation of 650 ft (200 m) above msl at the Fall Line to about 250 ft (75 m) above msl on its southeastern boundary. Primary depositional topography of the Upper Coastal Plain has been obliterated by fluvial erosion. The Upper Coastal Plain is separated from the Middle Coastal Plain by the Orangeburg scarp, which has a relief of approximately 100 ft (30 m) over a distance of a few miles. The Orangeburg scarp is the locus of Eocene, Upper Miocene, and Pliocene shorelines. The Middle Coastal Plain, separated from the Lower Coastal Plain by the Surry scarp, is characterized by lower elevations and subtle depositional topography that has been significantly modified by fluvial erosion. The Lower Coastal Plain is dominated by primary depositional topography that has been modified slightly by fluvial erosion.

The Upper Coastal Plain of South Carolina is divided into the Aiken Plateau and Congaree Sand Hills. The Aiken Plateau, where SRS is located, is bounded by the Savannah and Congaree Rivers and extends from the Fall Line to the Orangeburg scarp. The plateau's highly dissected surface is characterized by broad interfluvial areas with narrow, steep-sided valleys. Local relief is as much as 295 ft (90 m). The plateau is generally well drained, although many poorly drained sinks and depressions exist, especially on the topographically high (above 250 ft [76 m] above msl) “Upland unit.” The Congaree Sand Hills trend along the Fall Line northeast and north of the Aiken Plateau. The sand hills are characterized by gentle slopes and rounded summits that are interrupted by valleys of southeast-flowing streams and their tributaries.

The site region contains Carolina bays. (Carolina bays are discussed in detail in the previous section.)

The Piedmont province extends southwest from New York to Alabama and lies adjacent to the Atlantic Coastal Plain. It is the eastern-most physiographic and structural province of the Appalachian Mountains. The Piedmont is a seaward-sloping plateau whose width varies from about 10 mi (16 km) in southeastern New York to almost 125 mi (200 km) in North Carolina; it is the least rugged of the Appalachian provinces. Elevation of the inland boundary ranges from about 200 ft (60 m) above msl in New Jersey to over 1,800 ft (550 m) above msl in Georgia.

The Blue Ridge province extends from Pennsylvania to northern Georgia. It varies from about 30 mi (48 km) to 75 mi (120 km) wide north to south. Elevations are highest in North Carolina and Georgia, with several peaks in North Carolina exceeding 5,900 ft (1,800 m) above msl. Mount Mitchell, North Carolina, is the highest point (6,560 ft [2,000 m]) above msl in the Appalachian Mountains. The Blue Ridge front, with a maximum elevation of 4,000 ft (1,200 m) above msl in North Carolina, is an east-facing escarpment between the Blue Ridge and Piedmont provinces in the southern Appalachians.

1.3.5.1.7 General Geologic Setting at Savannah River Site

The 25 mi (40 km) radius study area is taken from DOE-STD-1022-94 (DOE 1996c) as the area in which to conduct geoscience investigations to locate possible seismogenic sources and surface deformation or to demonstrate that such features do not exist.

SRS is located on the Atlantic Coastal Plain, which is an essentially flat-lying, undeformed wedge of unconsolidated marine and fluvial sediments. The sediments are stratified sand, clay, limestone, and gravel that dip gently seaward and range in age from Late Cretaceous to Holocene. The sedimentary sequence thickens from zero at the Fall Line to more than 4,000 ft (1,200 m) at the coast. The Coastal Plain section is divided into several rock-stratigraphic groups, based principally on age and lithology. The details of Coastal Plain stratigraphy have been discussed in the preceding section.

Beneath the Coastal Plain sedimentary sequence and below a pre-Cretaceous unconformity are two geologic terranes: (1) the Dunbarton basin, a Triassic-Jurassic Rift basin, filled with lithified terrigenous and lacustrine sediments with possible minor amounts of mafic volcanic and intrusive rock; and (2) a crystalline terrane of metamorphosed sedimentary and igneous rock that may range in age from Precambrian to late Paleozoic. The Paleozoic rocks and the Triassic sediments were leveled by erosion, forming the base for Coastal Plain sediment deposition. The erosional surface dips southeast approximately 42 ft/mi (8 m/km).

Information about the Dunbarton basement and crystalline terrane comes primarily from deep borings. The U.S. Army Corps of Engineers drilled a single hole into basement rock in 1950 for the startup of the plant. In 1961, the Bedrock Waste Storage Project rock exploration program was conducted to determine the feasibility of long-term storage of radioactive waste in mined rock chambers. Twelve deep rock borings, the Deep Rock Borings (DRB) well series, were completed into basement to various depths greater than 980 ft (300 m) to accomplish this goal. This information is also augmented by deep borings used to constrain seismic reflection information both in the early 1970s (P-R series) and more recently acquired information (MMP and GCB series).

In addition to the direct information furnished by the deep borings, information about the composition, extent and structure of crystalline terrane and the Dunbarton basin are also provided by potential field geophysical methods. Detailed gravity information concerning SRS and vicinity exists and has been used to provide a detailed gravity map of the site. In addition, high resolution aeromagnetic data are available from the USGS and have been used to produce a high resolution aeromagnetic map of SRS and vicinity. Several recent studies have been the focus on integrating this geophysical information with the boring information listed above to evolve a fairly detailed model of the crystalline terrane and Dunbarton Basin.

1.3.5.1.7.1 Crystalline Terrane

The studies mentioned above have determined that the lithologies and structures in the crystalline terrane are basically similar to that seen in the eastern Piedmont province as exposed in other parts of the southeastern United States. The crystalline rocks form a volcanic – intrusive sequence of calc-alkaline composition, portions of which record both ductile and brittle deformational events. These relationships indicate that these rocks are the metamorphosed and deformed remnants of an ancient volcanic arc that are interpreted to be Carolina terrane equivalents.

The crystalline rocks were mapped as three formations. The Crackerneck Formation consists of weakly to unmetamorphosed and mildly to undeformed volcanic rocks of intermediate to felsic composition with minor amounts of mafic material. The rocks in this formation are represented mainly by tuffs and lapilli tuffs (extrusive volcanic rocks).

The DRB Formation (named after the Deep Rock Borings in which it is found) consists of moderately metamorphosed and highly to moderately deformed volcanic and plutonic rocks of mafic to intermediate compositions. The DRB Formation is cut by deformed amphibolite dikes and by undeformed dikes of basaltic and rhyolitic compositions, indicating that these rocks were intruded both before deformation and after the major episode of deformation had ceased. The DRB Formation may also contain a minor amount of quartz-rich sedimentary rock. However, the identification of this material is uncertain.

The Pen Branch Formation (named after the Pen Branch fault borings in which it is found) occurs as a thin slice between the Dunbarton Basin to the south and the DRB Formation to the north. This formation contains strongly metamorphosed gneisses and amphibolites that have experienced relatively high thermal effects and appear to be deeper equivalents of the DRB Formation. The plutonic rocks of both the DRB Formation and Pen Branch Formation have radiometerically dated crystallization ages of 620 Ma. Based on the association of these rocks with the Carolina terrane, the metavolcanic rocks of the Crackerneck Formation are interpreted to have been deposited unconformably on the DRB Formation at about 620 Ma.

Subsequent to the formation of this volcanic stratigraphy, these rocks underwent multiple deformational episodes and chemical changes. The rocks of the DRB Formation record highly developed deformational fabrics that indicate that these rocks have undergone significant amounts of ductile shearing at moderately high temperatures. These fabrics, in association with the superposition and juxtaposition of the higher temperature Pen Branch Formation, indicate that this deformation resulted from thrust and strike-slip faulting, which placed the Pen Branch Formation over the DRB Formation. Based on radiometric age dating of biotite in the fault zone, this deformation is Paleozoic in age (approximately 300 Ma). In addition to ductile deformation features, the sub-Cretaceous basement rocks also record the effects of brittle deformation episodes characterized by fractures, brittle faults, and frictional melting. The presence of mineralized veins associated with these fractures and brittle faults indicate that the brittle faulting was often accompanied by the movement of hot waters. Radiometric dating of these effects suggest that at least one phase of brittle deformation occurred around 220 Ma. This age would make this phase of brittle deformation most likely associated with formation of the Dunbarton basin. Other younger brittle deformation features are also present, and are most likely associated with Tertiary deformation in the basement such as the Pen Branch fault. Radiometric dating of fracture filling yielded an age of 23 Ma. However, the radiometric systematics of the mineral dated is not well known so the geologic meaning of this age is uncertain.

1.3.5.1.7.2 Dunbarton Triassic Rift Basin

The Dunbarton basin underlies the southeastern portion of SRS and was first identified based on aeromagnetic and well data. Subsequent seismic reflection surveys, potential field surveys, and additional well data have led to the current understanding of the basin. The structure is currently interpreted as an asymmetric graben approximately 30 mi (50 km) long and 6 to 9 mi (10 to 15 km) wide. The axis of the basin strikes north 63° east, which is parallel to the regional strike of crystalline basement. The basin extends 5 mi (8 km) southwest of the Savannah River and

24 mi (40 km) to the northeast of SRS, where it terminates against a granite body interpreted from magnetic data. The master border fault, named the Pen Branch fault, is on the northwest boundary of the basin and dips to the southeast.

The southeast boundary of the basin is poorly constrained but is interpreted as a fault. Southeast of the Dunbarton basin aeromagnetic and gravity data indicate a terrane heavily influenced by basalt flows and sills. The magnetic data contain numerous high-frequency, closed-contour features indicative of shallow structures, and lower frequency features indicative of deeper-seated features. The host rock is perhaps crystalline metamorphosed rock similar to what is found further to the northwest beneath SRS. It is suggested that this terrane separates the Piedmont orogeny from crust of a different affinity further to the southeast. In effect, the mafic intrusions define the southeastern boundary of the Dunbarton basin and the northern boundary of the South Georgia Rift basin.

Ten wells drilled in the southeastern half of SRS penetrated sedimentary rocks of the Dunbarton basin. Recovered core is clastic rock. Conglomerate, fanglomerate, sandstone, siltstone, and mudstone are the dominant lithologies. These rocks are similar to the clastic facies in other Newark Supergroup basins. In addition, four of the Pen Branch fault series wells penetrated Triassic rock. Conglomerate and red clayey siltstone are the dominant lithologies in these cores. The lithology and stratigraphy identified in these cores indicate that the proximal side of the basin is to the northwest. There is a larger component of coarse-grained rock types on the proximal side than on the southeast side of the basin. An upward increase of total fines is found in each core. Further, the sediments fine upward in each core. A detailed study of the Dunbarton Basin core that integrated the above observations with some new information grouped the sediments in the basin into four lithofaces:

1. A proximal fan facies occurs near the hanging wall of the Pen Branch fault and consists mainly of poorly sorted, matrix-supported conglomerates dominated by debris flows.

2. A distal fan facies includes silty and sandy mudstones interbedded with massive immature sandstones and wackes.

3. A fringe fan facies which is dominated by mudstones but also contains intervals with bioturbation, roots, and caliches, which indicate periods of flooding overprinted during periods of nondeposition by burrowing and soil formation.

4. A braided plain facies includes cross-stratified channel sandstones erbedded with bioturbated mudstones and fine sandstones containing caliches.

The facies relationships described above suggest an asymmetric basin that subsided faster to the northwest than to the southeast. The asymmetry led to greater local relief along the northern boundary, where high-energy fluvial processes dominated, and the resulting sediments were more coarse grained than farther out in the basin. The predominance of alluvial fan facies with abundant mud and debris flows, and caliches in paleosols suggests that the basin and surrounding areas were poorly vegetated, and an arid to semi-arid climate.

Gravity and magnetic modeling suggests that the Triassic section in the Dunbarton basin averages about 1.2 mi (2 km) thick. Boreholes have encountered up to 3,000 ft (899 m) of

Triassic fill, but the base of the Dunbarton was not encountered. Seismic reflection data do not unequivocally constrain the base of the basin, as the transition between the Triassic rock and the crystalline terrane is unclear. However, interpreted Triassic reflectors are at least as deep as 3,900 ft (1,188 m) to 12,100 ft (3,688 m).

1.3.5.1.8 Site Geologic Map

A geologic map of the SRS was completed by the USGS and provided to SRS in 1994. This map shows the Coastal Plain formations that crop out at the surface. Other, deeper Coastal Plain formations may not be observed at the surface within the boundaries of the site; however, these formations are known to exist in the subsurface based on drill core data and outcrops in nearby regions.

Erosion by the Savannah and Edisto Rivers and tributaries has truncated the uppermost stratigraphic units such as the "Upland unit" and the Tobacco Road Sand. This gives the geologic map its characteristic dendritic pattern and indicates that the strata are sub-horizontal. Deeper and older formations are exposed in stream valley walls Paleocene and Cretaceous formations crop out in nearby regions.

Superposed on the Coastal Plain sediments are a variety of alluvial and colluvial deposits that have resulted from streams cutting the valleys they occupy. The alluvial deposits are located in the stream valleys and on terraces and are indicated on the map as Qal 1, Qal 2, and Qt. The reworked sediments are derived from the uppermost Coastal Plain sediments and effectively cover up the deepest formations exposed in the stream valley bottoms.

Contacts separating the geological formations were mapped by examination of natural and manmade surface exposures and from subsurface drill core. Original compilation of field data was done at 1:100,000 scale. The subsequent SRS map is presented at 1:48,000 scale.

1.3.5.2 MFFF Site Geology

In calendar year 2000, 13 exploration borings and 63 CPT holes were used to define site-specific subsurface conditions at the MFFF site. Additional site geotechnical programs previously performed by others adjacent to and on this site were also used to evaluate site subsurface geologic and groundwater conditions. A detailed description of subsurface conditions encountered and previous SRS geotechnical references used for this investigation are described in detail in the *MOX Fuel Fabrication Facility Site Geotechnical Report* (DCS 2000). Exploration boring logs, CPT logs, and initial soil classification test results are also presented in this report. The location of exploration borings and CPT holes used to investigate the MFFF site are shown on Figure 1.3.5-22.

Information available from previous subsurface investigations was instrumental in development of the MFFF geotechnical field exploration program. Results of geotechnical exploration for the Actinide Packaging and Storage Facility (APSF), located adjacent to and south of the MFFF site, revealed the presence of subsurface soft zones. The *F-Area Northeast Expansion Report* (WSRC 1999b) contains results of additional explorations performed in the same vicinity, including the MFFF site area, that indicate that subsurface conditions at the MFFF site are similar to those previously encountered at APSF and nearby areas.

Initial layout of the CPT program for the MFFF site was patterned after the CPT layout that ultimately proved successful in adequately locating potential soft soil zones at the APSF site. As soft zones were encountered on the MOX site during field explorations, additional CPT and exploration holes were added to the plan to identify and delineate the extent of the soft zones found. The resulting CPT and exploration hole spacing, when combined with ones from previous explorations in the same area, was of greater concentration than was initially deemed necessary. The resulting data collection was found to be quite sufficient to identify potential loose soil zones that may be subject to liquefaction (see Section 1.3.7), as well as the soft zones present. Once the location and extent of the soft zones on the MFFF site were identified, the MFFF IROFS structures, such as the MOX Fuel Fabrication Building and the Emergency Diesel Generator Building were relocated to areas of the site found to be free of soft zones.

The approach for the layout of CPTs and exploration borings at the MFFF site provides confidence that soft and loose soil zones have been effectively identified in the vicinity of MFFF IROFS structures. Exploration spacing in the original geotechnical investigation was greater than desired because the drilling and CPT rigs could not access locations on the existing APSF spoils pile berm slopes. Grading of the slopes was performed in the summer of 2001 so that rig access could be provided to additional exploration hole locations. During the summer of 2002, MOX Services conducted a supplemental geotechnical investigation to acquire additional subsurface information to provide increased confidence that the size and extent of soft zones beneath the MFFF IROFS structures are adequately characterized. The results of these supplemental investigations are consistent with the results obtained during the initial site investigations, and they are described in detail in *MOX Fuel Fabrication Facility Site Geotechnical Report* (DCS 2005).

The CPT holes extended from approximately 64 to 140 ft (19.5 to 42.7 m) below present site grade. Each CPT hole provided a continuous profile of the soil conditions encountered at each test location. Seismic, resistivity, and piezometric measurements were obtained in many of the CPT holes. Some soft soil zones related to past solution and deposition activity were identified at depth on the MFFF site. The soft zones encountered were typical to those that have been described in previous F-Area investigations. The CPT holes were used to define limits of the soft zones. MFFF IROFS structures, the MOX Fuel Fabrication Building, and the Emergency Diesel Generator Building were adjusted on the MFFF site so that they are not directly over any identified thick soft zones and to minimize the potential impact of the underlying soft zones. Both static and dynamic analyses have been performed to evaluate the effect of soft zones near MFFF IROFS structures. The location of facilities at the MFFF site is shown on Figure 1.3.5-22 and on Figure 4-1 of the *MOX Fuel Fabrication Facility Site Geotechnical Report* (DCS 2005).

Subsurface soils at the MFFF site have also been evaluated to determine whether they have any potential for liquefaction during the design earthquake event. The potential for liquefaction has been determined using the established groundwater levels for the MFFF site, laboratory, geophysical, and CPT results, and blow count data from exploration borings.

Recognized industry practice methods used to define and determine the potential for liquefaction have been utilized. The design earthquake has been used to establish the potential for liquefaction. See Section 1.3.7.1 for a description of these evaluations.

The soil exploration borings extend from approximately 131 to 181 ft (39.1 to 55.2 m) below the present site grade. The exploration borings were used for correlation with the CPT holes and to obtain soil samples for laboratory testing. Three cased holes (exploration borings BH-2, BH-5, and BH-10) from the exploration program were used for downhole seismic testing.

The exploration borings and CPT holes indicate that subsurface conditions encountered at the MFFF site are consistent with previous investigations performed at SRS in F Area, at and near the site. No unusual subsurface geological or groundwater hydrologic conditions were encountered. Representative geotechnical cross sections at the MFFF site are shown on Figure 1.3.5-23, Figure 1.3.5-24, and Figure 1.3.5-25 and on Figures 5-1 through 5-7 of the *MOX Fuel Fabrication Facility Site Geotechnical Report* (DCS 2005).

The upper geologic units at the MFFF site are composed of the Barnwell Group. The exploration borings also extended through the Tinker/Santee Formation, Warley Hill Formation, and into the very dense Congaree Formation of the Orangeburg Group. See Table 1.3.5-1 for the correlation of geologic units and engineering units presently being used for geotechnical investigations at SRS. This correlation has been adopted for this geotechnical program, to be consistent with other SRS references being used for the MFFF site. The engineering units shown on the representative geotechnical cross sections are consistent with the correlation shown on Table 1.3.5-1 and the geologic units discussed in this section and the referenced reports.

The upper groundwater level is within the Upper Three Runs aquifer, as described in Section 1.3.4. Based on the results of pore water pressure dissipation testing, the groundwater level at the MFFF site was generally encountered at a depth of 60 ft (21 m) or more below grade at the time of the site exploration program. This groundwater level is expected to fluctuate seasonally.

A comprehensive laboratory testing program has been conducted to establish both static and dynamic design parameters for use in analysis. Laboratory results also indicate that the subsurface geologic units and soil properties at the MFFF site are consistent with those identified in previous investigations in F Area. The same geologic units described for SRS and F Area are found at the MFFF site.

The exploration borings, CPT holes, geophysical test results, and laboratory test results have been used to establish static and dynamic geotechnical design criteria. The geotechnical design criteria have been developed for each representative geologic unit, using the latest standard of practice for geotechnical engineering. The geotechnical design criteria have been correlated with available geotechnical design criteria developed for F Area and other relevant geologic units at SRS to confirm consistency.

1.3.5.3 Tectonic Features

1.3.5.3.1 Definition of Plate Tectonics

Plate tectonics within the 200 mi (322 km) radius of SRS provides the description of the major structural or deformational features of the region, as well as the origins, evolution, and interrelationship of these features. The implementation of natural phenomena hazards mitigation requires that the tectonic elements of the site region should be understood and described in sufficient detail to allow an evaluation of the safety of a proposed or existing facility. The major issue with respect to the tectonic framework and site suitability is concern for tectonic features influencing the seismicity of the region.

Based on previous studies at SRS and elsewhere, there are no known capable or active faults within the 200 mi (322 km) radius of the site that influence the seismicity of the region with the exception of the blind, poorly constrained faults associated with the Charleston seismic zone (see Section 1.3.6).

1.3.5.3.2 Definition of Seismogenic Faults

Various definitions have been established to evaluate the issues of describing the deformational features and relating specific features to seismicity. These definitions are derived from classical geology and regulatory geology. In some cases, the same concept is defined with different terminology. The definitions that follow are used in discussion of these features:

Active fault: A capable tectonic structure that demonstrates surface or near surface deformation of geologic deposits of a recurring nature within approximately the last 500,000 years or once in the last 50,000 years or/and associated with one or more large earthquakes or sustained instrumentally recorded earthquake activity.

Capable fault: A fault, which has one or more of the following characteristics:

1. Movement at or near the ground surface at least once in the past 35,000 years or repeatedly within the past 500,000 years
2. Macro-seismicity instrumentally determined with records of sufficient precision to demonstrate a direct relationship with the fault
3. A structural relationship to a capable fault according to characteristics 1 or 2 such that movement on one could be reasonably expected to be accompanied by movement on the other.

Capable tectonic source: A capable tectonic source is a tectonic structure that can generate both vibratory ground motion and tectonic surface deformation such as faulting or folding at or near the earth's surface in the present seismotectonic regime.

Fault: A geologic feature that demonstrates deformation or/and rupture of geologic deposits.

Seismic source: Seismic events, which contribute significantly (more than 5% to the total seismic hazards) to a probabilistic ground motion assessment.

Seismogenic source: A seismogenic source is a portion of the earth that is assumed to have uniform earthquake potential (same expected maximum earthquake and recurrence frequency) distinct from the seismicity of the surrounding regions. A seismogenic source will generate vibratory ground motion but is assumed not to cause surface displacement. Seismogenic sources cover a wide range of possibilities from a well-defined tectonic structure to simply a large region of diffuse seismicity (seismotectonic province) thought to be characterized by the same earthquake recurrence model. A seismogenic source is also characterized by its involvement in the current tectonic regime (the Quaternary, or approximately the last 2 million years).

SRS currently works to DOE-STD-1022-94 (DOE 1996c). At this time, there are no faults classified as active or capable at SRS.

1.3.5.3.3 Crustal Geometry of the Region and SRS Area

1.3.5.3.3.1 Thickness of the Crust

Along continental margins, the nature of the crust changes from continental-type crust to oceanic-type crust. Continental crust is generally thicker, less dense, and chemically distinct from ocean crust. The boundary at the base of either continental or oceanic crust also marks a fundamental change in physical parameters and is referred to as the Mohorovicic discontinuity. Density and P-wave velocity is significantly greater below this layer than above.

With the onset of continental rifting, the North American continent began to break away from Africa. Continental crust was stretched and thinned and was intruded with mafic magmas. At the point that one spreading center became dominant, the continental crust ceased to stretch and ocean crust was generated at the spreading center. This marked the initiation of a passive margin along the Atlantic continental margin.

In general, the thickness of continental crust thins from west to east across the eastern United States continental margin. The zone of transition from continental crust to oceanic crust is thought to underlie the offshore Carolina Trough and the Blake Plateau basin. A cross section is provided through the continental margin and Baltimore trough (offshore New Jersey). This is a typical Atlantic-type margin showing the geometry of oceanic crust to the east and continental crust to the west. The Moho deepens from east to west from about 9 mi (15 km) to about 25 mi (40 km), respectively. The continental crust along the margin has been extended and intruded during Mesozoic rifting and is described as rift stage crust. Further east in the middle of the cross section is a complicated zone of transition from continental crust to oceanic crust. The data that support this interpretive model come largely from seismic reflection and refraction surveys and potential field surveys. Offshore South and North Carolina show a similar geometry of thinning crust.

Further inland, the base of crust is discerned by following the configuration of the Moho on seismic refraction or reflection lines. From seismic reflection data collected at SRS, the Moho is interpreted at about 18.6 to 19.6 mi (30.0 to 31.5 km) depth. On the deep seismic profiles, a wide ban of reflections (200 to 300 milliseconds wide) at 10.5 to 11.05 seconds are interpreted to be the Moho. A survey from SRS southeast to Walterboro, South Carolina indicates a crust that thins from 23 mi (37 km) beneath the Dunbarton basin to 19.9 mi (32 km) near Walterboro, South Carolina. This interpretation is based on long seismic refraction and wide-angle seismic reflection data and constrained by gravity and aeromagnetic data. The effect of continental extension and thinning during the Mesozoic rifting event is thus observed in the configuration of the Moho as well as the geologic evidence from the existence of the Dunbarton basin.

1.3.5.3.4 Tectonic Structures

Tectonic structures of interest in the SRS region include faults, folds, arches, basins (rift and post-rift) and paleoliquefaction features from earthquakes. The various structural features in this section are discussed in terms of the age of the feature, starting with the oldest structures. The age of the structure is to be distinguished from the age of the rock in which the structure formed. The primary interest is on how the age of the feature can be discerned with greater or lesser confidence with respect to the definitions of active and capable features.

1.3.5.3.4.1 Paleozoic and Precambrian Structures

Modoc Fault Zone

The Modoc fault zone, located in South Carolina and Georgia, separates greenschist facies metamorphic rocks of the Carolina terrane (Carolina Slate and Charlotte belts) from the amphibolite facies migmatitic and gneissic rocks of the Kiokee belt. The Modoc fault zone is an east-northeast trending ductile shear zone that can be traced from central Georgia to central South Carolina based on geological and geophysical data. The Modoc fault zone dips steeply to the northwest and contains quartzites, phyllite, paragneiss, and button schists correlative with units in the Asbill Pond Formation of the Carolina terrane. The lower grade Carolina terrane rocks underwent significant granitic sheet intrusion, prograde metamorphism, and penetrative strain during the Alleghanian orogeny. Fabric in the fault zone is characterized by brittle and ductile deformation produced by ductile shear during an early phase of the Alleghanian orogeny (315 Ma). The Modoc zone is overprinted by the Irmo antiform near Columbia, South Carolina. Extension of the Modoc fault zone further to the northeast is uncertain but there are shear zones in North Carolina and Virginia that may be of the same deformational phase. An important normal-sense component exists in the Modoc zone on the northwest flank of the Kiokee belt. The significance of the age of myonitic fabric on this fault at 315 Ma is that the fault is very old and therefore not in the realm of active or capable in terms of regulatory guidance.

Augusta Fault Zone

The Augusta fault zone is located near Augusta, Georgia, and juxtaposes amphibolite grade rocks of the Kiokee belt against the greenschist facies rocks of the Belair belt. The fault trends east-northeast and dips approximately 45° southeast. The fault contains two distinct deformation fabrics: a mylonite about 820 ft (250 m) thick is overprinted by a brittle fabric. Kinematic analysis within the mylonite zone reveals a hanging wall down component during the movement history. Furthermore, the hanging wall consists of lower greenschist facies while the footwall contains upper amphibolite facies. Lower grade rocks structurally positioned above higher grade rocks in combination with shear sense indicators suggests a low-angle normal fault movement for the Augusta fault zone. This is a new view of the Augusta fault zone, which previously had been considered a ductile-to-brittle thrust fault or a strike-slip fault. It now appears that ductile faults with a normal sense component were an important aspect of late Alleghanian deformational history. From reported $^{40}Ar/^{39}Ar$ ages from samples along a traverse across the Modoc fault and Augusta fault zones, it is concluded that a 274 Ma cooling age closely dates initiation of extensional movement on the Augusta fault zone. This cooling age indicates the time when the ductile fabric was generated and therefore when the fault moved. This fault does not fall into the capable or active fault definitions of the regulatory guides.

Near Augusta, Georgia, the Augusta fault zone and the southeast edge of the Kiokee belt are offset by the north-northeast trending Belair fault. It has been suggested that the Belair fault was a tear fault linking two segments of the Augusta fault zone. Within the Atlantic Coastal Plain province sediments, the final stage of movement on the Belair fault occurred during the Cenozoic as high angle reverse faulting that offset the Late Cretaceous uniformity by 100 ft (30 m) and the Early Eocene uniformity by 40 ft (12 m).

It has been suggested that the Modoc fault zone, the Irmo shear zone, and the Augusta fault zone are part of the proposed Eastern Piedmont fault system, an extensive series of faults and splays extending from Alabama to Virginia. Aeromagnetic, gravity, and seismic data indicate that the Augusta fault zone continues in the crystalline basement beneath the Coastal Plain province sediments.

Paleozoic Basement Beneath SRS

Information concerning structural features in the basement beneath SRS is mainly derived from analysis of structural fabrics recorded in core samples from deep borings and at larger scales from geophysical techniques such as gravity and magnetic surveys and seismic reflection profiles. Seismic reflection surveys were conducted onsite in 1972 and 1987 to 1988 to image the basement reflector. In 1972, Seismograph Services Incorporated did a seismic reflection survey as part of the Bedrock Waste Storage Project.

Approximately 60 line miles of survey were completed. This was the first survey that indicated the presence of basement faults, some of which disturbed Coastal Plain sediments. Offset reflectors were interpreted as basement faults. No official report was written for the survey.

During the period 1987 to 1988, a more thorough seismic reflection survey of SRS was completed. The program consisted of two phases, which covered approximately 134 line miles distributed over much of SRS. These data were used to further define basement faults and to image any shallower or deeper structures. Subsequent seismic reflection and field potential geophysical data have led to various basement fault interpretations.

These data were reprocessed and re-interpreted to produce improved images of the Coastal Plain section and faults known to deform Coastal Plain sediments. Recovery of the shallow time section (40-200 milliseconds) in conjunction with recovery of the deep section (7-14 seconds) led to the discovery of additional faults clearly rooted in the midcrust and deforming Coastal Plain sediments.

An integrated analysis of the structural fabric in the basement core in addition to the geophysical data concluded that at least two regional scale ductile faults are present in the basement beneath SRS and vicinity, the Upper Three Runs fault and the Tinker Creek fault. These faults are expressed in the aeromagnetic data as lineaments and are interpreted to be associated with a thrust duplex that emplaces the rocks of the Pen Branch Formation (Tinker Creek Nappe) over the DRB Formation. The age of the faulting is constrained by a radiometric age on biotite that dates the movement at about 300 Ma, which would indicate that these faults are part of the Paleozoic Eastern Piedmont fault system.

In order to resolve faulting that deform Coastal Plain sediments, the topography of the basement surface was mapped utilizing the data listed above along with more recently acquired seismic reflection profiles. The map of basement topography indicates that offsets of the basement surface that range from approximately 100 ft (30 m) in magnitude down to the resolution limits of the data are present on the basement surface. However, most of these offsets are of relatively small magnitude and have limited lateral extents. Faults that involve Coastal Plain sediments that are considered regionally significant based on their extent and amounts of offset include Atta, Crackerneck, Martin, Pen Branch, and Tinker Creek. The Crackerneck and Pen Branch faults are relatively well constrained with borings. The other faults are projected from geophysical data only and their parameters are less well known. Of these faults the Pen Branch fault has been extensively studied and found to be not capable or not active.

1.3.5.3.4.2 Mesozoic: Extensional Tectonics and Rift Basins

A broad zone of extended (rifted) continental crust formed along the eastern continental margin of the United States, especially the southeastern portion during the early Mesozoic when North America broke away from Africa and South America. This region extends from Florida to Newfoundland and includes the area where SRS exists. The eastern seaboard domain encompasses this extended crust and is a sub-domain of the North American stable continental crust. Its significance is that within stable continental crust, areas of extended crust potentially contain the largest earthquakes. The Eastern Seaboard domain is bounded on the west by the western-most edge of Triassic-Jurassic onshore rift basins or the boundaries of the structural blocks in which they occur. The eastern boundary is the continental/ oceanic boundary which is coincident with the East Coast magnetic anomaly. Rifted crust is crust that has been stretched, faulted, and thinned slightly by rifting but is still recognizable as continental crust. The faulting is extensional or normal and down-dropped blocks form rift basins.

Geometric and kinematic arguments suggest that early Mesozoic normal faults may have been reactivated Alleghanian faults. Studies of exposed and buried rift basins in the eastern United States show that the faults controlling basin formation are complex, with border faults of variable dip, antithetic faults of variable displacement, and cross or transfer faults that fragment the basin into sub-basins. Within the SRS region, there is the Dunbarton rift basin, which is part of this tectonic setting. The fault that controls the basin formation, the Pen Branch fault, initially moved as a normal fault during the Triassic. However, it may have been a reactivated Paleozoic fault, and it has moved since the rifting episode.

One locus of major extension during early stages was in the South Georgia rift, which extends from Georgia into South Carolina. The Dunbarton basin, underlying SRS, is most likely structurally related to that rift basin. During the later stage of rifting (early Jurassic), the focus of extension was shifted eastward to the major marginal basins that would become the site of the Atlantic Ocean basin. The extension in the onshore, western-most basins, such as the Dunbarton, Florence, and Riddleville, waned. Eventually, rifting of continental crust ceased as sea floor spreading began in the Atlantic spreading center sometime around 175 Ma. The oldest ocean crust in contact with the eastern continental margin is late middle Jurassic. The significance of the age of transition from rifting to seafloor spreading is that the tectonic regime of rifting is no longer acting on the crust in the eastern seaboard domain. The basins are not continuing to form and for the most part, the crust is quiescent. The modern tectonic environment is partly based on ridge push from the Atlantic spreading center, and recent crustal stress measurements indicate a compressive northeast directed stress for the region.

1.3.5.3.4.3 Post-Rift and Cenozoic Structures

The following discussion includes tectonic features that have formed on the continental margin since the end of the Mesozoic rift stage (post-rift stage). Therefore, the discussion will include the late Mesozoic, as well as Cenozoic, tectonic elements. Post-rift tectonism is expressed along the eastern continental margin in a variety of structures originating in the crystalline basement and affecting the deposition of sediments and deformation of Coastal Plain sediments from the Cretaceous through the Cenozoic. These structures include offshore sedimentary basins, such as the Carolina trough and the Blake Plateau basin; transverse arches and embayments, such as the Cape Fear arch and the Southeast Georgia Embayment; Coastal Plain faulting; and paleoliquefaction features that provide information on the recurrence of the Charleston earthquake.

Outer Margin Basins

Sedimentary basins along the continental margin (offshore) have formed in response to subsidence in the outer continental margin crust. Outer margin subsidence resulted from (1) the extension and thinning of the crust during early Mesozoic rifting followed by thermal contraction as the lithosphere cooled, and (2) from sediment loading on the lithosphere. The outer margin sediment basins formed on this transitional crust. Toward the continent, continental crust was less altered and thicker. This portion of the margin subsided at a slower rate than the outer margin. Because of the differing rates and total amount of subsidence, a hinge zone developed all along the continental margin. Seaward of the hinge zone the crust is rift-stage continental crust. The crust here has subsided to greater depths. This is also the location of the outer margin basins. Landward of the hinge zone, the crust is the thicker, unaltered crust. The depth to crust in this region is significantly shallower with a corresponding thinner veneer of post-rift sediments. The Atlantic Coastal Plain is located landward of the hinge zone and has been affected by the outer margin subsidence.

Folding and Arching

Not all tectonism along the continental margin is due to outer margin subsidence. Lithospheric cooling and sediment loading were dominant processes during Middle Jurassic through early Cretaceous. The sediments now present in the outer margin basins are mostly Jurassic and early Cretaceous. Compressional faults, folds and thickness variations in the late Cretaceous and Cenozoic are due to intraplate stress fields rather than margin subsidence. These latest features are seen as highs and lows in the crust that control Coastal Plain sedimentation and are oriented perpendicular to the hinge zone. They are thought to be indicative of continued, episodic, differential crustal movements (tectonic) from Cretaceous through Pleistocene. The sedimentary sections are thinner, incomplete on the highs, or arches, and thicker with complete sections in the lows or embayments. The most prominent arch is the Cape Fear arch near the North Carolina-South Carolina border. Other arches in the region include the Norfolk arch near the North Carolina-Virginia border, and the Yamacraw arch near the South Carolina-Georgia border.

The Cape Fear arch has a variable history, receiving sediments during the Late Cretaceous and then acting as a sedimentary divide or arch from Latest Cretaceous through Late Tertiary. Upper Cretaceous Santonian sediments are the oldest strata to completely cover the Cape Fear arch.

Paleocene, Eocene, and Oligocene strata comprise 2,100 ft (640 m) of marine carbonate in the southeast Georgia embayment and thin to the northeast, toward the Cape Fear arch. The sediments become largely terrigenous on the flank of the arch and are completely missing over the crest of the arch; thus suggesting the arch was acting as a sedimentary divide beyond the Oligocene. Uplift on the arch may have continued through the Pleistocene.

Faulting

The most definitive evidence of crustal deformation in the Late Cretaceous through Cenozoic is the reverse sense faulting found in the Coastal Plain section of the eastern United States. In the late 1970s and early 1980s, USGS conducted a field mapping effort to identify and compile data on young tectonic faults in the Atlantic Coastal Plain. Consequently, many large, previously unrecognized Cretaceous and Cenozoic fault zones were found. Of 131 fault localities cited, 26 were within North and South Carolina. The identification of Cretaceous and younger faults in the eastern United States is greatly affected by distribution of geologic units of that age. Many of the faults are located in proximity to the Coastal Plain onlap over the crystalline basement. This may be due to the ease of identifying basement lithologies in fault contact with Coastal sediments.

The faults are characterized as mostly northeast trending reverse slip fault zones with up to 62 mi (100 km) lateral extent and up to 250 ft (76 m) vertical displacement in the Cretaceous. The faults dip 40° to 85°. Offsets were observed to be progressively smaller in younger sediments. This may be due to an extended movement history from Cretaceous through Cenozoic. Based on their similar characteristics, Cretaceous and younger faulting in the Coastal Plain is associated into several fault provinces. SRS falls into the Atlantic Coast fault province. A comparison of Cretaceous and younger faulting in SRS found that faulting on SRS shared similar characteristics with the faults in the Atlantic Coastal fault province including orientation and offset history. This comparison concluded that Cretaceous and younger faulting on SRS was not unique in comparison to the Atlantic Coast fault province in general and as a result shared the same seismic hazard.

Offset of Coastal Plain sediments at SRS includes the four Tertiary unconformities. Following deposition of the Snapp Formation, some evidence indicates oblique-slip movement on the existing faults. The offsets involve the entire Cretaceous to Paleocene sedimentary section. In A/M Area, this faulting formed a series of horsts and grabens bounded by subparallel faults that truncate at the fault intersections. The strike orientations of the individual fault segments vary from N 11°E to N 42°E, averaging about N 30°E. Apparent vertical offset varies from 15 to 60 ft (4.5 to 18 m), but throws of 30 to 40 ft (9 to 12 m) are most common.

This faulting was followed by erosion and truncation of the Paleocene section at the Lang Syne/Sawdust Landing unconformity. Subsequent sediments were normal faulted following deposition of the Santee Formation. Typically, the offset is truncated at the Santee unconformity, and the overlying Tobacco Road/Dry Branch formations are not offset. Locally, however, offset of the overlying section indicates renewed movement on new or existing faults after deposition of Tobacco Road/Dry Branch sediments.

In conjunction with these observations of Coastal Plain faults, modern stress measurements provide an indication of the likelihood of Holocene movement. There is a consistent northeast-southwest direction of maximum horizontal compressive stress (N 55-70°E) in the southeast United States. This determination is based on direct in situ stress measurements, focal mechanisms of recent earthquakes, and young geologic indicators. Shallow seismicity in the area, within crystalline terranes, is predominantly reverse character. It is concluded that the northeast directed stress would not induce damaging reverse and strike-slip faulting earthquakes on the Pen Branch fault, a northeast striking Tertiary fault in the area. These same conclusions may be implied for the other northeast trending faults.

In A/M Area at SRS, faulting appears to have been episodic and to have varied in style during the Tertiary. Oblique-slip faulting dominated the Cretaceous/Paleocene events, with a local north-south stress orientation. Subsequently, left-lateral shear on the pre-existing faulting and normal faulting occurred, with a corresponding shift in the direction of maximum compressional stress oriented N 20°E to N 30°E.

Pen Branch Fault

The Pen Branch fault has been regarded as the primary structural feature at SRS that has the characteristics necessary to pose a potential seismic risk. As stated below, studies have indicated that, despite this potential, the fault is not capable.

The Pen Branch fault is an upward propagation of the northern boundary fault of the Triassic Dunbarton basin that was reactivated in Cretaceous/Tertiary time. The fault dips steeply to the southeast. In the crystalline basement, slip was originally down to the southeast, resulting in the formation of the Dunbarton rift basin. However, movement during Cretaceous into Tertiary time was reverse movement, that is, up to the southeast. There could also be a component of strike-slip movement.

The bulk of evidence collected for the Pen Branch Fault Program supports the conclusion that the most recent faulting on the Pen Branch fault is older than 500,000 years. Therefore, the Pen Branch fault is not a capable fault. In a study designed to examine only the sediments with an age of 1 Ma or less, deformation was not found to exist.

The Pen Branch fault was identified in the subsurface at SRS in 1989. It was interpreted from seismic reflection surveys and other geologic investigations. A program was initiated at that time to determine the capability of the fault to release potentially damaging seismic energy. Separate actions completed under this program title include the following:

- Shallow drilling of Coastal Plain sediments with eight paired drill holes to bracket the location and the amount of displacement on the Pen Branch fault
- Formation of the Earth Science Advisory Committee for independent assessment and verification of the data gathered
- A deep drilling program into the fault zone in basement underlying Coastal Plain sediments
- A high-resolution, shallow seismic reflection survey over the fault trace

- Reprocessing seismic reflection data to enhance the shallow portions of the data and then the deeper portions of the data under separate processing protocols
- Quaternary geology investigation to examine the youngest surfaces and deposits onsite for indications of neotectonism
- Confirmatory Drilling Project: The final investigation carried out under the 1989 Pen Branch Fault Program. The investigation focused on a small zone over the fault where seismic reflection data had been collected previously and indicated that the fault deforms the subsurface reflector at 200 milliseconds two-way travel time. Eighteen drill holes, two to basement and the others to a depth of 300 ft (91.4 m), were arranged to adequately define the configuration of the layers deformed by the fault. Boreholes were spaced over a zone of 800 ft (245 m), north to south. Results suggest that deformation by the fault is limited to the Lang Syne/Sawdust Landing unconformity (~50 Ma) (Stieve et al. 1994). Other interpretations may be offered where offset on the Pen Branch fault involved the Tobacco Road and Dry Branch Formations. However, based on presently available data, the Pen Branch fault is not capable.

It is therefore concluded that the Pen Branch fault is not a capable fault.

Belair Fault Zone

The Belair fault is a Cenozoic fault located on the inner margin of the Coastal Plain near Augusta, Georgia. The fault is really a set of en echelon faults extending at least 15 mi (24 km) and trending northeast. Individual fault segments are 1.25 to 3 mi (2 to 5 km) long. The fault zone places Late Precambrian phyllites of the Belair belt over Middle Tertiary Coastal Plain sediments. The faults show oblique-reverse slip movement and as much as 100 ft (30 m) of vertical offset has taken place since the deposition of the Barnwell Group sediments. The Belair fault zone has a protracted history of movement in that it initiated as a tear fault on the Augusta fault during the late Alleghanian (Hercynian). The fault was later reactivated as an oblique-reverse slip fault during the Cretaceous. The age of latest movement on the Belair fault zone can only be determined based on available stratigraphic marker horizons. The age of last movement can be bracketed between the age of the sediment that is offset and the age of the stream terrace that caps this strata and is not deformed. The age of the deformed strata can be as young as 40 Ma and the age of the stream fill terrace is between 26,000 and 1,550 years based on carbon-14 dates of peat. This makes the age determination on the fault uncertain because the age of undeformed deposits capping the deformation is poorly defined and because the fault age can only be bracketed based on deposits that precede a large time period unconformity. However, it has been concluded that the Belair fault zone records movement from late Early Cretaceous through at least Eocene, which makes the fault approximately 40 Ma.

Buried or Blind Faulting in the Charleston Seismic Zone

Seismic activity in the southeastern United States has been dominated by the 1886 Charleston, South Carolina, earthquake, aftershocks, and the continuing low-level seismic activity that persists in the area today. The search for structures to explain seismicity near Charleston has been complicated by the absence of surface faulting, fault scarps, or other fault-generated topographic features. Because the seismic zone is buried in the subsurface, the presence of possible causal geologic structures at depth must be inferred through geophysical methods. Many geologic, geophysical, and seismic studies have been completed by a number of researchers since the mid-1970s resulting in the emergence of some widely diverse models and hypotheses. A review of the more recent models reveals that uncertainty still exists on details of the causal relationship between local geologic structures and seismic activity in the region. However, significant progress has been made.

Most hypotheses relating southeast United States seismicity to geologic structure assume activity to occur along preexisting zones of weakness favorably oriented with respect to the ambient stress field. Understanding the regional stress is an essential element in the formation of causative models.

Recent Models

Eastern United States coastal plain seismic activity occurred in distinct zones superposed on a regional background of very low level seismicity. The most active of these zones and the one assumed likely to be associated with the 1886 Charleston event is the Middleton Place-Summerville Seismic Zone (MPSSZ). The MPSSZ lies some 12 mi (20 km) northwest of Charleston, well within the mesoseismal area of the 1886 Charleston earthquake. It was in this area that the delineation of two possible intersecting faults was identified when relocating instrumentally recorded earthquakes from 1974 to 1980. The first was a shallow, northwest-trending fault defined by hypocenters 2.5 to 5 mi (4 to 8 km) deep striking parallel to the Ashley River. This was named the Ashley River fault. The second fault was labeled the Woodstock fault. The Woodstock fault trends north-northeasterly and is defined by planar distribution of hypocenters with depths between 5.6 and 8.1 mi (9 and 13 km). It intersects and appears deeper than the Ashley River fault. Recent studies refine and complement the 1982 effort by utilizing 58 additional well-recorded events located in the MPSSZ from 1980 to 1991. Fault-plane solutions from the new data reinforce the northeast-southwest maximum horizontal stress direction of previous studies. However, the epicentral distribution of this new data displayed no obvious pattern of association with the Ashley River fault or the Woodstock fault. Therefore, the seismicity was divided into sets according to focal mechanism in an attempt to infer a structural cause of the earthquakes. Results of this breakout revealed:

- The first set of data favored a northwest-southeast strike and southwest dip direction, suggesting compatibility with the Ashley River fault zone. Solutions were found to have components of mostly strike-slip and/or reverse faulting mechanisms.
- The second set of data was further divided into two subsets with the first displaying mainly vertical fault planes striking north-south and the second subset striking north northeast-south southwest with shallower dips to the southwest. These two subsets were classified as belonging to the Woodstock fault zone. Solutions of these events revealed mostly strike-slip motion on the vertical fault with a strong thrust component on the shallower dipping events.

Results indicated that the Ashley River and the Woodstock faults are not simple planar features, but resemble zones composed of short segments of varying strike and dip. When location was factored into the analyses, it was found that events associated with all sets of data occurred in the same area. From these observations, it was concluded that the seismicity in the MPSSZ defines the intersection of two fault zones, which are inferred to be the Ashley River fault zone and the Woodstock fault zone.

Paleoseismic Data

Estimating seismic recurrence intervals of moderate to large earthquakes within the southeastern United States is difficult. These difficulties stem from the relatively short (300 years) historical record coupled with an absence of surface faulting, offset features, or prehistoric ruptures.

Geologic field study methods developed to extend the seismic record assess both the temporal and spatial distribution of past moderate and large earthquakes. This assessment is carried out through identification and dating of secondary deformation features resulting from strong ground shaking. In the southeast, this extension of the seismic record has been accomplished through field search for earthquake-induced liquefaction flowage features called “sand blows” associated with prehistoric earthquake-induced paleoliquefaction features.

These features are attributed to prehistoric earthquake induced liquefaction as defined by the transformation of sediments from solid to liquid state caused by increased pore water pressure. The increased pore pressure is caused during or immediately after an earthquake. “Sand blows” are features formed where earthquake shaking causes liquefaction at depth followed by the venting of the liquefied sand and water to the surface.

The following section summarizes paleoliquefaction studies in the southeastern United States. Aspects that are of particular importance to SRS include the following:

- No conclusive evidence of large prehistoric earthquakes originating outside of coastal South Carolina has been found.
- Young fluvial terraces at or slightly above the level of the modern floodplain and Carolina bays are the most likely depositional environments for potentially liquefiable deposits in the SRS region.

Paleoliquefaction Studies in the Eastern United States

Widespread occurrences of earthquake-induced sand blows were originally reported throughout the meizoseismal area of the 1886 Charleston, South Carolina, earthquake. Excavation and detailed analyses of these liquefaction flow features provided the first insight into the pre-history of the Charleston earthquake. Other pre-1886 liquefaction flow features (mostly sand blows) were discovered and investigated near the town of Hollywood, about 15 mi (25 km) west of Charleston. Searches for sand blows were continued throughout the Charleston area and expanded to the remaining coastal South Carolina areas. Eventually, areas of study were broadened to include Delaware, Virginia, North Carolina, and Georgia. The objective was to identify other epicentral regions, if they existed, and to estimate the sizes of pre-1886 earthquakes assuming the areal extent of sand blows caused by an earthquake are a function of earthquake intensity in areas of similar geologic and groundwater settings. To date, no conclusive evidence of large prehistoric earthquakes originating outside of coastal South Carolina have been found.

In coastal South Carolina investigations, identification of paleoliquefaction features generally adheres to specific local geologic criteria. Some specific relations between liquefaction susceptibility and subsequent formation of liquefaction features (sand blows) are summarized below:

- A water table near the ground surface greatly increases susceptibility to liquefaction (depth <3 ft [<1 m]).
- Virtually all seismically induced liquefaction sites are located in either beach-ridge, backbarrier, or fluvial depositional environments. Of these, beach-ridge deposits were found to be the most favorable for the generation and preservation of seismically induced liquefaction features.
- Due primarily to the effects of chemical weathering, materials older than about 250 ka were less susceptible to liquefaction than were younger deposits. This indicates that the probabilities of sand blows forming in deposits of late Pleistocene and early Holocene age are extremely low.
- The liquefied materials are generally fine-grained, well-sorted (i.e., uniformly graded), clean beach sand. The principal properties of sand that control liquefaction susceptibility during shaking are degree of compaction (measured as relative density by geotechnical engineers), sand-grain size and sorting, and cementation of the sand at grain-to-grain contacts. Fine grained well-sorted sand of ancient and modern beaches are much more susceptible to liquefaction than compacted well-graded sand used in engineered construction.
- Features large enough to be interpreted as possibly having an earthquake origin in the low country were found only in sand deposits having total thickness greater than 7 to 10 ft (2 to 3 m).
- The depth of the probable source beds at liquefaction sites is generally less than 20 to 23 ft (6 to 7 m), and the groundwater table is characteristically less than 10 ft (3 m) beneath present ground surface.

Liquefaction features that typify the coastal South Carolina area have been described as sand blow explosion craters and sand-vents/fissures.

<u>Sand Blow Explosion Craters or Filled Sand Blow Craters</u>

Following the onset of seismic loading from a moderate to large earthquake, development of sand blow craters can be described by four sequential phases: (a) an explosive phase, (b) a flowage phase, (c) a collapse phase, and (d) a filling phase. These were first described based on historical accounts and the internal morphology of exhumed features. This feature illustrates characteristics consistent with earthquake-induced liquefaction origin. The soil horizon is cut by an irregular crater and filled with stratified to nonstratified and graded sediments. The fill materials are fine-to medium-grained sand and clasts from the original soil profile, as well as sand from source beds at depths below the exposed C horizon. Sand-blow explosion craters were found primarily on beach deposits, and are notably absent in fluvial settings.

Sand-Vents/Fissures or Sand Volcanoes

Sand volcanoes vent to the surface and leave relict sand mounds. These features generally form in circumstances where the liquefying source zone, at depth, is overlain by a cohesive, finer grained, non-liquefiable layer, or "cap." The thickest part of the mound ranges from a few centimeters to as much as 10 in (25 cm). The mounds are generally thickest directly above source feeder vents that extend downward through clay-bearing stratum. This type of liquefaction feature was rare in beach settings, but commonly found within backbarrier marine sediments and in interbedded fluvial deposits.

Dating paleoliquefaction episodes can be accomplished either qualitatively or quantitatively. Qualitative methods include degree of staining and weathering of sands within the feature, thickness of overlying profiles, and cross cutting relations of one feature compared to another. A more quantitative approach involves radiometric dating of organic material within or cut by the liquefaction feature. An example of a minimum age constraint is dating of roots that have grown into the feature. A maximum constraint can be determined from roots cut by the feature or by dating organic materials recovered from the collapsed area of the crater during the liquefaction episode. The most accurate estimates for the age of a liquefaction episode are obtained from radiometric dating of leaves, pine needles, bark, or small branches that were washed or blown into the liquefaction crater following formation.

Utilizing the above methods, at least four pre-1886 liquefaction episodes were described at approximately 580 ±104 (CH-2), 1311± 114 (CH-3), 3250 ± 180 (CH-4), and 5124 ± 700 (CH-5) years before the present. CH refers to Charleston source with CH-1 designated as the 1886 earthquake. An even older episode (CH-6) was found to be cut by a CH-5 feature.

Changes in hydrologic conditions (groundwater levels) play an important role in determining an area's susceptibility to liquefaction. On the basis of published sea-level curves, groundwater levels in the southeastern United States have been assumed at or near present levels for only the past 2,000 years. Consequently, the paleoliquefaction record is probably most complete for this period. However, beyond the 2,000 to 5,000 year range, knowledge of groundwater conditions is considerably less reliable, making gaps in the paleoseismic record much more probable.

Paleoliquefaction Assessment of the Savannah River Site Region

Reconnaissance surveys were performed in search of paleoliquefaction sites as far as 40 mi (65 km) inland along the Savannah River. However, no South Carolina paleoliquefaction surveys or studies have yet been performed as far inland as SRS. Several factors suggest that it would be difficult to locate and evaluate the origin of potential liquefaction features within the geomorphic and geologic environment of the SRS. Investigations elsewhere in South Carolina have shown that aerial photographs are useless for locating 1886 and pre-1886 sand blows. The SRS region has no Pleistocene beach ridges for sand-blow crater formation. Young fluvial terraces at or slightly above the level of the modern floodplain and Carolina bays are the most likely depositional environments for potentially liquefiable deposits in the SRS region. However, the search for liquefaction features in these areas is severely limited by the lack of access, high water table conditions, dense vegetative cover, and few exposures.

Existing exposures in the Savannah River fluvial terraces above the modern floodplain were examined for evidence of liquefaction. Extensive reconnaissance of the Bush Field and Ellenton terraces on the SRS revealed few exposures of adequate depth and extent to evaluate the presence or absence of liquefaction. Terrace alluvium associated with these terraces contains a high percentage of sand, but based on the degree and depth of pedogenic modification and probable depth to the water table, these terraces were judged to have had a relatively low susceptibility to liquefaction during the late Pleistocene and Holocene. In this fluvial environment, the most likely liquefaction features are sand vents or fissures. No evidence of sand vents, fissures, or other liquefaction features were observed in any of the available exposures examined. Recognition of paleoliquefaction features in the pre-Quaternary deposits at SRS would be extremely difficult, if not impossible.

A paleoliquefaction assessment of SRS was prepared by WSRC in 1996. This investigation indicated that several hydrologic, sedimentological, and logistical conditions must be met for seismically induced liquefaction (SIL) to occur and be identified. These included (1) the presence of Quaternary-age deposits; (2) the presence of a shallow groundwater table; (3) proximity to potential seismogenic features; (4) geologic sections of several different types of unconsolidated deposits; and (5) quality and extent of exposure.

Based on these considerations, the floodplains of the Savannah River and its tributaries were identified as the areas on SRS with the highest potential for generating and recording Holocene SIL features. The terraces of the Savannah River and tributaries were also considered potential areas for recording Quaternary SIL features, though these features would likely be older than ones in the floodplains. The upland areas on SRS have a low potential for recording Quaternary SIL because they are pre-Quaternary in age, partially indurated, and generally high above the water table. Paleoliquefaction investigations in the SRS uplands, therefore, only targeted those sites postulated by previous workers as containing evidence of SIL.

Conclusions from this paleoliquefaction assessment fell into two categories: (1) field studies of floodplain deposits along the Savannah River, and (2) evaluation of previously reported paleoliquefaction and neotectonic features located in pre-Quaternary sediments. A brief summary of findings in these two areas follows.

Investigation of banks along 68 mi (110 km) of the Savannah River adjacent to SRS revealed a large number of excellent exposures of floodplain deposits. Most of the exposed deposits were clay and silt, and had a low liquefaction potential. Locally, however, clean sand deposits with a high liquefaction potential were present. Given the extensive amount of exposure and the local presence of liquefiable materials, SIL features would likely be present in these deposits if strong earthquakes had occurred after they were deposited. However, the presence of buried historical objects and radiocarbon dates from these materials illustrated that most or all of the exposed floodplain deposits were historical in age. As no strong ground motions have occurred in historical times in the SRS area, SIL features could not exist in these deposits. Furthermore, the fact that they date to historical times precludes them from providing any information of earlier earthquake history.

The absence of SIL features in the bank exposures does not preclude the possibility that SIL features exist deeper in the section or on the older, higher terraces. In fact, the local presence of liquefiable materials in the Modern floodplain deposits suggests that, if strong prehistoric earthquakes had occurred, SIL features are probably present at depth in the floodplain deposits or on the older/higher terraces. These key areas were not investigated, and exposure is limited.

The upland areas of SRS were considered to have a low potential for recording Quaternary SIL because the deposits are old (pre-Quaternary), generally high above the water table (>30 ft [>10 m]), and are indurated. However, previous investigators described several features in the Tertiary section as clastic dikes, and attributed them to SIL and/or neotectonic activity. The sites were evaluated to determine if they have the diagnostic characteristics that have recently been documented for true SIL.

Four types of post-depositional features were identified: (1) irregularly shaped cutans; (2) structurally controlled cutans; (3) joints; and (4) faults. Cutans are a modification of the texture, structure, or fabric of the host material by pedogenic (soil) processes, either by a concentration of particular soil constituents or in situ modification of the matrix. These features were interpreted through the process of elimination procedure of multiple working hypotheses. None were thought to be the result of SIL. Summary observations of these four elements are given below.

Irregularly Shaped Cutans

The absence of offset on irregularly shaped cutans eliminated the possibility that they were faults, and the undisrupted bedding within and across the feature eliminated the possibility that they were clastic dikes, SIL features, or ice wedges. The higher density of these features near the ground surface and their similarity in appearance to the zone of more intense geochemical alteration at the top of each exposure suggested these features were pedogenic in origin. They were interpreted as an in situ, pedogenic modification of the texture, structure, and fabric of the host material, and therefore were referred to as “irregularly shaped cutans.”

Structurally Controlled Cutans

There was no evidence of rapid injection of liquefied material into structurally controlled cutans. The similarity of the material within the features and that of the host material, as well as undisrupted pebbly horizons within and across the features, demonstrated the features were not clastic dikes, ice wedges, or SIL features. The absence of offset across virtually all of the features demonstrated that they did not develop as faults. They were interpreted to have developed through pedogenic processes based on: (1) the similarity and relationships that illustrate the features formed concomitantly with the sub-horizontal zone of more intense geochemical alteration at the top of each exposure, and (2) an overall downward thinning and local pinch-out of the features. Strong preferred orientations at most exposures, parallelism with adjacent joints, and their occurrence along fault planes at one locality, suggested that the orientation of most of the features was controlled by pre-existing structures, and were therefore referred to as “structurally controlled cutans.”

Joints

Joints are common on SRS and vicinity. Though their mechanism of formation is not well understood, their age was determined to be constrained by interpretation that cutans often developed along pre-existing joints. The joints, therefore, pre-dated the pedogenic processes that formed the cutans. Highly variable orientations of cutans suggested that the orientation of joints on the SRS was also highly variable. A gradual and consistent change in orientation of cutans over 100 to 200 ft (30 to 60 m) at some outcrops suggested the orientation of joints also locally changed gradually and consistently. A lack of consistent preferred orientations of joints across SRS did not favor a tectonic origin for these features. Furthermore, no clear relationship existed between the joint-controlled cutans and the local topography. The joints, therefore, were probably not related to slope mass wasting. A local, gradual change in orientation over several hundred feet, and the common occurrence of closed depressions on SRS, are consistent with differential settling from subsurface dissolution. This hypothesis was not addressed directly during this study.

Faults

Small-scale faults were clearly present at several locations on and adjacent to SRS. Most faults had normal separations, although one small, sub-vertical feature had a component of reverse motion. Separations observed were less than 3 ft (1 m). The amount of horizontal slip was not determined for any of the faults. Low, medium, and high angle faults were also present. The presence of cutans on several faults suggested that these faults were older than the pedogenic processes that formed the cutans. A 2 ft (0.6 m) thick Pliocene loess deposit overlies one fault zone, indicating these faults are probably older than Pliocene. One fault zone was of particular interest because it was located at the approximate upward projection of the Pen Branch fault. Furthermore, the faults in outcrop trended northeast, sub-parallel to the Pen Branch fault. The relationship between the faults in outcrop and the Pen Branch fault, if any, was not investigated.

Table 1.3.5-1. Correlation of Geologic and Engineering Units for the MFFF Site

Geologic Unit	Engineering Unit Symbol
"Upland Unit" Formation	TR1 and TR1A Layers
Tobacco Road Formation	TR2A and TR2B Layers
Dry Branch Formation	TR3/4 and DB1/3 Layers
Tinker/Santee Formation	DB4/5, ST1, and ST2
Warley Hill Formation	GC Layer
Congaree Formation	CG Layer

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Figure 1.3.5-22. MFFF Site Exploration Programs

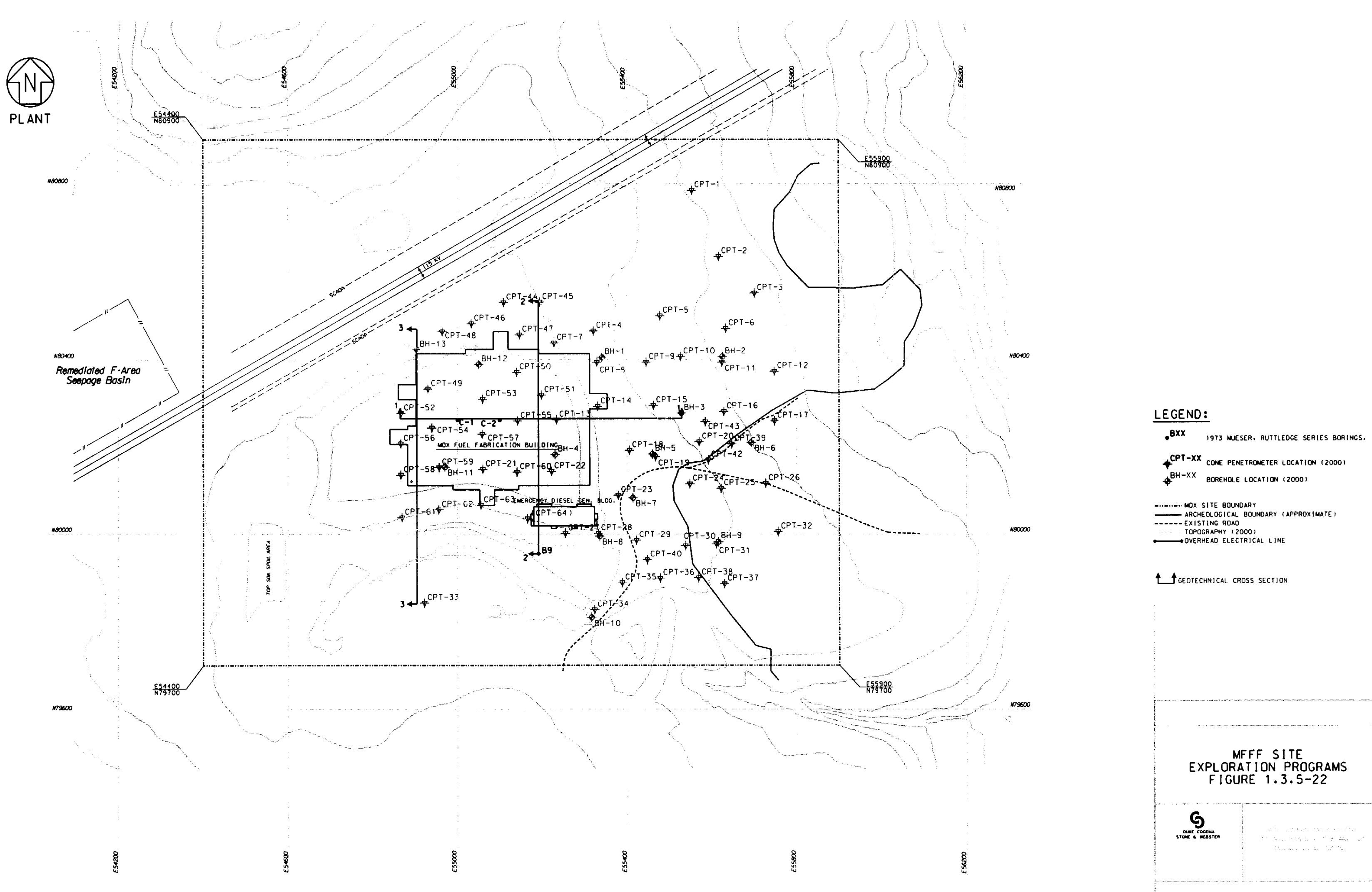


Figure 1.3.5-23. Geotechnical Cross Section 1

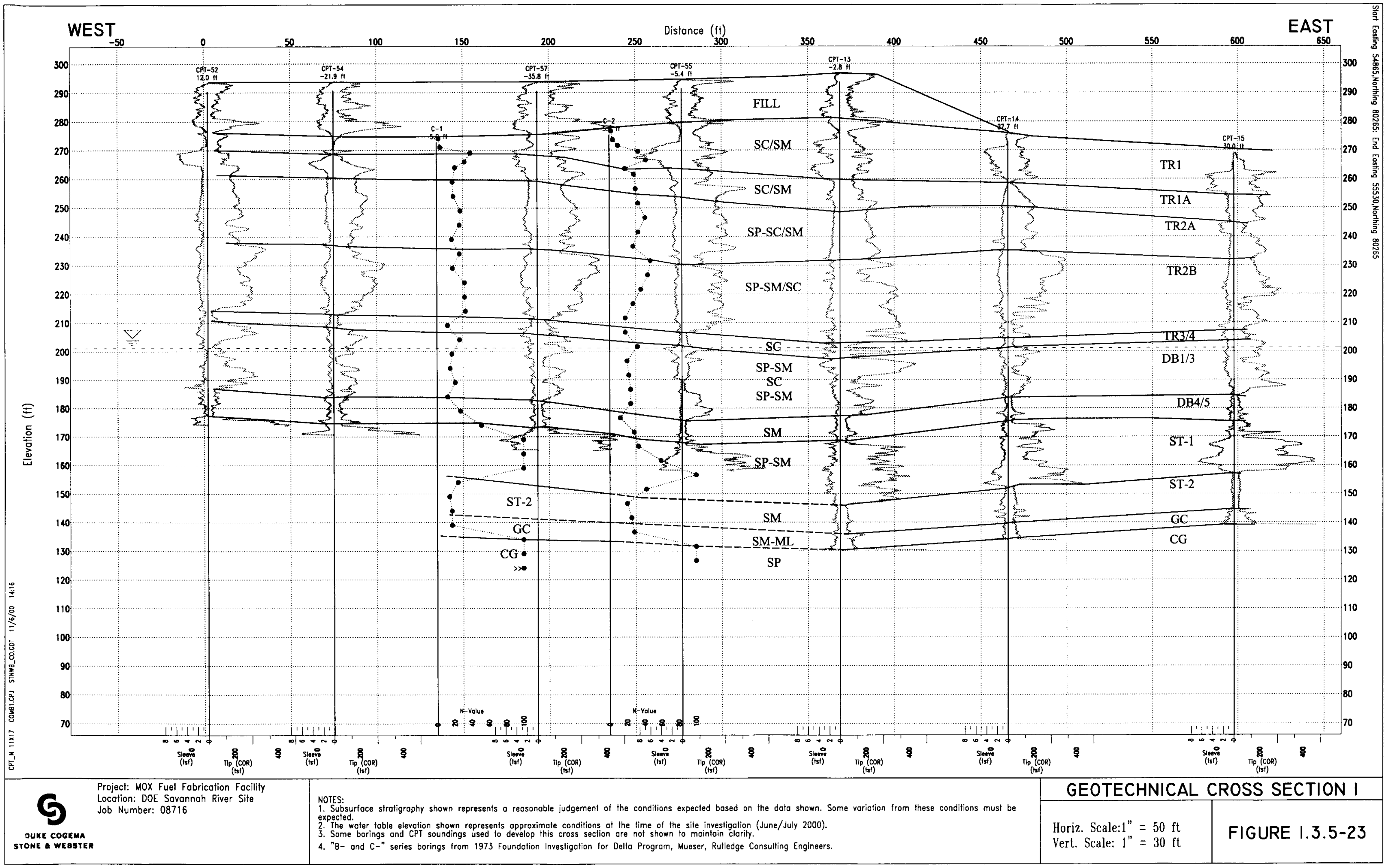


Figure 1.3.5-24. Geotechnical Cross Section 2

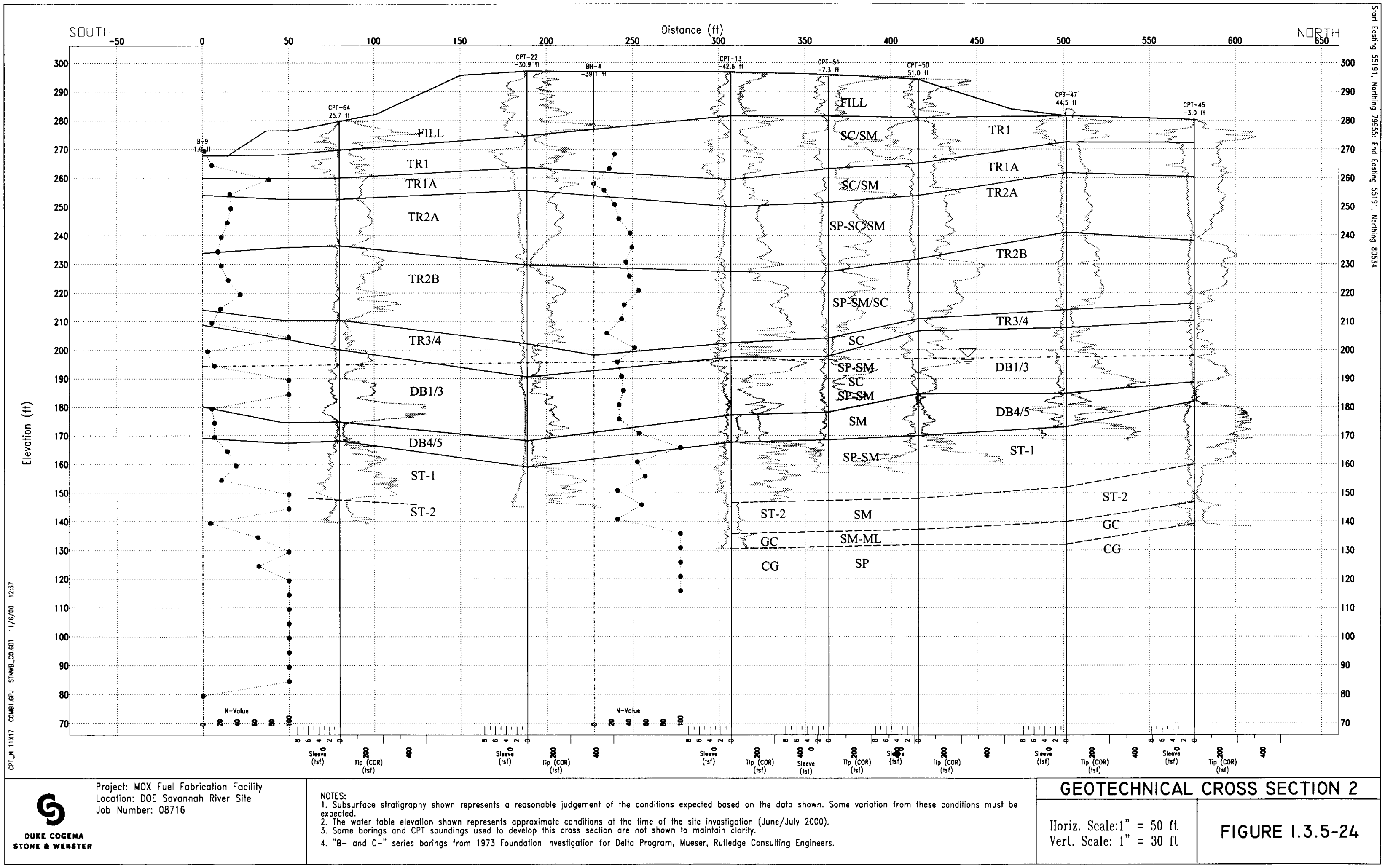
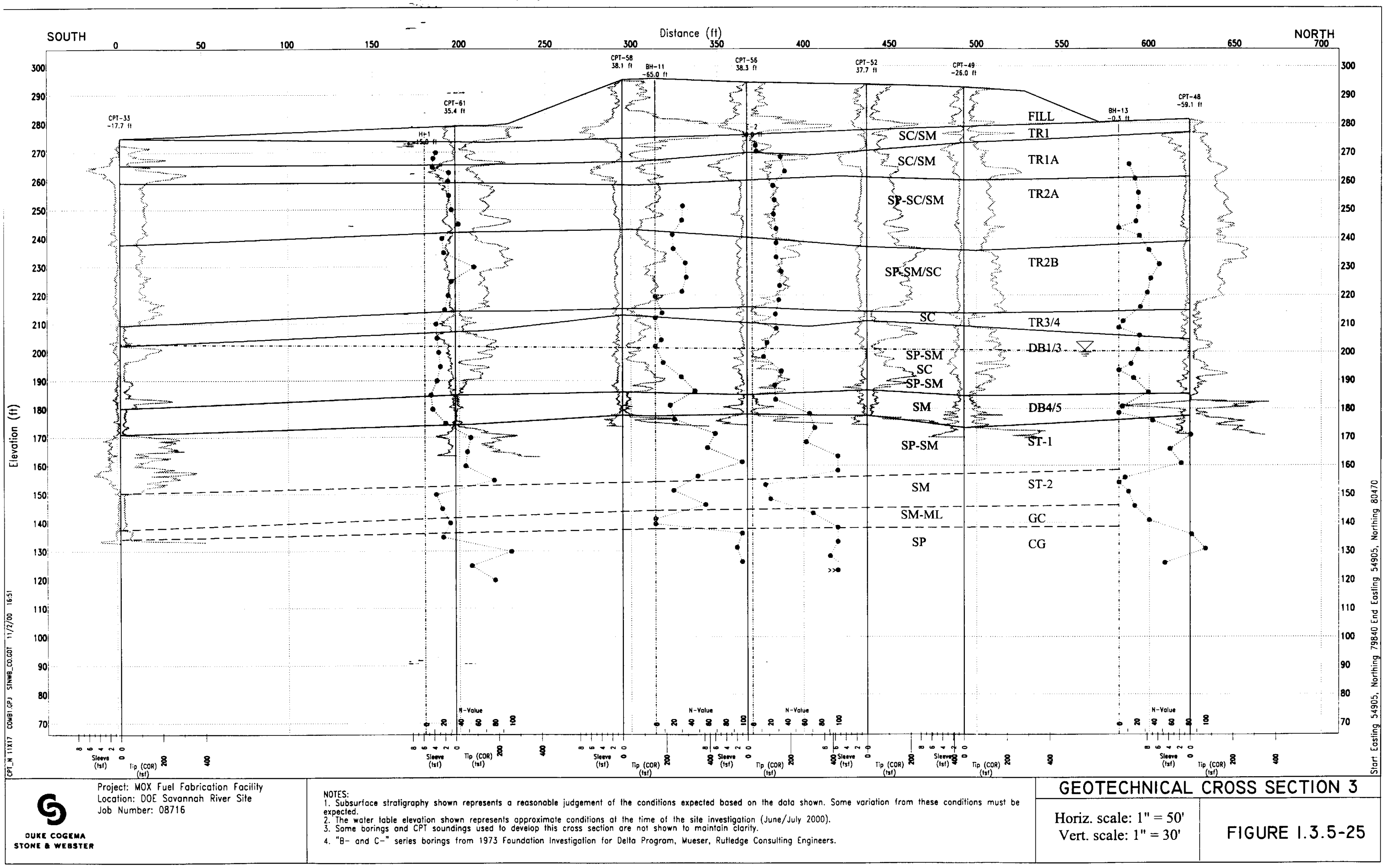


Figure 1.3.5-25. Geotechnical Cross Section 3



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1.3.6 Seismology

Significant studies of the local and regional seismology for SRS have been conducted to support operation of DOE facilities there. The Mixed Oxide (MOX) project has used these studies as a starting point in establishing appropriate design inputs for the MFFF. This section presents criteria that have been developed for DOE facilities at SRS and their application to developing design criteria for the MFFF.

Section 1.6.1 presents a broad description of the historic seismic record of the southeastern United States and SRS. This section also describes in detail the pre-instrumental and post-instrumental seismic records for SRS and the surrounding area. Section 1.6.2 discusses the relationship between geologic structure and seismic sources within the general site region. Section 1.6.3 summarizes the phased development of the seismic criteria for SRS facilities. Section 1.6.4 briefly describes the methodology used for ground motion prediction, earthquake source, path and site assumptions for H Area, and the most recent DE work conducted for SRS. Section 1.6.5 shows the current seismic design criteria for DOE facilities at the SRS. Section 1.6.6 provides a roadmap to the application of the Probabilistic Seismic Hazard Assessment (PSHA) process in the development of SRS site-wide seismic design criteria, and Section 1.6.7 describes the selection of the MFFF Design Earthquake using the technical bases of those site-wide criteria.

1.3.6.1 Earthquake History of the General Site Region

This section presents a broad description of the historic seismic record (non-instrumental and instrumental) of the southeastern United States and SRS. Aspects that are of particular importance to SRS and the MFFF site include the following:

- The Charleston, South Carolina area is the most significant seismogenic zone affecting SRS.
- Seismicity associated with SRS and the surrounding region is more closely related to South Carolina Piedmont-type activity. This activity is characterized by occasional small shallow events associated with strain release near small-scale faults, intrusive bodies, and the edges of metamorphic belts.

1.3.6.1.1 Historic Record

The earthquake history of the southeastern United States (of which SRS is a part) spans a period of nearly three centuries and is dominated by the catastrophic Charleston earthquake of August 31, 1886. The historical database for the region is essentially composed of two data sets extending back to as early as 1698. The first set is comprised of pre-network, mostly qualitative data (1698 to 1974), and the second set covers the relatively recent period of instrumentally recorded or post-network seismicity (1974 to present). A comprehensive catalog was created that successfully merged macroseismic, historical, pre-network data with instrumental, mostly microseismic, post-network data. See Table 1.3.6-1 for a listing of the significant earthquake locations within 200 miles (322 km) of SRS excerpted from this catalog. Today, seismic monitoring results from southeastern seismic networks are cataloged annually in the Southeast U.S. Seismic Network bulletins. This catalog is considered to be the most complete listing of seismic events for the Southeastern U.S. region. Other catalogs are available, and differences exist between catalogs, but the noted differences have no impact on the seismic assessment of SRS.

The information chronicled on earthquakes within the Southeast and the SRS region during the pre-network period consists of intensity data. Intensity refers to the measure of an earthquake's strength by reference to “intensity scales” that describe, in a qualitative sense, the effects of earthquakes on people, structures, and land forms. A number of different intensity scales have been devised over the past century, but the scale generally used in North America and many other countries is the modified Mercalli (MMI) Scale. See Table 1.3.6-2 for a definition of the scale. Using this intensity scale, it is possible to summarize the macroseismic data for an earthquake by constructing maps of the affected region that are divided into areas of equal intensity. These maps are known as isoseismal maps. It was through construction of isoseismal maps that epicenters of pre-network earthquakes were located at or near centers of areas experiencing highest ground shaking intensity. There is considerable uncertainty (up to several tens of miles) in locating the epicenters with this method because it depends heavily upon population density of the region in which the earthquake occurred.

The Charleston, South Carolina area is the most significant source of seismicity affecting SRS, in terms of both the maximum historical site intensity and the number of earthquakes felt at SRS. The greatest intensity felt at SRS has been estimated at MMI VI-VII and was produced by the intensity X earthquake that struck Charleston, South Carolina on August 31, 1886, at 9:50 p.m. local time. An earthquake that struck Union County, South Carolina (about 100 miles [161 km] north-northeast of SRS) on January 1, 1913, is the largest event located closest to SRS outside of the Charleston area. It had an intensity greater than or equal to MMI VII. This earthquake was felt in the Aiken-SRS area with an intensity of MMI II-III. Several other earthquakes, including some aftershocks of the 1886 Charleston event, were felt in the Aiken-SRS area with intensities estimated to be equal to or less than MMI IV.

Several large earthquakes outside the region were probably felt at SRS, including the earthquake sequence of 1811 and 1812 that struck New Madrid, Missouri (about 535 miles [861 km] west-northwest of SRS) and the earthquake that struck Giles County, Virginia (about 280 miles [451 km] north of SRS), on May 31, 1897. The temporal completeness of the existing earthquake catalog is considered to be complete for recent network data to mb = 2.5, the historical period between 1939 and 1977 complete to mb = 4.5, and the historical period between 1870 and 1930 complete to mb = 5.7.

1.3.6.1.1.1 SRS Activity within 50 Miles (80 km) Radius

SRS is located within the Coastal Plain physiographic province of South Carolina. However, seismic activity associated with SRS and the surrounding region displays characteristics more closely associated with the Piedmont province (i.e., a marked lack of clustering in zones). The activity is more characteristic of the occasional energy strain release occurring through a broad area of central Piedmont of the state. See Table 1.3.6-3 for the epicentral locations for events near (within 50 miles [80 km] from the center of the site) SRS.

Each historical event is described below. The numbers in parentheses refer to the numbers on Table 1.3.6-3.

- **1897, May 06, 24, and 27** (1,3,4): These three small earthquakes were reported to have occurred around the farming community of Blackville, South Carolina. They were lightly felt by residents of the town and surrounding farms. No intensity values have been assigned to these events because they have only been mentioned as being felt. When researching local newspapers of the area, the only reference found to any of these small events appeared as a small sentence in the May 13 issue of the *Barnwell People* from Blackville, which said, "Quite an earthquake shock was felt here on last Friday evening at 8:10." No mention of the 24th or 27th events was found in newspapers published shortly following those dates.
- **1897, May 09** (2): This has been documented as a small "lightly" felt event in the area of Batesburg, South Carolina. No intensity values have been assigned to this event.
- **1945, July 26**: This event was felt mostly in the Columbia and Camden, South Carolina areas. Historically, this event has been more closely associated with Lake Murray, near Columbia, South Carolina. However, it was relocated using some instrumental recordings at regional and teleseismic distances. Relocation moved the epicenter some 31 miles (50 km) to an area southwest of Columbia and to within the 50 miles (80 km) radius of interest for this study. This location, though instrumental, seems extremely questionable. An isoseismal map for this event defined the area of greatest intensity (VI) to be near Camden, South Carolina. Newspaper reports from Aiken, Columbia, and Camden in South Carolina the day following the event tend to confirm this original location. In this case, the location indicated from the reports of the intensity experienced is favored over the instrumental location.
- **1972, August 14** (5): This earthquake was reported to have been felt in Barnwell, Bowman, Cordova, Horatio, North, Springfield, and Summerton, South Carolina with an intensity of between I and III. The location of this earthquake also seems tenuous. Although the event was instrumentally located, the location can only be assumed approximate because the nearest station was over 62 miles (100 km) northeast of the computed epicenter. This event may possibly have occurred closer to the Bowman area and outside the area of interest for this study.
- **1974, October 28** (6) **and November 5** (7): These two events were estimated to have occurred in McCormick and southern Edgefield Counties, South Carolina. Magnitudes of 3.0 and 3.7, respectively, were assigned on the basis of felt reports collected at the time. An isoseismal map for the October event shows an elongated isoseismal roughly following the Fall Line with a maximum felt intensity of III-IV. No instrumental locations are available for either of these events.

1.3.6.1.2 Instrumental Record (Post-Network Seismicity)

By the middle of the 20th century, instrumental recordings from a few regional seismographic stations (less than ten for the entire southeastern United States) reduced uncertainty in locating epicenters to fewer than 10 miles (16 km). However, it was not until the early 1970s that the detection and location of earthquakes in the region greatly improved with the installation of seismic networks in South Carolina as well as other regions of the eastern United States.

The first seismic network in the region was deployed by the U.S. Geological Survey (USGS) and the University of South Carolina in 1974. Operation continues today under the management of the University of South Carolina and is known as the South Carolina Seismic Network. It currently consists of some 28 stations strategically located throughout the state. By 1976, a three-station short-period vertical component network was also established at SRS to monitor potential earthquake activity near SRS. A fourth station, consisting of a vertical and two horizontal instruments, was added to the network in 1986.

With the advent of modern seismic network installation, it was possible to estimate local magnitudes from collected data. Magnitudes are more quantitative estimates of an earthquake's size using instrumentally recorded data. They are based on the amplitude of motion on a standard instrument (seismograph) normalized to account for the separation of the instrument and the earthquake. Within South Carolina and the SRS region, the University of South Carolina developed a duration magnitude scale normalized to the worldwide seismic station in Atlanta, Georgia that has been commonly employed since the mid-1970s within South Carolina and the SRS region. Magnitudes reported using the duration scale are approximately equivalent to body wave magnitude. The uncertainty in the instrumentally determined duration magnitudes is about +0.3 magnitude units.

In addition to more accurate determinations of epicenters and magnitudes, a major benefit of instrumentation has been the ability to determine focal depths and focal mechanisms of locally recorded earthquakes. There is a systematic difference between the depths of earthquakes occurring in the Appalachian highlands and those occurring in the Piedmont and Coastal Plain. In the Appalachian highlands, the 90% depth (i.e., the depth above which 90% of all foci lie) is 12 miles (19.3 km), with a peak in the focal depth distributions at 6 to 7 miles (9.6 to 11.3 km). The corresponding depths for Piedmont and Coastal Plain earthquakes are 8 miles (12.9 km) and 4 to 5 miles (6.4 to 8 km), respectively. It has been argued that these depth variations indicate a significant difference in the thickness of the seismogenic crust between the adjacent provinces. Focal mechanism data for the region have been presented by many researchers through the years. Most focal mechanisms for the South Carolina-SRS region can be summarized to indicate thrust or strike-slip faulting, with the direction of the P-axis (inferred to be the direction of maximum horizontal compressive stress, oriented in a northeast-southwest to east-northeast, west-southwest direction).

1.3.6.1.3 Instrumental Locations (Post-Network)

A detailed review of existing data pertaining to instrumentally located earthquake activity within 50 miles (80 km) of SRS has recently been completed. The purpose of the review was to refine as much as possible the locations of reported event locations, both historical and instrumental. Historical activity was addressed above in the previous section, and with the exception of the 1945 event, the number of reported occurrences and locations did not change. Examination of data associated with instrumentally obtained epicenters revealed that many of the reported events would benefit from using a more detailed velocity model developed since the locations were originally noted. Additionally, waveform data not employed in some of the original locations was added from old records of the SRS network and incorporated into the location algorithm. New locations were derived using the HYPOELLIPSE computer program. Repeated trial runs revealed that the most stable locations were obtained when P and discernible S arrivals were used from stations within a 62-mile (100-km) radius of the computed hypocenter.

HYPOELLIPSE provides a multiple crustal structure option for refinement of locations by allowing the use of varying velocity structure models for groups of stations according to their proximity to geologically differing areas of South Carolina. Varying velocity models have been developed using 20 years of seismic refraction surveys completed throughout South Carolina. A total of five velocity models covering the entire state of South Carolina were developed from these data. These five velocity models change from one physiographic province to another and have been applied to each recording station accordingly. Further refinement to reflect the structure of a buried Triassic basin (Dunbarton Basin) lying beneath two SRS stations has also been provided.

Relocation results are presented in Table 1.3.6-3. The $'s represent old locations and #'s represent the new locations. Four events (26 July 1945, 15 November 1978, 16 January 1979, and 07 January 1992) have no # sign associated with them because their revised locations either plotted out of the 50 miles (80 km) radius (26 July 1945 and 07 January 1992), or upon closer inspection were discovered not to be real events at all (15 November 1978 and 16 January 1979). Consequently, these four events have been removed from consideration as reflected in Table 1.3.6-3. Relocations showed improvement in quality estimates. The revised locations show few, if any, changes between $'s and #'s. The depth estimate parameter returned by the HYPOELLIPSE on relocated events remained less than 7.5 miles (12.1 km). However, no relocated event had a depth of less than 1.4 miles (2.3 km), where original estimates had some events with depths at less than 0.6 miles (1 km).

The largest felt event to have occurred within a 50-mile (80-km) radius of SRS is the August 8, 1993 (09:24 UCT, 5:24 a.m. Eastern Daylight Savings Time [EDST]), Couchton earthquake near Aiken, South Carolina (approximately 40 miles [65 km] north of SRS). It was widely felt throughout the region in Williston, New Ellenton, and SRS. The MMI intensity for this event was estimated at IV-V with a duration magnitude of 3.2. No seismic alarms were triggered. The location of this event plotted on the flanks of a localized gravity low, indicating relation to Piedmont-type activity associated with the boundary of a buried intrusive rather than a large-scale regional feature.

1.3.6.1.3.1 Recorded Activity (Regional)

The distribution of eastern United States instrumentally located epicenters essentially coincides with pre-network, historical seismicity. That is, the pattern of historical activity, which is based on larger-magnitude, felt events, is reproduced in the pattern of smaller, instrumentally located events. A non-random spatial distribution of epicenters is noted with patterns that lie parallel as well as transverse to the northeasterly tectonic fabric of the Appalachians. Appreciable seismic activity is displayed trending along the Appalachian highlands (i.e., the Blue Ridge) with other broad trends of activity seen primarily in the Piedmont and Coastal Plain provinces of Virginia, South Carolina and Georgia. These apparent trends led to a zonal interpretation of southeast regional seismicity that includes the Appalachian zone, Virginia zone, and the South

Carolina-Georgia zone. A broader and simpler zonation concept has been developed that includes the dominant regional trend (along Appalachian highlands) and specific zones defined by areas of concentrated activity.

Results obtained from network data within the South Carolina-SRS region identified the Piedmont and Coastal Plain physiographic provinces as two diffuse areas of seismic activity. Through these studies, the Coastal Plain was further divided into three distinct clusters of seismicity that include the Bowman Seismogenic Zone, the MPSSZ, and the Jedburg-Adams Run Seismogenic Zone. The most active zone is the MPSSZ, which is the only one to coincide with the meizoseismal area of the 1886 Charleston earthquake. (Refer to Section 1.3.6.2 for more details on this zone.) Earthquake activity within the Piedmont not associated with reservoir-induced activity can best be characterized by occasional small shallow events associated with strain release near small-scale faults, intrusives, and edges of metamorphic belts.

1.3.6.1.3.2 SRS Onsite Earthquake Activity

Three earthquakes of MMI III or less have occurred with epicentral locations within the boundaries of SRS. On June 9, 1985, an intensity III earthquake with a local duration magnitude of 2.6 occurred at SRS. Felt reports were more common at the western edge of the central portion of the SRS plant site. Another event occurred at SRS on August 5, 1988, with an MMI I-II and a local duration magnitude of 2.0. A survey of SRS personnel who were at the site during the 1988 earthquake indicated that it was not felt at SRS. Neither of these earthquakes triggered seismic alarms (set point 0.002g) at SRS facilities. These earthquakes were of similar magnitude and intensity as several recent events with epicenters southeast of SRS (Table 1.3.6-3).

On the evening of May 17, 1997, at 23:38:38.6 UTC (7:38 pm EDST) an MD ~ 2.3 (Duration Magnitude) earthquake occurred within the boundary of SRS. It was reported as being felt by workers in K Area and by Wackenhut guards at a nearby barricade. A strong motion accelerograph (SMA) located 3 miles (4.8 km) southeast of the epicenter at Gun Site 51 was not triggered by the event. The SMA located approximately 10 miles (16 km) north of the event in the seismic lab building 735-11A was not triggered. The closest instrument to the epicenter (Gun Site 51) is set at a trigger threshold of 0.3% of full scale where full scale is 2.0g (0.006g). The more distant lab SMA is set to trigger at a threshold of 0.1% of full scale where full scale is 1.0g (0.001g).

1.3.6.1.4 Seismic Networks

1.3.6.1.4.1 Local

As discussed above, a short-period seismic network was established at SRS in 1976 with the installation of three single-component vertical stations. In 1987, digital recording capability and a fourth three-component (one vertical and two horizontal) site were added to the network. Other short-period instrumentation has been added through the years to more completely cover the site with the total number of short-period stations currently at eight. In addition to the short-period network, a ten-station SMA network was more recently (1998, 1999) installed throughout the SRS complex.

1.3.6.1.4.2 DOE SMA Network

Ten new SMAs have been installed in selected DOE structures at foundation level, other selected elevations, and in the free-field. In the event of an earthquake of sufficient size to trigger the installed instrumentation, free-field instrumentation data can be used by DOE to compare measured response to the design input motion for the structures and to determine whether the operating basis earthquake has been exceeded. The instruments located at the foundation level and at elevation in the structures can be used to compare measured response to the design input motion for equipment and piping and can be used in long-term evaluations. In addition, foundation-level instrumentation can provide data on the actual seismic input to the mission-critical structures and can be used to quantify differences between the vibratory ground motion at the free-field and at the foundation level. SMA instrumentation is set to trigger at 2.0% full scale with full scale being 1g (i.e., trigger set at 0.02g).

1.3.6.1.4.3 DOE Short-Period Seismic Monitoring Network (1991-Present)

From 1991 to the present, the following short-period instrumentation has been operated and maintained onsite:

- Vertical short-period digital seismic array, which consists of geophones (sensors) placed at different levels within a deep borehole located near the center of SRS to monitor effects of soil column for engineering analysis and design.
- Seven-station continuous-recording short-period telemetered seismic monitoring network for location and depth determination of locally occurring seismic activity.

1.3.6.1.4.4 Regional

To address the regional seismic issues within 150 to 200 miles (241 to 322 km) of SRS, the University of South Carolina operates and maintains the South Carolina Seismic Network, which includes regional statewide stations located east of SRS, as well as a small network of stations surrounding the most significant seismic source zone affecting SRS (the Charleston, South Carolina region). This program serves to complement current ongoing local SRS seismic data and studies by providing access to regional data and independent sources of data and expertise.

1.3.6.2 Relationship of Geologic Structure to Seismic Sources in the General Site Region

Within the southeastern United States, seismicity generally occurs in distinct zones superimposed on a regional background of very low level seismicity. These distinct zones of epicentral distribution are both parallel and oblique to the general northeastern trend of the tectonic structures in the region. As a general result, the relationship between the observed tectonic structures and seismic activity in the region remains unknown. Therefore, in most instances, the seismic sources are inferred rather than demonstrated by strong correlation with geologic structure. This diffuse characteristic of foci suggests the presence of multiple rather than specific seismogenic structural elements such as small-scale faults, intrusive bodies, and edges of metamorphic belts.

In this region, only about 65% of the instrumentally recorded earthquakes have focal depth determined, and only then with modest accuracy of about +/- 3 miles (+/- 4.8 km). About 90% of these earthquakes occur above a depth of 11 miles (17.7 km), and this depth defines the thickness of the brittle seismogenic crust. In the SRS region, the foci peak at about 3 miles (4.8 km) depth, although there is a smaller peak at about 5 miles (8 km).

For this discussion, a seismic zone is defined to extend from the Brevard zone in northwest South Carolina to just northwest of Charleston, South Carolina where another seismic zone has been defined. The length of the zone is about 250 miles (400 km), and the width is 93 miles (150 km) on each side of the Savannah River. This places SRS in about the center of the zone and includes the Consortium for Continental Reflection Profiling (COCORP) seismic reflection lines in Georgia.

SRS seismic reflection data reprocessed by Virginia Polytechnic Institute present a remarkably high-resolution image of the crust from within 65.6 ft (20 m) of the surface to the Moho. The upper crust is highly reflective and is dominated by southeast dipping bands of laminar reflective packages that are correlatable across SRS. Two of the most prominent of these packages appear to correspond to reflections identified in COCORP lines 5 and 8 in Georgia as the Augusta fault and a mid-crustal detachment. The midcrustal detachment at SRS is a discrete mappable southeastern dipping reflection that occurs at 8.7 to 13.7 miles (14 to 22 km) depth. The Augusta fault is denoted by a distinct laminar southeast dipping reflector at 2.2 to 7.4 miles (3.6 to 12 km) depth. In the southeastern portion of SRS, reflections from deformed Triassic-Jurassic strata are evident. These reflections are truncated by a complex southeast dipping package of reflections that may mark the detachment along which the Dunbarton basin formed.

The quality of the reflection seismic data outside of SRS is not as good except for the Appalachian Ultadeep Core Hole (ADCOH) data at the north-northwestern end of the Savannah River Corridor and the COCORP lines 1, 5, and 8 obtained on the Georgia side on the Savannah River. The ADCOH data clearly imaged highly reflective strata of lower Paleozoic age beneath the Blue Ridge allochthon. This interpretation now appears to be generally accepted. A similar seismic signature has also been imaged on COCORP line 5, suggesting that the lower Paleozoic platform rock extends southeastward at least as far as COCORP line 5. If these interpretations are correct, then the master decollement must lie above the highly reflective shelf strata.

Studies of the seismotectonics in central Virginia have shown a correlation between the distribution of hypocenters and seismic reflectors. They suggest that the earthquake activity might be associated with reactivation along existing faults above a major decollement. The seismic reflection data in the Savannah River Corridor also suggest that not only is the seismicity similar to that in central Virginia, but it may also be related to the seismic reflection data in a similar manner. That is, the seismicity is related to reactivation of existing faults above major detachments (Blue Ridge master decollement and August fault), but in general, does not penetrate below the midcrustal reflections until one approaches the East Tennessee seismic zone at the northwestern end of the corridor.

Although there are uncertainties in the determination of hypocentral depths, the earthquakes in the zone do appear to be localized above what is interpreted to be lower Paleozoic platform rock, which is separated by the master decollement from the overlying allochthon. It is reasonable to suggest that the earthquakes have been localized in the more brittle crystalline allochthon rather than in the more ductile underlying Paleozoic platform shelf strata. Indeed, this is generally the case for the seismic zones in the eastern United States. Thus, there does appear to be an association of the seismicity with pre-existing structure in the upper 7.5 miles (12 km) of the brittle crust, which forms the seismogenic zone. This is important in that, for earthquakes with a moment magnitude (Mw) greater than 5.5, the main shock usually occurs near the base of the seismogenic zone. This may then represent the largest earthquakes that possibly could occur in the SRS region due to the limits on size created by the depth of the seismogenic zone.

1.3.6.3 Development of SRS Design Earthquake

This section summarizes the phased development of the seismic criteria for SRS facilities. Probabilistic hazard, deterministic ground motion prediction methodologies, and the DE history for SRS are described. The summary of the evolution of the SRS design basis earthquake provides the necessary background for facility construction that spans four decades. This section also describes DOE seismic criteria. Ground motion prediction methodologies are described in Section 1.3.6.4. Current seismic design guidance is discussed in Section 1.3.6.5.

For engineering design of earthquake-resistant structures, empirically derived seismic response spectra are most commonly used to characterize ground motion as a function of frequency. These motions provide the input parameters used in the analysis of structural response and/or geotechnical evaluation. Response spectra are described in terms of oscillator damping, amplitude, and frequency and are defined as the maximum earthquake response of a suite of damped single-degree-of-freedom oscillators. The response spectra are related to earthquake source parameters, the travel path of the seismic waves, and local site conditions. Over the last two decades, SRS response spectra have evolved from the use of a single scaled record of a western United States earthquake to a composite spectrum that may represent the response of more than one earthquake. In the latter approach, controlling DEs represent a suite of earthquake magnitude and distance pairs that provide the maximum oscillator response in discrete frequency bands. The basis for controlling earthquakes is derived from detailed geologic and seismologic investigations conducted in accordance with 10 CFR Part 100 Appendix A. This approach is typically labeled the "deterministic" approach. This approach does not explicitly incorporate the rate of seismicity or the uncertainty in earthquake source parameters and ground motion.

An alternative to the deterministic approach is the PSHA. The PSHA incorporates the source zone definition and ground motion prediction assessments required for the deterministic approach, but also considers the estimated rates of occurrence of earthquakes and explicitly incorporates the uncertainties in all parameters. This approach predicts the probability of exceeding a particular ground motion value at a location during a specified period of time. This approach is essential for hazard mitigation of spatially distributed facilities having different risk factors. The current DOE criteria used for SRS facilities are probabilistic-based.

For SRS, design spectral shapes are employed for earthquakes of different magnitudes and travel paths. The following principal spectra were developed for SRS using deterministic methodologies or combinations of deterministic and probabilistic methodologies:

- Housner, *Earthquake Criteria for the Savannah River Plant* (Housner 1968)

- Blume, *Update of Seismic Criteria for the Savannah River Plant* (URS/Blume 1982)
- Geomatrix, *Ground Motion Following Selection of SRS Design Basis Earthquake and Associated Deterministic Approach* (Geomatrix Consultants 1991)
- WSRC, *Update of H Area Seismic Design Basis* (Lee 1994)
- WSRC, Savannah River Site Seismic Response Analysis and Design Basis Guidelines (Lee et al. 1997)
- WSRC, *Soil Surface Seismic Hazard and Design Basis Guidelines for Performance Category 1 & 2* SRS Facilities (Lee 1998).

Each of these portrays a step in the evolution of the understanding of the seismic process. The scientific and technical basis used in developing the DE is described herein.

The Housner spectrum was the response of a single record, the Taft record, from the 1952 Tehachippi earthquake. In contrast, the Blume study developed a composite free-field spectrum that enveloped three postulated events: (1) a random local earthquake (<15 miles [<25 km]), (2) a large earthquake originating near Bowman, South Carolina, and (3) a repeat of the 1886 Charleston, South Carolina earthquake. Although different methodologies were used to develop response spectra, the Geomatrix study used the same three earthquake sources except that the 1886 Charleston earthquake was increased slightly in magnitude and moved a few tens of kilometers closer to the site. In both the Geomatrix and Blume investigations, the postulated Bowman earthquake did not control motions at any spectral frequency; consequently, only two controlling events affected the design response spectra: (1) the random local earthquake, and (2) the larger, more distant, Charleston event.

The Housner and Blume spectra were based on western United States strong motion data because strong motion data were unavailable at that time in the eastern United States for earthquake magnitudes and distances necessary for design. Since the Blume study was conducted, ground motion studies have shown that seismic path and site properties are very different between the eastern United States and western United States. Current analytical approaches directly estimate spectra by using southeast U.S. coastal plain conditions to model path effects on wave propagation. Current SRS design basis spectra are based on a hybrid of deterministic and probabilistic approaches.

1.3.6.3.1 Criteria for DOE Facilities

Seismic design criteria for nonreactor DOE facilities are contained in DOE Order 420.1, DOE-STD-1020-94, and DOE-STD-1024-92 (DOE 1995, 1994, 1996a). Additionally, site characterization criteria can be found in DOE STD-1022-94 (DOE 1996c).

Earlier estimates of ground motion for SRS critical facilities generally adopted NRC regulatory guidance provided in 10 CFR Part 100, Appendix A. This deterministic guidance was applied, for example, at K Reactor. However, the more recent seismic evaluations employed the probabilistic guidance contained in DOE-STD-1024-92 and DOE-STD-1023-95 (DOE 1996a, 1996b).

DOE Order 420.1 provides requirements for mitigating natural phenomena hazards that include seismic, wind, flood, and lightning (DOE 1995). DOE-STD-1020-94 defines the performance goals for seismic, wind, tornado, and flood hazards (DOE 1994).

DOE-STD-1021-93 provides guidelines for selecting performance categories of SSCs, for the purpose of NPH design and evaluation (DOE 1996d). This standard recommends general procedures for consistent application of DOE's performance categorization guidelines.

DOE-STD-1020-94 and DOE-STD-1024-92 require the use of median input response spectra that are determined from site-specific geotechnical studies and anchored to peak ground accelerations (PGAs) determined for the appropriate facility-use annual rate of exceedance (DOE 1994, 1996a). Guidance regarding the specific characterization of seismic hazard is found in the Systematic Evaluation Program guidance and DOE-STD-1022-94 (DOE 1996c).

DOE-STD-1024-92 was an interim standard that required deterministic and probabilistic methodologies be used for hazard evaluation and was superseded by DOE-STD-1023-95 (DOE 1996a, 1996b). The guidelines for probabilistic hazard analyses are as follows: (1) sites can use a combined Electric Power Research Institute (EPRI) and Lawrence Livermore National Laboratory (LLNL) result if applicable, or (2) sites can complete a new estimate using site-specific data including definition of source zones, earthquake recurrence rates, ground motion attenuation, and computational methodologies that are spelled out in the Systematic Evaluation Program.

DOE-STD-1023-95 provides guidelines for developing site-specific probabilistic seismic hazard assessments and criteria for determining ground motion parameters for the design earthquakes (DOE 1996b). It also provides criteria for determination of design response spectra. Five performance categories are specified, from Performance Category 0 (PC-0) for SSCs that require no hazard evaluation, to PC-4, a desired performance level comparable to safety-related structures, systems, and components at commercial nuclear power plants. These criteria address weaknesses in prior guidance by specifying Uniform Hazard Spectrum (UHS) controlling frequencies, requiring a site-specific spectral shape and a historic earthquake check, to ensure that the DE contains sufficient breadth to accommodate anticipated motions from historic earthquakes with moment magnitude greater than 6.

The fundamental elements of the criteria for DOE moderate hazard (PC-3) and high hazard (PC-4) facilities are as follows:

1. A PSHA is conducted for the site (or use an existing PSHA that is less than 10 years old).

2. A target DE response spectrum is defined by the mean UHS.

3. Mean UHS shapes are checked by median site-specific spectral shapes, which are derived from deaggregated PSHA earthquake source parameters. The median site-specific spectral shapes are scaled to the UHS at two specific frequencies (average 1 to 2.5 and 5 to 10 Hz).

4. Estimated site-specific ground motions from historical earthquakes (significant felt or instrumental with Mw > 6) are developed using best-estimate magnitude and distance.

5. Spectral shapes are adjusted until DE response spectra have a smooth site-specific shape.
6. Probabilistic assessment of ground failure should be applied if necessary (i.e., wherever there may be instances of liquefaction or slope failure).

Recently, NEHRP-97 (BSSC [Building Seismic Safety Council] 1997) criteria were adopted by WSRC and DOE for evaluation of spectra for PC-1 and PC-2 facilities and structures (Lee 1998). DOE-STD-1023-95 (DOE 1996b) allows the use of building codes and/or alternate design criteria for DOE PC-1 and PC-2 design. The NEHRP design criteria are defined as two-thirds of the maximum considered earthquake ground motion (i.e., two-thirds of the 2,500-year UHS).

1.3.6.3.2 Historical Perspective on Design Earthquakes at the Savannah River Site

Because maximum potential causative fault structures within the Coastal Plain, Piedmont, and Blue Ridge provinces are not clearly delineated by lower-level seismicity or geomorphic features, past guidance prescribes the use of an assumed local earthquake. The magnitude/intensity of this local earthquake is conservatively assumed to be a repeat of the largest historic event in a given tectonic province located at that province's closest approach to the site. Application of this guidance resulted in the definition of two controlling earthquakes for the seismic hazard at SRS. One earthquake is a local event comparable in magnitude and intensity to the Union County earthquake of 1913 but occurring within a distance of about 15 miles (25 km) from the site. The other controlling earthquake represents a potential repeat of the 1886 Charleston earthquake. Selection of these controlling earthquakes for the design basis spectra for the SRS has not changed significantly in over 20 years. However, the assumed maximum earthquake moment magnitude estimates have increased in the more recent assessments of the 1886 Charleston earthquake. In addition, the assumed distance to a repeat of the 1886 Charleston-type earthquake has decreased slightly.

Until the late 1980s, investigations performed for the NRC focused on the uniqueness of the location of the Charleston earthquake, due to a lack of knowledge of a positive causative structure at Charleston. At issue was the possibility of a rupture on any one of the numerous northeast-trending basement faults located throughout the eastern seaboard. Further, there were no obvious geomorphic expressions that might suggest large repeated faulting.

Evidence that defines the Charleston Seismogenic Zone is as follows:

- The detailed analyses of isoseismals following the 1886 Charleston earthquake
- Instrumental locations and focal mechanisms of seismicity defining the 31-mile (50 km) long Woodstock fault lineament, which closely parallels the north-northeast trending isoseismals
- The remote-sensed 8.2-ft (2.5-m) high, 15.5-mile (25-km) long lineament that also parallels the Woodstock fault.

Paleoliquefaction investigations along the coasts of Georgia, North Carolina, and South Carolina identified and dated multiple episodes of paleoliquefaction that constrained the latitude of the episodes. Crater frequency and width are greatest in the Charleston area and decrease in frequency and width with increased distance along the coast away from Charleston.

The following sections contain, for historical reasons, brief summaries of the important deterministic and probabilistic seismic hazard investigations that were conducted at or applied to various facilities at SRS.

1.3.6.3.2.1 Housner

The earliest spectra used at SRS were developed by Housner who used a 5% damped response from the 1952 Taft earthquake (Housner 1968). For a repeat of the Charleston earthquake, Housner predicted 0.1g PGA at SRS and conservatively recommended 0.2g PGA for the DE. These spectra were used in an early evaluation of the seismic adequacy of production reactors at the site but are no longer considered acceptable for design basis analysis.

1.3.6.3.2.2 Blume

Recommended site acceleration and spectra in the Blume analysis were based on conservative assumptions for the occurrence of specific earthquakes (URS/Blume 1982). The anticipated ground motions from those events were developed from recorded earthquakes and synthetic seismograms for those postulated events. A probabilistic seismic hazard evaluation was also performed. Two hypothetical earthquakes consistent in size with earthquakes that have occurred in similar geologic environments were found to control SRS spectra and peak ground motion: (1) a hypothesized site MMI VII local earthquake, causing an estimated site PGA of 0.1g; and (2) a hypothetical MMI X earthquake (1886 Charleston-type), occurring at a distance of 90 miles (145 km) and causing an estimated site PGA of <0.1g. For added conservatism, the site PGA was increased to 0.2g, which corresponded to a site intensity of VIII. The PSHA indicated that the mean annual rate of exceedance of 2×10^{-4}, corresponding to 0.2g, was comparable to those probabilistic hazard studies developed for nearby nuclear power plants.

In the Blume study, the following three seismogenic source regions were considered for ground motion assessment:

- Appalachian Mountains, including the Piedmont and Blue Ridge geologic provinces, which were assessed at a maximum intensity VIII.
- Atlantic Coastal Plain, including SRS, assessed at a maximum intensity VII.
- The Charleston Seismogenic Zone with an epicentral intensity of X. A hypothetical Charleston event was also assumed to occur at Bowman for the purposes of estimating the distance for the attenuation of ground motion.

The length of the 1886 Charleston Seismogenic Zone was estimated as 31 miles (50 km) based on the elongation of the highest intensity isoseismal and on the length and location of the inferred Woodstock fault as determined by instrumental location and mechanisms of earthquakes. A displacement of 78.7 in. (200 cm) was estimated for the Charleston event based on the source dimension and the seismic moment. The source mechanism was assumed to be similar to the mechanisms recorded along the Woodstock fault: steeply dipping, right-lateral, strike-slip fault oriented N10°E.

The estimated PGAs for postulated maximum events were based on the following:

- A local earthquake of MMI VII: a maximum credible earthquake for the Atlantic Coastal Plain
- A Fall Line event, MMI VIII with distance > 28 miles (45 km): a maximum credible earthquake for the Piedmont
- A Middleton Place event of MMI X: a repeat of the Charleston 1886 earthquake
- A Bowman, MMI X: a postulated and considered extremely unlikely occurrence of a 1886 type-event at closest credible distance of 59 miles (95 km).

Blume applied a confidence margin of one intensity unit to the estimates in Table 1.3.6-4, resulting in a site intensity of VIII with a corresponding doubling of the estimated PGA (to 0.2g). Using the PSHA, Blume noted that a doubling of the PGA results in an approximate order of magnitude smaller probability of exceedance. Local and distant earthquake response spectral shapes were derived from statistical analysis of primarily western United States (western) data. The recommended response spectrum, computed from the envelope of the mean spectral shapes, is shown in Figure 1.3.6-11.

1.3.6.3.2.3 Geomatrix (K Reactor)

Geomatrix performed a deterministic analysis in accordance with Section 2.5.2 of NUREG-0800 for K Reactor (Geomatrix Consultants, Inc. 1991). The resulting spectra were developed for a distant Charleston source and a local source. The Charleston source was modeled for Mw 7.5 using the Random Vibration Theory (RVT) model. Site-specific soil data were used to address the impact of local conditions on the spectral content. The local source assumed Mw 5 and used empirical western United States deep-soil strong motion data corrected for eastern United States soil and rock conditions. The recommended response spectra for 5% damping are shown in Figure 1.3.6-11 for the two hypothetical controlling earthquakes, labeled EBE-Distant and EBE-Local, where “EBE” signifies “Evaluation Basis Earthquake”.

The primary uncertainty related to the 1886 Charleston earthquake moment magnitude estimate was the interpretation of intensity, which was derived from reported damage patterns. The fault rupture width was estimated to be 12.4 miles (20 km) based on a range of deepest Coastal Plain hypocenters. The rupture length was determined from regressions of worldwide M_o vs. rupture area. From the rupture dimensions and moment, Geomatrix estimated a stress drop of 65 bars and an average displacement of 157 in. (400 cm).

The Bowman seismicity zone, located in the Coastal Plain province, consists of magnitude 3.5 to 4.0 events occurring along a northwest trend from Charleston. Because of the timing and mechanisms of events, they are not believed to be associated with the Charleston Seismogenic Zone. The largest historical earthquake in the Piedmont Province was the 1913 Union County earthquake having an epicentral intensity of VI-VII. Based on Johnston isoseismal areas, that earthquake was estimated to be Mw 4.5. The largest Appalachian province earthquake was the

1875 Central Virginia event of MMI VII and Mw = 4.8. These earthquakes suggest Mw_{max} of 5.0 for Bowman, but because it was part of a diffuse north-west trend, Geomatrix used 6.0 for conservatism. The Bowman earthquake did not control site motions (similarly to the Blume study) and consequently was not used in specification of design basis motions.

For the local earthquake, the occurrence of a random earthquake within 15.5 miles (25 km) of K Reactor was assumed. With the largest site vicinity events limited to magnitudes within the range of 2 to 3, guidance suggests using the largest historical event in the Piedmont Province: Mw_{max} = 5.0.

Geomatrix developed 5% damped response of the horizontal component from an Mw 7.5, 150-bar, stress drop, Charleston-type earthquake using the parameters described above. The vertical component of motion was estimated to be half the horizontal. See Table 1.3.6-5 for a summary of the source parameters and predicted motions from these earthquakes and Figure 1.3.6-11 for the recommended, 5% damped, response spectrum (labeled "EBE-Distant").

Statistics for the local earthquake were selected using strong motion records from earthquakes of Mw 5.0 ± 0.5 within 15.5 miles (25 km) of the epicenter. The local earthquake spectral shape was scaled in accordance with DOE-STD-1024-92 guidance (DOE 1996a), and the recommended, 5% damped, response spectrum is shown in Figure 1.3.6-11 (labeled "EBE-Local").

1.3.6.3.3 Evaluation Basis Earthquake Spectra

For the 1993 liquefaction studies at the RTF, the design basis envelope spectra contained in the Blume report were not recommended because the spectra were not representative of a specific earthquake. Seismic hazard results show that the site can be characterized by local events, with R <25 km, controlling the PGA, and larger events, at some distance from the site, controlling the peak ground velocity. These results compared favorably with the deterministic analyses performed for the site by Blume (in 1982) and Geomatrix (in 1991).

The controlling earthquakes used in the liquefaction study at the RTF were selected to be consistent with the DOE probabilistic acceptance criteria. A spectral shape was taken from the local event spectrum developed by Geomatrix for K Reactor. The distant event spectrum developed by Geomatrix for the K Reactor was recommended for use unscaled (see Figure 1.3.6-11). The results were then compared to the past deterministic study of Blume and the deaggregated LLNL and EPRI hazard analyses. Induced stresses were calculated for the liquefaction analysis based on the two controlling earthquakes. Separate analysis was warranted based on the difference in shape of the two spectra.

The RTF spectra were later named the EBE and used to support initial geotechnical evaluations for the H-Area In-Tank Precipitation Facility (ITP) and H-Area Tank Farms. The EBE response spectra were used until site-specific spectra could be developed to judge adequacy. The EBE response spectra, which account for local and distant earthquakes, were consistent with DOE criteria and were used for the initial geotechnical evaluation of the RTF.

1.3.6.3.4 WSRC (H Area Spectrum)

Following initial site-specific evaluations performed for the ITP and H Area, a revised spectrum (84th percentile deterministic spectrum) was developed and recommended for structural engineering and geotechnical analyses of facilities in H Area. The geotechnical analyses utilized the bedrock results in a convolution analysis, and the structural engineering groups developed an envelope for use in analysis of SSCs. See Figure 1.3.6-12 for a diagram of the resulting structural design spectrum envelope, identified as the "Interim Site Specific Spectrum."

The fundamental change was to the distant earthquake component. The parameters used to develop 50th and 84th percentile spectra were based on site-specific soil properties and revised stress drop for a Charleston earthquake.

EPRI and LLNL hazard spectra were used to estimate the probability of exceedance of the spectrum. The local event spectrum was unchanged from the evaluation basis earthquake. The resulting local and distant spectra were then enveloped into a surface design spectrum, identified as the "Interim Site Specific Spectrum" in Figure 1.3.6-12.

1.3.6.3.5 WSRC (PC-3 and PC-4 Site-wide Design Spectra)

The PC-3 and PC-4 site-wide design response spectra fully implement DOE-STD-1023-95 (DOE 1996b) and are shown in Figure 1.3.6-13. (Note, the PC-3 spectral shape was revised in 1999, as discussed in Section 1.3.6.5). DOE-STD-1023-95 specifies a broadened mean-based UHS representing a specified annual probability of exceedance (for an SSC performance category) and a historical earthquake deterministic spectrum that ensures breadth of the UHS. For SRS, the deterministic spectrum is represented by a repeat of the 1886 Charleston earthquake. The development of the SRS design basis spectra used a statistical methodology to verify that a mean-based response is achieved at the soil free surface. The design response spectra were intended for simple response analysis of SSCs and are not appropriate for soil-structure interaction analysis or geotechnical assessments.

The EPRI and LLNL bedrock level uniform hazard spectra were averaged and broadened in accordance with DOE-STD-1023-95 (DOE 1996b). Available SRS soil data were used to parameterize the soil shear-wave velocity profile. The parameterization was used to establish statistics on site response for ranges of soil column thickness present at SRS. The mean soil UHS was obtained by scaling the bedrock UHS by the ground motion dependent mean site amplification functions.

The soil data used to develop the site-wide design response spectra incorporate the available SRS velocity and dynamic property data available to about mid-1996. The spectra are based on soil properties and stratigraphy from specific locations at the SRS and are parameterized to represent the variability in measured properties. Because of the potential for variation of soil properties in excess of what have been measured at SRS, the design basis spectra are issued as “committed” for DOE facilities at SRS in accordance with the WSRC quality assurance program. Each project is required to confirm the applicability of the site-wide spectra to its project site. The soil parameters available at the specific site or facility where it is being used must be reviewed and determined to be consistent with the data parameterized in the study. The results of this review for the MFFF site are provided in WSRC 2003.

DOE PC-3 (pre-1999) and PC-4 design spectra are compared to the SRS interim spectrum and the Blume envelope spectrum. There is broad general agreement between the PC-3 and interim spectral shapes. The SRS interim spectrum shape, however, is significantly more conservative in the frequency range of 0.5 to 2.0 Hz compared to the PC-3 spectrum, because the interim spectrum enveloped the 84^{th} percentile Charleston deterministic spectrum rather than the 50^{th} percentile, as required by DOE-STD-1023-95 (DOE 1996b).

Comparisons of the Blume 0.20g anchored spectrum to the PC-3 and PC-4 design spectra indicate significant shape differences. The Blume spectrum was derived from deep-soil recordings of western U.S. earthquakes and is not representative of eastern United States spectral shapes. The PC-4 design spectrum shows a generally more broadened shape as compared to the Blume spectra (see Figure 1.3.6-15). Low frequencies are enhanced with respect to Blume because the Blume spectrum does not contain the fundamental site resonance (about 0.6 Hz). High frequencies are also enhanced with respect to Blume because of differences between eastern and western United States attenuative properties. Both the PC-3 spectrum and the Blume spectrum have a dynamic amplification of about 2.7 at 3 Hz; however, the significantly larger Blume PGA scaling factor causes the excess (as compared to the PC-3 design basis spectrum) spectral values at the mid-range.

1.3.6.3.6 WSRC (PC-1 and PC-2 Site-wide Design Spectra)

Design spectra guidelines for DOE PC-1 and PC-2 facilities are reported by Lee (1998). The DOE PC-1 and PC-2 design spectra were derived using DOE-STD-1023-95 guidelines and NEHRP-97 (BSSC 1997) design criteria, and they account for the wide range in SRS material properties and geometries including soil shear-wave velocities, uncertainty or range in soil column thickness, and type of basement material. Additional design guidance is contained in the current revision of WSRC Engineering Standard 01060 (WSRC 1999a).

1.3.6.3.7 SRS Site-Specific Probabilistic Seismic Hazard Assessments

An SRS site-specific PSHA is dependent upon the local geological and geotechnical properties at the particular site or facility location. Past PSHAs, specifically those conducted by EPRI (NEI 1994) and LLNL (Bernreuter 1997; Savy 1996) for SRS, did not incorporate these detailed site properties. Consequently, though those bedrock outcrop hazards were appropriate for the site, the soil-surface hazard results were not appropriate for use at the SRS. An SRS site-specific PSHA should account for soil properties derived from site geological, geophysical, geotechnical, and seismic investigations (WSRC 1997c). An SRS site-specific PSHA was developed using EPRI and LLNL bedrock outcrop hazard and SRS site properties including soil column thickness, soil and bedrock shear-wave velocity, and dynamic properties (WSRC 1998).

The bedrock seismic hazard evaluations used for the SRS site-specific soil surface hazard were the EPRI and LLNL results for bedrock for SRS and vicinity (A later evaluation was completed using the U.S. National Map bedrock seismic hazard (WSRC 1999d; Frankel et al. 1996)). These evaluations did not revise or confirm in any way the experts' evaluations of activity rates, seismic source zonation, or the decay of ground motion with distance used in the LLNL or EPRI seismic hazard assessments. The analysis results in an SRS site-specific hazard evaluation for a soil site by continuing the hazard from bedrock to the soil surface using detailed soil response functions. Earthquake magnitude and ground motion level dependence of the site response are accommodated by applying site response functions consistent with the distribution of earthquake magnitude and ground motion levels obtained from deaggregating the bedrock uniform hazard spectrum.

Frequency and ground motion level dependent soil amplification functions developed in WSRC-TR-97-0085 (Lee et al. 1997) were used to account for the observed variations in properties throughout SRS, including soil column thickness, stratigraphy, shear-wave velocity, and material dynamic properties, as well as basement properties. Soil amplification functions (frequency-dependent ratio of soil response to bedrock input) were derived in WSRC-TR-97-0085 (Lee et al. 1997) by performing a statistical analysis of the response of bedrock spectra through realizable soil columns bounded by the observed variations in soil-column properties over SRS. Ground motion level-dependent distributions of soil amplification functions were derived for each of six soil categories: three on crystalline basement and three on Triassic basement. Those soil amplification function distributions were used to compute soil surface hazard.

The methodology used to compute soil surface hazard was formalized by Cornell (1997). The technique is to difference the bedrock hazard deaggregation for a suite of bedrock motions and sum the probability of exceedance of surface motions using the appropriate magnitude and ground motion level-dependent soil/rock transfer functions. The approach yields soil surface hazard that would be obtained from correctly applying local site soil transfer functions to the ground motion attenuation model used in a PSHA. The analysis is repeated at the oscillator frequencies available in the bedrock hazard deaggregation and for each soil column thickness and bedrock type. The envelope of the hazard curves is taken from the soil and bedrock categories. The curves representing hazard at the top of the soil column for oscillator frequencies of 1, 2.5, 5, and 10 Hz in terms of spectral velocity are shown in Figure 1.3.6-16. Figure 1.3.6-24 presents the same hazard information in terms of spectral accelerations at the soil surface.

High and low probability extrapolations of bedrock hazard curves were made to meet the ranges of probability required for engineering risk assessments (annual probabilities as low as 10^{-7} were considered). Soil surface hazard results computed in the range of bedrock hazard extrapolations are considered more uncertain. Consequently, computed ground surface hazard curves for annual probabilities greater than about 10^{-2} or less than about 10^{-6} should be used with caution. These results were computed using a 3σ truncation on the ground motion probability of exceedance and a lower bound of 0.5 on the soil amplification function.

PSHAs developed for SRS prior to the LLNL and EPRI studies, as well as the hazard derived from the combination of the original EPRI and LLNL soil surface hazard, were derived for PGA only and did not use SRS soils data. Historically, engineering applications and earthquake design used PSHAs that were PGA-based, a practice that has diminished over the last 20 years because of improved interpretations from broader-band seismic recordings and the better understanding of the broad-band nature of seismic hazard. The engineering use of PGA PSHAs is neither recommended nor consistent with DOE-STD-1023-95 (DOE 1996b).

1.3.6.4 SRS Ground Motion Prediction Methodologies

This section briefly describes the methodology for current ground motion prediction and earthquake source, path, and site assumptions used for H Area, the most recent DE work conducted for SRS.

1.3.6.4.1 Random Vibration Theory (RVT) Modeling

To model ground motion, an RVT model (also called Band Limited White Noise) is used to estimate ground motion for the distant Charleston-type event. The RVT model is widely accepted and, with proper parameterization, is found to predict ground motion as successfully as empirically derived relationships. Because of the model's simplicity, computational speed, ability to parameterize source, geometrical spreading, crustal attenuation, and site response, it is ideally suited to quantifying ground motion. The RVT methodology appears to be well suited in geologic environments where empirical strong motion data may not exist in the earthquake magnitude and distance ranges of interest. Nonlinear wave propagation within the soil column is accounted for by using a computer modeling program, such as SHAKE, or equivalent approach.

1.3.6.4.2 Earthquake Source Parameters

This section discusses the earthquake source parameter uncertainty affecting ground motion prediction for SRS. The distance from the SRS site center to the 1886 Charleston MMI X isoseismal contour is approximately 74.5 miles (120 km). The distance from the center of the SRS to the southern end of the Woodstock fault is approximately 80.8 miles (130 km) and to the center of the 1886 MMI X isoseismal, close to Middleton Place, is approximately 90 miles (145 km). Blume used 145 km as the distance from the SRS center to the 1886 Charleston earthquake epicenter (URS/Blume 1982). Current ground motion studies for the SRS analyze for a recurrence of the 1886 event at a distance of 74.5 miles (120 km). For estimates of median ground motions for a recurrence of the 1886 earthquake, a source distance of 74.5 miles (120 km) is conservative since the center of the isoseismal zone is at a distance of approximately 90 miles (145 km).

For simplicity, the RVT models of ground motion assume a point source. The effects of focal depth and crustal structure on predicted ground motion are described by Lee (1994).

The distance and stress drop effects on rock motion predictions for a repeat of the Charleston Mw 7.5 event were described by Lee (1994). The 100-150 bar range in stress drop is a probable range for the median value of an eastern United States earthquake. Somerville et al. (1987) found a value of 100 bars as the median stress drop for eastern United States earthquakes. The EPRI report *1993 Guidelines for Determining Design Basis Ground Motions* (EPRI 1993) estimated a value of 120 bars as a median for stress drop, from data with reported stress drops in the range of 20 to 600 bars.

Prior ground motion studies for SRS have used expected or median stress drops of 100 to 150 bars for a Charleston-type event. Peak ground motion is sensitive to the selection of stress drop.

The 1886 isoseismal data are consistent with ground motion models that have a moment magnitude of 7.3; i.e., slightly less than the Mw of 7.5 used by Geomatrix to model the Charleston source in developing the recommended response spectra for design of the K Reactor (Geomatrix 1991), but with a corresponding higher stress drop. The favored median model for the SRS uses a Mw of 7.3 at 74.5 miles (120 km) and a stress drop of 150 bars.

1.3.6.4.3 Bedrock and Crustal Path Properties

Ground motion estimates used a modified Herrmann crustal model developed from surface wave dispersion from Bowman, South Carolina to Atlanta, Georgia (Table 1.3.6-6).

For geometrical attenuation, a plane-layered crustal model approximation is used that accounts for the post-critical reflection. The effect of this approximation was to decrease the attenuating loss between about 49.7 to 74.5 miles (80 to 120 km). Using a point source and the local crustal structure for the Charleston event, the attenuation model predictions were found to be sensitive to source depth and source distance.

For development of the RVT rock spectra, anelastic attenuation is accounted for in two ways: (1) the crustal path operator, Q, which is frequency-dependent; and (2) the site-dependent factor, Kappa, which is related to Q by H/(Vs*Qs), where Qs is the average quality factor over a several kilometer range of the near-surface rock. The preferred Q model for these investigations is presented in the EPRI report, *1993 Guidelines for Determining Design Basis Ground Motions* (EPRI 1993).

The ranges of the rock site attenuation operator Kappa were estimated to be 0.010 to 0.004 seconds with a median of 0.006 seconds (EPRI 1993). RVT calculations for the SRS ground motion predictions used the median value of 0.006 seconds for Kappa.

For SRS ground motion predictions, bedrock properties underlying most of the SRS facilities were assumed to be uniform with a Vs of approximately 11,500 fps (3.4 km/sec). For facilities situated above the Triassic rift basin (Dunbarton basin), filled with 1.8 miles (3 km) of sedimentary rock, a Vs estimated to be 8,000 fps (2.4 km/sec) was used. This basin is surrounded by crystalline rock. For a first approximation to the ground motion effects of the basin, a one-dimensional plane-layer model was used to approximate the effect of contrasting velocities.

1.3.6.4.4 Soil Properties

SRS is located on soils (sedimentary strata) ranging in thickness from 600 to 1,500 ft (180 to 460 m) overlying crystalline or Triassic basement. A site-wide design response spectrum must account for the range and variability in SRS soil properties. Deep stiff soils, such as those present at SRS, severely condition bedrock spectra by frequency-dependent amplification or deamplification. Depending upon the frequency and amplitude of the bedrock motion, the key soil properties controlling the soil spectrum are the soil column thickness, the dynamic properties (strain dependent shear-modulus ratio and damping), low-strain soil shear-wave velocity structure, and impedance contrast with the basement (i.e., bedrock).

To accommodate the range of shear wave-velocity in the soil column, a database of velocity profiles was compiled for SRS. This database contains the range of soil and rock shear-wave velocities available from various borings and seismic surveys that have been conducted at SRS using seismic cross-hole, down-hole, velocity logger, and refraction techniques. The shallow profiles database for SRS is based primarily on site-specific seismic CPT with pore pressure measurement soundings (SCPTU). See Figure 1.3.6-17 for an example of SCPTU shear-wave velocity profile. Other velocity profiles consist of cross-hole and down-hole seismic surveys. The deeper soil profiles are based on measurements made in five deep boreholes drilled to basement at SRS.

Other, more numerous, deep holes are used for stratigraphic purposes and to estimate the elevation of the top of bedrock. Nearly all of the velocity data are from the SRS F, H, A, K, and L Areas, and the proposed New Production Reactor site.

Basement shear-wave velocities are estimated from compressional-wave velocities measured at SRS using seismic refraction techniques. These data show that there is a significant shear-wave velocity contrast in the SRS basement between the Dunbarton Triassic basin rock and crystalline rock. The Pen Branch fault is the demarcation for basement contrasts in velocity.

Predicted peak soil strains for SRS are sufficient to exceed the linear range of the constitutive relations (stress-strain). Consequently, laboratory testing of site-specific soil samples was required for reliable ground motion prediction of critical facilities.

Normalized shear modulus and damping ratio versus shear strain relationships were developed for specific stratigraphic layers. Stratigraphic formation identification and their corresponding dynamic properties were developed specifically for SRS by K.H. Stokoe of the University of Texas (Stokoe et al. 1995; Lee 1996).

Stokoe et al. compiled a dynamic soil property database from available SRS reports on dynamic soil properties and new dynamic measurements made by the University of Texas. The SRS areas from which data were obtained are as follows:

1. Area of the Pen Branch Fault Confirmatory Drilling Program
2. H Area ITP
3. H Area RTF
4. H Area Building 221-H
5. Proposed New Production Reactor site
6. Par Pond Dam
7. K Reactor Area
8. Burial Ground Expansion
9. L Reactor Area
10. L Area Cooling Pond Dam
11. F Area Sand Filter Structure.

These 11 areas represent eight general locations at SRS.

Figure 1.3.6-19 presents the recommended hysteric damping vs cyclic shear strain by formation. These curves form the basis for the dynamic properties used in the site response analyses performed for SRS facilities since 1996.

1.3.6.4.4.1 Velocity Model Parameterization

An SRS generic shear-wave velocity profile was developed from the location-specific data and includes randomness in both stratigraphic layer thickness and velocity. Because the area-specific simulations were generally consistent with the generic simulations, the SRS generic (site-wide) simulation is applied to all areas of SRS. There is no significant reduction in the site amplification variability by applying area-specific velocity model simulations for ground motion evaluations.

1.3.6.5 Current SRS Design Response Spectra

The WSRC Civil/Structural Committee reviewed the DOE PC-1 and PC-2 design response spectra and recommended to the Engineering Standards Board that the current Uniform Building Code be used for the Site Engineering Standard (WSRC 1999a). The basis for the decision was that the Uniform Building Code was more conservative than the WSRC (Lee 1998) spectra.

The current DOE PC-3 and PC-4 site-wide design response spectra are based on *Savannah River Site Seismic Response Analysis and Design Basis Guidelines* (Lee et al. 1997) developed in 1997 and incorporate variability in soil properties and soil column thickness. Following the development of PC-3 and PC-4 design basis spectra and the PC-1 and PC-2 design basis spectra, additional conservatisms were applied to the PC-3 spectral shape at high and intermediate frequencies. The shape change was incorporated in WSRC Engineering Standard 01060 (WSRC 1999a). The shape change, illustrated in Figure 1.3.6-20, increased the low-frequency (0.1 to 0.5 Hz) and intermediate-frequency (1.6 to 13 Hz) portions of the PC-3 design basis spectrum.

1.3.6.6 Summary of Methodology for Development of SRS Site-wide Probabilistic Seismic Hazard Assessment (PSHA)

A disciplined, systematic approach was used to develop the PC-3 and PC-4 site-wide design spectra, which included the contributions of national and international consultants and oversight groups and panels to validate the procedures and results. The resulting baseline data were used for selection of seismic design bases for the MFFF.

1.3.6.6.1 General

The development of the SRS PC-3 and PC-4 seismic design spectra that form the technical basis for selecting the MFFF Design Earthquake is documented in WSRC 1997c. The multi-discipline WSRC Site Geotechnical Services (SGS) Department, formed in 1992 to provide centralized geological, seismological, geotechnical (GSG), and geo-environmental services for SRS, uses modern, comprehensive, accurate GSG data and models. WSRC performed work in support of the MFFF in accordance with the WSRC Quality Assurance (QA) program and Criterion 1-6 and 15-18 of ASME/NQA-1-1989. MOX Services has approved WSRC as a supplier of services.

The section entitled Management Measures of the License Application provides additional details regarding quality control and review.

"Tier 1" documentation includes the reports prepared by WSRC's SGS in response to site-wide geoscience activities, including ground motion initiatives, and in support of critical mission facilities. For example, WSRC 1997c is an example of a report prepared in support of a site-wide initiative to develop seismic design spectra using a PSHA approach with a deterministic historical check. Other reports related to ground motion include WSRC 1998 and WSRC 1999d. National and international experts in geology, seismology, and geotechnical engineering supported the preparation of these reports.

"Tier 2" documentation consists of the much larger body of background information maintained by SGS that comprises the analysis documentation and the results of reviews by various oversight groups and panels. These documents are prepared and checked in accordance with WSRC procedures. WSRC 2003, which demonstrates that the soil properties at the MFFF site fall within the range used to develop the SRS PC-3 and PC-4 seismic design spectra, is an example of "Tier 2" documentation. "Tier 2" documentation also includes the records of reviews by independent oversight groups and panels.

In addition to peer reviews conducted in development of the SRS site-wide criteria, MOX Services also initiated a series of peer reviews of appropriate technical topics during the development of the MFFF design. The MFFF Structural Consulting Board (SCB) was formed and chartered to provide senior oversight for overall MFFF design approaches and to perform periodic reviews of in-process results. The SCB included recognized industry experts, as well as subject matter experts from within the MOX Services companies. SCB members have been involved in the selection of the design bases for the MFFF, and have concurred in their selection. Similarly, the *MOX Fuel Fabrication Facility Site Geotechnical Report* (DCS 2005) was the subject of a detailed peer review by a panel of industry experts.

1.3.6.6.2 Comparisons with Other PSHA Studies

Section 4.0 of NUREG/CR-5250 (Bernreuter et al. 1989) compares the results of seismic hazard characterization of 69 nuclear power plant sites east of the Rocky Mountains with previous PSHA results from LLNL and others. The comparisons show good agreement.

WSRC 1999d provided a comparison of the UHS derived from the computed site-specific hazard (referred to as USGS soil surface hazard) to the NEHRP (BSSC 1997) spectrum for the SRS. This comparison was of particular interest for deep-soil eastern U.S. sites, because it compared a building code design spectrum to a site-specific spectrum using the same hazard model and identical criteria. WSRC also compared the SRS site-specific bedrock hazard with the USGS hazard, corrected to account for SRS conditions (Frankel 1999; WSRC 1999d). The USGS hazard was prepared for use in building codes, and not for use in developing seismic hazard input for nuclear facility design. Since the DOE- and NRC-accepted hazard definitions are the EPRI and LLNL hazards, WSRC has maintained the site-wide criteria developed using those hazards, and MOX Services has accepted those criteria as inputs for selecting the MFFF Design Earthquake.

1.3.6.6.3 PSHA Methodology

A PSHA incorporates the source zone definition and ground motion prediction assessments required for a deterministic approach, but also considers the estimated rates of occurrence of earthquakes, and explicitly incorporates the uncertainties in all parameters. This approach predicts the probability of exceeding a particular ground motion value at a location during a specified period of time. This approach is useful for hazard mitigation of spatially distributed facilities having different risk factors. Details of implementation of PSHA methodology are provided in WSRC 1997c and WSRC 1998. DOE STD-1023-95 (DOE 1996b), which outlines overall guidelines for developing a site-specific PSHA, is discussed in Section 1.3.6.3.1, and the SRS site-specific PSHAs are discussed in Section 1.3.6.3.7.

1.3.6.6.4 PSHA Results

The PC-3 and PC-4 site-wide design spectra implement DOE-STD-1023-95 (DOE 1996b), which specifies a broadened mean-based UHS representing a specified annual probability of exceedance (for a SSC performance category) and a historical earthquake deterministic spectrum that ensures breadth of the UHS. For SRS, the deterministic spectrum is represented by a repeat of the 1886 Charleston earthquake. The development of the SRS design basis spectra uses a statistical methodology to verify that a mean-based response is achieved at the soil free surface.

The EPRI and LLNL bedrock level uniform hazard spectra were averaged and broadened per DOE-STD-1023-95. Available SRS soil data were used to parameterize the soil shear-wave velocity profile. The parameterization was used to establish statistics on site response for ranges of soil column thickness present at SRS. The mean soil UHS was obtained by scaling the bedrock UHS by the ground motion dependent mean site amplification functions.

The soil data used to develop the site-wide spectra incorporate the available SRS velocity and dynamic property database available to about mid-1996. The spectra are based on soil properties and stratigraphy from specific locations at the SRS, and are parameterized to represent the variability in measured properties. Because of the potential for variation of soil properties in excess of what have been measured at the SRS, the design basis spectra are issued as a site-wide commitment for DOE facilities. Each project is required to confirm the applicability of the site-wide spectra to its project site. As discussed in Sections 1.3.5 and 1.3.6.7, the analysis of the site-specific subsurface conditions at the MFFF site indicates that the geology and soils present at the MFFF site are consistent with subsurface conditions found throughout SRS and F Area. Therefore, the SRS site-wide hazard can be used for the MFFF seismic design.

1.3.6.7 Definition of the MFFF Design Earthquake

Previous sections present the bases for establishing seismic criteria for DOE PC-3 and PC-4 SSCs at the SRS. Soil surface hazard relationships (acceleration versus mean annual probability of exceedance) presented in WSRC 1998 are used to evaluate the relative probability of exceedance of the PC-3 and PC-4 accelerations and the accelerations of intermediate spectra.

Figure 1.3.6-20 presents the current PC-3 ground surface design response spectra at 5% damping, and Figure 1.3.6-14 presents the current PC-4 ground surface design response spectra for various levels of damping.

Section 1.3.5 indicates that the geology and soils present at the MFFF site are consistent with subsurface conditions found throughout the SRS and F Area. The *MOX Fuel Fabrication Facility Site Geotechnical Report* (DCS 2005) contains a detailed presentation of the site investigations that have been conducted and the results of site-specific analyses. Section 3.4 of DCS (2005) demonstrates the applicability of the SRS site generic PSHA to the MFFF site. Similarly, independent analyses by WSRC (WSRC 2003) confirmed that the site-wide "committed" criteria are "confirmed" to be applicable to the MFFF site. Sections 5 and 6 of DCS (2005) present the MFFF site subsurface conditions and engineering properties for the MFFF site, respectively. Similarly, as indicated in WSRC 2003, the SRS generic shear-wave velocity profile is appropriate for use at the MFFF site. The analysis of the site-specific subsurface conditions at the MFFF site "confirms" that they are consistent with development of SRS site-wide design spectra and that these can be used as design bases for MFFF seismic design. Consequently, another PSHA specific to the MFFF site is not required.

The MFFF geotechnical data are consistent with the SRS site-specific data used to develop the PC-3 and PC-4 design spectra. The application of the PC-3 and PC-4 design spectra is confirmed to be appropriate for the MFFF site in accordance with WSRC 1997c. Therefore, based on the site-specific MFFF geotechnical data (DCS 2005), the SRS PC-3 and PC-4 design spectra are also MFFF site-specific.

The PC-3 and PC-4 design spectra are conservative spectra with probabilities of exceedance of 5 x 10^{-4}/yr and 1 x 10^{-4}/yr, respectively, based on evaluation of SRS site-specific soil surface hazard curves (WSRC 1997c; WSRC 1998). Because the PC-3 design spectrum is also MFFF site-specific, it has a consistent probability of exceedance (5 x 10^{-4}/yr) at each oscillator frequency and envelops the 5 x 10^{-4}/yr uniform hazard spectrum.

Using the acceleration hazard relationships shown in Figure 1.3.6-24 for each of the four oscillator frequencies (1 Hz, 2.5 Hz, 5 Hz, and 10 Hz) represented in the hazard chart, the spectral acceleration can be read off each of the 5% damped response spectra (Figure 1.3.6-20 for PC-3, Figure 1.3.6-14 for PC-4, and Figure 1.3.6-21 for Regulatory Guide 1.60). These spectral accelerations are used to enter Figure 1.3.6-24 and to read the associated annual mean probability of exceedance. Inverting the annual mean probability of exceedance results in the return period. These surface accelerations represent approximately 2,700-year and 22,000-year surface accelerations at 5 Hz for the PC-3 and PC-4 spectra, respectively.

To achieve safety performance goals (i.e., to ensure that high consequence events are highly unlikely), the MFFF design response spectrum for 5% damping for horizontal motion at the ground surface was conservatively specified to be the Regulatory Guide 1.60 5% spectrum scaled to a PGA of 0.20g. Figure 1.3.6-21 shows that this spectrum falls between the surface spectra for PC-3 and PC-4 facilities. It can be seen that the Regulatory Guide 1.60 spectrum significantly envelops the PC-3 spectrum in frequency ranges of significant structural interest. The 0.2g Regulatory Guide 1.60 spectrum envelops the PC-3 spectrum, and therefore, has even lower probabilities of exceedance than the PC-3 spectrum. The return period of representative acceleration ordinates can be determined in the same way as it was for PC-3 and PC-4, above, and the results are shown in Table 1.3.6-7.

Figure 1.3.6-23 presents a comparison of the 0.2g Regulatory Guide 1.60 spectrum to the soil surface uniform hazard spectrum at four frequencies (1, 2.5, 5, and 10 Hz). It can be seen that at a frequency of 1 Hz, the spectral acceleration for the MFFF design spectrum is less than the 10,000-year UHS. For frequencies of practical structural interest, 2.5, 5 and 10 Hz, the spectral acceleration ordinates for the 0.2g Regulatory Guide 1.60 soil surface design earthquake are greater than the 10,000-year UHS.

Appendix C of DOE 1994 presents an evaluation that shows that a median annual probability of exceedance of 10^{-5} corresponds approximately to a mean annual probability of exceedance of 10^{-4}. By selecting accelerations consistent with a (10,000-year) 10^{-4}/yr mean annual probability of exceedance, this spectrum meets the intent of a 10^{-5}/yr median annual probability of exceedance.

On this basis, a 0.2g Regulatory Guide 1.60 horizontal spectrum was selected as the soil surface design response spectrum for use in the design of MFFF buildings and structures. For evaluation of subsurface conditions, to include liquefaction and dynamic settlements, bedrock motions based on the SRS PC-3 bedrock spectrum were used, scaled so that when amplified through the site soil profile, the resulting surface ground motion was 0.20g PGA.

Initial evaluations of SRS earthquake hazards for the MFFF did not indicate that near-field (closer than 9.3 miles [15 km]) earthquakes would be dominant. WSRC-TR-99-00271, *Computation of USGS Soil UHS and Comparison to NEHRP and PC-1 Response Spectra for the SRS* (WSRC 1999d), indicated that although the near-field earthquakes are not dominant, their contribution is potentially significant. WSRC-TR-2001-00342, *Development of MFFF-Specific Vertical-to-Horizontal Seismic Spectral Ratios* (WSRC 2001) indicated that the vertical component would be greater than the initially selected 2/3 ratio.

ASCE 4-98 recommends that if near-field earthquakes are dominant, the ratio of vertical to horizontal spectral ordinates be taken as, at least, unity for frequencies above 5 Hz, 2/3 for frequencies below 3 Hz, and a transition from 2/3 to 1 for frequencies between 3 Hz and 5 Hz. This is closely and conservatively approximated by the Regulatory Guide 1.60 vertical spectrum scaled to the same 0.2g PGA. Therefore, for the MFFF, the vertical component of earthquake motion at the soil surface was selected as the Regulatory Guide 1.60 vertical spectrum scaled to 0.2g PGA. This results in vertical and horizontal spectra that are consistent with the guidance in ASCE 4-98 and Regulatory Guide 1.60 and appropriately consider the effects of near-field earthquakes. Figure 1.3.6-22 presents the selected MFFF design earthquake response spectra.

Table 1.3.6-1. Significant Earthquakes Within 200 Miles of the SRS with Modified Mercalli Intensities ≥ IV and/or Magnitudes ≥ 3

Date (yr/mm/dd)	Latitude (Deg. N)	Longitude (Deg. W)	Depth (km)	Magnitude(s) *		MMI	Distance (Mile †)
1776/11/05	35.2	83.0				IV	155
1799/04/04	32.9	80.0				V	98
1799/04/11	32.9	80.0				V	98
1799/04/11	32.9	80.0				V	98
1817/01/08	32.9	80.0				V	98
1820/09/03	33.4	79.3				IV	135
1827/05/11	36.1	81.2				IV	198
1851/08/11	35.6	82.6				V	171
1853/05/20	34.0	81.2				VI	57
1857/12/19	32.9	80.0				V	98
1860/01/19	32.9	80.0				V	98
1861/08/31	36.1	81.1				VI	198
1869	32.9	80.0				IV	98
1872/06/17	33.1	83.3				V	97
1874/02/10	35.7	82.1				V	171
1874/02/22	35.7	82.1				IV	171
1874/03/17	35.7	82.1				IV	171
1874/03/26	35.7	82.1				IV	171
1874/04/14	35.7	82.1				IV	171
1874/04/17	35.7	82.1				IV	171
1875/11/02	33.8	82.5				VI	63
1876/12/12	32.9	80.0				IV	98
1879/12/13	35.2	80.8				IV	142
1885/08/06	36.2	81.6				V	203
1885/10/17	33.0	83.0				IV	81
1886/08/27	32.9	80.0				V	98
1886/08/28	32.9	80.0				VI	98
1886/08/28	32.9	80.0				IV	98
1886/08/28	32.9	80.0				IV	98
1886/09/01	30.4	81.7				IV	197
1886/09/01	32.9	80.0			6.9F	X	98
1886/09/01	32.9	80.0				V	98

Table 1.3.6-1 Significant Earthquakes Within 200 Miles of the SRS with Modified Mercalli Intensities ≥ IV and/or Magnitudes ≥ 3 (continued)

Date (yr/mm/dd)	Latitude (Deg. N)	Longitude (Deg. W)	Depth (km)	Magnitude(s) *		MMI	Distance (Mile †)
1886/09/02	32.9	80.0				V	98
1886/09/03	30.4	81.7				IV	197
1886/09/04	32.9	80.0				V	98
1886/09/04	30.4	81.7				IV	197
1886/09/05	30.4	81.7				IV	197
1886/09/06	32.9	80.0				V	98
1886/09/06	32.9	80.0				IV	98
1886/09/08	30.4	81.7				IV	197
1886/09/09	30.4	81.7				IV	197
1886/09/17	32.9	80.0				VI	98
1886/09/21	32.9	80.0				VI	98
1886/09/21	32.9	80.0				V	98
1886/09/27	32.9	80.0				VI	98
1886/09/27	32.9	80.0				V	98
1886/10/09	32.9	80.0				IV	98
1886/10/09	32.9	80.0				IV	98
1886/10/09	32.9	80.0				V	98
1886/10/22	32.9	80.0				VI	98
1886/10/22	32.9	80.0				VII	98
1886/10/23	32.9	80.0				IV	98
1886/11/05	32.9	80.0				VI	98
1886/11/28	32.9	80.0				IV	98
1887/01/04	32.9	80.0				V	98
1887/03/04	32.9	80.0				IV	98
1887/03/17	32.9	80.0				V	98
1887/03/18	32.9	80.0				IV	98
1887/03/19	32.9	80.0				IV	98
1887/03/24	32.9	80.0				IV	98
1887/03/24	32.9	80.0				IV	98

Table 1.3.6-1 Significant Earthquakes Within 200 Miles of the SRS with Modified Mercalli Intensities ≥ IV and/or Magnitudes ≥ 3 (continued)

Date (yr/mm/dd)	Latitude (Deg. N)	Longitude (Deg. W)	Depth (km)	Magnitude(s) *		MMI	Distance (Mile †)
1887/03/28	32.9	80.0				IV	98
1887/04/07	32.9	80.0				IV	98
1887/04/08	32.9	80.0				IV	98
1887/04/10	32.9	80.0				IV	98
1887/04/14	32.9	80.0				IV	98
1887/04/26	32.9	80.0				IV	98
1887/04/28	32.9	80.0				V	98
1887/05/06	32.9	80.0				IV	98
1887/06/03	32.9	80.0				IV	98
1887/07/10	32.9	80.0				IV	98
1887/08/27	32.9	80.0				V	98
1887/08/27	32.9	80.0				IV	98
1888/01/12	32.9	80.0				VI	98
1888/01/16	32.9	80.0				IV	98
1888/02/29	32.9	80.0				V	98
1888/03/03	32.9	80.0				IV	98
1888/03/03	32.9	80.0				IV	98
1888/03/04	32.9	80.0				IV	98
1888/03/14	32.9	80.0				V	98
1888/03/20	32.9	80.0				IV	98
1888/03/25	32.9	80.0				IV	98
1888/04/16	32.9	80.0				IV	98
1888/04/16	32.9	80.0				IV	98
1888/05/02	32.9	80.0				IV	98
1889/02/10	32.9	80.0				IV	98
1889/07/12	32.9	80.0				IV	98
1891/10/13	32.9	80.0				IV	98
1893/06/21	32.9	80.0				V	98
1893/06/21	30.4	81.7				IV	197
1893/07/05	32.9	80.0				IV	98
1893/07/06	32.9	80.0				IV	98

Table 1.3.6-1 Significant Earthquakes Within 200 Miles of the SRS with Modified Mercalli Intensities ≥ IV and/or Magnitudes ≥ 3 (continued)

Date (yr/mm/dd)	Latitude (Deg. N)	Longitude (Deg. W)	Depth (km)	Magnitude(s) *		MMI	Distance (Mile †)
1893/07/08	32.9	80.0				IV	98
1893/07/08	32.9	80.0				IV	98
1893/09/19	32.9	80.0				IV	98
1893/09/19	32.9	80.0				IV	98
1893/09/19	32.9	80.0				IV	98
1893/11/08	32.9	80.0				IV	98
1893/11/08	32.9	80.0				IV	98
1893/12/27	32.9	80.0				IV	98
1893/12/27	32.9	80.0				IV	98
1893/12/27	32.9	80.0				IV	98
1893/12/27	32.9	80.0				IV	98
1893/12/28	32.9	80.0				IV	98
1894/01/10	32.9	80.0				IV	98
1894/01/10	32.9	80.0				IV	98
1894/01/10	32.9	80.0				IV	98
1894/01/30	32.9	80.0				IV	98
1894/02/01	32.9	80.0				IV	98
1894/06/16	32.9	80.0				IV	98
1894/12/11	32.9	80.0				IV	98
1895/01/08	32.9	80.0				IV	98
1895/01/08	32.9	80.0				IV	98
1895/01/08	32.9	80.0				IV	98
1895/04/27	32.9	80.0				IV	98
1895/07/25	32.9	80.0				IV	98
1895/10/06	32.9	80.0				IV	98
1895/10/20	32.9	80.0				IV	98
1895/11/12	32.9	80.0				IV	98
1896/03/19	32.9	80.0				IV	98
1896/08/11	32.9	80.0				IV	98
1896/08/11	32.9	80.0				IV	98
1896/08/11	32.9	80.0				IV	98
1896/08/11	32.9	80.0				IV	98

Table 1.3.6-1 Significant Earthquakes Within 200 Miles of the SRS with Modified Mercalli Intensities ≥ IV and/or Magnitudes ≥ 3 (continued)

Date (yr/mm/dd)	Latitude (Deg. N)	Longitude (Deg. W)	Depth (km)	Magnitude(s) *		MMI	Distance (Mile †)
1896/08/12	32.9	80.0				IV	98
1896/08/14	32.9	80.0				IV	98
1896/08/30	32.9	80.0				IV	98
1896/09/08	32.9	80.0				IV	98
1896/11/14	32.9	80.0				IV	98
1899/03/10	32.9	80.0				IV	98
1899/12/04	32.9	80.0				IV	98
1900/10/31	30.4	81.7				V	197
1901/12/02	32.9	80.0				IV	98
1903/01/24	32.9	80.0				IV	98
1903/01/24	32.1	81.1				VI	85
1903/01/31	32.9	80.0				IV	98
1903/02/03	32.9	80.0				IV	98
1904/03/05	35.7	83.5		4.0F		V	200
1907/04/19	32.9	80.0				V	98
1911/04/20	35.1	82.7				V	141
1912/06/12	32.9	80.0				VII	98
1912/06/20	32.0	81.0				V	94
1912/09/29	32.9	80.0				IV	98
1912/10/23	32.7	83.5				IV	115
1912/11/17	32.9	80.0				IV	98
1912/12/07	34.7	81.7				IV	100
1913/01/01	34.7	81.7				VII	100
1913/04/17	35.3	84.2		3.9F		V	204
1914/03/05	33.5	83.5				VI	110
1914/03/07	34.2	79.8				IV	124
1914/07/14	32.9	80.0				IV	98
1914/09/22	32.9	80.0				V	98
1915/10/29	35.8	82.7				IV	186
1915/10/29	35.8	82.7				V	186
1916/02/21	35.5	82.5				VII	163
1916/03/02	34.5	82.7				IV	106

Table 1.3.6-1 Significant Earthquakes Within 200 Miles of the SRS with Modified Mercalli Intensities ≥ IV and/or Magnitudes ≥ 3 (continued)

Date (yr/mm/dd)	Latitude (Deg. N)	Longitude (Deg. W)	Depth (km)	Magnitude(s) *		MMI	Distance (Mile †)
1916/08/26	36.0	81.0				V	193
1924/01/01	34.8	82.5				IV	118
1924/10/20	35.0	82.6				V	133
1926/07/08	35.9	82.1				VII	184
1928/11/03	36.112	82.828	5.0		4.5N	VI	208
1928/11/20	35.8	82.3				IV	180
1928/12/23	35.3	80.3				IV	160
1929/01/03	33.9	80.3				IV	89
1929/10/28	34.3	82.4				IV	85
1930/12/10	34.3	82.4				IV	85
1930/12/26	34.5	80.3				IV	115
1931/05/06	34.3	82.4				IV	85
1933/12/19	32.9	80.0				IV	98
1933/12/23	32.9	80.0				V	98
1933/12/23	32.9	80.0				IV	98
1934/12/09	32.9	80.0				IV	98
1935/01/01	35.1	83.6				V	170
1938/03/31	35.6	83.6				IV	197
1940/12/25	35.9	82.9				IV	196
1941/05/10	35.6	82.6				IV	171
1943/12/28	32.9	80.0				IV	98
1944/01/28	32.9	80.0				IV	98
1945/01/30	32.9	80.0				IV	98
1945/07/26	33.750	81.376	5.0		4.4F	VI	37
1947/11/02	32.9	80.0				IV	98
1949/02/02	32.9	80.0				IV	98
1949/06/27	32.9	80.0				IV	98
1951/03/04	32.9	80.0				IV	98
1951/12/30	32.9	80.0				IV	98
1952/11/19	32.9	80.0				V	98
1956/01/05	34.3	82.4				IV	85
1956/01/05	34.3	82.4				IV	85

Table 1.3.6-1 Significant Earthquakes Within 200 Miles of the SRS with Modified Mercalli Intensities ≥ IV and/or Magnitudes ≥ 3 (continued)

Date (yr/mm/dd)	Latitude (Deg. N)	Longitude (Deg. W)	Depth (km)	Magnitude(s) *		MMI	Distance (Mile †)
1956/05/19	34.3	82.4				IV	85
1956/05/27	34.3	82.4				IV	85
1956/09/07	35.5	84.0			4.1F	V	206
1957/05/13	35.799	82.142	5.0		4.1F	VI	178
1957/07/02	35.6	82.7	7.0			VI	173
1957/11/24	35.	83.5			4.0F	VI	161
1958/05/16	35.6	82.6				IV	171
1958/10/20	34.5	82.7				V	106
1959/08/03	33.054	80.126	1.0		4.4F	VI	88
1959/10/27	34.5	80.2				VI	119
1960/01/03	35.9	82.1				IV	184
1960/03/12	33.072	80.121	9.0		4.0F	V	88
1960/07/24	32.9	80.0				V	98
1963/04/11	34.9	82.4				IV	122
1963/05/04	32.972	80.193	5.0		3.3M	IV	86
1963/10/08	33.9	82.5			3.2M		67
1964/01/20	35.9	82.3				IV	186
1964/03/07	33.724	82.391	5.0		3.3M		54
1964/03/13	33.193	83.309	1.0	4.4P	3.9M	V	97
1964/04/20	33.842	81.096	3.0		3.5M	V	51
1965/09/09	34.7	81.2			3.9M		103
1965/09/10	34.7	81.2			3.0M		103
1965/11/08	33.2	83.2			3.3M		91
1967/10/23	32.802	80.221	19.0	3.8P	3.4N	V	88
1968/07/12	32.8	79.7				IV	116
1968/09/22	34.111	81.484	1.0	3.7P	3.5M	IV	60
1969/05/09	33.95	82.58			3.3N		73
1969/05/18	33.95	82.58		3.5N			73
1969/12/13	35.036	82.846	6.0		3.7M	IV	142
1970/09/10	36.020	81.421	1.0		3.1N	V	191
1971/05/19	33.359	80.655	1.0	3.4P	3.7N	V	57
1971/07/13	34.76	82.98		3.8N		VI	130

Table 1.3.6-1 Significant Earthquakes Within 200 Miles of the SRS with Modified Mercalli Intensities ≥ IV and/or Magnitudes ≥ 3 (continued)

Date (yr/mm/dd)	Latitude (Deg. N)	Longitude (Deg. W)	Depth (km)	Magnitude(s) *		MMI	Distance (Mile †)
1971/07/13	34.7	82.9		3.0M			124
1971/07/31	33.341	80.631	4.0	3.8N		III	58
1971/08/11	33.4	80.7		3.5N			55
1971/10/09	35.795	83.371	8.0	3.4P	3.7N	V	202
1971/10/22	36.0	83.0		3.3M			205
1972/02/03	33.306	80.582	2.0	4.5P	4.5N	V	61
1972/02/07	33.46	80.58		3.2M		III	62
1972/02/07	33.46	80.58		3.2M		III	62
1972/08/14	33.2	81.4			3.0L	III	14
1973/12/19	32.974	80.274	6.0		3.0M	III	81
1974/08/02	33.908	82.534	4.0	4.3P	4.1N	V	69
1974/10/08	33.9	82.4		3.1P		III	63
1974/10/28	33.79	81.92			3.0L	IV	41
1974/11/05	33.73	82.22			3.7L	II	47
1974/11/22	32.926	80.159	6.0	4.7P	4.3N	VI	88
1974/12/03	33.95	82.50			3.6L	IV	69
1975/04/01	33.2	83.2			3.9M		91
1975/04/28	33.00	80.22	10.0		3.0N	IV	84
1975/10/18	34.9	83.0				IV	138
1975/11/25	34.943	82.896	10.0		3.2N	IV	137
1976/12/27	32.060	82.504	14.0		3.7N	V	97
1977/01/18	33.058	80.173	1.0		3.0N	VI	86
1977/03/30	32.95	80.18	8.0		2.9D	V	87
1977/08/04	33.369	80.699	9.0		3.1N		54
1977/08/25	33.369	80.698	3.4	3.1N	2.8D	IV	54
1977/12/15	32.944	80.167	7.5	3.0N	2.6D	V	87
1978/09/07	33.063	80.210	10.0	2.7N	2.6D	IV	83
1979/08/13	35.200	84.353	22.2	3.7N	3.7D	V	206
1979/08/13	33.90	82.54	23.0		4.1D		69
1979/08/26	34.93	82.97	2.0		3.7D		139
1979/09/06	35.298	83.241	10.0		3.2D		168
1979/09/12	35.579	83.941	27.1	3.2N	3.1D	V	208

Table 1.3.6-1 Significant Earthquakes Within 200 Miles of the SRS with Modified Mercalli Intensities ≥ IV and/or Magnitudes ≥ 3 (continued)

Date (yr/mm/dd)	Latitude (Deg. N)	Longitude (Deg. W)	Depth (km)	Magnitude(s) *		MMI	Distance (Mile †)
1979/12/07	33.008	80.163	5.0	2.8N	2.8D	IV	87
1980/06/10	35.458	82.815	0.6	3.0N	2.5D		167
1980/09/01	32.978	80.186	7.0	2.7N	2.9D	IV	86
1981/03/04	35.810	79.737	1.0	2.8N	2.2D	IV	206
1981/04/09	35.514	82.051	0.2	3.0N	3.3D	V	157
1981/05/05	35.327	82.422	10.2	3.5N	3.1D	V	150
1982/01/28	32.982	81.393	7.0	3.4N	2.4D		23
1982/03/01	32.936	80.138	6.7	3.0N	2.8D	IV	89
1982/07/16	34.32	81.55	2.0		3.1D	III	74
1982/10/31	32.671	84.873		2.9N	3.0D	V	193
1982/10/31	32.644	84.894		3.1N	3.1D		194
1982/12/11	32.853	83.532			3.0D		114
1983/01/26	32.853	83.558		3.5N	3.5D		115
1983/03/25	35.333	82.460	11.5	3.2N	3.3D	V	151
1983/11/06	32.937	80.159	9.6		3.3D	V	88
1985/12/22	35.701	83.720	13.4		3.3D		207
1986/02/13	34.76	82.94	5.0	3.5N		V	128
1986/03/13	33.229	83.226	5.0		2.4D	IV	93
1986/09/17	32.931	80.159	6.7		2.6D	IV	88
1987/03/16	34.560	80.948	3.0		3.1D		98
1988/01/09	35.279	84.199	12.2		3.2D	IV	203
1988/01/23	32.935	80.157	7.4		3.3D	V	88
1988/02/18	35.346	83.837	2.4	3.5N	3.3D	IV	192
1989/06/02	32.934	80.166	5.8		2.0D	IV	87
1990/11/13	32.947	80.136	3.4	3.5N	3.2D	V	89
1991/06/02	32.980	80.214	5.0		1.7D	V	84
1992/01/03	33.981	82.421	3.3		3.4D	V	68
1992/08/21	32.985	80.163	6.5	4.1N	4.1D	VI	87
1993/01/01	35.878	82.086	2.3		3.0D		183
1993/08/08	33.597	81.591	8.5	3.2N	2.9D	V	24
1995/04/17	32.997	80.171	8.4	3.9N			86
1998/04/13	34.471	80.603	6.6	3.9N			103

Table 1.3.6-1 Significant Earthquakes Within 200 Miles of the SRS with Modified Mercalli Intensities ≥ IV and/or Magnitudes ≥ 3 (continued)

Date (yr/mm/dd)	Latitude (Deg. N)	Longitude (Deg. W)	Depth (km)	Magnitude(s) *		MMI	Distance (Mile †)
1998/06/05	35.554	80.785	9.4	3.2N			165
1999/03/29	33.064	80.140	10.7		3.0D		87
2000/01/18	32.993	83.214	19.2	3.5N		V	93

Notes:

Table compiled for earthquakes through December 31, 2000. The primary source of data is the Southeastern U.S. Earthquake Catalog maintained by the Virginia Polytechnic Institute Seismological Observatory (VTSO). Secondary sources include the Southeastern Unites States Seismic Network Bulletin, the U.S. Geological Survey Earthquake Database, and the Advanced National Seismic System earthquake catalog. The December 12, 1987, magnitude 3.0 event appearing in the USGS database is not shown above since it is not contained in the VTSO master catalog or bulletins.

† Distances reported in the table are calculated from the center of the SRS. In some instances the distance is slightly more than 200 miles due to the point selected as the center of the SRS.

* In many instances location, depth and magnitude are determined and reported by more than one organization, which results in some minor discrepancies between catalogs. In general, when discrepancies exist, the VTSO data is reported. Generally when body-wave magnitude (m_b) is available it is reported. Magnitudes based on intensity are not reported, but the intensity is reported. In some instances more than one magnitude is reported. The magnitude type code follows the magnitude. The type codes are:

D - Duration magnitude (M_d) from duration or coda length

F - Body-wave magnitude (m_b) from felt area or attenuation data

L - Richter or local magnitude (M_L)

M - Body-wave magnitude (m_b) determined from modified instruments/formulae

N - Body-wave magnitude (m_b) from Lg wave data

P - Body-wave magnitude (m_b) from P wave data

Data from WSRC 2002b.

Table 1.3.6-2. Modified Mercalli Intensity Scale of 1931

Level	Definition
I.	Not felt except by a very few under especially favorable circumstances (I Rossi-Forel Scale).
II.	Felt by only a few persons at rest, especially on upper floors of buildings. Delicately suspended objects may swing (I and II, Rossi-Forel Scale).
III.	Felt quite noticeably indoors, especially on upper floors of buildings, but many people do not recognize it as an earthquake. Standing motor cars may rock slightly. Vibration like passing truck. Duration estimated (III Rossi-Forel Scale).
IV.	During the day felt indoors by many; outdoors by few. At night some awakened. Dishes, windows, doors disturbed; walls made creaking sound. Sensation like heavy truck striking building. Standing motor cars rocked noticeably (IV to V Rossi-Forel Scale).
V.	Felt by nearly everyone; many awakened. Some dishes, windows, and so on, broken, a few instances of cracked plaster, unstable objects overturned. Disturbance of trees, poles, and other tall objects sometimes noticed. Pendulum clocks may stop (V to VI Rossi-Forel Scale).
VI.	Felt by all; many are frightened and run outdoors. Some heavy furniture moved; a few instances of fallen plaster or damaged chimneys. Damage slight (VI to VII Rossi-Forel Scale).
VII.	Everybody runs outdoors. Damage negligible in buildings of good structures; considerable in poorly built or badly designed structures; some chimneys are broken. Noticed by persons driving motor cars (VIII Rossi-Forel Scale).
VIII.	Damage slight in specially designed structures; considerable in ordinary substantial buildings with partial collapse; great in poorly built structures. Panel walls thrown out of frame structures. Fall of chimneys, factory stacks, columns, monuments, and walls. Heavy furniture overturned. Sand and mud ejected in small amounts. Changes in well water. Disturbs persons driving motor cars (VIII+ to IX Rossi-Forel Scale).
IX.	Damage considerable in specially designed structures; well-designed frame structures thrown out of plumb; great in substantial buildings with partial collapse. Buildings shifted off foundations. Ground cracked conspicuously. Underground pipes broken (IX+ Rossi-Forel Scale).
X.	Some well-built wooden structures destroyed; most masonry and frame structures destroyed with foundations, ground badly cracked. Rails bent. Landslides considerable from riverbanks and steep slopes. Shifted sand and mud. Water splashed (slopped) over banks (X Rossi-Forel Scale).
XI.	Few, if any, masonry structures remain standing. Bridges destroyed. Broad fissures in ground. Underground pipe lines completely out of service. Earth slumps and land slips in soft ground. Rails bent greatly.
XII.	Damage total. Waves seen on ground surfaces. Lines of sight and level distorted. Objects thrown upward into the air.

Data from WSRC 2000b.

Table 1.3.6-3. Historic Earthquakes Recorded Within 50 Miles (80 km) of the SRS

Event #	Date	Latitude (Deg. N)	Longitude (Deg. W)	Depth (km)	Magnitude
1	05/06/1897	33.3	81.2		Felt
2	05/09/1897	33.9	81.6		Felt
3	05/24/1897	33.3	81.2		Felt
4	05/27/1897	33.3	81.2		Felt
5	8/14/1972	33.2	81.4		3.2
6	10/28/1974	33.79	81.92		3.0
7	11/5/1974	33.73	82.22		3.7
8	9/15/1976	33.144	81.413	4.5	2.4
9	6/5/977	33.052	81.412	3.5	2.7
10	2/21/1981	33.593	81.148	6.6	2.0
11	1/28/1982	32.980	81.390	7.0	3.4
12	6/9/1985	33.223	81.684	5.8	2.6
13	2/17/1988	33.511	81.697	11.7	2.5
14	8/5/1988	33.187	81.629	2.3	2.0
15	7/13/1992	33.480	81.192	7.6	1.9
16	10/2/1992	33.499	81.202	3.0	2.4
17	12/12/1992	33.280	81.833	11.8	1.2
18	6/29/1993	33.465	81.221	4.9	2.2
19	8/8/1993	33.589	81.585	10.2	3.2
20	8/8/1993	33.589	81.581	9.2	1.6
21	9/18/1996	33.692	82.125	2.4	2.8
22	5/17/1997	33.212	81.677	5.4	2.5
23	10/08/2001	33.324	81.667	3.9	2.6
24	10/08/2001	33.319	81.673	4.2	1.0
25	10/08/2001	33.332	81.676	4.2	1.4
26	10/14/2001	33.347	81.663	3.1	0.7

Table 1.3.6-3 Historic Earthquakes Recorded Within 50 Miles (80 km) of the SRS (continued)

Event #	Date	Latitude (Deg. N)	Longitude (Deg. W)	Depth (km)	Magnitude
27	10/15/2001	33.332	81.683	5.0	0.8
28	12/17/2001	33.328	81.675	4.1	1.1
29	12/27/2001	33.331	81.665	3.8	0.1
30	03/06/2002	33.331	81.679	4.6	1.4

Notes:

Table compiled for earthquakes through October 1, 2002.

Locations, depths, and magnitudes for events within 50 miles of the SRS were reevaluated by SRS personnel in 2002. Magnitudes, depths, and locations reported in this table will vary slightly from magnitudes, depths, and locations reported by other sources, including Table 1.3.6-1. The updated magnitudes and locations for events within 50 miles of the SRS will be provided to Virginia Polytechnic Institute Seismological Observatory for inclusion in their database.

Data from WSRC 2002b.

Table 1.3.6-4. Blume Estimated Site Motions for Postulated Maximum Events

Location	Epicentral Intensity (MMI)	R (km)	Site Intensity (MMI)	Site PGA (%g)
Local	VII	0-10	VII	0.10
Fall Line	VIII	45	VI	0.06
Bowman	X	95	VII	0.10
Middleton	X	145	VI-VII	0.075

Data from WSRC 2000b.

Table 1.3.6-5. Geomatrix Estimated Site Motions for Postulated Maximum Events

Location	Magnitude (Mw)	R (km)	Site PGA[a] (%g median, horizontal)
Local	5.0	<25	0.18
Bowman	6.0	80	0.06
Charleston	7.5	110	0.11

[a] 25 Hz

Data from WSRC 2000b.

Table 1.3.6-6. Modified Herrmann Crustal Model

H (km)	Vs (km/s)	Density (g/cc)
5.0	3.75	2.7
9.5	3.76	2.7
14	4.01	2.8
infinity	4.56	3.3

Data from WSRC 2000b.

Table 1.3.6-7. Return Periods for Spectrum Ordinates

Part 1: PC-3 Spectrum (0.16g)

Frequency (Hz)	Sa (g)	Return period (yrs)
1.00	0.250	4,000
2.50	0.375	3,300
5.00	0.375	2,700
10.00	0.360	5,600

Part 2: PC-4 Spectrum (0.23g)

Frequency (Hz)	Sa (g)	Return Period (yrs)
1.00	0.610	37,000
2.51	0.730	23,000
5.01	0.680	22,000
10.00	0.540	36,000

Part 3: 0.2g Regulatory Guide 1.60 Spectrum

Frequency (Hz)	Sa (g)	Return Period (yrs)
1.00	0.300	6,300
2.51	0.620	14,000
5.01	0.570	10,000
10.00	0.480	22,000

Figure 1.3.6-1. Figure Deleted

Figure 1.3.6-2. Figure Deleted

Figure 1.3.6-3. Figure Deleted

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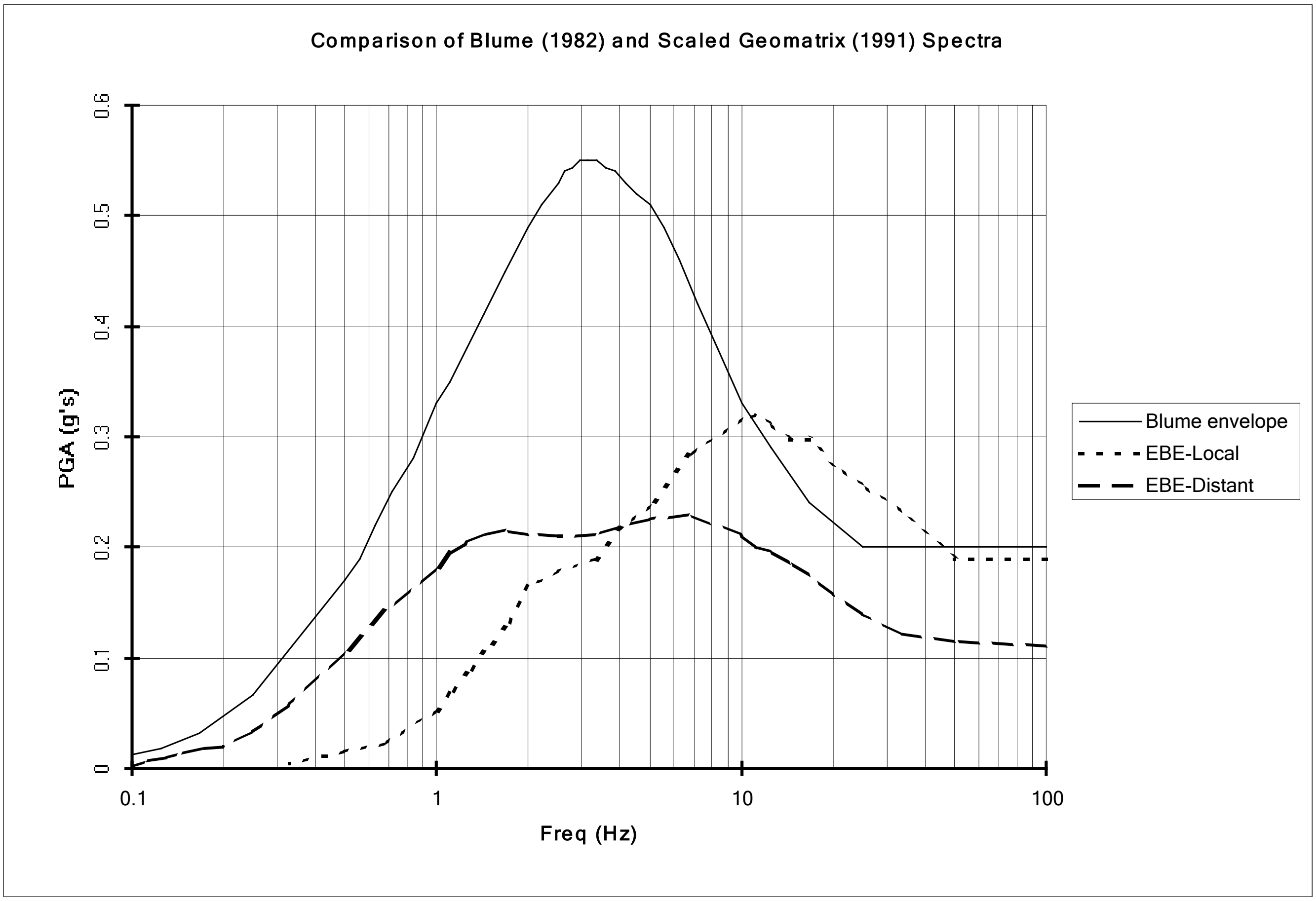
Figure 1.3.6-11. Response Spectrum Envelope Developed by URS/Blume (1982)



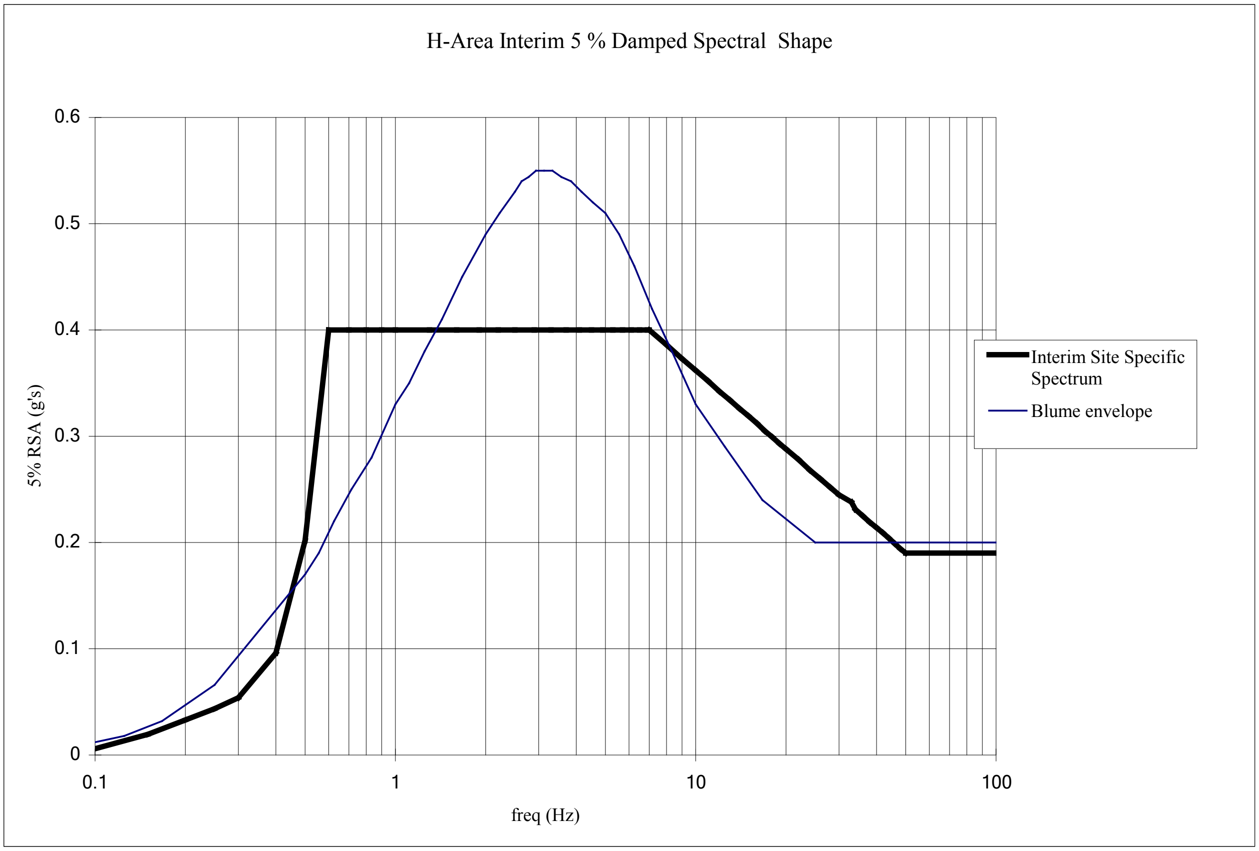
Notes:

5% damping

EBE = Evaluation Basis Earthquake, as described in Section 1.3.6.3.3.

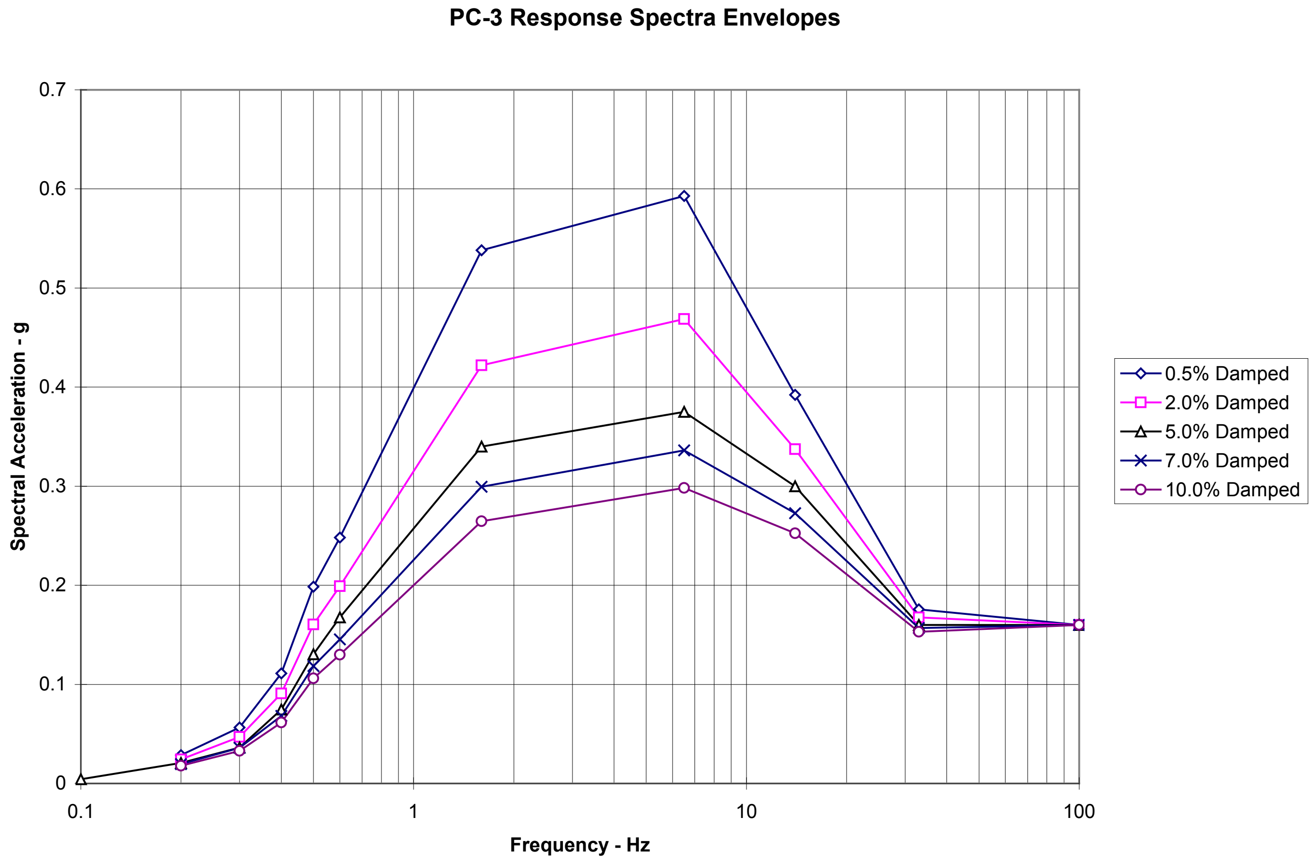
Data from WSRC 2000b.

Figure 1.3.6-12. Interim Site Spectrum Versus Blume Envelope



Note:
5% damping
Data from WSRC 2000b.

Figure 1.3.6-13. PC-3 Response Spectra Envelopes

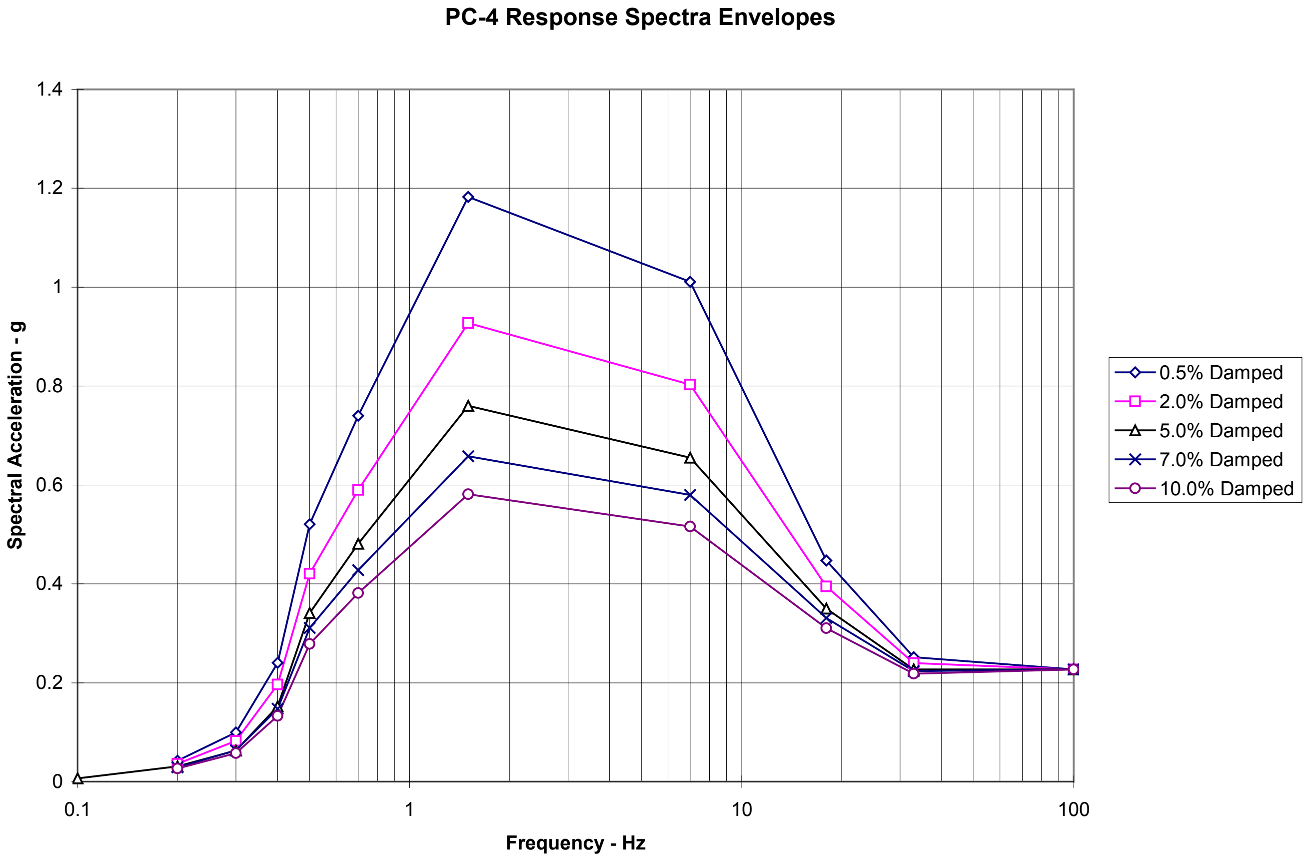


Note:

PC-3 design response spectrum revised in 1999, as shown in Figure 1.3.6-20.

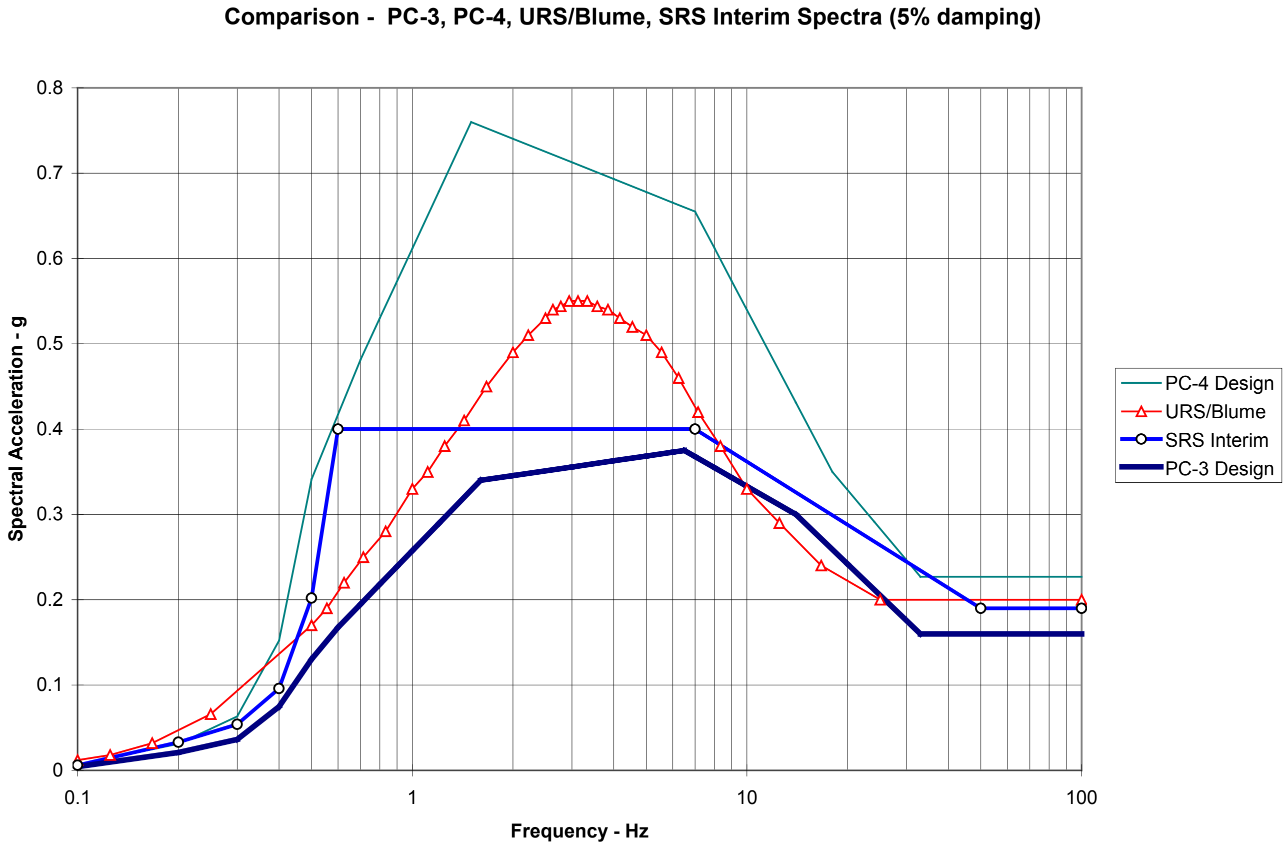
Data from WSRC 2000b.

Figure 1.3.6-14. PC-4 Response Spectra Envelopes



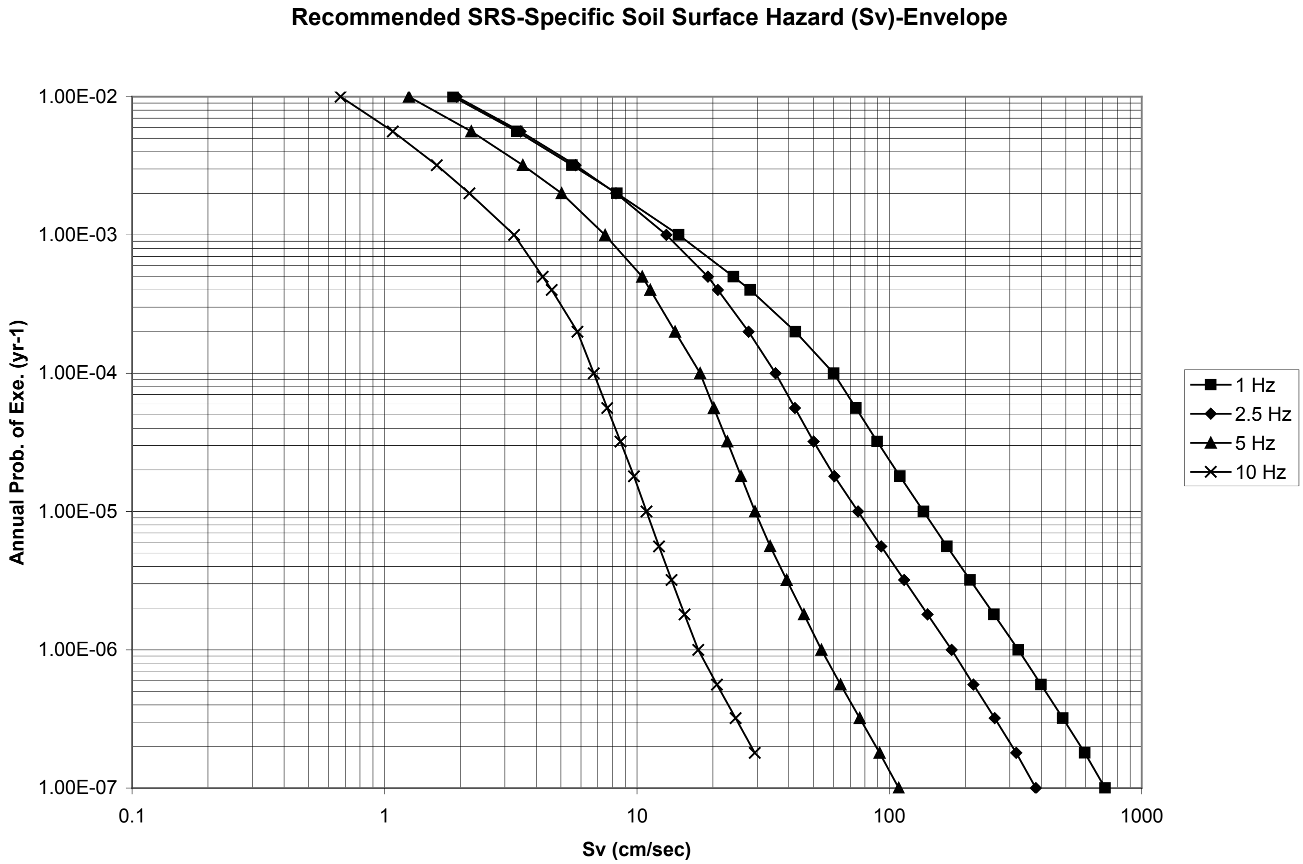
Data from WSRC 2000b.

Figure 1.3.6-15. Comparison – PC-3, PC-4, Blume, SRS Interim Spectra (5% Damping)



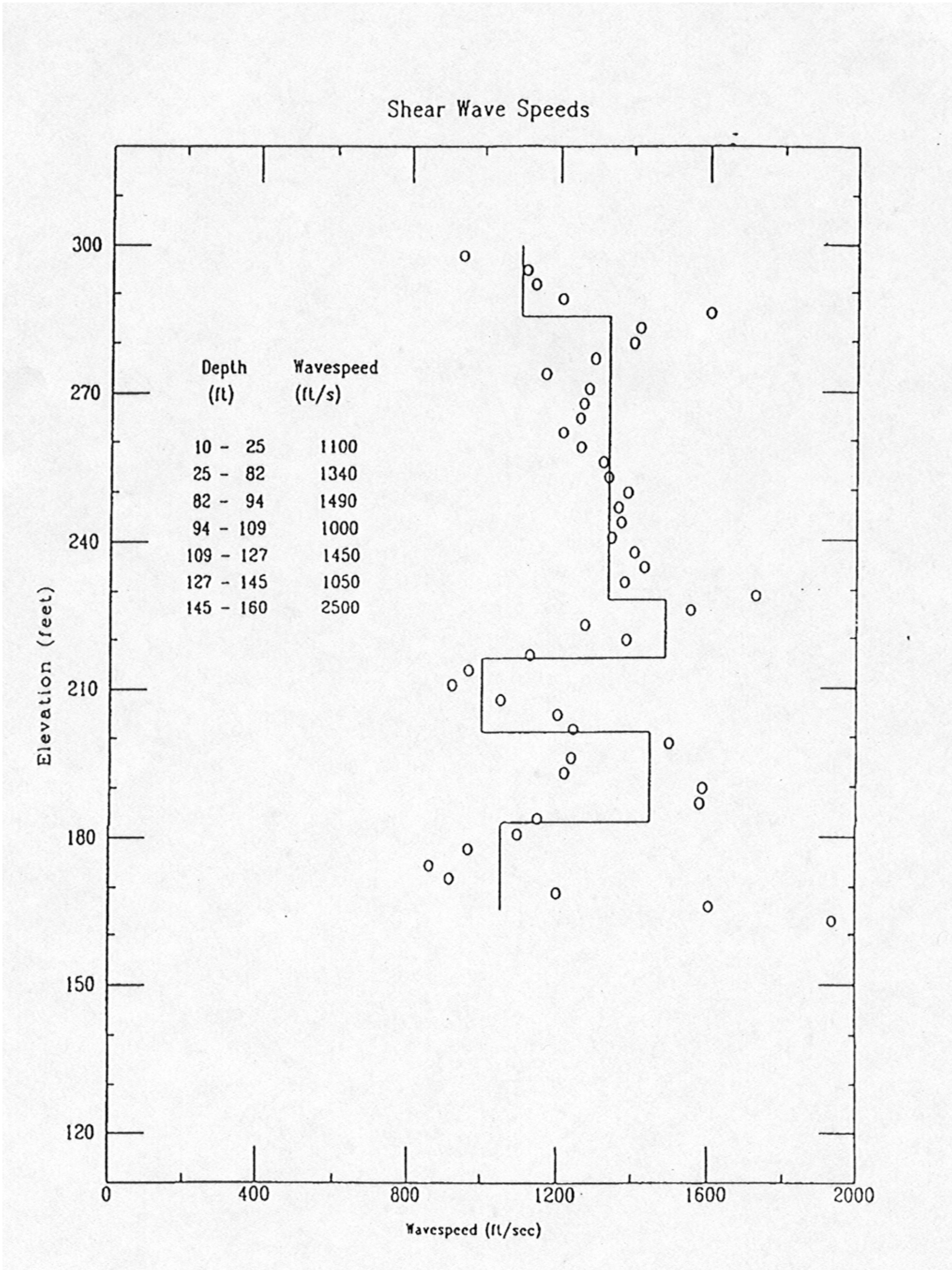
Note: PC-3 design response spectrum revised in 1999, as shown in Figure 1.3.6-20.
Data from WSRC 2000b.

Figure 1.3.6-16. Combined EPRI and LLNL Soil Surface Hazard Envelope (Probability of Exceedance vs 5% Damped Spectral Velocity) for Oscillator Frequencies of 1, 2.5, 5, and 10 Hz.



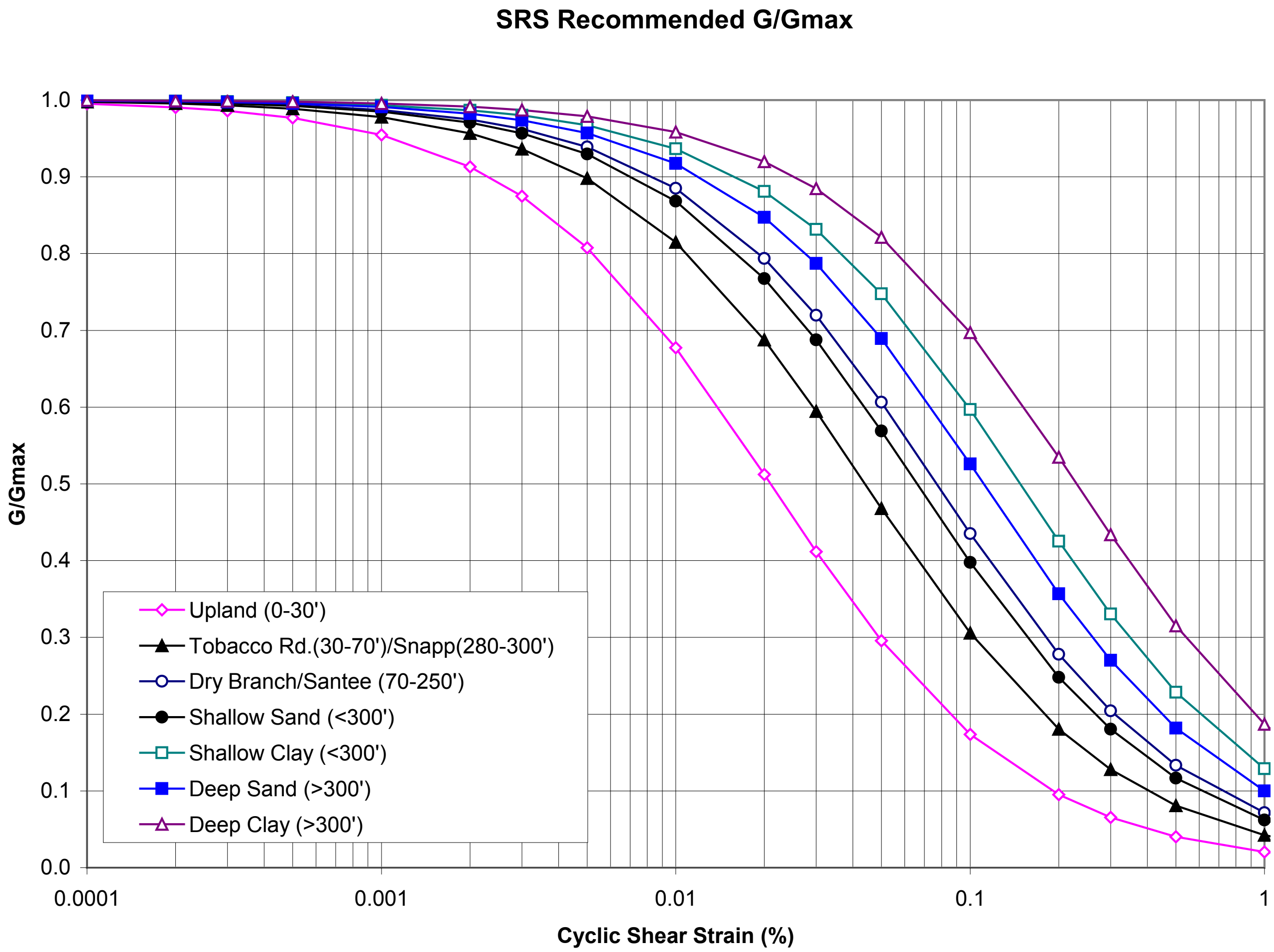
Data from WSRC 2000b.

Figure 1.3.6-17. Example Seismic Cone Penetrometer S-Wave Interpretation (Solid Lines) Measurement Taken in F Area



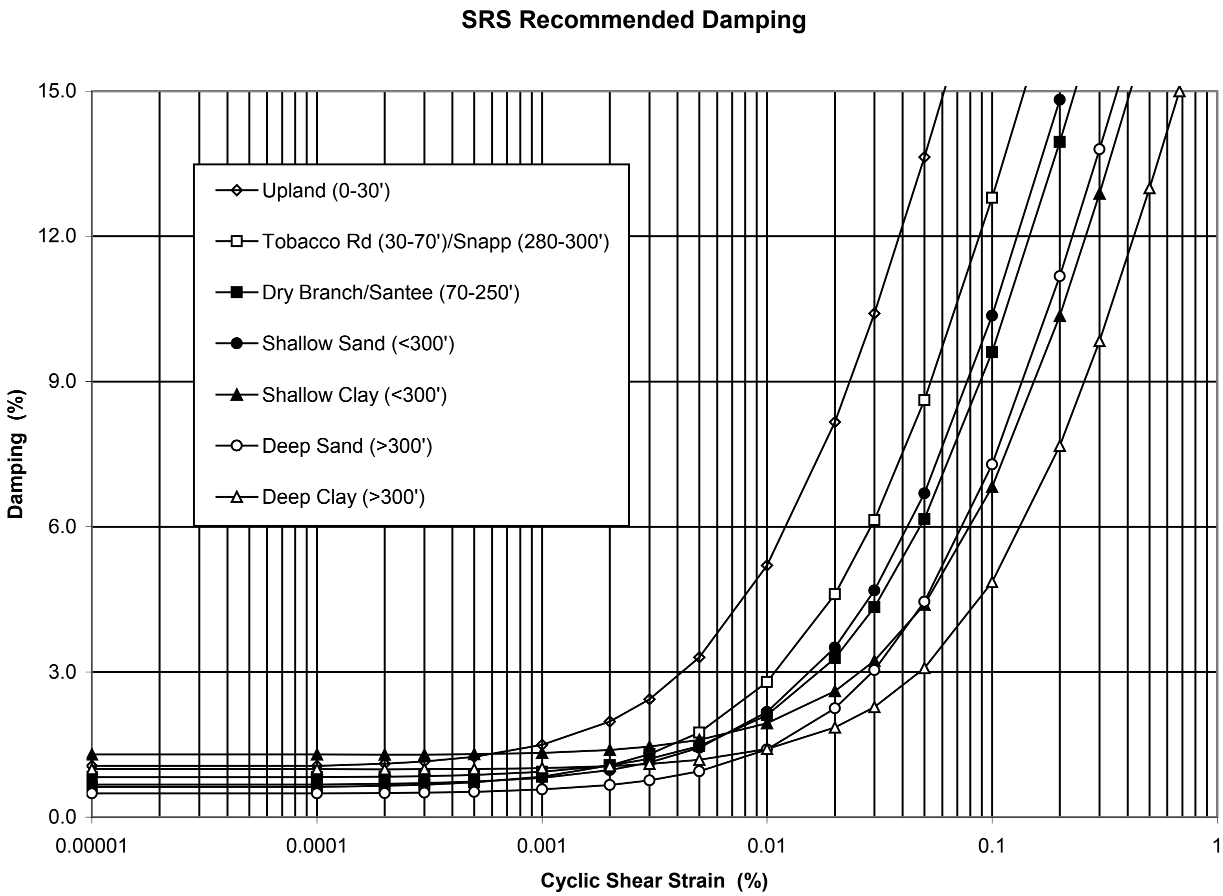
Data from WSRC 2000b.

Figure 1.3.6-18. SRS Recommended G/Gmax



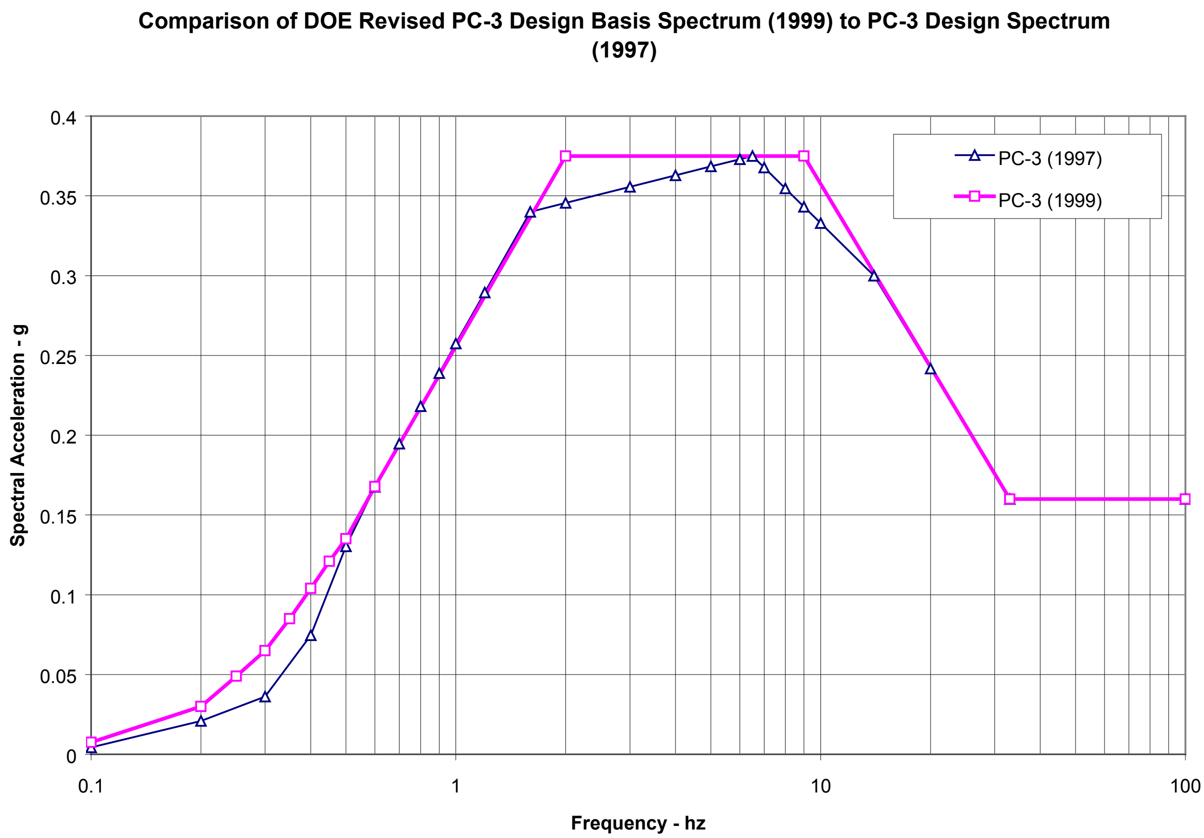
Data from WSRC 2000b.

Figure 1.3.6-19. SRS Recommended Damping



Data from WSRC 2000b.

Figure 1.3.6-20. Revised SRS PC-3 5% Damped Design Response Spectrum



Data from WSRC 2000b.

Figure 1.3.6-21. Comparison of 0.2g RG 1.60 Spectrum to PC-3 and PC-4

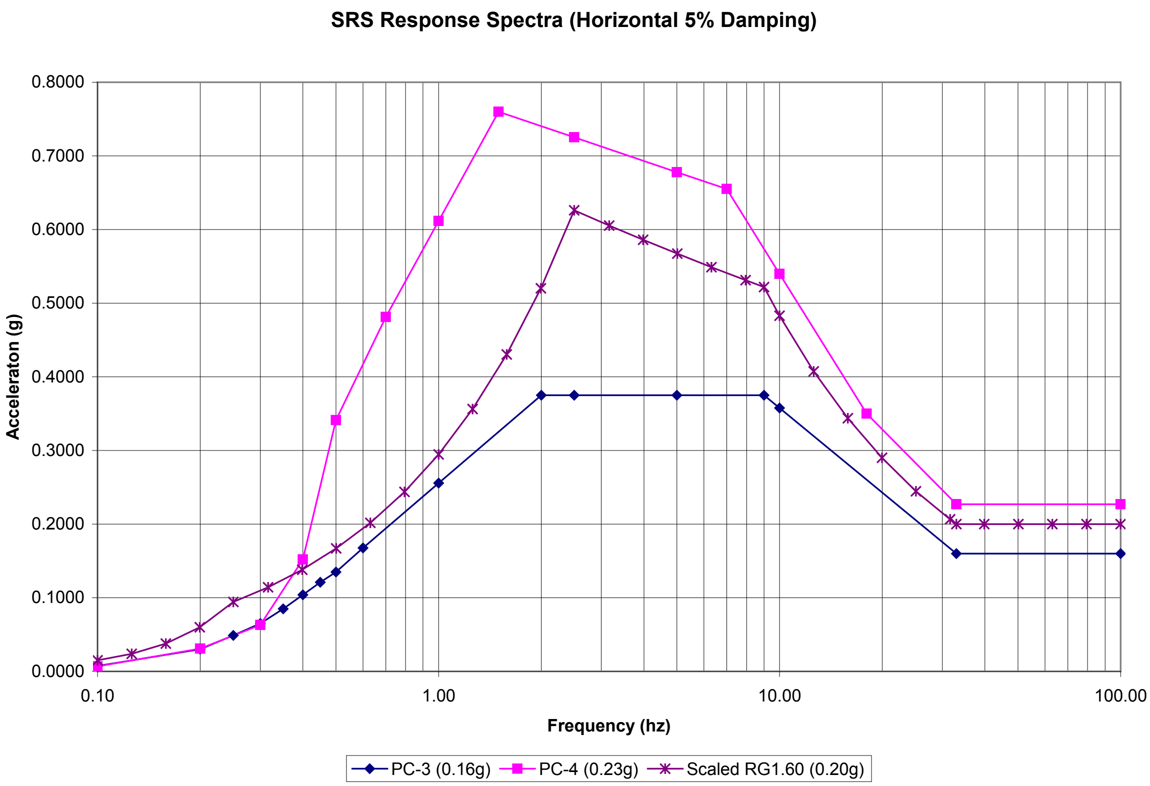


Figure 1.3.6-22. Design Earthquake for MFFF Systems, Structures, and Equipment

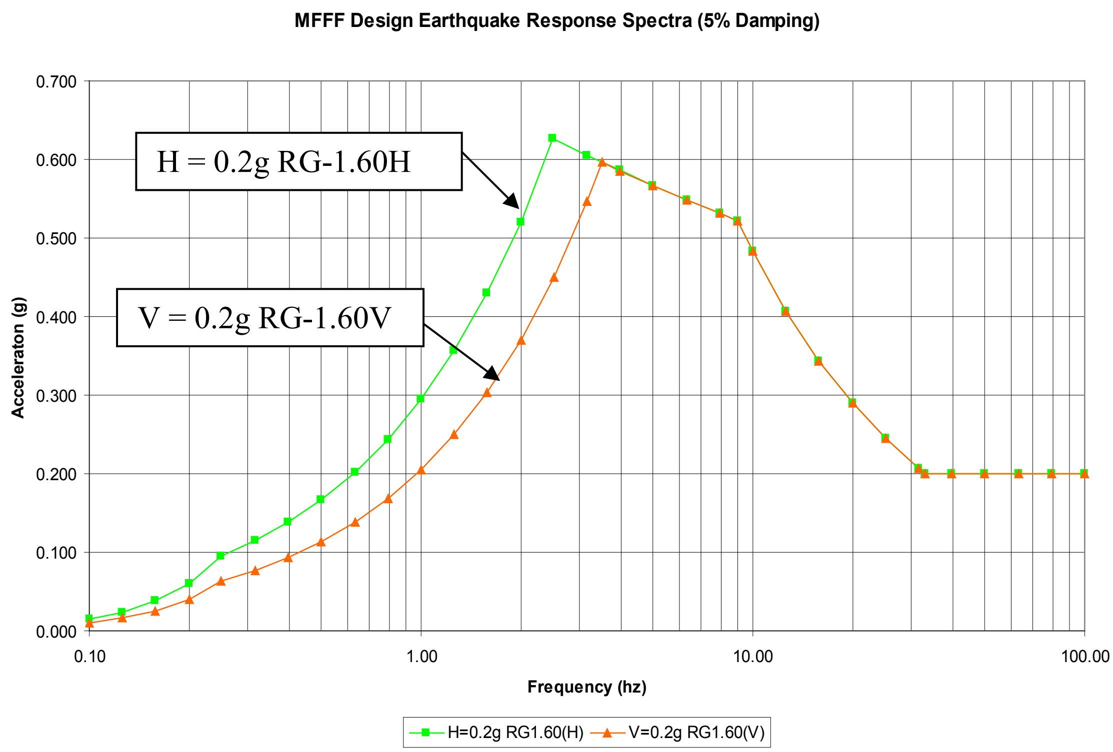
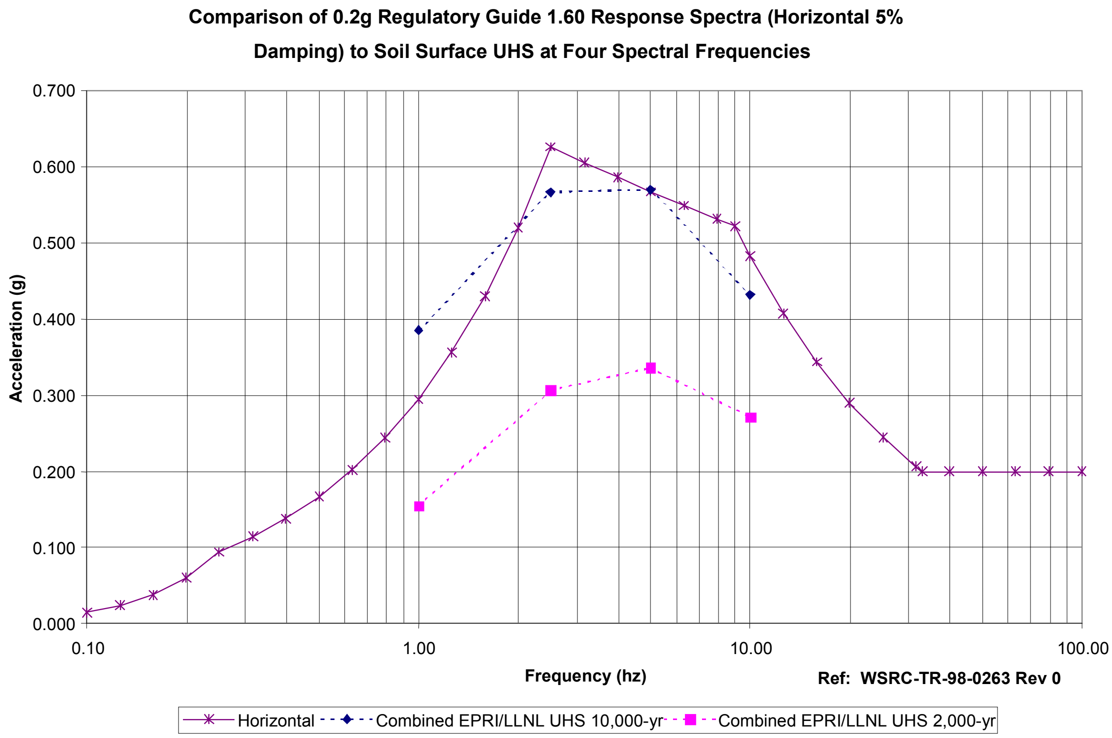
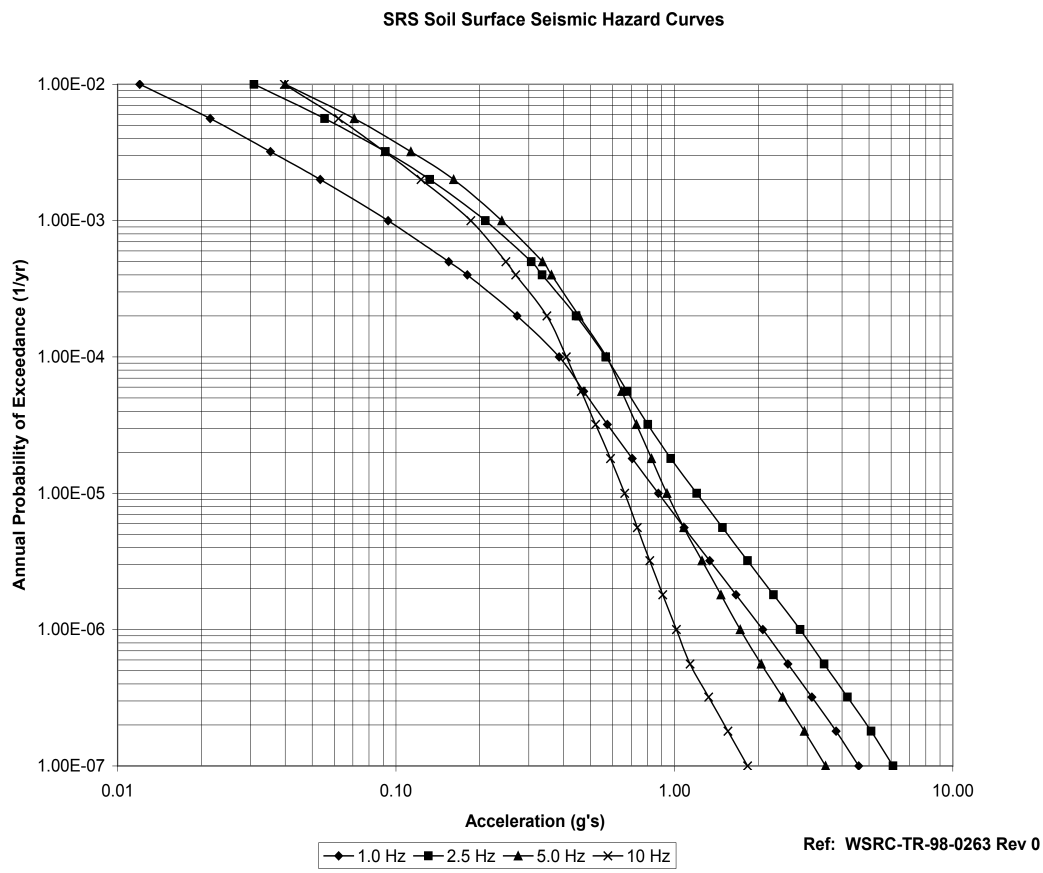


Figure 1.3.6-23. Comparison of 0.2g Regulatory Guide 1.60 Response Spectra (Horizontal 5% Damping) to Soil Surface UHS at Four Spectral Frequencies



Data from WSRC 2000b.

Figure 1.3.6-24. SRS Soil Surface Seismic Hazard Curves



Note: 5% Damping

Data from WSRC 1998.

1.3.7 Stability of Subsurface Materials

Subsurface geologic and soil conditions at the MFFF site are detailed in Section 1.3.5, and the engineering evaluation of foundations is presented in detail in Section 7 of the *MOX Fuel Fabrication Facility Site Geotechnical Report* (DCS 2005) and summarized in Section 1.1.2.1. As discussed in Section 1.3.5, the initial spacing of CPT exploration holes for the MFFF site was based on the approximate spacing patterns that were found to be successful in identifying potential soft zones at the nearby Actinide Packaging and Storage Facility (APSF) project site. The same process that was used successfully at the end of the APSF exploration program was used as the initial exploration hole spacing to identify potential soft zones on the MFFF site. When soft zones were encountered, additional CPT and exploration holes were added to delineate the extent of soft zones found. The resulting close spacing of exploration holes is also sufficient to identify loose soil zones that may be present at locations of MFFF IROFS structures, such as the MOX Fuel Fabrication Building (BMF) and Emergency Generator Building (BEG).

The BMF and BEG buildings were relocated to the western portion of the MFFF site when significant soft zones were encountered at the initial building locations. The same process used for initial layout of the CPT and exploration hole spacing was used for additional exploration in the western area of the MFFF site. In addition to the CPT holes and exploration borings made for the MFFF site, several exploration borings from previous exploration programs in the area were used for evaluation of subsurface conditions at the new building locations. The new CPT and exploration boring spacing, in conjunction with previous exploration holes, provided an exploration hole spacing closer than was deemed necessary for the initial facility layout.

The approach for the layout of CPTs and exploration borings at the MFFF site provides confidence that soft and loose soil zones have been effectively identified in the vicinity of the MFFF IROFS structures. Conservative assessments were used to define identified soft zones and loose soil zones in the vicinity of the MFFF IROFS structures. Exploration spacing in the original geotechnical investigation was greater than desired because the drilling and CPT rigs could not access locations on the existing APSF spoils pile berm slopes. After grading of the slopes facilitated rig access for additional exploration hole locations, MOX Services conducted a supplemental geotechnical investigation to acquire additional subsurface information to provide increased confidence that the size and extent of soft zones beneath the MFFF IROFS structures are adequately characterized. The results of these supplemental investigations are consistent with the results obtained during the initial site investigations, and they are described in detail in *MOX Fuel Fabrication Facility Site Geotechnical Report* (DCS 2005).

General geotechnical stability considerations are categorized and listed below with the intent of defining the approaches and methods used to address stability of subsurface materials. Geotechnical stability considerations at the MFFF site fall into the following two categories:

- Liquefaction (Section 1.3.7.1)
- Soft zones (Section 1.3.7.2).

1.3.7.1 Liquefaction Susceptibility

The liquefaction susceptibility of loose subsurface materials at the MFFF site was evaluated using qualitative and quantitative approaches. Site-specific investigations were conducted for the MFFF site, as discussed in Section 1.3.5. Approaches implemented included criteria for clayey soils, shear wave velocity evaluation, the stress method, and the strain method. Field programs were conducted and laboratory testing was performed to characterize site conditions and to define behavior characteristics of the native MFFF site soils. Section 1.3.5 addresses the MOX site soil profile and characteristics. The groundwater level at the time of the exploration programs was measured at an elevation of approximately 205 feet (62.5 m) above msl, which is more than 60 feet (18.3 m) below the planned site grade. The MFFF site exploration programs have identified only a few isolated pockets of loose soils at depth, below the groundwater table.

The *MOX Fuel Fabrication Facility Site Geotechnical Report* (DCS 2005) contains a detailed presentation of the site investigations that have been conducted and site-specific analyses, including liquefaction analyses. Section 8 (Stability of Subsurface Materials) of the report demonstrates the acceptability of the MFFF site with respect to liquefaction and post-earthquake dynamic settlement. This section of the report also demonstrates that the 1886 Charleston earthquake control motion is the controlling earthquake motion for liquefaction and post-earthquake dynamic settlement for the MFFF site. The results of the liquefaction analyses indicate that the soils within the MFFF Structure Vicinity will experience no liquefaction as a result of the design earthquake. The analyses also indicate that post-earthquake dynamic settlements are not excessive and considered acceptable for the design of principal SSCs.

1.3.7.2 Evaluation of Soft Zones

Across Savannah River Site (SRS), the soil zone between approximately 100 and 250 feet (30.5 and 76.2 m) below ground surface is a marine deposit labeled the Santee Formation. Within this interval are areas with locally high concentrations of calcium carbonate. Often found within these sediments, particularly in the upper third of this section, are weak zones interspersed in stronger matrix materials. These weak zones, which vary in thickness and lateral extent, are termed “soft zones.” The existence of soft zones and the potential for settlement are site-specific characteristics and require subsurface characterization and engineering evaluation on a site-specific basis.

At the MFFF site, the Santee Formation is generally found below an elevation of 180 feet (54.9 m) above msl, which is more than 90 feet (27.4 m) below the planned site grade. The soft zones found at the MFFF site are consistent with soft zones identified in the adjacent APSF and F Area geotechnical programs. The exploration programs indicated that soft zones at the MFFF site are isolated and found as soft soil pockets at depth. The field exploration program used close exploration hole spacing to identify and locate soft zones and to delineate their approximate boundaries, when encountered near planned structure locations. MFFF IROFS structures, such as the BMF and BEG buildings, were located on the MFFF site to avoid placement directly over significant soft zones identified during site explorations. The locations of IROFS structures were determined in accordance with detailed site-specific geotechnical analysis (DCS 2005), which demonstrates the acceptability of the soft zones identified at the MFFF site.

The soft zones at SRS and at the MFFF site are stable under static conditions. The Santee Formation, in which the carbonate and soft zones are found, is generally in the saturated zone well below the water table. Here the sediments are in a stable chemical environment, and carbonate dissolution is minimal. Further dissolution and removal of the Santee carbonate is not a concern in the time frame of interest for the MFFF (i.e., 50-100 years). The geologic record at SRS shows that soft zones encountered today have withstood the earthquakes that have occurred since their formation. No subsidence under static or seismic conditions is expected due to the presence of the soft zones identified at the MFFF site.

For the types of facilities to be constructed at the MFFF site, the increase in load on the soft zone soils is considered to be moderate to negligible. Potential load increases due to static and seismic design loads and deformations that may result in the soft zones from static or dynamic foundation loading were evaluated using appropriate geotechnical methods. Structure settlement that may result from deformation of any soft zones beneath or adjacent to critical structures has been defined (DCS 2005). MFFF IROFS structures are designed to accommodate anticipated settlements See Section 1.1.2.1 for more a detailed discussion of the settlement analyses that were performed of the BMF and BEG buildings.

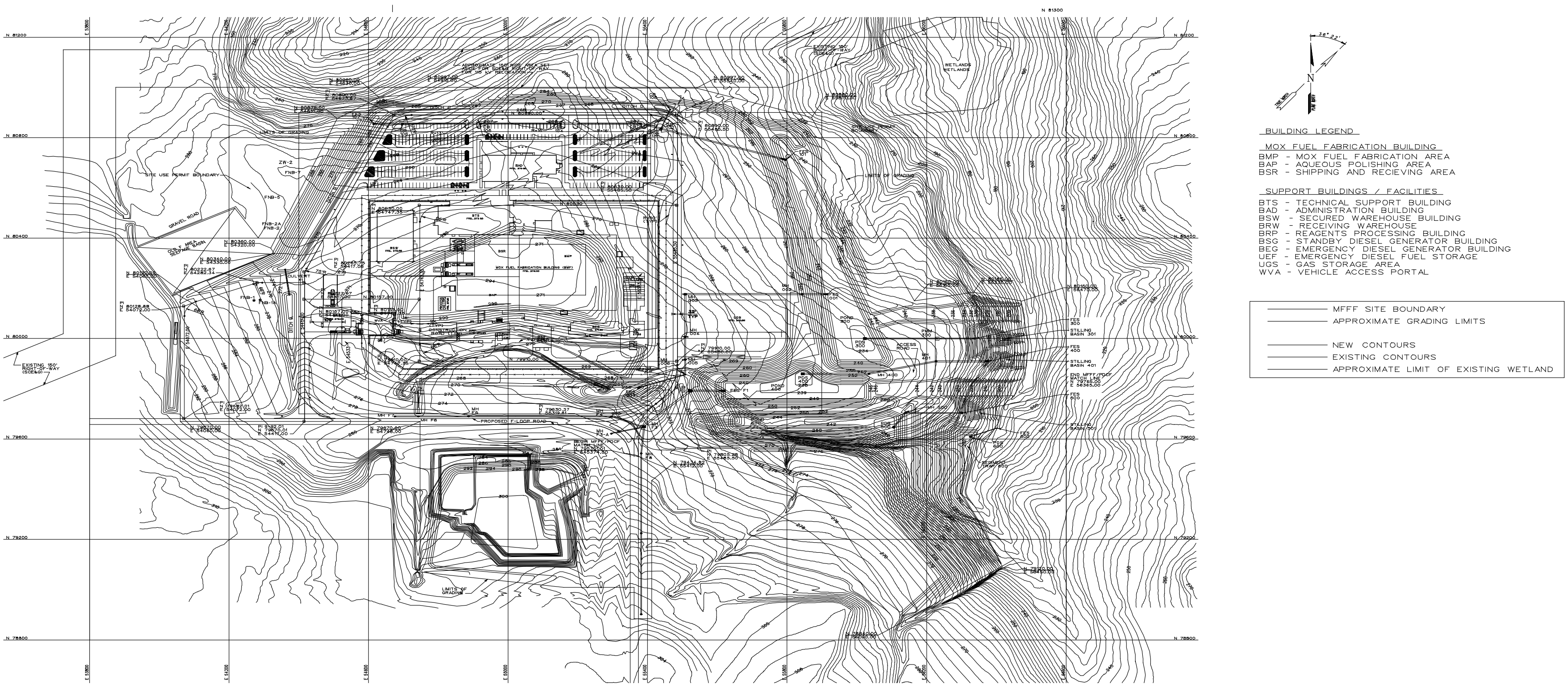
1.3.7.3 Slope Instability Hazard Evaluation

The preliminary site contour map defines the original topography, proposed finish grades, location of major cut and fill slopes, and location of the IROFS structures. See Figure 1.3.7-1 for the preliminary site contour map. The nearest cut slopes are over 400 feet (121.9 m) both north and west from the BMF and are only approximately 15 feet (4.6 m) high. The BMF and the BEG are located with finished floor elevations below the existing ground elevation, and are both over 400 feet (121.9 m) from the top of the nearest fill slope or steeper topographic slope. Figure 1.3.7-1 shows the fill and steep slopes to the northwest, northeast, and southeast of the BMF and BEG. Therefore, slope stability from existing topography and planned fills and cuts at the MFFF site do not have any adverse impact to IROFS structures.

1.3.7.4 Actinide Packaging and Storage Facility Spoil Evaluation

The preliminary site contour map shown in Figure 1.3.7-1 shows the location of the spoil pile created from the excavated materials removed from the APSF. During MFFF site grading, this APSF spoil pile will be removed and will not be used in connection with foundations for the IROFS structures. Therefore, the pile cannot adversely impact IROFS structures.

Figure 1.3.7-1. Preliminary Site Contour Map



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2.0 FINANCIAL QUALIFICATIONS

The purpose of financial qualifications information is to enable the U.S. Nuclear Regulatory Commission (NRC) to determine if the applicant appears to be financially qualified to engage in the proposed activities in accordance with the applicable NRC requirements. The information provided below demonstrates that CB&I AREVA MOX Services, LLC (MOX Services) is financially qualified to safely operate the Mixed Oxide (MOX) Fuel Fabrication Facility (MFFF).

2.1 PROJECT COSTS AND SOURCES OF FUNDS

The United States and the Russian Federation have concluded a bilateral agreement on plutonium disposition, "Agreement Between the Government of the United States of America and the Government of the Russian Federation Concerning the Management and Disposition of Plutonium Designated As No Longer Required for Defense Purposes and Related Cooperation" (September 2000). Under the agreement, the United States will dispose of surplus weapons-grade plutonium. The MFFF is intended to fulfill the United States' obligation for disposition of that plutonium. In light of the MFFF's importance to the United States' obligation and Congressional support for this program, there is significant continuing federal Government incentive to adequately fund the MFFF and to continue providing the necessary annual appropriations to support operation of the MFFF.

MOX Services operates the MFFF under a contract with the U.S. Department of Energy (DOE). During operations, DOE reimburses MOX Services for the full cost of operating the MFFF, minus fuel payments that MOX Services receives from the mission reactor utilities, plus a possible incentive fee. MOX Services does not intend to finance or rely on the proceeds from debt or equity securities, or any other source of external financing other than DOE funding, nor does it intend to rely on any revenue stream to cover such costs (with the exception of the revenue stream from the mission reactor utilities as described above).

2.2 CONTINGENCY FUNDS

In light of the structure of funding for operations, no contingency funds are necessary. In the unlikely event of a DOE funding shortfall, licensed materials would be placed in a safe condition.

2.3 FINANCIAL QUALIFICATIONS

Because the MFFF is a U.S. Government funded project, the specific financial resources and capabilities of MOX Services and its equity owners are not relevant to the determination of adequate financial resources to operate the facility. MOX Services does not intend to rely on its financial resources, or those of an equity partner or parent company, to provide financing.

MOX Services is not a publicly held entity, and as such, its financial statements are not publicly available. MOX Services previously submitted under separate cover proprietary financial statements providing information concerning MOX Services financial condition.

The structure of MOX Services reimbursement for MFFF operation is designed to support the MFFF project as a viable business enterprise. Thus, MOX Services is financially qualified to safely operate the MFFF, and that financial qualification is supported by the federal Government's obligation through the DOE – MOX Services contract for the MOX Project.

2.4 LIABILITY INSURANCE

MOX Services is a DOE contractor and is thus fully covered by DOE nuclear liability protection under the Price-Anderson Act, as amended. Section 170(d) of the Atomic Energy Act provides that the DOE Secretary shall enter into agreements of indemnification with certain persons "... who may conduct activities under a contract with the Department of Energy that involve the risk of public liability and that are not subject to financial protection requirements under subsection b. or agreements of indemnification under subsection c. or k." In accordance with this statutory authority, the contract between MOX Services and DOE contains the following "Nuclear Hazards Indemnity Agreement" excerpt from Department of Energy Acquisition Regulations (DEAR 952.250-70), which fully indemnifies MOX Services and its subcontractors up to the statutory limit of liability[1]:

"(d)(1) *Indemnification*. To the extent that the contractor and other persons indemnified are not compensated by any financial protection permitted or required by DOE, DOE will indemnify the contractor and other persons indemnified against (i) claims for public liability as described in subparagraph (d)(2) of this clause; and (ii) such legal costs of the contractor and other persons indemnified as are approved by DOE, provided that DOE's liability, including such legal costs, shall not exceed the amount set forth in section 170e.(1)(B) of the Act in the aggregate for each nuclear incident or precautionary evacuation occurring within the United States or $100 million in the aggregate for each nuclear incident occurring outside the United States, irrespective of the number of persons indemnified in connection with this contract.

"(2) The public liability referred to in subparagraph (d)(1) of this clause is public liability as defined in the Act which (i) arises out of or in connection with the activities under this contract, including transportation; and (ii) arises out of or results from a nuclear incident or precautionary evacuation, as those terms are defined in the Act."

The DOE indemnity agreement with MOX Services provides full protection and coverage for public liability arising from operation of the MFFF.

2.5 DESIGN BASIS

The DOE has agreed to indemnify MOX Services in accordance with the provision of the Price-Anderson Act set forth in Section 170(d) of the Atomic Energy Act of 1954, as amended. Based upon the DOE Indemnity Agreement, MOX Services requested and received an exemption from the NRC's requirements concerning agreements of indemnification and related financial protection requirements. The NRC granted this exemption in their "Final Safety Evaluation

[1] The Energy Policy Act of 2005 (Public Law 109-58; in particular Section 601, Price-Anderson Amendments Act of 2005) increases the limits in the DEAR.

Report for the License Application to Possess and Use Radioactive Material at the Mixed Oxide Fuel Fabrication Facility in Aiken, SC" (December 2010). This exemption removed potential ambiguity as to whether indemnification resided with DOE or NRC. The DOE indemnity agreement with MOX Services provides full protection and coverage for public liability arising from operation of the MFFF.

3.0 PROTECTION OF CLASSIFIED MATTER

Prior to issuance of the Part 70 license to possess and use by-product material, source material, and special nuclear material, CB&I AREVA MOX Services, LLC (MOX Services) will control classified matter in accordance with applicable DOE requirements. Upon receipt of the license, MOX Services will control classified matter in accordance with the Classified Matter Protection Plan for the Mixed Oxide (MOX) Fuel Fabrication Facility, which was submitted under separate cover.

The U.S. Department of Energy (DOE) has rendered a favorable foreign ownership, control, or influence (FOCI) determination of MOX Services, as discussed in Section 1.2.

4.0 ORGANIZATION AND ADMINISTRATION

The CB&I AREVA MOX Services, LLC (MOX Services) functional organizational structure for the operational phase of the Mixed Oxide Fuel Fabrication Facility (MFFF) is shown in Figure 4-1.

4.1 FACILITY ORGANIZATIONAL STRUCTURE

Functional responsibilities and authority are described below for key management functions. The authority to make commitments to the U.S. Nuclear Regulatory Commission (NRC) is only held by the explicitly stated positions. The key management functions are responsible for items relied on for safety (IROFS) and related activities.

The key MOX Services management functions with health, safety and environmental (HS&E) responsibilities are the MOX Services President, Plant Manager, Operations Manager, Engineering Manager, and Environmental Safety & Health (ES&H) Licensing Manager. Operations, Engineering and ES&H Licensing are independent functions allowing each organization to provide objective audits, assessments, and reviews. Independence means that none of these organizations report administratively to the other.

Qualification requirements for key management positions are provided below. Relevant work experience of at least five years, in addition to the minimum experience requirements specified below, may be substituted for the Bachelor's degree requirements. Where work experience in more than one field is required for a given position (e.g., four years of engineering experience and two years of management experience), the experience may be concurrent unless otherwise indicated. The MOX Services President may approve exceptions to the qualification requirements for the positions described in this chapter.

Stop work authority is vested in each MOX Services employee. Any employee may stop work when the continuation of such work could jeopardize the health and safety of workers or the public, result in adverse consequences to the environment, or produce results that do not comply with the MOX Services Quality Assurance (QA) program. Following a stop-work, activities related to safety are stopped until the deficiency or unsatisfactory condition has been resolved in accordance with MOX Services procedures.

4.2 KEY MANAGEMENT FUNCTIONS

4.2.1 Facility Management Function

The MOX Services President manages all aspects of the MFFF, including safety and nuclear fuel manufacturing activities at the facility. This individual directs licensed activities and staff functions through designated operations, engineering, safety, and business management personnel. The President provides for the health and safety of the public and workers and protection of the environment by delegating and assigning responsibility to qualified managers and personnel. The President's direct reports are shown on Figure 4-1. The President reports to the MOX Services Board of Governors (not shown on Figure 4-1).

The corporate officer that has the overall responsibility for health, safety, and environmental (HS&E) matters for MFFF is the President of MOX Services.

The minimum qualifications for the MOX Services President are a Bachelor's degree (or equivalent) in engineering or science, five years of experience in operations, and/or engineering of nuclear facilities, and five years of experience in management.

4.2.2 Quality Assurance Function

The manager of the quality assurance QA function is responsible for maintaining the MOX Services Project Quality Assurance Plan (MPQAP) and reports directly to the MOX Services President. This function is independent of the organizations responsible for performing quality-affecting work and is independent of cost and schedule considerations. This function may be assigned other duties; however, these duties are not allowed to compromise the independence of this function or to prevent attention to quality assurance matters. The manager of the QA function has the same access to the MOX Services President as the line managers of other functional areas of the MFFF.

The manager of the QA function is responsible for identifying quality problems, recommending and verifying implementation of solutions, and ensuring further work is controlled until the unsatisfactory conditions has been corrected. The manager of the QA function is responsible for approval of the subcontractor quality assurance programs, oversight, and audit functions. The manager of the QA function also interfaces with NRC, stakeholders and other governmental agencies regarding the QA requirements, compliance with QA requirements, and resolution of QA concerns. These functions are accomplished by delegating and assigning responsibility to qualified personnel.

The QA manager is the only key management position in the QA organization. The minimum qualifications for the QA Manager position are a Bachelor's degree (or equivalent), four years of quality assurance-related experience, two years of nuclear industry experience, and one year of supervisory or management experience.

4.2.3 Production Function Including the Operations Function

The managers of the production function are responsible for the production, operation, technical support activities for the MFFF, including aqueous polishing (AP), fuel fabrication (MP), and maintenance. The production function includes the Plant Manager, Operations Manager, Operations Shift Managers, Maintenance Manager, and Technical Support Manager. This position also is directly responsible for maintenance, analytical laboratory, balance of plant systems, logistics, waste disposal and product quality control. Production functions are accomplished by delegating and assigning responsibility to qualified managers, supervisors and other personnel. The plant manager has the authority to make commitments to the NRC

The Plant Manager reports directly to the President. The Operations Manager reports to the Plant Manager, and the Operations Shift Managers reports to the Operations Manager. The Plant Manager, Operations Manager, and Operations Shift Managers are the only key management personnel in the production function.

The manager of the production functions are responsible for the safety and control of operations, knowledgeable of safety program concepts as they apply to the overall safety of the facility, and compliance with MFFF licensing requirements. These managers are also responsible for ensuring the overall implementation of the configuration management program. The Operations Manager and the Operations Shift Managers are responsible for the day-to-day processing, handling, and storing of licensed materials. These managers ensure configuration control for the integrated safety of facility processes while meeting production objectives. Operations Managers and Shift Operations Managers accomplish these functions by ensuring that operations personnel are adequately trained and that approved written procedures are available and adhered to. They are knowledgeable of, and responsible for, the control of IROFS within their area of supervision.

The minimum qualifications for the Plant Manager are a Bachelor's degree, (or equivalent) in engineering or science, four years of operational or manufacturing production experience in a nuclear facility, and one year of supervisory or management experience.

The minimum qualifications for the Operations Manager are a Bachelor's degree (or equivalent) in engineering or science, four years of operational or manufacturing production experience in a nuclear facility, and one year of supervisory or management experience.

The minimum qualifications for Operations Shift Managers are a Bachelor's degree, (or equivalent) in engineering or science, one year of operations or manufacturing production experience in a nuclear facility, and one year of supervisory or management experience.

Supervisors shall have at least the qualifications required of personnel being supervised and either one additional year experience supervising the technical area at a similar facility or completion of supervisor training.

The minimum qualifications for Technical staff identified in the ISA Summary whose activities are relied on for safety to satisfy the performance requirements identified in 10 CFR Part 70, are a Bachelor's degree (or equivalent) in an appropriate technical field and experience and training appropriate for their activities, authority, and responsibilities.

Facility operators, technicians, maintenance personnel, and other staff whose actions are required to comply with NRC regulations shall have completed the training process or have equivalent experience or training.

The minimum qualifications for candidates for process operator positions are a high school education (or equivalent).

4.2.4 Engineering Functions

The manager of the engineering function is the MFFF design authority and is directly responsible for system engineering and facility upgrade engineering. The engineering function is independent of other MFFF functions. The engineering function is accomplished by delegating and assigning responsibilities to qualified managers, engineers, and designers.

The Engineering Manager reports directly to the President. Only the Engineering Manager is a key management position in the engineering function.

The minimum qualifications for the Engineering Manager are a Bachelor's degree (or equivalent) in engineering, four years of experience in engineering or operations of nuclear facilities, and one year of supervisory or management experience.

4.2.5 Environmental, Safety & Health Licensing Functions

The manager of the Environmental, Safety & Health (ES&H) Licensing function is independent of the production function and is directly responsible for the health, safety and environmental (HS&E) functions including fire safety, radiation protection, chemical safety, criticality safety, nuclear safety analysis, and environmental protection. The ES&H Licensing Manager is responsible for maintaining the MOX Services special nuclear material possession and use license, planning and executing licensing and regulatory compliance activities, maintaining licensing-related documents, and interfacing with the NRC and other regulatory agencies regarding licensing matters. The manager of ES&H Licensing function has the authority to make commitments to the NRC. These functions are accomplished by delegating and assigning responsibility to qualified personnel. The ES&H Licensing Manager reports directly to the President.

The ES&H Licensing Manager is the only key management position within ES&H Licensing.

The minimum qualifications for ES&H Licensing Manager are a Bachelor's degree (or equivalent), four years of experience in engineering, licensing, safety or operations of nuclear facilities, and one year of supervisory or management experience.

The fire protection function has implementation responsibility for the overall fire protection program and has input to organizations involved in fire protection activities. The individual responsible for the fire protection function has at least five years of experience as a fire protection engineer.

The manager of the radiological protection function (RPM) is responsible for setting radiological protection policy and for implementation of this policy. The RPM has a minimum of a Bachelor's degree, or equivalent, in science, health physics, or engineering, and has at least four years of experience in radiological protection. Certification by the American Board of Health Physics or an additional four years of relevant experience provides equivalency to the degree requirements. Experience should include supervision or management of operational radiological control programs. Management may waive specific qualifications for the RPM on a case by case basis when education, experience, certifications, and overall qualification of the supporting staff meet the above requirements.

The senior staff of the radiological protection function includes health physicists and other professionals with four-year degrees in science, engineering, or equivalent (as defined above for the RPM) and at least one year of experience in applied radiological controls at an operating nuclear facility.

Radiological support personnel provide radiological protection and radiological engineering, dosimetry, bioassay, independent oversight, instrumentation and calibration functions. These personnel have a high school diploma or equivalent, and technical qualifications pertinent to their assigned duties.

The manager of the nuclear criticality safety (NCS) function has the authority and responsibility to assign and direct activities for the NCS function. The minimum qualifications for the manager of the NCS function are a Bachelor's degree in science or engineering, or equivalent, with at least three years of nuclear industry experience in criticality safety. The manager of the NCS function has management or technical experience in the application and/or direction of criticality safety programs for nuclear facilities involving SNM.

A senior NCS engineer has the authority and responsibility to conduct activities assigned to the criticality safety function, as directed by the manager of the NCS function. The minimum qualifications for a senior NCS engineer are a Bachelor's degree in science or engineering, or equivalent, with at least three years of nuclear industry experience in criticality safety.

An NCS engineer has the authority and responsibility to conduct activities assigned to the criticality safety function, with the exception of independent verification of NCSEs. The minimum qualifications for an NCS engineer are a Bachelor's degree in science or engineering, or equivalent, with at least one year of nuclear industry experience in criticality safety.

4.2.6 Support Services Functions

The support services function includes business-related functions that are necessary to support the MFFF mission. The support services functions include training for employees, plant engineering, contracts, legal, finance and accounting, human resources, and procurement. The support services function manager reports to the President. The managers of this function are not responsible for the HS&E functions for the facility and are not key management personnel. Therefore, the minimum qualifications for support services managers are not included in the License Application.

4.3 ADMINISTRATION

The managers responsible for the above functions are appropriately available to perform their duties. In times of absence, their duties may be delegated to other qualified personnel, as determined by the responsible manager. While these managers have the authority to delegate tasks to other individuals, the responsible manager retains the ultimate responsibility and accountability for compliance with applicable requirements.

MOX Services procedures are used to implement HS&E functions associated with the MFFF and management measures that supplement IROFS. See Chapter 15 for a discussion of management measures, which include quality assurance, configuration management, maintenance, training and qualifications, plant procedures, audits and assessments, incident investigations, and records management. Plant procedures are formally approved and controlled. If a procedure cannot be adhered to, work is stopped and not resumed until the procedure has been corrected or changed.

4.4 DESIGN BASIS

The following information represents the design basis attributes for the functional organizational structure for the MFFF.

Minimum Qualifications:

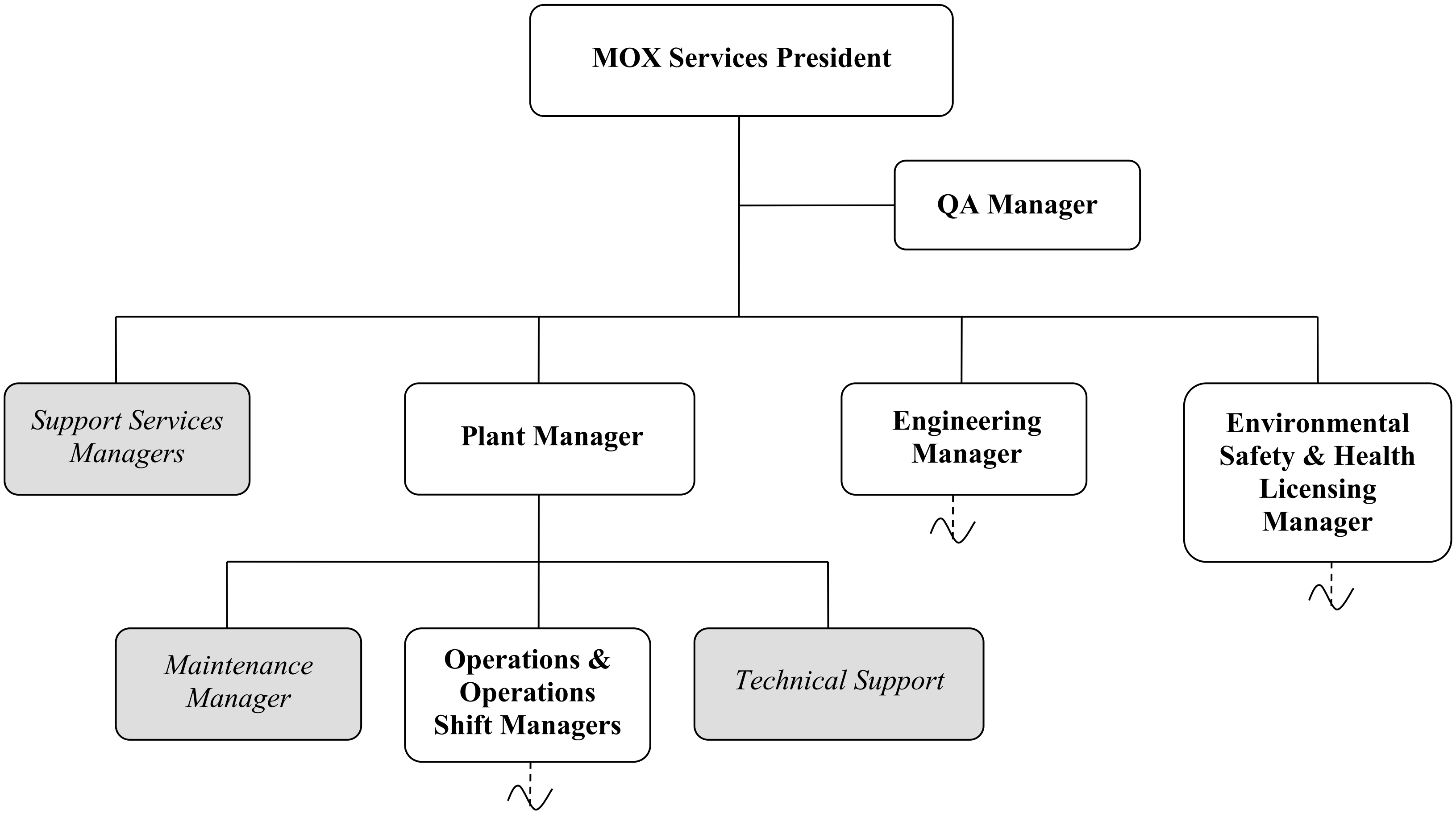
- Relevant work experience of at least five years, in addition to the minimum experience requirements specified below, may be substituted for the Bachelor's degree requirements.
- The minimum qualifications for the MOX Services President are a Bachelor's degree (or equivalent) in engineering or science, five years of experience in operations, and/or engineering of nuclear facilities, and five years of experience in management.
- The minimum qualifications for the QA Manager position are a Bachelor's degree (or equivalent), four years of quality assurance-related experience, two years of nuclear industry experience, and one year of supervisory or management experience.
- The minimum qualifications for the Plant Manager and the Operations Manager are a Bachelor's degree, (or equivalent) in engineering or science, four years of operational or manufacturing production experience in a nuclear facility, and one year of supervisory or management experience.
- The minimum qualifications for Operations Shift Managers are a Bachelor's degree, (or equivalent) in engineering or science, one year of operational or manufacturing production experience in a nuclear facility, and one year of supervisory or management experience.
- Supervisors shall have at least the qualifications required of personnel being supervised and either one additional year experience supervising the technical area at a similar facility or completion of supervisor training.
- The minimum qualifications for Technical staff identified in the ISA Summary whose activities are relied on for safety to satisfy the performance requirements identified in 10 CFR Part 70, are a Bachelor's degree (or equivalent) in an appropriate technical field and experience and training appropriate for their activities, authority, and responsibilities.
- Facility operators, technicians, maintenance personnel, and other staff whose actions are required to comply with NRC regulations shall have completed the training process or have equivalent experience or training.
- The minimum qualifications for candidates for process operator positions are a high school education (or equivalent).
- The minimum qualifications for the Engineering Manager are a Bachelor's degree (or equivalent) in engineering, experience in engineering or operations of nuclear facilities, and supervisory or management experience.
- The minimum qualifications for ES&H Licensing Manager are a Bachelor's degree (or equivalent), experience in engineering, licensing, safety or operations of nuclear facilities, and supervisory or management experience.

- The individual responsible for the fire protection function has at least five years of experience as a fire protection engineer.
- The RPM has a minimum of a Bachelor's degree, or equivalent, in science, health physics, or engineering, and has at least four years of experience in radiological protection. Certification by the American Board of Health Physics or an additional four years of relevant experience provides equivalency to the degree requirements. Experience should include supervision or management of operational radiological control programs. Management may waive specific qualifications for the RPM on a case by case basis when education, experience, certifications, and overall qualification of the supporting staff meet the above requirements.
- The senior staff of the radiological protection function includes health physicists and other professionals with four-year degrees in science, engineering, or equivalent (as defined above for the RPM) and at least one year of experience in applied radiological controls at an operating nuclear facility.
- The radiological support personnel have a high school diploma or equivalent, and technical qualifications pertinent to their assigned duties.
- The minimum qualifications for the manager of the NCS function are a Bachelor's degree in science or engineering, or equivalent, with at least three years of nuclear industry experience in criticality safety. The manager of the NCS function has management or technical experience in the application and/or direction of criticality safety programs for nuclear facilities involving SNM.
- The minimum qualifications for a senior NCS engineer are a Bachelor's degree in science or engineering, or equivalent, with at least three years of nuclear industry experience in criticality safety.
- The minimum qualifications for an NCS engineer are a Bachelor's degree in science or engineering, or equivalent, with at least one year of nuclear industry experience in criticality safety.

<u>Miscellaneous:</u>

- Operations, Engineering and ES&H Licensing are independent functions allowing each organization to provide objective audits, assessments, and reviews.
- The quality assurance function is independent of the organizations responsible for performing quality-affecting work and is independent of cost and schedule considerations.
- The manager of the engineering function is the MFFF design authority and is directly responsible for system engineering and facility upgrade engineering.

Figure **4-1**. MFFF Functional Organization



Non-key management functions are shaded.

Figure 4-1 MFFF Functional Organization (continued)

Engineering Manager

Functions & Responsibilities:

- Plant Engineering (Upgrade)
- MFFF Design Authority
- Mission Reactors
- System Engineering

Operations & Operations Shift Managers

Functions & Responsibilities:

- AP/MP
- Analytical Laboratory
- Balance of Plant
- Logistics and Waste
- Maintenance

ES&H Licensing Manager

Functions & Responsibilities:

- Criticality Safety
- Emergency Planning
- Environmental Compliance
- Fitness for Duty
- Industrial Safety
- Licensing

5.0 SAFETY PROGRAM AND INTEGRATED SAFETY ANALYSIS

CB&I AREVA MOX Services, LLC (MOX Services) has established and maintains a safety program, including an integrated safety analysis (ISA), that demonstrates compliance with the performance requirements of Title 10 of the Code of Federal Regulations (CFR) §70.61.

5.1 SAFETY PROGRAM

The Mixed Oxide (MOX) Fuel Fabrication Facility (MFFF) safety program consists of process safety information; an ISA that analyzes MFFF hazards and potential accident sequences, and identifies Items Relied Upon for Safety (IROFS); and management measures to ensure that IROFS are available and reliable to perform their function when needed. These three elements of the safety program as described in 10 CFR §70.62 and §70.65 are discussed below.

5.1.1 Process Safety Information

MOX Services compiles and maintains current written process safety information for the MFFF to identify and understand the hazards associated with the processes, and to update the ISA as required. This information is contained in documentation that is prepared, reviewed, and approved in accordance with the MFFF configuration management process (see Chapter 15) and includes the following:

- A description of the hazards, including information on the pertinent chemical or physical properties of hazardous materials (e.g., toxicity, acute exposure limits, reactivity, thermal and chemical stability, or other applicable information that would typically be included on Material Safety Data Sheets)
- A description of the equipment used in the process (e.g., information of a general nature on such topics as the materials of construction, piping and instrumentation diagrams, ventilation, design codes and standards employed, material and energy balances, safety systems, interlocks, fire detection or suppression systems, electrical classification, relief system design, and the design bases)
- A description of the technology of the process (e.g., block flow or simplified process flow diagrams, a brief outline of the process chemistry, upper and lower limits for controlled parameters, and an evaluation of health and safety consequences of process deviations).

5.1.2 Integrated Safety Analysis

An ISA is conducted with an appropriate level of detail for the complexity of the processes involved (10 CFR §70.62(c)). MOX Services has conducted this ISA to demonstrate compliance with 10 CFR §70.61. The ISA supports preparation of an ISA Summary (as a separate submittal that is not a part of this License Application—as specified by 10 CFR §70.65(b)), a document that summarizes the conclusions of the analyses done as a part of the ISA process. The ISA is a systematic analysis to identify: plant internal and external hazards and their potential for initiating event sequences; the potential event sequences; their likelihood and consequences; and the structures, systems, and components (SSCs) and activities of personnel that are relied on for safety (i.e., IROFS).

The consequence severity levels that are used in the hazard evaluation are based on 10 CFR §70.61 and are provided in Table 5.1-1. Risk is the product of the event likelihood and consequences. The risk of each credible event is determined by cross-referencing the severity of the consequence of the unmitigated accident sequence with the likelihood of occurrence in a risk matrix. A risk matrix, shown in Table 5.1-2, is used to determine the requirement for IROFS.

The ISA demonstrates that the IROFS are adequate to perform their intended safety functions when necessary. The ISA is an ongoing process and is maintained during all phases of the facility life cycle. MOX Services has completed an ISA in accordance with the methods and criteria contained in the ISA Summary and the programmatic commitments discussed below. MOX Services commits to maintaining the ISA.

5.1.3 Management Measures

Management measures are applied to IROFS by providing the administrative and programmatic framework for configuration management, maintenance, training and qualification, procedures, audits and assessments, incident investigation, and records management. IROFS and appropriate management measures are implemented based on the results of the ISA to ensure compliance with the performance requirements of 10 CFR §70.61. MOX Services implements and maintains these management measures, as described in Chapter 15, to ensure the required reliability and availability of IROFS. The application of management measures to IROFS is described in Section 5.2.5.2.4.

5.1.4 Control Of Facility And Process Changes

MOX Services maintains the ISA, ISA Summary, and License Application (LA) so that they are accurate and up-to-date by means of the MFFF configuration management processes, which include written procedures. MOX Services evaluates changes to the facility and its processes for impact on the ISA and LA, and updates the LA and ISA Summary, as needed, in order to ensure their continued accuracy. The evaluation of the facility and process changes includes identification and impact of changes to parameters used in the postulated accident sequences of the ISA (including event likelihood and consequences). Responsibility for maintaining and updating the ISA, ISA Summary, and the LA belongs to the ES&H Licensing Manager, as described in Chapter 4.

MOX Services will address safety-significant vulnerabilities or unacceptable performance deficiencies, if any are identified, in the evaluation of the proposed facility and process changes. MOX Services will take prompt and appropriate actions to address vulnerabilities that are identified.

MOX Services controls facility and process changes in accordance with the following requirements:

- A change to the facility or its processes is evaluated, as described above, before the change is implemented. The evaluation of the change determines, before the change is implemented, whether an application for an amendment to the license is required to be submitted in accordance with 10 CFR §70.34.
- The sites, structures, processes, systems, equipment, components, computer programs, and activities of personnel are described in both this License Application and in the accompanying ISA Summary. Pursuant to 10 CFR §70.72, MOX Services may make changes to these items, as described in the ISA Summary, without prior U.S. Regulatory Commission (NRC) approval, if the change:
  - Does not create new types of accident sequences that, unless mitigated or prevented, could exceed the performance requirements of 10 CFR §70.61, and that have not previously been described in the ISA Summary;
  - Does not use new processes, technologies, or control systems for which MOX Services has no prior experience;
  - Does not remove, without at least an equivalent replacement of the safety function, an IROFS that is listed in the ISA Summary and is necessary for compliance with the performance requirements of 10 CFR §70.61;
  - Does not alter an IROFS, listed in the ISA Summary, that is the sole item preventing or mitigating an accident sequence that exceeds the performance requirements of 10 CFR §70.61; and
  - Is not otherwise prohibited by 10 CFR §70.72, license condition, or order.
- If a change allowed under 10 CFR §70.72 is made, the affected onsite documentation will be updated promptly per written procedures.
- MOX Services maintains records of changes to its facility carried out under 10 CFR §70.72. These records include a written evaluation that provides the bases for the determination that the changes do not require prior NRC approval under paragraphs (b) and (c) of 10 CFR §70.72. These records are maintained until termination of the license.
- Changes are communicated to the NRC as follows:
  - For changes that require NRC pre-approval under 10 CFR §70.72, MOX Services submits an amendment request to the NRC in accordance with 10 CFR §70.34 and §70.65.
  - For changes that do not require NRC pre-approval under 10 CFR §70.72, MOX Services submits to the NRC annually, within 30 days after the end of the calendar year during which the changes occurred, a brief summary of the changes to the records required by 10 CFR §70.62(a)(2).
  - For changes that affect the ISA Summary, MOX Services submits to the NRC annually, within 30 days after the end of the calendar year during which the changes occurred, revised ISA Summary pages.

5.1.5 Records Of Failures

Deficiencies in IROFS or failure of management measures are addressed in accordance with the corrective action program described in the MOX Project Quality Assurance Plan (MPQAP). MOX Services maintains records of failures, readily retrievable and available for inspection by the NRC, documenting each discovery that an IROFS or management measure has failed to perform its function upon demand, or has degraded such that the performance requirements of 10 CFR §70.61 are not satisfied. These records identify the IROFS or management measure that has failed and the safety function affected, the date of discovery, date (or estimated date) of the failure, duration (or estimated duration) of the time that the item was unable to perform its function, other affected IROFS or management measures and their safety function, affected processes, cause of the failure, whether the failure was in the context of the performance requirements or upon demand or both, and corrective or compensatory action that was taken. Failure is recorded at the time of discovery, and the record of failure is updated promptly upon the conclusion of the failure investigation.

Table 5.1-1. Consequence Severity Categories Based on 10 CFR §70.61

Consequence Category	MFFF Facility & Site Workers	IOC	Environment
3: High (H)	TEDE ≥ 1 Sv (100 rem) CC ≥ AEGL3, ERPG3, TEEL3	TEDE ≥ 0.25 Sv (25 rem) CC ≥ AEGL2, ERPG2, TEEL2 Soluble uranium intake ≥ 30 mg; Insoluble uranium respirable intake ≥ 30 mg	
2: Intermediate (I)	1 Sv > TEDE □> 0.25 Sv (100 rem > TEDE >□□□ rem) AEGL3, ERPG3, TEEL3 > CC > AEGL2, ERPG2, TEEL2 Soluble uranium intake ≥ 30 mg; Insoluble uranium respirable intake ≥ 30 mg	0.25 Sv > TEDE >□0.05 Sv (25 rem > TEDE >□5 rem) AEGL2, ERPG2, TEEL2 > CC > AEGL1, ERPG1, TEEL1 10 mg ≤ soluble uranium intake < 30 mg; 10 mg ≤ Insoluble uranium respirable intake < 30 mg	Radioactive release > 5000 × (Table 2 in Appendix B of 10 CFR Part 20)
1: Low (L)	Events of lesser radiological and chemical exposures to workers than those above in this column	Events of lesser radiological and chemical exposures to the IOC than those above in this column	Radioactive releases producing effects less than those specified above in this column

TEDE – Total Effective Dose Equivalent (see Section 5.2.3.1.2)

CC – Chemical Consequences (see Section 5.2.4.2)

AEGL – Acute Exposure Guideline Level (1, 2, and 3 refer to the severity level; see Section 5.2.4.2)

ERPG – Emergency Response Planning Guideline (1, 2, and 3 refer to the severity level; see Section 5.2.4.2.)

TEEL – Temporary Emergency Exposure Limits (1, 2, and 3 refer to the severity level; see Section 5.2.4.2.)

Note: In the calculation of chemical consequences, AEGLs and ERPGs values were not established for many of the MFFF chemicals. Therefore, values issued by the DOE in WSMS-SAE-002-001, Revision 18, and listed in are used as quantitative standards for determining the consequence category thresholds. Listed values include ERPGs and TEELs. Additionally, intakes are used instead of concentration-based TEELs to establish consequence categories for uranium accidents.

Table 5.1-2. Event Risk Matrix

CONSEQUENCE		Highly Unlikely (1)	Unlikely (2)	Not Unlikely (3)
High	(3)	3 No IROFS Applied	6 IROFS Applied	9 IROFS Applied
Intermediate	(2)	2 No IROFS Applied	4 No IROFS Applied	6 IROFS Applied
Low	(1)	1 No IROFS Applied	2 No IROFS Applied	3 No IROFS Applied

LIKELIHOOD

Table 5.1-3. TEELs Used as Chemical Limits for Chemicals at the MFFF (Note 1) (mg/m^3)

Name	TEEL-1	TEEL-2	TEEL-3
Acetic Acid	35	75	125
Acetonitrile	100	100	750
Aluminum Nitrate	15	15	500
Argon	350,000	500,000	750,000
Ascorbic Acid	200	500	500
Azodicarbonamide	125	500	500
Boric Acid	30	50	125
Dry cement (i.e., calcium carbonate)	15	15	15
Calcium Nitrate	3.5	25	125
Chromic (VI) Acid	1	2.5	25
Chlorine*	3	7.5	60
Diluent (C10-C13 Isoalkanes) (Note 2)	5	35	200
• Decane (C10)	5	35	25000
• Undecane (C11)	6	40	200
• Dodecane (C12)	15	100	750
• Tridecane (C13)	60	400	500
Ethanol	500	3,500	15,000
Ethylene glycol	50	100	150
Ferrous sulfamate	3	5	25
Ferrous sulfate	7.5	12.5	350
Fluorine*	0.75	7.5	30
Hydrazine*	0.7	6.6	40
Hydrazine Monohydrate	0.0075	0.06	50
Hydrazine Nitrate	3	5	5
Hydrofluoric Acid*	1.5	15	40
Hydrochloric Acid*	4	30	200
Hydrogen Peroxide*	12.5	60	125
Hydroxylamine Nitrate	15	26	125
Iron	30	50	500
Isopropanol	1000	1000	5000
Manganese	3	5	500

Table 5.1-3. TEELs Used as Chemical Limits for Chemicals at the MFFF (Note 1) (mg/m^3) (continued)

Name	TEEL-1	TEEL-2	TEEL-3
Manganese Nitrate	10	15	500
Manganous Sulfate	7.5	12.5	500
Methanol*	262	1308	6540
Nitric Acid*	2.5	15	200
Nitric Oxide	30	30	125
Nitrogen Dioxide	7.5	7.5	35
Nitrogen Tetroxide	15	15	75
Oxalic Acid	2	5	500
Potassium Hydroxide	2	2	150
Potassium Iodide	0.75	6	300
Potassium Nitrate	3.5	20	500
Potassium Permanganate	7.5	15	125
Silver Nitrate	0.03	0.05	10
Silver Oxide	30	50	75
Sodium Acetate	30	500	500
Sodium Carbonate	30	50	500
Sodium Hydroxide*	0.5	5	50
Sodium Nitrate	1	7.5	100
Sodium Nitrite	0.125	1	60
Sodium Oxalate	30	50	50
Sodium Sulfite	30	50	100
Sulfuric Acid*	2	10	30
Sulfamic Acid	40	250	500
Thenoyl TrifluoroAcetone	3.5	25	125
Tributyl Phosphate	6	10	300

Table 5.1-3. TEELs Used as Chemical Limits for Chemicals at the MFFF (Note 1) (mg/m^3) (continued)

Name	TEEL-1	TEEL-2	TEEL-3
Xylene	600	750	4000
Zinc Stearate	30	50	400
Zirconium nitrate	35	35	50

* Values are based on Emergency Response Planning Guideline (ERPG) concentrations.

Notes:

1. Temporary Emergency Exposure Limits (TEELs) are derived from approved methodologies developed by Department of Energy Subcommittee on Consequence Assessment & Protective Actions (SCAPA) and are identified in WSMS-SAE-02-0001, Revision 18.
2. The TEEL values for diluent represent the most conservative value in each category among the following primary constituents: n-decane, n-undecane, n-dodecane, and n-tridecane.

Table 5.1-4. Application of Chemical Limits to Qualitative Chemical Consequence Categories

Consequence Category	Worker	IOC
High	Concentration ≥ TEEL-3	Concentration ≥ TEEL-2 Soluble uranium intake ≥ 30 mg Insoluble uranium respirable intake ≥ 30 mg
Intermediate	TEEL-3 > Concentration ≥ TEEL-2 Soluble uranium intake ≥ 30 mg Insoluble uranium respirable intake ≥ 30 mg	TEEL-2 > Concentration ≥ TEEL-1 30 mg > Soluble uranium intake ≥ 10 mg 30 mg > Insoluble uranium respirable intake ≥ 10 mg
Low	TEEL-2 > Concentration Soluble uranium intake < 30 mg Insoluble uranium respirable intake < 30 mg	TEEL-1 > Concentration Soluble uranium intake < 10 mg Insoluble uranium respirable intake < 10 mg

Notes:

1. Temporary Emergency Exposure Limits (TEELs) are derived from approved methodologies developed by Department of Energy Subcommittee on Consequence Assessment & Protective Actions (SCAPA) as identified in WSMS-SAE-02-0001, Revision 18, and listed in Table 5.1-3.
2. Intakes are used instead of concentration-based TEELs to establish consequence categories for uranium accidents.

5.2 INTEGRATED SAFETY ANALYSIS METHODS

MOX Services shall maintain an ISA that identifies and evaluates hazards associated with operation of the MFFF. The ISA is developed, used, and maintained during the life of the facility.

The major steps in the ISA process are as follows:

- Identify internal facility hazards, natural phenomena hazards (NPHs), and external man-made hazards (EMMHs) that could affect the safety of licensed material
- Identify radiological hazards related to possessing or processing licensed material at the facility
- Identify chemical hazards of licensed material and hazardous chemicals produced from licensed material
- Develop potential events involving the identified hazards
- Determine the consequence and the likelihood of potential events, and the methods used to determine the consequences and likelihoods
- Determine IROFS and the characteristics of their preventive, mitigative, or other safety function, and the assumptions and conditions under which the item is relied upon to support compliance with the performance requirements of 10 CFR §70.61

The following sections provide a description of the ISA steps.

5.2.1 Hazard Identification

Hazard identification is performed to identify the hazardous materials and hazardous energy sources associated with the operations of the MFFF process and auxiliary units. The ISA Team utilizes a checklist of hazardous materials and hazardous energy sources in the hazard identification process. The checklist is developed and used in accordance with the Checklist Analysis and What-If/Checklist methods of *Guidelines for Hazard Evaluation Procedures – Second Edition – With Worked Examples*, Center for Chemical Process Safety, American Institute of Chemical Engineers, New York, NY, 1992. The checklist is tailored for the MFFF and includes hazardous material, energy sources, confinement types and auxiliary systems.

A chemical interaction matrix is used to identify chemical hazards introduced by the mixing of incompatible chemicals and reagents. The matrix is facility specific and includes the chemicals and reagents used at MFFF. The chemical interaction matrix along with a listing of hazardous materials is provided in Chapter 8, Chemical Safety. Hazard identification is performed as part of the process hazards analysis discussed in Section 5.2.2.

External man-made and natural phenomena hazards are identified as part of the ISA process. A checklist analysis is also used to identify natural phenomena hazards (NPH) that may affect the MFFF. Initially, a comprehensive list of natural phenomena is assembled by performing a review of information provided in applicable documents including federal regulations, DOE Standards, DOE Orders, NRC NUREGS, and facility safety analysis reports. Examples include Savannah River Site Generic Safety Analysis, NUREG-0800, 10 CFR Part 100, and various DOE Standard addressing natural phenomena. Following preparation of the checklist, a screening process identified those NPH with the potential for affecting the MFFF during the period of operation. Screening criteria are based on NPH definition, NPH effects, NPH frequency of occurrence, and facility site characteristics.

A checklist analysis is also used to identify external man-made hazards that may affect the MFFF. The list was developed through an extensive documentation review of Savannah River Site (SRS) information, including site maps, site visits, and SRS Generic Safety Analysis Report. Information provided in U.S. Nuclear Regulatory Commission (NRC) regulatory requirements, U.S. Department of Energy (DOE) guidance documents, DOE Orders and NRC NUREGs is also used in the identification of potential external events. Following preparation of the comprehensive checklist, a screening process identified those external man-made events with the potential for affecting the MFFF during the period of operation. Guidance provided by NUREG/CR-4839 is used in the screening process.

Hazard identification checklists are maintained as part of the ISA. Changes or modifications to the facility will be reviewed to ensure that no new hazard is introduced. The evaluation of potential external man-made-made events shall be reviewed to consider proposed or projected changes that may affect the MFFF. It is not expected that new NPH be identified for the life of the facility.

5.2.2 Process Hazards Analyses

Potential events involving the identified hazards are developed and evaluated by the performance of Process Hazards Analyses (PrHA). PrHA are performed for each process unit to identify specific event scenarios in detail, including causes of the events, and associated prevention and mitigation features (IROFS) . All modes of operation are considered, including startup, normal operation, shutdown, and maintenance. PrHAs are performed in accordance with the guidance provided in *Guidelines for Hazard Evaluation Procedures – Second Edition – With Worked Examples*, Center for Chemical Process Safety, American Institute of Chemical Engineers, New York, NY, 1992 and *Integrated Safety Analysis Guidance Document,* NUREG-1513, U.S. Nuclear Regulatory Commission, 1999.

The specific PrHA methodologies utilized for each process unit are selected using the guidance of *Guidelines for Hazard Evaluation Procedures – Second Edition – With Worked Examples*, Center for Chemical Process Safety, American Institute of Chemical Engineers, New York, NY, 1992 and *Integrated Safety Analysis Guidance Document,* NUREG-1513, U.S. Nuclear Regulatory Commission, 1999.

The AP processes are chemical fluid systems in nature and complex. The MP processes are mechanical systems consisting of moving powder and pellets through various manufacturing steps. Utilization of the technique selection flowchart provided in the AIChE and NUREG-1513 guidelines resulted in the selection of the Hazard and Operability Analysis (HAZOP) technique for AP processes and the What-If/Checklist Analysis technique for the MP processes.

While HAZOP and What-If/Checklist are the main techniques used to evaluate MFFF events, supplemental hazard evaluations may be performed in specific instances to support the ISA. These supplemental analyses are performed to gain insight into event likelihoods, event sequences, single failure vulnerability and other safety aspects of hazards evaluation and may include such techniques as Preliminary Hazards Analysis, Failure Modes and Effects Analysis (FMEA), Fault Tree Analysis and Event Tree Analysis. Selection of techniques is based upon the specific application and the guidance of AIChE and NUREG-1513.

A team leader organizes and distributes technical information to a team of individuals with a variety of backgrounds and experiences. The team meets and together identifies event scenarios, causes, and prevention/mitigation features (IROFS and their safety function) in a step by step manner. As necessary, recommendations are made to modify the design, identify additional analyses to be performed, or actions to be taken to support the identification of the IROFS that are required to satisfy the requirements of 10 CFR §70.61.

For each credible accident event sequence determined to potentially result in unacceptable consequences, the PrHA identifies the IROFS necessary to support the argument that the performance requirements of 10 CFR §70.61 are satisfied.

The PrHAs utilize dose threshold calculations to screen event sequences whose consequences are acceptable to all potential receptors. For facility workers, dose threshold calculations identified the quantity of material (Material At Risk - MAR) that would result in dose consequences from radiation inhalation greater than the low consequence category defined in Table 5.1-1. The assessment of consequence is discussed in Sections 5.2.4 and 5.2.5.

Fire Hazards Analysis (FHA)

A Fire Hazards Analysis (FHA) will be maintained that documents the specific fire hazards, the fire protection features proposed to control those hazards, and the adequacy of MFFF fire safety program. The FHA provides information for each fire area and describes operational concerns that can affect fire safety in the MFFF. Additionally, a thorough systematic analysis of the fire potential at the MFFF ensures that adequate fire barriers and fire protection features are incorporated into the MFFF design.

The FHA verifies the combustible loading within the process areas, whether ignition sources are present, and that fires, if they occur, will remain within the initial fire area (that is, do not propagate). This information is then utilized to demonstrate that the fire barriers are not compromised, fires will not affect radioactive material within the C4/C3 confinement areas (See Chapter 11), and that the effects of a given fire will not affect the ability of the HEPA filters to mitigate a release that may accompany a fire.

To develop the design basis fire scenario(s) for each fire area, the bounding possible fire scenario(s) are determined. The determination of each bounding scenario includes the following:

- An evaluation of the types of potential fires that are based on the combustible form (for example, electrical insulation, furniture, and so on)
- The combustible type (for example, polyethylene, polyurethane, and polycarbonate)
- The quantities of combustible materials contained in the fire area (including an allowance for transient combustibles)
- Fire severity and intensity
- The potential hazards created
- Potential ignition sources.

Each postulated fire scenario includes, as necessary, a description of the characteristics that are associated with the possible fire(s), such as maximum fire loading, hazards of flame spread, smoke generation, toxic contaminants, contributing fuels, and ignition sources.

The FHA will be reviewed and updated as necessary at defined regular intervals to document that MFFF fire protection features are adequate to ensure fire safety. In addition to this periodic review/update, the FHA will be revised as needed to incorporate significant changes and modifications to the MFFF, its processes, or combustible inventories.

5.2.3 Radiological Consequence Evaluation

The methodology for assessing radiological consequences for events releasing radioactive materials is based on guidance provided in NUREG/CR-6410, *Nuclear Fuel Cycle Facility Accident Analysis Handbook*, U.S. Nuclear Regulatory Commission, March 1998. For the site worker, the individual outside the controlled area (IOC), and the environment, conservative quantitative consequences are calculated for both the unmitigated and mitigated cases. Unmitigated results are used to establish a safety strategy. For the facility worker, conservative qualitative consequences are determined. Consequences are categorized as high (H), intermediate (I), or low (L) based on the three severity levels; see Table 5.1-1 for a description of the severity levels.

The facility worker is considered to be located inside the MFFF, near a potential accident. Consequences for the site worker and the IOC are assessed from two postulated locations in the MFFF: (1) the MFFF building stack, and (2) the Secured Warehouse Building (BSW). Potential releases from other locations within the MFFF building (i.e., the truck bay) are further away from relevant boundaries and are analyzed using stack distances. The site worker is considered to be a fixed distance of 100 m from the release point. Both facility workers and site workers are deemed to be "workers." The IOC is defined as the maximally exposed individual outside the controlled area boundary. The controlled area is a minimum distance of 68 m from the BSW, and 160 m from the MFFF building stack. The IOC is not considered a worker.

The MFFF restricted area is coincident with the protected area, an area encompassed by physical barriers and to which access is controlled. Radiological consequences to the environment are assessed outside the MFFF restricted area (that is, at the restricted area boundary). This corresponds to a distance of 28 m from the BSW, and 52 m from the MFFF building stack.

Radiological consequences to the facility worker are qualitatively determined. Radiological releases for the site worker and the IOC are conservatively modeled using a 0- to 2-hour 95th percentile dispersion χ/Q. Radiological releases to the environment are conservatively modeled using an averaged 24-hour dispersion χ/Q. No evacuation is credited for the assessment of the unmitigated radiological consequences.

5.2.3.1 Quantitative Unmitigated Consequence Analysis to Site Worker and IOC

For each identified event sequence in the hazard evaluation, a bounding consequence for that event sequence is calculated. The bounding consequence is established by determining the applicable locations and locating the specific radioactive and chemical materials at risk. The applicable, bounding material-at-risk values are then established from the identified values by selecting the maximum value for each form and each compound. Values for each form and compound are conservatively selected due to the dependence on the airborne release fraction, the respirable fraction, the specific activity, and the dose conversion factors.

5.2.3.1.1 Source Term Evaluation

The first step in the evaluation of the unmitigated consequences is to determine the source term. The source term is determined based on the five-factor formula as described in NUREG/CR-6410. The five-factor formula consists of the following parameters:

- MAR – Material-At-Risk
- DR – Damage Ratio

- ARF – Airborne Release Fraction
- RF – Respirable Fraction
- LPF – Leak Path Factor.

These parameters are multiplied together to produce a source term (ST) representative of the amount of airborne respirable hazardous material released per a bounding scenario, as follows:

$$[\mathrm{ST}] = [\mathrm{MAR}] \times [\mathrm{DR}] \times [\mathrm{ARF}] \times [\mathrm{RF}] \times [\mathrm{LPF}]$$

Material at risk (MAR) is the amount of hazardous material subject to the event of interest (e.g., fire, drop). Conservative MAR values are determined based the associated event and process unit. MAR values are listed in Chapter 8. The damage ratio (DR) is the fraction of MAR impacted by the event.

The product of airborne release fraction (ARF) and respirable fraction (RF) is the fraction of respirable material released to the surrounding atmosphere and available for uptake. Applicable ARF and RF values are established for the material forms and the release mechanisms that could potentially occur at the MFFF from values presented in NUREG/CR-6410 and DOE-HDBK-3010, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*, U.S. Department of Energy.

The leak path factor (LPF) is the fraction of material leaving a defined confinement barrier. The LPF in all unmitigated cases is conservatively assumed to be one (that is, no credit is taken for removal of hazardous material). Section 5.2.4.4 contains a discussion of LPFs utilized in mitigated radiological consequence evaluations.

5.2.3.1.2 Dose Evaluation

The source term is used to calculate the total effective dose equivalent (TEDE). TEDE values are calculated for exposure via the inhalation pathway to a site worker (S) and IOC. Other potential pathways (for example, submersion and ingestion) are not considered to contribute a significant fraction to the calculated TEDE. The following expression is used to calculate the TEDE for potential radiological releases at the MFFF:

$$[\mathrm{TEDE}]^{\mathrm{S,IOC}} = \sum_{x=1}^{N} BR \times [\chi / \mathrm{Q}]^{\mathrm{S,IOC}} \times ST_x \times DCF_x$$

Where:

- $[\mathrm{TEDE}]^{\mathrm{S,IOC}}$ is the total effective dose equivalent to the site worker or IOC (rem),
- x represents an individual material,
- N represents the total number of materials,
- BR is the breathing rate (m^3/s),

- $[\chi/Q]^{S,IOC}$ is the relative concentration factor unique to the site worker or IOC (s/m^3),
- ST_x is the respirable source term of an individual material x (kg), and
- DCF_x is the committed inhalation dose conversion factor of each material x (rem/kg).

Atmospheric dispersion factors (χ /Q) for the site worker and IOC are established from SRS data using the ARCON96 computer code.

The breathing rate (BR) is based on the guidance of Regulatory Guide 1.25, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)*, U.S. Nuclear Regulatory Commission, March 1972.

To determine the mass-based inhalation DCF for each type of material, the isotopic mass fraction is multiplied by the mass-based DCF for each isotope y in the material and the products are summed. Expressed algebraically:

$$DCF_{material} = \sum_{y=1}^{N} f_y \times DCF_y$$

Where:

- $DCF_{material}$ is the committed inhalation dose conversion factor of the material x (rem/kg material),
- f_y is the mass fraction of each isotope in the material per kg (mass isotope/mass material),
- DCF_y is the committed inhalation dose conversion factor of each isotope (rem/kg isotope),
- y stands for a dose contributing isotope, and
- N stands for the total number of dose contributing isotopes in the material x.

Activity-based inhalation DCF values are taken from Federal Guidance Report No. 11 based on the form of the potential releases from the MFFF when received by the dose receptor.

5.2.3.2 Consequence Analysis for the Facility Worker

Facility worker consequences are qualitatively determined based on the material released, the release mechanism, and the location of the worker relative to the release. In most cases, events involving an airborne release of plutonium or americium are judged to have high consequences to the facility worker and IROFS are applied. However, threshold values of MAR below which facility worker exposures exceeding 25 rem are not possible are used to categorize some events as low.

5.2.3.3 Environmental Consequences

A 24-hour average effluent concentration (EC) is calculated for a release to the environment of each of the released radionuclides *x* using the following expression:

$$EC_x = \frac{\chi/Q}{24 \text{ hours}} \times \frac{ST_x}{RF}$$

Where:

- EC is the 24-hour average concentration at the restricted area boundary (kg/m^3),
- $\frac{\chi}{Q}$ is the 24-hour relative concentration factor at the restricted area boundary (s/m^3),
- ST_x is the respirable source term (kg),
- RF is the respirable fraction for the event (unitless)

The 24-hour average atmospheric dispersion factor (χ/Q) for ground-level releases at the restricted area boundary is calculated by ARCON96 for releases from the BSW and the MFFF building stack.

Since the radiological consequences to the environment are limited to an airborne effluent concentration and not a respirable quantity, the respirable fraction (RF) used in calculating the average effluent concentration in the above equation corrects the source term, from the previous five-factor formula, such that the source term reflects an airborne quantity.

The *EC* values are not additive, so a sum of the fractions rule is applied to determine the ratio of the calculated *EC* value to the performance limit, which is 5000 times the value specified in Table 2 of Appendix B to 10 CFR Part 20 for airborne releases. Thus,

$$EC\ Ratio = \sum_{x=1}^{N} \frac{EC_x}{5000 \times Part20EC_x}$$

Where:

- *EC Ratio* is the ratio of the *EC* value for the event to the performance limit (unitless)
- $Part20EC_x$ is the value specified in Table 2 of Appendix B to 10 CFR Part 20 (kg/m^3),
- *x* represents one radionuclide, and
- *N* represents the total number of radionuclides

The event scenarios are mapped during the PrHAs into event groups. The environmental consequences for the bounding event within an event group are analyzed in the NSE event consequence analysis calculations.

5.2.3.4 Quantitative Mitigated Consequence Analysis

The methodology used to establish the mitigated radiological consequences closely follows the methodology used to establish the unmitigated consequences. Mitigated consequences are calculated for those bounding events representing an event grouping in which mitigation features are utilized to reduce the risk in accordance with 10 CFR §70.61.

To perform the mitigated consequence analysis, applicable bounding LPFs are utilized based upon the IROFS providing mitigation. In the case of ventilation systems and associated HEPA filters, applicable bounding values for the LPF are established in NUREG/CR-6410.

Conservative LPFs utilized in the calculation of quantitative mitigated consequence will be determined in accordance with NUREG/CR-6410.

5.2.4 Chemical Consequence Evaluation

This section provides the methodology for the evaluation of chemical consequences that are associated with a release of hazardous chemicals produced from licensed materials as defined by 10 CFR Part 70.

According to 10 CFR Part 70, hazardous chemicals produced from licensed materials are identified as "substances having licensed material as precursor compound(s) or substances that physically or chemically interact with licensed materials; and that are toxic, explosive, flammable, corrosive, or reactive to the extent that they can endanger life or health if not adequately controlled. These include substances commingled with licensed material, but do not include substances prior to process addition to licensed material or after process separation from licensed material."

Hazards that involve only chemicals and that do not affect radiological safety are addressed in accordance with applicable Occupational Safety and Health Administration (OSHA) requirements. Non-routine work safety is addressed through the use of work authorization and task analysis or activity-based hazard analysis.

5.2.4.1 Methodology

A range of initial conditions is considered to identify the physical processes that control the nature and rate of vapor generation and release. Failure modes of storage containers and associated systems are also considered. The following release scenarios are addressed:

- Leaks and ruptures involving equipment vessels and piping leaks
- Evaporating pools formed by spills and tank failures
- Flashing and evaporating liquefied gases from pressurized storage.

Explosion events that could result in the release of hazardous chemical vapors are evaluated in the ISA. The chemical consequences are based on bounding analyses.

Facility worker consequences are qualitatively determined based on the material released, the release mechanism, and the location of the worker relative to the release. In most cases, events involving an airborne release of plutonium or americium are judged to have high consequences to the facility worker and IROFS are already applied. In lieu of a mechanistic calculation of the release, a conservative bounding release model is used to determine the consequences to the site worker and IOC from releases either from the BSW or the MFFF building stack, as applicable. Releases are modeled to occur using the total material at risk from the largest single tank or container. Furthermore, no credit is afforded to process equipment installed to remove/scrub some of the potentially released chemicals prior to release from the MFFF.

Estimates of hazardous chemical concentrations include techniques, assumptions, and models that are consistent with industry practice, are verified and/or validated, and follow the guidance on atmospheric and consequence modeling found in NUREG/CR-6410, *Nuclear Fuel Cycle Accident Analysis Handbook*. The analysis to determine the effects to the IOC is based on the following assumptions:

- A ground level release (conservative)
- No mechanical or buoyancy plume rise (conservative)
- Neutrally buoyant gas model (conservative).

These bounding assumptions envelop uncertainties inherent in realistic analyses.

Chemical consequence analyses are performed assuming the largest credible unmitigated spill or loss of containment accident involving these chemicals. Airborne concentrations are calculated for the site worker and the IOC. These concentrations are then compared to the chemical limits presented in Table 5.1-1. From this comparison, a consequence category is established (low, intermediate, high) using the guidance outlined in Table 5.1-4. These consequence categories correspond to those identified in 10 CFR §70.61.

Non-hazardous chemicals and gases are not evaluated. Except for oxygen, exposure to these gases poses an asphyxiant hazard only. Gas concentrations at asphyxiation levels are not credible at the distances corresponding to the site worker and the IOC. Oxygen has no established toxicity limit.

Several different methodologies are applied to the performance of chemical consequence analyses based on the nature of the chemical and the location of the receptor. For calculating airborne concentrations involving evaporative releases, the more conservative release rate from two separate evaporation models is used as input to the ARCON96 (Atmospheric Relative Concentrations in Building Wakes) computer code. The sources of these alternate evaporation models are: (1) a Journal of Hazardous Materials article (Kawamura, P. I., and D. Mackay, The evaporation of volatile liquids. J. Hazardous Materials 15:343-364, 1987), and (2) NUREG/CR-6410 (*Nuclear Fuel Cycle Facility Accident Analysis Handbook*, March 1998).

5.2.4.2 Chemical Consequence Criteria

Acute Exposure Guideline Level (AEGL) values and Emergency Response Planning Guideline (ERPG) values for chemical consequence categories meet the definitions for the qualitative chemical consequences performance criteria given in 10 CFR §70.61. However, since AEGL and ERPG values are not established for many of the MFFF chemicals, values presented in Table 5.1-3 are used as quantitative standards for determining the consequence category thresholds. Values in this table include ERPGs and Temporary Emergency Exposure Limits (TEELs). TEELs were adopted by the DOE Subcommittee on Consequence Assessment and Protective Action (SCAPA). The SCAPA-approved methodology was used to obtain hierarchy-derived TEELs.

The original TEEL methodology used only hierarchies of published concentration limits (that is, Permissible Exposure Levels [PELs] or Threshold Limit Values – Time-Weighted Averages [TLV-TWAs], Short-Term Exposure Levels [STELs], and Immediately Dangerous to Life and Health [IDLH] values) to provide estimated values approximating ERPGs. The expanded method for deriving TEELs also includes published toxicity data (Toxic Dose Low $[TD]_{Lo}$, Toxic Concentration Low $[TC]_{Lo}$, 50% Lethal Dose $[LD]_{50}$, 50% Lethal Concentration $[LC]_{50}$, Lethal Dose Low $[LD]_{Lo}$, and Lethal Concentration Low $[LC]_{Lo}$). Hierarchy-based values take precedence over toxicity-based values, and human toxicity data are preferred to animal toxicity data. Subsequently, default assumptions based on statistical correlation of ERPGs at different levels (for example, ratios of ERPG-3s to ERPG-2s) were used to calculate TEELs where there were gaps in the data. The TEEL hierarchy/toxicity methodology was used to develop community exposure limits for over 1,200 chemicals to date. The following are the TEEL definitions:

- TEEL-0 – The threshold concentration below which most people will experience no appreciable risk of health effects.
- TEEL-1 – The maximum concentration in air below which it is believed nearly all individuals could be exposed without experiencing other than mild transient adverse health effects or perceiving a clearly defined objectionable odor.
- TEEL-2 – The maximum concentration in air below which it is believed nearly all individuals could be exposed without experiencing or developing irreversible or other serious health effects or symptoms that could impair their abilities to take protective action.
- TEEL-3 – The maximum concentration in air below which it is believed nearly all individuals could be exposed without experiencing or developing life-threatening health effects.

For uranium accidents, intakes are used instead of concentration-based TEELs to establish consequence categories. An event that results in an intake of 30 mg soluble uranium or a respirable intake of 30 mg insoluble uranium may be considered to lead to irreversible or other serious, long-lasting health effects to any individual. An intake of 10 mg soluble uranium or a respirable intake of 10 mg of insoluble uranium may be considered to cause mild transient health effects (Hartmann, Heidi M., Frederick A. Monette, and Halil I. Avci, "Overview of Toxicity

Data and Risk Assessment Methods for Evaluating the Chemical Effects of Depleted Uranium Compounds," *Human and Ecological Risk Assessment*, Vol. 6, No. 5, pp. 851-874, 2000). Hence, controls are applied to events if the potential intake of soluble uranium or respirable intake of insoluble uranium exceeds 10 mg to the IOC or 30 mg to a worker.

Table 5.1-4 provides the corresponding chemical consequence categories for comparison to 10 CFR §70.61.

5.2.4.3 Latent Impacts

There are no residual, long-term impacts to facility workers, site workers, or the IOC that could result from an acute chemical exposure to licensed material or hazardous chemicals produced from licensed material. There are only two "potential carcinogens" at MFFF (that is, chemicals on the list of "potential carcinogens"). The two chemicals are hydrazine and uranium (soluble and insoluble).

For evaluating site workers exposed to a chemical release, the calculated concentration of an airborne chemical at 100 meters is compared to a TEEL-2 value. For evaluating the IOC exposed to a chemical release, the calculated concentration of an airborne chemical at the controlled area boundary is compared to a TEEL-1 value.

The TEEL determination process considers latent health effects (that is, cancer). The determination process (for TEEL-2 and TEEL-3 values) selects hierarchy-based values first, if available, followed by toxicity-based values. TEEL-2 values are based on Emergency Response Planning Guideline (ERPG-2) values when available, or on Permissible Exposure Limits (PEL), Threshold Limit Values (TLV), or Recommended Exposure Limit (REL) ceiling (C) values, or on 5 × TLV- Time Weighted Average (TWA) values, in order of availability, followed by toxicity-based values. TEEL-2 values, along with ERPG, PEL, TLV, or REL ceiling (C) values, take into account latent health effects (that is, cancer) where appropriate. TEEL-3 values are based on Emergency Response Planning Guideline (ERPG-3) values when available or on Immediately Dangerous to Life and Health (IDLH) values, in order of availability, followed by toxicity-based values. Since the ERPG committee considers latent health effects, TEEL-3 values also take into account latent health effects (that is, cancer) where appropriate. TEEL-1 values are less than or equal to TEEL-2 values and ensure that exposures do not result in latent health effects.

Therefore, by using the TEEL values as limits, the chemical consequence analysis has taken into account latent health effects (that is, cancer) from the two potential carcinogens at MFFF.

5.2.4.4 Uncertainty

Estimates of risks are often accompanied by uncertainty because of the complexity of the postulated scenarios and physical models used to describe them. Conservative models are utilized for the chemical releases with the intent to bound any anticipated uncertainty.

5.2.5 Likelihood Evaluation

Event sequence likelihoods are evaluated to show that the performance requirements of 10 CFR §70.61 are satisfied. The evaluation method is qualitative and is implemented through the definition of likelihood terms and specific criteria that demonstrate the reliability of identified IROFS.

Supplemental analyses may be performed to support likelihood determinations obtained with the qualitative method. These analyses provide insight into event likelihoods, event sequences, single failure vulnerability and other safety aspects of hazards evaluation and may include such techniques as Failure Modes and Effects Analysis (FMEA), Fault Tree Analysis and Event Tree Analysis. Selection and performance of these techniques are based upon the specific application and the guidance of the following documents:

- Guidelines *for Hazard Evaluation Procedures – Second Edition – With Worked Examples*, Center for Chemical Process Safety, American Institute of Chemical Engineers, New York, NY, 1992 and,
- *Integrated Safety Analysis Guidance Document,* NUREG-1513, U.S. Nuclear Regulatory Commission, 1999.

5.2.5.1 Likelihood Definitions

The following qualitative definitions are used in assessing event sequence likelihood:

- Not Unlikely – Events that may occur during the lifetime of the facility
- Unlikely – Events that are not expected to occur during the lifetime of the facility or events originally classified as Not Unlikely to which sufficient IROFS are applied to further reduce their likelihood to an acceptable level
- Highly Unlikely – Events originally classified as Not Unlikely or Unlikely to which sufficient IROFS are applied to further reduce their likelihood to an acceptable level
- Credible – Events that do not meet the definition of “Not Credible”
- Not Credible –
  - Natural phenomena or external man-made events with an extremely low initiating event frequency, conservatively estimated as less than once in a million years, or
  - A process upset or sequence of independent process deviations and/or human actions or errors, for which a convincing argument exists that it is unquestionably extremely unlikely, and no such sequence of events can ever have actually happened in any fuel cycle facility.

These likelihood definitions are described in a manner such that the application of the resulting requirements (which may consist of engineered controls, administrative controls, and management measures) will ensure that the performance requirements of 10 CFR §70.61 are satisfied. These definitions and methodology rely on specific identifiable characteristics of the process design that may affect the likelihood of an accident sequence, rather than subjective judgments of adequacy.

In applying the above definitions to address the performance requirements of 10 CFR §70.61, initiating events are assumed to be not unlikely. Postulated credible intermediate or high consequence events are made highly unlikely based on the application of IROFS features or controls without crediting the likelihood of the initiating event.

5.2.5.2 IROFS Reliability

To ensure that all event sequences with consequences exceeding the low consequence threshold of 10 CFR §70.61 meet the performance requirements identified in 10 CFR §70.61, the following qualitative design criteria and commitments are applied to those events and the associated IROFS:

- Application of the single failure criteria or double contingency (for nuclear criticality)
- Application of 10 CFR 50 Appendix B and NQA-1
- Application of Industry Codes and Standards
- Management Measures, including surveillance of IROFS (i.e., failure detection and repair, or process shutdown capability).

For those credible events where the single failure criteria or double contingency are not applicable (i.e., sole IROFS or passive IROFS feature), IROFS features are identified and the commitments for IROFS listed above are applied.

5.2.5.2.1 Application of Single Failure Criterion

The first design criterion, application of the single failure criterion or double contingency principle, is the most important attribute in providing adequate risk reduction for event sequences, and consequently ensuring that each respective event sequence is ultimately rendered highly unlikely. This design criterion ensures that even in the unlikely event of a failure of a single contingency, another unlikely, independent, and concurrent failure or process change is required prior to the occurrence of the event. This design criterion ensures that means are provided to protect against an event that could exceed the requirements of 10 CFR §70.61, including an inadvertent nuclear criticality.

The single failure criterion for MFFF means IROFS are required to be capable of carrying out their functions given the failure of any single active component within the system or in an associated system that supports its operation. Multiple failures resulting from a single occurrence are considered to be a single failure (also referred to as a common mode or common cause failure). Application of the single failure criterion is not required for IROFS performing a passive safety function (e.g., a glovebox providing confinement). The following hierarchy of controls has been established regarding the application of IROFS with respect to the single failure criterion:

- Protection by a single passive safety device, functionally tested on a pre-determined basis

- Protection by independent and redundant active-engineered features, functionally tested on a pre-determined basis
- Protection by a single hardware system/engineered feature, functionally tested on a pre-determined basis
- Protection by enhanced administrative controls
- Protection by simple administrative controls.

To ensure adequate implementation of the single failure criterion, the following principles are applied to the design of IROFS:

- Redundant equipment or systems – A piece of equipment or a system is redundant if it duplicates the operation of another piece of equipment or system to the extent that either may perform the required function (either identically or similarly), regardless of the state of operation or failure of the other.
- Diversity – Equipment or systems may satisfy single-failure criterion by providing diverse means of performing an IROFS safety function. This diverse means of performing the safety function is by equipment that does not duplicate the operation of another piece of equipment (redundancy), but still achieves the reliability required for the safety function. Each diverse system (means, paths, trains, etc.) or component is not required to provide for additional redundancy.
- Independence – IROFS are designed to ensure that the effects of natural phenomena and of normal operating, maintenance, testing, and postulated accident conditions on redundant equipment or systems do not result in the loss of their safety function, or are demonstrated to be acceptable on some other defined basis.
- Separation – IROFS are separated to the extent that failure of a single system component, or failure or removal from service of any IROFS that is common to the other systems and the IROFS leaves intact an IROFS satisfying applicable reliability, redundancy, and independence requirements.
- Fail-safe – IROFS are designed to fail into a safe state or into some other non-threatening defined basis if conditions such as disconnection of a system, loss of energy, or loss of pressure occur.

In cases where a single active system, component or activity of personnel is the sole IROFS preventing or mitigating an accident sequence that exceeds the performance requirements of 10 CFR § 70.61, justification is provided to demonstrate that it is designed to perform its safety function. This may include a discussion of additional management measures (e.g., increased surveillance frequencies), fail-safe characteristics, highly reliable components, or the application of non-credited additional protection features. Passive structures and components (such as buildings or tanks) are not designated as sole IROFS in the event that their design, or the design of any associated IROFS passive structure/component, precludes their failure under all credible natural phenomena and process conditions. However, these components, if relied upon, are designated as IROFS and will be specified to be quality level QL-1/QL-1LR.

5.2.5.2.2 Application of the MOX Project Quality Assurance Program (MPQAP)

The second design criterion, application of the MPQAP, ensures that the requirements for IROFS are correctly translated into specifications, drawings, procedures, and instructions. The MPQAP is implemented for quality affecting SSCs and their associated activities based on the significance of the SSC or activity to ensure the safety of workers, the public, and the environment. The highest level of QA and quality control is applied to all IROFS. Consistent with the criteria described in the MPQAP, IROFS are either QL-1 or QL-1LR. All IROFS are assigned the highest level of quality, QL-1/QL-1LR, which ensures a comprehensive application of quality assurance requirements covering all phases of the project including design, document, and configuration control, records management, procurement, materials control, installation, use of measurement and test equipment, and computer software and hardware. Within the MPQAP, quality assurance grading can also be used to identify the controls applied to IROFS and activities that support the MPQAP based upon an evaluation of the complexity and importance of the activity compared to quality, safety, risk, and the environment. Quality levels can be used to establish the level of programmatic requirements and procedural controls which are applied to SSCs and associated activities. The rigor of QA controls is commensurate with, but not limited to, the following criteria:

- The function or end use of the SSC
- The importance and end-user of the data collected or analyzed
- The consequence and likelihood of failure
- The complexity or uniqueness of the design, fabrication, or implementation
- The reproducibility of the results
- The reliability of the process
- The necessity for special controls or processes
- The ability to demonstrate functional compliance with applicable regulations.

The extent of QA controls applied to an SSC or activity varies as a function of the degree of confidence needed to achieve the desired quality. The grading process provides the flexibility to design and implement controls that best suit the facility or activity, but is not intended to reduce or in any way degrade the compliance with applicable requirements.

5.2.5.2.3 Application of Industry Codes and Standards

The third design criterion, application of recognized industry codes and standards, provides confidence in the ability of IROFS to perform their functions. The codes and standards provide the foundation for ensuring that IROFS are robust and incorporate lessons learned from the nuclear, mechanical, electrical, and instrumentation and control disciplines. Thus, they provide an effective set of engineering and procedural guidelines used to design, construct, and operate the IROFS. Application of codes and standards provides assurance that controls utilized to implement the single failure criterion or double contingency principle are sufficiently reliable.

5.2.5.2.4 Application of Management Measures

The fourth design criterion, application of management measures, is particularly important in the context of IROFS failure detection. IROFS failure detection is meant to include detection of IROFS failures and repair of the IROFS or the process is shutdown. As described in NUREG 1718, *Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility*, U.S. Nuclear Regulatory Commission, August 2000, IROFS failure detection can significantly reduce the likelihood of an accident scenario. For an accident scenario to proceed to completion, failure of one IROFS must occur, its failure must go undetected, and a second IROFS must fail.

Management measures are applied to the identified IROFS to ensure that they are reliable and available on demand. The MPQAP specifically describes the QA requirements, implementing procedural controls, and documentation requirements to address management measures as described in NUREG-1718. The set of applied management measures consists of applicable elements of the following management measures programs: quality assurance, configuration management, maintenance, training and qualification of plant personnel, plant procedures, audits and assessments, incident investigations, and records management.

Management measures are assigned based on the following types of IROFS classifications and the risk reduction level attributed to that particular IROFS:

- Passive Engineered Controls (PEC) – A device that uses only fixed physical design features to maintain safe process conditions without any required human action
- Active Engineered Controls (AEC) – A physical device that uses active sensors, electrical components, or moving parts to maintain safe process conditions without any required human action
- Enhanced Administrative Controls (EAC) – A procedurally required or prohibited human action, combined with a physical device that alerts the operator that the action is needed to maintain safe process conditions, or otherwise adds substantial assurance of the required human performance (i.e., augmented administrative control)
- Administrative Controls (AC) – A procedural human action that is prohibited or required to maintain safe process conditions (i.e., a simple administrative control).

Effective application of these well defined qualitative criteria will ensure that event sequences are highly unlikely. The application of the single failure criterion or double contingency principle and IROFS failure detection ensure that multiple undetected failures are required for an accident sequence to proceed to conclusion. Application of appropriate codes and standards and an NQA-1 QA program ensure that IROFS will be designed, operated, and maintained in a reliable manner. The application of these qualitative design criteria ensure that adequate risk reduction is achieved to satisfy the requirements of 10 CFR §70.61.

5.2.5.3 Additional IROFS Reliability Considerations

In addition to the four qualitative criteria discussed in Section 5.2.6.2, the following IROFS characteristics and qualities are defined and documented to ensure the reliability and availability of IROFS.

- Safety function – the credited safety function of each IROFS is stated in the safety evaluation with a description of the controlled safety parameter.
- Quality classification – IROFS are classified to the highest level of quality, i.e., a quality classification of QL-1/QL-1LR.
- Operating range and limits – the functional range of the IROFS is ensured to encompass both the normal operating range and the safety limit with an acceptable sensitivity over this full range.
- Emergency capabilities – operational requirements for an IROFS under emergency conditions (e.g., loss of power, etc.) is identified and demonstrated to be implemented in the design.
- Testing and maintenance requirements – testing and maintenance requirements are specified for each IROFS including a description of the means to detect failures, if available, and the applied management measures.
- Environmental design factors – environmental design characteristics necessary to ensure the IROFS remains available and reliable to perform its safety function are identified for each IROFS. These characteristics account for both short-term and long-term exposures to environmental conditions potentially detrimental to the operation of an IROFS (such as long-term chemical degradation impacts or short-term temperature transient impacts).
- Natural phenomena response – operational requirements of an IROFS during and/or after natural phenomena hazards (e.g., earthquakes, etc) are specified.
- Required instrumentation – instrumentation necessary to ensure an IROFS operation are specified.
- Applicable codes and standards – the design codes and standards applied to an IROFS (e.g., IEEE, ASME, ANS, etc.) are identified.
- Reliability – IROFS are procured under a 10 CFR 50 Appendix B, NQA-1 QA program.
- Protection from fires and explosions – Fires and explosions are specifically addressed in separate safety evaluations.

The following system level parameters are also considered in the safety evaluations and the process safety information:

- Safety margin, a comparison of the process parameter under normal conditions with the parameter's safety limit, is described.
- The type of control, passive, active, enhanced administrative control, or administrative control, is noted.

- Management measures are discussed.
- Fail-safe position, self announcing fault, or surveillance measures to limit down time are identified.
- Failure modes, if credited, are described.
- Demand rate, where specifically credited, is noted.
- IROFS failure rate is ensured by the implementation of 10 CFR 50 Appendix B, NQA-1 and commitments to industry codes and standards, and provides confidence that IROFS are at least unlikely to fail.

In addition to the individual qualities of each IROFS listed above, other reliability and availability qualities are related to the characteristics of the whole system of IROFS utilized to protect against an accident sequence. The following information is also addressed in the safety evaluations and the process safety information:

- Defense in depth features are described. These features may include normal process controls that are nearly identical to IROFS controls, but with lower set points, that reduce the potential demands on the IROFS.
- Degree of redundancy is identified. Usually, the degree of redundancy is dual, although diverse independent controls are sometimes used.
- Degree of independence is specified, usually by the use of two independent controls.
- Diversity is described where applicable. Often diversity is not practical, in which case independent controls are provided.
- Vulnerability to common cause failure is assessed and limited by having independent or diverse controls.

IROFS operability is defined in the MFFF Operating Limits Manual (OLM). A system, subsystem, component, or device is operable or has operability when it is capable of performing its specified function(s) and when all necessary support equipment required for the system, subsystem, component, or device to perform its specified IROFS function(s) is also capable of performing its related support function(s). The MFFF OLM defines operational modes, operability requirements, limiting conditions for operation and associated completion times, and required surveillances and frequencies. LCOs and surveillance frequencies are developed considering information from equipment supplier recommendations, reference plants, lessons learned from other appropriate facilities, and the safety classification (i.e., QL-1 or QL-1LR). The Limiting Condition for Operation (LCO) for MFFF IROFS components or system is defined as the lowest functional capability or performance level of the Systems, Structures or Components (SSC) required for safe operation. The use of compensatory measures in coordination with the LCOs for an IROFS is also defined in the MFFF OLM. Compliance with the LCOs ensures that the performance requirements of 10 CFR 70.61 are met.

The reliability and availability qualities of IROFS are assessed in the safety evaluations and the process safety information as described in the above three lists. This assessment ensures that the IROFS are sufficient and capable of performing their safety function(s) as described in the safety evaluations, with sufficient reliability and availability to ensure that each IROFS is at least unlikely to fail and thereby ensure the performance criteria of 10 CFR §70.61 are satisfied.

5.2.5.4 Setpoint Methodology

The determination of setpoints for Items Relied On For Safety (IROFS) is performed in accordance with the provisions of ISA standard 67.04.01-2006 and Regulatory Guide 1.105.

Safety limits for engineered and administrative IROFS are established in safety documents. From these safety limits, analytical limits are established by analysis to account for process system dynamics and transient behaviors. The analytical limits provide margin between the safety limits and the process response following activation of a protective response. From the analytical limits, the setpoints are established by analysis to account for effects of the measurement and response systems. The setpoints provide margin between the analytical limits and the protective response and include consideration of instrumentation drift and uncertainty.

Operating limits are established to provide sufficient margin between the established setpoints and normal process conditions. These limits are established to ensure sufficient margin between safety limits and normal processing conditions to prevent an event whose consequences could exceed the performance criteria of 10 CFR 70.61.

Operating procedures are developed to implement operating limits and control operations in a manner that ensures safety limits are not exceeded. An Operating Limits Manual documents the margin provided between the safety limits and normal processing conditions.

5.2.6 ISA Results

The integration of the necessary analyses and demonstration that the performance requirements of 10 CFR §70.61 are satisfied is performed in NSEs and NCSEs. NSEs/NCSEs are prepared at varying levels (e.g., the event, workshop, or process unit level) and demonstrate that the system can operate safely under normal and abnormal event conditions. For each event group, NSEs/NCSEs provide and/or summarize the information necessary to demonstrate that the requirements of 10 CFR §70.61 are satisfied. Selected analyses are performed to demonstrate that the IROFS are capable of performing their intended safety function in support of meeting the requirements of 10 CFR §70.61. For example, analyses are required to show that an IROFS can survive a seismic event and still perform its safety function before, during, and after a seismic event. These analyses are IROFS dependent and are determined on a case by case basis. The analyses performed in support of the ISA, including the PrHAs, FHA, chemical and radiological consequences, and criticality analyses, are integrated in the NSEs/NCSEs conclusions.

IROFS boundaries are defined by the IROFS assigned safety function. Associated components, including support systems, required to perform the assigned safety function are identified as IROFS. IROFS boundaries are maintained and controlled through MFFF Nuclear Safety Evaluations (NSE) and Nuclear Criticality Safety Evaluations (NCSE). These documents identify IROFS and safety functions at a group level (e.g., glovebox) in the event evaluations. Detailed listings of the associated component identifiers (e.g., NDP*GB1000) are provided in table form in the document body or attachments to the NSE/NCSE. The NSE/NCSEs are prepared and maintained in accordance with the design and records management controls of the MPQAP.

5.2.6.1 Nuclear Safety Evaluations

The NSE incorporates the results of the PHA, PrHAs and additional miscellaneous hazard evaluations to demonstrate that the performance requirements of 10 CFR §70.61 are satisfied. This demonstration includes identifying the selected safety strategy for each hazard event scenario and the IROFS required for implementation of the safety strategy. A description of each IROFS is included to show that the IROFS is capable of reliably performing its safety function. The safety function of the IROFS is identified together with the associated parameters, set points, justification for satisfying the single failure criteria, environmental qualification, failure modes, failure detection, and operating and surveillance requirements. Specific codes and standards, QA requirements and management measures applicable to the IROFS are described. A summary of the analyses demonstrating that the IROFS can perform the assigned safety function is provided.

The NSE contains a hazard assessment summary that identifies the applicable PHA and PrHA events for each event group being evaluated. Based on a review of the applicable PHA and PrHA events, the hazard assessment summary defines the NSE event groups to be evaluated.

The NSE events defined by the hazards assessment summary have been evaluated, with a list of credited IROFS and defense-in-depth features. A general description of each NSE event provides the causes of the event and the event location. The description includes a summary of process operations, sequence of events or event phenomena, as necessary to fully understand the event. The unmitigated consequences are provided for each receptor. The safety strategy is identified for each event, providing the basis for the selection of IROFS. Failure detection methods are identified for each of the cited IROFS. Defense-in-depth features that limit the challenges to these IROFS are also described. A summary is provided that includes how the performance requirements of 10 CFR §70.61 are met with the application of the identified IROFS.

The NSE identifies any specific operator actions required to implement the administrative control, the conditions related to the action, and any additional instrumentation and controls required to effectively perform the action.

Nuclear safety during design and operation is ensured for the MFFF through design and administrative practices. MFFF design and safety features are documented and controlled through the implementation of a rigorous configuration management program. Nuclear safety calculations and NSEs are maintained up-to-date and consistent with existing facility process and design features and administrative practices. Changes to these documents are controlled in accordance with the design change control and configuration management programs (see Chapter 15).

5.2.6.2 Nuclear Criticality Safety Evaluations

Operations with fissionable materials at the MFFF introduce risks of a criticality accident. Criticality safety must be ensured through design and administrative practices. Criticality safety is included in the ISA through the PrHAs described in Section 5.2.2.. NCSEs are performed to develop and document the safety basis for facility operations relating to the criticality events identified in the PrHAs. NCSEs are the main source of information demonstrating the adequacy of criticality controls and the effectiveness of administrative practices.

Criticality analysis design methods require a high level of validation. Criticality analysis methods used in MFFF design activities and facility safety programs comply with the technical guidance of ANSI/ANS-8.1-1983 (R1988), *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*, American Nuclear Society, Hinsdale, Illinois, September 9, 1998. Limits are developed specific to MFFF design applications (that is, limiting fissile material isotopic composition) using validated and approved computational methods. Validated and approved computational methods are also used to demonstrate criticality safety through analysis of specific design applications. Computational methods applied in MFFF design analysis include the KENO VI Monte Carlo criticality code and related computer code modules included in the SCALE system of codes for reactivity determination.

The validation process establishes method bias by comparing measured results from laboratory critical experiments to method-calculated results for the same systems. The verification and validation processes are controlled and documented as required by program QA procedures. Hardware system access controls are put in place to ensure that the same codes and data used in the validation are used in NCSE applications. Changes or maintenance to approved software is formally controlled and documented to the same level of control as the original verification and validation procedure.

The validation establishes a method bias by correlating the results of critical experiments with results calculated for the same systems by the method being validated. Critical experiments are selected to be representative of the systems to be evaluated in specific design applications. The range of experimental conditions (for example, material compositions and geometric arrangements) encompassed by a selected set of benchmark experiments establishes the “area(s) of applicability” over which the calculated method bias is applicable.

The validation process was documented and provided to NRC for review as part of the Construction Authorization process. MOX Services submitted its validation report in three separate parts by letter dated January 8, 2003. The latest revision to Parts I and III was submitted by letter dated July 2, 2003. The MFFF validation is described in three validation reports covering the five Areas of Applicability (AOAs) as follows:

Part I:

- Pu-nitrate aqueous solutions
- MOX pellets, fuel rods, and fuel assemblies

Part II:

- PuO_2 powders
- MOX powders

Part III:

- Aqueous solutions of Pu compounds (e.g., Pu-oxalate solutions).

Using critical benchmark experiments similar to the design conditions found in each of the AOAs, bias and uncertainty in the bias are determined. Additionally, an administrative margin of 0.05 was proposed in the validation reports. The validation reports were accepted by the NRC as part of the CAR review, as documented in NUREG-1821. The NRC agreed that using the administrative margin of 0.05 along with the established bias and uncertainty in the bias as described in the reports would provide an acceptable margin of subcriticality for safety, for both normal and credible abnormal conditions at the MFFF. Thus, that information is used along with calculated MFFF criticality application results to demonstrate that the MFFF units are subcritical.

NCSEs are performed to ensure that the entire process will be subcritical under both normal and credible abnormal conditions. NCSEs are documented with sufficient detail, clarity, and lack of ambiguity to allow independent evaluation and judgment of results. NCSEs identify the controlled nuclear and process parameters and their associated limits upon which criticality safety depends.

Thus, NCSEs form the basis for criticality safety for operations in which fissionable material is handled. That is, each NCSE evaluated a respective operation for credible accident sequences identified by the PrHA and identified sufficient controls such that double contingency protection is provided in those cases in which a criticality is credible. Utilizing the results of validated calculational methodologies, the NCSEs demonstrate that both normal and accident conditions meet the required minimum margin of subcriticality. Finally, the IROFS to provide double contingency protection, along with criticality accident sequences, are identified in NCSEs. Features that are required to ensure that the criticality controls identified in the NCSE are sufficiently available and reliable will be provided through the implementation of appropriate management measures.

Each potential credible criticality event sequence is shown to be highly unlikely by the application of well defined, qualitative criteria. In particular, to demonstrate that criticality events are highly unlikely, the NCSEs contain the following information:

- For each event for which a potential criticality is credible, the event is described and analyzed to demonstrate adherence to the double contingency principle.
- For each IROFS control identified, the IROFS is shown to be effective and perform the intended function.
- For each event for which a potential criticality is credible, the event is shown to be highly unlikely as follows:

- Summary description of each of the IROFS controls with cross reference to the IROFS information
- Description and justification of the failure of each of the IROFS being unlikely
- Description of failure detection or safety margin involved providing justification that the potential event is highly unlikely to occur.

For passive features, such as tanks, vessels, and storage areas whose failure is not credible, a potential criticality event is not credible. For this to be true, the following is shown:

- The passively controlled component is specified as an IROFS.
- The passively controlled equipment is evaluated and shown to be subcritical under all credible process conditions.
- The passively controlled equipment has management measures to ensure that the configuration is controlled and unchanging under the facility's configuration management program (see Chapter 15).

For other units for which potential events are credible, the criteria for judging events highly unlikely are as follows:

- At least two independent robust (that is, unlikely to fail) controls are provided.
- Active or passive engineered controls are unlikely to fail. This determination is based on consideration of all applicable "available and reliable" qualities per NUREG-1718; also the controls are identified as IROFS.
- Administrative controls are robust and unlikely to fail. This determination is based on consideration of all applicable "available and reliable" qualities per NUREG-1718; also administrative controls are simple and unambiguous.

For each independent and unlikely to fail control relied on for compliance with the double contingency principle, one of the following additional measures are utilized to ensure that the associated event sequences are highly unlikely to occur:

- A means to detect a failure of the control on a period (for example, of one month or less) is provided, as justified in the NCSEs, or
- A safety margin is shown that demonstrates that multiple (three or more) failures of each independent control (i.e., IROFS) does not result in a loss of subcriticality, or
- Other measure(s), with justification.

The rationale for demonstrating an event is highly unlikely is provided in the NCSEs.

An approved design configuration requires criticality safety design input.

Criticality safety during design and operation is ensured for the MFFF through design and administrative practices. MFFF design and safety features are documented and controlled through the implementation of a rigorous configuration management program. Criticality safety calculations and NCSEs are maintained up-to-date and consistent with existing facility process and design features and administrative practices. Changes to these documents are controlled in accordance with the design change control and configuration management programs (see Chapters 6 and 15).

5.3 ISA TEAM

Process Hazards Analyses (What-If/Checklist, HAZOP) are performed by a team of reviewers referred to as the ISA team. The ISA team consists of four basic participants: 1) team leader, 2) team scribe, 3) process or responsible engineer, and 4) discipline experts. The team leader provides direction for the team to ensure a thorough evaluation. The team scribe documents the discussions of the team during the evaluation. The responsible engineer provides detailed knowledge of process unit equipment and operations. Discipline experts provide input concerning the various design disciplines involved in the process and may include:

- Radiochemical process
- Chemical processes (i.e., aqueous polishing)
- Civil/structural/geotechnical
- HVAC
- Glovebox design
- Nuclear criticality safety
- Electrical
- Fire protection
- Instrumentation and control
- Mechanical
- MOX fuel process
- Operations
- Radiation protection
- Human Factors Engineering

Discipline experts are selected based on the process and associated hazards. Discipline experts may attend portions of the hazard evaluations or be placed on call based on the discretion of the team leader.

ISA team member responsibilities and qualification are listed below:

<u>ISA Team Leader:</u> The team leader is responsible for providing direction for the performance of the ISA hazard evaluation and ensuring the evaluation is conducted in an efficient and thorough manner. The team leader ensures that all materials and resources (i.e., drawings, support

analyses, design descriptions, meeting rooms, etc) required to perform the hazard evaluation are available. The ISA team leader is responsible for selecting the appropriate discipline experts for the process being evaluated. The ISA Team leader shall have a good working knowledge of the process being evaluated. The ISA team leader shall be knowledgeable and experienced in the method chosen for performance of the ISA hazard evaluation. This requirement may be satisfied by formal training in the specific method or one (1) year experience performing the specific method. The ISA team leader shall not be the responsible engineer for the process being evaluated.

<u>Team Scribe:</u> The team scribe is responsible for documenting the discussions that take place during performance of the hazard evaluation. Documentation is performed in a format dictated by the hazard evaluation method. The team scribe shall be familiar with the method chosen for performing the ISA hazard evaluation and the process unit being evaluated. The team scribe performs his duties under the direction of the team leader.

<u>Responsible Engineer:</u> The process expert is an experienced team member with detailed knowledge of the process unit being evaluated. The process expert provides information concerning process unit design, as well as the associated technology and theory of operation.

<u>Discipline Expert:</u> The discipline expert is an experienced team member with knowledge of a specific design discipline. The discipline expert provides the team with information used to identify and evaluate events, as well as determine applicable mitigative/preventive controls.

5.4 DESIGN BASIS

The following information represents the design basis attributes for the safety program and integrated safety analysis.

- A risk matrix, as shown in Table 5.1-2, is used to determine the requirements for IROFS.

- Deficiencies in IROFS or failure of management measures are addressed in accordance with the corrective action program described in the MOX Project Quality Assurance Plan (MPQAP).

- The hazard identification checklist is developed and used in accordance with the Checklist Analysis and What-If/Checklist methods of *Guidelines for Hazard Evaluation Procedures – Second Edition – With Worked Examples*, Center for Chemical Process Safety, American Institute of Chemical Engineers, New York, NY, 1992.

- PrHAs are performed in accordance with the guidance provided in *Guidelines for Hazard Evaluation Procedures – Second Edition – With Worked Examples*, Center for Chemical Process Safety, American Institute of Chemical Engineers, New York, NY, 1992 and *Integrated Safety Analysis Guidance Document,* NUREG-1513, U.S. Nuclear Regulatory Commission, 1999.

- The methodology for assessing radiological consequences for events releasing radioactive materials is based on guidance provided in NUREG/CR-6410, *Nuclear Fuel*

Cycle Facility Accident Analysis Handbook, U.S. Nuclear Regulatory Commission, March 1998.

- Radiological consequences to the facility worker are qualitatively determined.

- In determining radiological consequences, the site worker is considered to be a fixed distance of 100 m from the release point. Both facility workers and site workers are deemed to be "workers." The IOC is defined as the maximally exposed individual outside the controlled area boundary. The controlled area is a minimum distance of 68 m from the BSW, and 160 m from the MFFF building stack. Radiological consequences to the environment are assessed a distance of 28 m from the BSW, and 52 m from the MFFF building stack.

- Radiological releases for the site worker and the IOC are conservatively modeled using a 0- to 2-hour 95th percentile dispersion χ/Q. Atmospheric dispersion factors (χ/Q) for the site worker and IOC are established from SRS data using the ARCON96 computer code.

- Radiological releases to the environment are conservatively modeled using an averaged 24-hour dispersion χ/Q. The 24-hour average atmospheric dispersion factor (χ/Q) for ground-level releases at the restricted area boundary is calculated by ARCON96 for releases from the BSW and the MFFF building stack.

- The source term is determined based on the five-factor formula as described in NUREG/CR-6410.

- MAR values are listed in Table 8.3-1.

- Applicable ARF and RF values are established for the material forms and the release mechanisms that could potentially occur at the MFFF from values presented in NUREG/CR-6410 and DOE-HDBK-3010, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*, U.S. Department of Energy.

- The leak path factor (LPF) is the fraction of material leaving a defined confinement barrier. The LPF in all unmitigated cases is conservatively assumed to be one (that is, no credit is taken for removal of hazardous material). Conservative LPFs utilized in the calculation of quantitative mitigated consequence will be determined in accordance with NUREG/CR-6410.

- The breathing rate (BR) is based on the guidance of Regulatory Guide 1.25, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)*, U.S. Nuclear Regulatory Commission, March 1972.

- Activity-based inhalation DCF values are taken from Federal Guidance Report No. 11 based on the form of the potential releases from the MFFF when received by the dose receptor.

- Chemical consequences to the facility worker are qualitatively determined based on the material released, the release mechanism, and the location of the worker relative to the release.

- In lieu of a mechanistic calculation of the release, a conservative bounding release model is used to determine the consequences to the site worker and IOC from releases either from the BSW or the MFFF building stack, as applicable. Releases are modeled to occur using the total material at risk from the largest single tank or container.

- Estimates of hazardous chemical concentrations include techniques, assumptions, and models that are consistent with industry practice, are verified and/or validated, and follow the guidance on atmospheric and consequence modeling found in NUREG/CR-6410, *Nuclear Fuel Cycle Accident Analysis Handbook*.

- The analysis to determine the effects to the IOC is based on the following assumptions:

  - A ground level release
  - No mechanical or buoyancy plume rise
  - Neutrally buoyant gas model.

- For calculating airborne concentrations involving evaporative releases, the more conservative release rate from two separate evaporation models is used as input to the ARCON96 (Atmospheric Relative Concentrations in Building Wakes) computer code. The sources of these alternate evaporation models are: (1) a Journal of Hazardous Materials article (Kawamura, P. I., and D. Mackay, The evaporation of volatile liquids. J. Hazardous Materials 15:343-364, 1987), and (2) NUREG/CR-6410 (*Nuclear Fuel Cycle Facility Accident Analysis Handbook*, March 1998).

- Temporary Emergency Exposure Limits (TEELs) are listed in Table 5.1-3 and are derived from approved methodologies developed by Department of Energy Subcommittee on Consequence Assessment & Protective Actions (SCAPA) and are identified in WSMS-SAE-02-0001, Revision 18.

- Controls are applied to events if the potential intake of soluble uranium or respirable intake of insoluble uranium exceeds 10 mg to the IOC or 30 mg to a worker.

- The following qualitative definitions are used in assessing event sequence likelihood:

  - Not Unlikely – Events that may occur during the lifetime of the facility
  - Unlikely – Events that are not expected to occur during the lifetime of the facility or events originally classified as Not Unlikely to which sufficient IROFS are applied to further reduce their likelihood to an acceptable level
  - Highly Unlikely – Events originally classified as Not Unlikely or Unlikely to which sufficient IROFS are applied to further reduce their likelihood to an acceptable level

- Credible – Events that do not meet the definition of "Not Credible"
- Not Credible –
  - ✓ Natural phenomena or external man-made events with an extremely low initiating event frequency, conservatively estimated as less than once in a million years, or
  - ✓ A process upset or sequence of independent process deviations and/or human actions or errors, for which a convincing argument exists that it is unquestionably extremely unlikely, and no such sequence of events can ever have actually happened in any fuel cycle facility.

- To ensure that all event sequences with consequences exceeding the low consequence threshold of 10 CFR §70.61 meet the performance requirements identified in 10 CFR §70.61, the following qualitative design criteria and commitments are applied to those events and the associated IROFS:
  - Application of the single failure criteria or double contingency (for nuclear criticality)
  - Application of 10 CFR 50 Appendix B and NQA-1
  - Application of Industry Codes and Standards
  - Management Measures, including surveillance of IROFS (i.e., failure detection and repair, or process shutdown capability).

- In cases where a single active system, component or activity of personnel is the sole IROFS preventing or mitigating an accident sequence that exceeds the performance requirements of 10 CFR § 70.61, justification is provided to demonstrate that it is designed to perform its safety function.

- Passive structures and components (such as buildings or tanks) are not designated as sole IROFS in the event that their design, or the design of any associated IROFS passive structure/component, precludes their failure under all credible natural phenomena and process conditions.

- All IROFS are assigned the highest level of quality, QL-1/QL-1LR, which ensures a comprehensive application of quality assurance requirements covering all phases of the project including design, document, and configuration control, records management, procurement, materials control, installation, use of measurement and test equipment, and computer software and hardware.

- Within the MPQAP, quality assurance grading can also be used to identify the controls applied to IROFS and activities that support the MPQAP based upon an evaluation of the complexity and importance of the activity compared to quality, safety, risk, and the environment.

- IROFS operability is defined in the MFFF Operating Limits Manual (OLM).

- The determination of setpoints for IROFS is performed in accordance with the provisions of ISA standard 67.04.01-2006 and Regulatory Guide 1.105.

- The integration of the necessary analyses and demonstration that the performance requirements of 10 CFR §70.61 are satisfied is performed in NSEs and NCSEs.

- Associated components, including support systems, required to perform the assigned safety function are identified as IROFS.

- Using critical benchmark experiments similar to the design conditions found in each of the AOAs, bias and uncertainty in the bias are determined. Additionally, an administrative margin of 0.05 was proposed in the validation reports. The validation reports were accepted by the NRC as part of the CAR review, as documented in NUREG-1821. The NRC agreed that using the administrative margin of 0.05 along with the established bias and uncertainty in the bias as described in the reports would provide an acceptable margin of subcriticality for safety, for both normal and credible abnormal conditions at the MFFF.

- The ISA team leader shall be knowledgeable and experienced in the method chosen for performance of the ISA hazard evaluation. This requirement may be satisfied by formal training in the specific method or one (1) year experience performing the specific method. The ISA team leader shall not be the responsible engineer for the process being evaluated.

6.0 NUCLEAR CRITICALITY SAFETY

As described in this chapter, nuclear criticality safety (NCS) practices for the Mixed Oxide (MOX) Fuel Fabrication Facility (MFFF) are in accordance with U.S. Nuclear Regulatory Commission (NRC) regulations. The regulations for NCS are found in Title 10 of the Code of Federal Regulations (CFR) Part 70. In addition, MFFF practices for NCS draw, as needed, from guidance contained in Regulatory Guide 3.71, *Nuclear Criticality Safety Standards for Fuels and Materials Facilities,* Revision 1, October 2005 including the exceptions noted to American National Standards Institute (ANSI) and American Nuclear Society (ANS) ANSI/ANS 8 national standards.

6.1 ORGANIZATION AND ADMINISTRATION FOR NCS

The MFFF NCS program fosters ownership of nuclear criticality safety by the MFFF organization. The NCS program requires personnel to report defective NCS conditions to the manager of the regulatory function, directly or through a designated supervisor, and requires that the MFFF staff or management take no further action not specified by approved written procedure, until the NCS function has analyzed the situation.

The NCS organization, which reports to the manager Environmental Safety and Health Licensing function, is responsible for implementing applicable NCS practices for the MFFF. The NCS organization is independent of operations to the extent practical.

The NCS organization is responsible for implementing NCS practices of ANSI/ANS-8.1-1998, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*. The MFFF also implements the administrative practices for nuclear critical safety, as described in ANSI/ANS-8.19-2005, *Administrative Practices for Nuclear Criticality Safety*. The manager of the regulatory function and other key management functions are described in Chapter 4.

The NCS organization is administratively independent of production responsibilities, and has the authority and responsibility to shut down potentially unsafe MFFF operations. Specific responsibilities of the NCS organization are to:

- Establish the NCS program, including design criteria, procedures, and training
- Provide NCS support for integrated safety analyses and configuration control
- Assess normal and credible abnormal conditions
- Determine criticality safety limits for controlled parameters
- Develop and validate methods to support nuclear criticality safety evaluations (NCSEs)
- Perform criticality safety calculations and prepare NCSEs
- Review and approve proposed changes in process conditions or equipment involving fissionable material as part of the MFFF configuration management and design change process to determine whether the facility changes require prior NRC approval in accordance with the criteria of 10 CFR §70.72, *Facility Change Process*
- Specify NCS control requirements and functionality

- Review and approve MFFF operations and operating procedures that involve fissionable material
- Support emergency response planning and events
- Assess the effectiveness of the NCS program through the audit/assessment program
- Identify NCS posting requirements that provide administrative controls for operators in applicable work areas
- Maintain NCS programs for the MFFF in accordance with applicable regulatory guides and industry standards
- Be the single point of contact for nuclear criticality issues with internal and external groups or agencies, coordinating with and taking direction from the manager of the regulatory function.

The NCS organization is also responsible for the NCS function for analysis and corrective action. The nuclear criticality process requires that upon identification of a defective NCS condition, the MFFF organization take no further action not specified by approved written procedures, until the NCS function has analyzed the situation. The NCS organization shall be staffed by qualified engineers or technical staff with experience at nuclear facilities involving special nuclear material (SNM).

See Chapter 4 for discussion of minimum qualification requirements for the NCS organization.

6.2 MANAGEMENT MEASURES FOR NCS

The management practices for MFFF NCS are based on ANSI/ANS-8.1-1998, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*, which provides guidance on administration, technical practices, validation of calculational methods, and on various acceptable limits for fissile nuclides. MFFF NCS management practices are implemented in CB&I AREVA MOX Services, LLC (MOX Services) procedures, and provide reasonable assurance that NCS-related items relied on for safety (IROFS) are available and reliable to perform their designated safety functions when needed. Chapter 15 describes the MFFF management measures implemented to supplement IROFS, including training, audits and assessments, and procedures.

6.2.1 Nuclear Criticality Safety Training

The NCS practices and associated procedures comply with regulatory requirements and subscribe to ANSI/ANS industry standards. MOX Services implements the training requirements of ANSI/ANS-8.19-2005, *Administrative Practices for Nuclear Criticality Safety,* and ANSI/ANS-8.20-1991 (R1999), *Nuclear Criticality Safety Training*. The training is appropriately tailored to the staff's function within the MFFF.

In addition, the MFFF NCS staff develops:

1. NCS training that includes facility, materials, operations, methodologies, design solutions, work stations, and storage locations that provide operators with knowledge and rules to ensure MFFF maintains the nuclear safety margin
2. Instructions regarding the use of process variables for NCS control, when controls on such parameters are credited for nuclear criticality safety (e.g., IROFS)
3. Training that includes the policy to identify NCS posting requirements for administrative controls that provide operators with reference for ensuring conformance and safe operation
4. Training associated with the operation of plutonium containing systems to prevent criticality events.

NCS training is based on ANSI/ANS-8.20-1991 (R1999), *Nuclear Criticality Safety Training* and is appropriately tailored to the staff's function with the MFFF. NCS training is developed by the NCS organization and implemented in conjunction with the MFFF training function. The instructors of NCS-related material are selected by the manager of the NCS function, in cooperation and coordination with the MFFF training function. The manager of the NCS function ensures that the NCS training is current and adequate and contains the required skills and knowledge, by periodically reviewing training content. Records of currently trained MFFF employees are retained in accordance Chapter 15 requirements. Visitors are trained commensurate with the scope of their visit and/or are escorted by MOX Services employees who are fully trained for the scope of the visit, including the criticality safety requirements for the area(s) to be accessed.

6.2.2 Audits and Assessments

MOX Services utilizes distinct levels of activities to evaluate the effectiveness of the NCS program and other management measures to ensure that operations conform to criticality safety requirements and controls in accordance with ANSI/ANS-8.19-2005, *Administrative Practices for Nuclear Criticality Safety*. Internal or external audits, which are independently planned and documented evaluations, are performed by the quality assurance (QA) organization. Assessments are management directed evaluations, within their area of responsibility, to assess the adequacy, programmatic compliance, and implementation effectiveness of the NCS program and other management measures. The manager of the NCS function, or designee, is lead for NCS assessments, surveillances, and walk-downs. QA audits are consistent with MOX Project QA Plan (MPQAP) requirements. Representatives of the NCS function conduct scheduled assessments, surveillances, and/or walk-downs of applicable MFFF manufacturing and support areas in accordance with approved written procedures.

Quality-affecting activities of the NCS program are evaluated annually by either periodic audits or assessments. As a minimum, regularly scheduled internal audits of the NCS functional area quality-affecting activities shall be performed at least once every two years. The frequency for audits of operational phase IROFS related activities will be based on the risk-informed methodology determination which will consider the safety significance of the activity, results of the Integrated Safety Analysis (ISA) and/or performance history so that each area is evaluated annually (Assessment or Audit) and audited at least once every two years. Personnel performing audits shall be independent of the direct responsibility for performing the work being audited.

Written notification of a planned audit shall be provided to the functional organization at a reasonable time before the audit is to be performed.

Audit results are communicated in writing to the cognizant management of the audited function/organization. Internal management assessment results identifying findings and recommendations are communicated in writing to the cognizant management having responsibility for the area/activity evaluated and to the manager of the NCS function. Responsible management of the audited function/organization shall complete corrective action(s) including remedial action(s) and action(s) to prevent recurrence and document completion of the action(s) in a timely manner. An extent of condition will also be evaluated where appropriate for findings affecting the NCS function.

6.2.3 NCS Surveillance and Walk-downs

Periodic walkthroughs of all areas or activities involving fissile material operations are conducted and documented weekly. The frequency for walkthroughs, if less than weekly, will be based on the risk-informed methodology determination which will consider the safety significance of the activity, results of the Integrated Safety Analysis (ISA), and/or performance history. The manager of the NCS function may utilize a risk-informed methodology determination based upon the compliance results of these evaluations, to increase or decrease the scheduled frequency of these reviews or the scope of the evaluations. The evaluations are documented (e.g., by a checklist). Identified weaknesses are incorporated into the MFFF Corrective Action Program, and are promptly and effectively resolved.

6.2.4 NCS Procedures

Procedures are established and implemented for nuclear criticality safety in accordance with ANSI/ANS-8.19-2005, *Administrative Practices for Nuclear Criticality Safety*. NCS posting requirements at the MFFF are established that identify administrative controls applicable and appropriate to the activity or area. NCS procedures and postings are controlled to ensure that they are maintained current. Procedures and their implementation are reviewed periodically, but at least once every two years, to ascertain that procedures are being followed and that process conditions have not been altered to adversely affect NCS requirements and/or controls. The frequency for procedure reviews, if less than annually, will be based on the risk-informed methodology determination which will consider the safety significance of the activity, results of the Integrated Safety Analysis (ISA), and/or performance history. The reviews are conducted, in consultation with operating personnel, by MFFF staff that is knowledgeable in the nuclear criticality safety.

6.2.5 Change Management

The NCS functional organization shall review proposed changes to structures, systems and components (SSCs), hardware, software, processes and procedures to ensure that proposed facility changes are managed to maintain the integrity of the facility's safety basis and to ensure that proposed changes receive the appropriate level of NCS review. The NCS review assures that the ability of the NCS credited SSCs and/or IROFS to perform their function when needed is maintained. The NCS functional organization reviews and approves proposed changes in process conditions or equipment involving fissionable material as part of the MFFF configuration management and design change process to determine whether the facility changes require prior NRC approval in accordance with the criteria of 10 CFR §70.72, *Facility Change Process*.

6.3 NUCLEAR INCIDENT MONITORING SYSTEM

The purpose of the nuclear incident monitoring (NIM) system is to reduce risk to personnel by providing prompt warning and notification should a nuclear criticality event occur. The design and operation of the NIM system also takes into consideration the avoidance of false alarms. Alarm actuation setpoint(s) are specified with consideration of normal operating background radiation levels such that spurious actuations from sources other than criticality do not occur. The NIM system is designed in accordance with 10 CFR 70 and ANSI/ANS-8.3-1997 (R2003).

In the highly unlikely event of a nuclear criticality, the NIM system is intended to:

- Monitor for excessive radiation
- Monitor appropriate areas
- Warn personnel as quickly as possible.

The NIM system, which utilizes both fixed and portable (for maintenance only) monitoring units, is designed in accordance with generally accepted practices in R.G. 3.71, Rev 1 October 2005 and those required by 10 CFR §70.24. ANSI/ANS-8.3-1997 (R2003), *Criticality Accident Alarm System*, is the guidance document that defines the design criteria and functional operation requirements of the NIM system (or criticality accident alarm system). These features assure detection capability and prompt notification by clear audible alarm, visual light, or other notification means to warn personnel of a criticality condition. Criticality monitoring is performed by groups of detectors called “monitoring units.” Each NIM system monitoring unit contains multiple detectors that provide a redundant detector actuation logic thus minimizing false alarms. The design covers potentially affected areas with 3 detectors with a 2 out of 3 logic for alarm. The data from the NIM system monitoring units is sent real time to the emergency control consoles. Clearly audible alarms, visual lights, or other notification means are provided for areas that require evacuation.

If the NIM system, detection or alarm/notification capability, becomes unavailable, the allowable number of hours during which NIM system coverage is not available is determined on a process-by-process basis. The MFFF will maintain safe operations by immediately implementing compensatory measures (e.g., limit personnel access, halt SNM movement or activities) as necessary when the NIM system is unavailable or significantly degraded as approved by the nuclear criticality safety function.

The evaluation of the effectiveness of NIM system detectors (detection criteria and location/spacing) takes into account the effect of existing shielding. NIM system detector coverage is determined through the use of three dimensional radiation transport codes.

6.3.1 NIM System Principles of Operation

The NIM system is designed to detect radiation in the highly unlikely occurrence of a criticality event. The nuclear criticality audible alarm, visual light, or other notification means are clearly provided in accessible locations of the facility. Indication that a NIM system alarm condition has occurred is also sent to an emergency control console in the control room and/or a remote facility. The criticality alarm is designed to accommodate the working environment within the MFFF.

6.3.2 NIM System Design

NIM system design features:

- Prevent spurious alarms through the use of redundant detectors and alarm actuation setpoint determination
- Produce event records that will be monitored and recorded.

The design criteria for the NIM system are:

- **Reliability** – NIM system components do not require frequent servicing. The system is designed to reduce the effects of non-use, deterioration, power surges, and other adverse conditions. The design ensures reliable actuation of an alarm, while avoiding false alarms.
- **Seismic tolerance** – The NIM system is designed to remain operational in the event of a seismic shock equivalent to the MFFF design basis earthquake.
- **System vulnerability** – NIM system components are protected in order to reduce the potential for damage in case of fire, explosion, corrosive atmosphere, or other probable extreme conditions. The system is designed to reduce the potential of failure, including false alarms.
- **Failure warning** – The NIM system provides a visual or audible warning signal to indicate system malfunction or the loss of primary power.
- **Response time** – The NIM system produces a criticality alarm signal within one-half second of detector recognition of a criticality event.
- **Detection** – The NIM system is designed to detect the minimum event of concern. In areas where fissionable material is handled, used, or stored, the minimum event of concern is analytically determined based on the process, materials, geometry, and process equipment present in each covered area. The minimum event of concern delivers the equivalent of an absorbed dose in soft tissue of 20 rads of combined neutron and gamma radiation at an unshielded distance of 6.6 feet (2 meters), within one minute.
- **Coverage** – NIM system detector coverage is designed to detect the smallest criticality event as defined above. The location and spacing of detectors are chosen to account for the effect of shielding walls.

- **Electrical power –** The NIMS components will obtain facility power at 117 ± 15 volts AC or 102 to 132 volts AC, and a frequency range of 57 to 61 Hz from the essential power system
- **Alarm Response –** The system has been designed to produce the local criticality alarm within one-half (1/2) second of detector recognition of the minimum criticality accident of concern to enable local evacuation of the effected area. The alarm system covers all areas that may result in an absorbed dose of 12 rads or greater associated with the largest criticality event consequences.
- **Staff emergency response** – The nuclear criticality accident onsite emergency planning and response for the MFFF staff follows the guidance in ANSI/ANS-8.23-1997, *Nuclear Criticality Accident Emergency Planning and Response*. (As described in Chapter 14, an emergency plan is not required to be submitted.)
- **Emergency procedure** – The MFFF staff maintains an emergency procedure, which covers the entire facility including locations where licensed SNM is handled, used, or stored, to ensure that personnel can be withdrawn to a safe area upon the actuation of the NIM system alarm notification.

6.4 NCS TECHNICAL PRACTICES

6.4.1 Nuclear Criticality Safety Evaluations

When an MFFF component or system containing fissile materials is designed or modified that could potentially affect credible criticality sequences, an NCSE is developed or updated to determine that the entire process will be subcritical under both normal and credible abnormal conditions.

NCSEs are documented with sufficient detail and clarity to allow independent review and approval of results, and to explicitly identify the controlled nuclear and process parameters, and the associated limits on which nuclear criticality safety depends. NCSEs are only performed by qualified NCS Engineers or qualified Senior NCS Engineers. Prior to approval, NCSEs will be peer reviewed by a qualified Senior NCS Engineer or NCS Manager. The approval of NCSEs is performed in accordance with MFFF project procedures.

An evaluation is performed to determine credible event sequences and identify controls such that double contingency protection is provided. The evaluation may include criticality calculations using validated calculational methodologies to demonstrate that both normal and credible abnormal conditions are subcritical, including the required minimum margin of subcriticality. Permissible limits are used which are the minimum and maximum values of a parameter used in a criticality control evaluation where the k_{eff} is below the upper safety limit (USL). These values are the basic data for establishing criticality safety control limits used in the NCSEs. IROFS are identified in the NCSE. The environmental conditions required for qualification of IROFS are specified in the NCSEs and NSEs. Features that ensure that the criticality controls identified in the NCSE are sufficiently available and reliable are provided through implementation of management measures such as: procedures, training, maintenance procedures, and surveillance. The NCSE provides documentation that demonstrates that potential credible events are highly unlikely to cause a criticality.

6.4.2 Analytical Methodology

The double contingency principle specified in 10 CFR §70.64(a)(9) and ANSI/ANS-8.1-1998, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors* requires that the process incorporates sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality event can occur. NCSEs of the design of the MFFF demonstrate compliance with the double contingency principle and the adequacy of criticality controls. The NCSEs, which are part of the integrated safety analysis (ISA), identify the assumptions used in the criticality evaluations. The evaluations of the assumptions are based on realistic process conditions; conservative assumptions are analytically justified so as to demonstrate the level of conservatism added. The ISA also documents a comprehensive systematic review of MFFF hazards in Process Hazards Analysis (PrHAs), including criticality, and provides additional confirmation of the acceptability of the selected means of criticality control.

Compliance with the double contingency principle is demonstrated by identifying two or more controls on which reliance is placed to ensure criticality safety. Controls to prevent criticality are identified according to a preferential selection. Preferential selection manifests itself as first passive engineered controls, secondly active engineered controls, and then administrative controls, where practical. Common mode failures and the potential interaction between units containing fissionable material are appropriately taken into account. In addition to providing a basis for identifying IROFS, the hazard identification and review processes documented in the ISA are used to promote defense-in-depth practices in MFFF design and layout. Defense-in-depth practices are incorporated in the MFFF.

Acceptance criteria applied in performing double contingency and criticality hazard assessments are summarized as follows:

- When applying a single control to maintain limits on two or more controlled parameters, credit is taken for a single component only, for double contingency compliance.
- No single credible event or failure will result in a criticality.
- Geometry control constitutes the preferred controlled parameter, with fixed spacing or fixed neutron absorbers employed as necessary.
- Where practical, reliance is placed on equipment design that uses passive engineered controls, rather than on administrative controls.
- Controlled parameters are identified in the NCSE evaluations. IROFS associated with maintaining these controlled parameters are noted in the NCSE. All controls identified to prevent criticality are designated as IROFS. The criticality safety controlled parameters are transferred into appropriate operating and maintenance procedures.
- Calculations are performed to demonstrate that controlled parameters are maintained during both normal and credible abnormal conditions. For example, using IROFS, the controlled parameters are maintained in spite of abnormal conditions that may occur as a result of (non-safety system) control failures. Summaries of these calculations are provided in the NCSEs. Demonstrated in the NCSEs, it is highly unlikely that controlled

parameters exceed the safety limit. In cases where controlled parameters are controlled by measurement, reliable methods that ensure representative sampling and analysis are used.

- Optimum or worst-credible conditions are assumed for parameters unless they are specifically controlled.

6.4.3 Additional Technical Practices

A design application (system) for an MFFF unit is considered subcritical when the calculated multiplication factor for the design application (system) (ANSI/ANS-8.17 Section 5 [2004], *Criticality Safety Criteria for the Handling, Storage, and Transportation of Light Water Reactor (LWR) Fuel Outside Reactors*) is shown to be less than or equal to an established maximum allowed value that properly accounts for method bias, uncertainty, and administrative margin. An administrative margin of 0.05 is used for MFFF design applications. See Section 6.4.5 for discussion of the USL for each MFFF area of applicability (AOA).

6.4.4 Criticality Controls

Criticality controls are the methods of criticality safety control selected for various MFFF process stations and areas. Reliance is placed on equipment design using passive engineered controls, rather than administrative controls, where practical. Techniques for criticality control, listed in order of preference, are:

- **Passive Engineered Controls** – Controls that employ permanent and static design features or devices to preclude inadvertent criticality. No human intervention is required, except for maintenance and inspection.
- **Active Engineered Controls** – Controls that use active hardware to sense conditions and automatically place a system in a safe state or mode. Actuation and operation of these controls do not require human intervention.
- **Enhanced Administrative Controls** – Controls that rely on human judgment, training, and actions for implementation, and employ active warning devices (audible or visual) that prompt specific human actions to occur before the process can exceed established limits.
- **Simple Administrative Controls –** Controls that rely solely on human judgment, training, and actions for implementation.

The MFFF uses controls of hierarchical preference, to the extent practical, to provide correspondingly higher reliability when assessing criticality risks and demonstrating compliance with the double contingency principle. "To the extent practical" means that the hierarchy is followed wherever practicable as determined by the process. To ensure criticality control in activities involving significant quantities of fissionable materials, one or several of the following available controls are used:

- Geometry Control
- Mass Control

- Density control
- Isotopic control
- Reflection control
- Moderation control
- Concentration control
- Interaction control
- Neutron absorber control
- Volume control
- Heterogeneity control
- Physicochemical control
- Process variable control.

Geometry control constitutes the preferred control, with fixed neutron absorbers employed as necessary. Although geometry control is preferred, several methods of criticality control are employed in the aqueous polishing (AP) and MOX processing (MP) designs.

Controlled parameters and techniques for associated criticality controls that minimize the risk of inadvertent criticality are established and justified in the NCSEs. Tolerances on controlled parameters are conservatively taken into account in establishing operating limits and controls. The potential for neutron interaction between units is evaluated to ensure that the process remains subcritical under normal and credible accident conditions. Additional controls on spacing are identified as IROFS as necessary. Sensitivity studies are performed in calculations to demonstrate that the reactivity of units employing criticality controls are subcritical under all credible conditions. MFFF management measures described in Chapter 15 are generally required to ensure double contingency compliance. Where sampling of criticality controlled parameters is required, the sampling plan in accordance with requisite supporting analyses for the sampling plan ensures that the samples are appropriately representative.

6.4.4.1 Geometry Control

Geometry control involves the use of passive engineered devices to control worst-case geometry within ensured tolerances. Geometry limits are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded) using documented and approved methods. Geometry control is used in MFFF design wherever possible, including the following design applications:

- For storage systems containing large quantities of fissile material (for which mass or mass and moderation control is not applicable)
- For process equipment whenever the imposed geometry is compatible with the applicable process function.

When the possibility of neutron interaction with other fissile units exists, interaction control or neutron absorber control may also be indicated, in conjunction with geometry control.

Geometry control parameter limits are established and implemented as follows:

- Dimensions and nuclear properties of MFFF features relying on geometry control are subject to QA measures during design and fabrication, and are verified prior to beginning operations. The MFFF configuration management program (see Chapter 15) is used to maintain these dimensions and nuclear properties.
- Credible means of transferring fissile materials to an unfavorable geometry are identified and evaluated, and controls (i.e., IROFS) are established to ensure that such transfers are precluded. In particular, leaks from favorable-geometry process vessels are collected in favorable-geometry drip trays.
- Tolerances on nominal design dimensions are treated conservatively.
- Possible mechanisms for changes to fixed geometry are evaluated, and controls are established as necessary. Credible mechanisms that could result in component deformation or changes in geometry are identified and evaluated. Where such credible mechanisms exist, applicable design allowances and/or the surveillance program are specified.

6.4.4.2 Mass Control

Mass control involves the use of mass-based, single-parameter limits established on conservative geometry (i.e., spherical) and SNM form (e.g., metal, oxide, aqueous solution), unless these parameters are controlled by IROFS (i.e., implementation of another criticality control mode(s) in addition to mass control). Single-parameter limits are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded) using documented and approved methods, standards, or handbooks. Mass control is used in MFFF design applications where the process function is not compatible with geometry control. Mass control is generally used in combination with moderation control (i.e., allowable mass with moderation control is higher than without moderation control). The mass is generally controlled through a process variable control (i.e., required process controls include weighing and material mass balance functions). When the possibility of neutron interaction with other fissile units exists, interaction control or neutron absorber control may also be indicated, in conjunction with mass control.

Mass control is available as a control mode where the limitation of mass is compatible with the process function and where mass can be reliably controlled during process operations (e.g., by direct weighing and/or mass balances).

Mass control parameter limits are established and implemented as follows:

- Mass limits are derived for a material that is assumed to have a given weight percent of SNM, based on conservative assumptions. Determinations of mass are based on either (1) weighing the material and assuming the entire mass is SNM, or (2) taking physical measurements to establish the actual weight percent of SNM in the material. When process variables can affect the bounding weight percent of SNM in the mixture, the SSCs or procedures that affect the process variables are controlled as IROFS in the NCSEs and ISA Summary.

- Theoretical densities for fissile mixtures are used, unless lower densities are ensured or theoretical density is not credible.
- Reasonable batch sizes are considered:
  - When overbatching of SNM is possible, the mass of SNM in a single batch is limited so that the mass of the largest overbatch resulting from a single failure is safely subcritical, taking system uncertainties into account. Overbatching beyond double batching is considered when the unit allows additional material to be accepted, to establish the margin of safety.
  - When overbatching of SNM is not possible, the mass of SNM in a batch is limited to be safely subcritical, taking system uncertainties into account.
- Mass limits are established taking tolerances into account. The determination of minimum critical mass is based on spherical geometry, unless actual fixed geometry is controlled.
- Instrumentation used to physically measure mass is subject to QA controls.

Establishing a mass limit involves consideration of potential moderation, reflection, geometry, spacing, and material concentration. The evaluation considers normal operations and expected process upsets for determination of the actual mass limit for the system and for the definition of subsequent controls.

6.4.4.3 Density Control

Density control involves taking credit for controls on SNM density in which non-optimal SNM density characteristics are used in the performance of criticality safety design calculations. SNM density limits are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded) using documented and approved methods. Density control is used in the MFFF design, where the process function is not compatible with a worst-case SNM density assumption (i.e., maximum theoretical density), and is generally used in combination with mass, geometry, and/or moderation control.

Density control parameter limits are established and implemented as follows:

- Conservative assumptions are made about the density of the fissile material.
- Instrumentation used to physically measure density is subject to QA controls.
- When process variables can affect the density, controls to maintain the process variables are identified as IROFS in the related NCSE and ISA Summary.

6.4.4.4 Isotopic Control

Isotopic abundance control involves taking credit for established realistic or conservative assumptions regarding SNM isotopic abundance in the performance of criticality safety design calculations. Isotopic control includes both the $^{235}U/U$ concentration (enrichment) and the concentration of fissile and nonfissile plutonium isotopes (e.g., ^{239}Pu, ^{240}Pu, ^{241}Pu), as well as the relative abundance of plutonium to uranium. The presence of ^{240}Pu (5% to 9%) and ^{242}Pu (<0.02%) offsets the contribution from ^{241}Pu (<1%), such that their presence can be neglected for ^{239}Pu in the range from 90% to 95%, as is expected to be the case for the MFFF. This will be demonstrated in the criticality calculation to be referenced in the NCSEs. Justification will be provided in the NCSEs. SNM fissile and neutron absorption isotope abundance limits are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded) using documented and approved methods.

Isotopic control parameter limits are established and implemented as follows:

- When taking credit for isotopic mixtures (where different isotopic mixtures could coexist), controls are established to segregate clearly labeled SNM of different isotopic mixtures. This is provided by sample analysis and verification activities associated with MFFF and vendor (DOE)-supplied measurements. DOE (PDC-type/ARIES feed) and vendor data are qualified in accordance with an approved QA plan and are audited by the MFFF QA function. MFFF will comply with the double contingency principle for isotopic content of feed material. This will be based on the isotopic information supplied by the vendor (DOE). DOE will use sample destructive analysis such as thermal ionization mass spectrometry (TIMS), nondestructive assay (NDA), and/or other information to ensure that isotopic content is consistent with the isotopic characterizations specified in the safety documentation.
- Instrumentation used to physically measure isotopics is subject to QA controls.

6.4.4.5 Reflection Control

Reflection control involves the control of fissile unit geometry and the presence of neutron-reflecting materials in process areas to increase neutron leakage from a subcritical fissile system and thereby reduce the calculated subcritical multiplication factor for the system. Although reflection control is generally applied as a passive engineered feature (i.e., configuration of concrete walls or the construction of fixed personnel barriers), reflection control generally also requires surveillance procedures to ensure that neutron-reflecting materials are excluded from the process area, or to confirm continued efficacy of personnel barriers. Single-parameter limits for reflection are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded) using documented and approved methods.

Reflection control parameter limits are established and implemented as follows:

- When determining subcritical limits for an individual unit, the wall thickness of the unit and reflecting adjacent materials of the unit are conservatively bounded by the assumed reflection conditions, leaving allowances for transient reflectors as discussed below.
- Sufficient water reflection is conservatively used in evaluations to simulate potential personnel and/or other transient reflectors. At a minimum, reflection conditions equivalent to 1-in (2.5 cm) tight-fitting water jacket are assumed to account for personnel and other transient incidental reflectors not evaluated in the unreflected models.
- In cases where loss of reflection control can lead to criticality, by itself or in conjunction with another single failure, rigid and testable barriers are established and maintained by MFFF management measures (i.e., configuration management and maintenance programs) described in Chapter 15.
- Conservative design and spacing dimensions are required in cases where reflection control is not established.
- Conservative reflection conditions are established when evaluating the criticality safety of arrays. For example, conservative minimum distances from arrays to reflecting materials are established (e.g., concrete or water).

6.4.4.6 Moderation Control

Moderation control involves taking credit for non-optimal SNM moderator content or presence within process equipment or areas, in the performance of criticality safety design calculations. SNM moderator content limits or exclusion controls for areas are established in a manner that ensures a conservative margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded) using documented and approved methods, standards, or handbooks. Moderation control is used in MFFF design applications where the process function is not compatible with a worst-case SNM moderator content (i.e., optimum moderation). Moderation control is generally used in combination with mass or geometry control. Moderation control sometimes requires process variable control or other surveillance activities.

Moderation control is particularly useful in situations where process capacity requirements are not satisfied using mass control alone, and where the level of moderation is easily bounded or controlled (e.g., equipment in the powder handling stations confined within gloveboxes).

Potential sources of moderation that are considered include:

- Residual humidity present in powders
- Organic additives (e.g., lubricant, poreformer) used as part of a process
- Moderating fluids (e.g., water or certain oils), which could potentially enter process stations or storage areas under normal or abnormal conditions
- Presence of polyethylene, particularly in waste handling units.

Certain moderators (e.g., humidity and organic additives) exist during normal operations. Criticality safety calculations employ assumptions or process information to account for moderators normally anticipated being present in processes (see below). Moderation control parameter limits are established and implemented as follows:

- Moderation control is implemented consistent with guidance provided in ANSI/ANS-8.22-1997, Nuclear Criticality Safety Based on Limiting and Controlling Moderators.
- When process variables can affect moderation, the SSCs or procedures that affect those process variables are defined as IROFS in the NCSEs and the ISA Summary.
- Physical structures credited with performing moderator exclusion functions are designed to preclude ingress of moderator.
- When sampling of moderation properties is required, the sampling program is based on compliance with the double contingency principle (i.e. dual independent sampling).
- The sampling process incorporates independent verification as part of the sampling and analysis program.
- Fire protection system design, and fire-fighting procedures and training programs are developed with appropriate restrictions placed on the use of moderating materials as stated in section 7.3.3.1. The effects of credible fire events and the consequences associated with the potential use of moderating material in mitigating such fires are evaluated, as applicable.
- Credible sources of moderation are identified and evaluated for potential intrusion into moderator-controlled process stations or areas, and the ingress of moderator is precluded or controlled.
- The effects of varying levels of credible interstitial moderation are evaluated when considering neutron interaction between physically separated fissile units.
- Instrumentation used to physically measure moderators is subject to QA controls.
- Drains are provided to prevent water accumulation, if that accumulation could lead to unfavorable configurations of fissile material.
- Moderation control is implemented and maintained during transportation and storage by means of welded, triply contained, sealed containers.
- During maintenance there will be an administrative control to govern activities such as the removal of fissile material or otherwise ensure that any moderator used during maintenance is controlled.

6.4.4.7 Concentration Control

Concentration control involves the use of concentration-based single-parameter limits established based on conservative case geometry (i.e., spherical) and SNM fissile composition, unless these parameters are controlled by IROFS (i.e., implementation of another criticality control mode(s) in addition to concentration control). Concentration control is generally applied to process equipment handling solutions with low fissile material concentration. Single-parameter limits for concentration are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded), using documented and approved methods. These limits are based on conservative (full) reflection in addition to conservative (spherical) geometry. Concentration control typically includes process variable control to ensure that concentration limits are not exceeded.

Concentration control parameter limits are established and implemented as follows:

- When process variables can affect the concentration, those process variables are defined and controlled in the NCSEs and ISA Summary.
- Concentrations of SNM in excess of controlled parameter limits are precluded.
- When using a tank containing concentration-controlled solution, access to the tank is controlled so that a single operator cannot defeat the control mechanism.
- When sampling of the concentration is specified, a program based on duel independent sampling and analysis using independent verification sampling methods using two people is implemented.
- Concentration-controlled processes are designed and operated in a manner that ensures that possible precipitating agents are not inadvertently introduced to the process, or that the effects of precipitation are taken into account.
- Instrumentation used to physically measure concentration is subject to QA controls.
- Concentration-controlled processes are designed and operated in a manner that prevents overconcentration in excess of controlled parameter limits. Monitoring controls are implemented to detect and prevent long-term fissile material accumulation.

6.4.4.8 Interaction Control

Interaction control involves the use of spacing to limit neutron interaction between fissile units. Single-parameter limits for interaction are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded) using documented and approved methods. When interaction control is employed using passive engineered features (e.g., fuel assembly storage racks), interaction control is considered equivalent to geometry control in terms of hierarchical preference.

When neutron absorbers are used to limit interaction between fissile units, neutron absorber control is indicated in lieu of interaction control.

Interaction control parameter limits are established and implemented as follows:

- When maintaining physical separation between units, passive engineered features (i.e., spacers or other passive geometrical means) are used to the extent practical. The structural integrity of such engineered features is sufficient for normal and design basis conditions. Passive engineered features used as criticality safety controls are passive structural elements designed to withstand deformation. If needed, passive interaction controls are periodically inspected for deformation.

- When unit spacing is controlled by procedure, it is demonstrated that multiple procedural violations do not by themselves lead to criticality. Visual indicators and/or posting are used where interaction is procedurally controlled.
- When evaluating the criticality safety of units in an array or pairs of arrays, spacing is based on validated calculational methods.

6.4.4.9 Neutron Absorber Control

Neutron absorber control involves the use of supplemental neutron absorber features to limit subcritical multiplication of a single fissile unit (e.g., cadmium coatings and borated concrete), or to limit neutron interaction between multiple (spaced) fissile units. Single-parameter limits for neutron absorber features are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded) using documented and approved methods. When using fixed neutron absorbers, MFFF design and procedural controls are implemented consistent with guidance provided in ANSI/ANS-8.21-1995 (R2001), *Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors*.

6.4.4.10 Volume Control

Volume control involves the use of volume-based single-parameter limits established based upon worst-case geometry (i.e., spherical) and SNM form (e.g., metal, oxide, aqueous solution), unless these parameters are controlled by IROFS (i.e., implementation of another criticality control mode(s) in addition to volume). Single-parameter limits are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded) using documented and approved methods. When volume control is employed using passive engineered features (e.g., use of approved fixed-geometry containers), volume control is considered equivalent to geometry control in terms of hierarchical preference. When the possibility of neutron interaction with other fissile units exists, interaction control or neutron absorber control may be indicated in conjunction with volume control.

Volume control parameter limits are established and implemented as follows:

- When using volume control, geometric devices typically are used to restrict the volume of SNM, which limits the accumulation of SNM.
- Instrumentation used to determine volume is subject to QA controls.

6.4.4.11 Heterogeneity Control

Heterogeneity control involves taking credit for the distribution of fissile material. Heterogeneity control is applied in conjunction with another control mode (e.g., mass control, geometry control). Single-parameter limits for heterogeneity are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded) using documented and approved methods. Heterogeneity control is typically implemented through process variable control as well. Additionally, it may be important to control the lattice pitch (i.e., spacing) in a heterogeneous configuration, such as a fuel rod or for pellet fabrication.

Heterogeneity control parameter limits are established and implemented as follows:

- When process variables can affect heterogeneity, the SSCs or procedures that affect process variables and potential mechanisms affecting homogeneity or nonhomogeneity are controlled as IROFS in the NCSEs and ISA Summary.
- Computer calculations that take heterogeneity into account are appropriately validated.
- Assumptions about the physical scale of heterogeneity are based on the observed physical characteristics of the material and appropriately controlled (size of pellets, rod assemblies, etc.) and are conservatively bound. The reactivity in modeled conditions is conservatively bound as suggested by the physical data.

6.4.4.12 Physicochemical Control

Control of physicochemical characteristics is applied to several MFFF process units where non-optimal solution chemistry or specific values for some parameters (e.g., pellet diameter) are used in the definition of the fissile media and are assumed in criticality design calculations. The physicochemical form of the fissile material is defined by:

- Its chemical composition
- The pellet diameter (if applicable)
- The rod characteristics (if applicable)
- The assembly characteristics (if applicable).

For the AP process, a conservative or realistic (based on process information) assumption concerning the chemical form of the fissile matter is made for each step of the process, taking into account not only the nominal conditions, but also possible process upsets (e.g., failure of a PuO_2 filter or unwanted soda introduction that may cause precipitates) defined based on the double contingency principle. Single-parameter limits for physicochemical characteristic control are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded) using documented and approved methods. The different chemical forms used in the criticality analyses are:

- PuO_2
- $Pu(NO_3)_4$
- $Pu(NO_3)_3$
- Plutonium oxalate.

In the MP process, no chemical transformations take place. As a consequence, the oxide form of the fissile medium (PuO_2 and/or UO_2) is assumed.

When process variables can affect the physicochemical form, controls to maintain it are identified as IROFS in the NCSEs and ISA Summary.

6.4.4.13 Process Variable Control

The use of process variable control as a criticality control parameter is to control the process (Process Control) to affect the credited criticality safety of the system. MOX Services does not currently credit process control for demonstrating criticality safety. Thus, no process controls are credited currently in the NCSEs. However, it may be useful to do so in the future. The following is a description of Process Control that could be credited for criticality safety:

Process variable control involves taking credit in the criticality safety demonstration for the control of process conditions maintained within fissile systems. If SSCs or procedures that control the process parameters were credited for criticality safety, these would be identified as IROFS in NCSEs. Normal operational tolerances and variability in the process would be accounted for in establishing the NCS analyzed conditions. These bounding conditions would be addressed in calculations and the establishment of the IROFS controls. As noted above, MOX Services does not currently use process variable control for criticality.

6.4.5 Margin of Subcriticality and Double Contingency Principle

To develop the USL for each of the AOAs, accepted industry codes such as SCALE code packages using an accepted cross-section library (e.g., CSAS26 (KENO) sequence and the 238 energy group cross-section library 238GROUPNDF5) are used. (Other computation code systems may be used if they are qualified in accordance with the MPQAP.)

6.4.5.1 Regulatory Requirements, Guidance, and Industry Standards

Title 10 CFR §70.61(d) requires that "under normal and credible abnormal conditions, nuclear processes are subcritical, including use of an approved margin of subcriticality for safety." To comply with this requirement, an industry-accepted standard practice is used (i.e., ANSI/ANS-8.1-1998). According to industry standards, a validation report for computer codes is developed that describes the development of the USL, including (1) demonstrating the adequacy of the margin of subcriticality for safety by assuring that the margin is relatively large compared to the uncertainty in the calculated value of k_{eff}, and (2) determining the AOAs and use of the code within the AOA, including justification for extending the AOA by using trends in the bias. Only these validated methods with the corresponding validation reports are used for each methodology used to make an NCS determination.

6.4.5.2 Calculational Method

The SCALE code package is the computational system used for MFFF criticality analyses. (Other computation code systems may be used if they meet the requirements of the MPQAP.) This code package is available from the Radiation Safety Information Computational Center.

SCALE is a collection of modules designed to perform nuclear criticality, shielding, and thermal calculations. Each SCALE functional module may be run individually, or a sequence of functional modules may be executed using a special module referred to as a control module. For criticality analyses, various criticality safety analysis sequence (CSAS) control modules are available. The CSAS control modules differ in the specific functional modules executed and in the processing of cross sections used as input. As a practice, MFFF criticality analyses are performed using approved and industry-accepted control module and cross-section libraries. The calculation of k_{eff} is performed using the KENO Monte Carlo transport code.

6.4.5.3 Criticality Code Validation Methodology

To establish that a system or process is subcritical under normal and credible abnormal conditions, it is necessary to establish acceptable subcritical limits for the operation, and then show that the proposed operation will not exceed such subcritical limit. Software, meeting the requirements of the MPQAP, is used to determine the USL for each of the AOAs. Each documented, reviewed, and approved methodology validation report is incorporated into the configuration management program. Each report includes the following:

- A description of the theory of the methodology including the validity of assumptions and independent duplication of results.
- A description of the use of pertinent computer codes, assumptions, and techniques in the methodology.
- A description of the verification of the proper functioning of the mathematical operations in the methodology.
- A description of the benchmark experiments and data derived from them that were used for validating the methodology.
- A description of the bias, uncertainty in the bias, uncertainty in the methodology, uncertainty in the data, uncertainty in the benchmark experiments, and margin of subcriticality for safety, as well as the basis for these items.
- A description of the hardware and software used.

The criticality code validation methodology is divided into four steps:

- Identify general MFFF design applications. The MFFF design applications and key parameters are associated with normal and design abnormal conditions.
- Select applicable benchmark experiments and group them into AOAs.
- Model the criticality experiments and calculate k_{eff} values of selected critical benchmark experiments.
- Perform statistical analysis of results to determine computational bias and the USL.

There are several substeps associated with selecting and grouping benchmark experiments. First, based on the key parameters, the AOA and expected range of the key parameter are identified. ANSI/ANS-8.1-1998 defines the AOA as "The range of material composition and geometric arrangements within which the bias of a calculational method is established." AOAs covering plutonium (Pu) and MOX applications are as follows: (1) Pu-nitrate solutions; (2) MOX pellets, fuel rods, and fuel assemblies; (3) PuO_2 powders; (4) MOX powders; and (5) aqueous solutions of Pu compounds. After identifying the AOAs, a set of critical benchmark experiments is selected. Benchmark experiments for the AOAs are selected from industry-accepted data.

6.4.5.4 Determination of Bias

ANSI/ANS-8.1-1998, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors* requires a determination of the calculational bias by "correlating the results of critical and exponential experiments with results obtained for these same systems by the calculational method being validated." The correlation must be sufficient to determine if major changes in the bias can occur over the range of variables in the operation being analyzed. The standard permits the use of trends in the bias to justify extension of the area of applicability of the method outside the range of experimental conditions.

The recommended approach for establishing subcriticality based on numerical calculations of the neutron multiplication factor is prescribed in Section 5.1 of ANSI/ANS-8.17-2004, *Criticality Safety Criteria for the Handling, Storage, and Transportation of Light Water Reactor (LWR) Fuel Outside Reactors*. The criteria to establish subcriticality requires that for a design application (system) to be considered subcritical, the calculated multiplication factor for the system, k_s, is noted to be less than or equal to an established maximum allowed multiplication factor, based on benchmark calculations and uncertainty terms. That is:

$$k_s \leq k_c - \Delta k_s - \Delta k_c - \Delta k_m \qquad \text{(Eq. 6.4.5.4-1)}$$

where:

k_s = the calculated allowable maximum multiplication factor, (k_{eff}) of the design application (system)

k_c = the mean k_{eff} value resulting from the calculation of benchmark critical experiments using a specific calculation method and data

Δk_s = the uncertainty in the value of k_s

Δk_c = the uncertainty in the value of k_c

Δk_m = the administrative margin.

Sources of uncertainty that determine Δk_s include:

- Statistical and/or convergence uncertainties
- Material and fabrication tolerances
- Limitations in the geometric and/or material representations used.

Sources of uncertainty that determine Δk_c include:

- Uncertainties in critical experiments

- Statistical and/or convergence uncertainties in the computation
- Extrapolation outside the range of experimental data
- Limitations in the geometric and/or material representations used.

Subcriticality requires the determination of an acceptable margin, based on known biases and uncertainties. The USL is defined as the upper bound for an acceptable calculation, as follows:

$$k_s + \Delta k_s \leq USL \qquad \text{(Eq. 6.4.5.4-2)}$$

The USL takes into account bias, uncertainties, and administrative and/or statistical margins, such that the calculated configuration is subcritical with a high degree of confidence.

6.4.5.5 Summary of USL for Each AOA

The development of the USLs takes into account bias and uncertainties, as well as an administrative margin. See Section 6.4.3 for a discussion of the administrative margin used for MFFF design applications within the AOAs. The USLs are applied as the basis for each nuclear criticality evaluation performed for MFFF. Table 6.4-1 identifies the USL, the key parameters and a definition of the MFFF AOAs.

6.4.6 Implementation of NCS in the ISA

Nuclear criticality calculations are performed for potentially fissile-bearing systems. In the design process, criticality safety calculations are performed to specify requirements for the design concept. The NCSEs assess both normal operating and process upset conditions. Where practical, nuclear criticality is precluded by demonstrating that the design is subcritical without the need to implement active engineered or administrative controls. In those cases in which it is not possible to demonstrate that a criticality is not credible, criticality control parameters are selected and limits on these parameters are established. Using the results of validated calculational methodologies, NCSEs demonstrate that both normal and process upset conditions meet the required minimum margin of subcriticality, and IROFS are identified to provide double contingency protection.

The NCSE evaluates normal and credible abnormal conditions developed in the component/ system Process Hazards Analysis (PrHA). The NCSEs demonstrate compliance with the double contingency principle. Passive engineered, active engineered, and administrative criticality safety controls are relied on to meet double contingency and to demonstrate that a criticality is highly unlikely. Controls are based on criticality calculations for conservative geometries (e.g., spheres, cylinders, and slabs, and supporting criticality safety calculations) that evaluate normal and credible abnormal conditions. Nominal configurations are also used to define the margin of safety. The criticality calculations determine and identify the criticality control (e.g., favorable geometry, safe spacing, process variables, concentration, content, and configuration) for the components or system being evaluated.

Criticality safety during design and operation is ensured for the MFFF. MFFF design and safety features are evaluated in NCS calculations and NCSEs that are documented, controlled, and maintained by implementing the management measures described in Chapter 15.

6.5 DESIGN BASIS

The following information represents the design basis attributes for Nuclear Criticality Safety.

- Reflection conditions equivalent to 1-in (2.5 cm) tight-fitting water jacket are assumed to account for personnel and other transient incidental reflectors not evaluated in the unreflected models.
- In cases where reflection control is not indicated, water reflection of process stations or fissile units is represented by a minimum of 12-in (30 cm) tight-fitting water jacket, unless consideration of other materials present in the design (e.g., concrete, carbon, or polyethylene) may be a more effective, more conservative assumption, than water.
- During maintenance there will be an administrative control to govern activities such as the removal of fissile material or otherwise ensure that any moderator used during maintenance is controlled.
- To develop the USL for each of the AOAs, accepted industry codes such as SCALE code packages using an accepted cross-section library [e.g., CSAS26 (KENO) sequence and the 238 energy group cross-section library 238GROUPNDF5] are used.
- The SCALE code package is the computational system used for MFFF criticality analyses. The calculation of k_{eff} is performed using the KENO Monte Carlo transport code.
- Regulatory Guide 3.71, *Nuclear Criticality Safety Standards for Fuels and Materials Facilities,* Revision 1, October 2005, (Including section 4.3.6 which states that licensees and applicants should provide the details of validation.)
- ANSI/ANS-8.1-1998, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors.* Clarifications are noted as follows:

  Section 4.2.2: MFFF process, material handling, or storage area designs incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality event is possible. For the purposes of demonstrating compliance with this requirement, "unlikely" is defined as events or event sequences that are not expected to occur during the facility lifetime, but are considered credible.

  Section 4.2.3: MFFF process design relies on engineered features where practical, rather than administrative controls.

  Section 4.3.2: In cases where an extension in the area(s) of applicability of a NCS analysis methodology is required, the method is supplemented by other calculational methods to provide estimate of bias in the extended area(s). As an alternative, the extension in the area(s) of applicability may also be addressed through an increased margin of subcriticality.

- ANSI/ANS-8.3-1997 (R2003), *Criticality Accident Alarm System*, is part of the design basis of MFFF process and fissile material handling and storage areas.

  The MFFF will maintain safe operations by immediately implementing compensatory measures (e.g., limit personnel access, halt SNM movement or activities) as necessary when the NIM system is unavailable or significantly degraded as approved by the nuclear criticality safety function

- ANSI/ANS-8.17-2004, *Criticality Safety Criteria for the Handling, Storage, and Transportation of Light Water Reactor (LWR) Fuel Outside Reactors.* Clarifications are noted as follows:

  Section 4.11: Fuel units and rods are handled, stored, and transported in a manner that provides a sufficient factor of safety to require at least two unlikely, independent, and concurrent changes in conditions before a criticality event is possible.

  Section 5.1: The criticality experiments used as benchmarks in computing k_c have physical compositions, configurations, and nuclear characteristics (including reflectors) similar to those of the system being evaluated.

  When the calculated multiplication factor for the design application (system) (ANSI/ANS-8.17 Section 5 [2004]) is shown to be less than or equal to an established maximum allowed value that properly accounts for method bias, uncertainty, and administrative margin. An administrative margin of 0.05 is used for MFFF design applications.

- ANSI/ANS-8.19-2005, *Administrative Practices for Nuclear Criticality Safety.* Clarifications noted as follows:

  Guidance for planned response to nuclear criticality events are addressed by ANSI/ANS-8.23-1997. Therefore, no commitment is made to satisfy the guidance or recommendations of this section.

  Procedures and their implementation are reviewed periodically at least once every two years.

- ANSI/ANS-8.20-1991 (R1999), *Nuclear Criticality Safety Training.*
- ANSI/ANS-8.21-1995 (R2001), *Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors.*
- ANSI/ANS-8.22-1997, *Nuclear Criticality Safety Based on Limiting and Controlling Moderators.*
- ANSI/ANS-8.23-1997, *Nuclear Criticality Accident Emergency Planning and Response.*

Table 6.4-1 identifies the USL, key parameters and a definition of the MFFF AOAs. AOAs covering plutonium (Pu) and MOX applications are as follows: (1) Pu-nitrate solutions; (2) MOX pellets, fuel rods, and fuel assemblies; (3) PuO_2 powders; (4) MOX powders; and (5) aqueous solutions of Pu compounds.

Table 6.4-1. Withheld from Public Disclosure Under 10CFR2.390

7.0 FIRE PROTECTION

The Mixed Oxide (MOX) Fuel Fabrication Facility (MFFF) fire protection program establishes policies and institutes a program to promote life safety, the conservation of property, and the continuity of operations through provisions of fire prevention and fire protection measures. The program establishes defense-in-depth practices for the protection of items relied on for safety (IROFS) and the procedures, equipment, and personnel required to implement the program. The fire protection program extends the concept of multiple layers of defense in fire protection to:

- Prevent fires from starting
- Detect fires rapidly and determine their location
- Inform MFFF workers of fires
- Inform the Savannah River Site (SRS) Operations Center of fires
- Control and limit the spread of fires
- Promptly extinguish fires
- Maintain safe egress paths for plant personnel in the event of fire
- Protect IROFS when a fire is not promptly extinguished by the fire suppression systems, so that neither an uncontrolled release of radioactive materials nor a criticality event occurs.

MFFF conduct of operations, administrative controls, and fire protection features and systems provide protection against fires and explosions based on defense-in-depth practices described in this chapter.

7.1 ORGANIZATION AND CONDUCT OF OPERATIONS

Organizational responsibilities, lines of communication, and personnel qualification requirements are defined in the fire protection program. Program documentation includes an organization chart and functional descriptions of the responsibilities of fire protection program personnel. MFFF key management functions are described in Chapter 4. Specific management responsibilities for fire protection are described below.

The manager of the plant has overall responsibility for formulation, implementation, effectiveness, and assessment of the MFFF fire protection program. This position is responsible for the development and administration of MFFF operations and fire response plans, and the fire protection and prevention program including post-fire safety considerations. The manager of the plant is the single point of control and contact for fire contingencies.

The manager of the production function is responsible for implementing periodic inspections to minimize the amount of combustibles in areas with IROFS, and for determining the effectiveness of housekeeping practices. This position is responsible for assuring the availability and acceptable condition of fire protection systems and equipment, fire stops, and fire-rated penetration seals; and for assuring that prompt and effective corrective actions are taken to remedy conditions adverse to fire protection, and to preclude their recurrence.

The manager of the maintenance function is responsible for periodic inspection and testing fire protection systems and equipment in accordance with established procedures, which include evaluation of test results and determination of the acceptability of the system under test. This position ensures that personnel responsible for the maintenance and testing of the fire protection systems are qualified by training or experience for such work.

The manager of the quality assurance function is responsible for assuring the effective implementation of the quality-affecting aspects of the fire protection program by planned inspections and scheduled audits, identifying adverse conditions or trends, and reporting adverse conditions or trends to management.

The manager of the regulatory function is responsible for fire safety. The fire protection function reports to the manager of the regulatory function. The fire protection function has implementation responsibility for the overall fire protection program and has input to organizations involved in fire protection activities. Refer to Chapter 4 for the minimum qualification requirements for the individual responsible for the fire protection function. This position is responsible for reviews and evaluations of proposed work activities to identify potential transient fire loads. He periodically assesses the effectiveness of the fire protection program, including fire drills and training. The results of these assessments are reported to management, with recommendations for improvements or corrective actions, as deemed necessary. Fire fighting training is implemented by the fire protection function, consistent with the requirements of the MFFF training program. The fire protection function ensures that the content of fire protection training is current and adequate, by reviewing the training content on a regularly scheduled basis.

The manager of the training function is responsible for providing MFFF specific training to the SRS Fire Department (FD). This position assists in the critique of fire drills to determine how well the training objectives have been met. He is also responsible for implementing a program for indoctrination of MFFF personnel (including contractor personnel) in administrative procedures that implement the fire protection program and emergency procedures relative to fire protection, including handling of leaks or spills of flammable materials that may be related to fire protection.

The SRS FD is responsible for fighting fires at the MFFF. Coordination with the SRS FD and responsibilities of the SRS FD are defined in work-task agreements or procedures between the MFFF and SRS.

The Authority Having Jurisdiction for licensed activities required to meet 10 CFR 70 is the U.S. Nuclear Regulatory Commission.

7.2 ADMINISTRATIVE CONTROLS

7.2.1 Fire Prevention

A key element of fire protection is fire prevention. The goal of fire prevention is to prevent a fire from starting. The basic components of fire prevention are:

- Prevention of fires and fire spread by placing controls on operational activities
- Design features such as the use of spark resistant electrical components where appropriate
- Design and administrative controls that restrict the use of combustible materials.

7.2.2 Surveillance Procedures

Fire protection surveillance procedures include inspections of combustible loading, fire protection equipment and systems, general housekeeping, and transient combustibles.

7.2.3 Control of Flammable and Combustible Materials

Flammable and combustible materials are controlled by design, and by procedures that limit:

- Bulk storage of combustible materials inside, or adjacent to, buildings or systems containing IROFS during operation or maintenance periods.
- Handling and use of ordinary combustible materials, combustible and flammable gases and liquids, combustible high efficiency particulate air and charcoal filters, dry ion exchange resins, pyrophoric materials, and other combustible supplies in areas containing IROFS. Flammable and combustible liquids are stored, handled, and used in accordance with applicable sections of National Fire Protection Association (NFPA) 30, *Flammable and Combustible Liquids Code*, 1996 edition.
- Storage and handling of pyrophoric metals to methods in the applicable codes and/or industry standards and require that an adequate supply of extinguishing agent for pyrophoric metals is present. Procedures for pyrophoric metals also establish operating limits and controls. Combustible loading in areas containing IROFS is in accordance with applicable guidance in NFPA 801, *Standard for Fire Protection for Facilities Handling Radioactive Materials*, 1998 edition. Flammable and combustible liquids are stored, handled, and used in accordance with applicable sections of NFPA 30, *Flammable and Combustible Liquids Code*, 1996 edition. Flammable and combustible gases are stored, handled, and used in accordance with applicable portions of NFPA 50A, *Standard for Gaseous Hydrogen Systems at Consumer Sites*, 1999 edition and NFPA 55, *Standard for the Storage, Use, and Handling of Compressed and Liquefied Gases in Portable Cylinders*, 1998 edition. As part of the integrated safety strategy for explosion events EXP03 and EXP06, explosion prevention measures follow the guidelines of NFPA 69, *Standard on Explosion Prevention Systems,* 1997 edition.

- Handling of transient fire loads, such as combustible and flammable liquids, wood and plastic products, or other combustible materials in buildings containing IROFS during the phases of operation, and especially during maintenance or modification activities.
- Use of wood is permitted only when noncombustible products are not practical from a process consideration. Where used, wood is treated with a flame retardant.
- Unpacking of transient combustible materials is done outside of MFFF production areas as much as practical. When necessary, transient combustible packing materials may be unpacked inside MFFF production areas; however, the materials are removed from the area following unpacking. Loose combustible packing material — such as wood or paper excelsior or polyethylene sheeting — is placed in metal containers with tight-fitting, self-closing metal covers if the material remains in production areas.
- Work-generated combustible waste is removed from buildings containing IROFS following completion of the activity, or at the end of the shift, whichever comes first.

7.2.4 Control of Ignition Sources

Ignition sources are controlled by design such as selection of appropriate electrical equipment in gloveboxes where combustible material is present, absence of electrical equipment in process cells, and where appropriate, use of spark resistant electrical equipment. Ignition sources are also controlled by work control procedures requiring:

- Permits to control welding, grinding, flame cutting, brazing, or soldering operations; separate permits for each area where work is to be performed; and allowable duration for validity of permits
- Welding and grinding in accordance with applicable portions of NFPA 51B, *Standard for Fire Prevention During Welding, Cutting, and Other Hot Work*, 1999 edition
- Prohibition of open flames or combustion-generated smoke for leak testing
- Smoking is restricted to designated areas outside of the MFFF buildings.

7.2.4.1 Selection of Electrical Equipment in Gloveboxes

The following national codes and standards are used for selection of electrical equipment in gloveboxes to assure that the risk of electricity as an ignition source of fires and explosions is minimized:

- Occupational Safety and Health Administration (OSHA) 29 CFR 1910 (2004), Occupational Safety and Health Standards
- UL 508 (1993 edition), Industrial Control Equipment

The cable insulation within the gloveboxes meets the flame requirements from one or more of the following standards:

- IEEE 383 (1974 Reaffirmed 1992), Standard for Qualifying Class 1E Electric Cables and Field Splices for Nuclear Power Generating Stations or code that meets equivalent flame requirements.
- NFPA 262-2002, Standard Method of Test for Flame Travel and Smoke of Wires and Cables for Use in Air-Handling Spaces
- IEEE 1202-1991, -1996, Standard for Flame Testing of Cables for Use in Cable Tray in Industrial and Commercial Occupancies
- UL-910 – 1998, Standard for Safety Test for Flame-Propagation and Smoke-Density Values for Electrical and Optical Fiber Cables used in Spaces Transporting Environmental Air
- UL-1666-2002, Test for Flame Propagation Height of Electrical and Optical-Fiber Cables Installed Vertically in Shafts
- UL-1685-1997, Standard for Safety Vertical-Tray Fire-Propagation and Smoke Release Test for Electrical and Optical-Fiber Cables
- FT4 of CSA C22.2 No. 0.3-92 Para 4.11.4, Test Methods for Electrical Wires and Cables (Acceptable Cable Flame Test)
- FT6 of CSA C22.2 No. 0.3-92 Appendix B, Test Methods for Electrical Wires and Cables (Acceptable Cable Flame Test)
- IEC 60332-3, Tests on Electric Cables under Fire Conditions Part 3: Tests on Bunched Wires or Cables (1992) (Acceptable Cable Flame Test)
- ICEA T-29-520, Conducting Vertical Cable Tray Flame Tests with Theoretical Heat Input Rate of 210,000 B.T.U/Hour (Acceptable Cable Flame Test
- ASTM D2633, Standard Test Methods for Thermoplastic Insulating and Jackets for Wire and Cable

The suitability of electrical distribution equipment, (i.e., wire, cable, and other equipment covered by the NFPA 70) for use within the process gloveboxes, is demonstrated by listing and appropriate labeling by a Nationally Recognized Testing Laboratory (NRTL), such as UL or FM.

NFPA 70 provides the primary criteria used in the design of electrical power distribution to the gloveboxes to minimize the risk of electricity as an ignition source of fires and explosions. Electrical power provided to the glovebox equipment is designed and installed according to the requirements of NFPA 70. The industrial electrical safety requirements of 29 CFR 1910 generally adopt and may supplement the electrical distribution criteria found in NFPA 70. Electrical distribution to the glovebox equipment is via electrical components selected, sized, assembled, and protected as required by NFPA and OSHA for industry.

Engineering studies were performed in accordance with Article 500 of NFPA 70 to determine the Hazardous Location classification of each glovebox environment. Electrical distribution and utilization equipment selected for each glovebox is based upon the classification assigned.

Within the process gloveboxes, electrical utilization equipment is considered acceptable per 29 CFR 1910.399 if it is:

- Listed by an NRTL; or
- Inspected by another federal, state, municipal, or local authority responsible for enforcing the National Electrical Code (NFPA 70) if the equipment is not covered by an NRTL; or
- Custom made and intended for use by a particular customer and documented by the manufacturer to be safe.

It is required that glovebox electrical utilization equipment either have evidence of this acceptability, be exempted by meeting the low power criteria of UL 508, or be accepted based on an evaluation in accordance with 29 CFR 1910.303, for its mechanical strength, and durability, electrical insulation, heating effects under conditions of use, arcing effects, and classification by type, size, voltage, current capacity, specific use, and any other factors that contribute to its safety and determine it to be safe from electrical fire potential.

7.2.5 Testing, Inspection, and Maintenance

The MFFF fire protection systems and features are inspected, tested, and maintained. Inspection, testing, and maintenance are documented by means of written procedures, with the results and follow-up actions recorded. Water-based MFFF fire protection systems and equipment are inspected, tested, and maintained in accordance with applicable portions of NFPA 25, *Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems,* 1998 edition. Other MFFF fire protection systems are inspected, tested, and maintained in accordance with the applicable requirements of their applicable NFPA codes, manufacturer's guidelines, and operating experience. Safety controls and interlocks for combustible liquids, flammable liquids, and flammable gases and their associated delivery systems are tested periodically and after maintenance activities.

A test plan lists the responsible personnel positions in connection with routine tests and inspections of the fire detection and protection systems. The test plan contains the types, frequency, and identification of the testing procedures.

A penetration seal-tracking program records pertinent information regarding the installation and modification of fire-rated penetration seals that are IROFS.

Emergency lighting and communications systems are inspected, tested, and maintained in accordance with vendor recommendations.

Onsite and offsite emergency communications systems are tested periodically in accordance with the site emergency preparedness program.

7.2.6 Impairments

Fire protection is maintained during those periods when a fire protection system is impaired, or during periods of maintenance. To achieve this continuity of fire protection, written procedures address impairment of MFFF fire protection systems. Disarming of MFFF fire detection or fire suppression systems is controlled by a permit system. Together, the impairment procedure and permit system include the following:

- Identification and tracking of impaired equipment
- Identification of personnel to be notified
- Determination of needed compensatory fire protection and fire prevention measures.

If protection system impairment is planned, the necessary parts and personnel are assembled prior to removing the system from service. When an unplanned impairment occurs, or when a system has discharged, the repair work or fire protection system restoration is expedited.

Compensatory measures (e.g., fire watches) are implemented as appropriate in accordance with procedures when IROFS fire protection features and systems (i.e., Quality Level [QL]-1 fire barriers, fire doors, fire dampers, fire-rated penetration seals, fire suppression systems, and fire detection systems) are not operable.

Acceptable outage times are specified in work control procedures for fire protection system impairments. Exceeding the acceptable outage times for IROFS fire protection systems requires additional compensatory measures, which could include shutdown of processes in affected areas. Once repairs are completed, tests are conducted to ensure proper operation and restoration of fire protection equipment capabilities.

7.2.7 Fire Response Planning

Procedures identify actions to be taken by an individual discovering a fire, including guidance for notifying appropriate personnel. Procedures specify means and methods that may be used by MFFF staff to extinguish a fire (see also Section 7.4.2).

The response procedures specify actions to be taken to determine the need for assistance when a fire is reported or a fire alarm is received at an annunciation panel. For example, these actions may include announcing the location of a fire over the MFFF public address system, sounding fire alarms, notifying the shift supervisor, and notifying the SRS Operations Center.

7.2.8 Pre-fire Plans

Pre-fire plans are developed by the SRS FD. They define the strategies that are used at MFFF for fighting fires in areas containing IROFS or that present a hazard to IROFS. For those areas, pre-fire plans identify:

- Fire hazards in each area covered by the specific pre-fire plans
- Fire extinguishing agents best suited for controlling the fires associated with the fire hazards in that area and the nearest location of these extinguishing agents
- The direction from which to attack a fire in each area in view of the ventilation direction, access hallways, stairs, and doors that are likely to be free of fire, and best station or

elevation for fighting the fire. The access routes that involve locked doors are specifically identified with the appropriate precautions and methods for access specified

- Management of MFFF systems to reduce the damage potential during a fire and the location of local and remote controls for such management
- Heat-sensitive system components or hazardous combustibles that need to be kept cool while fighting a fire
- Coordination between MFFF staff and the SRS fire department
- Potential radiological and toxic hazards
- Operations requiring control room coordination or authorization
- Instructions for MFFF operators and general MFFF personnel during a fire.

7.3 FIRE PROTECTION FEATURES AND SYSTEMS

Fire protection features and systems consist of fire barriers, fire detection and alarm systems, fire suppression systems, fire protection water supply system, and smoke control features.

7.3.1 Fire Barriers

The function of the fire barriers is to separate fire areas from one another. Fire areas confine fires, smoke, and gases to their areas of origin to prevent fires from spreading to adjacent areas. Fire barriers are used to separate IROFS and to separate areas that contain materials and processes that contain fire hazards into fire areas. The major components of fire barriers are walls, floors, ceilings, doors, fire dampers, fire wrap, and fire-rated penetration seals. The passive boundaries of the fire areas are fire-rated for a minimum of 2 hours; fire area barriers rated at less than 2 hours are justified in the Fire Hazards Analysis. Some passive boundaries are fire-rated for 3 or 4 hours. Firewalls, floors, and ceilings are constructed of noncombustible materials. Firewalls maintain sufficient structural stability under fire conditions to allow the collapse of structures on either side without collapse of the wall itself. Structural members that support firewalls, floors, and ceilings have a fire-resistance rating that is equal to or greater than the barrier supported.

Fire doors are Factory Mutual (FM) approved or Underwriters Laboratory (UL) listed or are qualified through additional testing or analysis to U.S. standards or an equivalent method. Their fire-resistance ratings are compatible with the hazard expected in the area. Fire doors that do not have automatic closures can be manually operated. These doors are the fire doors of the Pellet Handling (PML) unit and the revolving fire doors of the Jar Storage and Handling (NTM) unit. These doors are not UL listed or FM approved.

Mechanical and electrical penetrations between fire areas have a fire rating (specifically an F-rating) consistent with that of the fire barriers. Vertical shafts are enclosed by fire barriers and have a rating no less than 2 hours. Fire barrier penetration seals are also capable of withstanding fire induced pressures and remain sufficiently leak tight. Penetration seals in 2 and 3 hour rated fire barriers that enclose areas with gaseous suppression systems are capable of withstanding a room pressure increase as a result of gaseous suppression system discharge and remain sufficiently leak tight.

Closure devices with fire-resistance ratings are provided where ventilation penetrates fire barriers. These devices have fire-resistance ratings that are consistent with the designated fire-resistance ratings of the fire barriers penetrated. Five different fire damper configurations exist for the four different HVAC systems:

- In C2 ventilation areas (for example, process rooms containing rods or assemblies and corridors around C3 areas), automatic fire dampers are provided in the Medium Depressurization Exhaust (MDE) system supply and exhaust ductwork.
- In process rooms and other C3 ventilation areas (process rooms) with dispersible radioactive material, the High Depressurization Exhaust (HDE) system exhaust fire dampers have manual controls. The room supply fire dampers for these areas are automatic. The manual fire damper on the HDE exhaust from the process room is located in the hallway outside the process room. The operation of the manual fire damper is by either a chain wheel operator or a remote push-button electric motor operator accessible from the corridor.
- For the Process Cells Exhaust (POE) system, room exhaust fire dampers are manually operated. The room supply fire dampers for these areas are automatic.
- Fire block dampers may be mounted in the exhaust duct within a room. In these installations, the exhaust duct shall be directly routed out of the above rooms.
- In C4 ventilation areas (gloveboxes), fire isolation valves in the Very High Depressurization Exhaust (VHD) headers are manually controlled.

Where shutdown of a ventilation system is not desirable (that is, where the loss of confinement might pose a greater threat than the spread of fire), fire dampers are not required for ventilation duct penetrations; alternative means of protecting against fire propagation are provided, such as manually closed isolation dampers, duct wrapping, duct enclosure, and/or rerouting.

Redundant IROFS systems and components are separated by fire barriers that are sufficient to ensure that a fire in one train of IROFS equipment shall not affect the operation of the redundant train, even if fire suppression systems fail to operate. Where fire barriers do not separate redundant IROFS systems or components, these configurations are identified and justified. For instance, within gloveboxes, it is not feasible to separate redundant IROFS, such as sensors. The justification for not separating these IROFS is that operations are expected to halt if there is a fire in the glovebox, obviating the need for the sensors. As another example, where train A and train B cables are not separated by fire barriers, the justification is that the cables are self-extinguishing, separated by suitable distance and in an area subject to the combustible control program.

7.3.2 Fire Detection and Alarm Systems

The functions of the MFFF fire detection and alarm system are: monitor for fire conditions, fire suppression system actuation, fire protection system fault conditions, and actuate automatic systems it controls. The fire alarm system is designed according to NFPA 72, National Fire Alarm Code.

Fire detection and alarm communication devices include the proprietary supervising workstation (PSW), annunciator panels, local fire alarm control panels, the fire alarm panel data network, the digital alarm communications transmitter, and firefighter telephones.

The fire detection and alarm system has a PSW, which is located in the polishing and utilities control room (PUCR) (room D-301). The PSW serves as an interactive interface between the control room operator and the network system. Fire alarm information is also sent from the PSW to workstations in each process control room, which monitor alarms (for example, radiation, fire, toxic chemicals, and so on).

The PSW includes an integral Digital Alarm Communications Transmitter (DACT), which retransmits these signals to the Savannah River Site (SRS) Operations Center.

The MFFF fire detection system alarm signals are transmitted to fire annunciator panels in the Central Alarm Station and Secondary Alarm Station. Additional fire annunciator panels are provided in the emergency control rooms (D-318 and D-319), the operations support center (G-111), and the utilities control room (B-319) for use during emergency situations. The fire annunciator panels primarily provide alarm information based on the location.

The local microprocessor-based fire alarm control panels (FACPs) receive fire alarm signals that are generated by alarm initiating devices, such as detectors, manual pull stations, and water flow switches, as well as supervisory and trouble signals. The FACP identifies the device or circuit in alarm and then audibly and visibly annunciates the off-normal condition by location and type of device. The FACP automatically routes alarm, supervisory, and trouble signal information, which are programmed as “public points,” to other control panels (nodes) on the network, including the PSW and fire annunciator panels, such as those in the Central Alarm Station and Secondary Alarm Station. Fire safety functions (for example, activation of automatic fire suppression systems, glovebox or process fire door closure, and elevator capture) are initiated by the local control panels. Notification devices, such as strobes and horns, are automatically activated by the local FACP to warn personnel in the area to evacuate to safe havens during fire events.

In the event of a fire in an area protected by a clean agent suppression system, clean agent controls, either in the FACPs or in separate panels, control the clean agent suppression system (pre-release alarm, time delay, abort switch operation, clean agent release, and damper operation). If clean agent is released, a supervisory signal is sent throughout the network with the associated details.

The PSW, local FACPs, and additional fire annunciator panels are connected together as nodes. A node that receives an alarm, supervisory, or trouble signal automatically retransmits that signal to the next node on the ring, which indicates the location and type of device, circuit, or panel that is originating the signal. Fire detection system information is available to other panels on the network and displayed if the system is programmed to do so. Communications on this network are dedicated to fire alarm service.

Alarm, supervisory, and trouble signals from any FACP are transmitted from the DACT to the SRS Operations Center.

Two-way firefighter telephones are provided for emergency communications in local FACPs and other locations. There is at least one per floor and at least one per exit stairway.

Alarm initiating devices include smoke detectors, heat detectors, duct smoke detectors, water flow detectors, and manual pull stations. Upon actuation, these devices notify the fire detection system. Manual pull stations are readily accessible, and located in the normal path of exit from each floor.

Automatic smoke and heat detectors are located throughout the MFFF. The type of detector used in a given location is based on the fire hazards in the area, the function of the detector, and potential for false alarms. The BAP glovebox fire detectors are stainless steel heat detectors. The MFFF laboratory gloveboxes that are provided with glovebox decontamination solution capability contain fire detectors that are stainless steel heat detectors. Smoke and/or heat detectors are also located in the HVAC Supply Air (HSA) ventilation intake header. Heat detectors are provided upstream of HVAC final filters. Smoke detectors are also installed in the ventilation exhaust ducts of the process cells, which are inaccessible during plant operation.

Installation of smoke and heat detectors is in accordance with NFPA 72.

Water flow sensors and alarms that respond to the sensors are provided wherever a sprinkler system preaction, wet-pipe, or deluge is installed.

Alarm notification devices are located throughout the facility. Strobes outside of a room warn personnel not to enter that room after a fire detector within that room is activated. In areas that are protected by clean agent suppression systems, both audible and visible pre-discharge alarms are provided.

During a fire emergency, emergency notification may be activated over the PA speakers. Provisions for manual control of the PA system are provided, including visible on/off status indication.

7.3.3 Fire Suppression Systems

The fire suppression systems provide fire suppression in the form of the appropriate extinguishing agents for MFFF areas. Fire suppression systems for the MFFF are composed of:

- Water-based suppression systems (preaction, wet-pipe, deluge)
- Carbon dioxide systems
- Clean agent systems
- Standpipe systems
- Portable fire extinguishers.

7.3.3.1 Water-Based Suppression Systems

The water-based fire suppression systems provide fire suppression in areas where water is the preferred means of suppression.

Sprinkler systems are preaction type systems, which consist of closed-head sprinklers and normally closed preaction valves, wet-pipe sprinkler systems, or deluge systems. Due to nuclear criticality safety concerns, hydrogenous material (e.g. water) is not used as a suppression agent in process rooms and in areas that contain nuclear material. Water-based suppression system piping is not routed through process rooms.

Preaction-type water suppression systems are the predominant water-based suppression system used in the MP, SR, and AP areas and in the Emergency Generator building. A preaction system is a supervised design that uses an electrically activated preaction valve operated by fire detectors. The preaction systems are interlocked and require two simultaneous activating signals to operate. The preaction-type system requires independent actions to allow the discharge of water. These actions are the opening of the sprinkler by heat from the fire and the opening of the preaction valve by heat or smoke detectors. The automatic control valves of the preaction systems are independent of detection devices and sprinklers.

Wet-pipe sprinkler systems are used in buildings and areas of the MFFF that would not be significantly impacted by water damage from inadvertent operation of the sprinklers (for example, the Administration, Technical Support, Secured Warehouse, and Reagent Processing buildings). The wet-pipe sprinkler system uses automatic sprinklers that are attached to a piping system containing water supplied by the firewater supply system. When heat from a fire actuates a sprinkler, water discharges immediately from the opened sprinkler(s). The system is supervised and annunciated through the use of flow-sensing devices.

The Emergency Fuel Storage Vault (UEF) and the truck bay areas in the Shipping and Receiving building are equipped with automatic deluge systems. In the deluge system, sprinklers are always open. When the smoke or heat from a fire actuates a fire detector, the fire detection system sends a signal to open the appropriate deluge valve. Water is discharged from sprinklers on the piping system. After the fire is extinguished, closing the appropriate deluge valve stops water flow. The automatic control valves of the deluge systems are provided with hydraulic, pneumatic, or mechanical manual means for operation that is independent of detection devices and of the sprinklers.

7.3.3.2 Carbon Dioxide Systems

The carbon dioxide fire suppression systems (portable Carbon Dioxide (CO_2) bottles) provide for manual fire suppression of incipient fires in MFFF gloveboxes that present significant fire hazards.

Manually operated carbon dioxide fire suppression systems are provided for the MFFF gloveboxes. The carbon dioxide systems consist of portable CO_2 bottles with hose and quick connection fittings, and glovebox fittings and piping to direct the CO_2 to the interior areas of the gloveboxes. The glovebox connections are equipped with flow-restricting orifices to prevent overpressurization of the glovebox. The injection points on gloveboxes provide one injection point for each 4 m^3 of glovebox. The portable CO_2 bottles are located in close proximity to the glovebox to be protected.

In the event of a fire in a glovebox, the smoke/heat from a fire is detected by the fire detection system, which alerts the operators in the PUCR. As long as the glovebox fire is still in the incipient phase, a specially configured portable CO_2 bottle may be manually connected to the affected glovebox and the CO_2 bottle actuated (without impact to confinement) to attempt to extinguish the fire.

7.3.3.3 Clean Agent Systems

The clean agent fire suppression systems provide fire suppression in areas where clean agent is the preferred means of suppression (that is, water-based suppression is undesirable).

A typical clean agent actuation system consists of an electronic control panel, appropriate fire detectors for the type of hazard, valve actuators, pilot cylinder valves, and slave valves.

7.3.3.3.1 Non-halogenated Clean Agent Suppression System

The non-halogenated clean agent supply is provided by high-pressure storage cylinders that are located in dedicated cylinder rooms and in various discrete locations near the room(s) that they are protecting. A reserve quantity of non-halogenated clean agent is provided equal to the largest demand from the largest non-halogenated clean agent storage location. The non-halogenated clean agent reserve, which is an unconnected reserve, is maintained on the MFFF site. The non-halogenated clean agent system is designed to activate automatically by fire detectors. A typical non-halogenated clean agent delivery system consists of agent stored in a bank of cylinders, a cylinder manifold, a distribution manifold, a pressure-reducing orifice and selector valve for each fire area, distribution piping to each fire area, and a network of distribution piping and discharge nozzles for each room or protected space within the fire area.

The number of cylinders that are required for a given cylinder bank for a given fire area is based on the required minimum design concentration for the given volume of protected spaces within the fire area. Cylinder banks are sectioned by check valves to deliver agent from a pre-determined number of cylinders to a given protected area.

Typically, no more than 40 cylinders are connected to a cylinder manifold. The cylinder manifold supplies agent to the distribution manifold for delivery of agent to the signaling fire area. Based on the number of cylinders in a cylinder bank and the location of the cylinder bank, cylinders are arranged in single-row, double-row, or back-to-back double-row configurations. Cylinders are located and supported in cylinder racks.

The distribution piping serves to connect the storage cylinders to the intended fire area or protected space.

A pressure reducing orifice, or pressure reducer, is provided in the distribution piping to restrict the flow of non-halogenated clean agent and thus the agent pressure down stream of the pressure reducer.

The selector valve is located in the distribution piping after the pressure-reducing orifice. The selector valve is provided to direct agent to a specific fire area connected to the distribution manifold. The selector valve is actuated by the electronic control panel, which is part of the non-halogenated clean agent actuation system.

Following the injection of non-halogenated clean agent, the supply ventilation damper(s) to the room(s) is (are) automatically closed by the fire detection alarm and control system. The exhaust ventilation of the room is automatically secured following the injection of non-halogenated clean agent except in rooms containing dispersible radioactive materials; where needed, the design includes a specified quantity of clean agent for extended discharge to account for the loss of suppressant through the exhaust ducting. This ensures that the oxygen concentration in the area will not increase above 15 percent.

Discharge nozzles are provided for each protected space to control the distribution of non-halogenated clean agent and the rate of non-halogenated clean agent flow into the protected space.

The design and installation of non-halogenated clean agent systems and agent quantity requirements complies with the requirements of NFPA 2001. The non-halogenated clean agent system has been designed to prevent overpressurization of the process room when it actuates. As such, the design of the non-halogenated clean agent system incorporates the following details:

1. Extended discharges are utilized, where needed, in rooms containing gloveboxes. The extended discharge of non-halogenated clean agent utilizes dedicated clean agent cylinders.
2. The minimum clean agent design concentration is based on a 20 percent safety factor for flame extinguishment.
3. The soak time is 20 minutes to allow for the SRS Fire Department to arrive and enter the affected area.

7.3.3.3.2 Halogenated Clean Agent Suppression System

The halogenated clean agent supply is provided by cylinders located inside or near the room(s) that they are protecting. A reserve quantity of halogenated clean agent is provided equal to the largest demand from the largest halogenated clean agent storage location. The halogenated clean agent reserve, which is an unconnected reserve, is maintained on the MFFF site. The halogenated clean agent system is designed to activate automatically by fire detectors. A typical halogenated clean agent delivery system consists of a cylinder or cylinders, a network of distribution piping and discharge nozzles for each room or protected space within the fire area.

The number of cylinders required for a given fire area is based on the required minimum design concentration for the given volume of protected spaces within the fire area. The distribution piping serves to connect the storage cylinders to the intended fire area or protected space.

Discharge nozzles are provided for each protected space to control the distribution of halogenated clean agent and the rate of halogenated clean agent flow into the protected space.

7.3.3.4 Standpipe Systems

Standpipe systems provide manual fire suppression capabilities for the MFFF.

The MP, SR, and AP areas have a dry standpipe system (instead of a normally pressurized wet standpipe system). Wet standpipe systems provide manual fire protection for MFFF areas other than the MP, SR, and AP areas.

Fire fighters operate the standpipe systems closest to the fire. For the standpipe systems that are normally dry, operation of the system requires the opening of an isolation valve. After the fire is extinguished, the standpipe water supply is secured and the standpipe is drained.

7.3.3.5 Portable Fire Extinguishers

Portable fire extinguishers are provided throughout the MFFF and inside buildings to provide for extinguishing fires during their incipient phase.

The portable extinguishers are distributed as a function of their effectiveness in fighting fires in the area. The portable fire extinguishers are primarily multipurpose dry chemical A, B, C type. Metal fire extinguishers (Class D) are provided in areas where cladding material (zirconium alloy swarf) could result in a metal fire, and CO_2 extinguishers are provided in areas that contain energized electrical equipment. Wheeled 100-lb CO_2 extinguishers are placed in personnel and material corridor locations to support fire fighting in process rooms with moderation controls and other areas where the use of CO_2 is appropriate.

In the event of a fire in an MFFF building, the person who discovers the fire notifies the operators in the PUCR of the location and the extent of the fire. If the fire is small and still in the incipient phase, a portable extinguisher, which is located nearby, may be used to attempt to quickly extinguish the fire. After the fire is extinguished, the fire extinguisher is replaced or recharged.

7.3.4 Fire Protection Water Supply System

The distribution system of the fire protection water supply system is composed of the necessary piping, valves, flow and pressure control equipment, and instrumentation and controls to provide a reliable source of fire protection water for fighting onsite fires. This system is designed to provide for the largest sprinkler system demand plus 500 gpm for hose streams.

The fire protection water supply system is a 12 inch underground firewater loop around the MFFF site. The loop consists of piping and valves to convey water from the SRS fire protection water distribution system to MFFF yard hydrants and water based fire protection systems. The yard fire hydrants are located around the MFFF for use by fire fighters. The SRS F area firewater loop, via multiple supply lines to the MFFF firewater loop, provides the necessary water to support the MFFF requirements.

Calculations were performed to determine the demand of each MFFF sprinkler system as well as the system with the highest demand. For conservatism, the most direct flow path of the MFFF underground firewater loop was assumed to be isolated and unavailable so that the hydraulically most demanding pressure drop path to the building with the highest flow demand was analyzed. It was determined that the system with the largest demand that provides mitigation for a fire, defense-in-depth protection against a nuclear material release or criticality is the deluge system protecting the Shipping and Receiving Area truck bays. The maximum demand for this system is within the capabilities of the firewater supply from the SRS fire protection water distribution system.

Potentially contaminated firewater is collected in contaminated drain systems.

7.3.4.1 Other Fire Protection Features and Systems

A nonflammable hydrogen/argon mixture is utilized in the MP area within the sintering furnaces.

Exposed interior walls and ceilings (including ceilings formed by the underside of roofs) and factory-installed facing material have a flame spread rating of 25 or less and a smoke developed rating of 50 or less.

Facility evacuation routes are provided for evacuation of facility personnel. Exit routes are clearly marked and provided with egress lighting.

Safe havens provide for detaining personnel during emergency egress from the BMF until such time as personnel can be processed by security personnel for release.

Emergency lighting is provided for critical operations areas (that is, areas where personnel might be required to operate valves, dampers, and other controls in a fire emergency).

Drainage in areas handling radioactive materials is sized to accommodate a spill of the largest single container of any flammable or combustible liquid in the area. Floor drainage from areas containing flammable or combustible liquids is trapped to prevent the spread of burning liquid beyond the area. Drainage and prevention of equipment flooding is accomplished by one or more of the following methods:

- Floor drains
- Floor trenches
- Open doorways or other wall openings.

Laboratories that use chemicals or nuclear materials are operated in accordance with applicable safety criteria of NFPA 45, *Standard for Fire Protection for Laboratories Using Chemicals*, 1996 edition.

Noncombustible storage racks within the MFFF are used for the storage of plutonium oxide, uranium oxide, or mixed oxide in powder, pellet, or rod form. Additionally, the areas where these storage racks are located are free of combustible material storage.

7.3.5 Smoke Control Features

Smoke control features prevent the spread of smoke and combustion gases during a fire and remove smoke and combustion gases after a fire has been extinguished. The heating, ventilation, and air conditioning system uses filters and fire dampers to control smoke and combustion gases during and following a fire.

7.4 MANUAL FIRE FIGHTING CAPABILITY

The MFFF manual fire fighting capability consists of the SRS FD. The SRS FD is a full time professional fire department sufficiently trained and qualified to fight MFFF fires. Manual fire fighting needs assessments conducted by the SRS FD and CB&I AREVA MOX Services, LLC determined that minimum required onsite fire fighting capabilities are met by the SRS FD.

7.4.1 Equipment

Fire fighting equipment, including portable fire extinguishers, is maintained and inspected based on experience, manufacturer's recommendations, and applicable codes to assure the safe operational condition of the equipment.

7.4.2 Training

The SRS FD is provided training in operational precautions, radiological protection, and special hazards that could be present when fighting fires on the MFFF site. General employee training provides MFFF employees with training on actions to take upon discovering a fire, including notifications, when it is appropriate to attempt to extinguish a fire, and fire fighting methods that may be used, including manual activation of suppression systems.

7.4.3 Fire Drills

Fire drills are performed at regular intervals in the MFFF with the SRS FD.

7.5 FIRE HAZARDS ANALYSIS

MFFF's fire hazards analysis is reviewed and updated periodically, as necessary, following changes and modifications to the facility, processes, or inventories in accordance with MOX Services' configuration management process (Chapter 15).

MOX Services will perform analyses using guidance of Regulatory Guide 1.189 that consider the improper operation of IROFS equipment due to spurious signals induced by fire damage. The results of these analyses will be factored into the FHA, as appropriate.

7.6 CODES AND STANDARDS FOR FIRE PROTECTION

In addition to codes and standards identified in the Sections above, the codes and standards listed in Section 7.7.3 are applied to the design, construction, operation and/or maintenance of Fire Protection Systems.

7.7 DESIGN BASES

The design bases of fire protection at the MFFF ensure that adequate protection is provided against fires and explosions. The following information represents the design basis attributes for fire protection.

7.7.1 Equivalencies and Exemptions to Codes and Standards

Equivalencies to, and exemptions taken to codes and standards are provided in Section 7.7.3. The Authority Having Jurisdiction for licensed activities required to meet 10 CFR 70 is the NRC. Issuance of the license requested by this License Application constitutes approval of exemptions and equivalencies described in the application. NFPA recognizes that there is no one “right way” to meet the goal or intent of a code. Therefore, an alternate method providing an equivalent level of safety (i.e., performance-based design and/or documented analysis) may be used to satisfy the provisions of the code or standard. These alternate approaches are referred to as equivalencies. Equivalencies are deemed in compliance with the NFPA code and must be maintained as records for NRC inspection. Exemptions (deviations) to NFPA codes, however, require submittal to NRC for approval.

7.7.2 Design Basis for Fire Protection Structures, Systems and Components (SSCs)

The following criteria apply to fire protection SSCs:

- A fire protection program will be established at the MFFF.
- Management measures will be implemented at the MFFF to monitor and maintain the performance of the MFFF fire protection systems.
- Fire area barriers in the MOX Fuel Fabrication Building will have a minimum fire rating of two hours; fire area barriers rated at less than 2 hours are justified in the Fire Hazards Analysis.
- Following the injection of clean agent, the supply ventilation dampers to the room are automatically closed by the fire detection and alarm control system. The exhaust ventilation to the room is automatically secured following the injection of clean agent except in rooms containing dispersible radioactive materials.
- Sprinkler systems are designed in accordance with the applicable requirements of NFPA 13-1996.
- The MFFF fire main and fire water supplies are sized in accordance with the applicable requirements of NFPA 13-1996, 14-1996, and 24-1995.
- Flammable and combustible liquids are stored and handled in accordance with the applicable requirements of NFPA 30-1996.
- The emergency response team for fires at the MFFF is the SRS FD. The SRS FD is a full time professional fire department trained and qualified to fight MFFF fires.
- The MFFF fire detection and alarm system is designed in accordance with the applicable requirements of NFPA 72-1996.

- MFFF buildings containing IROFS are designed as Type I construction in accordance with the applicable requirements of NFPA 220-1995.
- The fire loading in each fire area of the MFFF considers transient combustible material.
- As a matter of the fire strategy for the MFFF, all fire doors in the facility remain closed until they are used to transfer personnel or material. Upon the detection of a fire on either side of fire doors that are an intrinsic part of the glovebox processes, the fire detection and alarm system automatically notifies the controlling normal programmable logic controller (NPLC) to remove the permissive to open the door and give the close command to doors that may happen to be open at the time of fire detection. The controlling NPLC has automated software routines to deal with an obstruction in a door that is commanded closed by the fire detection system. Upon clearing an obstruction, the close command is given to complete door closure. These processes are fully automatic and implemented by the fire detection system in concert with the controlling NPLC and its software.
- Fire dampers located in the ventilation exhaust of process rooms in areas with dispersible radioactive material are manually closed. Ventilation supply fire dampers are automatic.
- MFFF buildings containing IROFS are designed as Type I construction in accordance with NFPA 220-1995.
- Process room ventilation exhaust will normally remain open during a fire unless the inlet temperature at the final filter boxes exceeds the HEPA filter maximum design temperature. The High Depressurization Exhaust System (HDE) will ensure that any potential releases caused by a fire are filtered or will continue to filter other C3 rooms should the affected process room be isolated. Intermediate filters with fire screens are provided at each common glove box exhaust header for each process room containing glove boxes to protect the VHD final HEPA filters.
- The final Medium Depressurization Exhaust (MDE), HDE, Process Cell Exhaust (POE) and Very High Depressurization Exhaust (VHD) HEPA filters are qualified for the maximum temperature loading anticipated to result from credible fires within the MOX Fuel Fabrication Building.
- The final MDE, HDE, POE and VHD HEPA filters are qualified to maintain design flow for the maximum soot loading and maximum differential pressure anticipated to result from credible fires within the MOX Fuel Fabrication Building. See Chapter 11 for additional information on heating, ventilating, air conditioning and confinement systems.
- Compensatory measures (e.g., fire watches) are implemented as appropriate in accordance with procedures when IROFS fire protection features and systems (i.e., Quality Level [QL]-1 fire barriers, fire doors, fire dampers, fire-rated penetration seals, fire suppression systems, and fire detection systems) are not operable.
- MFFF's fire hazards analysis is reviewed and updated periodically, as necessary, following changes and modifications to the facility, processes, or inventories in accordance with MOX Services' configuration management process (Chapter 15). MOX Services will perform analyses using guidance of Regulatory Guide 1.189 that consider

the improper operation of IROFS equipment due to spurious signals induced by fire damage.

Fire barriers are used to separate IROFS and to separate areas that contain materials and processes that contain fire hazards into fire areas. The major components of fire barriers are walls, floors, ceilings, doors, ventilation dampers, fire propagation barriers, fire wrap, and fire-rated penetration seals. The passive boundaries of the fire areas are fire-rated for a minimum of 2 hours; fire area barriers rated at less than 2 hours are justified in the Fire Hazards Analysis.

The MFFF fire detection and alarm system monitors the facility for a fire condition. Upon detection of a fire, this system initiates, as appropriate, the following actions; (1) provides appropriate alarms locally and in the various control rooms, (2) provides control signals to various plant equipment related to the shutdown of transfer of flammable fluids (NFPA 221 considerations), (3) provides signals to fire suppression sequencing panels to coordinate fire suppression release actions with HVAC damper action, (4) provides signals to close appropriate automated fire doors, and (5) provides signals to miscellaneous equipment such as elevators. These automated actions occur individually or in combination as appropriate to the fire area and the circumstance. The fire detection and alarm system is designed in accordance with NFPA 72 and NFPA 221.

The fire suppression systems provide extinguishing agents appropriate to areas served. Fire suppression systems consist of water-based suppression systems (preaction, wet-pipe and deluge), carbon dioxide systems, clean agent systems (halogenated and non-halogenated), standpipe systems and portable fire extinguishers.

7.7.3 Codes and Standards for Fire Protection

- ASTM E-119-1995, Fire Tests of Building Materials
- NFPA 50A-1999, Standard for Gaseous Hydrogen Systems at Consumer Sites
- NFPA 51B-1999, Standard for Fire Prevention During Welding, Cutting, and Other Hot Work
- NFPA 55-1998, Standard for the Storage, Use, and Handling of Compressed and Liquefied Gases in Portable Cylinders
- NFPA 70-1999, National Electrical Code

  NFPA 70 Chapter 9, Table 1 allows a maximum of 40% fill for conductors where there are over 2 conductors installed in a conduit. There exists several underground ducts where the fill is at 41%.

  NFPA 70 Article 384-15 restricts the number of over current devices to 42 in a single panel. Due to space consideration, it is necessary to use panels with more than 42 circuits.

  NFPA 70 limits power and control cable fills to 30% and instrument fill to 40%. The project has increased these limits to 40% and 50%, respectively.

NFPA 70 Article 362-5 states that a wireway shall not be filled more than 20 percent of its cross-sectional area. At MFFF, trays and wireways containing power cable with random fill are designed for a 40 percent maximum capacity. Control or instrumentation trays and wireways are designed for a 50 percent maximum capacity.

- NFPA 72-1996, National Fire Alarm Code

  Where there is an accessible room that is devoid of combustibles, the room can be made inaccessible by locking its access door(s) and controlling the access to the room. Based on the requirements of Section 5-1.4.2 of NFPA 72, rooms that are inaccessible and devoid of combustible materials do not require fire detection. This approach has been taken for some MFFF rooms.

- NFPA 80-1999, Standard for Fire Doors and Fire Windows

  Some fire doors are not UL listed or FM approved, and therefore these doors will not be labeled. These fire doors include the revolving fire doors of the Jar Storage and Handling (NTM) unit and the cut-off fire doors of the Pellet Handling (PML) unit. The use of fire doors that are not UL listed or FM approved deviates from the guidance of Section 1-6.1 of NFPA 80, which requires fire doors be labeled. The use of these fire doors has been justified by equivalency analysis.

- NFPA 80A-1996, Recommended Practice for Protection of Buildings from Exterior Fire Exposures
- NFPA 221-1997, Standard for Fire Walls and Fire Barriers
- NFPA 801-1998, Standard for Fire Protection for Facilities Handling Radioactive Materials

  Polycarbonate glovebox windows, which are a "combustible" as defined in NFPA 801, have been selected for use in MFFF process gloveboxes. NFPA 801, Section 5-4.4.1 requires the use of noncombustible materials in the construction of gloveboxes. However, as allowed by NFPA 801, an equivalent level of fire protection is achieved for this material due to difficulty of polycarbonate to ignite or sustain combustion, absence of significant ignition sources, and other fire protection features implemented.

  NFPA 801, Section 3-9.1 requires the design of the MFFF ventilation systems to be in accordance with NFPA 90A; and NFPA 90A, Section 2-3.8 requires MFFF fire dampers to be listed. However, the fire dampers in the BMP, BAP, and BSR will not be UL listed or FM approved because these dampers have been modified from a UL listed or FM approved design to meet MFFF design requirements. Although not UL listed or FM approved, these modified fire dampers are equivalent to UL listed/ FM approved fire dampers because they are tested in accordance with UL 555, which is required in order for a fire damper to be UL listed/FM approved.

- NFPA 2001-2004, Standard on Clean Agent Extinguishing Systems

  As an NFPA 2001 equivalency, the discharge of the non-halogenated clean agent in rooms with gloveboxes is designed to release over an extended period to avoid overpressurization and compromising confinement within these rooms. To confirm the effectiveness of the minimum design concentrations provided in NFPA 2001 to extinguish a fire when the agent is released over an extended period, the clean agent was tested in accordance with Section 36 of UL 2127, "Inert Gas Clean Agent Extinguishing System Units," 1st edition revised through March 22, 2001 using the materials listed in UL 2127 and additional combustibles to represent those found in significant quantities in the MFFF. The results of these tests indicate that to extinguish all the materials tests, the minimum design concentration must be 20% greater than the values provided in NFPA 2001. For conservatism, the concentration of clean agent is maintained in the area of fire origin after the non-halogenated clean agent has been fully discharged.

- Regulatory Guide 3.16, General Fire Protection Guide for Plutonium Processing and Fuel Fabrication Plants, January 1974
- UL 555 (1995 edition), Fire Dampers

  Fire dampers installed in the BMF deviate from UL 555 Fifth Edition as follows:

  Actuation temperatures are greater than or equal to 375 degree Fahrenheit, in lieu of maximum code actuation temperature of 286 degree Fahrenheit. Elevated temperatures were required to support fire analysis.

  Fire endurance and hose stream testing was accomplished in the horizontal position during the fire test and vertically for the hose stream test, in lieu of conducting each test in the vertical and horizontal positions. This is the bounding condition for vertical and horizontal installed fire dampers.

  Vertical position fire dampers were tested with the jack shaft side facing the furnace and the jack shaft side was subjected to the hose stream test, in lieu of testing two assemblies with one jack shaft side facing the furnace and the other facing out from the furnace. This is the bounding condition of the two orientations.

  Three-hour fire test was conducted in gypsum board wall in lieu of concrete wall. Wall has no effect on testing and gypsum is allowed in later editions of UL 555.

  Temperature degradation testing commenced at 350 degree Fahrenheit in lieu of 250 degree Fahrenheit.

  Chain and motor actuators are tested with the actuators not subjected to the elevated temperatures. The actuators are outside of the fire area of concern during a fire event in which the dampers must be closed.

  The code requires fire damper sleeves to extend specific lengths beyond wall or floor openings. Fire dampers installed in the BMF in through-wall penetrations were designed and manufactured based on a given wall thickness. Through design evolution and construction tolerances, the out-of-wall section of the fire damper sleeve may exceed code requirements.

- Underwriters Laboratory listed and/or Factory Mutual approved

The following national codes and standards are used for selection of electrical equipment in gloveboxes to assure that the risk of electricity as an ignition source of fires and explosions is minimized:

- Occupational Safety and Health Administration (OSHA) 29 CFR 1910 (2004), Occupational Safety and Health Standards
- UL 508 (1993 edition), Industrial Control Equipment

The cable insulation within the gloveboxes meets the flame requirements from one or more of the following standards:

- IEEE 383 (1974 Reaffirmed 1992), Standard for Qualifying Class 1E Electric Cables and Field Splices for Nuclear Power Generating Stations or code that meets equivalent flame requirements.

- NFPA 262-2002, Standard Method of Test for Flame Travel and Smoke of Wires and Cables for Use in Air-Handling Spaces
- IEEE 1202-1991, -1996, Standard for Flame Testing of Cables for Use in Cable Tray in Industrial and Commercial Occupancies
- UL-910 – 1998, Standard for Safety Test for Flame-Propagation and Smoke-Density Values for Electrical and Optical Fiber Cables used in Spaces Transporting Environmental Air
- UL-1666-2002, Test for Flame Propagation Height of Electrical and Optical-Fiber Cables Installed Vertically in Shafts
- UL-1685-1997, Standard for Safety Vertical-Tray Fire-Propagation and Smoke Release Test for Electrical and Optical-Fiber Cables
- FT4 of CSA C22.2 No. 0.3-92 Para 4.11.4, Test Methods for Electrical Wires and Cables (Acceptable Cable Flame Test)
- FT6 of CSA C22.2 No. 0.3-92 Appendix B, Test Methods for Electrical Wires and Cables (Acceptable Cable Flame Test)
- IEC 60332-3, Tests on Electric Cables under Fire Conditions Part 3: Tests on Bunched Wires or Cables (1992) (Acceptable Cable Flame Test)
- ICEA T-29-520, Conducting Vertical Cable Tray Flame Tests with Theoretical Heat Input Rate of 210,000 B.T.U/Hour (Acceptable Cable Flame Test
- ASTM D2633, Standard Test Methods for Thermoplastic Insulating and Jackets for Wire and Cable

Table 7-1. Table Deleted

8.0 CHEMICAL SAFETY

Chemical safety for the Mixed Oxide (MOX) Fuel Fabrication Facility (MFFF) has two main aspects. The first aspect of chemical safety is control of the chemical hazards that apply to the chemicals that do not interact with licensed materials and do not impact the safety of licensed materials. For this set of chemical hazards, which is not regulated by the U.S. Nuclear Regulatory Commission (NRC), and hence outside the scope of this license application, CB&I AREVA MOX Services, LLC (MOX Services) has established and maintains a safety program that includes protection against industrial chemical hazards. The second aspect of chemical safety for the MFFF is control of chemical hazards of licensed material, hazardous chemicals produced from licensed material, and plant conditions impacting the safety of licensed material (resulting in an increased radiological risk).

This chapter describes the chemical hazard identification process, process chemistry and potential interactions, chemical hazards analysis methodology, and chemical process safety interfaces with programmatic areas and management measures.

The integrated safety analysis (ISA) includes identification and evaluation of chemical hazards that may impact radiological safety and chemical hazards directly associated with NRC-licensed radioactive material. The ISA process includes designation of items relied on for safety (IROFS) to provide adequate protection against chemical risks from licensed material, facility conditions that affect the safety of licensed material, and hazardous chemicals produced from licensed material. The ISA identifies the controls—both engineered and administrative IROFS—that are used either to prevent the occurrence of chemical-related accidents, or to mitigate the consequences of potential accidents to acceptable levels. MOX Services maintains continuity of control over IROFS during design, construction, and operations by implementing the MFFF configuration management processes. This control extends to chemical safety, which is an integral component of the ISA process. Chapter 5 describes MOX Services' programmatic commitments for the conduct and maintenance of the ISA, and summarizes the ISA methodology, including chemical safety aspects.

An overview of the MFFF processes is provided in Chapter 1. Further descriptions of the chemical processes are provided in Chapter 11 and evaluated as part of the ISA.

8.1 CHEMICAL INFORMATION

A wide variety of chemical products, several of which are hazardous, are used in the Aqueous Polishing (AP) and MOX Process (MP) processes at the MFFF. Table 8.1-1 lists the hazardous characteristics and incompatibilities associated with these chemicals. Of the chemicals used in the AP and MP processes, at least 20 exhibit one of the following hazardous characteristics:

- Corrosivity
- Flammability
- Explosivity

- Chemical burn
- Toxicity.

Table 8.1-2 through Table 8.1-5 list the process chemicals used at the MFFF by location, with chemical formula, chemical state, and Central Abstract System Registry Number (CASRN). In addition, two chemicals (uranium dioxide and uranyl nitrate) that are stored in the Secured Warehouse Building (BSW) are listed in Table 8.1-6.

Numerous reagents are used in the AP process, with a summary description of these reagent systems provided in Chapter 11. These reagent systems are designed such that segregation/separation of vessels/components from incompatible chemicals is assured to prevent chemical explosions under normal, off-normal, and accident conditions including earthquakes. Rigid control of the chemical makeup of the reagents introduced into the cells or AP reagent rooms prevents explosions caused by chemical reactions. Chemicals, piping, tanks, and other components in the Nitric Acid (RNA) system are clearly labeled to prevent reagent preparation errors. Safety precautions are used in handling reagents, in accordance with Material Safety Data Sheet requirements.

The following reagents support the AP process functions:

- Nitric Acid
- Tributyl Phosphate
- Hydroxylamine Nitrate
- Hydrazine
- Sodium Hydroxide
- Oxalic Acid
- Hydrogenated Polypropylene Tetramer (diluent)
- Sodium Carbonate
- Dinitrogen Tetroxide
- Hydrogen Peroxide
- Manganese Nitrate
- Aluminum Nitrate
- Zirconium Nitrate
- Silver Nitrate
- Sodium Sulfite
- Sodium Nitrite
- Uranyl Nitrate.

Table 8.1-7 provides a list of these reagents along with the downstream transfer unit, and the normal operating range.

In addition, the following reagents will be used in either the MP process or as oxygen scavengers in the steam and condensate system:

- Zinc stearate
- Azodicarbonamide
- Carbohydrazide
- Morpholine.

Zinc stearate is a lubricant used in the MP process. It is packed in small ready-to-use plastic bags that are manually introduced into the relevant powder process glovebox to be mixed with powders in process. The bags are introduced into the gloveboxes via a glove port using a "bag-in bag-out" procedure.

Azodicarbonamide is a poreformer used in the MP process. It is also packed in small ready-to-use plastic bags that are manually introduced into the relevant powder process glovebox to be mixed with powders in process. The bags are introduced into the gloveboxes via a glove port using a "bag-in bag-out" procedure.

Carbohydrazide and Morpholine are used as oxygen scavengers in the steam and condensate (SPS) system (see Section 11.8). Carbohydrazide will be purchased as a solid, while morpholine will be delivered as a liquid.

Table 8.1-1. Process Chemical Hazardous Characteristics and Incompatibilities

Chemical	Chemical Formula	Corrosivity	Flammability	Explosivity	Chemical Burn	Toxicity	Incompatibilities
Aluminum Nitrate	$Al(NO_3)_3$	x			x	x	Flammable and combustible materials, strong reducing agents, finely powdered metals, water and strong acids. Strong reducing agents such as hydrogen peroxide and nitrates may be incompatible under certain conditions.
Argon	Ar						None (noble gas).
Butanol	$C_4H_{10}O$		x	x	x	x	Strong acids, strong oxidizing agents
Butyl nitrate	$C_4H_9NO_3$	x	x	x	x	x	Strong acids, sodium hydroxide, reducing agents.
Carbohydrazide	CH_6N_4O	x	x	x	x	x	Oxidizing Agents.
Dibutyl phosphate	$C_8H_{19}NO_4P$		x		x	x	Strong oxidizing agent.
Diluent (Dodecane isomer mix), HPT	$C_{12}H_{26}$ (mixture)		x	x		x	Vapors can form an explosive mix with air. Nitric acid may be incompatible due to potential to form short-chain hydrocarbons and nitro compounds (i.e. potential fire hazards) under specific conditions. Oxidizing agents, oxygen.
Dinitrogen Tetroxide	N_2O_4	x		x	x	x	Water, bases, flammable and combustible materials, hydrocarbons, copper, aluminum, chrome. Explosion may occur on contact with ammonia, boron trichloride, carbon disulfide, cyclohexane, fluorine, formaldehyde, nitrobenzene, toluene, propylene, alcohols and ozone. Reducing agents and metals.
Helium	He						None (noble gas).
Hydrazoic Acid	HN_3	x	x	x	x	x	Unstable. Violently explosive in the concentrated or pure states. Readily forms explosive compounds with heavy metals. This is a dangerous and hard to handle substance which must not be prepared or handled by non-experts.
Hydrazine	N_2H_4	x	x	x	x	x	Hydrogen peroxide, nitric acid and other oxidants. May react violently with water to emit toxic gasses. Oxidizing agent, metal, asbestos.
Hydrogen	H_2		x	x			None.

Table 8.1-1. Process Chemical Hazardous Characteristics and Incompatibilities

Chemical	Chemical Formula	Corrosivity	Flammability	Explosivity	Chemical Burn	Toxicity	Incompatibilities
Hydrogen peroxide	H_2O_2		x	x	x	x	Copper, chromium, most metals or their salts, flammable liquids and other combustible materials, aniline and nitromethane, hydrazine, sodium carbonate, hydroxylamine nitrate, and metal salts. Organics, nitric acid, manganese.
Hydroxylamine Nitrate (HAN)	NH_2OH HNO_3	x		x	x	x	Stable under normal conditions. If hydroxylamine nitrate solutions are concentrated and heated, there will be a decrease in stability. Evaporation in a stainless steel tank had been reported to have led to an autocatalytic decomposition which led to the rapid release of gas that resulted in an explosion in a confined vessel. Alkaline materials leading to a pH of 5 or greater. Strong oxidizers and strong reducing agents should be tested for compatibility before use. Combustible materials. If hydroxylamine nitrate solutions are allowed to dry on combustible materials such as paper or wood, the material may ignite. Iron and other metals have been reported to lead to the decomposition of hydroxylamine nitrate solutions. Decomposition may lead to a very rapid gas release. Combustible material. nitric Acid, hydrogen peroxide, bichromate and permanganate of potassium, copper sulfate, zinc.
Manganese Nitrate	$Mn(NO_3)_2$	x			x	x	Strong reducing agents, combustible materials, aluminum nitrate, N_2O_4, hydrogen peroxide.
Methane-Argon Mixture (P10)	CH_4 (10%) - Ar (90%)						None.
Morpholine borane	$C_4H_{12}BNO$	x	x	x	x	x	Sodium hydroxide, oxidizing agents.
Nitric acid (13.6N)	HNO_3	x			x	x	Organics. Acetic, chromic and hydrocyanic acids, aniline, carbon, hydrogen sulfide, fluids or gases and substances that are readily nitrated. Can be explosive with organic materials. Corrodes most metals, hydrogen peroxide, Hydroxylamine, hydrazine and sodium carbonate, sodium hydroxide, sodium sulfite and oxalic acid.
Nitrogen Dioxide	NO_2					x	Hydrocarbon, metals, reducing agents.

Table 8.1-1. Process Chemical Hazardous Characteristics and Incompatibilities

Chemical	Chemical Formula	Corrosivity	Flammability	Explosivity	Chemical Burn	Toxicity	Incompatibilities
Oxalic Acid Dihydrate (also present as solid)	$H_2C_2O_4 \cdot 2H_2O$				x	x	Vigorous reaction with strong bases, alkali metals, acid chlorides and ferric salts. Explosive reaction with silver and mercury. Nitric acid, silver, sodium chloride, sodium hypochlorite reacts with sulfuric acid to form carbon monoxide, solid is hygroscopic.
Oxygen	O_2		x			x	Organics.
Pre-mixed Argon-Hydrogen	Ar (95%) – H_2 (5%)						None.
Silver Nitrate	$AgNO_3$	x			x	x	Ammonia, alkalis, antimony salts, arsenites, bromides, carbonates, chlorides, iodides, thiocyanates, ferrous salts, phosphates, tannic acid and tartrates, nitric acid aluminum nitrate, alcohols, magnesium, strong bases, strong reducing agents (Note: solid is air, moisture and light sensitive), oxalic acid.
Sodium Carbonate	Na_2CO_3					x	Violent reaction with phosphorus pentoxide, sulphuric acid, and fluorine. Ignites magnesium and is explosive with hot aluminum. Hydrogen peroxide and acids (e.g., nitric acid reactions).
Sodium Hydroxide	NaOH	x			x	x	Can react violently with water and ignite combustible materials. Explosive reaction with nitro- and chloro- compounds and some metals. Acid, aluminum, organic halogens (especially trichloroethylene) and sugars.
Sodium Nitrite	$NaNO_2$				x	x	Ammonium nitrate and other ammonium salts, strong acids and reducing agents, organic materials, cyanides, cellulose, sodium thiosulfate, sodium bisulfite, acetanilide, chlorates, hypophosphites, iodides, mercury salts, permanganates, sulfites and tannic acid. Combustible materials.
Sodium Sulfite	Na_2SO_3				x	x	Forms acids with water and steam. Contact with strong oxidizers such as perchloric acid causes vigorous exothermic reactions. Contact with acids will release sulphur dioxide. Organics, combustible materials. (Note: solid is air and moisture sensitive).
Tri-Butyl Phosphate (TBP)	$(C_4H_9)_3PO_4$		x	x	x	x	Strong oxidizers and strong bases. Avoid wet alkaline conditions, especially when the material is heated, because TBP undergoes hydrolysis to produce butyl alcohol and alkyl phosphoric acid salts. Zirconium nitrate, ammonia, oxidizing agents.
Zirconium Nitrate	$Zr(NO_3)_4$			x		x	Strong bases or reducing agents, TBP and aluminum nitrate.

Table 8.1-2. Process Chemicals in the Reagents Processing Building (BRP)

CHEMICAL			
Name	**Formula**	**CASRN**	**State**
Diluent, HTP (C10-C13 Isoalkanes)	$C_{12}H_{26}$ (mixture)	68551-17-7	Liquid
Hydrazine	N_2H_4	302-01-2	Liquid
Hydrogen Peroxide	H_2O_2	7722-84-1	Liquid
Hydroxylamine Nitrate (HAN)	$NH_2OH\text{-}HNO_3$	13465-08-2	Liquid
Nitric Acid	HNO_3	7697-37-2	Liquid
Nitrogen Dioxide (Note 1)	NO_2	10102-44-0	Gas
Dinitrogen tetroxide	N_2O_4	10544-72-6	Liquid/Gas
Oxalic Acid	$H_2C_2O_4$	144-62-7	Solid/Liquid
Sodium Carbonate	Na_2CO_3	497-19-8	Liquid
Sodium Hydroxide	NaOH	1310-73-2	Liquid
Sodium Sulfite	Na_2SO_3	7757-83-7	Liquid
Tributyl Phosphate (TBP)	$(C_4H_9)_3PO_4$	126-73-8	Liquid
Zirconium Nitrate	$Zr(NO_3)_4$	13746-89-9	Liquid

Notes:

1. Nitrogen dioxide is the coexisting monomer of dinitrogen tetroxide in gas form.

Table 8.1-3. Process Chemicals in the Aqueous Polishing Building (BAP)

CHEMICAL			
Name	**Formula**	**CASRN**	**State**
Aluminum Nitrate	$Al\ (NO_3)_3$	13473-90-0	Liquid
Butanol (Note 3)	$C_4H_{10}O$	71-36-3	Liquid
Butyl nitrate (Note 3)	$C_4H_9NO_3$	928-45-0	Liquid
Chlorine (Note 1)	Cl_2	7782-50-5	Gas
Dibutyl phosphate (Note 3)	$C_8H_{19}NO_4P$	107-66-4	Liquid
Diluent, HPT (C10-C13 Isoalkanes)	$C_{12}H_{26}$ (mixture)	68551-17-7	Liquid
Hydrazine (0.1 N)	N_2H_4	302-01-2	Liquid
Hydrazoic Acid	HN_3	7782-79-8	Liquid
Hydrogen Peroxide	H_2O_2	7722-84-1	Liquid
Hydroxylamine Nitrate (HAN)	$NH_2OH\text{-}HNO_3$	13465-08-2	Liquid
Manganese Nitrate	$Mn(NO_3)_2$	10377-66-9	Liquid
Nitric Acid	HNO_3	7697-37-2	Liquid
Nitric Oxide (Note 1)	NO	10102-43-9	Gas
Nitrogen	N_2	7727-37-9	Gas
Nitrogen Dioxide	NO_2	10102-44-0	Gas
Nitrogen Oxides (Note 1)	NO_x	N/A	Gas
Oxalic Acid	$H_2C_2O_4$	144-62-7	Liquid
Oxygen	O_2	N/A	Gas
Plutonium Dioxide	PuO_2	N/A	Solid
Plutonium Oxalate (Note 2)	$Pu(C_2O_4)_2$	N/A	Solid/Liquid
Plutonium Nitrate (Note 2)	$Pu(NO_3)_4$	N/A	Liquid
Silver Nitrate	$AgNO_3$	7761-88-8	Liquid
Sodium Carbonate	Na_2CO_3	497-19-8	Liquid
Sodium Hydroxide	NaOH	1310-73-2	Liquid
Sodium Nitrite	$NaNO_3$	7632-00-0	Liquid
Sodium Sulfite	Na_2SO_3	7757-83-7	Liquid
Tributyl Phosphate (TBP)	$(C_4H_9)_3PO_4$	126-73-8	Liquid
Uranyl Nitrate	$UO_2(NO_3)_2$	36478-76-9	Liquid
Zirconium Nitrate	$Zr(NO_3)_4$	13746-89-9	Liquid

Notes:

1. Chlorine and nitrogen oxides are by-products of AP processing.
2. Plutonium oxalate and plutonium nitrate are intermediate products of AP processing.
3. Butanol, dibutyl phosphate, monobutyl phosphate and butyl nitrate are byproducts of TBP degradation.

Table 8.1-4. Process Chemicals in the MOX Processing Building (BMP)

CHEMICAL			
Name	**Formula**	**CASRN**	**State**
Argon-Hydrogen	95% Ar; 5% H	N/A	Gas
Azodicarbonamide (poreformer)	$H_2NCONNCONH_2$	123-77-3	Solid
Carbohydrazide	CH_6N_4O	497-18-7	Solid
Helium	He	7440-59-7	Gas
Isopropanol	C_3H_7OH	67-63-0	Liquid
Morpholine borane	$C_4H_{12}BNO$	4856-95-5	Solid
Nitrogen	N_2	7727-37-9	Gas
Plutonium Dioxide	PuO_2	N/A	Solid
Uranium Dioxide	UO_2	1344-57-6	Solid
Zinc Stearate	$Zn(C_{18}H_{35}O_2)_2$	557-05-1	Solid

Table 8.1-5. Process Chemicals in the Laboratories

CHEMICAL			
Name	**Formula**	**CASRN**	**State**
Aluminum Nitrate	$Al (NO_3)_3$	13473-90-0	Liquid
Acetonitrile	C_2H_3N	75-05-8	Liquid
Ammonium bi-fluoride	F_2H_5N	1341-49-7	Liquid
Argon	Ar	N/A	Gas
Argon-Hydrogen	95% Ar; 5% H	N/A	Gas
Argon-Methane (P10)	90% Ar; 10% CH_4	N/A	Gas
Ascorbic Acid	$C_6H_8O_6$	50-81-7	Liquid
Chromic (VI) Acid	CrO_3	7738-94-5	Liquid
Ethanol	C_2H_6O	64-17-5	Liquid
Ethylene Glycol	$C_2H_6O_2$	107-21-1	Liquid
Ferrous sulfate	$FeSO_4$	7720-78-7	Liquid
Helium	He	N/A	Gas
Hydrofluoric Acid	HF	7664-39-3	Liquid
Hydrogenated Propylene Tetramer (diluent)	$C_{12}H_{26}$ (mixture)	68551-17-7	Liquid
Hydroxylamine Nitrate	NH_2OH-HNO_3	13465-08-2	Liquid
Liquid Nitrogen	N_2	N/A	Liquid
Methanol	CH_4O	67-56-1	Liquid
Nitric Acid	HNO_3	7697-37-2	Liquid
Nitrogen	N_2	7727-37-9	Gas
Nitrogen/Helium (70%/30%)	N_2/He	N/A	Gas
Oxygen	O_2	N/A	Gas
Silver Nitrate	$AgNO_3$	7761-88-8	Liquid
Silver Oxide	AgO	1301-96-8	Liquid
Sodium Carbonate	Na_2CO_3	497-19-8	Liquid
Sodium Hydroxide	NaOH	1310-73-2	Liquid
Sodium Nitrite	$NaNO_2$	7632-00-0	Liquid
Sulfamic Acid	HSO_3NH_2	5329-14-6	Liquid
Tetrahexyl Ammonium Bromide	$C_{24}H_{52}BrN$	12124-97-9	Liquid
Tributyl Phosphate (TBP)	$C_{12}H_{27}O_4P$	126-73-8	Liquid

Table 8.1-6. Chemicals in the Secured Warehouse Building (BSW)

CHEMICAL			
Name	**Formula**	**CASRN**	**State**
Uranium Dioxide	UO_2	1344-57-6	Solid
Uranyl Nitrate	$UO_2(NO_3)_2$	36478-76-9	Liquid

Table 8.1-7 Reagents Used in the AP Process

Chemical Name	Reagent Formula	Downstream Transfer Unit of Concern	Normal Operating Range
RNA Nitric Acid	HNO_3	KDB KDD KCA KCD KPA KPC LGF	13.6 N
		KDD KDB KCD LGF	6 N
		KDD KDB KPA KWD KCD KPG KPB KPC	1.5 N
RHN Hydroxylamine Nitrate	$[NH_3OH][NO_3]$	KPA	0.4, 0.78 M
Hydrazine (with HAN)	$[NH_3NH_2]$	KPA	0.1 M
RTP Tributyl Phosphate	$(C_4H_9O)_3PO$	KPB	99 wt%
		KPA	30 wt%
RDO Diluent Hydrogenated Polypropylene Tetramer (HPT)	hydrocarbon mixture of C_{10}, C_{11}, C_{12} and C_{13} isomers	KPA KPB	70 wt%
RHP Hydrogen Peroxide	H_2O_2	KDB KDD	10 wt%
		RHN	5 wt%
RSN Silver nitrate	$AgNO_3$	KDB KDD	3.0 M solution in 3.0 N HNO_3
RSC Sodium Carbonate	Na_2CO_3	KPB	0.3 M
RSH	NaOH	KDD	10 N

Table 8.1-7 Reagents Used in the AP Process

Chemical Name	Reagent Formula	Downstream Transfer Unit of Concern	Normal Operating Range
Sodium Hydroxide		KWD RNA	
		KPB	0.1 N
RSS Sodium Sulfite	Na_2SO_3	KDD	0.5 M
ROA Oxalic Acid	$H_2C_2O_4$	KCA	0.05-0.7 M solution in 2.0 N HNO_3
RZN Zirconium nitrate	$Zr(NO_3)_4$	KPA KPC	10 g/L solution in 3.5 N HNO_3
GNO Nitrogen Tetroxide	Mixture of N_2O_4, NO_2	KPA	1.85 Nm^3/hr 4.1 Nm^3/hr
RUN Uranyl Nitrate	$UO_2(NO_3)_2$	KDB KDD KPA	200 g U/L solution in 1.0 N HNO_3
RMN Manganese Nitrate	$Mn(NO_3)_2$	KCA	0.01 M solution in 13.6 N HNO_3
RAN Aluminum Nitrate	$Al(NO_3)_3$	KPA	1 g Al/L solution in 1.5 N HNO_3
RSI Sodium Nitrite	$NaNO_2$	KWD	400 g/L

8.2 CHEMICAL PROCESS INFORMATION

This section discusses the evaluation of potential chemical interactions to identify those chemicals that cannot be mixed under specified conditions and those mixtures that could create a safety hazard (e.g., a fire or explosion). Potential adverse reactions between the reagents used in the AP process are examined. In addition, interactions between the reagents and actinides of plutonium and uranium are examined to identify possible hazards related to colloids formation, polymerization of plutonium, precipitate formation, or explosion. Furthermore, interactions between the reagents and water are assessed, as well as with reactive species that are generated in situ during the AP process. Finally, interactions between the reagents used in the AP process and those used in the MP process or as oxygen scavengers, is investigated for possible hazards.

The chemical processes that take place as a part of normal operations at the MFFF are described in Chapter 11 of this License Application, with further descriptions and evaluations developed in the ISA.

In the chemical conditions encountered in the PUREX-based process at the reference La Hague facility, chemical incompatibilities between the reagents have been mitigated or prevented through the control of process parameters. Table 8.2-1 presents a chemical interaction matrix created as a part of the ISA to assess the chemical compatibility/incompatibility of the reagents that are postulated to be mixed either by failure of operations or equipment within the AP process itself, or an inadvertent mixing by a technician in the MFFF laboratories. A similar chemical interaction matrix was created to assess the possible chemical reactions between the reagents and actinides [U(VI), Pu(III), Pu(IV) and Pu(VI)] and water as depicted in Table 8.2-2. Table 8.2-3 displays the compatibility of the reagents present in the AP process with reactive chemical species that may be generated in situ during the AP process. Finally, the chemical interaction for the reagents that are used in the MOX process and steam supply unit can be found in Table 8.2-4.

The ISA includes an analysis of the potential for explosions and the IROFS that are required to prevent these events. In addition, events involving chemical releases, alone or in combination with radioactive releases, are evaluated. IROFS are identified to protect against these chemical risks at the MFFF.

Normal process conditions allow interactions of some chemicals listed as incompatible in Table 8.1-1, provided that process parameters are controlled to allow safe operating conditions. For chemicals listed in Table 8.1-1 as incompatible that are mixed under normal process conditions, the conditions that are controlled as necessary to maintain safe operating conditions are provided below:

- Aluminum nitrate – The low concentrations of aluminum nitrate (1 g/L) used in the AP process are compatible with nitric acid.
- Diluent – For the chemical incompatibility between the diluent and oxygen, Section 8.3.3.6 of the LA discusses IROFS controls to ensure that either (1) all vapor compositions in the AP process are maintained at or below 60% of the LFL; or (2) by application of purge gas, effluent gas remains out of the flammability range throughout the system being protected.

- Dinitrogen tetroxide (equivalent entry for nitrogen dioxide) – Concentration and NOx flow controls in KPA*CLMN6000 allow controlled (i.e., intentional) HAN and hydrazine destruction by NOx, with adequate venting capacity provided for off gases that are generated.
- Hydrazine – Concentration and temperature controls ensure that hydrazine and nitric acid can coexist with minimal reactivity such that safe operating conditions are maintained.
- Hydrogen peroxide - Concentration controls in KDD/KDB*TK3000 allow for the safe use of hydrogen peroxide to reduce Ag(II) to Ag(I) and Pu(VI) to Pu(IV) in the presence of nitric acid.
- HAN - Concentration and temperature controls are employed to limit the depletion of HAN by nitric acid to prevent a HAN autocatalytic reaction.
- Nitric acid – The concentration of degradation products from reactions of organic compounds such as TBP and HPT by nitric acid is limited to safe levels during normal process conditions by processing spent solvent through the solvent recovery unit KPB, which removes solvent degradation compounds from the AP process.
- Oxalic acid – Manganese nitrate is used to catalyze the destruction of oxalic acid by nitric acid in the KCD Unit.
- Silver nitrate – Silver nitrate is stable in acidic solutions in the absence of strong reducing agents, which are not used in AP process vessels that contain silver nitrate. The reduction of silver nitrate to silver metal by hydrogen peroxide is extremely slow and poses no safety hazard in KDB*TK3000, KDD*TK3000, and KDD*TK4000.
- Sodium carbonate – Adequate vent capacity for vessels KPB*MIXS1000 and KWD*TK4015 allow for safe neutralization reactions with nitric acid.
- Sodium hydroxide – Adequate vent capacity for vessels KPB*MIXS1000 and KWD*TK4015 allow for safe neutralization reactions with nitric acid.
- Sodium nitrite – Flow controls of nitric acid into KWD*TK4015, combined with adequate vent capacity for this vessel, allows for safe acidification of sodium nitrite into nitrous acid for the purpose of azide destruction.
- Sodium sulfite – Flow control of sodium sulfite allows for controlled reduction of chlorine (Cl2) in KDD CLMN7000.
- TBP – The hydrolytic degradation of TBP in alkaline solutions used for solvent washing in the KPB unit is mitigated by the separation of TBP from the alkaline stream (i.e., contact between TBP and sodium hydroxide is limited by solvent separation).

8.2.1 Process and Reagent Sampling

IROFS sampling has been identified as an enhanced administrative control for the prevention of explosion, loss of confinement, and criticality events. IROFS sampling combines administrative and active engineered controls to perform the following functional elements:

- Isolation of AP process and reagent tanks, and powder vessels and containers, prevents inadvertent transfer from a tank/vessel/container being sampled. Supporting features include IROFS valves, hand switches, and level monitors.
- Homogenization of AP process tanks and powder samples. Supporting features include mixing jets/spargers, valves, level and flow monitors, a Sampling Plan, and a Quality Assurance Plan.
- Automatic sampling of AP process tanks and powder samples, and manual sample extraction of a select group of powder samples and AP process and reagent tanks, portable containers and drip trays.
- Identification and tracking of all samples (including sample receipt, labeling/marking of vials and transfer within the laboratory). Supporting features include vials labeled with indelible ink, bar code readers, position switches, pneumatic transfer systems for vials (MMIS and LIMS), and a Sampling Plan.
- Laboratory analyses and communication of sampling results. Supporting features include analytical procedures, analytical equipment and technique, analytical limits, record keeping, and reporting of analytical results to Operations.

The safety function of IROFS sampling is to ensure that: (1) reagents are correctly identified upon receipt and are introduced into the process at their proper concentrations; (2) process solutions remain within the correct composition; and (3) the contents of drip trays are identified (as necessary) and appropriately recovered.

For quality control, operating procedures will be developed in an administrative IROFS Sampling Plan which will include information for training personnel and analysts, establishing the technical bases for the analytical methods used in the laboratory, and documenting the processes used in the analyses. The Sampling Plan will be developed under the MOX Project Quality Assurance Plan and will include provisions for the following:

- Reagent tank isolation
- Downstream/upstream AP process tank isolation
- Homogenization of sampled tank
- Automatic sampling of AP process tanks
- Manual sampling of AP process tanks or reagent tanks
- Manual sampling of portable containers
- Manual sampling of drip trays
- Powder sampling
- Analytical techniques
- Communication of results

Table 8.2-1. through Table 8.2-4 Withheld from Public Disclosure Under 10CFR2.390

8.3 CHEMICAL HAZARDS ANALYSIS

8.3.1 Consequence Analysis Methodology

As a part of the hazard assessment process performed as a part of the ISA, potential accident events are identified and evaluated that could result in acute chemical exposure from licensed material or hazardous chemicals produced from licensed material. The materials-at-risk (MAR) used to assess the radiological consequences are provided by event type and event group in Table 8.3-1. These MAR values produce the bounding consequence for an event group and are the maximum inventory associated with a particular location (e.g., fire area or process vessel) where an event can credibly occur. For chemicals, Table 8.3-2 provides the maximum inventory used in the chemical consequence analysis for each of the chemicals evaluated. The baseline design criteria require (Title 10 of the Code of Federal Regulations (CFR) §70.64(a)(5)) that the design provide for adequate protection against chemical risks produced from licensed material, facility conditions that affect the safety of licensed material, and hazardous chemicals produced from licensed material. This section describes the methodology for the evaluation of the chemical consequences associated with a release of hazardous chemicals.

8.3.1.1 Quantitative Standards for Chemical Consequences Levels

Chemical concentration limits are required to be established to evaluate the potential consequences to the individual outside of the controlled area (IOC) and to workers for an accidental release of chemicals. Three levels, High (H), Intermediate (I), and Low (L), based on 10 CFR §70.61, are used to define these limits.

A high consequence event is one that results in any of the following:

- An intake of 30 mg or greater of uranium in soluble form by an individual located outside the controlled area;
- An acute chemical exposure to an individual from licensed material or hazardous chemicals produced from licensed material that could endanger the life of a worker;
- An acute chemical exposure to an individual from licensed material or hazardous chemicals produced from licensed material that could lead to irreversible or other serious, long-lasting health effects to an individual located outside the controlled area.

An intermediate consequence event is one that results in any of the following:

- An acute chemical exposure to an individual from licensed material or hazardous chemicals produced from licensed material that could lead to irreversible or other serious, long-lasting health effects to a worker;
- An acute chemical exposure to an individual from licensed material or hazardous chemicals produced from licensed material that could cause mild transient health effects to an individual located outside the controlled area.

Quantitative standards are required to correctly categorize exposures per the qualitative criteria established in 10 CFR §70.61. Limits are based on Acute Exposure Guideline Level (AEGL) values and Emergency Response Planning Guideline (ERPG) values. However, since AEGL and ERPG values are not established for MFFF chemicals, Temporary Emergency Exposure Limits (TEELs) have been adopted for use in chemical consequence analysis for those chemicals where AEGL or ERPG values have not been established. TEELs were adopted by the DOE Subcommittee on Consequence Assessment and Protective Action (SCAPA). The SCAPA-approved methodology was used to obtain hierarchy-derived TEELs.

The original TEEL methodology used only hierarchies of published concentration limits (that is, Permissible Exposure Levels [PELs] or Threshold Limit Values – Time-Weighted Averages [TLV-TWAs], Short-Term Exposure Levels [STELs], and Immediately Dangerous to Life and Health [IDLH] values) to provide estimated values approximating ERPGs. The expanded method for deriving TEELs also includes published toxicity data (Toxic Dose Low $[TD]_{LO}$, Toxic Concentration Low $[TC]_{LO}$, 50% Lethal Dose $[LD]_{50}$, 50% Lethal Concentration $[LC]_{50}$, Lethal Dose Low $[LD]_{LO}$, and Lethal Concentration Low $[LC]_{LO}$). Hierarchy-based values take precedence over toxicity-based values, and human toxicity data are preferred to animal toxicity data. Subsequently, default assumptions based on statistical correlation of ERPGs at different levels (for example, ratios of ERPG-3s to ERPG-2s) were used to calculate TEELs where there were gaps in the data. The TEEL hierarchy/toxicity methodology was used to develop community exposure limits for over 1,200 chemicals to date. The following are the TEEL definitions:

- TEEL-0 – The threshold concentration below which most people will experience no appreciable risk of health effects.
- TEEL-1 – The maximum concentration in air below which it is believed nearly all individuals could be exposed without experiencing other than mild transient adverse health effects or perceiving a clearly defined objectionable odor.
- TEEL-2 – The maximum concentration in air below which it is believed nearly all individuals could be exposed without experiencing or developing irreversible or other serious health effects or symptoms that could impair their abilities to take protective action.
- TEEL-3 – The maximum concentration in air below which it is believed nearly all individuals could be exposed without experiencing or developing life-threatening health effects.

The definitions of TEEL severity levels are consistent with 10 CFR §70.61. See Table 8.3-3 for a listing of the TEEL and ERPG values for MFFF chemicals. ERPG values are noted in the table.

For uranium accidents, intakes are used instead of concentration-based TEELs to establish consequence categories. An event that results in an intake of 30 mg soluble or a respirable intake of 30 mg of insoluble uranium may be considered to lead to irreversible or other serious, long-lasting health effects to an individual. An intake of 10 mg of soluble uranium or a respirable intake of 10 mg of insoluble uranium may be considered to cause mild transient health effects (Hartmann, Heidi M., Frederick A. Monette, and Halil I. Avci, "Overview of Toxicity Data and Risk Assessment Methods for Evaluating the Chemical Effects of Depleted Uranium

Compounds", *Human and Ecological Risk Assessment, Vol. 6, No. 5,* 2000). Hence, controls are applied to uranium events if the potential intake of soluble uranium or respirable intake of insoluble uranium exceeds 10 mg to the IOC or 30 mg to a worker.

The chemical consequence categories used to define the level of risk are provided in Table 8.3-4.

8.3.1.2 Chemical Event Release Scenarios

The chemical consequences for the facility worker, site worker, and IOC are assessed for events identified in the hazard evaluation as part of the ISA. The facility worker is considered to be located inside the MFFF, near a potential accident. When evaluating chemical consequences for the site worker and the IOC, two release points in the MFFF are evaluated: (1) the MFFF building stack, and (2) the Secured Warehouse Building (BSW). The site worker is considered to be 100 m from the release point. Both facility workers and site workers are deemed to be "workers." The IOC is defined as the maximally exposed individual outside the controlled area boundary, either 68 m (for BSW releases) or 160 m (for MFFF building stack releases) from the release point.

A range of initial conditions was considered to identify the physical processes that control the nature and rate of vapor generation and release. Failure modes of storage containers and associated systems were also considered. The following release scenarios were addressed:

- Leaks and ruptures involving equipment vessels and piping leaks
- Evaporating pools formed by spills and tank failures
- Flashing and evaporating liquefied gases from pressurized storage.

The associated explosion events that could result in the release of hazardous chemical vapors are evaluated in the ISA. The chemical consequences are based on bounding analyses.

8.3.1.3 Atmospheric Dispersion

Chemical releases are conservatively modeled for the site worker and the IOC using a 0- to 2-hour 95th percentile atmospheric dispersion factor (χ/Q). The ARCON96 computer code is used to compute the downwind relative air concentrations (χ/Q) for the site located within 100 m of a ground-level release from the MFFF to account for low wind meander and building wake effects, and for the IOC located at either 68 m or 160 m from the release point. The 0- to 2-hour atmospheric dispersion factor (χ/Q) for ground-level releases to the site worker at 100 m is 6.1×10^{-4} sec/m^3. For the IOC, the 0- to 2-hour atmospheric dispersion factor (χ/Q) for ground-level releases is (1) 1.25×10^{-3} sec/m^3 at 68 m from the release point, and (2) 2.5×10^{-4} sec/m^3 at 160 m.

8.3.1.4 Chemical Consequences

Facility worker consequences are qualitatively determined based on the material released, the release mechanism, and the location of the worker relative to the release. In most cases, events involving an airborne release of plutonium or americium are judged to have high consequences to the facility worker and IROFS are already applied. In lieu of a mechanistic calculation of the release, a conservative bounding release model was used to determine the consequences to the site worker and IOC from releases either from the BSW, or the MFFF building stack, as applicable. Releases were modeled to occur using the total material at risk from the largest single tank or container. Furthermore, no credit was afforded to process equipment installed to remove/scrub some of the potentially released chemicals prior to release from the MFFF.

Estimates of hazardous chemical concentrations include techniques, assumptions, and models that are consistent with industry practice, were verified and/or validated, and follow the guidance on atmospheric and consequence modeling found in NUREG/CR-6410, *Nuclear Fuel Cycle Accident Analysis Handbook*.

The chemical consequence analyses were performed assuming the largest credible unmitigated spill or loss of containment accident involving these chemicals. Airborne concentrations were calculated at distances correlating to the site worker (100 m) and the IOC (either 68 m or 160 m). These concentrations were then compared to the chemical limits presented in Table 8.3-3. From this comparison, a consequence category was established (low, intermediate, high) using the guidance outlined in Table 8.3-4. These consequence categories correspond to those identified in 10 CFR §70.61.

8.3.2 Description of Explosion Events Safety Strategy

8.3.2.1 Sintering Furnace Hydrogen Explosions (EXP01)

The safety strategy for sintering furnace hydrogen explosions involves the application of IROFS to meet the performance criteria of 10 CFR § 70.61. Sintering furnace hydrogen explosions are considered in the Pellets Processing Area where hydrogen explosions may occur in the high temperatures of the furnaces that sinter mixed oxide fuel pellets.

IROFS controls are provided to ensure the detection and prevention of potential leaks of hydrogen into the process room and the isolation of the Ar-H supply prior to reaching explosive conditions.

8.3.2.2 Steam Explosions (EXP02)

The safety strategy for steam explosions involves the application of IROFS to meet the performance criteria of 10 CFR § 70.61. Steam explosions/overpressurizations are postulated to occur within the sintering furnace due to overfill of the humidifier system or within the humidifier itself due to the introduction of water onto a hot humidifier heater. Also, loss of cooling water to the LFT laboratory furnace can result in the water in the cooling water jackets flashing to steam (i.e., steam pressure transient).

IROFS controls are provided to ensure the flow of demineralized water to the sintering furnace is restricted and the Ar-H supply is isolated on detection of high water level in the humidifier mixer drain tank. IROFS pressure relief valves on the LFT laboratory furnace chiller minimize the potential for an internal glovebox pressure transient.

8.3.2.3 Radiolysis Explosions (EXP03)

The safety strategy for radiolysis explosions involves the application of IROFS to meet the performance criteria of 10 CFR § 70.61. Radiolysis explosions are postulated in AP vessels where fluids are exposed to radiation fields from radionuclides such as plutonium and americium, generating potentially explosive quantities of hydrogen. The following parameters are important relative to maintaining process safety:

- quantity and isotopic composition of radionuclides (plutonium and/or americium) – identified as the Material at Risk (MAR)
- composition of the fluid (organic versus aqueous)
- solution acidity (lower pH is more susceptible to radiolysis)

There are three cases for consideration for this explosion event:

- vessels not requiring scavenging air
- vessels requiring scavenging air
- waste containers

For the case of vessels not requiring scavenging air, IROFS controls are provided to limit the fluid content and MAR to maintain hydrogen concentrations at or below 25% of its lower flammable limit (LFL).

For the case of vessels requiring scavenging air, IROFS controls are provided to limit the MAR content and ensure a qualified air supply is hard-piped to the requisite AP process vessels to maintain hydrogen concentrations at or below 25% of its lower flammable limit (LFL).

For the case of waste containers, the waste container design allows for release of Hydrogen making radiolysis explosion events highly unlikely.

8.3.2.4 HAN Explosions (EXP04)

The safety strategy for HAN explosions involves the application of IROFS to meet the performance criteria of 10 CFR § 70.61. HAN explosions are considered in AP process vessels where HAN may be present with nitric acid. The following parameters are important relative to maintaining process safety:

- HAN concentration (HAN/nitric acid system stability decreases with decreasing HAN concentration for systems with constant nitric acid concentration; however, the total potential energy and offgas release from autocatalytic reaction is also reduced with lower HAN concentrations)
- Nitric acid concentration (HAN/nitric acid system stability decreases with increasing nitric acid concentration above 2 N; below 2 N the influence of acidity is more complex in the presence of an organic phase)

- Process temperature (HAN/nitric acid system stability decreases with increasing temperature)
- Room (cell) temperature (affects the ability to transfer heat due to exothermic chemical reactions from process vessels to the surroundings; higher room temperatures are less favorable for heat transfer)
- Flow rate of HAN-bearing streams (affects the consumption rate of HAN, which in turn affects both the resulting temperatures and off-gas rates. Low or no flow conditions can lead to depletion of HAN and lower concentrations where the autocatalytic reaction dominates)
- Plutonium concentration (process solution temperature and offgas release increases with increasing plutonium concentration)

IROFS controls are provided to prevent process deviations from initiating HAN autocatalytic reactions that may generate explosive gases.

8.3.2.5 Hydrogen Peroxide Explosions (EXP05)

The safety strategy for hydrogen peroxide explosions involves the application of IROFS to meet the performance criteria of 10 CFR § 70.61. Hydrogen peroxide explosions are considered in AP process vessels (e.g., KDD/KDB) where hydrogen peroxide may be present with substrates that can be oxidized or reduced. The following parameters are important relative to maintaining process safety:

- Sequence, rate, and quantity of hydrogen peroxide addition
- Flow controls for substrates that can be oxidized or reduced
- Process vessel off-gas venting capacity

IROFS controls are provided to prevent hydrogen peroxide reactions from generating explosive gases.

8.3.2.6 Solvent Explosions (EXP06)

The safety strategy for solvent explosions involves the application of IROFS to meet the performance criteria of 10 CFR § 70.61. Solvent explosions are postulated in AP vessels where solvent may exceed flammability safety limits. The following may induce a solvent explosion event:

- process temperature events involving TBP and/or HTP
- room temperature events involving TBP and/or HTP
- fire events affecting TBP and/or HTP
- events involving transfer of solvent to heated vessels

- events involving lowering the lower flammability limit of the solvent

In order to ensure that solvent explosions remain highly unlikely, the MFFF facility deploys the following strategies depending upon the location and operation associated with each vessel:

- all vapor compositions in the AP process are maintained at or below 60% of the LFL

IROFS controls are provided to prevent process deviations from generating explosions and/or exceeding the solvent flammability limit.

8.3.2.7 TBP-Nitrate (Red Oil) Explosions (EXP07)

The safety strategy for TBP-nitrate (red oil) explosions involves the application of IROFS to meet the performance criteria of 10 CFR § 70.61. Red oil explosions are postulated in AP due to the possible commingling of TBP and nitric acid. In order to ensure that red oil explosions remain highly unlikely, the MFFF facility deploys a three-pronged safety strategy (heat transfer, evaporative cooling, and TBP prevention) that includes the following measures:

- Temperature controls < 130°C
- Pressure controls (venting)
- Mass controls (separate phase TBP)
- Nitric acid concentration < 10 N where separate phase TBP may be present

In this preventative safety strategy, IROFS controls ensure that: 1) heating of ambient temperature (< 80°C) solutions containing nitric acid and TBP is prevented to the extent that it can contribute to runaway conditions (heat transfer); and 2) TBP is prevented from heated vessels above a mass limit to allow safe venting of off-gases (TBP prevention and/or evaporative cooling).

8.3.2.8 AP Vessel Over-Pressurization (EXP08)

The safety strategy for AP vessel over-pressurization explosions involves the application of IROFS to meet the performance criteria of 10 CFR § 70.61. Vessel over-pressurization events may occur in AP vessels with high levels or where process conditions can generate off-gases. The causes for vessel over-pressurization include:

- pressurized utilities supplied to process vessels
- heating of fluids and vapor space
- introduction of excessive quantities of fluids to high temperature environments
- loss of the KWG vent flow path
- chemical reactions that produce gaseous products
- overfilling process vessels which leads to excessive static head beyond pressure rating of vessel

IROFS controls are provided to prevent over-pressurization of AP process vessels.

8.3.2.9 Pressure Vessel Explosions (EXP09)

The safety strategy for pressure vessel explosions involves prevention of the event by design to meet the performance criteria of 10 CFR § 70.61. Pressure vessel explosions are postulated to occur from failure of an auxiliary pressure vessel or compressed gas bottle.

No IROFS have been identified for this event as the auxiliary system pressure vessels and the compressed gas bottles are designed not to fail in a manner that could affect the functionality of the IROFS.

8.3.2.10 Hydrazoic Acid Explosions (EXP10)

The safety strategy for hydrazoic acid explosions involves the application of IROFS to meet the performance criteria of 10 CFR § 70.61. Hydrazoic acid explosions are postulated in AP due to the formation of hydrazoic acid from the oxidation of hydrazine.

The following parameters are important relative to maintaining process safety:

- Limiting hydrazine concentrations
- Temperature controls on process solutions
- Preventing hydrazoic acid from high temperature vessels
- Employing oxidation reactions (e.g., with NOx) that destroy hydrazoic acid

IROFS controls are provided to limit the amount of hydrazoic acid that can form, control the temperature of process fluids, prevent hydrazoic acid from entering high temperature vessels, ensure hydrazoic acid destruction in appropriate vessels, and ensure that azides in alkaline solution do not come into contact with concentrated nitric acid.

8.3.2.11 Metal Azides Explosions (EXP11)

The safety strategy for metal azide explosions involves the application of IROFS to meet the performance criteria of 10 CFR § 70.61. Metal azide explosions are postulated in AP due to the presence of hydrazoic acid and metals.

The following parameters are important relative to maintaining process safety:

- Metal ion concentration
- Formation of azide ions from hydrazoic acid
- Solution pH and the presence of substrates that influence acid/base chemistry
- Redox reactions that affect metal oxidation state
- Metal azide solubility and/or solution evaporation

IROFS controls are provided to limit the formation and potential precipitation of metal azides or promote downstream destruction of hydrazoic acid.

8.3.2.12 Pu(VI) Oxalate Explosions (EXP12)

Pu(VI) oxalate explosions in the KCA calcining furnace have been demonstrated to be not credible. Therefore, no IROFS controls are required to prevent their occurrence.

8.3.2.13 Electrolyzer Related Hydrogen Explosions (EXP13)

The safety strategy for electrolyzer related hydrogen explosions involves the application of IROFS to meet the performance criteria of 10 CFR § 70.61. Electrolyzer related hydrogen explosions are postulated to occur from hydrogen (H_2) that may be generated either (a) electrochemically at the cathode of the electrolyzer, or (b) due to the presence of metal (e.g., Al, Be) in the PuO_2 feed.

The following parameters are important relative to maintaining process safety:

- Acid normality (lower acidity increases the risk of electrolytic hydrogen formation)
- Catholyte solution flow
- Hydrogen concentration in the off-gas

IROFS controls are provided to prevent the electrochemical formation of hydrogen at the cathode of the electrolyzer and to prevent the accumulation of explosive hydrogen generated due to the presence of Al and/or Be metal in the PuO_2 feed.

8.3.2.14 Laboratory Explosions (EXP14)

The safety strategy for laboratory explosions involves the application of IROFS to meet the performance criteria of 10 CFR § 70.61. Laboratory explosions are postulated to occur from two possible events:

- a hydrogen event
- an event produced by an unintended chemical reaction

For a hydrogen explosion event, IROFS controls ensure that the introduction of argon, Ar-H gas mixture, and air into the LAC unit furnace is controlled.

For an event produced by an unintended chemical reaction, IROFS controls ensure that an explosion is prevented.

8.3.2.15 Outside Explosions (EXP15)

The safety strategy for outside explosions involves the application of IROFS to meet the performance criteria of 10 CFR § 70.61. Explosion events occurring outside of the MOX Fuel

Fabrication Building (BMF) that could potentially impact MFFF operations or safety support systems are postulated to occur on the MFFF site in the following specific areas:

- Reagent Processing Building (BRP)
- MFFF Gas Storage Area (UGS)
- MFFF Site Roadways.

IROFS controls mitigate the consequences of external explosions and associated missiles by providing robust heavily reinforced concrete structures in the case of the BMF, BEG, and UEF structures, as well as providing buried doubled-walled KWD high alpha waste transfer lines. In addition, the chemical safety controls program is a preventive feature that controls the receipt, storage and handling of chemicals delivered and used by the MFFF process.

8.3.2.16 Miscellaneous Explosions (EXP16)

The safety strategy for miscellaneous explosions involves prevention of the event by design to meet the performance criteria of 10 CFR § 70.61. Within the BAP, the BMP, the BEG, and the BSR, there are some potential explosion hazards that either do not directly involve radiological material or involve only trace quantities of radiological material. However these hazards may have the potential to damage nearby IROFS (principally fire area boundaries). The following areas have been identified as having or creating potential explosion hazards that could indirectly result in the unacceptable release of radiological material from the MFFF by damaging IROFS:

- The solvent/diluent reagent room in the BAP, room C-513
- The HAN/hydrazine/nitric acid reagent room in the BAP, room C-444
- The reagent room where leaks and overflows from the previous two rooms are collected in the BAP, room C-209
- The two reagent pipe chases connecting in the BAP, rooms C-104 and C-108
- The KWS vessels room in the BAP, room C-138
- The gas pad where the nitrogen system interfaces with the hydrogen system
- The emergency battery rooms in the BSR, rooms D-214 and D-217
- The battery rooms in the BSR, rooms D-202 and D-210
- The electrical equipment rooms A and B in the BEG, rooms F-101 and F-103.
- BAP room C-435 containing the KDD scrubbing column

Trace quantities of radiological material may exist in the following rooms listed above: C-513 (in the KPB recovered solvent feeding pot and the associated pipes and pumps), C-138 (in the KWS final waste organic solvent tank and associated pipes and pumps), and C-435 (in KDD filters within KDD*GB7000).

No IROFS are identified for these event scenarios because no radiological material is directly involved with the event, nor are the trace quantities of radiological material that may be present sufficient enough to exceed the performance criteria of 10 CFR §70.61.

8.3.2.17 Perchlorate Explosions (EXP17)

Perchlorate explosions have been demonstrated to be not credible. Therefore, no IROFS controls are required to prevent their occurrence.

Table 8.3-1. through Table 8.3-2 Withheld from Public Disclosure Under 10CFR2.390

Table 8.3-3. TEELs (mg/m^3) Used as Chemical Limits for Chemicals at the MFFF (Note 1)

Name	TEEL-1	TEEL-2	TEEL-3
Acetonitrile	100	100	750
Aluminum Nitrate	15	15	500
Argon	350,000	500,000	750,000
Ascorbic Acid	200	500	500
Azodicarbonamide	125	500	500
Boric Acid	30	50	125
Dry cement (i.e., calcium carbonate)	15	15	15
Calcium Nitrate	3.5	25	125
Chromic (VI) Acid	1	2.5	25
Chlorine*	3	7.5	60
Diluent (C10-C13 Isoalkanes) (Note 2)	5	35	200
• Decane (C10)	5	35	25000
• Undecane (C11)	6	40	200
• Dodecane (C12)	15	100	750
• Tridecane (C13)	60	400	500
Ethanol	500	3,500	15,000
Ethylene glycol	50	100	150
Ferrous sulfamate	3	5	25
Ferrous sulfate	7.5	12.5	350
Fluorine*	0.75	7.5	30
Hydrazine*	0.7	6.6	40
Hydrazine Nitrate	3	5	5

Table 8.3-3. TEELs (mg/m^3) Used as Chemical Limits for Chemicals at the MFFF (Note 1) (continued)

Name	TEEL-1	TEEL-2	TEEL-3
Hydrofluoric Acid*	1.5	15	40
Hydrochloric Acid*	4	30	200
Hydrogen Peroxide*	12.5	60	125
Hydroxylamine Nitrate	15	26	125
Iron	30	50	500
Isopropanol	1000	1000	5000
Manganese	3	5	500
Manganese Nitrate	10	15	500
Manganous Sulfate	7.5	12.5	500
Methanol*	262	1308	6540
Nitric Acid*	2.5	15	200
Nitric Oxide	30	30	125
Nitrogen Dioxide	7.5	7.5	35
Dinitrogen Tetroxide	15	15	75
Oxalic Acid	2	5	500
Potassium Hydroxide	2	2	150
Potassium Iodide	0.75	6	300
Potassium Nitrate	3.5	20	500
Potassium Permanganate	7.5	15	125
Silver Nitrate	0.03	0.05	10
Silver Oxide	30	50	75
Sodium Acetate	30	500	500
Sodium Carbonate	30	50	500
Sodium Hydroxide*	0.5	5	50

Table 8.3-3. TEELs (mg/m^3) Used as Chemical Limits for Chemicals at the MFFF (Note 1) (continued)

Name	TEEL-1	TEEL-2	TEEL-3
Sodium Nitrate	1	7.5	100
Sodium Nitrite	0.125	1	60
Sodium Oxalate	30	50	50
Sodium Sulfite	30	50	100
Sulfuric Acid*	2	10	30
Sulfamic Acid	40	250	500
Thenoyl TrifluoroAcetone	3.5	25	125
Tributyl Phosphate	6	10	300
Xylene	600	750	4000
Zinc Stearate	30	50	400
Zirconium nitrate	35	35	50

* Values are based on Emergency Response Planning Guideline (ERPG) concentrations.

Notes:

1. Temporary Emergency Exposure Limits (TEELs), Revision 18, are derived from approved methodologies developed by Department of Energy Subcommittee on Consequence Assessment & Protective Actions (SCAPA) and are identified in WSMS-SAE-02-0001.
2. The TEEL values for diluent represent the most conservative value in each category among the following primary constituents: n-decane, n-undecane, n-dodecane, and n-tridecane.

Table 8.3-4. Application of Chemical Limits to Qualitative Chemical Consequence Categories

Consequence Category	Worker	IOC
High	Concentration ≥ TEEL-3	Concentration ≥ TEEL-2 Soluble uranium intake ≥ 30 mg Insoluble uranium respirable intake ≥ 30 mg
Intermediate	TEEL-3 > Concentration ≥ TEEL-2 Soluble uranium intake ≥ 30 mg Insoluble uranium respirable intake ≥ 30 mg	TEEL-2 > Concentration ≥ TEEL-1 30 mg > Soluble uranium intake ≥ 10 mg 30 mg > Insoluble uranium respirable intake ≥ 10 mg
Low	TEEL-2 > Concentration Soluble uranium intake < 30 mg Insoluble uranium respirable intake < 30 mg	TEEL-1 > Concentration Soluble uranium intake < 10 mg Insoluble uranium respirable intake < 10 mg

Notes:

1. Temporary Emergency Exposure Limits (TEELs) are derived from approved methodologies developed by Department of Energy Subcommittee on Consequence Assessment & Protective Actions (SCAPA) as identified in WSMS-SAE-02-0001, Revision 18, and listed in Table 5.1-3.
2. Intakes are used instead of concentration-based TEELs to establish consequence categories for uranium accidents.

8.4 ORGANIZATIONAL STRUCTURE

MOX Services key management functions with responsibilities for IROFS and related activities are described in Chapter 4. These IROFS include those established by the ISA to protect against chemical risks from licensed material, facility conditions that affect the safety of licensed material, and hazardous chemicals produced from licensed material. Responsibility for performing and maintaining the ISA is described in Chapter 5.

8.5 CHEMICAL PROCESS SAFETY INTERFACES

Aspects of MFFF chemical process safety have interfaces with the following programmatic areas and management measures:

- Human factors engineering
- Emergency management
- Quality assurance
- Configuration management
- Maintenance
- Training and qualification
- Plant procedures
- Audits and assessments
- Incident investigations
- Records management.

8.5.1 Interfaces with Programmatic Areas

MOX Services applies criteria for human factors engineering to the design of MFFF IROFS with associated personnel activities for operation or maintenance (i.e., the scope of human factors engineering is associated with IROFS, whose function is protection against radiological, chemical, and criticality hazards). The MFFF is a highly automated facility based in large part on the design and operating experience of existing facilities. The highly automated nature of the facility limits the number of personnel activities designated IROFS. The application of human factors engineering to MFFF IROFS is described in Chapter 12.

As described in Chapter 14, an emergency plan is not required to be submitted.

8.5.2 Interfaces with Management Measures

Management measures supplement MFFF IROFS by providing the administrative and programmatic framework for configuration management, maintenance, training and qualification, procedures, audits and assessments, incident investigation, and records management. The MOX Project QA Plan (MPQAP) and management measures are described in Chapter 15.

Personnel responsible for performing activities involving chemical safety are qualified and trained in accordance with the MFFF training and qualification program, specifically, applicable training for IROFS associated with chemical hazards. A general discussion of qualification and training of personnel is provided in Chapter 15.

Activities associated with IROFS are conducted in accordance with approved procedures. MFFF plant procedures govern operations, maintenance, and administrative actions to ensure that IROFS are operated in a manner consistent with the results of the ISA. Plant procedures associated with items relied on for chemical safety take into account chemical hazards, as well as radiological and criticality hazards, as appropriate for the activity. A general discussion of procedures is provided in Chapter 15.

Audits and assessments are used to determine the effectiveness of management measures, including those associated with chemical safety. Audit and assessment attributes (e.g., independence of auditors from personnel responsible for the chemical safety activities being audited, reports to management) are consistent with those for other MFFF IROFS. A general discussion of the audit and assessment program is provided in Chapter 15, with a more detailed description given in the MPQAP.

Incident investigation activities identify corrective actions for, and root causes of, incidents that involve MFFF IROFS, including those related to chemical safety. A general discussion of the incident investigation /corrective action implementation is provided in Chapter 15, with a more detailed description given in the MPQAP.

Chemical safety records are controlled in accordance with configuration management processes, the requirements of the MPQAP, and the records management program. Chemical safety records are processed and retained in the same manner as records associated with other IROFS and related programs, as described in Chapter 15.

8.6 DESIGN BASIS

The following information represents the design basis attributes for chemical safety.

A Sampling Plan will be developed which will address reagent tank isolation, downstream/upstream AP process tank isolation, homogenization of sampled tank, automatic sampling of AP process tanks, manual sampling of AP process tanks or reagent tanks, manual sampling of portable containers, manual sampling of drip trays, powder sampling, analytical techniques, and communication of results.

The chemical consequence analysis uses maximum chemical inventories.

Table 8.3-3 provides a listing of TEEL value limits for MFFF Chemicals.

Chemical releases are conservatively modeled for the site worker and the IOC using a 0- to 2-hour 95th percentile atmospheric dispersion factor (χ/Q).

The chemical consequence analyses were performed assuming the largest credible unmitigated spill or loss of containment accident involving these chemicals.

Airborne concentrations were calculated at distances correlating to the site worker (100 m) and the IOC (either 68 m or 160 m).

The safety strategy for TBP-nitrate (red oil) explosions involves the application of the following three strategies:

- Heat transfer strategy
- Evaporative cooling strategy
- TBP prevention strategy

The safety strategy for HAN explosions involves the following parameters relative to maintaining process safety:

- HAN concentration
- Nitric acid concentration
- Process temperature
- Room (cell) temperature
- Flow rate of HAN-bearing streams
- Plutonium concentration

9.0 RADIATION SAFETY

The radiological protection program provides assurance that facility radiation safety measures protect the health and safety of workers and comply with the regulatory requirements of Title 10 of the Code of Federal Regulations (CFR) Part 20, *Standards for Protection Against Radiation,* and 10 CFR Part 70, *Domestic Licensing of Special Nuclear Material* during routine and nonroutine operations, including anticipated events. Public and environmental radiation protection is addressed in Chapter 10.

Facility management will implement a quality radiation protection program consistent with Regulatory Guide 8.8 C.1.a & b. The program is supported throughout the facility lifetime and it is documented in project documents.

The potential for occupational exposure at the Mixed Oxide (MOX) Fuel Fabrication Facility (MFFF) exists primarily as a result of processing plutonium (i.e., potential internal exposure from inhalation) and secondarily as a result of proximity to photon and neutron radiation sources (i.e., direct external exposure). The primary design features that limit exposure in accordance with as low as is reasonably achievable (ALARA) goals are automated and remote systems operation, confinement systems (e.g., gloveboxes, process cells, and ventilation), monitoring, alarms, and radiation shielding.

The radiological protection program applies to MFFF activities that manage radiation and radioactive materials, and that may potentially result in radiation exposure to facility workers and the individual outside of the controlled area (IOC). The radiological protection program guides the actions of personnel involved in radiological work at the MFFF.

9.1 RADIATION SAFETY DESIGN FEATURES

The MFFF design objectives, along with the programmatic measures, ensure that operation of the MFFF is in accordance with 10 CFR Parts 20 and 70, and ALARA principles. Occupational doses are ALARA through engineering design features and management controls implemented during operation.

9.1.1 ALARA Design Considerations

9.1.1.1 Responsibilities for ALARA Design

The design function is split between the regulatory and engineering functions. The nuclear safety function within the regulatory function provides design criteria associated with radiation protection. The nuclear safety function reviews the MFFF designs for radiation safety concerns including criticality control and potential radiation exposure as well as shielding considerations. These reviews provide the MFFF designers with the information necessary to ensure that operational exposures are maintained ALARA as a result of the design.

The manager of the radiological protection function is part of the design review process and evaluates the MFFF system and structural design to ensure that ALARA principles are incorporated. His review evaluates potential radiological concerns so that they can be mitigated during the design process to provide for adequate radiation protection of personnel during facility operations, including maintenance activities.

The manager of the engineering function is responsible for implementation of radiation protection design criteria. Facility design engineers report to the manager of the engineering function. The nuclear safety function reviews the design, performs radiation protection analyses, and confirms that the design meets radiation protection design criteria.

Design personnel are qualified in radiation protection design and ALARA concepts, including personnel experienced in radiation protection, radiation shielding, radiation monitoring and general radiation safety. Design personnel are trained to recognize potential radiation hazards and to minimize the effects of these hazards on operations.

The primary radiation analyses performed in support of the radiation protection design are radiation shielding calculations and occupational radiation dose assessments during routine and nonroutine operations.

9.1.1.2 MFFF Design and Design Activities

The MFFF design reflects ALARA principles. Specific ALARA considerations in the MFFF design include:

- Control of plutonium particulate to prevent inhalation by confining radioactive materials in process equipment and in gloveboxes
- Multiple-zone ventilation system design, sweeping from low to high potential contamination zones
- Continuous remote monitoring for airborne contamination in accessible areas with local and remote readout and alarm functions
- Use of automated and remotely operated equipment to minimize personnel exposure
- Provisions for removing radioactive material before most maintenance operations are included in facility maintenance procedures
- Shielding between radioactive sources and operators, according to the intensity, nature, and penetrating power of the radiation
- Design of structures, systems, and components (SSCs) that require a minimum of maintenance or repair, to minimize personnel stay time in radiation areas
- Shield wall penetrations between high radiation areas and personnel access areas are located and oriented so that there is no direct line of sight to the source(s), thus precluding streaming without reduction due to scatter
- Placement of piping containing radioactive fluids in nonaccessible pipe chases
- Placement of equipment requiring maintenance in separate shielded areas having a minimum of radioactive piping

- Placement of administrative, security, and radiation protection administrative activities away from radiation areas
- Areas of continuous occupancy are zoned to maintain dose rates at a low level while areas of higher dose rates are limited access.

9.1.1.3 Collective Dose Estimates

The design process includes an occupational dose assessment for the facility. Dose assessments are performed for each process unit with known personnel access requirements and are evaluated to determine reasonably achievable design enhancements to reduce exposures. Dose assessments are performed using guidance from U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 8.19, *Occupational Radiation Dose Assessment in Light-Water Reactor Power Plant — Design Stage Man-Rem Estimates*, and Regulatory Guide 8.34, *Monitoring Criteria and Methods to Calculate Occupational Radiation Doses*.

The dose assessments take into account both direct and internal dose. The direct dose assessment is determined by dose rate analyses and a dose assessment process called the ABAQUES Method (see Section 9.1.4.6). The internal dose assessment was determined based on the MFFF design and review of MELOX and La Hague experience. The internal dose and direct dose sum meet MFFF's design goals and are ALARA.

9.1.1.4 Design Review Process

Competent personnel are responsible for the review of, and concurrence on, preliminary and final designs. The design reviews incorporate experience from the MELOX and La Hague plants. Project design reviews include ALARA evaluations to a level of detail commensurate with the potential radiation hazard. Recommendations made in the ALARA evaluations are tracked to completion as part of the review of design products.

The MFFF design incorporates applicable radiation protection experience from MELOX and La Hague, such as the following:

- Descriptions of process unit operations
- Personnel access times
- Source configurations
- Radiation monitoring
- Radiation exposure problem areas
- ALARA design features and performance
- Contamination estimates
- Radiation monitoring design and operations
- Process unit shielding design
- Ventilation system design.

MELOX and La Hague are reference facilities for the MFFF design. Much of the MFFF facility design is the same as that used at the reference facilities. Occupational exposures at the MFFF facility should be similar to occupational exposure at the reference facilities, with adjustments to account for differences in radiation source terms, differences in shielding design, and personnel access requirements.

Radiation protection design improvements that have been made at the MELOX and La Hague facilities are incorporated into the MFFF facility design. For example, the grinding unit vacuum system minimizes loose contamination in the glovebox. Project team members have direct experience with the MELOX and La Hague facilities, and design documentation is available to the design team. Such improvements are incorporated to the maximum extent practical in the MFFF facility.

Continuing radiation safety (ALARA) design reviews for facility or process modifications are conducted during construction and operations. Environmental Safety & Health (ES&H) Licensing is responsible for reviewing facility or process modifications to maintain exposures ALARA.

9.1.1.5 Other Design Considerations

Experience from the MELOX and La Hague facilities is incorporated into the MFFF design to ensure that the occupational exposure from the MFFF is maintained ALARA. Airborne and loose surface contamination is prevented during normal operations by plutonium recovery operations, glovebox design, and ventilation system design, to maintain direct exposure and inhalation dose ALARA. Most of the aqueous polishing (AP) process is installed in process cells. Entry to those process cells is physically prevented.

Design features such as automation and remote controls reduce the time spent in radiation areas. MFFF zone classification (see Table 9.1-1) minimizes occupational radiation exposure through access control and shielding design to meet exposure criteria.

The design minimizes the distribution and retention of radioactive material throughout plant systems by:

- Designing the process equipment containing radioactive material to confine the material to the maximum extent practical to reduce glovebox contamination
- Designing the gloveboxes to prevent accumulation of contamination and allow easy access for cleaning
- Using a vacuum system in gloveboxes so that airborne dust is collected in dust pots and the radioactive material is recycled.

9.1.2 Facility Design Features

This section describes the primary design features and equipment that directly or indirectly reduce radiation exposure for facility workers and provide monitoring capability.

9.1.2.1 Drawings and Descriptions

Facility drawings, process descriptions, and other facility documents associated with the radiation protection design include:

- Scaled drawings of the general arrangement of the facility with superimposed radiation zones based on expected worker occupancy.
- Radiation shielding calculations that use design drawings for locations and configurations of radiation sources, shielding and plant operators in order to specify requirements for each process unit design.
- A summary report of radiation protection design that provides definitions of the radiation sources, dose rates, and worker dose estimates for process units. The report identifies features relied on to reduce doses to ALARA, and shows how the design meets the requirements of 10 CFR Part 20 during routine and nonroutine operations including anticipated events.
- Location for radiation protection equipment both for fixed detectors and for storage of portable equipment.
- General requirements and descriptions for radiation detectors and alarm systems
- Locations of permanent shielding and confinement design (e.g., penetrations, labyrinth seals, shield doors).
- Locations and access control points for radiation areas.
- The controlled area, including the means to limit access to the controlled area as necessary.
- The restricted area.
- Change rooms, showers, and locker rooms.
- Contamination control and waste minimization design features.

9.1.2.2 Radiation Sources and Exposure

The greatest potential for occupational radiation exposure at the MFFF is from plutonium inhalation. Therefore, the design incorporates multiple systems and barriers to prevent the release of radioactive material into personnel access areas. Depending on the stage in the process, confinement of radioactive material and worker protection is obtained by process vessels in cells (AP), gloveboxes (AP Sampling, Powder Area, and Pellet Process Area), or other sealed containers (fuel rods, containers). Gloveboxes are used to prevent personnel contamination. The gloveboxes are kept at a negative pressure with respect to the area occupied by personnel, to ensure that contamination will be contained in the event of a breach. A second ventilation system in the personnel access areas sends clean air through registers located near the ceiling toward the floor, providing a slow downwash of clean air at work stations, to minimize the potential for inhalation of contaminants. Airborne contamination in all C3b rooms is monitored using continuous air monitors and pressure is monitored to detect changes in containment barriers. The ventilation system is equipped with differential pressure monitors and alarms as identified in Chapter 11.

A second source of potential occupational radiation exposure is from direct exposure to radiation sources within gloveboxes. Although previous exposure rates are low (MELOX and La Hague), various design features have been implemented to attenuate ionizing radiation and to further limit operator exposures, including (1) limiting exposure times through automation and remote control of production workstations, and (2) placing shielding between radiation sources and operators.

For process cells in the AP Area, the primary feature is remote operations capability, with few operations performed in radiation areas. System sampling and inspections are designed to be performed from access areas outside of high radiation areas. Sources of radiation often can be removed from the work area prior to extensive work being performed. Routine access to process cells is precluded. Radiation shielding consists of multiple barriers including concrete cell walls and borated concrete panels around process equipment for neutron absorption.

MOX Processing (MP) Area work is primarily performed in the process rooms; thus these rooms are routinely accessed. Radiation and pressure monitoring are performed to detect changes in the confinement barriers. Shielding is designed so that dose rates in radiation work areas are low, to accommodate required access. Existing data from the MELOX and La Hague facilities are used to estimate access requirements. Radiation shielding for both neutron and gamma sources is designed permanently into the glovebox system (inside the glovebox for large radiation sources when this does not impair operation, and outside the glovebox whenever practical). Shielding is separate from the confinement barrier to allow for changes, if needed, without the potential for spreading contamination. The radiation shielding concepts in the MFFF include the following:

- **AP cells** – thick concrete walls constitute the primary shielding
- **AP gloveboxes** – shielding on the gloveboxes as needed; limited access – primarily for sampling
- **MP gloveboxes** – shielding inside the gloveboxes when necessary; external shielding outside the gloveboxes in general based on access requirements
- **MP areas** – have separate areas for each process unit shielded by concrete and sealed to prevent the spread of contamination.

Standard shielding materials are used to attenuate radiation intensity at the worker. Calculations are performed using shielding material properties contained in American National Standards Institute/American Nuclear Society (ANSI/ANS)-6.4.2-1985, R1997. Materials used for shielding include: leaded glass, plastic, borated polymers and plasters, carbon and stainless steel, cadmium, ordinary and borated concrete, and pourable plasters.

Glovebox design incorporates use of shielding to protect workers from direct radiation. Interior shielding is provided to ensure that radiation from specific sources is minimized. Glovebox walls incorporate appropriate shield materials to reduce worker exposures. Regular glovebox maintenance is conducted to preserve operability. Irregular, longer duration glovebox maintenance is scheduled at times when radiation sources are not present, to minimize radiation exposures to the maintenance personnel and to limit the potential for a release of airborne radioactive material.

Glovebox design complies with 10 CFR §20.1406 requirements for the minimization of contamination and uses the MELOX and La Hague facility design experience for guidance. The design includes permanent shielding in the process rooms.

Project quality assurance applies to shielding design, procurement, installation, maintenance, and operation. Radiation shielding testing verifies the efficacy of installed shielding materials in meeting radiation shielding design goals and the direct dose regulatory requirements of 10 CFR Part 20.

Shielding materials are selected for the source term to effectively reduce dose rates to meet ALARA goals. Borated polymers are used for neutron attenuation, and stainless steel, plastic, and leaded glass are used for photon shielding in the glovebox units.

9.1.2.3 Ventilation Systems, Glovebox Design, and Waste Minimization

The design of ventilation systems and gloveboxes ensures that during routine and nonroutine operations and anticipated events, the airborne concentration in occupied operating areas remains well below the limits of 10 CFR Part 20, Appendix B. Engineering controls are preferred over the use of respiratory protection.

The MFFF process implements recycling and reuse for waste minimization. For example, the recycling process minimizes the quantity of plutonium in the final waste by using systems that return (recycle) radioactive material to previous steps of the main process. Liquid waste is minimized in the AP process by use of recycling to the maximum extent practical. Nitric acid is recovered by evaporation from the process and partly reused as reagent feedstock for the plutonium dissolution subprocess. Distillates from the evaporation process are collected and partly reused in the process. Spent solvent from the plutonium separation step is regenerated by washing with sodium carbonate, sodium hydroxide, and nitric acid to remove degradation products from organic compounds, including trace amounts of plutonium and uranium.

Solid waste is minimized by reuse of solid scrap material from fuel fabrication. Many other system design features perform contamination control, confinement, and associated waste minimization functions. The process design reduces the distribution and retention of radioactive materials throughout plant systems by using vacuum systems in the gloveboxes. Airborne dust is collected in dust pots in dedusting systems installed in the gloveboxes, and the material is recycled. These design features control contamination to ensure that secondary waste production is minimized during plant operation.

9.1.2.3.1 Ventilation System Design

The ventilation (heating, ventilation, and air conditioning [HVAC]) system is designed to incorporate features that ensure workers are protected, to the greatest extent practical, from airborne radioactive material during normal and anticipated conditions. Many ventilation system design features described in this section also promote reduced airborne effluent releases, thus minimizing exposure to site workers and the IOC.

The HVAC systems maintain a negative pressure gradient between building confinement zones, and between the buildings and outdoors to ensure that airflow is from zones of lesser to greater contamination potential. Confinement zones are bounded by confinement system boundaries, across which a well-defined pressure gradient is maintained. This ensures that an air exchange, and consequently airborne contaminants, across a breach is also from zones of lesser to greater contamination potential. For example, air flows from clean areas (C1 or C2 zones) to the most contaminated areas (C4 zones) (e.g., gloveboxes), before being exhausted via high-efficiency particulate air (HEPA) filters to the plant stack. C4 zones are the primary confinement zones containing process equipment and enclosures. C3 zones are broken down into two levels depending on the contamination hazard: C3a zones have a low occasional hazard, while C3b zones have a moderate hazard. C2 zones have a low occasional contamination hazard, and C1 zones have no potential for contamination.

In the AP and MP Areas, dynamic confinement of C4 zones is ensured by the Very High Depressurization Exhaust (VHD) system. In the AP Area, dynamic confinement of process cells is provided by the Process Cell Depressurization Exhaust (POE) system. In the AP and MP Areas, dynamic confinement of C3a and C3b zones within secondary confinement is provided by the High Depressurization Exhaust (HDE) system. In the AP and MP Areas, dynamic confinement of C2a and C2 zones within tertiary confinement is provided by the Medium Depressurization Exhaust (MDE) system. For the AP process cells, the typical cascading sequence of pressure gradients between neighboring zones is as follows:

C1 → C2a → C2 → process cells

For the AP and MP Areas with gloveboxes containing dispersible material, the typical sequence is as follows:

C1 → C2a → C2 → C3a → C3b → C4

In both examples, leakage airflow is from high pressure to low pressure.

Airlocks/drop tubes for access are provided between zones. Cascading air from the cleaner areas through the airlock/drop tube minimizes potential for migration of airborne contaminants into clean areas during personnel access.

Monitors and alarms indicate changes in confinement pressure to warn personnel so that appropriate action is taken. The instrumentation for a glovebox or enclosure ventilation system includes devices to indicate the differential pressure across the glovebox or enclosure, filter resistance, and the exhaust flow rate from the glovebox or enclosure. An alarm will signify abnormal pressure at a location where operations personnel are stationed.

The ventilation systems operate continuously to protect personnel from exposure to airborne and transferable contamination. Redundancy ensures continuous operation of an HVAC system in the event of the failure of an active component (e.g., a fan or a damper) during normal or anticipated conditions. The Emergency Alternating Current (AC) Power system provides uninterruptible power to the VHD glovebox exhaust fans.

Room airflow in some rooms is designed to reduce the possibility of airborne radioactive materials being released in the vicinity of workers during abnormal conditions. Air is supplied above the worker and exhausted as close to floor level as possible. This design provides a "wash" across the worker, resulting in the air around the worker being maintained free of contaminants.

These design features minimize the potential that workers are exposed to airborne radioactive material during normal operations, maintenance, or anticipated events.

Airborne radioactivity monitoring and warning systems are provided for worker protection and safety. Systems are located near the glove ports and are placed to maximize sensitivity. The location was determined based on air flow characteristics. The monitoring and warning systems are connected to a data network, providing numerous communication links and readout capabilities. Alarms and instrument readouts are provided in the Radiological Protection Control Area (RPCA) of the Polishing and Utilities Control Room (PUCR), Emergency Control Rooms, and the Operations Support Center in the Technical Support Building (BTS) during postulated events.

9.1.2.3.2 Glovebox System Design

The primary function of the glovebox is to protect workers from radioactive materials. The gloveboxes are considered primary confinement and are designed to meet ALARA objectives for both direct and internal radiation sources, and to ensure worker safety.

Glovebox design incorporates design techniques to minimize pockets and sharp corners. Smooth surfaces and rounded corners provide for ease of cleaning and recovery of material. This design reduces the localized collection of radioactive material and thereby reduces worker radiation exposure. Periodic cleaning inside the gloveboxes removes dust and minimizes contamination.

Gloveboxes are designed to withstand anticipated conditions (e.g., the design basis earthquake, over- or underpressure). The design ensures that, for anticipated conditions, personnel are provided appropriate protection from a release of radioactive material. Glovebox design is based on providing adequate airflow and sealing surfaces to preclude releases from the glovebox. Glovebox penetrations are designed with glove ports that are sealed to prevent release of contamination.

9.1.2.3.3 Design Features to Reduce Contamination and Waste Production

Many of the design features addressed in previous sections perform contamination control functions. In addition, the design reduces the distribution and retention of radioactive materials throughout plant systems by using a vacuum system in gloveboxes. Airborne dust is collected in glovebox dust pots, and the material is recycled. Contamination entrained in the C4 exhaust is collected on HEPA filters at the glovebox boundary. Design features control contamination so that secondary waste production is minimized. These design features ensure that contamination is confined to specific areas and that contamination is minimized at the time the plant license is terminated, to facilitate eventual deactivation.

9.1.3 Radiation Protection Design Analysis

Potential occupational radiation exposure from external radiation sources is evaluated and minimized throughout the facility design process using general radiation zoning criteria, the ABAQUES dose assessment method, and design ALARA evaluations.

Each source of radiation within the facility is identified and included in the shielding analysis to estimate radiation dose-rate fields throughout the facility. Radiation sources are identified for each source configuration and "collapsed" for computer code input. Radiation transport codes are used to predict dose rates at work locations. Shielding is designed to meet radiation zone criteria and assures that exposures are below MFFF goals and ALARA.

Based on MELOX and La Hague operating experience, a residual source of contamination was conservatively estimated for loss-of-confinement and extremity dose analyses.

The occupational dose for normal operations and maintenance is assessed during the design phase. Significant occupational doses are evaluated for design enhancements to reduce the potential doses. ALARA analyses are performed to evaluate design alternatives to reduce occupational dose.

9.1.3.1 Source-Pertinent Information

Five primary radiation sources are used for radiation protection design: nonpolished plutonium, polished plutonium, raffinates, master blend, and final blend. Nonpolished plutonium, as received at the MFFF, contains daughter products from the original product that has decayed for about 40 years. As the facility nears the end of life, the original product received will have decayed about 70 years. These inventories are decayed to maximize the photon source term. Neutrons are produced by spontaneous fission and through alpha-neutron (α, n) reactions. Impurities associated with input materials are incorporated into the alpha-neutron (α, n) reaction for the unpolished source.

The sources identified are used to:

- Evaluate consequences of nonroutine events for the radiation protection design
- Provide input to shielding codes used in the design
- Establish design features, along with controls and responsibilities for restricted, controlled, and unrestricted areas
- Develop plans and procedures
- Assess occupational dose.

9.1.4 Shielding Evaluations

MELOX and La Hague operating experience is used throughout the MFFF design process to minimize occupational and public radiation exposure. Operating experience that defines the occupancy for each of the process units is used to estimate the occupational exposures for each glovebox. Radiation sources are determined for the MFFF. The redesign of some process units for process reasons and/or to optimize radiation protection is taken into account in the analysis. These sources are used to calculate the dose rates and thus establish the radiation shielding requirements. Process units that result in higher occupational exposure are reviewed to maximize productivity, minimize maintenance, and thus minimize radiation exposures. The types of MELOX and La Hague data used for the MFFF design for personnel access requirements are as follows:

- Description of activities
- Proximity to radiation sources
- Definition of radiation sources
- Duration of activities
- Duration of time that hands are in the gloveboxes.

Permanent shielding is designed in the facility to lower dose rates to comply with 10 CFR Part 20 during routine and nonroutine operations and anticipated events. Radiation zone drawings are used to locate equipment.

Design goals for internal and direct doses are based on fractions of 10 CFR Part 20 limits. These were developed by making use of the design features and experience of the MELOX and La Hague facilities. Exposure data and the difference in the source terms between MELOX, La Hague, and MFFF material are used in setting these design goals. The permanent and temporary shielding developed as part of this design meets these design goals.

The Total Effective Dose Equivalent (TEDE) is the effective dose equivalent from external exposures plus the Committed Effective Dose Equivalent (CEDE) from internal exposures. Design goals for TEDE which were established early in the design process for individual workers are applied to facility operations (see Table 9.1-2).

Design drawings and descriptions of the shielding for high and very high radiation areas clearly identify the penetrations, shield doors, and labyrinths incorporated to meet the shielding design criteria. Radiation shielding analyses are used to verify the shielding for each process room, including the dose rates for each position workers are required to take to perform routine and nonroutine maintenance. This design is based on experience and the design features of the reference facilities. A radiation shielding test program will be implemented prior to the start of operations for protection of personnel from high radiation dose rates.

Shielding calculations are performed using several standard industry computer codes (Monte Carlo N-Particle [MCNP], SCALE, Perceval, SN1D). Dose rate estimates are determined using flux-to-dose conversion factors contained in ANSI 6.1.1-1977, *Neutron and Gamma-Ray*

Fluence-to-Dose Factors. The 1977 version is more conservative than ANSI 6.1.1-1992 for MFFF's photon spectra.

The shielding design complies with 10 CFR §20.1406 requirements for the minimization of contamination and uses the reference facilities' design experience for guidance. The MFFF minimizes waste of shielding materials. The design includes permanent shielding in the process rooms.

9.1.4.1 Shielding Information for Each Radiation Source

Shielding is specified in each radiation shielding calculation to reduce dose rates and occupational doses to below levels established in the radiation zone drawings and below administrative goals. For those areas with estimated exposures greater than administrative goals, an ALARA evaluation is performed to determine if design changes should be implemented to reduce the dose.

9.1.4.2 Criteria for Penetrations

Penetrations in shielding for high radiation sources are minimized in the design. For lower dose-rate sources, the impacts are analyzed in shielding analyses and determined to meet the ALARA goal. Radiation protection design features are provided in accordance with Regulatory Guide 8.8, *Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable*.

9.1.4.3 Shielding Materials

Standard shielding materials are used to attenuate the radiation intensity at the worker. Materials such as leaded glass, plastic, borated concrete, borated polymers, borated plasters, stainless steel, and ordinary concrete are used. ANSI/ANS-6.4.2-1985, R1997, *Specification for Radiation Shielding Materials*, is used as the reference for shielding material properties for performing calculations.

9.1.4.4 Dose Assessment and ALARA Evaluations

The general design requirements established for the various radiological attributes addressed below include those that maintain exposures ALARA during normal operation and minimize exposures during off-normal conditions.

Potential occupational radiation exposure from external radiation sources were evaluated and minimized throughout the facility design process using general radiation zoning criteria, the ABAQUES dose assessment method, and design ALARA evaluations.

9.1.4.5 Radiation Zoning

Radiation zoning (see Table 9.1-1) is developed based on estimates of the access required for each area and radiation dose limits for personnel from 10 CFR Part 20. Shielding for the process units and access areas is designed to satisfy radiation zoning criteria. The final dose assessment verifies that the facility can be operated within the occupational exposure limits of 10 CFR Part 20 and ALARA principles.

Radiation zone drawings show the design occupancy for radiation zones as follows: Zone Z1 is a continuous occupancy area for staff and visitors. Zone Z2 is a continuous occupancy area for trained workers. Zone Z3 is a limited occupancy area in which routine maintenance may be performed by trained workers. Zone Z4 and zone Z5 are conservatively estimated and are expected to be higher radiation areas. Access to zone Z5 radiation area is controlled in accordance with 10 CFR §20.1601.

Radiation shielding design as documented in the shielding analyses satisfies radiation zone criteria for restricted access areas. The design criteria for occupational exposures inside the MFFF are supported by the radiation zone criteria.

In zones Z1 and Z2, residence time is not restricted. The design basis maximum area radiation dose rates shown on radiation zone drawings allow continuous occupancy. The design basis maximum area radiation dose rate limit is the only shielding design criterion. Residence time is restricted in zones Z3, Z4, and Z5 of the AP Area, and access is permitted only intermittently.

Access to zone Z3 process rooms in the process areas is necessary for normal operations and routine maintenance. The annual dose equivalent for workers was evaluated with reasonable assumptions (in the form of time-motion studies). Access to zones Z4 and Z5 is restricted to nonroutine maintenance or intervention.

9.1.4.6 The ABAQUES Method

The facility design and resultant occupational dose are evaluated using the ABAQUES dose assessment method, which is similar to that provided in Regulatory Guides 8.19 and 8.34. Radiation shielding is selected to minimize personnel occupational exposures based on facility occupancy for normal operations and facility maintenance. Personnel exposures are estimated based on facility experience for access requirements, and standard shielding methods are used to estimate radiation fields. The method is iterated to minimize the number of personnel that have the potential of receiving doses in excess of the design goal. The general equation used to satisfy this prerequisite is as follows:

$$\frac{\sum_i \mathrm{f_i} \times t_i \times DER_i}{\sum_i \mathrm{f_i} \times t_i} \leq \frac{\text{design objective for individual doses}}{\mathrm{T}} \qquad \text{(Eq. 9.1.4.6-1)}$$

where:

$\mathrm{f_i}$ = the frequency of each task associated with a given process unit or group of process units

$\mathrm{t_i}$ = the time of exposure for the task

DER_i = the dose equivalent rate for the task

T = the worker average estimated annual working time in radiation areas

$\sum_i \mathrm{f_i} \times t_i =$ the total yearly duration of the tasks performed by the same work group associated with the process unit or group of process units.

The DERs are adjusted by varying the shielding thickness, and/or the operating conditions (operation duration and frequency) are changed to reduce the exposures to below the design goal. *T* is an estimate of the average time an individual spends in the radiation area per year based on industry operating experience. This is approximately 50% of the total working time, or 1,100 hours per year. The remaining time is associated with training, administrative duties, and work in the facility but outside of the radiation area. This approximation gives a rough estimate of the number of personnel required to perform normal operations and routine maintenance for each process unit.

9.1.4.7 ALARA Evaluations

This process includes a preliminary estimate of the occupational exposure, an ALARA evaluation of the activities that produce exposures, and recommendations for design enhancements to reduce occupational exposures. Lessons learned from facility operations and industry guidance are used to evaluate potential design enhancements. ALARA cost-benefit analyses are performed in accordance with NUREG/CR-0446, *Determining Effectiveness of ALARA Design and Operational Features.*

Occupational exposure data based on data from MELOX and La Hague were estimated. These data were used during the design phase to evaluate occupational radiation exposures and to recommend potential enhancements to the design to effectively reduce doses. Final design shielding calculations were performed to estimate dose rates and doses using the ABAQUES dose assessment method.

Several areas were further examined for cost-effective design changes to reduce the estimated occupational dose. Examples include:

- The receiving area, where transport casks with feed material are received and processed for counting and storage is evaluated. Impurities associated with the alternate feedstock feed material cause higher neutron radiation. Recommendations were made to reduce dose rates and personnel occupancy time to reduce potential doses.
- The assembly fabrication unit was evaluated for dose reduction. The MOX assembly is fabricated in a manner similar to a standard uranium fuel assembly. Design changes were made to automate the process as much as possible and to reduce worker time in the radiation area.
- The assembly packaging unit was extensively reviewed for ALARA design changes. Several design changes were made to reduce the dose rate and reduce the access time.

9.1.4.8 Predicted Occupational Doses

Estimated doses for operations meet 10 CFR Part 20 and ALARA criteria.

9.1.4.9 Dose Assessment Estimate

Occupational exposure was estimated for process units with expected occupancy for normal operations and preventive maintenance. MELOX and La Hague experience shows that outage maintenance contributes about 50% of the normal operating doses. The inhalation dose for MFFF is expected to be small.

9.1.4.10 Contribution from Internal Exposure

As previously noted, there are two primary sources of radiation risk to the MFFF worker: plutonium inhalation and direct radiation exposure. Plutonium inhalation is the most significant potential hazard at the MOX facility. Design engineers are instructed on the risks and the methods of controlling plutonium contamination. Process units that handle powder have the greatest potential for generating respirable particulate, releasing contamination, and causing worker inhalation exposure. The process areas for these units provide radiation protection through the following multiple system barriers and controls:

- The operations for the units are controlled remotely and are automated to minimize access to the work area.
- The plutonium is contained in a sealed glovebox. This internal environment is kept under negative pressure relative to the worker environment. A leakage would be into the glovebox, thus preventing the release of contamination.
- Pressure within the glovebox is monitored.
- Glove ports are provided for maintenance access to the process equipment.
- When practical, process material is removed prior to maintenance activities.
- Workers evacuate the area upon radiation monitoring alarms.

Events that are expected to occur over the lifetime of the facility and their consequence are estimated and added to occupational exposure estimates.

Design features and management measures at the reference facilities are similar to MFFF; thus, the normal internal exposure received at the reference facilities, which is a small fraction of the total dose, is assumed to represent a reasonable estimate for the MFFF.

9.2 OPERATIONAL RADIOLOGICAL PROTECTION

The radiological protection program implements the requirements of 10 CFR Part 20, *Standards for Protection Against Radiation*, and the appropriate sections of 10 CFR Part 19, *Notices, Instructions and Reports to Workers: Inspection and Investigations*, and 10 CFR Part 70, *Domestic Licensing of Special Nuclear Material*. The radiological protection program implements the programmatic requirements necessary to ensure that radiological work activities are performed in a manner that protects the health and safety of workers, the IOC, and the environment.

The radiological protection program ensures the following:

- The individual worker's exposure to radiological hazards is ALARA.
- Personnel responsible for performing radiological work are appropriately trained.
- Personnel responsible for implementing and overseeing the radiological protection program are well qualified.
- The ALARA process is incorporated into the facility design, modifications, and work processes.
- Line management is involved and accountable for radiological performance.
- Radiological measurements, analyses, worker monitoring results, and estimates of public exposure are accurately and appropriately conducted.
- Radiological operations are conducted in a manner that controls the spread of radioactive material and reduces exposure to the work force and the public, and a process is used that maintains exposure levels ALARA.
- Employees have the authority and responsibility to stop radiological work activities suspected of being unsafe.
- Oversight is provided for radiography activities.

Contracted radiation technical support and services (e.g., instrument calibrations, dosimeters) are subject to controls under the Quality Assurance Program, which is described in Chapter 15.

MFFF is operated in a manner to not exceed radiological dose limits and to meet the goals of ALARA, as defined in 10 CFR Part 20. Radiological work activities, including those performed by subcontractors, meet the requirements of the radiological protection program.

Actions taken to maintain doses ALARA are documented as part of the radiological protection program.

9.2.1 ALARA Program

The purpose of the ALARA program is to maintain exposure to the public and occupational radiation exposures as low as reasonable achievable by the use of sound engineering controls, radiation protection practices and radiation protection procedures. Line management and the work force are committed to this policy and work to establish goals that are as far below the regulatory limits as reasonable. Management shall ensure that the work force is committed to the ALARA goals to take every reasonable effort to maintain exposures to radiation as far below the regulatory limits consistent with the plant operations, current technology, and benefits to the health and safety of personnel. Management will make every effort to ensure that plant personnel are aware and committed to the ALARA principles.

The ALARA program is composed of the following:

- ALARA program description
- ALARA principles incorporation into plant procedures involving radioactive materials
- ALARA Committee
- ALARA Chairman
- ALARA program coordinator – An appointed member of the radiological protection staff who assists the ALARA Chairman in implementing the ALARA program.

9.2.1.1 Management Commitment

The responsibility for complying with radiological safety requirements and for maintaining radiation exposures ALARA starts with the individual worker and broadens as it progresses upward through the organization. Line management is fully responsible for the radiological performance of their personnel and takes necessary actions to ensure that personnel are properly trained and that performance is monitored and corrected as necessary. As part of their commitment to radiological safety, senior management ensures that the ALARA program is implemented and that line management is held accountable.

Management commitment to ALARA principles is communicated to plant personnel through policy statements, instructions to personnel, and similar documents, as well as by direct communication, training, and inspection of the workplace.

Management ensures that personnel are made aware of the commitment to keep exposures ALARA through the use of audits of radiation work activities, evaluation of training programs, and evaluation of plant procedures. The results of the systematic review process provides for the communication of management expectations for maintaining exposures ALARA. Radiation protection audits include reviews of operating procedures, exposure records, inspections of Radiologically Controlled Areas (RCAs) and interviews with radiation protection personnel and plant operating personnel. Training program reviews include classroom observation, evaluation of training content for regulatory requirements, incorporation of lessons learned information as well as on the job training for radiation protection staff and plant operating personnel. Plant procedure evaluations ensure that lessons learned are incorporated to achieve a strong position for ALARA concerns.

Management also takes an active role in the evaluation of maintenance and modification activities for the opportunity to have ALARA principals incorporated in maintenance operations. Management ensures that there is a well supervised radiation protection program that oversees the maintenance activities with the authority to enforce safe operations. Management empowers radiation safety personnel to have the authority to take appropriate actions to prevent unsafe practices and communicate with senior management the concerns with the activities.

9.2.1.2 ALARA Committee

The ALARA Committee provides the focus and direction for improving the radiological protection program. The ALARA Committee includes the ALARA Chairman (who is a member of line management and nominated by senior management); the ALARA program coordinator; the manager of the radiological protection function; and personnel from line management, operations, engineering, criticality safety, and maintenance functions. ALARA Committee members are made up of personnel who have had an intimate role in the design of the facility as well as personnel with operating experience at the reference facilities and similar facilities within the United States. Radiological protection personnel act as advisors to the committee. All ALARA Committee members are qualified in their respective areas of expertise through training, experience and operational knowledge. In addition to the radiation protection personnel on the Committee, all members have experience in radiation protection through operational experience, training and formal education. The ALARA Committee meets frequently according to project procedures, and more often for the evaluation of upcoming maintenance activities, following abnormal events and unusual exposures to personnel. Reports on the status of the program are provided at least annually.

The ALARA Committee evaluates major design activities, operations activities, or plant modifications that could affect radiation levels, doses, and radioactivity levels in liquid and gaseous effluents. The ALARA Committee considers the results of the Integrated Safety Analysis in determining whether further reductions in occupational radiation doses are reasonable. The ALARA Committee evaluates trend analyses and the adequacy and implementation of radiological performance (ALARA) goals. Reviews and recommendations of the ALARA Committee are tracked to completion.

9.2.1.3 Administrative Control Levels and Dose Limits

The objective of minimizing radiation exposure is to maintain individual radiation doses ALARA, but in all cases below regulatory limits. To accomplish this objective, administrative control levels are established below the regulatory limits to control individual and collective radiation dose (see Table 9.1-2). The administrative control levels are multi-tiered with increasing levels of authority required to exceed higher administrative control levels. Unless otherwise indicated, administrative control levels and dose limits are stated in terms of the TEDE.

9.2.1.4 Internal Audits and Assessments

Internal audits and assessments are performed under the Quality Assurance Program such that over a 12-month period, functional elements of the radiological protection program are evaluated for program compliance and implementation (10 CFR §20.1101(c), *Radiation Protection Programs*). The results of these evaluations provide valuable feedback to line management on those areas requiring additional management attention. Areas of review include, but are not limited to, access control (including proper posting, labeling, and operability of access controls), proper identification of restricted areas to prevent the spread of contamination, numbers and appropriate locations of step-off pads, change facilities, personal protective equipment facilities, personnel monitoring equipment, contamination and overexposure events, Radiation Work Permits (RWPs), instrumentation, and respiratory protection.

Radiological protection program performance is periodically evaluated using performance indicators measured against specific goals. These indicators are collective dose (person-rem), skin and clothing contaminations (number), radioactive material intakes (number), radioactive waste (volume), and airborne radioactive releases (curies). Trends in these areas provide information on the performance of the radiological protection program.

9.2.2 Radiological Protection Organization and Administration

The radiological protection function is independent of the operations and maintenance functions and has direct access to senior management through the Manager of Environmental Safety & Health (ES&H) Licensing. The radiological protection function provides relevant support to facility operations. The radiological protection function develops policies and procedures to ensure compliance with 10 CFR Part 20, and to ensure that the policies and procedures are implemented as necessary for compliance with 10 CFR §20.1101(b).

The radiation protection program oversight is under the responsibility of the MFFF senior plant management. Senior management is responsible for ensuring that the radiation protection organization is provided with adequate resources to manage an effective program and maintain exposures ALARA. Senior management also supports the radiation protection program by re-enforcing the ALARA principals throughout the MFFF organization. Senior management participate in the establishment of administrative goals and limits, evaluation of ALARA goals, and the review of information (exposure records, waste minimization, etc) concerning meeting MFFF goals.

Individuals responsible for developing and implementing measures necessary for ensuring compliance with the requirements of 10 CFR Part 20 have the appropriate education, training, and skills to discharge these responsibilities. The radiological protection function, working with facility management, ensures adherence to the radiological protection program in operations and provides the required radiological support to the facility organization.

The manager of the radiological protection function (RPM) is responsible for setting radiological protection policy and for implementation of this policy. He reports to the Manager of Environmental Safety & Health (ES&H) Licensing.

The RPM has the responsibility for

- Planning, administering, and maintaining the radiological protection program — with support from line management
- Reviewing facility modifications and operations activities
- Ensuring that radiological protection program elements are appropriately implemented and maintained through radiological policies, procedures, and documents
- Approval of radiological protection policies and procedures

- Management of the Respiratory Protection Program
- Ensuring that staffing for the radiological protection function is adequate to conduct routine radiation functions in a timely manner and ensures radiation requirements can be met during routine operations and nonroutine operations, such as anticipated events and accidents
- Participation in design reviews for radiation protection concerns
- Identification of potential areas and operations that may be a significant source of radiation exposure
- Participation in the development of training programs to ensure all MFFF personnel are knowledgeable of the radiation protection programs, concerns and ALARA policies
- Participation on the ALARA Committee and developing ALARA policies
- Supervision of the radiation protection surveys, radiation work activities and the collection of data and information concerning radiation and contamination
- Supervision of training of the radiation protection staff.

The RPM is an experienced professional in radiological protection and is familiar with the design features and operations of the facility that affect the potential for exposures of persons to radiation.

The RPM has the technical competence and experience to establish effective radiological protection programs and the supervisory capability to direct the implementation and maintenance of the radiological protection program.

The RPM is supported by a staff of radiation protection technicians assigned to various shift activities so as to provide around the clock radiation protection coverage. Each shift is managed by a senior technician or supervisor who represents the RPM in all activities so that there is a continuity of radiation protection management for all MFFF operations. The senior technicians are responsible for the operation of the radiation protection program when the RPM is absent and have the authority to act in the absence of the RPM.

See Chapter 4 for discussion of minimum qualification requirements for the radiological protection organization.

9.2.3 Radiation Safety Procedures and Radiation Work Permits

The primary methods used to control workplace exposure are operating procedures and facility and equipment design features. These controls are augmented with the use of area entry/exit requirements to control access to and from radiological areas, and RWPs to provide specific requirements for all work within the RCAs. Personnel entering the RCAs are required to read and understand the Radiation Work Permit (RWP) requirements and monitor their exposures during the conduct of their activities.

RWPs are issued and controlled in accordance with approved radiological protection procedures for activities conducted within the RCAs. RWPs may be general in nature for normal access and rounds, daily operations and activities that do not require access into high or very high radiation areas.

Specific tasks such as maintenance activities require RWPs for that individual task to provide specific requirements and documentation of exposures for those workers. Maintenance work packages will include the specific RWP for that activity and all personnel working on the activity are required to read, understand and control their work according to the RWP.

RWPs are initiated by the individual or group that intends to perform an activity (operations, maintenance, laboratory, etc.) and provide the location of the work, duration and specific information concerning the activity such as the work package detailing the maintenance activity. The radiation protection staff provides the radiological conditions of the work areas, establishes stay times, protective clothing requirements, shielding (if required), dosimetry requirements, etc. The Radiation Protection Manager reviews and approves the RWP. Other RWP approvals may include other organizational groups' reviews and/or approvals, when appropriate, to ensure that provisions of the RWP or related documentation address potential hazards (including non-radiological hazards) and compliance with applicable regulations. The radiation protection staff reviews the RWP with the associated work group to ensure that personnel are aware of all the requirements to ensure exposures are minimized.

RWPs include a list of safety requirements for authorized work, and include at least the following, as applicable:

- The identification of personnel working on the task
- Expected radiological conditions (radiation, contamination, and airborne levels)
- Type and frequency of monitoring and dosimetry (e.g., continuous air monitor [CAM], self-alarming dosimetry)
- Estimated doses for the authorization
- Limiting doses for the authorization
- Allowable stay times
- Special instructions or equipment (e.g., special shielding required)
- Hold points or monitoring points, if applicable
- Personnel protective equipment requirements
- Authorization signature and date
- Actual doses, time, or other information resulting from the completed work authorization recorded on the RWP
- Expiration/termination date of the RWP
- Sufficient information on RWPs to allow independent inspection and reconstruction of the circumstances necessitating the RWP, the factors included, and the results.

Specific operations such as calibrations using licensed and non-licensed sources may not require the use of a RWP. Procedures that involve the use of licensed or non-licensed radioactive materials without an RWP require review and approval by the RPM and include equivalent information as identified in a RWP. The RPM reviews procedures that require the use of licensed or non-licensed radioactive materials for the inclusion of requirements for the control of personal radiation exposure and any protective measures.

Administrative controls (RWP expiration/termination date) ensure RWPs are not used past their termination dates. Procedures define the types of records to be kept, retention time for these records, and the final disposition of the RWP. The record system allows independent auditors to reconstruct the circumstances necessitating the RWP, the factors included, and results. Routine (e.g., long-duration maintenance) RWPs are reviewed periodically to identify improvements in worker protection.

Procedures and administrative controls ensure current copies of radiological protection procedures and RWPs are provided to appropriate personnel.

Radiological protection procedures and RWPs are developed, maintained, and used under quality assurance (QA) controls.

9.2.3.1 Radiological Work Planning

Work planning is the responsibility of line management, with support from the radiological protection organization. Radiological surveys are used to develop radiological protection requirements and are documented on the RWP. Specific radiological controls based on the surveys, and from formal ALARA reviews that were performed because established planning thresholds were exceeded, are incorporated into the work documents.

9.2.3.2 Radiation Area Access Control

Specific requirements for entering and exiting radiation areas are established. Radiation safety training commensurate with the hazards and required controls is required before unescorted access to radiation areas is permitted. The primary control for entry into radiation areas is the RWP, which is augmented by signs and barricades.

Administrative procedures implement radiation area access controls. These procedures address measures implemented to ensure the effectiveness and operability of entry control devices, such as barricades, alarms, and locks. Periodic inspections of the physical access controls to high and very high radiation areas are made to verify controls are adequate to prevent unauthorized entry. Worker access controls for high and very high radiation areas meet the requirements of 10 CFR §20.1601 and §20.1602.

9.2.3.3 Radiological Work Controls

Positive control of personnel is established through RWPs. Only trained and qualified personnel who have the information available to properly respond to the radiological conditions that they will encounter during the work activity are allowed to enter the restricted area unescorted. In special circumstances, specialists who have not completed unescorted access training may be allowed escorted access to perform specific tasks, with permission granted by the RPM.

The RWP is the administrative mechanism used to establish radiological controls for intended work activities. The RWP informs employees of area radiological conditions and entry requirements, and provides a mechanism to relate employee exposure to specific work activities.

9.2.3.4 Posting and Labeling

Posting and labeling of radiation areas, high radiation areas, and radiologically contaminated areas, equipment, and material are used to alert personnel to the radiological status of the item or area, and to prevent an inadvertent dose to the worker. This includes the use of the standard radiological posting and labeling to meet the requirements of 10 CFR Part 20 Subpart J, and posting signs that are clear and conspicuous. As stated in Chapter 1, an exemption request has been submitted related to container labeling requirements.

9.2.3.5 Release of Materials and Equipment

Material and equipment that are contaminated or potentially contaminated are considered contaminated until they are surveyed and released. This ensures that no contaminated material or equipment is inadvertently released. Movement of material and equipment from contamination areas, and between controlled areas and release of material and equipment from controlled areas, and from the site, is controlled. See Table 9.2-1 for contamination limits.

9.2.3.6 Sealed Radioactive Source Accountability and Control

Radioactive sealed sources are controlled by accountability and monitoring requirements to prevent loss or unintentional exposures. Sealed sources are leak tested in accordance with procedures that include limits and actions to be taken if limits are exceeded. Frequency of leak testing is no less than annually and is described in program documentation. Sealed sources in excess of limits in 10 CFR §20.1601 or §20.1602, when not in use, are kept in locked storage areas where access is controlled by the RPM.

9.2.3.7 Receipt of Packages Containing Radioactive Material

MFFF ensures that appropriate controls are implemented from the time of package receipt to final destination. Receipt and offsite transfer of radioactive materials is conducted in accordance with 10 CFR 20.1906, 10 CFR 71 and 49 CFR 171 – 178. Unauthorized access to packages is prevented to ensure that radiation dose is ALARA.

9.2.4 Radiation Safety Training

Radiation safety training is commensurate with the employee's duties. Standardized courses are used to the extent practical and are supplemented by facility-specific information. Personnel and visitors entering restricted areas receive either radiation safety training, or are provided a general indoctrination in site-specific safe practices and are escorted by an individual who has received such radiation safety training. To be granted unescorted access to the MFFF restricted area, individuals are required to pass site-specific general employee training.

Radiation safety training addresses the following topics, to the extent appropriate to each individual's prior training, work assignments, and degree of exposure to potential radiological hazards:

- Risks of exposure to radiation and radioactive materials, including prenatal radiation exposure, health risks, effects of exposure, internal and external exposure, fatality risks, cancer risks, and embryo/fetus risks
- Background exposure
- Regulatory limits and planned special exposures
- Administrative limits
- Basic radiological fundamentals and radiological protection concepts
- Controls, limits, policies, procedures, alarms, and other measures implemented at the facility to control doses, including both routine and emergency actions
- Identification of potential loss of confinement events
- Individual rights and responsibilities as related to implementation of the facility radiological protection program
- Individual responsibilities for implementing ALARA measures
- Individual exposure reports that may be requested.

Individuals likely to receive an occupational dose in excess of 100 mrem in a year will be instructed on procedures and equipment used to maintain exposure ALARA. All MFFF personnel will receive training commensurate with the requirements of 10 CFR 19.12.

Examinations are used to demonstrate satisfactory completion of theoretical and classroom material. Examinations are written; however, the RPM may approve alternatives to accommodate special needs. Alternative examinations are equivalent in content to written examinations. Trainees acknowledge in writing that the training was received and understood. Records of the most recent training and testing are maintained.

All MFFF radiation protection training courses are reviewed as a minimum on a three year cycle by the RPM for applicability, modification of the MFFF, and revisions to regulatory positions. Each course includes a portion on lessons learned and is updated on an annual basis to ensure that information is accurate on the conditions within the MFFF.

Training addresses both normal and abnormal situations in radiological protection.

General employee training is completed annually. Changes to the program are incorporated as they are identified and a decision made if retraining prior to the annual period is needed.

Radiological worker retraining also is completed annually.

MFFF site-specific general employee training and refresher training includes changes in requirements and updates of lessons learned from operations and maintenance experience and occurrence reporting for the MFFF site.

9.2.5 Air Sampling

Airborne radioactivity monitoring uses air samplers and/or CAMs, with usage based on working conditions. Frequency of air sampling is based on area conditions and planned activities. Counting techniques, action levels, and alarm setpoints are described in radiological programs and procedures. Controls minimize internal exposure to the radiation workers as part of the overall ALARA program. The estimation of internal dose is based on airborne radioactivity concentrations. In the event of suspected high exposure, the internal dose is verified from bioassay data.

Air monitoring equipment is used in situations where airborne radioactivity levels can fluctuate and early detection of airborne radioactivity could prevent or minimize inhalation of radioactive material by personnel. Selection of air monitoring equipment is based on the specific job being monitored. Air monitoring equipment includes portable and fixed air sampling equipment, and CAMs.

Air sampling equipment is used in occupied areas where, under normal operating conditions, a person is likely to receive an annual intake of 2% or more of the specified annual limit on intake (ALI) value (40 Derived Air Concentration [DAC]-hours).

CAMs are installed in rooms where radioactive materials are handled or there is a need to ensure that there is no airborne contamination present. The CAM system consists of Work Station CAMs, Area CAMs and Duct CAMs as well as Main Stack Exhaust CAMs. The Work Station CAMs are movable and are used by the operator when working within a glovebox or on an open system. The CAM is placed so as to detect the air passing the operators breathing zone (down draft across the face of the glovebox). Area CAMs are placed close to the room exhaust to sample the air exiting the room and detect any minute release of material that may escape the Work Station CAM detection. To further sample the potential airborne contamination, specific Duct CAMs are installed to detect leakage in inaccessible rooms (Process Cells). These CAMs sample a combined duct work so that individual cells may be sampled using installed sample ports. The Main Exhaust Stack CAMs provide for immediate notification of potential releases from the MFFF.

When specific maintenance activities are being conducted, portable CAMs or air samplers are placed within the work area to detect any airborne contamination. Air samples are taken upon opening systems and periodically during the maintenance. CAMs provide both an active alarm when specific set points are exceeded and a sample that is analyzed under laboratory conditions to determine the gross activity as well as the specific isotopes of concern. Air samples are also performed in conjunction with radiation and contamination surveys to validate the installed CAMs or to ensure rooms without installed CAMs are free of airborne contaminants.

In addition to the CAMs, personnel who perform work with radioactive materials are equipped with lapel air samplers. These samples are analyzed upon completion of the work shift or if the individual is in a room when a CAM alarm is activated. The combination of the CAM air filter analysis and the lapel air sample are used to calculate an individual's internal exposure if required.

All CAM, air samples and lapel air samples are recorded including the location, date, time of sample, volume, activity, isotopic concentration (if required), instrument used to analyze the sample, calibration date and name of the individual performing the analysis.

Laboratory analytical equipment minimum detection levels are based on the specific instrument, background radiation, type and sixe of the detector and counting times. Each instrument will be calibrated and have an established minimum detection level.

CAMs are used to provide early warning to individuals of events that could lead to substantial unplanned exposures to airborne radioactivity. Such exposures could result from a breakdown of engineered controls or improper establishment of boundaries during work that creates airborne radioactivity. Real-time air monitoring detects and provides warnings of airborne radioactivity concentrations that warrant immediate action to terminate inhalation of airborne radioactive material. Radiation protection procedures define the immediate actions upon receipt of a CAM alarm (8 DAC-hours) including the process of investigating and determining the cause of the alarm, the levels of contamination and processes for mitigating the release.

Air sampling equipment is positioned to measure air concentrations to which persons are exposed.

Air monitoring equipment is calibrated and maintained at a frequency specified in the radiological protection program. CAMs are capable of measuring 1 DAC when averaged over 8 hours (8 DAC-hours) under laboratory conditions.

Continuous air monitoring equipment has sufficient sensitivity to alert personnel that immediate action is necessary to minimize or terminate inhalation exposures.

The proper operation of continuous air monitoring equipment is verified by performing an operational check. Operational checks include positive air-flow indication, non-zero response to background activity, and internal check sources (or electronic checks when available). Continuous air monitoring equipment is verified by checking for instrument response with a check source.

Air sample results are evaluated as quickly as practical for evaluation of the need for respiratory protection, area evacuation (if necessary), worker intake, and worker relief from respirator use.

9.2.6 Contamination Monitoring and Control

Contamination monitoring and control measures prevent the movement of radioactive contamination from controlled areas to uncontrolled areas, and "clears" personnel and equipment when leaving contaminated areas. Radioactive contamination is controlled by using engineering controls, by containing contamination at the source, by monitoring, and by promptly decontaminating areas that become unintentionally contaminated. The use of personnel monitoring equipment is required when personnel leave a potentially contaminated area such as a

C3b ventilation controlled room (glovebox room). All C3b rooms are provided with airlocks containing personnel contamination monitors in the form of hand and foot monitors and "friskers." Personnel use the installed equipment to self-monitor prior to exiting the airlock. Personnel are considered contaminated if contamination levels are detected in excess of levels given in Table 9.2-1. When the self-monitoring results in an alarm, the alarm is recorded in the Polishing and Utilities Control Room and radiation protection personnel are dispatched to assist in decontamination efforts for the personnel as well as the room.

A controlled area is any area, outside of a restricted area but inside the site boundary, access to which can be limited by the licensee for any reason to protect individuals from exposure to radiation and/or radioactive materials. Individuals who enter only the controlled area without entering radiological areas are not expected to receive a total effective dose equivalent of more than 0.1 rem (0.001 sievert) in a year. Controlled areas are posted in accordance with 10 CFR 20.1900

A radiological control area/RCA is an intermediate area established to prevent the spread of radioactive contamination and to protect personnel from radiation exposure. RCAs are posted in accordance with 10 CFR 20.1900.

A contamination area is any area where the loose surface and/or fixed contamination levels exceed those of Table 9.2-1. Contamination areas are barricaded and posted in accordance with 10 CFR 20.1900 until the contamination levels are reduced below the established limits.

A high contamination area is any area where the loose surface of fixed contamination levels exceed ten (10) times the levels established in Table 9.2-1. High contamination areas are barricaded and posted in accordance with 10 CFR 20.1900 until the contamination levels are reduced below the established limits.

A radioactivity area is any area where there is a natural and spontaneous process by which the unstable atoms of an element emit or radiate excess energy from their nuclei and, thus, change (or decay) to atoms of a different element or to a lower energy state of the same element. Radioactivity areas are posted in accordance with 10 CFR 20.1900.

A radiation area is an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 0.005 rem (0.05 mSv) in 1 hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates. Radiation areas are posted in accordance with 10 CFR 20.1900

A high radiation area is an area, accessible to individuals, in which radiation levels could result in an individual receiving a deep- dose equivalent in excess of 0.1 rem (1 mSv) in 1 hour at 30 centimeters from the radiation source. High Radiation areas barricaded and locked to prevent entry and are posted in accordance with 10 CFR 20.1900

An Airborne Contamination area is an area where the concentration of airborne radioactive materials, composed wholly or partly of licensed material exist in concentrations in excess of the derived air concentrations specified in appendix B to 10 CFR 20 or where an individual present in the area without respiratory protection could exceed an intake of 0.6 percent of the annual limit of intake. Airborne Contamination areas are barricaded and posted in accordance with 10 CFR 20.1900 until the contamination levels are reduced below the established limits.

All areas which are identified as being controlled for radiological concerns (radiation, radioactive materials, airborne contamination, etc.) are posted in accordance with 10 CFR 20.1900.

To monitor and control contamination, instrumentation appropriate for the contaminant is used; most often this will be an alpha-sensitive instrument. Some beta/gamma instruments will be used in areas where there is sufficient design information (source terms) that would indicate that these instruments will provide adequate survey results. Tritium contamination surveys will be conducted in those areas that have been determined to have a potential for tritium leakage. Radiation and contamination surveys are performed on a continuous basis of all areas within the MFFF Radiation Controlled Area and well as selected areas outside the controlled areas. Survey frequencies are based on the engineering design of the ventilation system where the air flow is from the uncontrolled areas to areas where there is a potential for contamination. RCAs that have a high occupancy rate are surveyed on a frequent basis of normally once per week. Areas where there is an infrequent entry are surveyed on a less frequent basis normally monthly or quarterly. Access and egress areas are surveyed daily to ensure the control of any potential contamination. Areas such as lunch rooms, change rooms and offices are surveyed on a quarterly basis since they are located outside the RCAs. Survey frequencies may be changed based on experience and historically information derived for the performance of the survey program.

All surveys are performed in accordance with approved procedures that encompass the objectives of the survey, methodology of the survey, expected equipment necessary to perform the survey, general survey frequencies, type and format of the survey records, review and approval of survey results, reporting of survey results and document of the surveys.

The radiation protection organization is responsible for all radiation and contamination surveys except personnel contamination surveys using installed monitoring equipment in the airlocks and at the egress points of the RCAs.

Surveying contaminated areas is performed to determine the level of contamination. Survey results are also used to determine if postings are correct, if additional controls are required, and to determine the appropriate personnel protective equipment. Contamination surveys, investigations, corrective actions, and reviews (along with deficiencies) are documented. These records are maintained for historical purposes including decommissioning activities. The radiological protection organization reviews this documentation for possible trends and needed corrective actions. Contaminated areas and contamination levels are tracked as part of ALARA goals along with decontamination efforts and results.

A surface is considered contaminated if either the removable or total surface contamination is above the levels in Table 9.2-1. Contamination surveys incorporate techniques to detect both removable and fixed contamination. Contamination survey results that indicate surface contamination above the levels of Table 9.2-1 are isolated, personnel are notified of the area isolation, and clean up activities initiated as soon as practical. Contaminated areas are posted and controls established to limit access until decontamination efforts are complete. Additional surveys are taken to validate the decontamination work.

Initially, contamination surveys (i.e., instrument, swipe and large-area wipes) are conducted in the Radiological Control Area established for the control of contamination, and other areas with the potential for becoming contaminated. After historical data have been collected, the frequencies of surveys are adjusted based on the need to perform surveys in those specific areas. Survey frequency adjustments are based on sufficient historical data to ensure adequate determination of conditions. Survey frequencies may be adjusted no more than annually or when at least ten consecutive surveys show no significant changes in radiation levels and contamination.

Radiation and contamination surveys records contain the individual area radiation readings including any locations that are in excess of the general area radiation levels, contamination locations including the level of contamination, the type and serial number of the instruments used in the performance of the survey, the name of the technician performing the survey, date and time of the survey, actions taken to mitigate and contamination levels, and the signature of the individual performing the review of the results..

To prevent internal contaminations, procedures and policies restrict eating, drinking, and smoking within the Radiological Control Area.

The MFFF design is an enclosed system and features low contamination estimates, which allows protective clothing requirements to be optimized. Depending on the contamination at the work location, the minimum type of clothing is either a lab coat for lab areas, or plastic (disposable) coveralls for minor maintenance.

Personnel wear protective clothing during the following activities:

- Handling contaminated materials with removable contamination in excess of prescribed levels
- Work in contamination, high contamination, and airborne radioactivity areas
- As directed by the radiological protection organization, or as required by an RWP.

In cases of skin contamination, decontamination is performed by radiological protection technicians, with wounds treated by the medical staff. As a minimum, nonabrasive methods, such as soap and water, are used. In cases of dry contamination or nondiscrete radioactive particles, masking tape is used. Personnel decontamination methods are provided as part of the radiation worker training and prior to the commencement of any actual decontamination effort.

Once materials or equipment have entered the Radiological Control Area, surveys are required before releasing material or equipment. See Table 9.2-1 for contamination limits.

When specific work activities are to be performed, areas are established around the work site so as to limit the spread of contamination. The area may be a small site covering only the component being worked on or may consist of the complete room where the work is being performed. In all cases a step off pad will be installed along with a buffer area where work personnel are able to remove protective clothing and check themselves for contamination. Normally the airlock self-monitoring equipment is used by the work force to survey themselves prior to exiting. However, if the work area is small, portable monitoring equipment may be installed to ease monitoring and movement of personnel.

Hampers are placed at the step off pads for the placement of used protective clothing upon removal. Additional protective clothing is available in the case of an individual becoming contaminated in the process of removing the clothing.

Protective clothing is stored in the Technical Support Building adjacent to the fuel fabrication structure. Personnel entering to perform maintenance will obtain the protective clothing as defined on the Radiation Work Permit and proceed to the work site. At the work site personnel will don the protective clothing and proceed to perform the work. Upon completion of the work or a break period, personnel will remove the protective clothing and place it in the designated hampers then self-monitor and proceed out of the fuel fabrication building.

Procedures are established for the donning and removal of protective clothing and personnel are trained as part of radiation worker training. Included in radiation worker training, personnel are instructed in the proper methods of self-monitoring and what actions to take should they determine that they are contaminated. Specific personnel decontamination procedures are established to provide basic decontamination with other methods provided by Savannah River Site SRS medical department in the case of gross contamination beyond the capabilities of the MFFF.

If an individual becomes contaminated in the process of performing any maintenance or normal activities, they are provided with a decontamination/first aid station in the Shipping and Receiving Building on the third floor between the personnel contamination monitoring rooms. This facility provides for the removal of contamination. If the contamination is of such an extent that there is a potential medical concern, SRS medical will be contacted and the individual transported to site medical for further decontamination. Decontamination liquids are contained and disposed of as liquid waste within the normal facility waste systems.

Dirty protective clothing is bundled for cleaning and laundry in accordance with established procedures. The dirty protective clothing is removed on an as needed basis for large work activities and at the end of the shift for small activities consuming small amounts of clothing. Clothing is bagged, surveyed for loose surface contamination on the exterior of the bag and transported to the waste handling area for placement in drums. The drums are then counted for activity, concentrations and weighted. The drums are then staged for transportation to the laundry facility.

9.2.7 Direct Exposure Control

Personnel working at the MFFF may be exposed to both photon and neutron radiation. The criteria for personal dosimetry are to:

- Measure both photon and neutron radiation from the primary isotopes of plutonium, uranium, and americium
- Provide reproducible results.

The direct exposure controls provide the following:

- Exposure monitoring
- Dosimeters and their processing
- Dose determinations
- Dose record maintenance
- Dose reporting
- Records maintenance.

The purpose of direct exposure controls is to ensure that the radiation worker doses do not exceed dose limits. Controls include:

- Measurement of the direct radiation dose received by workers using a dosimeter
- Control, as practical, of personnel who have received radiopharmaceuticals
- Planned special exposures
- Exposure limit for minors and the public
- Radiological protection for an embryo/fetus.

Personnel dosimetry is required for the following:

- Personnel who are expected to receive an annual external whole body dose greater than 100 mrem, or an annual dose to the extremities, or organs and other tissues (including lens of the eye and skin), greater than 10% of the corresponding limits specified in Table 9.1-2
- Declared pregnant workers who are expected to receive from external sources a dose equivalent of 50 mrem or more to the embryo/fetus during the gestation period
- Visitors, and public expected to receive an annual external whole body dose equivalent of 50 mrem or more in a year
- Minors for whom access and monitoring requirements are approved by the RPM
- Neutron dosimetry provided when a person is likely to exceed 100 mrem annually from neutrons.

Administrative goals are established to minimize the direct exposure of individuals. The Administrative goals are specified on Table 9.1-2. Individuals who in the course of their work approach the administrative limits are evaluated as to the continuation in the performance of that specific task that exposes the individual. In order that the possibility of exceeding the annual administrative goals is minimized, the goals are further subdivided into quarterly goals for ease of monitoring. The RPM, in coordination with the Vice President of Environmental Safety & Health (ES&H) Licensing may, upon a thorough review, authorize the individual to exceed the administrative limits. When an individual's exposure exceeds the administrative limits, a condition report will be initiated in accordance with MFFF procedures to ensure that the cause of the exposure is identified and corrective actions are implemented.

Thermoluminescent Dosimeters (TLDs) and Albedo (reflected) TLDs are the primary measuring devices at the MFFF. These dosimeters have the appropriate range and sensitivity to accurately measure exposures from plutonium and the other primary isotopes. Personal dosimeters are analyzed at a frequency described in approved procedures but not less than quarterly. Dosimetry is processed and evaluated by a processor accredited by the National Voluntary Laboratory Accreditation Program. TLDs are the source of exposure information for records. See Section 9.2.13 for exposure records. Radiation protection program policies and approved procedures establish action levels for personal dosimetry analyses results.

9.2.8 Internal Exposure Control

Internal exposure controls monitor workplace activities for potential and actual intakes of radioactive material. Both discretionary and nondiscretionary bioassay sampling are employed to monitor internal uptakes and to determine the quantity of the uptake. The bioassay program is conducted consistent with ANSI/HPS N13.22 criteria.

Baseline bioassay monitoring of personnel who are likely to receive intakes resulting in a CEDE greater than 100 mrem is conducted before they begin work that may expose them to internal radiation exposure. The 100 mrem action level is difficult to achieve; therefore, workplace monitoring is also used to identify potential intakes so that special bioassay monitoring can be initiated. All personnel who have or might receive more that 10% of the Annual Limits on Intake (ALI) will be subject to routine bioassay monitoring.

All personnel who in the course of their work have entered or been in a room where a CAM has alarmed or their lapel air sampler indicates that there is a potential uptake of material will be required to undergo bioassay monitoring upon indication of a potential uptake. The CAM air sample analysis or lapel air sample will be used to track the uptake and validate the results of the bioassay.

Routine bioassay monitoring methods and frequencies are established for personnel who in the course of their work handle radioactive materials or perform maintenance on radioactive systems. As a minimum personnel in the routine bioassay program will be required to submit bioassay samples (urinalysis and/or fecal as necessary) and receive whole body scans annually. If an individual is determined to have potentially received an uptake, additional bioassays will be required.

When an individual is suspected of receiving an uptake, a condition report will be initiated in accordance with MFFF procedures to ensure that the cause of the exposure is identified and corrective actions are implemented.

Termination bioassays are required when a person who participated in bioassay monitoring terminates employment.

Bioassay analyses are also performed when any of the following occurs:

- Facial or nasal contamination is detected that indicates a potential for internal contamination
- Airborne monitoring indicates the potential for intakes exceeding 100 mrem committed effective dose equivalent
- Upon direction of the radiological protection organization when an intake is suspected.

Levels of intakes that warrant the consideration of medical intervention are based on site-specific radionuclides. The effectiveness of medical intervention, such as blocking or chelating agents, is documented using bioassay results.

A preliminary assessment of the intakes detected is conducted prior to permitting an employee to return to radiological work.

Internal dosimetry relies on radionuclide standards from, or traceable to, the National Institute of Standards and Technology (NIST).

Summation of the internal dose includes the methodology that evaluates the doses from inhalation, oral ingestion, and an intake through wounds or absorption through skin.

Interpretation of bioassay results and subsequent dose assessments includes the following:

- Characteristics of the radionuclide, such as chemical and physical form
- Bioassay results and the person's previous exposure history
- Exposure information, such as route of intake, and time and duration of exposure
- Biological models used for dosimetry of radionuclides
- Models to estimate intake or deposition and to assess dose
- Minimal Detection Levels for the potential primary contaminants – plutonium, uranium, and americium – are based on implementation of ANSI/HPS N13.30-*1996 Performance Criteria for Radiobioassay*
- Coordination between the radiological protection organization and medical personnel for doses that may require medical intervention
- DAC and ALI values – presented in Table 1 of 10 CFR Part 20, Appendix B; used to determine the individual's dose and to demonstrate compliance with occupational dose limits
- In estimating exposure of individuals to airborne radioactive materials, the respirator protection factor for respiratory protection equipment worn is considered.

Radiation protection policies and approved procedures establish action levels for internal contaminations. Bioassays are documented in accordance with the QA controls. Bioassays analytical quality control is described in the appropriate laboratory manual. Analytical procedures are consistent with national or international consensus standards or have equivalent or superior performance to such methods based on industry accepted methodologies. Analytical instrumentation is standardized and calibrated in accordance with the manufacturer's recommendations. Calibration standards are traceable to NIST.

9.2.9 Summing of Internal and Direct Exposure

The maximum doses allowed for occupationally exposed workers are contained in 10 CFR §20.1201. These limits apply to radiation workers 18 years of age or older. These limits are expressed in units of dose equivalent (DE) in rem and Sv. Internal dose to a specific organ is given as committed dose equivalent (CDE), while the internal dose relative to a whole-body exposure is given as CEDE. Direct dose is expressed as deep dose equivalent (DDE), shallow dose equivalent (SDE), and lens of the eye dose equivalent (LDE). Extremities are considered to be the hand, elbow, arm below the elbow, foot, knee, and leg below the knee.

In accordance with 10 CFR 20.1202, the internal and external exposures are summed when applicable. Recording and reporting of radiation exposures are conducted as provided for in Regulatory Guide 8.7, Instructions for Recording and Reporting Occupational Radiation Exposure Date. Monitoring of occupational exposures is conducted in accordance with Regulatory Guide 8.34, Monitoring Criteria and Methods to Calculate Occupational Radiation Doses. Specific radiation protection procedures provide for the implementation of these Regulatory Guides.

The annual occupational exposure limits from 10 CFR Part 20 are:

- Total (CEDE + DDE) = TEDE 5 rem (0.05 Sv)
- Lens of Eye (LDE) 15 rem (0.15 Sv)
- Other Organs (CDE + DDE) 50 rem (0.5 Sv)
- Skin or Extremity (SDE) 50 rem (0.5 Sv).

9.2.10 Respiratory Protection

Using ALARA concepts, the use of respiratory protection is minimized to ensure that the TEDE dose is optimized for the work activity. Specialized training and a medical evaluation are required for individuals required to wear respiratory protection. Procedures direct the supervision and training of respirator users, fit testing, respirator selection, inventory and control, storage, issuance, repair, testing and quality control of respiratory equipment, recordkeeping and limitations on respirator use and duration. Respiratory protection records include the training of the individual user, duration of use, type of respirator, including the cartridge if required, and respirator maintenance.

It is MFFF policy to limit the intake of hazardous material by its workers to ALARA. Engineering and process controls (contamination control, use of containments, ventilation, and other technology) are used to the extent practical to minimize airborne hazards. When these are not practical to control levels below the appropriate limits for a hazard, the radiological protection organization will limit intake by control of access, limitation of exposure times, and use of respiratory protection equipment.

Respiratory protection is worn (unless ALARA analysis indicates TEDE for an operation would be lowered by not wearing respiratory protection) when air sample analysis indicates concentrations equal to or greater than the 20% of the DACs listed in 10 CFR Part 20, Appendix B.

Respiratory protection is selected to give a protection factor greater than the multiple by which the peak concentration exceeds the DACs listed in 10 CFR Part 20, Appendix B.

Use of respiratory protection is reduced to the minimum practical by implementing engineering controls and work practices to contain radioactivity at the source.

Equipment used is within limitations for type and mode of use and provides proper visual, communication, and other special capabilities (such as adequate skin protection), when needed.

Adequate numbers and locations of respiratory protection equipment are available.

9.2.11 Instrumentation

Fixed and portable radiological protection instrumentation used for the radiological protection program are calibrated and maintained to ensure accurate and reproducible results.

MFFF radiological protection equipment comprises a broad spectrum of analytical instruments used to determine the presence of radioactive material and to quantify the amount of contamination. Instrumentation ranges from gross measurements to specific isotopic analytical analyzers that can determine the constituents and quantity of each isotope. The instrumentation also includes installed personnel monitors and hand-held survey equipment.

Airborne contamination monitors are installed to detect barrier failure. These monitors are placed in each room where either personnel access is allowed or that contains the first confinement barrier. In rooms with no routine personnel access, airborne contamination monitors obtain air samples taken from the ventilation exhaust ducts exiting rooms (cells) as appropriate.

To ensure that workers are provided adequate monitoring, there may be more than one CAM in a room. The actual number of CAMs is determined based on the anticipated number of operations and the potential for an uptake. Where there is a potential for airborne contamination, a monitor is installed so that the workers are provided coverage. The initial number and location of monitors is based on MELOX and La Hague experience.

A person working in a glovebox (i.e., hands/arms extended into glovebox gloves) has an airborne contamination monitoring device (i.e., CAM) located in close proximity to the breathing air zone. To ensure coverage at glovebox workstations, some CAM sample heads are movable. In addition to the CAMs provided for workstations, CAMs are also strategically placed in routinely occupied areas surrounding gloveboxes. Readout and alarm monitors are located in the PUCR and the RM/RPR. The system also provides an alarm in the glovebox room and in the airlocks for the glovebox room if the airborne contamination exceeds preset limits. Portable CAMs are available for use during maintenance and provide additional coverage.

Alarm setpoints are provided at two distinct levels to enable the worker to take appropriate action if a release should occur. The lower (first) setpoint provides a local warning of increasing airborne contamination so that the worker can exit the room or don appropriate respiratory protection equipment. This alarm also warns other workers outside the room that there is an increase in airborne contamination and that they should not enter the room without respiratory equipment. The higher (second) alarm setpoint provides local alarm and readout, indicating that personnel are in danger and that immediate actions are required to provide protective measures to the workers. This setpoint is less than the 10 CFR Part 20, Appendix B limit, but above the warning level. The alarms have remote readouts in the PUCR and the RM/RPR so that the process can be terminated and corrective actions can be initiated to stop the release.

During maintenance activities when a glovebox or a system boundary is opened, portable air samplers are used to monitor personnel inside contamination control enclosures. The use of portable monitors allows for closer supervision of the airborne activity in the area of the work.

9.2.11.1 Types of Instrumentation

9.2.11.1.1 Alpha/Beta Counters

Due to the nature of plutonium, the ability to detect minute quantities of plutonium requires the use of sensitive equipment. The MFFF radiological protection equipment is capable of detecting extremely low levels of alpha contamination in a relatively short counting-time cycle.

The radiological protection laboratories, MP Area, and AP Area are equipped with alpha/beta counters to enable the processing of swipes and airborne contamination surveys on a continuous basis. Additional counters are located as necessary to support incoming radioactive material and shipments of waste, fuel, and excess materials.

9.2.11.1.2 Isotopic Analytical Equipment

The laboratories are equipped with instrumentation capable of quantifying the radioactive material on swipes, air samples, and other sample configuration. When necessary, the detector portion of the instruments is installed in counting shields to reduce the background effects and minimize background counts.

9.2.11.1.3 Personal Surveys Between Contaminated Areas

At transitions between contaminated areas, personnel are monitored for contamination. Personnel monitoring equipment is placed as close to the source as practical to ensure that contamination is controlled close to the source.

9.2.11.1.4 Whole Body Contamination Monitors

Prior to exiting MFFF production areas, personnel are surveyed at control points by multidetector personnel contamination monitors to ensure that no contamination leaves the area.

9.2.11.2 Instrument Calibration

Radiological instruments are used to measure the radiation for which their calibrations are valid and follow the requirements contained in ANSI N323 for radiological instrumentation calibration. Calibration sources are traceable to NIST.

Calibration procedures are developed for each radiological instrument type and include frequency of calibration, pre-calibration requirements, primary calibration requirements, periodic performance test requirements, calibration record requirements, and maintenance requirements.

Radiological instruments are calibrated based on instrument performance and manufacturer's recommendations. The effects of environmental conditions, including interfering radiation, on an instrument are known prior to use. Operational checks are performed on continuously operating radiation protection instruments at a frequency based on instrument performance and manufacturer's recommendations.

When necessary to use an instrument in an application other than that envisioned by the manufacturer, the instrument is adjusted, calibrated, and labeled to identify the special conditions and used only under the special conditions for which it was calibrated.

Instruments bear a label or tag with the date of calibration and date calibration expires.

Instruments whose "as found" readings indicate that the instrument may be out of calibration are reported to the radiological protection organization. The radiological protection organization reviews surveys performed with the instrument while it was out of calibration.

Calibration facilities perform inspections, calibrations, and performance tests, and select calibration equipment in accordance with the recommendations of ANSI N323, *Radiation Protection Instrumentation Test and Calibration*, and take the following actions:

- Locate calibration activities in a manner to minimize radiation exposure to operating personnel and to personnel in adjacent areas
- Minimize sources of interference, such as backscatter and non-ionizing radiation, during the calibration of instrumentation and correct for interference as necessary
- Operate in accordance with the referenced standards
- Generate records of calibration, functional tests, and maintenance in accordance with the referenced standards.

9.2.11.3 Instrument Maintenance

The radiological protection program includes preventive and corrective maintenance of radiological instrumentation. Preventive and corrective maintenance are performed using components and procedural recommendations at least as stringent as those specified by the manufacturer of the instrument. Radiological instruments undergo calibration prior to use and following preventive or corrective maintenance, or adjustment that voids the previous calibration. A battery change is not considered maintenance.

9.2.11.4 Radiological Protection Work Areas and Labs

The radiological protection working spaces consist of radiological protection laboratories and a radiological protection storage room, which contain instruments and areas where technicians may prepare their survey results and store hand-held instruments. These laboratories contain multisample alpha/beta counters, as well as hand-held survey instruments, portable air samplers and isotopic analyzers. The space allows personnel to perform surveys, count the samples, perform isotopic analyses, and record results.

Level 3 of the Shipping and Receiving Area contains the access control point into the Radiological Control Area, which serves as the egress point for both the MP and AP Areas. This area has the personnel contamination monitors and the Decontamination Area/Contaminated First Aid Area. The Decontamination Area/Contaminated First Aid Area contains a shower and sinks to perform minor decontamination of individuals, and supplies to treat minor injuries.

The Technical Support Building has three rooms dedicated to radiological protection activities:

- **RM/RPR**– Houses the respiratory equipment and issue area for the MFFF. This room provides for the minor repair of respiratory protection and storage of spare equipment and emergency supplies.
- **Clean Anti-Contamination Storage Room** – Provides storage for anti-contamination clothing to be used during maintenance activities.
- **Locker Room Area** – Contains storage racks for respiratory protection and dosimetry devices. Space is provided for an increase in staff during maintenance outages.

In the RPCA and the RM/RPR, there are visual displays of alarms and radiation levels for the MFFF radiation monitoring equipment. These visual displays provide identification of specific alarms and the locations of the radiation monitors in the workplace.

The radiation monitoring system uses trending software to identify increasing direct radiation levels over a period of time. The system provides the initial warning of increasing radioactivity in gloveboxes and production rooms and releases to the environment.

9.2.12 Significant Exposure or Contamination Response Capabilities

Personnel assigned to MFFF have dosimetry (activation foils installed in the SRS security badges) that can be used to determine if significant exposures have occurred. Personnel within the MFFF process areas wear a TLD, and an electronic pocket dosimeter. The electronic pocket dosimeter may be exposed to an excessive amount of radiation beyond the capabilities of the instrument. In that case, significant exposure dosimetry will be used to quickly identify personnel with high levels of exposure. Response personnel are trained to survey personnel, including significant exposure dosimetry, for indications of significant exposures. TLDs can be rapidly processed for a more accurate exposure determination. The combined readings are then used to determine the necessity of long-term medical treatment.

Personnel involved in a significant exposure event will initially be transported to the Decontamination/First Aid Room located in the Shipping and Receiving Building. MFFF radiological protection staff will then initiate treatment and decontamination efforts to remove gross amounts of contamination as necessary.

SRS staff physicians and nurses are trained in the proper treatment of high levels of exposure and contamination. The SRS is equipped with medical facilities, ambulances, and technicians to rapidly provide appropriate medical treatment.

MFFF radiation protection procedures direct personnel to specific actions if an accident occurs and personnel are exposed to extremely high levels of radiation and/or contamination. In coordination with the SRS medical personnel, mitigating actions will be implemented as soon as possible. Personnel will be transported to the SRS medical facilities or designated off site hospitals following initial efforts to remove gross amounts of contamination.

9.2.13 Exposure Records

Complete and accurate radiological protection records of areas, including the records of individuals who work in or visit them, are maintained in accordance with 10 CFR Part 20, Subpart L. Reports are formatted in accordance with 10 CFR §20.2110. These records are used to document the radiation exposures of individuals and are available as prescribed by the Privacy Act of 1974. These records are also used for (1) evaluation of the effectiveness of the radiological protection program, (2) demonstration of compliance with regulations and requirements, and (3) personnel records. These dose records are sufficient to evaluate compliance with applicable dose limits, and monitoring and reporting requirements. Occupational exposures are reported annually to the NRC as required by 10 CFR 20.2206(b) as well as reporting exposures of individuals exceeding dose limits in accordance with 10 CFR 20.2205.

As a minimum, exposure reports are provided to individuals under the following conditions:

- Upon request from an individual terminating employment, records of exposure are provided to that individual when the data become available.
- If requested, a written estimate of radiation dose, based on available information at the time of termination, is provided.
- Annual radiation dose reports are provided to individuals monitored during the year.
- If requested, detailed exposure information is provided.

- Reports are provided to individuals when required to report to the NRC pursuant to occurrence reporting and processing, or planned special exposures.

9.2.14 Additional Program Elements

Occupational exposures in excess of prescribed limits are referred to the corrective action program (see Section 15.7.1).

Internal audits of the radiological protection program shall be conducted such that over a one-year period, all functional elements are assessed for program performance, applicability, content and implementation. The audits may be performed by the radiation safety staff or the quality assurance organization.

The following functional elements shall be in the assessment program:

- Personnel dosimetry and dose assessment
- Portable and fixed instrumentation
- Contamination control
- Radiological monitoring (area and item monitoring)
- ALARA program
- Accident and emergency dose controls
- Radioactive material control, including sealed radioactive source control and material release
- Entry controls
- Training
- Posting and labeling
- Records and reports
- Radiological design and administrative controls.

Concerns identified in the assessments shall be incorporated into the MFFF corrective action program.

9.3 DESIGN BASIS

The following information represents the design basis attributes for radiation safety:

- Facility management will implement a quality radiation protection program consistent with Regulatory Guide 8.8 C.1.a &b.

- Occupational doses are ALARA through engineering design features and management controls implemented during operation.

- Dose assessments are performed using guidance from U.S. Nuclear Regulatory commission (NRC) Regulatory Guide 8.19, *Occupational Radiation Dose Assessment in Light-Water Reactor Power Plant – Design Stage Man-Rem Estimates*, and Regulatory Guide 8.34, *Monitoring Criteria and Methods to Calculate Occupational Radiation Doses*.

- The direct dose assessment is determined by dose rate analyses and a dose assessment process called the ABAQUES Method.

- Environmental Safety & Health (ES&H) Licensing is responsible for reviewing facility or process modifications to maintain exposures ALARA.

- Calculations are performed using shielding material properties contained in American National Standards Institute/American Nuclear Society (ANSI/ANS) – 6.4.2 – 1985, R1997.

- Design goals for TEDE, which were established early in the design process for individual workers are applied to facility operations (see Table 9.1-2).

- Shielding calculations are performed using several standard industry computer codes (Monte Carlo N-Particle [MCNP], SCALE, Perceval, SN1D).

- Dose rate estimates are determined using flux-to-dose conversion factors contained in ANSI 6.1.1-1977, *Neutron and Gamma-Ray Fluence-to-Dose Factors*.

- Radiation protection design features are provided in accordance with Regulatory Guide 8.8, *Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable*.

- ALARA cos-benefit analyses are performed in accordance with NUREG/CR-0446, *Determining Effectiveness of ALARA Design and Operational Features*.

- Personnel entering the RCAs are required to read and understand the Radiation Work Permit (RWP) requirements and monitor their exposures during the conduct of their activities.

- RWPs are issued and controlled in accordance with approved radiological protection procedures for activities conducted within the RCAs.

- Air sampling equipment is used in occupied areas where, under normal operating conditions, a person is likely to receive an annual intake of 2% or more of the specified annual limit on intake (ALI) value (40 Derived Air Concentration [DAC]-hours).

- CAMs are installed in rooms where radioactive materials are handled or there is a need to ensure that there is no airborne contamination present.

- The bioassay program is conducted consistent with ANSI/HPS N13.22 criteria.

- Minimal Detection Levels for the potential primary contaminants – plutonium, uranium, and americium are based on implementation of ANSI/HPS N13.30-*1996 Performance Criteria for Radiobioassay*.

- Recording and reporting of radiation exposures are conducted as provided for in Regulatory Guide 8.7, *Instructions for Recording and Reporting Occupational Radiation Exposure Data*.

- Monitoring of occupational exposures is conducted in accordance with Regulatory Guide 8.34, *Monitoring Criteria and Methods to Calculate Occupational Radiation Doses*.

- Respiratory protection is worn (unless ALARA analysis indicates TEDE for an operation would be lowered by not wearing respiratory protection) when air sample analysis indicates concentrations equal to or greater than the 20% of the DACs listed in 10 CFR Part 20, Appendix B.

- Radiological instruments are used to measure the radiation for which their calibrations are valid and follow the requirements contained in ANSI N323 for radiological instrumentation calibration. Calibration sources are traceable to NIST.

- Calibration facilities perform inspections calibrations and performance tests, and select calibration equipment in accordance with the recommendations of ANSI N323, *Radiation Protection Instrumentation Test and Calibration*.

Table 9.1-1. MFFF Radiation Zoning Criteria

Zone	Design Basis Maximum Area Radiation Dose Rate (mrem/hr)
Z1 - High access area	<0.05
Z2 - Intermediate access area	<0.25
Z3 - Low access area	<5.0
Z4 - Very low access area	<100
Z5 - Restricted access area	>100

Table 9.1-2. Summary of Dose Limits and Goals

	10CFR20 Limits	Administrative Goals
General Employee: Whole Body (internal CEDE + external EDE) (TEDE)	5 rem/yr	0.5 rem/yr
General Employee: Lens of Eye 15 rem (LDE)	15 rem/yr	10 rem/yr
General Employee: Skin and extremities (external shallow dose) (SDE)	50 rem/yr	10 rem/yr
General Employee: Any organ or tissue 50 rem (other than lens of eye) and skin	50 rem/yr	5 rem/yr
General Employee: Soluble uranium intake	10 mg/week	1 mg/week
Declared Pregnant Worker: Embryo/Fetus (TEDE)	0.5 rem/gestation period	0.5 rem/gestation period

Notes:

1. The annual limit of dose to “any organ or tissue” is based on the committed dose equivalent to that organ or tissue resulting from internally deposited radionuclides over a 50-year period after intake plus any deep dose equivalent to that organ from external exposures during the year.
2. Exposures due to background radiation, as a patient undergoing therapeutic and diagnostic medical procedures, and participation as a subject in medical research programs shall not be included in either personnel radiation dose records or assessment of dose against the limits in this table.
3. Whole body dose (TEDE) = effective dose equivalent from external exposures + committed effective dose equivalent from internal exposures.
4. Lens of the eye dose equivalent = dose equivalent from external exposure determined at a tissue depth of 0.3 cm.
5. Shallow dose equivalent = dose equivalent from external exposure determined at a tissue depth of 0.007 cm
6. The soluble uranium intake limit is in consideration of the chemical toxicity.
7. Minors (below age 18) are allowed to enter radiation areas only with RPM permission. Dose limits for minors will be in accordance with 10 CFR §20.1207.

Table 9.2-1. Summary of Contamination Values

Radionuclide[1]	Removable[2] (dpm/100 cm^2)	Total[3] (Average) (dpm/100 cm^2)
U-natural, ^{235}U, ^{238}U, and associated decay products	1,000 alpha	5,000 alpha
Transuranics (including Pu isotopes), ^{226}Ra, ^{228}Ra, ^{230}Th, ^{228}Th, ^{231}Pa, ^{227}Ac	20	100
Th-nat, ^{232}Th, ^{223}Ra, ^{224}Ra, ^{232}U	200	1000
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission). Includes mixed fission products containing ^{90}Sr [4,5]	1,000 beta-gamma	5,000 beta-gamma
Tritium and tritiated compounds	10,000	N/A

Notes:

1. Except as noted in Footnote 5 below, the values in this table apply to radioactive contamination deposited on, but not incorporated into the interior of, the contaminated item. Where contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for the alpha- and beta-gamma-emitting nuclides apply independently.
2. The amount of removable radioactive material per 100 cm^2 of surface area shall be determined by swiping the area with dry filter or soft absorbent paper while applying moderate pressure and then assessing the amount of radioactive material on the swipe with an appropriate instrument of known efficiency. (Note: The use of dry material may not be appropriate for tritium.) For objects with a surface area less than 100 cm^2, the entire surface shall be swiped, and the activity per unit area shall be based on the actual surface area. It is not necessary to use swiping techniques to measure removable contamination levels if direct scan surveys indicate that the total residual contamination levels are below the values for removable contamination.
3. The levels may be averaged over 1 square meter provided the maximum activity in an area of 100 cm^2 is less than three times the values in the table.
4. This category of radionuclides includes mixed fission products, including the ^{90}Sr, which is present in them. It does not apply to ^{90}Sr that has been separated from the other fission products or mixtures where the ^{90}Sr has been enriched.
5. These values shall be applied to total $^{90}Sr/^{90}Y$ activity resulting from the presence of ^{90}Sr in mixed fission products.

10.0 ENVIRONMENTAL PROTECTION

The components of the Mixed Oxide (MOX) Fuel Fabrication Facility (MFFF) Environmental Protection Program include:

- Radiation safety controls to assess the level of radioactive releases to the environment, maintain public dose as low as is reasonably achievable (ALARA), minimize facility and environmental contamination, facilitate eventual deactivation, and minimize waste generation
- Effluent monitoring to measure and monitor radioactive effluents released from the facility during normal and off-normal operations
- Environmental surveillances to monitor environmental impact from operations during normal and off-normal operations.

10.1 RADIATION SAFETY

This section describes the methods used to maintain dose outside the Restricted Area Boundary (RAB) ALARA, in accordance with Title 10 of the Code of Federal Regulations (CFR) §20.1101. Facility radiation safety is described in Chapter 9.

The radiation protection organization is responsible for the analysis of the MFFF Main Exhaust Stack samples. The radiation protection organization is also responsible for sampling all internal areas of the MFFF for airborne contamination, analyzing the radioactive liquid waste prior to transfer to the Savannah River Site facilities, and providing the surveys of the solid waste drums prior to shipment.

10.1.1 ALARA Goals for Effluent Control

Calculations performed in accordance with 10 CFR §20.1302(b)(1) using the guidance provided in Regulatory Guide 4.20, Section 2.2, demonstrate that the Total Effective Dose Equivalent (TEDE) to an individual outside of the RAB likely to receive the highest dose from licensed operation does not exceed 100 mrem/yr, as required in 10 CFR §20.1301(a)(1).

The ALARA goal for TEDE to the individual outside of the RAB likely to receive the highest dose from air emissions of radioactive material to the environment during normal operations, excluding ^{222}Rn and its daughters, is less than 10 mrem/yr, which is 10% of dose stated in 10 CFR §20.1301(a)(1) and is consistent with 10 CFR §20.1101(d). Reports are made in accordance with 10 CFR §20.2203 if the 10 mrem/yr dose constraint is exceeded during all operating conditions.

No radioactive liquid effluents are predicted or anticipated for normal operations.

Dose estimates are monitored and compared to ALARA goals. CB&I AREVA MOX Services LLC (MOX Services) management is apprised of data in accordance with the ALARA program.

10.1.2 Effluent Controls

Effluent controls, consisting of airborne, liquid, and solid waste management, reduce exposure to individuals outside of the RAB and minimize releases to the environment.

10.1.2.1 Control of Airborne Emissions

Airborne emissions are controlled by the heating, ventilation, and air conditioning (HVAC) system and the Offgas Treatment (KWG) unit ventilation system that removes radionuclides, nitrous fumes, and other hazardous materials from the Aqueous Polishing (AP) process systems offgas. Airborne waste from MFFF processes is routed through the HVAC system. The HVAC system is designed to handle the expected volume of potentially radioactive waste, compartmentalize the airborne waste to reduce the potential for cross-contamination, safely handle the chemical characteristics of the airborne waste, achieve an acceptable decontamination factor for each radionuclide, and be capable of safe shutdown consistent with the operating status. Several design features of the HVAC system support specific areas of the facility, such as the MOX Processing (MP) and AP Areas. These features include items relied on for safety (IROFS) to provide for confinement of radioactive materials. Ventilation exhaust from contaminated gloveboxes is passed through multiple banks of filters, including high-efficiency particulate air (HEPA) filters. The arrangement and control of IROFS ensure that contaminated exhaust does not bypass confinement controls.

Airborne emissions are monitored and controlled to maintain dose outside the RAB ALARA.

Trending results from effluent monitors, samplers, and other MFFF airborne monitoring equipment provide early indication of increased radioactivity in ventilation exhaust. Procedures identify evaluations and actions to be taken when the concentrations of airborne radioactivity exceed prescribed limits.

10.1.2.2 Liquid Waste Management

The AP process uses recycling to the maximum extent practical to minimize liquid waste. Liquid waste management is integrated into the fluid transport systems. The fluid transport systems are designed to handle the maximum expected volume of potentially radioactive waste, compartmentalize the liquid waste to reduce the potential for cross-contamination, safely handle the chemical characteristics of the liquid waste, and be capable of safe shutdown consistent with the operating status. Liquid radioactive waste is transferred to U.S. Department of Energy (DOE) facilities at the Savannah River Site (SRS) in a manner consistent with the SRS Waste Acceptance Criteria (WAC) for appropriate storage and disposition. SRS will take possession of the waste prior to reaching the RAB and is responsible for the safe movement of the waste.

Liquid radioactive wastes and liquid nonradioactive wastes are collected and managed in separate systems that have no opportunity for interconnection. Radioactive process fluids are maintained within at least two levels of confinement. Radioactive process fluids are transferred using means such as gravity flow, airlifts, air jets, and steam jets, when practical. Drains within the radiation control area are routed to the liquid waste system. Liquid radioactive wastes are collected in the aqueous liquid waste system or the solvent liquid waste system, and are sent to

SRS for disposition. Outside the radiation control area, liquid nonradioactive wastes are collected and sent to SRS for disposition. Systems containing nonradioactive hazardous fluids are of fully-welded construction and are accessible for inspection.

Prior to transfer to SRS, liquid wastes from storage tanks are sampled and analyzed to ensure that waste transfers meet the SRS WAC.

10.1.2.3 Solid Waste Management

Solid wastes are transferred to SRS for disposition. MFFF quantifies the activity in radioactive solid waste containers to ensure that waste shipments meet the SRS WAC.

Hazardous solid waste is waste that is, or contains, a listed hazardous waste, or that exhibits one of the four U.S. Environmental Protection Agency (EPA) hazardous waste characteristics (i.e., ignitability, corrosivity, reactivity, and toxicity). Hazardous waste includes nonradioactive laboratory wastes. Mixed low-level waste is waste that is radioactive and contains chemical components regulated by EPA as hazardous waste, while mixed transuranic waste is waste that meets the criteria for transuranic waste and contains chemical components regulated by EPA as hazardous waste.

Mixed low-level waste and mixed transuranic waste are packaged and transferred to SRS in a manner consistent with the SRS WAC for processing and disposal within 90 days of generation. SRS will take possession of the waste prior to reaching the RAB and is responsible for the safe transfer of the waste. To the extent practical, commingling of waste from streams requiring different treatment technologies is prevented. Containers of hazardous waste known or suspected to be contaminated with radioactive material are uniquely labeled and tracked through storage and shipping.

10.1.3 ALARA Reviews and Reports to Management

Reports summarizing the ALARA program are provided to MOX Services management. They include trending information, so that analytical results can be compared to ALARA goals. Emission and effluent radionuclide concentrations and radionuclides transferred to SRS as liquid and solid waste are included in trend analyses. Abnormal increases in the trend of analytical results are reported to MOX Services senior management as soon as practical. To ensure that releases are maintained ALARA, MOX Services management is informed quarterly of the trends measured against ALARA goals. ALARA goals are reevaluated annually, and new goals are established for the upcoming year as appropriate. Recommendations are made to MOX Services senior management, as needed, for changes in facilities and procedures to achieve ALARA goals. Effluent controls are reviewed annually as part of the radiological protection program annual review to ensure public doses are ALARA.

If an adverse trend is noted, an evaluation is made to determine if a detrimental effect is evident in the environment or the surrounding biota. The evaluation considers the information provided by the environmental surveillance network. Based on facility operating history and the data obtained from environmental surveillances during operations, the sampling and/or analysis programs are adjusted to optimize reliability.

10.1.4 Waste Minimization and Pollution Prevention

Waste management is guided by the principles of ALARA, waste minimization, and pollution prevention. Waste minimization is accomplished through a design that reduces the potential for waste generation, and an operations philosophy that minimizes the introduction of excess materials that can become contaminated.

The MFFF process implements recycling and reuse for waste minimization. For example, the recycling process minimizes the quantity of plutonium in the final waste by using systems that return (recycle) radioactive material to previous steps of the main process. The distribution and retention of radioactive materials throughout plant systems is reduced by using vacuum systems in the gloveboxes. Airborne dust is collected in dust pots in dedusting systems installed in the gloveboxes, and the material is recycled. Liquid waste is minimized in the AP process by use of recycling to the maximum extent practical. Nitric acid is recovered by evaporation from the process and partly reused as reagent feedstock for the plutonium dissolution subprocess. Distillates from the evaporation process are collected and partly reused in the process. Spent solvent from the plutonium separation step is regenerated by washing with sodium carbonate, sodium hydroxide, and nitric acid to remove degradation products from organic compounds, including trace amounts of plutonium and uranium.

Waste minimization procedures will require separation and segregation of solid and liquid wastes and the removal of packing and shipping materials prior to entry into contaminated areas. Waste minimization reduces worker and public exposure to radiation and to radioactive and hazardous materials.

Waste minimization programmatic documentation includes a statement of senior management support and identification of management, employees, and organizational responsibilities for waste minimization. Waste minimization includes periodic characterization of waste and assessment of waste management practices to identify opportunities to enhance waste minimization. Goals for waste minimization are established based on operational data. To ensure that waste generation is minimized, management is informed quarterly of the trends measured against waste minimization goals. The goals are reevaluated annually, and new goals are established for the upcoming year as appropriate. Recommendations are made to MOX Services senior management, as needed, for changes in facilities and procedures to achieve waste minimization goals.

10.2 EFFLUENT MONITORING

10.2.1 Air Emissions

The maximum annual concentrations of radioactive airborne effluents are expected to be much less than the values in 10 CFR Part 20, Appendix B, Table 2. Estimated isotopic distribution of emissions is shown in Table 10.2-1. MOX Services does not plan to request U.S. Nuclear Regulatory Commission (NRC) approval to adjust effluent concentrations shown in 10 CFR Part 20 Appendix B; therefore, physical and chemical properties are not described here.

10.2.1.1 Discharge Locations

Exhaust from MFFF processes is filtered and discharged to the environment via a stack located on top of the MOX Fuel Fabrication Building.

10.2.1.2 Sample Collection, Frequency, and Analytical Methods

Based on Regulatory Guide 4.16, Revision 2, a representative sample of the particulate effluent from the stack is continuously collected during operations. The representative sample is collected on a filter for determination of quantities and average concentrations of principal radionuclides that are released. The analytical methodologies used to characterize airborne emissions are listed in Table 10.2-2.

To investigate abnormal stack releases and/or anomalies, sample connections are installed at key locations in process area ventilation ducts. The placement and use of sample connections are based on minimizing the risk to facility workers, site personnel, and members of the public. The potential for leakage from process systems, equipment, and confinement is also considered. The evaluation focuses on the equipment and spaces with the highest potential for leakage of airborne contaminants. During MFFF operations, elevated readings from continuous air monitors (CAMs) are used to identify the need to perform maintenance, or to take other action to reduce effluent releases. To quantify the contribution from each source, CAMs sample the discharged air from the MP and AP process areas and, as appropriate, other areas that are not used for processing special nuclear material.

Analytical quality control methodology is described in the appropriate laboratory manual and is subject to Quality Assurance controls. Analytical procedures are consistent with national or international consensus standards or have equivalent or superior performance to such methods. Analytical instrumentation is standardized and calibrated in accordance with the manufacturer's recommendations. Calibration standards are traceable to the National Institute of Standards and Technology.

10.2.2 Liquid Effluents

Liquid radioactive waste is collected by the liquid aqueous liquid waste system or the solvent liquid waste system and transferred to SRS for disposition. The MFFF does not discharge radioactive liquid to the environment during normal and off-normal operations. The expected nonradioactive liquid release is from stormwater that is released to the storm drains and water from HVAC noncontact condensate that is released to the sanitary sewerage facilities system.

10.2.2.1 Discharge Locations

The MFFF does not discharge process effluents. The National Pollutant Discharge Elimination System (NPDES) discharge for stormwater runoff is designated in South Carolina Department of Health and Environmental Control (SCDHEC) NPDES Construction and Industrial General Permit and related documents (e.g., Storm Water Pollution Prevention Plan).

10.2.2.2 Leak Detection Systems for Ponds, Lagoons, and Tanks

The MFFF does not use wastewater treatment ponds, lagoons, or other process water holding ponds. The only pond on the MFFF site is the stormwater detention basin, which does not receive process liquid discharges from the MFFF. Tanks used for storage of radioactive material are located inside MFFF buildings and are equipped with drip pans and leak detection.

10.2.3 Recording/Reporting Procedures

Data from the sampling and monitoring are reviewed on a regular basis. Radionuclide activities are trended over a period of time at each sampling location for each media to determine the effects of facility operation. If an increasing trend is noted, an evaluation is made to determine if a detrimental effect has been seen in the environment or in the surrounding biota. The appearance of an increasing activity trend in itself is not cause for action. Based upon the operating history of the facility and operational data, sampling and/or analysis programs are adjusted as necessary.

MOX Services submits a summary of the effluent monitoring to the NRC semiannually.

Table 10.2-1. Estimated Radiological Releases from the MFFF during Normal Operations

Isotope	Airborne Radiological Releases(μCi/yr)
^{3}H	3.0E+6 [1]
^{237}Np	7.2E-04
^{236}Pu	1.3E-08
^{238}Pu	8.5
^{239}Pu	91
^{240}Pu	23
^{241}Pu	101
^{242}Pu	6.1E-03
^{241}Am	48
^{234}U	5.1E-03
^{235}U	2.1E-04
^{238}U	0.012

Note 1: Value is based on revision of feedstock specifications.

Table 10.2-2. Analytical Methods for Characterization of Airborne Emissions

Parameter	Analytical Method	Lower Limit of Detection[1] (μCi/ml)
Gross alpha	Gas-flow proportional counter	1.0E-15 [2]
Gross beta	Gas-flow proportional counter	1.0E-15 [2]
^{3}H	Liquid scintillation	5.0E-09
^{237}Np	Alpha spectrometer	5.0E-16
^{241}Am	Alpha spectrometer	1.0E-15
^{238}Pu	Alpha spectrometer	1.0E-15
^{239}Pu, ^{240}Pu	Alpha spectrometer	1.0E-15
^{235}U	Alpha spectrometer	3.0E-15
^{238}U	Alpha spectrometer	3.0E-15

Note 1: Lower limit of detection values are 5% of the values in 10CFR20, Appendix B, Table 2.

Note 2: It is estimated that this LLD can be met based on design basis, which is susceptible to change.

10.3 ENVIRONMENTAL SURVEILLANCES

Environmental surveillances assess the environmental impact of licensed activities, which include preoperational and operational environmental monitoring activities. Radionuclide analyses are performed more frequently if there is an unexplained increase of gross radioactivity in airborne emissions, or when a process change or other circumstance might cause a variation in radionuclide concentration.

Radiological impacts to the environment from airborne emissions during operation of the MFFF are expected to be minimal. Because the MFFF does not discharge radioactive liquids directly to the environment, the environmental surveillances focus on the environmental media impacted by the airborne pathway for the anticipated types and quantities of radionuclides released from the facility.

10.3.1 Pathway Analysis Methods to Estimate Public Dose

As noted above, the MFFF does not release radioactive effluents to the aquatic environment. Consequently, the pathways for radionuclides to reach the public or environment are associated with airborne emissions. The dominant pathway for MFFF releases to reach human consumption is inhalation of airborne emissions. Deposition of airborne particulates on crops and ingestion of the contaminated agricultural products is a secondary pathway for radionuclides to reach the environment and human consumption. However, because the MFFF is located on a DOE reservation, there are no consumable crops within 5 miles of the MFFF. A tertiary pathway is deposition of airborne particulates to water, or contaminated runoff to nearby streams and ingestion of the water or fish. Again, since the MFFF is on a DOE reservation, the importance of this pathway is significantly reduced. The analysis of public dose considers inhalation uptake, external exposure to the airborne plume, ingestion of terrestrial foods and animal products, and inadvertent soil ingestion.

10.3.2 Environmental Media to be Monitored and Sample Locations

The environmental surveillances track each pathway for the release of MFFF radioactivity to the environment. Environmental surveillances include monitoring of airborne particulates and deposition of particulates on surrogates for crops, such as grass and soil, and nearby streams. Environmental surveillances evaluate the effects of both short-term and long-term deposition.

Locations and sampling frequencies during operations phase monitoring are adjusted, based on the results of the preoperational surveillances or operational emissions monitoring results.

10.3.3 Preoperational Surveillances

The DOE has monitored the SRS site for many years. MFFF preoperational environmental surveillances provide a link between the long-term DOE data and the MFFF operational environmental surveillances. Preoperational environmental surveillances begin approximately two years prior to production of commercial fuel. The objectives of the preoperational environmental surveillances are:

- Establish a baseline of existing radiological and biological conditions at and nearby the MFFF site
- Evaluate procedures, equipment, and techniques used in the collection and analysis of environmental data, and train personnel in their use
- Determine the presence of contaminants that could be a safety concern for personnel.

Preoperational surveillances establish a baseline for operational environmental surveillance for radioactivity levels of environmental media (e.g., air, soil, sediments, and vegetation), as appropriate, with analyses for uranium, plutonium, and other radionuclides of interest.

10.3.3.1 Air Sampling and Analysis

Preoperational air quality sampling establishes the baseline to be used during the operational monitoring period. The airborne monitoring provides a comprehensive baseline of radiological conditions related to airborne emissions in the environs of the MFFF. The airborne radiological monitoring program, including the sampling locations, is outlined in Table 10.3-1.

Preoperational monitoring is used to establish the baseline for both isotopic composition and concentrations, which are then compared to observations during MFFF operations. Environmental observations are evaluated in conjunction with MFFF emissions data and atmospheric transport and diffusion modeling projections.

Air quality monitoring points are subject to emissions from not only the MFFF, but also from other nearby SRS operations. Accordingly, three additional air sampling locations, corresponding to existing SRS monitoring points to monitor exposure at the SRS boundary had been considered to assist in estimating dose to the offsite public, conservatively assuming a member of the offsite public spends all their time at the SRS boundary. These additional sampling locations are no longer necessary due to the termination of the nearby F-Canyon and FB-Line operations.

Analytical methods and lower limit of detection (LLD) for analyses of airborne isotopes are listed in Table 10.3-2. For rainwater samples, the rainwater is evaporated and then the dry material is counted. Sufficient volumes of samples are collected to ensure the attainment of LLD thresholds in the analysis. Samples are processed and packaged in a manner to ensure the integrity of each sample.

10.3.3.2 Water Sampling and Analysis

The MFFF does not discharge process water to the environment. Deposition rates of airborne contaminants to water bodies are estimated based on airborne environmental surveillances and confirmed by water and sediment sampling.

10.3.3.3 Terrestrial Sampling and Analysis

Preoperational terrestrial radiological monitoring is outlined in Table 10.3-3. It provides a comprehensive baseline of radiological conditions related to airborne emissions in the environs of the MFFF.

Soil samples are collected, using hand augers or equivalent devices, from uncultivated and undisturbed areas. Grassy vegetation is collected at locations adjacent to the soil sample by hand picking vegetation.

Analytical methods and LLDs for terrestrial environmental samples are listed in Table 10.3-4. Sufficient volumes of samples are collected when available, using accurate sample collection methods to ensure the attainment of LLDs in the analyses. Samples are processed and packaged in a manner to ensure the integrity of each sample.

10.3.4 Operational Monitoring

Locations and sampling frequency during the operational monitoring period may be altered based on the results of the preoperational monitoring or operational emissions monitoring results. The frequency of the monitoring described in this section may be reduced when a consistent radionuclide composition in effluents is established.

10.3.4.1 Air Sampling and Analysis

Operational air quality sampling is based on the results of preoperational and emission monitoring. The operational airborne radiological monitoring is outlined in Table 10.3-5.

Analytical methods and LLDs are listed in Table 10.3-6. For rainwater samples, the rainwater is evaporated and then the dry material is counted. Sufficient volumes of samples are collected to ensure the attainment of LLDs in the analyses. Samples are processed and packaged in a manner to ensure the integrity of each sample.

10.3.4.2 Water Sampling and Analysis

The MFFF does not discharge process water to the environment. Deposition rates of airborne contaminants into water bodies are estimated based on airborne environmental surveillances and confirmed by water and sediment sampling.

10.3.4.3 Terrestrial Sampling and Analysis

Operational terrestrial radiological monitoring is outlined in Table 10.3-7. It provides an evaluation of radiological impacts related to deposition of airborne emissions in the environs of the MFFF. Terrestrial samples are collected in the vicinity of the air quality monitors to allow association of the particulate and rainwater analyses with vegetation analyses.

Soil samples are collected, using hand augers or equivalent devices, from uncultivated and undisturbed areas. Grassy vegetation is collected at locations adjacent to the soil sample by hand picking vegetation.

Analytical methods and LLDs for analyses of terrestrial environmental samples are listed in Table 10.3-8. Sufficient volumes of samples are collected when available, using accurate sample collection methods to ensure the attainment of LLDs in the analyses. Samples are processed and packaged in a manner to ensure the integrity of each sample.

10.3.5 Action Levels and Actions

Title 10 CFR §20.1301 establishes regulatory limits for dose to the public. To ensure that the regulatory limits are not exceeded, MOX Services has established administrative limits and action levels as shown in Table 10.3-9. If an action level is exceeded for sampling, an investigation is performed to determine the source of the elevated activity. Emission data are trended as an analytical tool.

10.3.6 Recording/Reporting Procedures

Data from the sampling are reviewed on a regular basis.

Radionuclide activities are trended at each sampling location for each media to determine the effects of facility operation. If an increasing trend is noted, an evaluation is performed to determine if a detrimental effect has been seen in the environment or in the surrounding population.

Based upon the operating history of the facility and operational data, sampling and/or analysis programs are adjusted as necessary.

Results of the environmental surveillances are summarized annually.

Reports and notifications of theft or loss of licensed material are submitted as required. Reports and notifications of concentrations of principal radionuclides released are provided, and include the minimum detectable concentration for the analysis. Reports and notifications of exposure incidents above acceptable levels are submitted as required.

10.3.7 Monitoring Procedures, Analytical Methods, and Instrumentation

Analytical quality control is described in laboratory procedures and is consistent with the MOX Project Quality Assurance Plan. Analytical procedures are consistent with national or international consensus standards or have equivalent or superior performance to such methods. Analytical instrumentation is standardized and calibrated in accordance with the manufacturer’s recommendations. Calibration standards are traceable to the National Institute of Standards and Technology.

Table 10.3-1. Preoperational Airborne Radiological Monitoring

A-04	Burial Ground North, SE from MFFF Boundary	Air filters collected biweekly	Air particulate samplers using glass fiber filters	Gross alpha/beta
		Air filters monthly composite	Air particulate samplers using glass fiber filters	Pu-238, Pu-239+240, U-235, U-238, Am-241, Np-237
		Monthly	Rainwater collected and passed through ion exchange columns in the field	Gross alpha/beta, Pu-238, Pu-239+240, U-235, U-238, Am-241, Np-237, H-3
		Biweekly	Silica Gel	H-3
A-05	400-D, SRS boundary in the principal wind direction	Air filters collected biweekly	Air particulate samplers using glass fiber filters	Gross alpha/beta
		Air filters monthly composite	Air particulate samplers using glass fiber filters	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np
		Monthly	Rainwater collected and passed through ion exchange columns in the field	Gross alpha/beta, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np, ^{3}H
		Biweekly	Silica Gel	^{3}H
A-06	West Jackson - SRS boundary at centerline to nearest residence	Air filters collected biweekly	Air particulate samplers using glass fiber filters	Gross alpha/beta
		Air filters monthly composite	Air particulate samplers using glass fiber filters	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np
		Monthly	Rainwater collected and passed through ion exchange columns in the field	Gross alpha/beta, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np, ^{3}H
		Biweekly	Silica Gel	^{3}H

Table 10.3-1. Preoperational Airborne Radiological Monitoring (continued)

Location	Description of Monitor Location	Frequency	Collection Methodology	Analyses
A-07	Aiken Barricade - SRS boundary in the 2nd principal wind direction	Air filters collected biweekly	Air particulate samplers using glass fiber filters	Gross alpha/beta
		Air filters monthly composite	Air particulate samplers using glass fiber filters	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np
		Monthly	Rainwater collected and passed through ion exchange columns in the field	Gross alpha/beta, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np, ^{3}H
		Biweekly	Silica Gel	^{3}H
A-08	Dark Horse	Air filters collected biweekly	Air particulate samplers using glass fiber filters	Gross alpha/beta
		Air filters monthly composite	Air particulate samplers using glass fiber filters	Pu-238, Pu-239+240, U-235, U-238, Am-241, Np-237
		Monthly	Rainwater collected and passed through ion exchange columns in the field	Gross alpha/beta, Pu-238, Pu-239+240, U-235, U-238, Am-241, Np-237, H-3
		Biweekly	Silica Gel	H-3
A-09	Patterson Mill Road	Air filters collected biweekly	Air particulate samplers using glass fiber filters	Gross alpha/beta
		Air filters monthly composite	Air particulate samplers using glass fiber filters	Pu-238, Pu-239+240, U-235, U-238, Am-241, Np-237
		Monthly	Rainwater collected and passed through ion exchange columns in the field	Gross alpha/beta, Pu-238, Pu-239+240, U-235, U-238, Am-241, Np-237, H-3
		Biweekly	Silica Gel	H-3

Table 10.3-2. Preoperational Methodology and Lower Limits of Detection for Airborne Environmental Samples

Analyte	Method	Lower Limit of Detection[1] (μCi/ml)
Particulate		
Gross alpha	Gas-flow proportional counter	1.0E-13 [2]
Gross beta	Gas-flow proportional counter	5.0E-13 [2]
^{3}H	Liquid scintillation	5.0E-09
^{237}Np	Alpha spectrometer	5.0E-16
^{241}Am	Alpha spectrometer	1.0E-15
^{238}Pu	Alpha spectrometer	1.0E-15
^{239}Pu, ^{240}Pu	Alpha spectrometer	1.0E-15
^{235}U	Alpha spectrometer	3.0E-15
^{238}U	Alpha spectrometer	3.0E-15
Rainwater		
Gross alpha	Gas-flow proportional counter	9.0E-09
Gross beta	Gas-flow proportional counter	1.5E-08
^{3}H	Liquid scintillation	5.0E-05
^{237}Np	Alpha spectrometer	1.0E-09
^{241}Am	Alpha spectrometer	1.0E-09
^{238}Pu	Alpha spectrometer	1.0E-09
^{239}Pu, ^{240}Pu	Alpha spectrometer	1.0E-09
^{235}U	Alpha spectrometer	1.5E-08
^{238}U	Alpha spectrometer	1.5E-08

Note 1: Lower limit of detection values are 5% of the values in 10CFR20, Appendix B, Table 2.

Note 2: Lower limit of detection value is based on potential contractor minimum detectable activity.

Table 10.3-3. Preoperational Terrestrial Radiological Monitoring

Location	Description of Monitor Location	Media[1]	Frequency	Analyses[2]
VS-04 (A-04)	Burial Ground North, SE from MFFF Boundary	V, S	Annual	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np, ^{3}H
VS-05 (A-05)	400-D, SRS boundary in the principal wind direction	V, S	Annual	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np, ^{3}H
VS-06 (A-06)	West Jackson - SRS boundary at centerline to nearest residence	V, S	Annual	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np, ^{3}H
VS-07 (A-07)	Aiken Barricade - SRS boundary in the 2nd principal wind direction	V, S	Annual	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np, ^{3}H
VS-08 (A-08)	Dark Horse	V, S	Annual	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np, ^{3}H
VS-09 (A-09)	Patterson Mill Road	V, S	Annual	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np, ^{3}H

Note 1: V = Vegetation; S = Soil.

Note 2: For terrestrial radiological monitoring ^{3}H is only analyzed for in vegetation.

Table 10.3-4. Preoperational Methodology and Lower Limits of Detection for Terrestrial Environmental Samples

Analyte	Method	Lower Limit of Detection[1] (pCi/g)
Soil		
^{237}Np	Alpha spectrometer	6.0 E-03
^{241}Am	Alpha spectrometer	8.0E-03
^{238}Pu	Alpha spectrometer	6.0 E-03
^{239}Pu, ^{240}Pu	Alpha spectrometer	6.0 E-03
^{235}U	Alpha spectrometer	8.0E-03
^{238}U	Alpha spectrometer	8.0E-03
Vegetation		
^{3}H	Liquid scintillation	5.0E-03
^{237}Np	Alpha spectrometer	4.0E-03
^{241}Am	Alpha spectrometer	4.0E-03
^{238}Pu	Alpha spectrometer	4.0E-03
^{239}Pu, ^{240}Pu	Alpha spectrometer	4.0E-03
^{235}U	Alpha spectrometer	2.0E-03
^{238}U	Alpha spectrometer	2.0E-03

Note 1: Lower limit of detection values are based on potential contractor minimum detectable activity.

Table 10.3-5. Operational Airborne Radiological Monitoring

Location	Description of Monitor Location	Frequency	Collection Methodology	Analyses
A-04	Burial Ground North, SE from MFFF Boundary	Air filters collected biweekly	Air particulate samplers using glass fiber filters	Gross alpha/beta
		Air filters monthly composite	Air particulate samplers using glass fiber filters	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np
		Monthly	Rainwater collected and passed through ion exchange columns in the field	Gross alpha/beta, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np, ^{3}H
		Biweekly	Silica Gel	^{3}H
A-05	400-D, SRS boundary in the principal wind direction	Air filters collected biweekly	Air particulate samplers using glass fiber filters	Gross alpha/beta
		Air filters monthly composite	Air particulate samplers using glass fiber filters	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np
		Monthly	Rainwater collected and passed through ion exchange columns in the field	Gross alpha/beta, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np, ^{3}H
		Biweekly	Silica Gel	^{3}H
A-06	West Jackson - SRS boundary at centerline to nearest residence	Air filters collected biweekly	Air particulate samplers using glass fiber filters	Gross alpha/beta
		Air filters monthly composite	Air particulate samplers using glass fiber filters	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np
		Monthly	Rainwater collected and passed through ion exchange columns in the field	Gross alpha/beta, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np, ^{3}H
		Biweekly	Silica Gel	^{3}H

Table 10.3-5. Operational Airborne Radiological Monitoring (Continued)

Location	Description of Monitor Location	Frequency	Collection Methodology	Analyses
A-07	Aiken Barricade - SRS boundary in the 2nd principal wind direction	Air filters collected biweekly	Air particulate samplers using glass fiber filters	Gross alpha/beta
		Air filters monthly composite	Air particulate samplers using glass fiber filters	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np
		Monthly	Rainwater collected and passed through ion exchange columns in the field	Gross alpha/beta, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np, ^{3}H
		Biweekly	Silica Gel	^{3}H
A-08	Dark Horse	Air filters collected biweekly	Air particulate samplers using glass fiber filters	Gross alpha/beta
		Air filters monthly composite	Air particulate samplers using glass fiber filters	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np
		Monthly	Rainwater collected and passed through ion exchange columns in the field	Gross alpha/beta, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np, ^{3}H
		Biweekly	Silica Gel	^{3}H
A-09	Patterson Mill Road	Air filters collected biweekly	Air particulate samplers using glass fiber filters	Gross alpha/beta
		Air filters monthly composite	Air particulate samplers using glass fiber filters	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np
		Monthly	Rainwater collected and passed through ion exchange columns in the field	Gross alpha/beta, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np, ^{3}H
		Biweekly	Silica Gel	^{3}H

Table 10.3-6. Operational Methodology and Lower Limits of Detection for Airborne Environmental Samples

Analyte	Method	Lower Limit of Detection[1] (µCi/ml)
Particulate		
Gross alpha	Gas-flow proportional counter	1.0E-13[2]
Gross beta	Gas-flow proportional counter	5.0E-13[2]
^{3}H	Liquid scintillation	5.0E-09
^{237}Np	Alpha spectrometer	5.0E-16
^{241}Am	Alpha spectrometer	1.0E-15
^{238}Pu	Alpha spectrometer	1.0E-15
^{239}Pu, ^{240}Pu	Alpha spectrometer	1.0E-15
^{235}U	Alpha spectrometer	3.0E-15
^{238}U	Alpha spectrometer	3.0E-15
Rainwater		
Gross alpha	Gas-flow proportional counter	9.0E-09
Gross beta	Gas-flow proportional counter	1.5E-08
^{3}H	Liquid scintillation	5.0E-05
^{237}Np	Alpha spectrometer	1.0E-09
^{241}Am	Alpha spectrometer	1.0E-09
^{238}Pu	Alpha spectrometer	1.0E-09
^{239}Pu, ^{240}Pu	Alpha spectrometer	1.0E-09
^{235}U	Alpha spectrometer	1.5E-08
^{238}U	Alpha spectrometer	1.5E-08

Note 1: Lower limit of detection values are 5% of the values in 10CFR20, Appendix B, Table 2.

Note 2: Lower limit of detection value is based on potential contractor minimum detectable activity.

Table 10.3-7. Operational Terrestrial Radiological Monitoring

Location	Description of Monitor Location	Media[1]	Frequency	Analyses[2]
VS-04 (A-04)	Burial Ground North, SE from MFFF Boundary	V, S	Annual	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np, ^{3}H
VS-05 (A-05)	400-D, SRS boundary in the principal wind direction	V, S	Annual	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np, ^{3}H
VS-06 (A-06)	West Jackson - SRS boundary at centerline to nearest residence	V, S	Annual	^{238}Pu, ^{239}Pu, ^{240}Pu, $^{23}5U$, ^{238}U, ^{241}Am, ^{237}Np, ^{3}H
VS-07 (A-07)	Aiken Barricade - SRS boundary in the 2nd principal wind direction	V, S	Annual	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np, ^{3}H
VS-08 (A-08)	Dark Horse	V, S	Annual	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np, ^{3}H
VS-09 (A-09)	Patterson Mill Road	V, S	Annual	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{237}Np, ^{3}H

Note 1: V = Vegetation; S = Soil.

Note 2: For terrestrial radiological monitoring ^{3}H is only analyzed for in vegetation.

Table 10.3-8. Operational Methodology and Lower Limits of Detection for Terrestrial Environmental Samples

Analyte	Method	Lower Limit of Detection[1] (pCi/g)
Soil		
^{237}Np	Alpha spectrometer	6.0E-03
^{241}Am	Alpha spectrometer	8.0E-03
^{238}Pu	Alpha spectrometer	6.0E-03
^{239}Pu, ^{240}Pu	Alpha spectrometer	6.0E-03
^{235}U	Alpha spectrometer	8.0E-03
^{238}U	Alpha spectrometer	8.0E-03
Vegetation		
^{3}H	Liquid scintillation	5.0E-03
^{237}Np	Alpha spectrometer	4.0E-03
^{241}Am	Alpha spectrometer	4.0E-03
^{238}Pu	Alpha spectrometer	4.0E-03
^{239}Pu, ^{240}Pu	Alpha spectrometer	4.0E-03
^{235}U	Alpha spectrometer	2.0E-03
^{238}U	Alpha spectrometer	2.0E-03

Note 1: Lower limit of detection is based on potential contractor minimum detectable activity.

Table 10.3-9. Administrative Limits and Action Levels for Air Emissions

Parameter	Action Level[1] (µCi/ml)	Action
Alpha activity	3.2 E-14	Recount sample(s), including full isotope spectroscopy and compare to individual isotope regulatory limits
Alpha activity	6.4 E-14	Evaluate operations for possible source of positive activity
Alpha activity	3.2 E-13	Releases are potentially above allowable 10 CFR 20, Appendix B, Table 2 effluent limits. Initiate orderly shutdown of associated processes for repair or correction.

Note 1: Calculated values at the MFFF BMF stack.

10.4 ENVIRONMENTAL PERMITS

Table 10.4-1 lists the status of environmental permits and plans that are required during construction and prior to operation of the MFFF.

Table 10.4-1. Status of Federal, State and Local Licenses, Permits and Approvals

Requirement	Status	Comments
Federal Laws and Enabling Regulations		
Negative declaration on cultural resources from the State Historic Preservation Officer (SHPO) 43 CFR Part 7; 36 CFR Parts 60, 61, 63, 65, 67, 68	Completed	SHPO approved mitigation plan on 11 April 2001. Mitigation completed August 2002.
Negative declaration on endangered species from the U.S. Fish and Wildlife Services (USFWS) 50 CFR Parts 13, 17, 222, 226, 227, 402, 424, 450-453	Completed	USFWS issued negative declaration on 20 June 2001.
Negative declaration on prime or unique farmlands from U.S. Natural Resources Conservation Service (USNRCS) 7 CFR Part 658	Not required	USNRCS does not identify SRS as prime farmlands because the land is not available for agricultural production.
Negative declaration on 404 Permit from U.S. Army Corps of Engineers (COE)	Not required	No jurisdictional wetlands exist on MFFF site.
Floodplain Assessment	Completed	Floodplain Assessment incorporated into the design basis.
Construction Environmental Plans and Permits		
Construction Emissions Control Plan (CECP) 40 CFR 60 South Carolina Regulation 61.62-6	Completed	CECP was completed and does not need to be approved by SCDHEC.
Bureau of Air Quality (BAQ) Construction Permit 40 CFR 60 South Carolina Regulation 61.62-5	Completed	BAQ Construction Permit for BMF Stack, Diesel Generators, and Diesel Fuel Tanks has been received from SCDHEC in 2006. BAQ Construction Permit for two Concrete Batch Plants and a 175-MW diesel generator has been received from SCDHEC in 2007.

Table 10.4-1. Status of Federal, State and Local Licenses, Permits and Approvals (Continued)

Requirement	Status	Comments
BAQ National Emission Standard for Hazardous Air Pollutants (NESHAP) Construction Permit 40 CFR 61 Subpart H 10 CFR 20 South Carolina Regulation 61.62-5	Completed	Alternative Calculation methodology approved by EPA Region IV and SCDHEC in April 2002. Exemption from NESHAP Construction Permit granted.
Bureau of Water Quality (BWQ) **Construction NPDES General Permit** 40 CFR 122 South Carolina Regulation 61-9 South Carolina Regulation 61-68 South Carolina Regulation 72-300 through 72-316 (GR)	Completed	Access to BWQ Construction General Permit granted in May 2005 upon acceptance of Notice of Intent (NOI) and Storm Water Pollution Prevention Plan (SWPPP) by SCDHEC.
BWQ Sanitary Wastewater Construction Permit 40 CFR 122 South Carolina Regulation 61-9 South Carolina Regulation 61-67	Completed	Permit from tie-in to interface point has been received. Permit from interface point to MFFF was received from SRS Delegated Authority in 2007.
BWQ Construction Storm Water Pollution Prevention Plan (SWPPP) 40 CFR 122 South Carolina Regulation 61-9 South Carolina Regulation 61-68 South Carolina Regulation 72-300 through 72-316 (GR)	Completed	Accepted by SCDHEC with NOI in May 2005. SWPPP has been updated to account for improved stormwater controls through 2012.
BWQ Domestic Water Distribution Construction Permit 40 CFR 141 South Carolina Regulation 61-58 South Carolina Regulation 61-71 South Carolina Regulation 61-101	Completed	Permit from tie-in to interface point has been received. Permit from interface point to MFFF was received from SRS Delegated Authority in 2007.
Bureau of Land and Waste Management (BLWM) **Underground Storage Tank (UST) Installation Permit** 40 CFR 112 40 CFR 280 South Carolina Regulation 61-92	No longer required	UST Installation Permit is no longer required with the deletion of the secondary diesel fuel tanks from the design.

Table 10.4-1. Status of Federal, State and Local Licenses, Permits and Approvals (Continued)

Requirement	Status	Comments
Waste Minimization and Pollution Prevention Plan 40 CFR 261 40 CFR 262 40 CFR 264 40 CFR 268 South Carolina Regulation 61-66 South Carolina Regulation 61-79 South Carolina Regulation 61-99 South Carolina Regulation 61-104	Completed	Issued in 2006. WMPPP has been updated to account for improved waste management practices through 2012.
Operational Environmental Plans and Permits		
BAQ Air Operating Permit 40 CFR 71 South Carolina Regulation 61.62-70	In progress	BAQ Air Operating Permit will be completed approximately 2 years prior to MFFF operations.
Risk Management Plan 40 CFR §68.130 Tables 1 & 3 South Carolina Regulation 61.62-68	Not required	MFFF will impose administrative limits on 40 CFR §68.130 and South Carolina Regulation 61.62-68 extremely hazardous chemicals, which will preclude the need for a Risk Management Plan.
BWQ Utility Water Permit 40 CFR 122 South Carolina Regulation 61-9 South Carolina Regulation 61-67	No longer required	BWQ Utility Water Permit is no longer required.
BWQ Sanitary Wastewater Operating Permit 40 CFR 122 South Carolina Regulation 61-9 South Carolina Regulation 61-67	In progress	BWQ Sanitary Wastewater Permit will be completed approximately 2 years prior to MFFF operations.
BLWM UST Operating Permit 40 CFR 112 40 CFR 280 South Carolina Regulation 61-92	No longer required	UST Operating Permit is no longer required with the deletion of the secondary diesel fuel tanks from the design.
Spill Prevention Control and Countermeasures (SPCC) Plan 40 CFR 112 Section 110 South Carolina Regulation 61-9	Completed	SPCC Plan will be completed approximately 2 years prior to MFFF operations
BWQ Domestic Water Distribution Operating Permit 40 CFR 141 South Carolina Regulation 61-58 South Carolina Regulation 61-71 South Carolina Regulation 61-101	In progress	BWQ Domestic Water Permit will be completed approximately 2 years prior to MFFF operations.
BLWM Resource Conservation and Recovery Act (RCRA) Generator Identification Number South Carolina Regulation 61-79	Not required	SRS M&O Contractor requested MOX as a tenant of the site use the SRS EPA ID number for non-radiological hazardous waste generated during construction and operations.

Table 10.4-1. Status of Federal, State and Local Licenses, Permits and Approvals (Continued)

Requirement	Status	Comments
Bureau of Land and Waste Management **RCRA Part B Permit** South Carolina Regulation 61-66 South Carolina Regulation 61-79 South Carolina Regulation 61-99 South Carolina Regulation 61-104	Not required	Generated hazardous waste will be stored and accumulated for less than 90 days prior to being sent to SRS, which will preclude the need to obtain a RCRA Part B Permit.
Waste Minimization and Pollution Prevention Plan 40 CFR 261 40 CFR 262 40 CFR 264 40 CFR 268 South Carolina Regulation 61-66 South Carolina Regulation 61-79 South Carolina Regulation 61-99 South Carolina Regulation 61-104	Completed	Construction Waste Minimization and Pollution Prevention Plan was last updated in 2012 and will be updated approximately 2 years prior to MFFF operations.
Emergency Planning and Community Right-to-Know Notifications 40 CFR 355 40 CFR 370 40 CFR 372	Completed	MFFF has been filing Tier II reports and Toxic Release Inventory reports on an annual basis from 2007-2012.

10.5 DESIGN BASIS

The following information represents the design basis attributes for environmental protection.

- Liquid radioactive waste is transferred to Department of Energy facilities at the Savannah River Site (SRS) in a manner consistent with the SRS Waste Acceptance Criteria (WAC) for appropriate storage and disposition.
- Table 10.2-2 provides the analytical methodologies and lower limits of detection (LLD) used to characterize airborne emissions.
- The MFFF does not discharge radioactive liquid to the environment during normal and off-normal operations.
- Table 10.3-1 identifies the preoperational airborne radiological monitoring program, including the sampling locations, frequency of sampling, collection methodology, and radionuclide analyses.
- Preoperational analytical methods and lower limits of detection (LLD) for analyses of airborne isotopes are listed in Table 10.3-2.
- Table 10.3-3 identifies the preoperational terrestrial radiological monitoring program, including a comprehensive baseline of radiological conditions related to airborne emissions in the environs of the MFFF.
- Preoperational analytical methods and lower limits of detection (LLD) for terrestrial environmental samples are listed in Table 10.3-4.
- Table 10.3-5 identifies the operational airborne radiological monitoring program, including the sampling locations, frequency of sampling, collection methodology, and radionuclide analyses.
- Operational analytical methods and lower limits of detection (LLD) for analyses of airborne isotopes are listed in Table 10.3-6.
- Table 10.3-7 identifies the operational terrestrial radiological monitoring program, including an evaluation of radiological impacts related to deposition of airborne emissions in the environs of the MFFF.
- Operational analytical methods and lower limits of detection (LLD) for terrestrial environmental samples are listed in Table 10.3-8.
- To ensure that the regulatory limits for doses to the public are not exceeded, administrative limits and actions levels for air emissions are provided in Table 10.3-9.

CHAPTER 11 WITHHELD FROM PUBLIC DISCLOSURE UNDER 10CFR2.390

12.0 HUMAN FACTORS ENGINEERING

This chapter describes the application of Human Factors Engineering (HFE) to the Mixed Oxide (MOX) Fuel Fabrication Facility (MFFF). HFE is the application of the knowledge of human cognitive and physical capabilities and characteristics to the development of engineered systems, facilities, and equipment. By applying this knowledge, human performance, and therefore system performance, can be improved dramatically. Human-System Interfaces (HSI) designed with the human subsystem as a key element are inherently safer and more reliable than those systems that do not address human integration during design. What is typically not emphasized is that in many instances, causes of "human or operator error" result from poorly designed Human-Machine Interface (HMI), or poorly designed equipment requiring human interfacing. It is the goal of HFE to ensure that the potential for "design-induced" human error is eliminated or minimized as it pertains to identified Items Relied on for Safety (IROFS) Administrative Controls. This is necessary to ensure that the MFFF design supports safe, effective and efficient human performance.

HFE principles and practices are applied specifically to the MFFF active and passive Engineered IROFS (for maintainability, testing, and surveillance purposes) and to those personnel activities that are identified by the Integrated Safety Analysis (ISA) as Enhanced Administrative Control (EAC) IROFS and Administrative Control (AC) IROFS (i.e., Administrative IROFS). An EAC is a procedurally required or prohibited human action, combined with a physical device that alerts the operator that the action is needed to maintain safe process conditions, or otherwise adds substantial assurance of the required human performance (i.e., augmented administrative control). An Administrative Control is a human action that is prohibited or required to maintain safe process conditions (i.e., simple administrative control(s) typically implemented in procedures). MOX Services will also review the operator actions identified as Defense-in-Depth in the Nuclear Criticality Safety Evaluations (NCSEs) and Nuclear Safety Evaluations (NSEs) using a graded approach as defined in the Human Factors Engineering Program Plan (HEPP) and the Human Factors Engineering Implementation Plan (HFIP).

The MOX Fuel Fabrication Building (BMF) is a multi-functional complex containing all of the plutonium handling, fuel processing, and fuel fabrication operations of the MFFF. This building includes the Aqueous Polishing Area, the MOX Processing Area, the Shipping and Receiving Area, and the Laboratory Area.

The design of the MFFF is based on two successfully operating AREVA facilities (i.e., "reference plants"): MELOX facility and La Hague facility, located in southern and northwestern France, respectively. Both of these operating facilities successfully and safely convert plutonium to MOX fuel. This design represents over 25 years of operating experience and associated evolving process design in producing MOX fuel. The existing design is adapted to meet required United States codes, regulations and standards, and is culturally calibrated to better accommodate North American operators.

MFFF modifications incorporate the reference plants' operating experience reviews or "lessons learned," and implements requirements of the U.S. Nuclear Regulatory Commission (NRC). Existing designs of systems, facilities, and equipment that are adapted or modified for use in the MFFF are reviewed to evaluate the efficacy of human factors design elements as they relate to the active and passive Engineered IROFS and the Administrative IROFS. The depth and rigor of the evaluations is dependent on a determination of the complexity and importance to safety of the component or system, and consequences of human error. The final MFFF design will manifest human factors principles and practices resulting from HFE programmatic reviews.

Using detailed review criteria and guidelines, HFE program reviews are integrated into the MFFF final design and design change control processes. Focus areas of the HFE review are based on and consistent with the findings of the ISA process in the identification of active and passive Engineered IROFS and Administrative IROFS, as discussed in Chapter 5. The ISA may be viewed as a developmental process starting with the Safety Assessment (SA) phase in support of the development of the Construction Authorization Request (CAR) that progressively becomes more sophisticated (i.e., Detailed ISA Phase) in support of the development of the ISA Summary and MFFF License Application (LA). Initially, a broad set of hazards is identified and analyzed to most efficiently identify and evaluate events. Events with unmitigated consequences satisfying the low dose limits established by 10 CFR §70.61 (i.e., less than "intermediate") or events with event likelihood meeting the requirements of 10 CFR §70.61 (i.e., "not credible" events) are dispositioned and not analyzed further. For the remaining events, progressive layers of more detailed analyses are performed until the risks of all identified events satisfy the requirements of 10 CFR §70.61. The ISA is developed, used, and maintained during the life of the facility.

The focus during the development of the ISA is on the identification of IROFS through functional requirements analyses. The identified IROFS are functionally allocated as the necessary and sufficient set of design features (Engineered IROFS) and the administrative controls (activities of personnel) to be implemented in the final design. The selected IROFS satisfy the performance requirements of 10 CFR §70.61. Baseline design criteria, as described in 10 CFR §70.64, are applied from the outset of MFFF design work and are primarily focused on physical design and facility features, with the intent to achieve a conservatively designed facility tolerant of both process upsets and human errors.

The detailed phase of the ISA process, described in Chapter 5, builds upon the information and analyses performed as a part of the SA. The main purpose of the more detailed analyses is to identify the IROFS at the component level to implement the safety strategy established in the SA. The detailed analyses also demonstrate that the selected IROFS are sufficiently robust to ensure the performance criteria of 10 CFR §70.61 are satisfied.

The ISA process includes Process Hazards Analysis (PrHA) execution by a qualified team with HFE representation. A PrHA is performed for each process unit or workshop to identify specific event causes and event scenarios in detail, and associated prevention and mitigation features (IROFS) at the component level. All modes of operation are considered, including startup, normal operation, shutdown, maintenance, testing, and surveillance. Software malfunctions, including communication malfunctions, common mode malfunctions, and human errors in the Human-Computer Interface (HCI) are included in the PrHAs. Specific causes evaluated also include faults caused by operation of a support system outside of normal operating ranges.

Other event causes evaluated include personnel actions and in-actions (e.g., operator errors regarding commission or omission) that could result in adverse consequences. A detailed review of the various operational sequences is performed to identify process upsets and deviations, including human errors of omission and commission. Manual and semi-automatic processes and sequences are evaluated for potential human errors that could result in adverse consequences.

The detailed ISA process, which includes HFE reviews, was applied during the initial design phase and will be maintained through final design, construction, start-up, and operation phases. HFE will be sustained during the operational phase through the human performance monitoring strategy identified and described in both the Human Factors Engineering Program Plan (HEPP) and the Human Factors Engineering Implementation Plan (HFIP).

To provide additional details of the MFFF Human Factors Engineering program implementation and execution, a HFIP is developed that addresses the HFE program elements described below. The HFIP follows and supplements the MFFF HEPP.

12.1 PERSONNEL ACTIVITIES DESIGNATED IROFS

Operation and control of the MFFF relies predominantly on automated control systems for production and facility safety. The MFFF control systems are discussed in Section 11.0, Plant Systems. MFFF operations staff will perform the following types of procedural tasks, mostly through soft control systems (i.e., via HCI):

- Initiate batch or continuous operations through a permissive control system
- Monitor the progress of the production processes
- Perform sampling quality control checks at preprogrammed process hold points
- Monitor and confirm the status of confinement systems, fluid systems, and other facility systems
- Identify and respond to MFFF process alarms and warnings using procedural based instructions
- Respond to or recover from off-normal or emergency conditions (e.g., facility fire scenarios, or seismic event).

The highly automated nature of the facility reduces the number of personnel activities designated as IROFS Administrative Controls. Each IROFS Administrative Control, as identified in the ISA and documented in the Nuclear Criticality Safety Evaluations (NCSE) or the Nuclear Safety Evaluations (NSE), will be executed by approved procedures. A second operator or supervisor is typically included in many IROFS procedural actions to provide verification of procedure accomplishment or provide an additional permissive action. Examples include reviewing sample results in order to allow a process to feed forward, or inputting data into safety programmable logic controllers (SPLC).

The HFE review of personnel actions credited as IROFS administrative controls includes the identification of the associated HSIs and the consequences of incorrectly performing or omitting actions for each activity.

12.2 HUMAN FACTORS ENGINEERING DESIGN REVIEW

The MOX Project HFE program function for design review resides in Engineering, however, the HFE program activities cross-matrix the MOX organization, as needed. The chain of authorities and responsibilities for the HFE program flow down from the MOX Project Manager, to the Vice President of Operations, to the Vice President of Engineering, to the Manager of Electrical, Instruments and Controls (Electrical/I&C), to the Lead Engineer (LE) for Electrical Group, to the Responsible Engineer (RE) for Human Factors Engineering.

The MOX Project HFE program documented in the HEPP includes identification of HFE programmatic goals, scope, a description of the various HFE processes used for HFE review; the MOX HFE team composition; and preparation of a human performance monitoring strategy. The HFE elements described in the HEPP and implemented in the HFIP are: HFE Program Management and basis for the program; Operating Experience Review (i.e., "Lessons Learned"); Functional Requirements Analysis and Functional Allocation; Task Analysis; Staffing and Qualifications; Human-System Interface (HSI) Design; Procedure Development; Training Program Development; Human Factors Verification and Validation (V & V); Design Implementation; and Human Performance Monitoring. The human performance monitoring strategy will assure that no significant safety degradation occurs because of changes that are made in the MFFF and to provide adequate assurance that the conclusions that have been drawn from the HFE review and evaluation remain valid over time. The ISA process identifies the sensors, instruments, actuators, and Administrative IROFS. The associated HSIs are then identified, and appropriate human performance requirements established, during the design process. Human performance is verified and validated before final turn over of the MFFF design to operations.

12.2.1 Goals and Scope

The scope of the MFFF HFE design review encompasses Engineered and Administrative IROFS controls associated with design, design changes, operations, maintenance, testing, and surveillance activities and will be documented in the final Human Factors Report. There are a series of summary reports for each element of the HEPP that is implemented and completed. The MFFF HFE program will include evaluations of personnel activities that support safety, such as maintenance, testing, and surveillances of IROFS systems or instrumentation. Also included are evaluations to ensure operators have the appropriate controls and instrumentation available to confirm proper operation of the safety systems and controls, including automated safety actions, confinement systems, HVAC systems, alarms and warnings, and during off-normal or emergency conditions (e.g., facility fire scenario, or adverse weather phenomena such as tornado). Evaluation of the characteristics of HSI uses the MFFF Human Factors Design Guideline (HFDG) with backup design review guidance of NUREG-0700 Rev. 2, Human-System Interface Design Review Guidelines. Implementation guidance only from NUREG-0711, Human Factors Engineering Program Review Model, (1994) is also used for the writing of the HFE Implementation Plan.

The goals of the MFFF HFE program for IROFS Administrative Controls are to:

- Determine the functional requirements and functional allocations of IROFS Administrative Controls by participation in the ISA process
- Include HFE principles in the design to preclude challenges to the IROFS Administrative Controls, (e.g., provide enough time for the performance of a human action)
- Verify that the design is appropriate with respect to HFE principles and practices
- Demonstrate the adequacy of the human factors design by integrated system validation and final HSI verification
- Institute procedures that ensure HFE principles are appropriately applied to facility changes.

12.2.2 HFE Team Composition, Organizational Activity, and Responsibilities

The MOX Project HFE Team has authorities and responsibilities for implementing the HEPP and HFIP program elements. The HFE Team assists with the development of the HEPP and the HFIP by providing review and comments to these plans. In addition, the HFE Team provides input in the HFE procedures development; reviews HFE design development, test and evaluation activities; initiates, recommends, and provides solutions to HFE concerns; and has the authority to make design changes within the scope of the MOX project change management procedures. The HFE team ensures the human engineering effort is integrated into the MFFF design. The HFE team is coordinated in carrying out its activities to ensure that other technical disciplines are receiving the proper support from HFE and vice versa.

During the MFFF design phase the HFE team is composed of a core group of persons from functional groups having substantial HSI interest (e.g., Operations or Software Design Group). As design progresses to final design, additional expertise is added to the team (e.g., training and procedure writing). Team communications is through emails, teleconferences, face-to-face meetings, and when necessary committee meetings. In summary, the HFE team is responsible for execution and documentation of the HFE functions, including:

- Recommending actions to ensure HFE principles are integrated into the design
- Coordinating implementation of HFE recommendations
- Verifying implementation of HFE design criteria as part of the final design review

During the operations phase, this function transitions to become a part of plant production, and the manager of the production function is responsible for continuation of the HFE program and leading the HFE team. The HFE team ensures that HFE criteria are appropriately applied to active and passive Engineered IROFS and IROFS Administrative Controls, with particular emphasis on review of plant changes, events and incidents, and procedures.

12.2.3 Process and Procedures

HFE is applied to the MFFF in a structured approach (see §12.2) using approved Human Factors plans and procedures.

The HFE design review process includes:

- Operating Experience Review – to identify and evaluate safety issues regarding operation and design experiences, particularly the lessons learned garnered from the reference plants.
- Functional Requirements Analysis – to identify functions that must be performed to satisfy MFFF safety objectives. This is accomplished in the ISA.
- Functional Allocation – to analyze and assign the requirements for safe MFFF control and operation to system elements (e.g., automatic functions, or to the human for manual procedural action).
- Task Analysis – to further define and analyze the necessary human tasks by decomposition of the required tasking and determining the necessary requirements for displays, data processing, controls, and job support aids needed to accomplish tasks.
- Technical support for Staffing and Qualifications – to understand and verify systematically the need for the number and qualifications of personnel required to safely operate the MFFF. Initial staffing levels are established based on experience of the reference plants.
- HSI design, inventory, and characterization – to translate function and task requirements into HSI characteristics and functions by using a structured approach to guide the designers. This includes the detailed design of alarms, displays, controls, and other aspects of the HSI through systematic application of HFE principles and criteria.
- Technical support for Procedure Development – to verify the application of HFE principles and guidance, along with all other design requirements, to develop procedures that are technically accurate, comprehensive, explicit, easy to use, and validated.
- Technical support for Training Program Development – Training of MFFF personnel is an important factor in ensuring safe and reliable operation of the MOX facility. Training design is elucidated by systematic analysis of job and task requirements and the development of learning objectives derived from an analysis of desired performance following the training.
- Identification of HFE Verification and Validation (V&V) activities to support construction and startup – there are several important processes accomplished during HFE V&V. First, to verify that the HSI inventory and characterization accurately describes all HSI displays, controls, and related equipment that are within the scope of the HSI design review (i.e., Administrative IROFS). Second, to comprehensively determine that the design conforms to HFE design principles and practices, and that the design enables MFFF personnel to successfully perform their tasks to achieve safe operational goals. And third, any open item Human Engineering Discrepancy (HED) is resolved.
- Design Implementation – the final MFFF HSIs, procedures, and personnel training are compared with the design description to verify that they conform to the design that resulted from the HFE design process and V&V activities. Any identified discrepancies

are corrected or justified. All HFE-related issues documented in the issue tracking system are verified as adequately addressed and closed.

- Human Performance Monitoring (HPM) – as a part of the HEPP, this strategy will help to provide reasonable assurance that the confidence developed by the completion of the integrated system validation is maintained over time. HPM is passed onto and maintained by the operations personnel.

12.2.4 Issue Tracking

During the design phases of preliminary design, final design, and verification/validation human factors engineering design reviews will identify HSI concerns, anomalies, or other issues affecting the human integration into design. An HFE concern will be identified as a Human Engineering Discrepancy (HED) and tracked accordingly in the HFE program and an approved MOX Project tracking system, per project procedure. HEDs are defined as departures from a benchmark of system design suitability for the roles and capabilities of the human operator. The benchmark is the NUREG-0700 Rev. 2 with backup of MFFF HFDG. Also, this may include a deviation from a standard or convention of human engineering practice, an operator preference or need, or an instrument/equipment characteristic that is implicitly or explicitly required for an operator's task, but is not provided to the operator. HEDs are not considered in isolation and, to the extent possible, their potential interactions are considered when developing and implementing solutions. For example, if the HSI for a single plant system is associated with many HEDs, then the set of design solutions should be coordinated to enhance overall performance and avoid incompatibilities between individual solutions. Approaches that develop design solutions to some HEDs before all have been identified from a particular verification or validation activity are acceptable provided that the potential interactions between HEDs are specifically considered prior to implementing the design solutions.

HFE deviations discovered during the human factors review are resolved, or justification of the acceptability is documented, prior to Operations acceptance of the final design. HEDs are tracked to resolution.

12.2.5 Functional Grouping

Functional units of the MFFF are grouped by plant area or by control rooms. HFE design reviews are performed during final design for each plant area or control room where personnel actions designated IROFS, or where operations, maintenance, or surveillance activities associated with IROFS, are performed. The HFE design review process documents this review and confirms that the final design is acceptable for the applicable plant area or control room.

12.3 FUNCTIONAL REQUIREMENTS ANALYSIS AND FUNCTIONAL ALLOCATION

Functional Requirements Analysis (FRA) is the identification of functions that must be performed to satisfy the MOX facility safety objectives to prevent or mitigate the consequences of postulated accidents that could damage the facility or cause undue risk to the health and safety of the public. The FRA is conducted to:

- Determine the objectives, performance requirements, and constraints of the design
- Define the high-level functions that have to be accomplished to meet the objectives and desired performance
- Define the relationships between high-level functions and MOX facility systems (e.g., facility configurations or success paths) responsible for performing the function
- Provide a framework for understanding the role of controllers (whether the human subsystem or automated subsystems) for controlling the facility.

The FRA identifies the control actions (IROFS) that are required to achieve the functional goals. Ultimately, it is the human subsystem that retains control over the safe operation of the facility.

The Functional Allocation (FA) is the analysis of the requirements for MOX facility control and the assignment of control functions to:

- The human subsystem (Administrative Control)
- The machine subsystem elements (automatic control; passive and active engineered features)
- Combinations of human subsystem and machine subsystem elements (shared control and automatic systems with manual backup).

The functional requirements and function allocations of the MFFF design are primarily based on one or more reference plant designs. Many of the functional requirements and function allocations for the new MFFF may be similar to those of the reference plant design iterations; however, a major difference is the selection and incorporation of IROFS for the MFFF. This reflects the evolutionary nature of technology development in a complex, but high-reliability system like the MFFF. Functional allocation is the process of assigning responsibility for function accomplishment to human or machine resources, or to a combination of human and machine resources. Functional allocation addresses the MFFF design goal to automate operations as much as practical. The MFFF functional allocation takes into account the results of the PrHA.

The functional allocation methodology is embedded within the ISA. Per NUREG-1718, Section 12.0 Human Factors Engineering for Personnel Activities, the functional allocation analysis is based on the Operating Experience Review (OER). Personnel activities are functionally allocated to take advantage of human strengths and to avoid demands that are not compatible with human capabilities. PrHAs for each process unit or workshop are performed and include the OER review for the unit. PrHAs typically employ techniques such as hazard and operability studies and the use of "What-If" checklist analyses. As the PrHAs are performed, additional analyses may be identified as necessary to satisfy the requirements of 10CFR70.61. IROFS at the component level are identified and described by generating IROFS descriptions, including the unique IROFS identification tag, safety function, operating requirements, failure detection description, and a summary of the functional analyses demonstrating the IROFS design will function as necessary to perform their safety function. This is where the allocation is made to either an appropriate engineering control or to an Administrative Control (personnel action).

Human Factors Engineering (HFE) evaluations or analyses of proposed Administrative IROFS are conducted to consider that the human interaction requirement will be compatible with human capability, under stated conditions or hypotheses. The HFE evaluations support the ISA in evaluating operator actions and inactions, including errors of omission and commission. An example of allocating to a human strength is a decision-making action; an example of allocating away from a human weakness is allocating complex calculations to computers or automating a task.

12.4 TASK ANALYSIS

The functions allocated to plant personnel define their roles and responsibilities. Human actions (HAs) are performed to accomplish these functions or jobs. HAs can be further divided into tasks. A task is a group of related activities that have a common objective or goal. Task analysis is the identification of requirements for accomplishing these tasks (i.e., for specifying the requirements for the displays, data processing, controls, and job support aids needed to accomplish tasks). As such, the results of task analyses are identified as inputs in many HFE activities; e.g., it forms the basis for:

- Staffing, qualifications, job design, and training
- HSIs, procedures, and training program design
- Task support verification criteria definition.

Many tasks have already been carried over from the existing Reference plant design. The scope of task analyses for the MFFF design is limited to that range of personnel activities already identified as IROFS Administrative Controls (Enhanced Administrative Controls and Administrative Controls) and personnel activities that support safety, such as maintenance, testing, and surveillance activities. Included are detailed descriptions of personnel demands (e.g., input, processing, and output); iterative nature of the analysis; and incorporation of job design issues. The task analysis addresses each operating mode for each personnel activity identified as IROFS (e.g., startup, normal operations, emergency operations, and shutdown). The task analysis will re-enforce the functional allocation or dictate a change in the allocation choice.

Task analysis methodologies are annotated in the HFIP. Generally, the process will involve a gross level of analyses involving the development of detailed narratives of what the human subsystem has to do. The nature of the input, process and output needed by and of personnel is defined and the appropriate detailed task descriptors will address task considerations. The task analysis is generally an iterative process and becomes progressively more detailed over the design cycle.

The HFE Team will agree on what task analysis method will be applied for a particular process. Some processes may be very simple to analyze while others may be more complex and need more detailed analyses. The Responsible Engineer (RE) for a particular subsystem or functional unit will participate in the HFE task analysis review meetings during final design and the information gained will be captured and used by the control system designers to guide the application of human factors engineering criteria into the final design work. The task analysis will address issues such as:

- The minimum number of operators and supervisors required to do the job and associated tasks
- The operator and supervisor minimum skills needed to perform the job and tasking
- Allocation of monitoring and control tasks to achieve the formation of a meaningful job, and to address the management of operator's physical and cognitive workloads.

Another purpose for the task analysis is confirming the ISA provided inventory of IROFS alarms, displays, and controls necessary to perform operator or maintainer tasks. In addition, the task analysis may identify and examine adjustments made to the HSIs by the users, such as notes and external memory aids, which suggest that the users' needs may not be fully accommodated in the current HSI design. Task analysis identifies the specific operator and the operator's information and control requirements (e.g., instruments, controls, communication, instrument ranges) that enable the operator to perform the task. Task analysis considers representative tasks from the areas of operations, maintenance, inspection, and surveillance, and considers various plant operating modes. Task analysis also considers tasks for monitoring automated systems and responding to off-normal conditions. This analysis process will identify the potential causes and modes of human error.

"Problem tasks" (e.g., those tasks that cannot be performed well, that don't have the right tools, that confuse the operator, or create safety concerns) will be identified and written into the Human Engineering Discrepancy (HED) tracking system, using an approved procedure, for resolution of the concern. The HED is tracked accordingly in the HFE program, and an approved MOX Project tracking system per project procedure.

12.5 OPERATING EXPERIENCE REVIEW

MFFF control system architecture, control philosophy, and human system interfaces emphasize proven control methods. An important early step in the design process was to review operating experience, focusing on the lessons learned, associated with the existing AREVA facilities or systems. Conditions that could contribute to or alleviate human performance problems were identified mainly through operator interviews at the existing facilities and factored into the MFFF facility design. Other sources of operating experience information included interviews with plant operations, maintenance, systems engineering personnel, or other supporting staff who are familiar with the reference plants. Insights gained from operational lessons learned are incorporated into the MFFF design.

12.6 HSI DESIGN, INVENTORY, AND CHARACTERIZATION

The HSI design is derived from the existing and proven design of the reference plants HSIs, modified for both cultural calibration purposes and U.S. safety requirements. The layout of HSIs within consoles, panels, and workstations is based upon (1) predecessor plant design, (2) analyses of operator roles (job analysis) and (3) systematic strategies for organization such as arrangement by importance, frequency of use, and sequence of use.

Functional allocation and task analysis are incorporated into the design of IROFS components (e.g., IROFS alarms, normal monitor displays, controls, and operator aids) during final design. Based on the results of the task analysis, the control, display, and communications equipment requirements for IROFS systems, facilities, and equipment that are operated and maintained by personnel or have a significant HSI are identified. Evaluation of the characteristics of the HSI incorporates the review criteria of NUREG-0700, Rev. 2. The following sources of information provide additional inputs to the HSI design process:

- Analyses of personnel task requirements performed in the earlier stages of the design process and are used to identify the requirements for the HSIs. These analyses included the OER, the FRA and FA, and the Task Analysis, along with the evaluations of staffing, qualifications and job analyses.
- System requirements which are interpreted as constraints imposed by the overall instrumentation and control (I&C) system and are considered throughout the HSI design process.
- Regulatory Requirements which are identified as inputs to the HSI design process.
- Human Factors Design Guidelines (HFDG) are used not only for review of the HSI but also for design guidance for new or modified IROFS equipment having human interfacing. This utility document provides a MFFF oriented easy-to-use source of human factors guidance and is consistent with NUREG-0700 Rev. 2. The HFDG additionally will be used appropriately as best engineering practices for non-IROFS equipment and systems.
- Any other requirements that may be identified and are inputs to the HSI design.

The resulting HSI design addresses work environment, the work space layout, control panel and console design, control and display device layout, and information and control interface design. The HSI design avoids extraneous controls and displays, and minimizes the incorporation of information, displays, controls, and features that unnecessarily complicate operator activities. The HSI characterization consists mainly of alarms, warnings, safety control panels, displays, controls, and local process HMI consisting of IROFS alarms and Administrative Controls called out by the ISA process. The HSI detailed design supports personnel in their primary role of monitoring and controlling the MFFF while minimizing personnel demands associated with the use of HSIs (for example, window manipulation, display selection, display system navigation). NUREG-0700 describes high-level HSI design review principles that the detailed design incorporates. For IROFS Administrative Controls, the design minimizes the probability that errors will occur and maximizes the probability that an error will be detected if one should be made. The layout of HSIs within consoles, panels, and workstations is based upon the analyses of operator role (job analysis), and the systematic strategies for organization such as arrangement by importance, frequency of use, and sequence of use. Consideration is given to HSIs used over the duration of a shift where decrements in performance due to fatigue may be a concern. HSIs are designed to support inspection, maintenance, surveillances and repair of MFFF equipment. The HSIs are designed so that inspection, maintenance, tests, surveillance, and repair of the HSIs do not interfere with other MFFF control activities.

The HSI design documentation includes a complete HSI inventory and the basis for the HSI characterization.

12.7 STAFFING

The MFFF staff and their qualifications are important considerations. The MFFF organization will be comprised of five major subgroups – Business, Engineering, Licensing, Quality, and Plant Operations. Of interest here is the Operations, Maintenance and Technical Support groups within Plant Operations. The initial staffing levels are estimated and established based on experience with the reference plants that have been in operation for the last few decades and discussions with the NRC licensed US Fuel Assembly Manufacturers. Ultimately, the initial staffing requirements will be determined by the Operations Manager, based in part on human factors engineering inputs such as tasks analysis, OER, HSI design, procedures development, and V&V. MFFF operational staffing levels will be determined, in part, by evaluating the number of tasks required to be performed by operators, the complexity of the tasks, and the coincidence of the tasks for various plant operating conditions. The evaluation incorporates results from the functional allocation, task analysis, operating experience review, HSI design, procedure development, and the HFE V&V. The number and complexity of tasks determined from the startup and testing phase will be factored into this evaluation.

12.7.1 Operations

12.7.1.1 Plant Management

Operation of a nuclear fuel fabrication facility requires a comprehensive organization of trained personnel as well as an underlying program of documented procedures to support this organization. Overall responsibility for safe and efficient operation of the facility and the quality of the fuel product lies with Plant Management. Responsibilities include preparation of operating procedures, staffing and training of qualified plant personnel, implementation of a maintenance program and preparation of maintenance procedures, implementation of safe work practices and emergency response programs, facilitating a quality assurance program that assures product quality and promotes overall process quality.

12.7.1.2 Plant Operations

The plant will be licensed for operations by the NRC. As such, specific and detailed operating procedures must be followed to ensure safety of the public and plant workers. All activities conducted by plant personnel are controlled by detailed written procedures and work instructions. Procedures exist for both normal and emergency situations. Operators and technicians are qualified to conduct specific functions and can only perform those functions for which they have a current qualification. During normal operations, there are relatively few manual operations related to fuel fabrication. Many of the processes are controlled automatically by computers and programmable logic controllers (PLCs), with operator oversight from remote control rooms with only as-required operator interventions. The computer programming logic is verified during development and thoroughly tested during plant startup. A “supervision” computer directs material control and accountability.

Shift staff teamwork and communications is based on the reference plants and NRC licensed fuel assembly manufacturers. Generally, the Operations Manager should meet with the staff to provide top level instructions for the work. The AP Manager and MP Manager should develop their workbooks containing work instructions for their shifts. This information should be provided to the Shift Leaders who in turn should be composing a unit by unit workbook to accomplish the instructions given by the Managers. The Operators should have their own workbook containing very specific instructions for the work and also to record results, generally maintaining a record of the process. It is expected that during shift changeover there will be approximately 30 minutes of overlap to allow the operating shift to brief and update the on-coming shift.

12.7.2 Maintenance

The Maintenance Department is headed by the Maintenance Department Manager (MDM) who will report to the Operations Manager.

The MDM is responsible for the overall management of the MFFF Maintenance Department. The maintainability function (i.e., maintenance, testing, surveillance, and availability) for plant operation is supported by HFE design principles and practices (for example, accessibility to IROFS components).

12.8 PROCEDURE DEVELOPMENT

Procedures are essential to the MFFF safety because procedures support and guide personnel interactions with plant systems and their response to process-related events.

MFFF procedures for IROFS, including IROFS administrative controls, incorporate HFE principles and other design criteria to develop procedures that are technically accurate, comprehensive, easy to utilize, and validated. As appropriate, these procedures include generic technical guidance, plant and system operations, abnormal and emergency operations, tests (e.g., preoperational, startup, and surveillance), and alarm response. The MFFF operations, maintenance, testing, and surveillance procedures are developed during the design and construction management phase. MFFF procedures are described in Chapter 15.

Procedures are developed or modified to reflect the characteristics and functions of the modifications for “Americanizing” the MFFF from the reference facilities (for example, instead of using the nomenclature “lighting executive switch” we would label as “master light switch”). The scope of the procedures for HFE review will include Administrative IROFS covering emergency operating procedures; procedures for startup, operation, and shutdown; procedures for recovery from a “frozen” process; alarm response; and possible abnormal conditions. All applicable procedures will be verified and validated for correctness and ability to be carried out as required. Changes or modifications to procedures will be again verified.

12.8.1 Procedures

Procedures will incorporate appropriate HFE principles, practices and guidance criteria (NUREG-0700) into the text format and presentation to aid legibility, readability, and comprehension. Procedures will be technically accurate, comprehensive, explicit and validated. The HFE Team will verify the hard copy Administrative Control procedures (IROFS) of the ISA required Administrative Controls and conduct a validation of the administrative procedures by observing operator “walk through” or “talk through” of the administrative procedure the operator is required to perform. The validation will include all task analyses identified as “generic” ISA Administrative Controls.

12.9 TRAINING PROGRAM DEVELOPMENT

The operator training program for the MFFF will address active and passive Engineered IROFS and IROFS Administrative Controls. The operator training incorporates the results of the task analysis. All personnel (except visitors) that have a need for MFFF plant access will be provided a General Employee Training (GET), along with more specified training according to position requirements. All the various functional groups of employees will be required to be trained in the aspects of their job responsibilities. Required training is all the training exercises that have been assigned to an individual. An individual will be assigned role-required training and other training that may develop skills and knowledge but is not necessary for role qualification. Training requirements are tied to individuals whereas qualification requirements are tied to roles. MFFF training is described in Chapter 15.

12.10 VERIFICATION AND VALIDATION

Verification and validation (V&V) that the MFFF final design adequately incorporates HFE for IROFS administrative controls will be completed prior to plant operation. The MFFF V&V evaluation comprehensively determines that the design conforms to MFFF HFE design principles, practices and guidelines and that it enables personnel to successfully perform their tasks to achieve MFFF safety and other operational goals. The MFFF HFE Program will review all of the Administrative Controls (IROFS) procedures using first a review of the procedure itself, followed by observing operator “walk through” or “talk through” of the administrative procedure the operator is required to perform. It should be noted that with the exception of Integrated System Validation, the majority of this section mainly addresses verification of HSIs associated with Enhanced Administrative and Administrative IROFS and personnel activities that support safety, such as maintenance, testing, and surveillance on IROFS equipment. HFE will review the Operations’ Reliability, Accessibility, Maintenance, and Inspection (RAMI) checklists to ensure HFE guidance is provided in the RAMI checklists; then, the checklist is used for review of IROFS equipment. The MFFF HFE verification will involve two types of Design Verification: HSI Task Support Verification and HFE Design Verification. HSI Task Support Verification is an evaluation to verify that the HSI supports personnel task requirements as defined by task analyses. HEDs (see § 12.2.4) are identified for: (1) personnel task requirements that are not fully supported by the HSI, and (2) the presence of HSI components which may not be needed to support personnel tasks or which may impede personnel tasks. HFE Design Verification is a static evaluation that verifies the HSI is designed to accommodate human capabilities and limitations as reflected in NUREG-0700 Rev 2. If necessary, additional resources will be used such as using MIL-STD-1472G as a backup verification tool. HEDs are identified if the design is inconsistent with HFE guidelines.

Integrated System Validation is an evaluation using performance-based tests to determine whether an integrated system design (i.e., hardware, software, and personnel elements) meets performance requirements and acceptably supports safe operation of the MFFF. HEDs are identified if performance criteria are not met. HED resolution is an evaluation to provide reasonable assurance that the HEDs identified during the V&V activities have been acceptably assessed and resolved. HED resolution is an activity that should be performed iteratively with V&V. The MFFF process Lead Engineer or process Responsible Engineer may address and resolve issues identified during a V&V activity prior to conducting other V&V activities.

The preferred order is HSI Task Support Verification, HFE Design Verification, and Integrated System Validation, although iteration may be necessary, and these verifications may be conducted in parallel where the tasks, design, and integration present the opportunity for parallel verification. V&V is considered a test that final design requirements are met. The V&V process is conducted in accordance with MFFF design control and configuration management requirements, and includes:

- HSI task support verification HFE design verification
- Integrated system validation
- Human factors issue resolution verification
- Final HFE/HSI design verification.

12.10.1 HSI Task Support Verification

The MFFF HFE review will verify that the HSI design is appropriately provided yet minimizes the incorporation of alarms, information, displays, and control capabilities that unnecessarily complicate personnel actions.

12.10.2 HFE Design Verification

The HFE review compares the IROFS HSI items inventoried, against the Human Factors guidelines. Resolution of identified discrepancies is documented prior to MFFF operation.

The HFE verification includes verifying the HSI inventory and characterization accurately describes all HSI displays, controls, and related equipment that are within the defined scope of the HEPP. This means verifying the Administrative IROFS HSI. Documentation of HSI components is included in the HSI inventory and characterization. The HSI inventory is based on the best available information sources. Equipment lists, design specifications, and drawings describe HSI components. These descriptions should be compared by directly observing the components, both hardwired and computer-generated, to verify that the inventory accurately reflects their current state.

Of particular importance is the HSI task support verification wherein the HFE program verifies that the HSI provides all alarms, information, and control capabilities required for personnel tasks. The criteria for task support verification will come from earlier task analyses of HSI requirements for performance of personnel tasks. The HSIs and their characteristics (as defined in the HSI inventory and characterization) are compared to the personnel task requirements identified in the task analysis.

To simplify the application of guidelines and reduce redundancy when reporting findings, the guidelines may be applied to features of the HSI as follows:

- Global features - global HSI features are those relating to the configuration and environmental aspects of the HSI, such as Control Room(s) layouts, general workstation configurations, lighting, noise, heating, and ventilation. These aspects of the review (e.g., control room lighting) tend to be evaluated only once.
- Standardized features - standardized features are those that were designed using HFE guidelines applied across individual controls and displays (e.g., display screen organization, display format conventions, and coding conventions). Therefore, their implementation should be more consistent across the interface than features that were not designed with guidelines. Thus, for example, if display labeling is standardized by NUREG-0700 HFE guidelines, then display labels can be spot-checked rather than being verified individually.
- Detailed features - detailed features are the aspects of individual HSIs that are not addressed by general HFE guidelines. The latter can be expected to be more variable than the standardized design features. For each guideline, it should be determined whether the HSI is "acceptable" or "discrepant" (an HED) from the guideline (therefore, potentially unacceptable). “Acceptable” will be indicated only if there is total compliance - only if every instance of the item is fully consistent with the criteria established by the HFE guidelines. If there is any instance of noncompliance, full or partial, then an evaluation of “discrepant” will be given, and a notation made as to where noncompliance occurs. The identified discrepancies may be justified and accepted during their review.

Discrepancies should be evaluated as potential indicators of additional issues. For example, identifying an inappropriate format for presenting data on an individual display should be considered a potential sign that other display formats could be incorrectly used or that the observed format is inappropriately used elsewhere. As a result, the sampling strategy could be modified to encompass other display formats. In some cases, discovering these discrepancies could warrant further review in the identified areas of concern.

12.10.3 Integrated System Validation

Integrated system validation is the process by which an integrated system design (i.e., hardware, software, and personnel elements) is evaluated using performance-based tests to determine whether it acceptably supports safe operation of the MFFF. Walk-throughs and talk-throughs of plant procedures will be performed to determine the adequacy of HSIs to support plant operation. The design verification includes both HSI task support verification and HFE design verification. HSI task support verification evaluates that the HSI supports operator task requirements as defined by task analysis. HEDs are identified when the HSI does not fully support the identified operator task requirements (i.e., controls or information is not available or not displayed in the proper format for the specific task) or when HSI components exist that may not be needed to support operator tasks or HSI components that may impede operator tasks. HFE design verification is a static evaluation that verifies that the individual HSI components and details accommodate the human capabilities and limitations reflected in HFE guidelines. HEDs are identified if the design is inconsistent with the project specific HFE guidelines. Accomplishing these verifications does not eliminate the need for integrated system validation. MFFF personnel should perform operational events using computer-based interactive displays or other suitable representation of the system to determine its adequacy to support safety operations. This should be undertaken after significant HEDs that were identified in verification reviews have been resolved, since these will negatively affect performance and, therefore, the results of validation.

The HFE Team will determine the methodology used for the ISV and the scenarios for consideration in the review. The most probable methodology will be hands on, meaning walkthrough and observation of operation actions. Scenarios for control room operator IROFS actions will be evaluated by the HFE Team. Performance measures will need to identify that the operator understands the state of the system process, can navigate process screens on the SCADA units, knows how to carry out emergency procedures, and has all the information and controls needed to accomplish the task. Acceptance criteria for Administrative Controls (IROFS) must demonstrate that human actions are 100% correct (i.e., that the right actions are selected and that the operator can perform those actions without error). The final procedure for Administrative Controls is evaluated and consulted by the HFE review team to ensure all actions are accomplished in their proper order and that the operator has the required information and controls to accomplish the Administrative Control.

For the case of the MFFF modification from the reference facilities, the applicability and scope of integrated system validation may vary. An integrated system validation is reviewed for all modifications that may (1) change personnel tasks; (2) change tasks demands, such as changing task dynamics, complexity, or workload; or (3) interact with or affect HSIs and procedures in ways that may degrade performance. Integrated system validation may not be needed when a modification results in minor changes to personnel tasks such that they may reasonably be expected to have little or no overall effect on workload and the likelihood of error.

Detailed objectives will be developed to provide evidence that the integrated system adequately supports MFFF personnel in the safe operation of the MFFF. The test objectives and scenarios will be developed to address aspects of performance that are affected by the modification design, including personnel functions and tasks affected by the modification. The objectives should be to:

- Validate that for each Administrative IROFS, the design provides adequate alerting, information, control, and feedback capability for human functions to be performed under normal plant operations, transients, design-basis accidents
- Validate that the Administrative IROFS tasks can be accomplished within time and performance criteria, with a high degree of operating crew situation awareness, and with acceptable workload levels that provide a balance between a minimum level of vigilance and operator burden
- Validate that the operator interfaces minimize operator error and provide for error detection and recovery capability when errors occur

- Validate that the crew can make effective transitions between the HSIs and procedures in the accomplishment of their tasks and that interface management tasks such as display configuration and navigation are not a distraction or undue burden
- Validate that the integrated system performance is tolerant of failures of individual HSI features
- Identify aspects of the integrated system that may negatively affect integrated system performance.

12.10.4 Human Factors Issue Resolution Verification

The resolution of HEDs identified during the design process is verified. Any significant HFE issues are reviewed, dispositioned, and documented.

12.10.5 Final HFE/HSI Design Verification

The final HFE/HSI design verification process will include verification of the needed controls and displays identified in the task analysis, categorizing each by type, and identifying the applicable HFE specifications and requirements prior to MFFF operation.

Aspects of the design not addressed during the V&V are evaluated later using an appropriate strategy or method. Aspects of the design addressed at this stage may include design characteristics (e.g., displays for plant-specific design features) and features that cannot properly be evaluated during the MFFF HFE program review (e.g., main control room noise and HVAC).

The final (i.e., as-built) HSIs, procedures, and training program are compared with the detailed design description to verify that they conform to the design that resulted from the HFE design process activities. The start-up and test program is the process by which the constructed facility is reconciled against the final design. The HFE Team will ensure IROFS having HMI are reconciled between the "as built" and the final detailed design description. All of the Control Room IROFS will be reviewed. Emergency Procedures are reviewed to ensure the procedure, the required displays (including labeling) and controls, and the human action are compatible and in accordance with the final design description. Alarms and the required operator alarm responses will be reviewed for comparison between the "as built" and the detailed final design. Identified discrepancies are either corrected or justified. HFE-related issues will be verified as having been adequately resolved.

The HFE/HSI design verification verifies the HFE considerations of the following aspects of the HSI design against the HFE HFDG and NUREG-0700 Rev. 2:

- Layout and arrangements for control rooms with HSI equipment
- Communications equipment
- Environmental guidelines are achieved, especially lighting in regards to HSI with visual displays
- Habitability systems

- Operating procedures and job aids
- Training manuals.

The design implementation verifies:

- Aspects of the design are either partially verified or unverified prior to operation at the MFFF site
- The as-built HSI design is consistent with final design specifications, user and trainee manuals, and operating and maintenance procedures
- The final control rooms and local control station layouts
- Any design modifications (such as, display changes) resulting from pre-operational and start-up testing
- Resolution of any open HFE issues
- The final installed design and its performance criteria are described and documented.

12.10.6 Design Basis

The MOX Services Human Factors Engineering Program is applied to personnel activities identified as IROFS (i.e., Item Relied On For Safety) consistent with the findings of the Integrated Safety Analysis (ISA). The application of HFE to personnel activities ensures that the potential for human error in the facility operations is addressed during the design of the MFFF by facilitating correct, and inhibiting wrong, decisions by personnel and by providing means for detecting and correcting or compensating for error. In addition, it will be important for HFE to determine through review that the operators have the appropriate controls and instrumentation available to confirm proper operation of the automated safety systems and controls under all design basis conditions. HFE Program, "personnel activities" represent personnel activities identified as IROFS (i.e., Enhanced Administrative Controls and Administrative Controls) and personnel activities that support safety, such as maintenance, testing, and surveillance. The following information represents the design basis attributes for Human Factors Engineering.

The MOX Project Human Factors Engineering Program Plan (HEPP) establishes the human factors engineering program requirements and management measures to control or direct the human factors engineering activities related to the design, and operation of the MOX Fuel Fabrication Facility (MFFF). The plan includes:

- HFE principles and practices that are applied to those personnel activities identified by the Integrated Safety Analysis (ISA) as Enhanced Administrative Controls (IROFS), Simple Administrative Controls (IROFS) as well as active and passive engineered IROFS.
- MFFF modifications that incorporate the reference plant's Operating Experience Reviews or Lessons Learned.
- HFE program reviews which are integrated into the MFFF final design and design change control processes.

- Task analysis of the Administrative Controls that is conducted to provide identification of operator activities and understanding of operator actions, the Human-System Interfaces (HSIs) needed in performance of operator activities, and the consequences. The task analysis includes startup, emergency procedures, shutdown, and operator defense-in-depth actions.
- A Control Room summary report lists HSI for each control room, critical operator actions, control room environment, and explains the how and why of the control rooms arrangements.
- The HFE Team observation of operator "walk through" or "talk through" of the administrative procedure the operator is required to perform to validate the adequacy of HSIs to support plant operation.
- Verification and validation (V&V) to confirm that the design incorporates HFE to HSI in a manner that enables the successful completion of operator activities. The HFE Verification and Validation Plan addresses HFE Operational Condition Sampling, HFE Design Verification, Integrated System Validation, and Human Engineering Discrepancies (HEDs).

13.0 SAFEGUARDS AND SECURITY

13.1 PHYSICAL PROTECTION

CB&I AREVA MOX Services, LLC (MOX Services) has submitted under separate cover the Physical Protection Plan for the Mixed Oxide (MOX) Fuel Fabrication Facility.

13.2 TRAINING AND QUALIFICATION

MOX Services has submitted under separate cover the Training and Qualification Plan for Security Personnel for the MOX Fuel Fabrication Facility.

13.3 MATERIAL CONTROL AND ACCOUNTING

MOX Services has submitted under separate cover the Fundamental Nuclear Material Control Plan for the MOX Fuel Fabrication Facility.

13.4 SAFEGUARDS CONTINGENCY

MOX Services has submitted under separate cover the Safeguards Contingency Response Plan for the MOX Fuel Fabrication Facility.

14.0 EMERGENCY MANAGEMENT

CB&I AREVA MOX Services, LLC has submitted under separate cover to the U.S. Nuclear Regulatory Commission an evaluation showing that the maximum dose to a member of the public offsite due to a release of radioactive materials will not exceed 1 rem effective dose equivalent, or an intake of 2 milligrams of soluble uranium. Therefore, in accordance with Title 10 of the Code of Federal Regulations §70.22(i)(1)(i), an Emergency Plan is not required to be submitted.

15.0 MANAGEMENT MEASURES

CB&I AREVA MOX Services, LLC (MOX Services) has established management measures, an administrative and programmatic framework that ensures that facility items relied on for safety (IROFS) are available and reliable to perform their safety function when needed, and that work is conducted efficiently and in a manner that protects workers, the public, and the environment. This framework includes configuration management, maintenance, training and qualification, procedures, audits and assessments, incident investigations, and records management. Within this framework are the administrative and programmatic measures implemented for Mixed Oxide (MOX) Fuel Fabrication Facility (MFFF) IROFS to ensure safety. This chapter describes the management measures implemented for MFFF IROFS. These management measures are implemented in accordance with a quality assurance (QA) program established in accordance with Title 10 of the Code of Federal Regulations (CFR) Part 50, Appendix B.

This chapter makes frequent reference to the MOX Services QA program described in the MOX Project Quality Assurance Plan (MPQAP), because management measures are closely related to quality assurance requirements. The MPQAP has previously been approved by the U.S. Nuclear Regulatory Commission (NRC).

Application of Management Measures

Management measures are applied to IROFS to ensure that they are reliable and available upon demand. The set of applied management measures consists of applicable elements of the following management measures programs: quality assurance, configuration management, maintenance, training and qualification of plant personnel, plant procedures, audits and assessments, incident investigations, and records management.

Management measures are assigned based on the following types of IROFS classifications and the risk reduction level attributed to that particular IROFS:

- Passive Engineered Controls (PEC) – A device that uses only fixed physical design features to maintain safe process conditions without any required human action
- Active Engineered Controls (AEC) – A physical device that uses active sensors, electrical components, or moving parts to maintain safe process conditions without any required human action
- Enhanced Administrative Controls (EAC) – A procedurally required or prohibited human action, combined with a physical device that alerts the operator that the action is needed to maintain safe process conditions, or otherwise adds substantial assurance of the required human performance (i.e., augmented administrative control)
- Administrative Controls (AC) – A procedural human action that is prohibited or required to maintain safe process conditions (i.e., a simple administrative control).

The following information provides a brief overview of the MOX management measures programs assigned to IROFS.

Quality Assurance – The MOX QA program is described in the MPQAP and established in accordance with Title 10 of the CFR Part 50, Appendix B. The MPQAP describes the quality assurance requirements for quality-affecting activities on the project and coincide with the 18 criteria of 10 CFR 50, Appendix B.

Configuration Management – Configuration management processes and requirements are required to maintain effective control of the MFFF as-designed facility arrangement and operation. This provides reasonable assurance that IROFS safety functions are properly controlled, and that changes to the facility are properly addressed, evaluated, and approved. The configuration management processes and requirements are described in the MFFF Configuration Management Plan. The plan consists of the following five basic plan elements: (1) plan management, (2) technical requirements, (3) change control, (4) document control, and (5) audits and assessments. MOX implements these five elements to maintain consistency among design requirements, design basis, physical configuration, and facility documentation throughout the life cycle of the facility.

Maintenance – MOX implements a maintenance program that includes provisions for planned, scheduled, and unplanned maintenance to ensure MFFF equipment will be available and reliable to perform their intended safety functions. Maintenance for IROFS is developed and conducted to maximize availability and reliability for assurance that the safety functions and ISA requirements will be achieved. Maintenance activities include surveillances, preventive maintenance, and corrective maintenance. Surveillances are planned and scheduled systematic procedures conducted at required intervals to monitor the performance of IROFS equipment for assurance that they continue to meet their performance specifications, including availability and reliability goals. Surveillances may consist of measurements, inspections, functional tests, and calibration checks. Preventive maintenance activities are planned and scheduled and include actions that detect, preclude, or mitigate degradation and to sustain or extend the useful life of SSCs. Corrective maintenance is performed to repair or replace equipment that has failed or is significantly degraded to the point that failure is imminent and no longer conforms to or is incapable of performing its intended safety function. Post maintenance functional tests are performed to confirm equipment functions have been restored to normal conditions. Maintenance work is performed through a coordinated and structured work control process that integrates with ongoing production activities and requirements. This work control process includes representation from various disciplines, such as radiation protection, safety, operations and others, as necessary, for complete pre-planning of the required work. Coordination of work activities includes items such as work orders, procedures, schedules, radiation work permits, and lockout/tagout requirements.

Training and Qualification – Training and qualification of MOX employees is essential to the safe and successful design, construction, testing, and operation of the MFFF. Training is commensurate with the complexity of assigned tasks. Lesson plans are used for classroom and on-the-job training as required to assure consistent presentation of subject matter. When design changes or plant modifications are implemented, updates of applicable lesson plans are included in the change control process of the configuration management system. A needs/job analysis is performed and tasks identified to ensure that appropriate training is provided to personnel. Learning objectives identify the training content established by needs/job analyses and position-specific requirements. Lesson plans are developed from learning objectives, which are based on job performance requirements. Trainee mastery of learning objectives is evaluated through observation/demonstration, or oral or written tests. In addition to appropriate classroom training, on-the-job training is used for selected activities when appropriate. Completion of on-the-job training is demonstrated by task performance, where feasible and appropriate. The training program is periodically and systematically evaluated to measure the program's effectiveness in producing competent employees. Trainees provide feedback after completing their classroom training as their evaluation for program improvements. Training records are maintained to support management information needs associated with personnel training, job performance, and employee qualifications.

Procedures – Plant procedures are developed and controlled in accordance with the requirements of the MPQAP. They are broadly categorized as either administrative procedures or operating procedures. Administrative procedures specify controls that apply to specific functions or specific interfaces with other organizational functions. Operating procedures provide specific direction for functional task-based work within an organizational function. Operating procedures include production, maintenance, and emergency procedures. Operating procedures include operating limits and controls, and specific IROFS administrative controls to ensure nuclear criticality safety, chemical safety, fire protection, emergency planning, and environmental protection. Prior to initial use and after major revisions, production and maintenance procedures are verified and validated. The MFFF training program ensures that employees are trained in the use of approved procedures before implementation. To ensure technical accuracy, operating procedures are periodically reviewed by qualified individuals to verify their continued applicability and accuracy.

Audits and Assessment – MOX utilizes two distinct levels of activities (audits and assessments) to evaluate the effectiveness and implementation of QA Program elements and other management measures for IROFS and to address the technical adequacy of the items evaluated. Audits are independently planned and documented evaluations performed by the QA organization. Audits evaluate the scope, status, adequacy, programmatic compliance, and implementation effectiveness of quality-affecting activities. Assessments are management directed evaluations of the scope, status, adequacy, programmatic compliance, and implementation effectiveness of QA and other management measures in their area of responsibility.

Incident Investigations – MOX implements two programs for investigating discrepancies: the corrective action process and incident investigations. The MOX corrective action process is used for identifying, investigating, reporting, tracking, correcting, and preventing recurrence of conditions adverse to quality. Nonconforming materials, parts, or components are identified and controlled in accordance with the MPQAP. Incident investigations are used for investigating unplanned events such as accidents, unexpected transients, operator error, and unacceptable performance deficiencies. An incident investigation is performed by one or more individuals assigned by the manager of production. The process used for the investigation may be similar to that of the corrective action process. Upon completion, a report on the incident and its investigation is made to the production manager, who initiates appropriate action(s), if determined necessary.

Records Management – MOX records are managed in accordance with the records management program under the requirements of the MPQAP. Records management program procedures have been established to address the receipt, processing, indexing, filing, storage, access control, preservation, retrieval, correction, and retention of QA records developed or received by the MOX project.

15.1 QUALITY ASSURANCE

MOX Services, based on option A of NUREG-1718, implements and maintains the QA program in conformance with the applicable requirements of Parts I and II of ASME-NQA-1-1994, as revised by the ASME NQA-1a-1995 Addenda or equivalent as described in the MPQAP. The MPQAP, including any exceptions or alternatives, has been approved by the NRC. A change that would reduce the commitments of the NRC approved QA program is submitted with written justification to the NRC for acceptance, prior to implementation by MOX Services. The MPQAP will be updated as necessary during testing, operation, and deactivation of the MFFF. MOX Services implements the requirements of 10 CFR Part 21, *Reporting of Defects and Noncompliance,* for design, construction, procurement, testing, and operations of Quality Level 1 structures, systems, and components (SSCs) (i.e., IROFS). MPQAP Section 4, *Procurement Document Control*, requires that 10 CFR Part 21 be invoked for procurements of IROFS, unless the procurement is for a Commercial Grade Item.

15.2 CONFIGURATION MANAGEMENT

15.2.1 Configuration Management Policy

MOX Services implements configuration management (CM) processes to ensure design and operation within the design basis of IROFS by: identifying and controlling preparation and review of documentation associated with IROFS; controlling changes to IROFS; and maintaining the physical configuration of the facility consistent with the approved design.

The Integrated Safety Analysis (ISA) of the design determines the IROFS and establishes the safety function(s) associated with each IROFS. Configuration control is accomplished during design through the use of procedures for controlling design, including preparation, review (including interdisciplinary review), design verification where appropriate, approval, release and distribution for use. Quality level classifications are established for the MFFF structures, systems, components, and associated documents. Changes to the approved design are subject to a review to ensure consistency with the design bases of IROFS. Configuration management is also accomplished through design review and design verification, which ensures that design documents are consistent and that design requirements for IROFS are met. Changes identified during construction or testing must be approved by Engineering via a documented engineering change process or an approved non-conformance report prior to change implementation to ensure configuration is maintained and that testing that is specified to demonstrate performance of IROFS is accomplished successfully. Periodic audits and assessments of the configuration management program and of the design confirm that the system meets its goals and that the design is consistent with the design bases. The corrective action process occurs in accordance with the MPQAP and associated procedures in the event problems are identified. Prompt corrective actions are developed as a result of incident investigations or in response to audit or assessment results.

Configuration management provides the means to establish and maintain the essential features of the design basis of IROFS, including the ISA. As the project progresses from design and construction to operation, configuration management is maintained. Procedures will define the turnover process and responsibilities.

The administrative instructions for modifications during the operations phase are contained in procedures that are approved, including revisions, by the Functional Area Manager. The change procedure contains the following items necessary to ensure quality in the modification program:

- The technical and quality requirements which shall be met to implement a modification
- The requirements for initiating, approving, monitoring, designing, verifying, and documenting modifications. The facility modification procedure shall be written to ensure that policies are formulated and maintained to satisfy the MPQAP, as applicable.

Each change to the facility or to activities of personnel shall have an evaluation performed in accordance with the requirements of 10 CFR 70.72, as applicable. Each modification shall also be evaluated for any required changes or additions to the facility's procedures, personnel training, testing program, or regulatory documents, as applicable.

For any change (i.e., new design or operation, or modification to the facility or to activities of personnel, e.g., site structures, systems, components, computer programs, processes, operating procedures, management measures), that involves or could affect the integrated safety analysis, the impacts shall be evaluated and documented. Prior to implementing the change, it shall be demonstrated that the change does not affect the safety basis in accordance with 10 CFR 70.72. Changes that impact the safety basis require NRC approval prior to implementation.

Each modification is also evaluated and documented for radiation exposure to minimize worker exposures in keeping with the facility as low as reasonably achievable (ALARA) program, criticality and worker safety requirements and/or restrictions. Other areas of consideration in evaluating modifications may include, but are not limited to the review of:

- Modification cost
- Lessons learned from similar completed modifications
- QA requirements
- Potential operability or maintainability concerns
- Constructability concerns
- Post-modification testing requirements
- Environmental considerations
- Human factors
- Integrated safety analysis.

After completion of a modification to a structure, system, or component, the modification Responsible Manager, or designee, shall ensure that applicable testing has been completed to ensure correct operation of the system(s) affected by the modification and documentation regarding the modification is complete. In order to ensure operators are able to operate a modified system safely, when a modification is complete, documents necessary (e.g., the revised process description, checklists for operation and flowsheets) are made available to operations and maintenance departments prior to the start-up of the modified system. Appropriate training on the modification is completed before a system is placed in operation. A formal notice of a modification being completed is distributed to the appropriate managers. Drawings incorporating the modification are completed in accordance with the design control procedures. These records shall be identifiable and shall be retained in accordance with the records management procedures.

15.2.2 Implementation of Configuration Management

During the design phase of the project, configuration management is based on the design control provisions and associated procedural controls over design documents to establish and maintain the technical baseline. Design documents, including the ISA, that provide design input, design analysis, or design results specifically for IROFS are identified with the appropriate quality level. These design documents undergo interdisciplinary review during the initial issue and during each subsequent revision. During the construction phase of the project, changes to drawings and specifications issued for construction, procurement, or fabrication are systematically reviewed and verified, evaluated for impact, including impact to the ISA, and approved prior to implementation. Proper implementation is verified by the Quality Assurance organization.

In order to provide for the continued safe and reliable operation of the MFFF IROFS, measures are implemented to ensure that the quality of these IROFS is not compromised by planned changes (modifications). Upon acceptance by Operations, the Plant Manager is responsible for the design of, and modifications to, facility items relied on for safety. The design and implementation of modifications are performed in a manner so as to assure quality is maintained in a manner commensurate with the remainder of the system which is being modified, or as dictated by applicable regulations.

15.2.3 Organization

The MOX Services President is responsible for the overall implementation of the configuration management program. This includes development and approval of plans and policies necessary to provide overall program direction.

The configuration management program is administered by the Vice President - Engineering during design. Engineering includes engineering disciplines. The discipline engineers have primary technical responsibility for the work performed by their disciplines. The Responsible Managers are responsible for the conduct of interdisciplinary reviews as discussed previously in this section. Reviews are also conducted, as appropriate, by construction management, operations, ES&H, QA, and support services personnel. The design control process also interfaces with the document control and records management process via procedures.

During construction, the Vice President - Construction has responsibility for configuration management through establishment and maintenance of processes and procedures used during construction of the facility.

During operational testing, operation, and deactivation, the Plant Manager is responsible for ensuring the implementation of configuration management.

The various MOX Services departments and subcontractors perform quality-related activities. The primary MOX Services subcontractors work to the MPQAP. Some MOX Services subcontractors are responsible for development of their respective QA Programs, which shall be consistent with the requirements of the MPQAP for those activities determined to be within the scope of the MPQAP. The interfaces between subcontractors and MOX Services or among subcontractors shall be documented. MOX Services and subcontracted personnel have the responsibility to identify quality problems. Disagreements that cannot be resolved are elevated to the next level of management for resolution. If this level of management cannot resolve the issue, then the issue is elevated through successive layers of management until resolution is achieved.

15.2.4 Scope of CM Program

The scope of configuration management includes IROFS identified by the integrated safety analysis and any items which may affect the safety function of the IROFS. Documents subject to configuration management include calculations, safety analyses, design criteria, engineering drawings, system descriptions, technical documents, operating procedures and specifications that establish design and safety requirements for IROFS. During the design phase, these documents are maintained under configuration management when initially approved.

The number of documents included in the configuration management program increases throughout the design process. As drawings and specifications related to IROFS are prepared and issued for procurement, fabrication, or construction, these documents are included in configuration management.

During construction, initial startup, and operations, the scope of documents under configuration management similarly increase to include, as appropriate: vendor data; test data; inspection data; initial startup, test, operating and administrative procedures as applicable to IROFS and nonconformance reports. These documents include documentation related to IROFS that is generated through functional interfaces with QA, maintenance, and training and qualifications of personnel. Configuration management procedures will provide for evaluation, implementation, and tracking of changes to IROFS, and processes, equipment, computer programs, and activities of personnel that impact IROFS.

Configuration management is implemented through or otherwise related to other management measures. Key interfaces and relationships to other management measures are described below:

Quality Assurance - The QA program establishes the framework for configuration management and other management measures for IROFS and items that affect the function of the IROFS.

Records Management - Records associated with IROFS are generated and processed in accordance with the applicable requirements of the QA Program and provide evidence of the conduct of activities associated with the configuration management of those IROFS.

Maintenance - Maintenance requirements are established as part of the design basis, which is controlled under configuration management. Maintenance records for IROFS provide evidence of compliance with preventative and corrective maintenance schedules.

Training and Qualifications - Training and qualification are controlled in accordance with approved project procedures. Personnel qualifications and/or training to specific processes and procedures are management measures that support the safe design, operation, maintenance, and testing of IROFS. Also, work activities that are themselves IROFS, (i.e., administrative controls) are proceduralized, and personnel are trained and qualified to these procedures. Training and qualification requirements and documentation of training may be considered part of the design basis controlled under configuration management. Training and Qualification of plant personnel is described in Section 15.4.

Audits and Assessments / Incident Investigation - Audits, assessments, and incident investigations are described in Sections 15.6, Audits and Assessments, and 15.7, Incident Investigations and Corrective Action Process. Corrective actions identified as a result of these management measures may result in changes to design features, administrative controls, or other management measures (e.g., operating procedures). The Corrective Action Program (CAP) is described in Section 15.7. Changes are evaluated under the provisions of configuration management through the QA Program and procedures. Periodic assessments of the configuration management program are also conducted in accordance with the audit and assessment program described in Section 15.6.

Procedures - Operating, administrative, maintenance, and emergency procedures are used to conduct various operations associated with IROFS and will be reviewed for potential impacts to the design basis. Also, work activities that are themselves IROFS, (i.e., administrative controls) are contained in procedures.

15.2.5 Change Control

Configuration management includes those activities conducted under design control provisions for ensuring that design and construction documentation is prepared, reviewed, and approved in accordance with a systematic process. This process includes interdisciplinary reviews appropriate to ensure consistency between the design and the design bases of IROFS. During construction, it also includes those activities that ensure that construction is consistent with design documents. Finally, it includes activities that provide for operation of the IROFS in accordance with the limits and constraints established in the ISA, and that provide for control of changes to the facility in accordance with 10 CFR 70.72.

Configuration management also includes records to demonstrate that personnel conducting activities that are IROFS are appropriately qualified and trained to conduct that work.

Implementing documents are controlled within the document control system. These documents support configuration management by ensuring that only reviewed and approved procedures, specifications and drawings are used for procurement, construction, installation, testing, operation, and maintenance of IROFS, as appropriate.

Procedures control changes to the design documents. The process includes an appropriate level of technical, management, and safety review and approval prior to implementation. During the design phase of the project, the method of controlling changes is the design control process described in the implementing procedures. This process includes the conduct of interdisciplinary reviews, design reviews and design verification that constitute a primary mechanism for ensuring consistency of the design with the design bases. During both construction and operation, appropriate reviews to ensure consistency with the design bases of IROFS and the ISA, respectively, will similarly ensure that the design is constructed and operated/modified within the limits of the design basis. Additional details are provided below.

Changes to the design include a systematic review of the design bases for consistency. In the event of changes to reflect design or operational changes from the established design bases, the integrated safety analysis are properly modified, reviewed, and approved prior to implementation. Approved changes are made available to personnel through the document control function discussed previously in this section.

During design, the method of ensuring consistency between documents, including consistency between design changes and the safety analyses, is the interdisciplinary review process. The interdisciplinary reviews ensure design changes either (1) do not impact the ISA, (2) are accounted for in subsequent changes to the ISA, or (3) are not approved or implemented. Prior to issuance of the License, MOX will notify the NRC of potential changes that reduce the level of commitments or margin of safety in the design bases of IROFS.

When the project enters the construction phase, changes to documents issued for construction, fabrication, and procurement will be documented, reviewed, approved, and posted against each affected design document. Vendor drawings and data also undergo an interdisciplinary review to ensure compliance with procurement specifications and drawings, and to incorporate interface requirements into facility documents.

During construction, design changes will continue to be evaluated against the approved design bases. Changes are expected to the design as detailed design progresses and construction begins. A systematic process consistent with the process described above will be used to evaluate changes in the design against the design bases of IROFS and the ISA. The configuration change process fully implements the provisions of 10 CFR 70.72, including reporting of changes made without prior NRC approval as required by 10 CFR 70.72(d)(2) and (3). Changes that require Commission approval, will be submitted as required by 10 CFR 70.72(d)(1) and the change will not be implemented without prior NRC approval.

During the operations phase, changes to design will also be documented, reviewed, and approved prior to implementation. MOX will implement a change process that fully implements the provisions of 10 CFR 70.72. Measures are provided to ensure responsible facility personnel are made aware of design changes and modifications that may affect the performance of their duties.

In order to provide for the continued safe and reliable operation of MFFF IROFS, measures are implemented to ensure the quality of these IROFS are not compromised by planned changes (modifications). Upon acceptance by Operations, the Plant Manager is responsible for the design of and modifications to IROFS. The design and implementation of modifications are performed in a manner so as to assure quality is maintained in the remainder of the system that is being modified, or as dictated by applicable regulations.

During deactivation, configuration management incorporated into the original design and modifications throughout operation facilitate deactivation of the facility.

The administrative instructions for modifications are contained in a facility administrative procedure that is approved, including revisions, by the Functional Area Manager. The modification procedure contains the following items necessary to ensure quality in the modification program:

- The technical and quality requirements which shall be met to implement a modification.
- The requirements for initiating, approving, monitoring, designing, verifying, and documenting modifications. The facility modification procedure shall be written to ensure that policies are formulated and maintained to satisfy the MPQAP, as applicable.

Each change to the facility or to activities of personnel shall have an evaluation performed in accordance with the requirements of 10 CFR 70.72, as applicable. Each modification shall also be evaluated for any required changes or additions to the facility's procedures, personnel training, testing program, or regulatory documents.

For changes (e.g., new design or operation, or modification to the facility or to activities of personnel, IROFS, computer programs, processes, operating procedures, management measures), that involves or could affect the integrated safety analysis, the impacts shall be evaluated and documented. Prior to implementing the change, it shall be demonstrated that the change does not affect the safety basis in accordance with 10 CFR 70.72.

15.2.5.1 Identification of Changes

Design requirements and associated design bases are established and maintained by the Engineering organization during design and construction and by the Plant Manager during operations. The configuration management controls on design requirements and the integrated safety analysis of the design bases are described previously in this section.

The design bases are documented in the design documents (e.g., calculations, safety analysis, engineering drawings, system descriptions, technical documents, and specifications) and Licensing Bases Documents. Design requirements are derived from the design bases identified above. The design documents are controlled under the design control provisions of the configuration management program.

IROFS are designated as Quality Level 1. The associated design documents are subject to interdisciplinary reviews, design review and verification. Analyses constituting the integrated safety analysis are subject to the same requirements. Changes to the design are evaluated to ensure consistency with the design bases. Computer codes used in safety analyses and design of IROFS are also subject to these design control measures, with additional requirements as appropriate for software control, verification, and validation. IROFS are summarized in the Integrated Safety Analysis Summary.

A qualified individual who specifies and includes the appropriate codes, standards, and licensing commitments within the design documents prepares each design document, such as a calculation, specification, procedure, or drawing. This individual also notes any deviations or changes from such standards within the design documentation package. Each design document is then reviewed by another individual qualified in the same discipline. These design inputs are in sufficient detail to permit verification of the document. The manager having overall responsibility for the design function approves the document. The Responsible Manager documents the entire review process in accordance with approved procedures. These procedures include provisions to assure that appropriate quality standards are specified in design documents, including quantitative or qualitative acceptance criteria. The QA Manager conducts audits on the design control process using independent technically qualified individuals to augment the QA audit team.

During the review, emphasis is placed on assuring conformance with applicable codes, standards and license application design commitments. The individuals in engineering assigned to perform the review of a document have full and independent authority to withhold approval until questions concerning the work have been resolved. Design reviews, alternative calculations, or qualification testing accomplish verification of design. The bases for a design, such as analytical models, theories, examples, tables, codes and computer programs must be referenced in the design document and their application verified during check and review. Model tests, when required to prove the adequacy of a concept or a design, are reviewed and approved by the responsible qualified individual. Testing used for design verification shall demonstrate adequacy of performance under conditions that simulate the most adverse design conditions. The tests used for design verification must meet the design requirements.

Qualified individuals other than those who performed the design but may be from the same organization perform design verification. Verification may be performed by the supervisor of the individual performing the design, provided the supervisor did not specify a singular design approach or rule out certain design considerations, and did not establish the design inputs used in the design.

Independent design verification shall be accomplished before the design document (or information contained therein) is used by other organizations for design work or to support other activities such as procurement, construction, or installation. When this is not practical due to time constraints, the unverified portion of the document is identified and controlled. Design verification shall be completed before relying on the item to perform its function. Changes to the design and procurement documents, including field changes, must be reviewed and approved commensurate with the original approval requirements.

15.2.5.2 Review and Approval of Changes

Configuration control is accomplished during design through the use of procedures for controlling design, including preparation, review (including interdisciplinary review and preparation of NSEs and NCSEs as applicable), and design verification where appropriate, approval, and release and distribution for use. Engineering documents are assessed for Quality level classification. Changes to the approved design also are subject to a review to ensure consistency with the design bases of IROFS.

Configuration verification is also accomplished through design verification, which ensures that design documents are consistent and that design requirements for IROFS are met. During construction, in-process verification is conducted by the construction and quality control organizations. During testing to demonstrate performance of IROFS, configuration is verified by the startup and quality organizations.

The MPQAP requires procedures that ensure that work performed shall be accomplished in accordance with the requirements and guidelines imposed by applicable specifications, drawings, codes, standards, regulations, quality assurance criteria and site characteristics.

Acceptance criteria established by the designer are incorporated in the instructions, procedures and drawings used to perform the work. Documentation is maintained, including test results, and inspection records, demonstrating that the work has been properly performed. Procedures also provide for review, audit, approval and documentation of activities affecting the quality of items to ensure that applicable criteria have been met.

Maintenance, modification, and inspection procedures are reviewed by qualified personnel knowledgeable in the quality assurance disciplines to determine:

- The need for inspection, identification of inspection personnel, and documentation of inspection result
- That the necessary inspection requirements, methods, and acceptance criteria have been identified.

Facility procedures shall be reviewed by an individual knowledgeable in the area affected by the procedure on a frequency determined by the age and use of the procedure to determine if changes are necessary or desirable. Procedures are also reviewed to ensure procedures are maintained up-to-date with facility configuration. These reviews are intended to ensure that any modifications to facility items relied on for safety are reflected in current maintenance, operations and other facility procedures.

15.2.5.3 Implementation of Changes

After design documents have been properly prepared, reviewed, and approved by the appropriate parties, the responsible engineer sends the document to document control for distribution. After input into Documentum, documents are electronically routed (distributed) to employees identified on the record submittal form.

When deficiencies are identified which affect the design of IROFS, such deficiencies are documented and resolved in accordance with approved CAP procedures. In accordance with the CAP, the report is forwarded for appropriate review to the responsible manager, who coordinates further review of the problem and revises the design documents affected by the deficiency as necessary. Where required, the responsible manager forwards the report to the engineers in other areas, who coordinate necessary revisions to their affected documents

Design interfaces are maintained by communication among the Functional Area Managers. Methods by which this is accomplished include the following:

- Design documents are reviewed by the responsible engineer or authorized representative.
- Project interface meetings are scheduled and held to coordinate design, procurement, construction and pre-operational testing of the facility. These meetings provide a primary working interface among the organizations.
- Reports of nonconformances are transmitted and controlled by procedures.

During the operational phase, measures are provided to ensure responsible facility personnel are made aware of design changes and modifications that may affect the performance of their duties.

15.2.6 Document Control

15.2.6.1 Storage of Documents

Procedures are established which control the preparation and issuance of documents such as manuals, instructions, drawings, procedures, specifications, procurement documents and supplier-supplied documents, including any changes thereto. Measures are established to ensure documents, including revisions, are adequately reviewed, approved, and released for use by authorized personnel.

Approved documents included in the CM program are stored in the MOX Services electronic document management system (Documentum). Documentum is a tool capable of reporting the status of documents. Records not suitable for storage in this system are stored in accordance with the requirements of MPQAP Section 6, *Document Control*.

Document control procedures require documents to be transmitted and received in a timely manner at appropriate locations including the location where the prescribed activity is to be performed. Distribution of controlled documents is made in accordance with applicable document control procedures.

Superseded documents are retained within Documentum and are controlled by Document Control utilizing a versioning process and by updating the status attribute. Indexes of current documents are generated using Documentum functionality.

15.2.6.2 Identification of Documents

Capabilities to track and retrieve current documents included in the CM program, historical records, and other information by multiple attributes (e.g., document number, document subject, component number, component name, status) are accomplished in accordance with approved procedures.

The system is capable of generating indices of controlled documents, which are uniquely numbered (including revision number). Controlled documents are maintained until cancelled or superseded, and cancelled or superseded documents are maintained as a record, currently for the life of the project or termination of the license, whichever occurs later. Hardcopy distribution of controlled documents is provided when needed in accordance with applicable procedures (e.g., when the electronic document management system is not available).

A part of the configuration management program, the document control and records management procedures, as appropriate, capture various documents. For example:

- Design requirements
- The integrated safety analysis, through the controlled copies of supporting analyses
- Nuclear Safety Evaluations
- Nuclear Criticality Safety Evaluations
- Drawings
- Specifications
- Calculations
- Technical Reports
- Project procedures
- QA Documents
- Maintenance Documents
- Audit and assessment reports
- Operating procedures
- Emergency response plans
- System modification documents

15.2.7 Audits and Assessments

Initial assessment(s) of the configuration management program is performed as part of system turnover upon entering the operations phase. Periodic assessments of the configuration management and design control program are conducted to determine the system's effectiveness and to correct deficiencies. These assessments include review of the adequacy of documentation. Such audits and assessments are scheduled, conducted and documented in accordance with approved procedures.

Periodic audits and assessments of the configuration management program and of the design confirm that the system meets its goals and that the design is consistent with the design bases. Incident investigations occur in accordance with the MPQAP and associated CAP procedures in the event problems are encountered. Prompt corrective actions are developed as a result of incident investigations or in response to adverse audit/assessment results, in accordance with CAP procedures.

15.3 MAINTENANCE

This section outlines the maintenance and functional testing programs to be implemented for the operations phase of the facility. Preventive maintenance activities, surveillance, and performance trending provide reasonable and continuing assurance that IROFS will be available and reliable to perform their safety functions in accordance with the integrated safety analysis (ISA).

The purpose of planned and scheduled maintenance for IROFS is to ensure that the equipment and controls are kept in a condition of readiness to perform the planned and designed functions when required. Appropriate plant management is responsible for ensuring the operational readiness of IROFS under this control. For this reason, the maintenance organization is administratively closely coupled to operations. Maintenance is developed using information from such sources as equipment suppliers, reference plants, lessons learned from other appropriate facilities, and the safety classification (i.e., QL-1 or QL-1LR). A work management group is assigned to plan, schedule, coordinate, track work activities through completion, and maintain the associated records for analysis and trending of equipment performance and conditions. This information is assessed for indicators of areas for adjustments and improvements to methods and frequencies. Should an incident investigation be initiated in accordance with the MFFF Incident Investigation Program, recommendations and corrective actions identified are assessed by the work management group and applied to the respective portions of the Maintenance Program.

In order to provide for the continued safe and reliable operation of the IROFS, measures are implemented to ensure that the quality of the IROFS is not compromised by planned changes (modifications) or maintenance activities. Upon acceptance by Operations, the Plant Manager is responsible for the design of and modifications to IROFS and maintenance activities. The design and implementation of modifications are performed in a manner so as to assure quality is maintained in a manner commensurate with the remainder of the system which is being modified, or as dictated by applicable regulations. The two categories of MFFF equipment are IROFS and non-IROFS.

Maintenance for IROFS is developed and conducted to maximize availability and reliability for assurance that the designed safety functions and ISA requirements will be achieved, when needed. This maintenance is performed under strict procedural controls and the resultant records are maintained as proof of compliance to safety requirements.

Non-IROFS equipment will be maintained commensurate with designed functions. In general, non-IROFS maintenance will be performed to standard industrial practices.

Procedures used to perform maintenance use the applicable requirements of the design and safety analysis documents and meet the requirements of MPQAP Section 5, *Instructions, Procedures, and Drawings*. Where applicable, grading of QA controls is performed in accordance with requirements of MPQAP Section 2.1.2, *Graded Quality Assurance*. Spare and replacement parts are procured, received, accepted, stored, and issued according to the requirements of MPQAP Section 4, *Procurement Document Control*, Section 7, *Control of Purchased Material Equipment, and Services*, Section 8, *Identification and Control of Materials, Parts, and Components*, and Section 13, *Handling, Storage, and Shipping*. Required special processes are performed to meet the requirements of MPQAP Section 9, *Control of Special Processes*. Equipment used to measure and record maintenance and inspection parameters is calibrated in accordance with the requirements of MPQAP Section 12, *Control of Measuring and Test Equipment*. Nondestructive examination, inspection, and test personnel are qualified and certified in accordance with MPQAP Section 2.2.6, *Personnel Indoctrination, Training, and Qualification*. Inspections are performed to meet the requirements of MPQAP Section 10, *Inspection*, and testing required after maintenance conforms to the requirements of MPQAP Section 11, *Test Control*. Maintenance activities meet the requirements of MPQAP Section 14, *Inspection, Test, and Operating Status*. Completed records of maintenance are maintained in the records management system, which meets the requirements of MPQAP Section 17, *Quality Assurance Records.*

15.3.1 Maintenance Categories

Maintenance activities generally fall into the following categories:

- Surveillance/monitoring
- Preventive maintenance
- Corrective maintenance
- Functional tests.

Audits and assessments are performed to assure that these maintenance activities are conducted in accordance with the written procedures and that the processes reviewed are effective. These maintenance categories are discussed in the following sections.

15.3.1.1 Surveillance / Monitoring

Surveillance/monitoring is utilized to detect degradation and adverse trends of IROFS so that action may be taken prior to component failure. The monitored parameters are selected based upon their ability to detect the predominant failure modes of the critical components. Data sources include: surveillance, periodic and diagnostic test results, plant computer information, operator rounds, walk downs, as-found conditions, failure trending, and predictive maintenance. Surveillance/monitoring and reporting is required for IROFS and any administrative controls that could impact the functions of an IROFS.

Plant performance criteria are established to monitor plant performance and to monitor IROFS functions and component parameters. These criteria are established using industry experience, operating data, surveillance data, and plant equipment operating experience. These criteria ensure the reliability and availability of IROFS. The performance criteria are also used to demonstrate that the performance or condition of an IROFS is being effectively controlled through appropriate predictive and repetitive maintenance strategies so that IROFS remain capable of performing their intended function.

Surveillance of IROFS is performed at specified intervals. The purpose of the surveillance program is to measure the degree to which IROFS meet performance specifications. The results of surveillances are trended, and when the trend indicates potential IROFS performance degradation, preventive maintenance frequencies are adjusted or other appropriate corrective action is taken.

Surveillances may consist of measurements, inspections, functional tests, and calibration checks. Incident investigations may identify root causes of failures that are related to the type or frequency of maintenance. The lessons learned from such investigations are factored into the surveillance/monitoring and preventive maintenance programs as appropriate.

Maintenance procedures prescribe compensatory measures, if appropriate, for surveillance tests of IROFS that can be performed only while equipment is out of service.

Records showing the current surveillance schedule, performance criteria, and test results for IROFS will be maintained in accordance with the Record Management System.

Results of surveillance/monitoring activities related to IROFS via the configuration management program will be evaluated by the safety disciplines to determine any impact on the ISA and any updates needed.

15.3.1.2 Preventive Maintenance

Preventive maintenance (PM) includes preplanned and scheduled periodic refurbishment, partial or complete overhaul, or replacement of IROFS, if necessary, to ensure their continued safety function even with unplanned outages. Planning for preventive maintenance includes consideration of results of surveillance and monitoring, including failure history. PM also includes instrument calibration and testing.

The PM program procedures and calibration standards (traceable to the national standards system) enable the facility personnel to calibrate equipment and monitoring devices important to plant safety and safeguards. Testing performed on IROFS that are not redundant will provide for compensatory measures to be put into place to ensure that the IROFS function is performed until it is put back into service.

Industry experience, vendor recommended intervals and data derived from the reference facilities, as applicable, is used to determine initial PM frequencies and procedures. In determining the frequency of PM, consideration is given to appropriately balancing the objective of preventing failures through maintenance against the objective of minimizing unavailability of

IROFS because of PM. In addition, feedback from PM and corrective maintenance and the results of incident investigations and identified root causes are used, as appropriate, to modify the frequency or scope of PM. The rationale for deviations from industry standards or vendor recommendations for PM shall be documented.

After conducting preventive maintenance on IROFS, and before returning an IROFS to operational status, functional testing of the IROFS, if necessary, is performed to ensure the IROFS performs its intended safety function. Functional testing is described in detail in Section 15.3.1.4, Functional Tests.

Records pertaining to preventive maintenance will be maintained in accordance with the Records Management System.

Results of preventive maintenance activities related to IROFS via the configuration management system will be evaluated by safety disciplines to determine any impact on the ISA and whether updates are needed.

15.3.1.3 Corrective Maintenance

Corrective maintenance involves repair or replacement of equipment that has unexpectedly degraded or failed. Corrective maintenance of IROFS restores the equipment to acceptable performance through a planned, systematic, controlled, and documented approach for the repair and replacement activities.

Following any corrective maintenance on IROFS, and before returning an IROFS to operational status, functional testing of the IROFS, if necessary, is performed to ensure the IROFS performs its intended safety function.

The CAP requires facility personnel to determine the cause of conditions adverse to quality and promptly act to correct these conditions.

Results of corrective maintenance activities related to IROFS via the configuration management program will be evaluated by the safety disciplines to determine any impact on the ISA and whether updates are needed.

15.3.1.4 Functional Tests

A test control program will be implemented that incorporates plant procedures for test control that delineates the criteria for determining when, why, and how tests are required along with other elements of the test control program. Compensatory measures will be applied in accordance with the limiting conditions for operation as provided in the Operating Limits Manual (OLM). See Chapter 5 for a description of the OLM.

The overall testing program is broken into the three major testing programs:

Cold Startup Testing Program

- Component and Qualification Testing

- Preoperational Testing – Functional Testing of IROFS
- Response to General Incident (RGI)
- Reference Period

Hot Startup

- Verification of Parameters
- Verification of Safe Operation with Nuclear Material

Operational Testing Program

- Periodic Testing
- Special Testing.

Results of surveillance/monitoring activities related to IROFS via the configuration management program will be evaluated by the safety disciplines to determine any impact on the ISA and any updates needed.

The objectives of the overall facility Cold Startup, Hot Startup, and Operational Testing Programs are to ensure that items relied on for safety:

- Have been adequately designed and constructed
- Meet licensing requirements
- Do not adversely affect worker or the public health and safety
- Can be operated in a dependable manner so as to perform their intended function.

Additionally, the Cold Startup Testing Program, Hot Startup, and Operational Testing Program ensure that operating, emergency and surveillance procedures are correct and that personnel have acquired the correct level of technical expertise. The facility operating, emergency and surveillance procedures are progressively use-tested throughout the testing programs. The use of operating procedures by the Operations group during Cold Startup serves to familiarize personnel with plant operation during the testing phases and also serves to ensure the adequacy of the procedures under actual or simulated operating conditions.

Cold Startup Testing Program

Cold Startup functional tests are completed prior to introduction of special nuclear material (SNM). Other tests, not required prior to SNM introduction and not related to IROFS, such as office building ventilation tests, may be completed following SNM introduction. Tests (or portions of tests), which are not required to be completed before SNM introduction are identified in the test plan. Cold Startup Testing consists of Component Testing and Qualification Testing for non-IROFS equipment and Preoperational Tests for IROFS equipment. Component Tests and Qualification Tests functionally test equipment to ensure it meets required design functionality but do not take credit for testing IROFS. Component and Qualification Functional Testing at the facility consists of that testing conducted to initially determine various facility parameters and to initially verify the functionality of process equipment and systems. The Preoperational Tests conducted are primarily associated with IROFS (Quality Level 1) and certain Quality Level 2 structures, systems and components. The major objective of Preoperational functional testing is to verify that IROFS essential to the safe operation of the plant are capable of performing their intended function.

Functional testing of IROFS is performed as appropriate following initial testing, as part of periodic surveillance testing, and after corrective or preventive maintenance or calibration to ensure that the item is capable of performing its safety function when required.

The Response to General Incident (RGI) testing consists of a general shutdown of the plant while simulated operations are taking place. In this phase of Cold Startup, the plant will be subjected to an unexpected simulated outage resulting in one or more critical systems being shut down. This will test the facility's safety system reaction to adverse conditions and its ability to recover from a catastrophic event.

The Reference Period phase of Cold Startup consists of a period of time where the facility will simulate operations under normal conditions. During this period, the facility's normal operating procedures will be used to the extent possible without SNM introduced demonstrating the adequacy of procedures, personnel training, qualification and facility design. Prior to the beginning of this period the Startup group will turn over the facility to the Operations group and the Operations group will lead the performance of the Reference Period.

Hot Startup

Hot Startup is performed beginning with the introduction of SNM and ending with the start of operation. The purpose of Hot Startup is to ensure safe and orderly SNM processing and to verify parameters assumed in the ISA.

Records of the Cold Startup tests required prior to operation are maintained. These records include testing schedules and the testing results for IROFS.

The use of properly reviewed and approved test procedures is required for Cold Startup tests. The results of each Cold Startup test are reviewed and approved by the responsible Functional Area Manager or designee before they are used as the basis of continuing the test program. In addition, preoperational tests and test results will be reviewed and approved by the Joint Test Group (JTG) comprised of Operations, Engineering, Quality Assurance, Safety, and Startup groups. Modifications to IROFS that are found to be necessary are subjected to an evaluation per 10 CFR 70.72 prior to making the change.

The impact of modifications on future and completed testing is evaluated during the 10 CFR 70.72 evaluation process and retesting is conducted as required.

The overall Cold Startup functional testing program is reviewed, prior to initial SNM introduction, by the Plant Manager, Functional Area Managers, and JTG to ensure that prerequisite testing is complete.

The facility operating, emergency and surveillance procedures are use-tested throughout the testing program phases to the extent practicable. The trial use of operating procedures serves to familiarize operating personnel with systems and plant operation during the testing phases and also serves to ensure the adequacy of the procedures under actual or simulated operating conditions before plant operation begins.

Procedures which cannot be use-tested during the testing program phase are revised based on initial use-testing, operating experience and comparison with the systems. This ensures that these procedures are as accurate and comprehensive as practicable.

Operational Testing Program

The operational testing program consists of periodic testing and special testing. Periodic testing is conducted at the facility to monitor various facility parameters and to verify the continuing integrity and capability of facility IROFS. Special testing which may be conducted at the facility is testing which does not fall under any other testing program and is of a non-recurring nature.

The Maintenance Manager has overall responsibility for the development and conduct of the operational testing program and in conjunction with the Operations Manager and the Licensing Manager ensures that testing commitments and applicable regulatory requirements are met.

Surveillance commitments, procedures identified to satisfy these commitments and surveillance procedure responsibility assignments for the facility are identified in a computer database. The database is also used to ensure surveillance testing is completed in the required time interval.

Periodic Testing

The periodic testing program at the facility consists of testing conducted on a periodic basis to verify the continuing capability of IROFS to meet performance requirements.

The facility periodic test program verifies that the facility:

- Complies with regulatory and licensing requirements
- Does not endanger health and minimizes danger to life or property
- Is capable of operation in a dependable manner so as to perform its intended function.

The facility periodic testing program begins during the preoperational testing stage and continues throughout the facility's life.

A periodic testing schedule is established to ensure that required testing is performed and properly evaluated on a timely basis. The schedule is revised periodically, as necessary, to reflect changes in the periodic testing requirements and experience gained during plant operation. Testing is scheduled such that the safety of the plant is not dependent on the performance of an IROFS that has not been tested within its specified testing interval. Testing is scheduled consistent with the limiting conditions for operation as identified in the Operating Limits Manual such that the performance requirements of 70.61 continue to be met. Periodic test scheduling is handled through the Maintenance department.

In the event that a test cannot be performed within its required interval due to system or plant conditions, appropriate actions will be taken.

Periodic testing and surveillance associated with Quality Level 1 and 2 structures, systems and components are performed in accordance with written procedures.

Special Testing

Special testing is testing conducted at the facility that is not a facility preoperational test, periodic test, post-modification test, or post-maintenance test. Special testing is of a non-recurring nature and is conducted to determine facility parameters and/or to verify the capability of IROFS to meet performance requirements. Purposes of special testing include, but are not necessarily limited to, the following:

- Acquisition of particular data for special analysis
- Determination of information relating to facility incidents
- Verification that required corrective actions reasonably produce expected results and do not adversely affect the safety of operations
- Confirmation that facility modifications reasonably produce expected results and do not adversely affect systems, equipment and/or personnel by causing them to function outside established design conditions; applicable to testing performed outside of a post-modification test.

The determination that a certain plant activity is a Special Test is intended to exclude those plant activities which are routine surveillances, normal operational evolutions, and activities for which there is previous experience in the conduct and performance of the activity. At the discretion of the Plant Manager, any test may be conducted as a special test.

15.3.2 Measuring and Test Equipment

The MFFF Measuring and Test Equipment / Calibration (M&TE) program is responsible for the calibration and maintenance of active engineered components used as IROFS, including storage of test equipment, control of calibration standards, collection and storage of performance data used in the development of calibration procedures, and repair of active engineered IROFS that fail in service. This program identifies the processes and plans for maintenance and control of calibration instruments and calibrations standards for the facility and provides a description of how instrument maintenance activities will take place. This program identifies the method by which calibration standards are maintained within the environmental conditions needed to assure their accuracy sufficient to appropriately calibrate and maintain components used as IROFS.

15.3.3 Work Control

Maintenance work, as described above, is performed through a coordinated and structured work control process that integrates with ongoing production activities and requirements and is managed by the Maintenance Work Management Group. The purpose of this structure is to minimize challenges to safety requirements, minimize challenges to production requirements, and maximize work efficiency. This work control process includes representation from various organizations, such as radiation protection, safety, operations and others, as necessary, for complete pre-planning of the required work. Coordinated work support functions include such items as work requests, procedures, schedules, radiation work permits, and lockout/tagout requirements.

Should modifications be identified to plant structures, systems, or components, the change will be prepared in accordance with the Configuration Management process. A modification package will be prepared that will contain the description and rationale for the change and the applicable instructions for implementation. Implementation of the modification is done through the work control process for consistency in implementing work activities in the MFFF.

15.3.4 Relationship of Maintenance Elements to Other Management Measures

The maintenance elements, as described above, interface with other management measures, for example:

- Maintenance activities are implemented in accordance with the quality assurance (QA) program described in the MOX Project Quality Assurance Plan (MPQAP).
- Configuration Management, for obtaining the current approved and controlled documents necessary to support the maintenance activity, such as drawings, specifications, and procedures. Training and Qualification to ensure maintenance personnel are trained to perform their assigned tasks.

Audits and assessments are performed to assure that Maintenance activities are conducted in accordance with the written procedures and that the processes reviewed are effective.

Plant Procedures for the applicable operating and maintenance procedures pertinent to support the maintenance activity.

Records Management provides the framework for reviewing, maintenance, approving, handling, identification, retention, and retrieval of Maintenance related quality assurance records

- Incident investigations may identify root causes of failures that are related to the type or frequency of maintenance. The lessons learned from such investigations are factored into the surveillance/monitoring and preventive maintenance programs as appropriate.

15.4 TRAINING AND QUALIFICATION

Training and qualification of plant personnel is essential to the safe and successful design, construction, testing, and operation of the MFFF. This section describes the training program for the operations phase of the facility, including preoperational functional testing and initial startup testing. The training program requirements apply to those plant personnel who perform activities related to IROFS.

The MPQAP provides training and qualification requirements during the design, construction, and operations phases, for QA training of personnel performing Quality levels 1 and 2 work activities; for nondestructive examination, inspection, and test personnel; and for QA auditors.

The principle objective of the MOX training program system is to ensure job proficiency of facility personnel through effective training and qualification. The training program system is designed to accommodate future growth and meet commitments to comply with applicable established regulations and standards. Employees are provided with training to establish the knowledge foundation and on-the-job training to develop work performance skills. Continuing training is provided, as required, to maintain proficiency in these knowledge and skill components, and to provide further employee development.

Qualification is indicated by successful completion of prescribed training, demonstration of the ability to perform assigned tasks and the maintenance of requirements established by regulation. Training is designed, developed and implemented according to a systematic approach. A systematic approach includes a variety of methods to accomplish the analysis, design, development, implementation, and evaluation of training.

15.4.1 Organization and Management of Training

Line managers have the responsibility and authority to manage, supervise and implement training for their personnel. Accountability of the line managers is indicated on the organizational chart shown in Figure 4-1. Training responsibilities for line managers are included in position descriptions. The training organization provides support to line managers by facilitating the planning, direction, development, conduct, evaluation, and control of a systematic performance-based training process that fulfills job-related training needs.

Performance-based training is a function of analyzing, designing, developing, conducting, and evaluating training. Plant procedures establish the requirements for the training of personnel performing activities related to IROFS. Additionally they ensure the training program is conducted in a reliable and consistent manner. Procedures also allow for exceptions from training when justified and properly documented and approved by appropriate management.. The training process incorporates human factor engineering analysis results. The human factors task analysis of the IROFS identified in the Integrated Safety Analysis (ISA) will be incorporated into plant procedures. Personnel training will be developed based on the plant procedures.

Lesson plans or other approved process controlling documents are used for classroom and on-the-job training as required to assure consistent presentation of subject matter. When design changes or plant modifications are implemented, updates of applicable lesson plans are included in the change control process of the configuration management system.

Training programs and training records at the facility are the responsibility of the Training Manager. Training records are maintained to support management information needs associated with personnel training, and qualification. Records are maintained on each employee's qualifications, experience, and training. The employee training file shall include records of general employee training, technical training, and employee development training conducted at the facility. The employee training file shall also contain records of special company sponsored training conducted by others. The training records for each individual, relative to the employee's performance in completing training and qualification activities, are maintained so that they are accurate and retrievable. Training records are retained in accordance with the records management procedures. Training and qualification records are maintained in a learning management system. The data is backed up nightly by the MOX Information Technology organization and copies of the backup tapes are stored in a remote location. Data entry activities are peer reviewed within the Training organization to ensure data is entered accurately.

15.4.2 Analysis and Identification of Functional Areas Requiring Training or Qualification

A needs/job analysis is performed and tasks identified to ensure that appropriate training is provided to those responsible for managing, supervising, performing, and verifying activities related to IROFS. Identification of job hazards are referred to as precautions and limitations in the procedure related to that task. These limits and precautions will be part of the needs/job analysis performed for that task.

The training organization will identify, document, and address areas requiring training for competent and safe job performance. The training organization consults with relevant subject matter experts, as necessary, to develop a list of tasks for which personnel training for specific jobs is appropriate. The list of tasks selected for training is reviewed and compared to the training materials as part of the systematic assessment of training effectiveness. The task list is also updated periodically as necessitated by changes in procedures, processes, plant systems, equipment, or job scope. The task list is matrixed to supporting procedures and training materials.

15.4.3 Position Training Requirements

Minimum training requirements are developed for those positions whose activities are relied on for safety. Initial identification of job-specific training requirements is based on experience from the MFFF reference facilities of MELOX and La Hague, and other United States fuel cycle facilities. Entry-level criteria (e.g., education, technical background, experience, and/or physical fitness requirements) for these positions are contained in position descriptions. Exceptions from training requirements may be granted when justified and documented in accordance with the approved MFFF procedure.

Radiation safety training is commensurate with the employee's duties. Standardized courses are used to the extent practical and are supplemented by facility-specific information. MFFF personnel will receive training commensurate with the requirements of 10 CFR 19.12. MOX Services commits to ALARA principles as outlined in Chapter 9.2.1.

The training program is designed to prepare initial and replacement personnel for safe, reliable and efficient operation of the facility. Appropriate training for personnel of various abilities and experience backgrounds is provided. The level at which an employee initially enters the training program is determined by the employee's past experience, level of ability, and qualifications.

Facility personnel may be trained through participation in prescribed parts of the training program that consists of the following:

- General Employee Training
- Technical Training.

Training is made available to facility personnel to initially develop and maintain minimum qualifications outlined in Chapter 4, Organization and Administration. The objective of the training shall be to ensure safe and efficient operation of the facility and compliance with applicable established regulations and requirements. Training requirements shall be applicable to, but not necessarily restricted to, those personnel within the plant organization who have a direct relationship to the operation, maintenance, testing or other technical aspect of the facility IROFS. Training courses are updated prior to use to reflect plant modifications and changes to procedures when applicable.

15.4.3.1 General Employee Training

Site General Employee Training is required prior to gaining access to the Savannah River Site and the MOX facility. General Employee Training/new hire training encompasses those Quality Assurance, radiation protection, safety, emergency and administrative procedures established by facility management and applicable regulations. People under the supervision of facility management (including subcontractors) must participate in General Employee Training. Temporary maintenance and service personnel receive General Employee Training to the extent necessary to assure safe execution of their duties.

15.4.3.2 Technical Training

Technical training is designed, developed and implemented to assist facility employees in gaining an understanding of applicable fundamentals, procedures, and practices related to IROFS. Also, technical training is used to develop manipulative skills necessary to perform assigned work related to IROFS. Technical training consists of three segments:

- Initial Training
- On-the-Job Training
- Continuing Training.

Initial job training is designed to provide an understanding of the fundamentals, basic principles, and procedures involved in work related to IROFS that an employee is assigned. This training may consist of, but is not limited to, live lectures, taped and filmed lectures, required reading, self-guided study, demonstrations, laboratories and workshops and on-the-job training.

Certain new employees or employees transferred from other sections within the facility may be partially or wholly qualified by reason of previous applicable training or experience. The extent of further training for these employees is determined by applicable regulations, performance in review sessions, comprehensive examinations, or other techniques designed to identify the employee's present level of ability.

Initial job training and qualification programs are developed for operations, maintenance and technical services classifications. Training for each program is grouped into logical blocks or modules and presented in such a manner that specific behavioral objectives are accomplished. Trainee progress is evaluated using written examinations, oral or practical tests.

On-the-job training (OJT) is a systematic method of providing the required job related skills and knowledge for a position. This training is conducted in an environment as close to the work environment as feasible. Applicable tasks and related procedures make up the OJT/qualifications program for each technical area. Technical areas will be derived based on the activities identified in the ISA Summary, job/task analyses and associated procedures. Training is designed to supplement and complement training received through classroom training.

Continuing training courses shall be established when applicable to ensure that personnel remain proficient. Continuing training is any training not provided as initial qualification or basic training that maintains and improves job-related knowledge and skills. Continuing training may consist of periodic exercises, computer or classroom instruction or any other type of training identified appropriate and performed on a frequency needed to maintain proficiency on the job. Once the objectives for Continuing Training have been established, the methods for conducting the training may vary. The method selected must provide clear evidence of objective accomplishment and consistency in delivery.

15.4.4 Basis for and Objectives of Training

Training requirements shall be applicable to, but not necessarily restricted to, those personnel within the plant organization who have a direct relationship to the operation, maintenance, testing or other technical aspect of the facility IROFS. The objective of the training shall be to ensure safe and efficient operation of the facility and compliance with applicable established regulations and requirements.

Learning objectives identify the training content established by needs/job analyses and position-specific requirements. The task list from the needs/job analysis is used to develop action statements that describe the desired post-training performance. Objectives include the knowledge, skills, and abilities the trainee should demonstrate; the conditions under which required actions will take place; and the standards of performance the trainee should achieve upon completion of the training activity.

15.4.5 Organization of Instruction

Lesson plans are developed from learning objectives, which are based on job performance requirements. Lesson plans and other training guides are developed under guidance by the training organization. Lesson plans are reviewed by the training organization and, generally, by the organization responsible for the subject matter. Lesson plans are approved prior to issue or use. Lesson plans are used for classroom training and on-the-job training as required and include standards for evaluating acceptable trainee performance.

15.4.6 Evaluation of Trainee Learning

Trainee mastery of learning objectives is evaluated through observation/demonstration, or oral or written tests. Such evaluations measure the trainee's skills and knowledge of job performance requirements.

15.4.7 Conduct of On-the-Job Training

On-the-job training is used in combination with classroom training for selected activities when appropriate. Lesson plans are used for classroom training and on-the-job training, as required, and include standards for evaluating acceptable trainee performance. On-the-job training is conducted by personnel who are competent in the program standards and methods of conducting the training using well-organized and current performance-based training materials. Completion of on-the-job training is demonstrated by task performance, where feasible and appropriate. When the actual task cannot be performed in the work environment (e.g., conflicting plant operations), a simulation of the task is conducted, with the trainee explaining task actions in consideration of the conditions that would be encountered during actual performance of the task. This simulation ("walk-through") would use references, tools, and equipment appropriate for the actual task, to the extent practical.

15.4.8 Systematic Evaluation of Training Effectiveness

Periodically the training program is systematically evaluated to measure the program's effectiveness in producing competent employees. The trainees are encouraged to provide feedback after completion of classroom training sessions to provide data for this evaluation for program improvements. These evaluations identify program strengths and weaknesses, determine whether the program content matches current job needs, and determine if corrective actions are needed to improve the program's effectiveness. The training organization is responsible for leading the training program evaluations and for implementing any corrective actions. Program evaluations may consist of an overall periodic evaluation, or a series of topical evaluations over a given period.

Evaluation objectives that are applicable to the training program or topical area being reviewed may address the following elements of training:

- Management and administration of training and qualification programs
- Development and qualification of the training staff
- Position training requirements
- Determination of training program content, including its facility change control interface with the configuration management system
- Design and development of training programs, including lesson plans
- Conduct of training
- Trainee examinations and evaluations
- Training program assessments and evaluations.

Evaluation results are documented, and noteworthy practices and weaknesses are highlighted in the training program. Identified deficiencies are reviewed, improvements are recommended, and changes are made to procedures, practices, or training materials, as necessary. Training materials are updated prior to use to reflect plant modifications and changes to procedures when applicable.

Periodically, training and qualifications activities are monitored by designated facility and/or contracted training personnel. The QA organization audits the facility training and qualification system. In addition, trainees and vendors may provide input concerning training program effectiveness. Methods utilized to obtain this information include, among other things surveys, questionnaires, performance appraisals, staff evaluation, and overall training program effectiveness evaluation instruments. Frequently conducted classes are not evaluated each time. However, they are routinely evaluated at a frequency sufficient to determine program effectiveness.

15.4.9 Personnel Qualification

The qualification requirements for technical personnel are determined as discussed in Sections 15.4.2 and 15.4.3. Training and qualification requirements associated with quality-affecting activities are provided in the MPQAP. Such requirements include QA training for project personnel, and qualification of nondestructive examination personnel, inspection and test personnel, personnel performing special processes, and auditors. Qualification requirements for key management positions are provided in Chapter 4.

15.4.10 Provisions for Continuing Assurance

Personnel performing activities relied on for safety are evaluated at least every two years to verify that they continue to understand, recognize the importance of, and have the qualifications to perform their activities that are relied on for safety. The evaluation may be by written test, oral test, or on-the-job performance evaluation. The results of the evaluation are documented. When the results of the evaluation dictate, retraining or other appropriate action is provided. Retraining is also required due to plant modifications, procedure changes, and QA program changes that result in new or changed information.

15.5 PLANT PROCEDURES

This section describes the procedures used for control of overall facility operations, including IROFS. Activities involving special nuclear material (SNM) will be conducted in accordance with approved procedures. This includes procedures for the conduct of all operations involving controls identified in the Integrated Safety Analysis (ISA) as activities relied on for safety and for all management control systems supporting those controls. Management policies require strict adherence to procedures when performing work. In the event that a procedure cannot be executed as written, personnel are required to notify their supervisor. Time-out authority within MOX Services is vested in each MOX Services employee, with respect to work within their scope of responsibility, whenever the health and safety of workers, the public, or the environment is involved, or when continued work will produce results that are not in compliance with the MOX Services QA Program.

Plant procedures are developed and controlled under the requirements of the MPQAP. Specifically, the associated activities are implemented by personnel who are trained in accordance with the requirements of MPQAP Section 2, *Quality Assurance Program*. Plant maintenance, testing, and operating procedures meet the requirements of MPQAP Section 5, *Instructions, Procedures, and Drawings*. Plant procedures are distributed and otherwise controlled in accordance with the requirements of MPQAP Section 6, *Document Control*. When completed, procedure results (e.g., sign-offs, checklists, data sheets) are maintained in the records management system in accordance with the requirements of MPQAP Section 17, *Quality Assurance Records*.

15.5.1 Types of Procedures

Plant procedures are broadly categorized as either administrative procedures or operating procedures. Administrative procedures apply to functions or specific interfaces with other organizational functions. Operating procedures provide specific direction for functional task-based work. Operating procedures can apply MOX Services-wide or to a specific organization.

15.5.1.1 Administrative Procedures

Administrative procedures specify controls that apply to specific functions or specific interfaces with other organizational functions. They address administration and conduct of process activities in the following areas:

- Training and qualification
- Audits and assessments
- Incident investigation
- Records management
- Configuration management
- Human systems interface
- Reporting
- Quality Assurance
- Equipment control (lockout/tagout)
- Shift turnover
- Work control
- Management control
- Procedure management
- Nuclear criticality safety
- Fire protection
- Radiation protection

- Radioactive waste management
- Maintenance
- Environmental protection
- Chemical process safety
- Operations
- Calibration control
- Preventive maintenance
- Design control
- Test control.

15.5.1.2 Operating Procedures

Operating procedures provide specific direction for functional task-based work within an organizational function. Operating procedures include production, maintenance, and emergency procedures that address startup, operation, shutdown, control of process operations, and recovery after a process upset. These procedures address: Ventilation; Criticality alarms; Shift routines, shift turnover, and operating practices; Decontamination operations; Plant utilities (air, other gases, cooling water, firewater, steam); Temporary changes in operating procedures; and Abnormal operation/alarm response including: Loss of cooling water; Loss of instrument air; Loss of electrical power; Loss of criticality alarm system; Loss of containment; Fires; and Chemical process releases. The results of the ISA are used to identify specific IROFS Administrative Controls that are developed.

Operating procedures include operating limits and controls, and specific IROFS Administrative Controls to ensure: nuclear criticality safety, chemical safety, fire protection, emergency planning, and environmental protection. If needed, safety checkpoints (e.g., hold points for radiological or criticality safety checks, QA verifications, independent operator verification) are identified at appropriate steps.

Operating procedures, with different types of documents, are organized to a consistent architecture, which include:

- General rules for production, maintenance, operational safety, security, abnormal operating procedures, emergency planning and emergency operating procedures, and environmental protection program
- Unit Operating Instructions or Maintenance Instructions – Provide instructions for operating and maintaining process units, systems, and/or equipment.

The scope of these procedures is as follows:

- Production procedures – startup, operation, shutdown, off-normal, alarm response, control of process and laboratory operations, and recovery after a process upset condition

- Maintenance procedures – preventive and corrective maintenance, calibration, surveillance, functional testing, and work control
- Emergency procedures – response to a criticality event, a hazardous chemical release, or an emergency external to the MFFF that may affect the MFFF.

15.5.1.2.1 Production Procedures

Production procedures control process operations and apply to utility, workstation, and control room operations.

Production procedures contain the following elements, as applicable:

- Purpose of the activity
- Regulations, policies and guidelines governing the procedure
- Type of procedure
- Steps for each operating process phase
- Initial startup, Periodical startup / shutdown
- Normal operations
- Off-normal operations
- Temporary operations
- Emergency shutdown
- Emergency operations
- Normal shutdown
- Startup following an emergency or extended downtime
- Hazards and safety considerations
- Operating limits
- Precautions necessary to prevent exposure to hazardous chemicals or SNM
- Measures to be taken if contact or exposure occurs
- Safety controls and their functions that are associated with the process
- Specified time period or other limitations on the validity of the procedure.

15.5.1.2.2 Maintenance Procedures

Where appropriate, maintenance procedures include requirements for pre-maintenance activities involving reviews of the work to be performed, work controls, and reviews of procedures. When appropriate, maintenance work may require clearance from the operations organization to begin work, as well as notification when the work and associated post-maintenance functional testing are complete. Maintenance activities will be monitored/assessed in accordance with the MPQAP.

Maintenance of facility structures, systems and components is performed in accordance with written procedures, documented instructions, checklists, or drawings appropriate to the circumstances. Maintenance activities that address repair, calibration, surveillance, and functional testing include: Repairs and preventive repairs of items relied on for safety (IROFS); Testing of criticality alarm units; Calibration of IROFS; High efficiency air particulate (HEPA) filter maintenance; Functional testing of IROFS; Relief valve replacement/testing; Surveillance/monitoring; Pressure vessel testing; Piping integrity testing; and Containment device testing.

The facility's maintenance department under the Maintenance Manager has responsibility for preparation and implementation of maintenance procedures. The maintenance, testing and calibration of facility IROFS is performed in accordance with approved written procedures. Testing conducted on a periodic basis to determine various facility parameters and to verify the continuing capability of IROFS to meet performance requirements is conducted in accordance with approved, written procedures. Periodic test procedures are utilized to perform such testing and are sufficiently detailed that qualified personnel can perform the required functions without direct supervision. Testing performed on IROFS that are not redundant will provide for compensatory measures to be put into place to ensure that the IROFS performs until it is put back into service.

15.5.1.2.3 Emergency Procedures

Emergency procedures address the preplanned actions of operators and other plant personnel in response to an incident, criticality event, hazardous chemical release, or external emergency that may affect MFFF. In addition, applicable procedures will be reviewed after unusual incidents, such as an accident, unexpected transient, significant operator error, or equipment malfunction, or after any modification to a system, and procedures will be revised as needed.

15.5.2 Preparation of Procedures

MFFF procedures are prepared using a consistent format, and are clear, concise and comprehensive in addressing the procedure subject. MFFF procedures are well organized, and may include (approved) checklists or data sheets as documented records of completion. Initial procedure drafts are reviewed by other members of the facility staff and vendors as appropriate for inclusion and correctness of technical information, including formulas, set points, and acceptance criteria. Procedures that are written for the operation of equipment related to IROFS shall be subjected to a peer review. The Functional Area Manager shall determine whether or not any additional, cross-disciplinary review is required and shall approve procedures. Applicable safety limits associated with IROFS are clearly identified in the procedures.

15.5.2.1 Identification and Preparation

The results of the ISA and other processes are used to identify specific operating and administrative procedures that are developed. Plant procedures are prepared by qualified individuals assigned by the organization's management responsible and accountable for the associated operation.

MOX Services will incorporate methodology for identifying, developing, approving, implementing, and controlling operating procedures. Identifying needed procedures will include consideration of ISA results or changes in ISA results. The method will ensure that, as a minimum:

- Operating and safety limits related to IROFS are specified in the procedure
- Procedures include required actions for off-normal conditions of operation, as well as normal operations
- If needed safety checkpoints are identified at appropriate steps in the procedure
- Procedures are validated as prescribed in applicable project procedures
- Procedures are approved by Functional Area Managers responsible and accountable for the operation
- A mechanism is specified for revising and reissuing procedures in a controlled manner
- The QA elements and CM Program at the facility provide reasonable assurance that current procedures are available and used at work locations
- The facility training program trains the required persons in the use of the latest procedures available.

15.5.2.2 Review/Approval

Operating and administrative procedures are reviewed and approved by management responsible and accountable for the associated operation. The functional management may specify a review to be performed by another functional group. Prior to initial use or after major revisions, production and maintenance procedures are verified and validated.

15.5.2.3 Revisions

Procedure revisions, including temporary changes, are prepared and approved in the same manner as the original. The procedure change process shall be defined in a MFFF procedure.

15.5.3 Use of Procedures

Compliance with operating and maintenance procedures is required, and operators and technicians are trained to report inadequate procedures or the inability to follow procedures. Dependent on the nature of the procedure and work location, procedures are either available at work stations, or are readily accessible where needed to perform work.

15.5.4 Control of Procedures

Following approval, plant procedures are processed for entry into the Electronic Data Management System (EDMS) and issued for use. The MFFF training program, addressed in

Section 15.4, ensures that necessary personnel are trained in the use of approved procedures before implementation.

Change control for operating and administrative procedures is the same as for other items in the document management system. Document management procedures ensure that changes to the facility, including procedures, are entered into the EDMS and address control and distribution of changes (including those for emergency conditions, temporary procedure changes, temporary modifications, etc.). The MPQAP provides requirements for QA procedures, which detail the controls for design input, design output, processes, verification, interfaces, changes, approval, and records.

To ensure technical accuracy, radiation protection procedures, respiratory protection procedures, operating, maintenance, and administrative procedures are reviewed every five years to verify their continued applicability and accuracy. Respiratory protection procedures are reviewed as appropriate whenever the MFFF undergoes a modification, change in process or replacement of equipment. Emergency procedures are reviewed annually for the first two years of MFFF operation and at least every two years thereafter. These periodic reviews are performed by qualified individuals assigned by the functional management responsible and accountable for the associated operation. Reissue/approval of a procedure meets the requirements for procedure periodic review. Additionally, if procedural inadequacy is identified as a root cause from an incident investigation, applicable procedures are reviewed and modified, as necessary.

15.6 AUDITS AND ASSESSMENTS

MOX Services maintains the program for audits and assessments described in the MPQAP, Section 18, *Audits*. MPQAP changes are reviewed to ensure that audit and assessment program revisions are reflected in the program description. MOX Services will have a tiered approach to verifying compliance to procedures and performance to regulatory requirements. Audits are focused on verifying compliance with regulatory and procedural requirements (including compliance with selected operating limits) and licensing commitments. Assessments are focused on effectiveness of activities and ensuring that IROFS, and items that affect the function of IROFS, are reliable and are available to perform their intended safety functions. This approach includes performing Assessments and Audits on critical work activities associated with facility safety, results of the ISA, environmental protection and other areas as identified via trends.

Assessments are divided into two categories that will be owned and managed by the line organizations as follows:

- Management Assessments conducted by the line organizations responsible for the work activity
- Independent Assessments conducted by individuals not involved in the area being assessed.

Audits of the Quality Level 1 work activities and items required to satisfy regulatory requirements for which Quality Level 1 requirements are applied will be the responsibility of the QA Department.

Audits and assessments are performed to assure that facility activities are conducted in accordance with the written procedures and that the processes reviewed are effective. As a minimum, they shall assess activities related to radiation protection, criticality safety control, hazardous chemical safety, industrial safety including fire protection, and environmental protection. Technical and programmatic audits and assessments are performed internally and externally to provide a comprehensive independent verification and evaluation of procedures and activities for IROFS.

Audits and assessments shall be performed routinely by qualified staff personnel that are not directly responsible for production activities. Deficiencies identified during the audit or assessment requiring corrective action shall be forwarded to the responsible manager of the applicable area or function for action in accordance with the CAP procedure. The audit and assessment program provides for on-the-spot corrective actions with appropriate documentation in accordance with the CAP procedure. Future audits and assessments shall include a review to evaluate if corrective actions have been effective.

The Quality Assurance Department shall be responsible for audits. Audits shall be performed in accordance with a written plan that identifies and schedules audits to be performed. Audit team members shall not have direct responsibility for the function and area being audited. Team members shall have technical expertise or experience in the area being audited and shall be indoctrinated in audit techniques. Audits shall be conducted on an annual basis.

The results of the audits shall be provided in a written report in a timely manner to the Plant Manager and the Managers responsible for the activities audited. Any deficiencies noted in the audits shall be responded to promptly by the responsible Managers or designees, entered into the CAP and tracked to completion and re-examined during future audits to ensure corrective action has been completed.

Records of the instructions and procedures, persons conducting the audits or assessments, and identified violations of license conditions and corrective actions taken shall be maintained.

15.6.1 Activities to be Audited or Assessed

Audits and assessments are conducted for the areas of:

- Radiation safety
- Nuclear criticality safety
- Chemical safety
- Other ISA safety areas
- Industrial safety including fire protection
- Environmental protection
- Emergency management
- QA

- Configuration management
- Maintenance
- Training and qualification
- Procedures CAP/Incident investigation
- Records management.

Assessments of nuclear criticality safety, performed in accordance with ANSI/ANS-8.19, will ensure that operations conform to criticality requirements.

15.6.2 Scheduling of Audits and Assessments

A schedule is established that identifies audits and assessments to be performed and the responsible organization assigned to conduct the activity. The frequency of audits and assessments is based upon the status and safety importance of the activities being performed and upon work history. Major activities will be audited or assessed on an annual basis. The audit and assessment schedule is reviewed periodically and revised as necessary to ensure coverage commensurate with current and planned activities.

Nuclear Criticality safety audits are conducted and documented such that aspects of the Nuclear Criticality Safety Program will be audited at least every two years. The Operations Group is assessed periodically to ensure that nuclear critical safety procedures are being followed and the process conditions have not been altered to adversely affect nuclear criticality safety. The frequency of these assessments is based on the controls identified in the NCS analyses and NCS evaluations. Assessments are conducted annually.

15.6.3 Procedures for Audits and Assessments

Internal and external audits and assessments are conducted using approved procedures that meet the QA Program requirements. These procedures provide requirements for the following audit and assessment activities:

- Scheduling and planning of the audit and assessment
- Certification requirements of audit personnel
- Development of audit plans and audit and assessment checklists as applicable
- Performance of the audit and assessment
- Reporting and tracking of findings to closure
- Closure of the audit and assessment.

The applicable procedures emphasize reporting and correction of findings to prevent recurrence.

Audits and assessments are conducted by:

- Using the approved audit and assessment checklists as applicable

- Interviewing responsible personnel
- Performing plant area walkdowns (including accessible out-of-the way and limited access areas)
- Reviewing controlling plans and procedures
- Observing work in progress
- Reviewing completed QA documentation.

Audit and assessment results are tracked in the CAP. The data is periodically analyzed for potential trends and needed program improvements to prevent recurrence and/or for continuous program improvements. The resulting trend is evaluated and reported to applicable management. This report documents the effectiveness of management measures in controlling activities, as well as deficiencies. Deficiencies identified in the trend report require corrective action in accordance with the applicable CAP procedure. The QA organization also performs follow-up reviews on identified significant deficiencies and verifies completion of corrective actions reported as a result of the trend analysis.

The audit and /or assessment team leader is required to develop the audit and /or assessment report documenting the findings, observations, and recommendations for program improvement. These reports provide management with documented verification of performance against established performance criteria for IROFS. These reports are developed, reviewed, approved, and issued following established formats and protocols detailed in the applicable procedures. Responsible managers are required to review the reports and provide any required responses due to reported findings.

Corrective actions following issuance of the audit and/or assessment report require compliance with the CAP procedure. Audit reports are required to contain an effectiveness evaluation and statement for each of the applicable QA program elements reviewed during the audit. The audit/assessment is closed with the proper documentation as required by the applicable audit and assessment procedure. The QA organization will conduct follow-up audits or assessments to verify that corrective actions were taken in a timely manner. In addition, future assessments will include a review to evaluate if corrective actions have been effective.

15.6.4 Qualifications and Responsibilities for Audits and Assessments

The QA Manager initiates audits. The responsible Lead Auditor and QA Manager determine the scope of each audit. The QA Manager may initiate special audits or expand the scope of audits. The Lead Auditor directs the audit team in developing checklists, instructions, or plans and performing the audit. The audit shall be conducted in accordance with the checklists, but the scope may be expanded by the audit team during the audit. The audit team consists of one or more auditors.

Auditors and lead auditors are responsible for performing audits in accordance with the applicable QA procedures. Auditors and lead auditors hold certifications as required by the QA Program. Before being certified under the MFFF QA Program, auditors must complete training on the following topics:

- MFFF QA Program
- Audit fundamentals, including audit scheduling, planning, performance, reporting, and follow-up action involved in conducting audits
- Objectives and techniques of performing audits
- On-the-job training.

Certification of auditors and lead auditors is based on the QA Manager's evaluation of education, experience, professional qualifications, leadership, sound judgment, maturity, analytical ability, tenacity, and past performance and completion of QA training courses. A lead auditor must also have participated in a minimum of five QA audits or audit equivalent within a period of time not to exceed three years prior to the date of certification. Audit equivalents include assessments, pre-award evaluations or comprehensive surveillances (provided the prospective lead auditor took part in the planning, checklist development, performance, and reporting of the audit equivalent activities). One audit must be a nuclear-related QA audit or audit equivalent within the year prior to certification.

Personnel performing assessments do not require certification, but they are required to complete QA orientation training, as well as training on the assessment process. The nuclear criticality safety assessments are performed under the direction of the criticality safety staff. Personnel performing these assessments do not report to the production organization and have no direct responsibility for the function or area being assessed.

15.7 INCIDENT INVESTIGATIONS

MOX Services implements two programs for investigating discrepancies: the Corrective Action Process and Incident Investigations. This section describes these programs.

15.7.1 Corrective Action Process

The MFFF Corrective Action Process is used for identifying, investigating, reporting, tracking, correcting, and preventing recurrence of conditions that are adverse to quality or that may affect radiation protection, safety, quality, regulatory compliance, reliability, human performance or project performance. The corrective action process is performed in accordance with MPQAP Section 16, “Corrective Action”. Nonconforming materials, parts, or components are identified and controlled in accordance with MPQAP Section 15, *Nonconforming Materials, Parts, or Components*. The MPQAP requires regularly scheduled audits and assessments to ensure that needed corrective actions are identified. MOX Services employees have the authority and responsibility to initiate the corrective action process if they discover deficiencies. Reports of conditions adverse to quality are analyzed to identify trends in quality performance. Significant conditions adverse to quality and significant trends are reported to senior management in accordance with corrective action process procedures.

15.7.2 Incident Investigations

Incident investigations are used for investigating abnormal events, other than those that involve conditions adverse to quality identified in Section 15.7.1. Identification of the need for an incident investigation may come from anyone in the MFFF organization. An incident investigation is performed by one or more individuals assigned by the manager of production. The process used for the investigation may be similar to that of the CAP. Each event will be considered in terms of its requirements for reporting in accordance with regulations and will be evaluated to determine the level of investigation required. The process of incident identification, investigation, root cause analysis, environmental protection analysis, recording, reporting, and follow-up shall be addressed in and performed by written procedures. Radiological, criticality, hazardous chemical, other ISA related safety requirements shall be addressed. Guidance for classifying occurrences shall be contained in procedures, including examples of threshold off-normal occurrences. The depth of the investigation will depend upon the severity of the classified incident in terms of the levels of special nuclear material released and/or the degree of potential for exposure of workers, the public or the environment.

MOX Services shall maintain a record of corrective actions to be implemented as a result of off-normal occurrence investigations in accordance with CAP procedures. These corrective actions shall include documenting lessons learned, and implementing worker training where indicated, and shall be tracked to completion.

Specifics of the Incident Investigation process are as follows:

1. MOX Services will establish a process to investigate abnormal events that may occur during operation of the facility, to determine their specific or generic root cause(s)and generic implications, to recommend corrective actions, and to report to the NRC as required by 10 CFR 70.50 and 70.74. The investigation process will include a prompt risk-based evaluation and, depending on the complexity and severity of the event, one individual may suffice to conduct the evaluation. The investigator(s) will be independent from the line function(s) involved with the incident under investigation and are assured of no retaliation for participating in investigations. Investigations will begin within 48 hours of the abnormal event, or sooner, depending on safety significance of the event. The record of IROFS failures required by 10 CFR 70.62(a)(3) for IROFS will be reviewed as part of the investigation. Record revisions necessitated by post-failure investigation conclusions will be made following completion of the investigation.
2. Qualified internal or external staff are appointed to serve on investigating teams when required. The teams will include at least one process expert and at least one team member trained in root cause analysis.
3. MOX Services will monitor and document corrective actions through completion.
4. MOX Services will maintain auditable records and documentation related to abnormal events, investigations, and root cause analyses so that "lessons learned" may be applied to future operations of the facility. For each abnormal event, the incident report includes a description, contributing factors, a root cause analysis, findings, and recommendations. Relevant findings are reviewed with affected personnel. Details of the event sequence will be compared with accident sequences already considered in the ISA. As appropriate, the ISA and the ISA Summary will be modified to include evaluation of the risk associated with accidents of the type actually experienced.

MOX Services will develop CAP procedures for conducting an incident investigation, and the procedures will contain the following elements:

1. A documented plan for investigating an abnormal event.
2. A description of the functions, qualifications, and/or responsibilities of the manager who would lead the investigative team and those of the other team members; the scope of the team's authority and responsibilities; and assurance of cooperation of management.
3. Assurance of the team's authority to obtain the information considered necessary and its independence from responsibility for or to the functional area involved in the incident under investigation.
4. Retention of documentation relating to abnormal events for two years or for the life of the operation, whichever is longer.
5. Guidance for personnel conducting the investigation on how to apply a reasonable, systematic, structured approach to determine the specific or generic root cause(s) and generic implications of the problem.
6. Requirements to make available original investigation reports to the NRC on request.
7. A system for monitoring the completion of appropriate corrective action and that actions are completed in a timely manner.

15.8 RECORDS MANAGEMENT

Records management shall be performed in a controlled and systematic manner in order to provide identifiable and retrievable documentation during design, construction and operation of the MFFF. Records management procedures establish the requirements and responsibilities for record selection, verification, protection, transmittal, distribution, retention, maintenance, and disposition. Procedures have been established that promptly detect and correct deficiencies in the records management system or its implementation.

The MPQAP requires procedures for reviewing, approving, handling, identifying, retention, retrieval and maintenance of quality assurance records. These records include the results of tests and inspections required by applicable codes and standards, construction, procurement and receiving records, personnel certification records, design calculations, purchase orders, specifications, procedures, corrective action records, source surveillance and audit reports, and any other QA documentation required by specifications or procedures. These records are maintained at locations where they can be reviewed and audited to establish that the required quality has been assured. Applicable design specifications, procurement documents, or other documents specify the QA records to be generated by, supplied to, or held, in accordance with approved procedures. QA procedures are not considered valid until they are authenticated by authorized personnel.

Classified records are managed in accordance with an approved project procedure which identifies both the physical protection and access control measures for classified records. A limited area has been established as a satellite records retention facility in accordance with the records management procedure.

For computer codes and electronic data used for IROFS, procedures are established for the control and management of computer codes over the life cycle of the facility. The Records Center maintains control over access and use of records entered into the EDMS. Documents in EDMS shall be legible and shall be identifiable as to the subject to which they pertain. Documents shall be considered valid only if stamped, initialed, signed or otherwise authenticated by authorized personnel. Documents in EDMS may be originals or reproduced copies. Computer storage of data may be used in EDMS.

In order to preclude deterioration of records in EDMS, the following requirements are applicable:

- Provisions shall be made in the storage arrangement to prevent damage from moisture, temperature and pressure.
- For hardcopy records, approved filing methods shall require records to be:
  - Firmly attached in binders, placed in folders, or placed in envelopes for storage in steel file cabinets; or
  - In containers appropriate for the record medium being stored on shelving.
- The storage arrangement shall provide adequate protection of special processed records (e.g., radiographs, photographs, negatives, microform and magnetic media) to prevent damage from moisture, temperature, excessive light, electromagnetic fields or stacking, consistent with the type of record being stored.

The EDMS shall provide for the accurate retrieval of information without undue delay. Records shall be stored and preserved in the Records Center in accordance with an approved QA procedure that provides:

- A description of the storage facility;
- A description of the filing system to be used;
- A method for verifying that the records received are in agreement with the transmittal document;
- A method for verifying that the records are those designated and the records are legible and complete;
- A description of rules governing control of the records, including access, retrieval and removal;
- A method for maintaining control of and accountability for records removed from the storage facility;
- A method for filing supplemental information and disposition of superseded records;
- A method for precluding entry of unauthorized personnel into the storage area to guard against larceny and vandalism; and

- A method for providing for replacement, restoration or substitution of lost or damaged records.

One-of-a-kind records shall be stored in 2 hour fire rated cabinets to assure records are adequately protected from damage.

Records related to environment, safety and health, including radiological protection, shall be maintained in accordance with the records management procedural requirements. Records shall be retained for at least the periods indicated in accordance with the records management procedures that specify retention periods.

The following are examples of records that will be retained:

- Operating logs
- Procedures
- Supplier QA documentation for equipment, materials, etc.
- Nonconforming item reports
- Test documentation/test results – preoperational/operational
- Facility modification records
- Drawings/specifications
- Procurement documents (e.g., purchase orders)
- Nuclear material control and accounting records
- Maintenance activities including calibration records
- Inspection documentation (plant processes)
- Audit reports
- Reportable occurrences and compliance records
- Completed work orders
- License conditions records
- Software verification records
- System description documents
- Dosimetry records
- Effluent records
- As-built design documentation packages
- Regulatory reports and corrective action

Other retention times are specified for other facility records as necessary to meet applicable regulatory requirements. These retention times are indicated in facility administrative procedures.

Section 17, "*Quality Assurance Records,*" of the MPQAP provides additional details regarding records management requirements.

15.9 DESIGN BASIS

This chapter describes the management measures implemented for MFFF IROFS. These management measures are implemented in accordance with a MOX QA program established in accordance with Title 10 of the CFR Part 50, Appendix B. The MPQAP describes the quality assurance requirements and management measures to control quality-affecting activities on the project and coincide with the 18 criteria of 10 CFR 50, Appendix B. The MPQAP has previously been approved by the NRC. A change that would reduce the commitments of the NRC approved QA program is submitted with written justification to the NRC for acceptance, prior to implementation by MOX Services. The application of management measures to IROFS is further described in Chapter 5.

Table 15-1. Deleted

16.0 AUTHORIZATIONS AND EXEMPTIONS

16.1 EXEMPTIONS

16.1.1 Decommissioning

U.S. Department of Energy (DOE) will assume responsibility for decommissioning the Mixed Oxide (MOX) Fuel Fabrication Facility (MFFF) as discussed in SECY 99-177, "Current Status of Legislative Issues Related to U.S. Nuclear Regulatory Commission (NRC) Licensing a Mixed Oxide Fuel Fabrication Facility," Issue 8. MOX Services requested an exemption from decommissioning requirements. Based on review of the exemption request, the NRC concluded that the requested exemption is authorized by law based on the agreement for DOE to assume responsibility for decommissioning.

Therefore, the method of financial assurance is in accordance with 10 CFR §70.25(f)(5) and 10 Code of Federal Regulations (CFR) §40.36(e)(5).

16.1.2 Financial Protection

SECY 99-177, Issue 7, addresses the issue of Price-Anderson liability coverage. DOE has agreed to indemnify MOX Services in accordance with Section 170(d) of the Atomic Energy Act of 1954, as amended, 42 U.S.C. 2210(d), and Department of Energy Acquisition Regulation (DEAR) 952.250-70 (48 CFR §952.250-70). Because the DOE indemnity will apply to the MFFF, there is no need for the application of the NRC financial protection requirements. Based upon the DOE Indemnity Agreement, MOX Services requested and received an exemption from the NRC's requirements concerning agreements of indemnification and related financial protection requirements. This exemption removed potential ambiguity as to whether indemnification resided with DOE or NRC. The DOE indemnity agreement with MOX Services provides full protection and coverage for public liability arising from operation of the MFFF.

16.1.3 Labeling

MOX Services requested an exemption from the labeling requirements of 10 CFR 20.1904(a) because of the nature of the MFFF operations. In lieu of labeling each container, MOX Services will post in areas that house or temporarily store radioactive material, signs that incorporate the radiation symbol with the warning "CAUTION RADIOACTIVE MATERIAL; ANY CONTAINER IN THIS AREA MAY CONTAIN RADIOACTIVE MATERIAL". Based on review of the exemption request, the NRC concluded that the requested exemption is authorized by law because of the impracticality of labeling each container at the MFFF and because of the appropriate training of personnel in radiation protection requirements.

16.2 AUTHORIZATIONS

16.2.1 Prior Commitments

The License Application reflects the current commitments by MOX Services to meet the regulatory requirements for a Part 70 license. Previous commitments are included in the License Application, as appropriate.

16.2.2 Frequencies

When measurement, surveillance, and/or other frequencies are specified in this License Application or other license commitments, the following shall apply:

- DAILY means once each 30-hour, or less, period.
- WEEKLY means once each eight, or less, consecutive days.
- MONTHLY means 12 per year, with each covering a span of 40 days or less.
- SEMIMONTHLY means twice a month, each covering a span of 20 days or less.
- BIMONTHLY means every 2 months, with each covering a span of 70 days or less.
- QUARTERLY means four per year, with each covering a span of 115 days or less.
- SEMIANNUAL (or BIANNUAL) means two per year, with each covering a span of 225 days or less.
- ANNUAL means once per year, not to exceed a span of 15 months.
- BIENNIAL means once every two years, with each covering a span of 30 months or less.
- TRIENNIAL means once every three years, with each covering a span of 45 months or less.

16.2.3 Changes to the License Application

MOX Services maintains the License Application (LA) and licensing basis so that it is accurate and up-to-date by means of the MFFF configuration management process, which is implemented via project procedures. MOX Services evaluates changes to the facility to determine if prior NRC approval is required and to maintain the accuracy of the LA. The configuration management process for maintaining the LA is implemented on a graded approach. With respect to changes, “graded approach” includes the level and type of documentation required (i.e., initial broad screening may be simple checklist approach versus licensing evaluation where there is an impact to the LA).

16.2.3.1 Changes Prior to Receipt of License to Possess and Use

An initial screening of IROFS related documents is performed for consistency with the LA at document issuance. Changes to key LA basis documents (proceduralized list of documents) are evaluated before the change is implemented in accordance with the project procedure that maintains the project licensing basis. The evaluation of the change determines whether NRC approval of an LA revision is required prior to implementation. Responsibility for maintaining and updating the LA belongs to the Licensing Manager.

MOX Services may make changes that impact the LA without prior NRC approval, if the change:

- Maintains the effectiveness of the design basis as described in the LA (e.g., does not impact compliance with 10CFR70.61 performance requirements);
- Does not result in a departure from a method of evaluation described in the LA used in establishing the design bases;
- Does not adversely affect compliance with applicable regulatory requirements (e.g., 10CFR20); and
- Is not otherwise prohibited by a Construction Authorization (CA) condition or order.

If a change is made, the relevant affected onsite documentation (e.g., required to support construction) will be updated promptly per written procedures. MOX Services maintains records of changes to the key LA basis documents. These records include a written evaluation that provides the bases for the determination that the changes do not require prior NRC approval. These records are maintained until termination of the license. Changes are communicated to the NRC as follows:

- For changes that require NRC pre-approval, MOX Services submits a revision to the LA to NRC.
- For changes that do not require NRC pre-approval, MOX Services submits to the NRC annually, within 30 days after the end of the calendar year during which the changes occurred, a brief summary of the changes.
- For changes that affect the LA, MOX Services submits to the NRC annually, within 30 days after the end of the calendar year during which the changes occurred, revised LA pages.

16.2.3.2 Changes After Receipt of License to Possess and Use

A change to the facility or its processes is evaluated before the change is implemented. The evaluation of the change determines whether a license amendment is required to be submitted in accordance with 10 CFR 70.34 (i.e., requires NRC approval prior to implementation). Responsibility for maintaining and updating the LA belongs to the manager of the support services functions, as described in Chapter 4.

The site, structures, processes, systems, equipment, components, computer programs, and activities of personnel are described in the LA. MOX Services may make changes to these items, as described in the LA, without prior NRC approval, if the change:

- Maintains the effectiveness of the design basis as described in the LA (e.g., does not impact compliance with 10CFR70.61 performance requirements);

- Does not result in a departure from a method of evaluation described in the LA used in establishing the design bases;
- Does not adversely affect compliance with applicable regulatory requirements (e.g., 10CFR20); and
- Is not otherwise prohibited by a license condition or order.

If a change to the LA is made, the relevant affected onsite documentation (e.g., IROFS testing procedure) will be updated promptly per written procedures. MOX Services maintains records of changes to its facility. These records include a written evaluation that provides the bases for the determination that the changes to the LA do not require prior approval. These records are maintained until termination of the license. Changes are communicated to the NRC as follows:

- For changes that require NRC pre-approval, MOX Services submits an amendment request to the NRC in accordance with 10 CFR 70.34 and 70.65.
- For changes that do not require NRC pre-approval of the LA, MOX Services submits to the NRC annually, within 30 days after the end of the calendar year during which the changes occurred, a brief summary of the changes.
- For changes that affect the LA, MOX Services submits to the NRC annually, within 30 days after the end of the calendar year during which the changes occurred, revised LA pages.

16.2.3.3 Code Deviations

MOX Services evaluates code deviations for LA impact consistent with the LA change process described above. If an identified code deviation is more than a limited exception, the deviation will be summarized in the LA; otherwise, the LA deviation is documented in *MFFF Deviation Log* (DCS01-AAJ-DS-ECA-D-40124) in accordance with project procedures. For purposes of maintaining the *MFFF Deviations Log*, MOX Services has defined deviation as:

> *A code "deviation" with respect to the LA exists when the resolution of a nonconformance is not consistent with LA design basis industry codes or standards. For the code "deviation" to be acceptable, it must be evaluated and documented in accordance with project procedures. The evaluation should include justification that the IROFS will perform its IROFS function with the implementation of the resolution (i.e., does not impact compliance with 10CFR70.61 performance requirements). Note, with respect to the LA, a code deviation does not exist if additional analysis is performed that demonstrates the IROFS will perform its IROFS function without crediting the feature associated with the nonconformance.*

If an NFPA Code/Standard cannot be met and an alternate method that provides an equivalent level of safety cannot be identified (i.e., performance-based design and/or documented analysis), a formal request to approve that exemption (deviation) from the NFPA Code/Standard shall be submitted to the NRC for review and approval.

16.2.4 Changes to the Integrated Safety Analysis Summary

MOX Services maintains the Integrated Safety Analysis Summary (ISAS) in accordance with 10CFR70.72 so that it is accurate and up-to-date by means of the MFFF configuration management process which is implemented in project procedures. MOX Services evaluates changes to the facility and its processes for impact on the ISAS, and updates the ISAS, as needed, in order to ensure its continued accuracy. The 10CFR70.72 configuration management process for the ISAS is implemented on a graded approach similar to the CM process for the LA. In addition, while the requirements of 10CFR70.72 in general are applicable, there are areas where the implementation is not applicable during construction and testing as these areas are still in development. An example is operating procedures. MOX Services will implement 10CFR70.72 on operating procedures after the procedures are released for use with licensed materials. An additional example is a 10CFR70.72 evaluation of temporary modifications. A 10CFR70.72 screening/evaluation is only applicable for temporary modifications that will be in place only when MOX is in actual receipt of licensed material. The 10CFR70.72 applicability is documented in project procedures.

16.2.4.1 Changes Prior to Receipt of License to Possess and Use

Changes to project documents that potentially impact the ISAS are evaluated before the change is implemented. The documents to be evaluated for changes are the key basis documents (e.g., Process Safety Information, Integrated Safety Analysis) for the ISAS such that a change to these documents could potentially result in a change to the ISAS. In addition, changes to management measures requirements are evaluated for impact. During construction and testing, the key management measures requirements are documented in the MPQAP. The list of these documents is maintained in the project procedure controlling the licensing basis configuration management. The evaluation of the change determines whether NRC approval is required prior to implementation (e.g., construction) of the associated ISAS revision. Responsibility for maintaining and updating the ISAS belongs to the Licensing Manager.

MOX Services may make changes that impact the ISAS without prior NRC approval, if the change:

- Does not create new types of accident sequences that, unless mitigated or prevented, would exceed the performance requirements of 10 CFR 70.61, and that have not previously been described in the ISA Summary;
- Does not use new processes, technologies, or control systems for which MOX Services has no prior experience;
- Does not remove, without at least an equivalent replacement of the safety function, an IROFS that is listed in the ISA Summary and is necessary for compliance with the performance requirements of 70.61;
- Does not alter an IROFS listed in the ISA Summary, that is the sole item preventing or mitigating an accident sequence that exceeds the performance requirements of 10 CFR 70.61; and

- Is not otherwise prohibited by this section [10CFR70.72], Construction Authorization condition, or order.

If a change is made, the relevant affected onsite documentation (e.g., required to support construction) will be updated promptly per written procedures. MOX Services maintains records of changes to the key ISAS basis documents. These records include a written evaluation that provides the bases for the determination that the changes do not require prior NRC approval. These records are maintained until termination of the license. Changes are communicated to the NRC as follows:

- For changes that require NRC pre-approval, MOX Services submits a revision to the ISAS to NRC.
- For changes that do not require NRC pre-approval, MOX Services submits to the NRC annually, within 30 days after the end of the calendar year during which the changes occurred, a brief summary of the changes.
- For changes that affect ISAS, MOX Services submits to the NRC annually, within 30 days after the end of the calendar year during which the changes occurred, revised LA pages.

16.2.4.2 Changes After Receipt of License to Possess and Use

The change process for the ISAS after receipt of a license is described in LA Section 5.1.4.

16.2.5 Changes to MPQAP

MOX Services may make changes to the MPQAP without NRC approval as described in the MPQAP. Changes are communicated to the NRC as follows:

- For changes that require NRC pre-approval, MOX Services submits an amendment to the MPQAP to the NRC;

- For changes that do not require NRC pre-approval, MOX Services submits annually, within 30 days after the end of the calendar year during which the changes occurred, a brief summary of the changes

- For changes that affect the MPQAP, MOX Services submits to the NRC annually, within 30 days after the end of the calendar year during which the changes occurred, revised MPQAP pages.

16.3 DESIGN BASIS

The following information represents the design basis attributes for authorizations and exemptions.

- MOX Services maintains the LA and licensing basis so that it is accurate and up-to-date by means of the MFFF configuration management process, which is implemented via project procedures.

- MOX Services maintains the ISAS in accordance with 10 CFR 70.72 so that it is accurate and up-to-date by means of the MFFF configuration management process which is implemented in project procedures.
- For changes that require NRC pre-approval, MOX Services submits a revision or amendment, as applicable, to the LA or ISAS, to NRC.
- For changes that do not require NRC pre-approval, MOX Services submits to the NRC annually, within 30 days after the end of the calendar year during which the changes occurred, a brief summary of the changes.
- For changes that affect the LA and/or ISAS, MOX Services submits to the NRC annually, within 30 days after the end of the calendar year during which the changes occurred, revised LA and/or ISAS pages.
- If an identified code deviation is more than a limited exception, the deviation will be summarized in the LA; otherwise, the LA deviation is documented in *MFFF Deviation Log* in accordance with project procedures.
- If an NFPA Code/Standard cannot be met and an alternate method that provides an equivalent level of safety cannot be identified (i.e., performance-based design and/or documented analysis), a formal request to approve that exemption (deviation) from the NFPA Code/Standard shall be submitted to the NRC for review and approval.
- For changes to the MPQAP that require NRC pre-approval, MOX Services submits an amendment to the MPQAP to NRC.
- For changes to the MPQAP that do not require NRC pre-approval, MOX Services submits to the NRC annually, within 30 days after the end of the calendar year during which the changes occurred, a brief summary of the changes.
- For changes that affect the MPQAP, MOX Services submits to the NRC annually, within 30 days after the end of the calendar year during which the changes occurred, revised MPQAP pages.

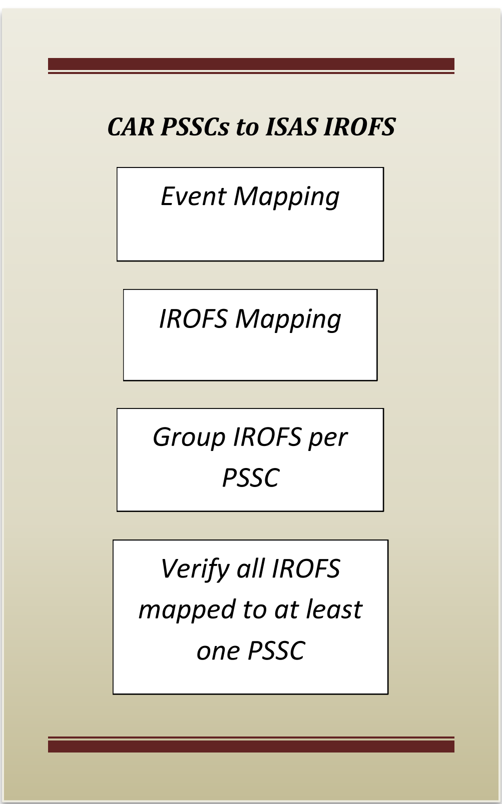
17.0 PSSC Completion

17.1. Purpose

The purpose of Chapter 17 is to describe the process that will provide the basis to notify NRC that MOX Services has completed construction of a Principal Structure, System or Component (PSSC) in accordance with the application as per 10 CFR 70.23(a)(8). The following sections describe the relationship between the PSSCs identified in the Construction Authorization Request (CAR) and the IROFS that are identified in the ISAS. Completing construction of a PSSC is a step in the process as MOX Services prepares to possess and use special nuclear material for operations. For simplicity, MOX Services will only refer to this phase as "operations". In order to operate, MOX Services must also complete any remaining construction punchlist items, demonstrate that IROFS will perform their credited safety function (also referred to as preoperational testing), and develop the required procedures and programs as well as a trained and qualified staff (also referred to as operational readiness).

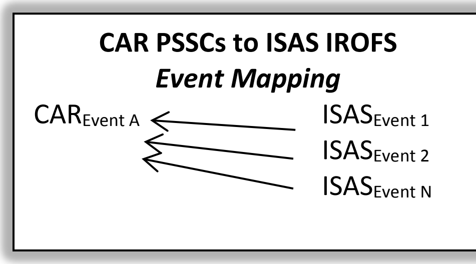
17.2. CAR PSSCs and ISAS IROFS Relationship

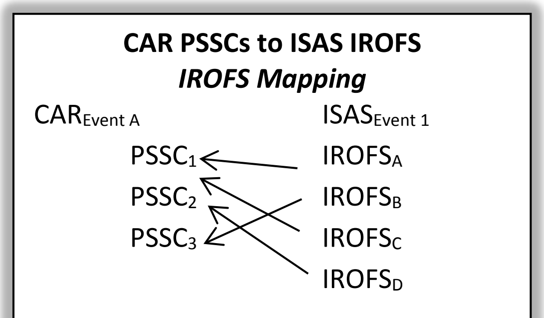
10CFR70.23(a)(8) requires the NRC to verify construction of PSSCs prior to issuance of a license. PSSCs are effectively preliminary Items Relied on for Safety (IROFS) that were identified in the CAR as part of the safety assessment of the design basis. PSSCs are items identified in the CAR that are required to provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents. These PSSCs were identified at a level sufficient to support the NRC review for compliance with 10CFR70.22 and subsequent issuance of a Construction Authorization (CA). CAR Table 5.6-1, MFFF Principal SSCs summarizes the PSSCs that supported the NRC issuance of CAMOX-001 in March 2005.



In order to receive a license to possess and use Special Nuclear Material (SNM), an Integrated Safety Analysis (ISA) is required by 10CFR70.62. MOX Services provided a summary of this ISA to the NRC in 2006 in its Integrated Safety Analysis Summary (ISAS), which was submitted along with the License Application (LA). The NRC completed its review of the LA, ISAS, and other licensing submittals in 2010 – as documented with the issuance of the Final Safety Evaluation Report (FSER) in December 2010. As noted in the FSER, the NRC staff will not issue a license to possess and use SNM before a determination that construction of the PSSCs is in accordance with the application. 10 CFR 70.23(a)(8) refers to "approved pursuant to paragraph (b)" and paragraph (b) cites the "design bases of principal structures, systems and components … provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents" as one of the requirements that must be met prior to the NRC issuing a construction authorization. MOX Services has paraphrased 10 CFR 70.23(a)(8) PSSC verification to "verification that construction of PSSCs is in accordance with the design bases in the Construction Authorization Request as refined in the License Application IROFS design basis sections" – which is consistent with the Construction Authorization (CAMOX-001) issued by the NRC.

Since the ISA is a more detailed evaluation, events were evaluated and the associated IROFS were identified with more specificity than in the CAR phase. With the more detailed evaluation, some events evolved into multiple events, some were determined to be not credible, while others were combined to facilitate evaluation. This usually resulted in a more detailed identification of the IROFS for the events – which significantly increased the number of IROFS described in the ISAS from the 53 PSSCs identified in the CAR. Since the 10CFR70.23(a)(8) specifically requires NRC verification of construction of PSSCs (and not IROFS), a PSSC to ISAS IROFS correlation was developed as a tool to facilitate the NRC finding that must be made regarding PSSC construction prior to issuance of a license. The results of the correlation (or mapping) are provided in Tables 17-1, 17-2, and 17.3. The LA tables were developed by initially mapping ISAS events to the events in the CAR that resulted in the PSSCs identified in CAR Table 5.6-1. Subsequently, for each event, ISAS IROFS (also referred to as control groups) were mapped to the PSSCs associated with the corresponding CAR event.





As discussed in FSER Section 1.2.1.3.1, the NRC will complete verification of construction of the 53 PSSCs using a sample-based inspection program of the IROFS that relate to a PSSC. In order to assure that each ISAS IROFS can be considered for the sample-based PSSC inspection program, all the ISAS IROFS must be mapped to at least one PSSC. As such, an additional verification of PSSC to IROFS mapping was performed to validate that the IROFS identified in the ISAS were mapped to at least one PSSC. This validation, which is documented in project document *CAR PSSCs to ISAS IROFS Summary Compliance Crosswalk*, ensures that, upon verification of the construction of the 53 PSSCs, all the IROFS identified in the ISAS were included in the inspection pool for at least one PSSC.

The terms "ISAS IROFS", "IROFS Control Group", and "PSSC Control Group" are used throughout LA Chapter 17 to support PSSC verification and are defined below:

- ISAS IROFS – IROFS credited in the ISAS to meet 10CFR70.61 performance objectives. These IROFS are summarized in tables at the end of each event section (e.g., loss of confinement, explosion) in the ISAS.
- IROFS Control Group – IROFS systems, structures, and components required to perform the credited ISAS IROFS safety function. The use of ISAS IROFS and IROFS Control Group for PSSC completion and verification are interchangeable.
- PSSC Control Group – the grouping of IROFS components to support PSSC completion and verification. For non-criticality PSSCs, "PSSC Control Group" and "IROFS Control Group" are synonymous. For PSSC-009, criticality IROFS are grouped by unit/system for both engineered IROFS and administrative IROFS into PSSC Control Groups. PSSC Control Group is only used to support PSSC completion and verification.

Changes to the ISA (e.g., revisions to events, added or deleted IROFS control groups) are reflected in updates to Tables 17-1 through 17-3, as appropriate.

In summary,

a) 10 CFR 70.23(a)(8) requires the NRC to verify construction of principal structures, systems, and components prior to issuance of a license
b) A PSSC to ISAS IROFS correlation has been developed as a tool to facilitate PSSC verification
    1) Configuration management of PSSC to ISAS IROFS correlation is maintained
    2) All ISAS IROFS are mapped to at least one PSSC
c) As discussed in FSER Section 1.2.1.3.1, the NRC must complete verification of construction of 53 PSSCs.

17.3. PSSC Completion

One of the goals of MOX Services approach to PSSC completion is to provide NRC the earliest reasonable notification that a PSSC has been constructed in accordance with the application. The early notification allows NRC reviews and inspections to be performed in closer time proximity to the actual MOX Services completion. This approach can balance resources associated with PSSC verification as well as supporting near "real time" NRC PSSC verification. As discussed in Section 17.1, PSSC completion is one step in preparing for operations. In support of early NRC notification and recognizing additional future activities (e.g., preoperational testing, operational readiness reviews), MOX Services has developed PSSC completion options that recognizes processes embedded in the construction process (system turnover process with punchlist items), planned delayed final completion (e.g, Temporary Construction Openings, jumpers to facilitate testing), and recognition that some items are not required for initial operations (e.g., cask loading equipment).

PSSCs can involve multiple components and safety functions. To facilitate early completion notifications, PSSCs have been subdivided into PSSC control groups (see Tables 17-1 through 17-3). A PSSC control group may be associated with ISAS IROFS (e.g., VHD system, PSSC-006-015) or a process unit (e.g., KPA Criticality IROFS, PSSC-009-032). For PSSC completion, the PSSC control groups are only used to facilitate PSSC verification.

For engineered IROFS Systems, Structures, and Components (SSCs), MOX Services will provide NRC notification of completion of a PSSC or PSSC control group. If there are items associated with the PSSC control group that have not been completed, these items will be identified consistent with the factors discussed below.

- Any remaining items (e.g., punchlist items) shall be tracked to completion by a formal project process (e.g., PP11-42, *Construction Project Turnover*, PP16-4 *Temporary Modification Request*)
- While many of these punchlist items are expected to be minor and would not affect startup testing, a subset of the punchlist items include planned delayed completion items. Examples of these planned items include:

    - Temporary Construction Openings (TCOs). TCOs exist to facilitate introduction of components into the facility/room to facilitate construction.
    - Jumpers. Jumpers are temporary configurations (e.g., temporary pipe to reroute testing fluids) that are necessary to support preoperational testing and are requested and controlled under the Temporary Modification Request process. The temporary modification process tracks the use of these temporary configurations until the final permanent configuration is completed (i.e., after the completion of the associated preoperational test(s).
    - Filters. In order to not diminish filtration capability during construction, the filters that will be used for operations may not be installed until close to the operating phase (e.g., operational readiness).
    - Glovebox panels. In order to minimize the potential for damage during nearby construction activities, the final glovebox panels installation may be delayed until their installation is required to support preoperational testing.

  Project processes will ensure any remaining items are completed prior to operations.
- SSCs that are not required for initial operation. As operations is a phased process, not all SSCs may be necessary to support the initial operations (e.g., only 1 of the two pellet

presses). Similarly, some SSCs may not be required until later in operations (e.g., assembly loading into a cask for transport). PSSC control groups where this condition may exist at the time of the associated PSSC completion letter are identified in Tables 17-1 and 17-2. When proceeding to operations without the "not required IROFS" being installed, the planned operations must be bounded by the existing licensing basis or the licensing basis will be updated consistent with the processes discussed in LA Chapter 16. Provisions must also exist to ensure installation of the remaining items is completed and any necessary preoperational testing is performed prior to placing these items in service.

Tables 17-1 through 17-3 include the proposed PSSC completion basis and the identified PSSC control groups that contain IROFS that are not required for initial operations. As PSSC control groups are completed, the PSSC completion column will be updated to 1) reference the applicable PSSC completion letter, 2) summarize any items remaining that affect PSSC completion and 3) identify IROFS that are not installed and are not required for initial operation. If there are IROFS that are not installed, the ISA will be updated, as necessary, consistent with initial operations configuration. Any impacts on the licensing basis will be evaluated consistent with processes described in LA Chapter 16.

Administrative IROFS are considered PSSC complete when they are implemented in the first level approved document (e.g., Operating Limits Manual). First level document refers to an approved document that supports operations which establishes requirements that may also be flowed down into lower level documents. For example, an administrative IROFS that requires power to be removed prior to maintenance can be PSSC complete with a specification in the OLM that establishes the operating requirement even though subsequent maintenance procedures may duplicate that requirement.

The PSSC Completion Letter provided to the NRC will identify any remaining items along with the process that will assure completion. In all cases, operations will be consistent with the licensing design basis as well as the Integrated Safety Analysis.

17.4. PSSC Completion Packages

The completion of construction of a PSSC or PSSC control group is documented in a PSSC Completion Package. PSSC completion packages are not intended to include all the information that supports the conclusion the PSSC has been constructed in accordance with the application. PSSC completion packages are predominantly expected to be crosswalks that provide references to the supporting completion documentation. PSSC completion packages are prepared using controlled project processes and are maintained, as necessary, until the NRC verifies completion of the PSSCs and issues a license to MOX Services.

17.5. PSSC Completion Letters

When a PSSC or PSSC control group has been completed in accordance with the application and the associated PSSC Completion Package has been developed, MOX Services will notify the NRC by a PSSC completion letter. At a minimum, the PSSC completion letter will provide the following information:

- o ISAS IROFS associated with PSSC or PSSC control group
- o Summary description of key components
- o Completion basis with references
- o Discussion of any NRC inspection open items
- o Summary of any remaining items to complete and identification of the process(es) that will ensure completion
- o Summary of any IROFS that are not required for initial operation and supporting licensing basis.

17.6. PSSC All Complete Letter

MOX Services will provide NRC a "PSSC all complete" letter which will include the MOX basis that PSSCs have been constructed in accordance with the application. While the letter is predominantly a summary listing of the submitted completion letters, it may include additional references to demonstrate IROFS commodities that may not be directly associated with a PSSC control group have also been constructed.

17.7. PSSC Configuration Management

Included in MOX Services processes for PSSC completion is PSSC Configuration Management (CM). PSSC CM is necessary to ensure that the NRC is notified of any material changes to the PSSC after the PSSC has been declared complete. A graded approach is utilized for PSSC CM as 1) some changes may not require changes to the PSSC Completion Package or NRC notification, 2) some changes may require changes to the PSSC Completion Package but do not require NRC notification, and 3) some changes may require changes to the PSSC Completion Package and NRC notification. Changes that impact a completed PSSC that would require updates to the PSSC Completion Package and NRC notification are:

- Result of a license amendment, or
- Change in the associated licensing design basis, or
- Add IROFS
- Software changes/updates - reserved

Impacts of changes/updates to software on PSSC configuration management is under development and will be added to Chapter 17 at later date, but no later than the submittal of a completion package that include IROFS software.

Changes associated with a completed PSSC that are the result of a design change associated with an IROFS or software changes/updates (reserved), but do not meet the above criteria, require an update of the PSSC Completion Package but do not require formal NRC notification. These changes are not considered material to the PSSC completion notification. For example, a design change that deletes an IROFS in a previously completed PSSC does not significantly impact the completion determination unless it meets one of the other criteria (e.g., requires a license amendment) since the other IROFS within the PSSC remain "completed" and remain consistent with the licensing design basis. The PSSC Completion Package and any revisions are available for NRC inspection.

Changes that do not meet the above criteria are considered not material to the PSSC completion determination or the PSSC Completion Package. Records are available to support NRC verification that MOX Services has not performed activities that materially impact the PSSC Completion except as described above.

17.8. PSSC Completion Process

The primary goal of the PSSC Completion Process is to provide timely NRC notification of PSSC construction completion. Timely notification refers to notification as close to the actual PSSC construction completion as reasonably achievable recognizing a PSSC may contain many IROFS components that are installed over a period of time. The MOX Services PSSC Completion Process, which is documented and controlled in project processes, is a step in receiving authorization to operate the MOX facility although other key steps (preoperational testing, Operational Readiness Review) remain to be completed prior to being able to receive and possess special nuclear material.

CAR Table 5.6-1 summarizes the 53 PSSCs that were identified in the Construction Authorization Request and provides the starting point for correlating the IROFS identified in the Integrated Safety Analysis Summary with the CAR PSSCs. The correlation is developed to support the NRC in making their PSSC verification in accordance with 10CFR70.23(a)(8) prior to issuing a license. In addition, MOX Services has divided the PSSCs into PSSC Control Groups to allow timely notification of PSSC completion. The PSSC Control Groups are provided in Tables 17-1 (non-criticality IROFS), 17-2 (criticality engineered IROFS) and 17-3 (criticality administrative IROFS).

The tables also identify duplicate PSSC Control Groups. Duplicate PSSC Control Groups include PSSC Control Groups whose IROFS components are a subset of another Control Group. In order to eliminate redundant tracking and closure, these duplicate Control Groups are closed to the appropriate PSSC Control Group that will provide the PSSC Closure Package/NRC notification. In this context, the subject PSSC control group that is "closed" is considered PSSC complete and the scope for PSSC completion is included in another PSSC control group. The closure of these duplicate PSSC control groups is documented in the Chapter 17 tables and additional NRC notification is not planned. If all PSSC control groups of a given PSSC are duplicates and are closed to other PSSC control groups, a PSSC completion letter will be provided to the NRC providing the basis for the closure (i.e., the associated IROFS are included in other PSSC control groups). Therefore, a minimum of one PSSC completion letter will be provided for each PSSC.

PSSC Completion basis will be based on one of the following methods

- Approved procedure(s) – approved procedure(s) that demonstrates implementation of administrative IROFS
- Verification of criticality dimensions – completed forms that verifies construction meets required criticality dimensions
- Construction turnover process – process used to verify and document system has been constructed in accordance with design and system is ready for pre-operational testing. A defined set of action items may exist as a result of the construction turnover process. These action items are tracked to completion under a controlled MOX Services process.
- Field walkdown – documented field walkdown to verify completed construction. When this option is utilized, it must include appropriate documentation to support the SSC(s) has been constructed in accordance with the application.
- Operating Limits Manual – approved OLM is used to demonstrate implementation of an IROFS (e.g., administrative IROFS).
- Programs – approved program document (e.g., Criticality Safety Program) that implements IROFS
- Work Package Verification – closed construction work package is used to demonstrate completed construction of an SSC.
- Procurement specification – approved procurement specification that establishes requirements consistent with the licensing design basis for IROFS SSCs that are not permanently installed in the facility (e.g., conveyance cart). This method is complemented by approved project process that allows the use of IROFS in this category consistent with the approved specification.
- Other – method is described and justification provided that the chosen method demonstrates completed construction in accordance with application.

Tables 17-1, 17-2, and 17-3 provide the methodology expected to be used for each PSSC Completion Control Group; however, the completion methodology that is used as the basis for PSSC completion will be provided in each PSSC Completion Letter.

17.9. Design Basis

PSSC and IROFS Relationship

- All ISAS IROFS are mapped to at least one CAR PSSC
- Changes to ISA are incorporated into Tables 17-1 through 17-3, as appropriate

PSSC Completion

- MOX Services will provide notification of completion of a PSSC or PSSC control group. Any remaining items to complete will be identified consistent with the following factors, as appropriate
  - Punchlist items are tracked to completion by a formal project process
  - Provisions exist for planned delayed completion items to ensure remaining items are completed prior to operations
  - For SSCs not required for initial operation
    - Plant configuration and operations are bounded by the licensing basis
    - Provisions exist that ensure installation of the remaining items and any required preoperational testing is performed prior to placing these SSCs in service
- Administrative IROFS are considered PSSC complete when are implemented in the first level approved document (e.g., Operating Limits Manual)
- Tables 17-1 through 17-3 will be updated as PSSCs or PSSC control groups are completed with the following information
  - Applicable PSSC completion letter reference
  - Summary of any remaining items to complete
  - Identification of any IROFS that are not required for initial operation

PSSC Completion Packages

- Completion of PSSC/PSSC control group is documented in a PSSC completion package
- PSSC completion packages are prepared and maintained, as necessary, until the NRC verifies completion of PSSCs and issues an operating license to MOX Services

PSSC Completion Letters

- MOX Services will notify the NRC when a PSSC or PSSC control group has been completed
- PSSC completion letter will provide the following information
  - ISAS IROFS associated with the Completion Letter
  - Summary description of key components
  - Completion basis with references
  - NRC Open items
  - Summary of any remaining PSSC completion items to complete and identification of the process(es) that will ensure completion

    - Summary of any IROFS that are not required for initial operation and supporting licensing basis

PSSC All Complete Letter

- MOX Services will notify the NRC when all PSSCs are complete.
- PSSC all complete letter will include basis that all PSSCs have been constructed in accordance with the application

PSSC Configuration Management

- Impacted PSSC Completion Package(s) will be updated and revised PSSC Completion Letter(s) will be provided to the NRC as a result of
    - License amendment
    - Change in the associated licensing design basis
    - Added IROFS
    - Software changes/updates - reserved
- Impacted PSSC Completion Package(s) will be updated as a result of
    - Design change
    - IROFS deletion
    - Software changes/updates – reserved
- Records are available to support NRC verification that MOX Services has not performed activities that materially impact PSSC completion except as described above.

PSSC Completion Process

- Closure of duplicate/subset PSSC control groups is documented in Tables 17-1 through 17-3 and additional NRC notification is not required
- At least one PSSC completion letter will be provided for each PSSC
- PSSC completion basis is based on one of the following methods
    - Construction turnover process
        - Action items are tracked to completion under controlled process
    - Approved procedure(s)
    - Verification of criticality dimensions
    - Field walkdown
        - Must include/reference appropriate documentation to support the SSC(s) have been constructed in accordance with the application
    - Operating Limits Manual
    - Approved program(s)
    - Work Package Verification
    - Procurement specification
        - Used for SSCs not permanently installed in the facility

    - Must be complemented by approved process required use consistent with the specification
  - Other
    - Method is described and justified that it demonstrates completion in accordance with the application
- PSSC completion basis will be provided in PSSC Completion Letter

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
PSSC-001 3013 Canister	3013 container administrative control	PSSC-001-001	Admin	OLM specification				
	Position Sensors	PSSC-001-002	Engineered Components	Closed to PSSC-032-007		**Complete** Scope included in PSSC-032-007	N/A	
PSSC-002 3013 Transport Cask	9975 shipping package administrative control	PSSC-002-001	Admin	OLM specification				
	3013 container administrative control	PSSC-002-002	Admin	Closed to PSSC-001-001		**Complete** Scope included in PSSC-001-001	N/A	
PSSC-003 Backflow Prevention Features	Administrative - Maintain adequate buffer space in KWD unit to complete Pu and solvent flushing activities during shutdown	PSSC-003-001	Admin	Closed to PSSC-045-006		**Complete** Scope included in PSSC-045-006	N/A	
	Administrative - Operator response to high level alarm	PSSC-003-002	Admin	Procedure(s)				
	Administrative - Process solution transfer	PSSC-003-003	Admin	Procedure(s)				
	Administrative - Secondary confinement	PSSC-003-004	Admin	OLM Specification or Procedure(s)				
	Administrative - Utility and reagent system usage	PSSC-003-005	Admin	Procedure(s)				
	AP Process Temperature Controls	PSSC-003-006	Engineered Components	Turnover or Work Package				

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	AP tank high level instrumentation	PSSC-003-007	Engineered Components	Turnover or Work Package				
	Component drains and overflows	PSSC-003-008	Engineered Piping	Turnover or Work Package				
	Pressure monitors for detection of primary-secondary interface failures	PSSC-003-009	Engineered Components	Turnover or Work Package				
	Process vessels and pipes	PSSC-003-010	Engineered Components	Closed to PSSC-041-001		**Complete** Scope included in PSSC-041-001	N/A	
PSSC-004 C2 Confinement System Passive Barrier (MDE)	MDE static confinement	PSSC-004-001	Engineered System	Turnover Package				
PSSC-005 C3 Confinement System (HDE)	HDE system	PSSC-005-001	Engineered System	Turnover Package				
	HVAC control system override	PSSC-005-002	Engineered Components	Turnover or Work Package				
	HVAC supply air (HSA) Emergency Supply system	PSSC-005-003	Engineered System	Closed to PSSC-050-001		**Complete** Scope included in PSSC-050-001	N/A	
	MDE static confinement	PSSC-005-004	Engineered System	Closed to PSSC-004-001		**Complete** Scope included in PSSC-004-001	N/A	

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	Pneumatic transfer room air supply HEPA filters	PSSC-005-005	Engineered Components	Turnover or Work Package				
	HDE final filter temperature and differential pressure instrumentation	PSSC-005-006	Engineered Components	Turnover or Work Package				
	MDE final filter temperature and differential pressure instrumentation	PSSC-005-007	Engineered Components	Turnover or Work Package				
	VHD system	PSSC-005-008	Engineered System	Closed to PSSC-006-015		**Complete** Scope included in PSSC-006-015	N/A	
PSSC-006 C4 Confinement System (VHD)	Administrative - Facility worker evacuation in response to alarms	PSSC-006-001	Admin	Closed to PSSC-019-003		**Complete** Scope included in PSSC-019-003	N/A	
	Administrative - Manually switch HEPA filter trains on high differential pressure or high temperature affecting the VHD or HDE final filters	PSSC-006-002	Admin	Procedure(s)				
	Administrative - Restricted use of double door docking system bins	PSSC-006-003	Admin	OLM specification				
	Exterior feeding head process ventilation	PSSC-006-004	Engineered Components	Turnover or Work Package				

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	system pressure boundary							
	Glovebox	PSSC-006-005	Engineered Components	Closed to PSSC-024-001		**Complete** Scope included in PSSC-024-001	N/A	
	Glovebox differential pressure instrumentation and alarms	PSSC-006-006	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			.
	Glovebox dump valves	PSSC-006-007	Engineered Components	Closed to PSSC-025-002		**Complete** Scope included in PSSC-025-002	N/A	
	Glovebox ventilation gas supply isolation valves	PSSC-006-008	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	HDE system	PSSC-006-009	Engineered System	Closed to PSSC-005-001		**Complete** Scope included in PSSC-005-001	N/A	
	Feeding head blower suction and exhaust HEPA filters	PSSC-006-010	Engineered Components	Turnover or Work Package				
	KWG static confinement	PSSC-006-011	Engineered System	Turnover Package				
	NTP/LTP/LLP pneumatic transfer	PSSC-006-012	Engineered Components	Closed to PSSC-025-007		**Complete** Scope included in PSSC-025-007	N/A	

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	system pressure boundary							
	SMT system	PSSC-006-013	Engineered Components	Closed to PSSC-046-001		**Complete** Scope included in PSSC-046-001	N/A	
	VHD final filter temperature and differential pressure instrumentation	PSSC-006-014	Engineered Components	Turnover or Work Package				
	VHD system	PSSC-006-015	Engineered System	Turnover Package				
PSSC-007 Chemical Safety Controls	Administrative - Chemical safety controls	PSSC-007-001	Admin	Procedure(s)				
	Administrative - Drip tray sampling controls	PSSC-007-002	Admin	Procedure(s)				
	Administrative - Process level controls	PSSC-007-003	Admin	Procedure(s)				
	Administrative control - hydrogen peroxide sequence of operations in dissolution receiving tanks	PSSC-007-004	Admin	Procedure(s)				
	Administrative Control Related to KWG Operation	PSSC-007-005	Admin	Closed to PSSC-038-001		**Complete** Scope included in PSSC-038-001	N/A	
	AP process tank isolation	PSSC-007-006	Engineered Components	Turnover or Work Package				
	Automatic sampling controls	PSSC-007-007	Engineered Components	Turnover or Work Package				

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	Density controls for the detection of aqueous phase	PSSC-007-008	Engineered Components	Turnover or Work Package				
	Density controls for the detection of HTP/TBP	PSSC-007-009	Engineered Components	Turnover or Work Package				
	Fire detection and suppression system	PSSC-007-010	Engineered Systems/ Components	Closed to PSSC-022-001 and PSSC-022-002 completion		**Complete** Scope included in PSSC-022-001 and PSSC-022-002	N/A	
	Administrative Control: disposition of aqueous solution in the KPA back-end rework tank	PSSC-007-011	Admin	Procedure				
	Process flow controls	PSSC-007-012	Engineered Components	Turnover or Work Package				
	Process level controls	PSSC-007-013	Engineered Components	Turnover or Work Package				
	Process sampling controls (administrative)	PSSC-007-014	Admin	Procedure(s)				
	Process vessel offgas venting	PSSC-007-015	Engineered Components	Closed to PSSC-038-003		**Complete** Scope included in PSSC-038-003	N/A	
	Radiation detection controls	PSSC-007-016	Engineered Components	Turnover or Work Package				
	Reagent sampling controls (administrative)	PSSC-007-017	Admin	Procedure(s)				
	Reagent tank isolation	PSSC-007-018	Engineered Components	Turnover or Work Package				

Table 17-1: PSSC Completion (not including criticality IROFS)								
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PSSC-008 Combustible Loading Controls	Administrative - Combustible material loading controls	PSSC-008-001	Admin	OLM Specification				
	Administrative - Facility worker action for self protection	PSSC-008-002	Admin	Closed to PSSC-019-002		**Complete** Scope included in PSSC-019-002	N/A	
	Administrative - Operator actions regarding handling of transfer containers	PSSC-008-003	Admin	Procedure(s)				
	Administrative - Transient ignition source controls	PSSC-008-004	Admin	OLM Specification				
	Administrative - Truck bay door closure controls	PSSC-008-005	Admin	OLM specification				
	3013 container administrative control	PSSC-008-006	Admin	Closed to PSSC-001-001		**Complete** Scope included in PSSC-001-001	N/A	
PSSC-009 Criticality Controls	See LA Tables 17-2 and 17-3							
PSSC-010 Double-Walled Pipe	Administrative - Facility worker evacuation of process room C-217	PSSC-010-001	Admin	Procedure(s)				

Table 17-1: PSSC Completion (not including criticality IROFS)								
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	Administrative Controls Related to KWG Operation	PSSC-010-002	Admin	Closed to PSSC-038-001		**Complete** Scope included in PSSC-038-001	N/A	
	AP process offgas treatment unit (KWG)	PSSC-010-003	Engineered Systems/ Components	Closed to PSSC-038-003		**Complete** Scope included in PSSC-038-003	N/A	
	AP process tank isolation	PSSC-010-004	Engineered Components	Closed to PSSC-007-006		**Complete** Scope included in PSSC-007-006	N/A	
	Automatic sampling controls	PSSC-010-005	Engineered Components	Closed to PSSC-007-007		**Complete** Scope included in PSSC-007-007	N/A	
	Double walled process pipe	PSSC-010-006	Engineered Components	Turnover or Work Package				
	Process filters	PSSC-010-007	Engineered Components	Turnover or Work Package				
	Process sampling controls (administrative)	PSSC-010-008	Admin	Procedure(s)				
	Process vessels and pipes	PSSC-010-009	Engineered Components	Closed to PSSC-041-001		**Complete** Scope included in PSSC-041-001	N/A	
PSSC-011 Electrolyzer Structure	Electrolyzer insulators and isolators.	PSSC-011-001	Engineered Components	Closed to PSSC-026-001		**Complete** DCS-NRC-000462 NRC ML 17339A183	N/A	
PSSC-012 Emergency	Emergency power systems - 4.16 KV	PSSC-012-001	Engineered Components	Turnover or Work Package				

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AC Power System								
	Emergency power systems- 480 VAC	PSSC-012-002	Engineered Components	Turnover or Work Package				
	Emergency power systems- 120 VAC UPS	PSSC-012-003	Engineered Components	Turnover or Work Package				
	Emergency power systems- 240/120 VAC	PSSC-012-004	Engineered Components	Turnover or Work Package				
	Emergency power systems - 125 VDC	PSSC-012-005	Engineered Components	Turnover or Work Package				
	Emergency power systems- 480/277 VAC	PSSC-012-006	Engineered Components	Turnover or Work Package				
	Emergency power systems - Fuel Oil Emergency Diesel Generator	PSSC-012-007	Engineered Components	Closed to PSSC-018-001		**Complete** Scope included in PSSC-018-001	N/A	
	Emergency power systems - Emergency Control Panels	PSSC-012-008	Engineered Components	Turnover or Work Package				
	HSA Emergency Supply System	PSSC-012-009	Engineered Components	Closed to PSSC-050-001		**Complete** Scope included in PSSC-050-001	N/A	
	HDE System	PSSC-012-010	Engineered Components	Closed to PSSC-005-001		**Complete** Scope included in PSSC-005-001	N/A	
	HVD System	PSSC-012-011	Engineered Components	Closed to PSSC-017-001		**Complete** Scope included in PSSC-017-001	N/A	

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	HVC System	PSSC-012-012	Engineered Components	Closed to PSSC-013-001		**Complete** Scope included in PSSC-013-001	N/A	
	VFD Cooling Units	PSSC-012-013	Engineered Components	Turnover or Work Package				
PSSC-013 Emergency Control Room Air-Conditioning System (HVC)	HVC	PSSC-013-001	Engineered System	Turnover Package				
PSSC-014 Emergency Control System (ECS)	IROFS included in other control groups and are not repeated for PSSC-014 (HDE, VHD, EGF HVC, HVD, Emergency Power Systems Train A/B, SMT, POE)	PSSC-014-001	Engineered Systems/ Components	Closed to PSSC-005-001, PSSC-006-015, PSSC-013-001, PSSC-017-001, PSSC-018- 001, PSSC-012-001 through PSSC-012-013, PSSC-046-001, PSSC-044-003		**Complete** Scope included in PSSC-005-001, PSSC-006-015, PSSC-013-001, PSSC-017-001, PSSC-018-001, PSSC-012-001 through PSSC-012-013, PSSC-046-001, PSSC-044-003	N/A	
PSSC-015 Emergency DC Power System	The Emergency power systems IROFS associated with EMMH-03 are captured in PSSC-012 and not repeated here.	PSSC-015-001	Engineered Systems/ Components	Closed to PSSC-012-001 through PSSC-012-013		**Complete** DCS-NRC-000463 NRC ML 17339A186	N/A	

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
PSSC-016 Emergency Generator Building Structure (BEG)	Emergency Generator Building Structure (BEG)	PSSC-016-001	Structure	Turnover or Work Package(s)				
	Emergency Fuel Storage Vault (UEF)	PSSC-016-002	Structure	Turnover or Work Package(s)				
	Structures (BMF, BEG, UEF)	PSSC-016-003	Structure	Closed to PSSC-036-005, PSSC-016-001 and PSSC-016-002		**Complete** Scope included in PSSC-036-005, PSSC-016-001, and PSSC-016-002	N/A	
	Drum Handling Controls	PSSC-016-004	Admin	Procedure(s)				
PSSC-017 Emergency Generator Ventilation System (HVD)	Emergency power systems/emergency diesel generators (EDGs) including associated support systems)	PSSC-017-001	Engineered Components	Turnover Package				

Table 17-1: PSSC Completion (not including criticality IROFS)								
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PSSC-018 Emergency Diesel Generator Fuel Oil System (EGF)	Emergency power systems/emergency diesel generators (EDGs) including associated support systems	PSSC-018-001	Engineered Components	Turnover or Work Package				
PSSC-019 Facility Worker Action	Administrative - Combustible material loading controls	PSSC-019-001	Admin	Closed to PSSC-008-001		**Complete** Scope included in PSSC-008-001	N/A	
	Administrative - Facility worker action for self-protection	PSSC-019-002	Admin	Procedure(s)				
	Administrative - Facility worker evacuation in response to alarms	PSSC-019-003	Admin	Procedure(s)				
	Administrative - Facility worker evacuation in response to observation of event	PSSC-019-004	Admin	Closed to PSSC-031-008		**Complete** Scope included in PSSC-031-008	N/A	
	Administrative - Transient ignition source controls	PSSC-019-005	Admin	Closed to PSSC-008-004		**Complete** Scope included in PSSC-008-004	N/A	
	Administrative controls of gas feed sequence	PSSC-019-006	Admin	Procedure(s)				
	Fire wrap of LLP/LTP/NTP piping in C2 areas	PSSC-019-007	Engineered Components	Turnover or Work Package				

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
PSSC-020 Facility Worker Controls	Administrative - Hoist/crane/winch operation	PSSC-020-001	Admin	Procedure(s)				
	Administrative - Radioactive material removal from glovebox during maintenance activities requiring crane movements	PSSC-020-002	Admin	OLM Specification				
	Administrative - Radioactive material removal from primary confinements during maintenance	PSSC-020-003	Admin	OLM Specification				
	Administrative - Respiratory protection during bag in/bag out activity	PSSC-020-004	Admin	Procedure(s)				
	Administrative - Respiratory protection during crane movements in or over gloveboxes	PSSC-020-005	Admin	Procedure(s)				
	Administrative - Respiratory protection during crane movements over the sintering furnace	PSSC-020-006	Admin	Procedure(s)				

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	Administrative - Respiratory protection during maintenance on primary confinements	PSSC-020-007	Admin	Procedure(s)				
PSSC-021 Fire Barriers	Administrative - Combustible material loading controls	PSSC-021-001	Admin	Closed to PSSC-008-001		**Complete** Scope included in PSSC-008-001	N/A	
	Administrative - Transient ignition source controls	PSSC-021-002	Admin	Closed to PSSC-008-004		**Complete** Scope included in PSSC-008-004	N/A	
	NFPA 221 compliance controls for specified process cells/rooms	PSSC-021-003	Engineered Components	Turnover or Work Package				
	Connections and flex hose at press upper enclosure interface in room B-334	PSSC-021-004	Engineered Components	Turnover or Work Package				
	Fire dampers	PSSC-021-005	Engineered Components	Turnover or Work Package				
	Fire detection system	PSSC-021-006	Engineered Components	Closed to PSSC-022-001		**Complete** Scope included in PSSC-022-001	N/A	
	Fire Detection System in the Hydraulic Pump Room and the Test Line (LCT and LFT) Room	PSSC-021-007	Engineered Components	Turnover or Work Package				
	Fire isolation valves	PSSC-021-008	Engineered Components	Turnover or Work Package				
	Fire propagation barriers and fire wraps	PSSC-021-009	Engineered Components	Turnover or Work Package				

Table 17-1: PSSC Completion (not including criticality IROFS)								
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	Fire suppression system	PSSC-021-010	Engineered Components	Closed to PSSC-022-002		**Complete** Scope included in PSSC-022-002	N/A	
	Fire wrap sealing the exterior of the horizontal portion of the secondary containment and any of its supports in rooms B-119 and B-121	PSSC-021-011	Engineered Components	Turnover or Work Package				
	Fire stop at the transition from the horizontal to the vertical portion of the secondary containment in rooms B-119 and B-121	PSSC-021-012	Engineered Components	Turnover or Work Package				
	Flexible secondary containment connection at press upper enclosure in process rooms B-119 and B-121	PSSC-021-013	Engineered Components	Turnover or Work Package				
	Hydraulic oil lines	PSSC-021-014	Engineered Components	Turnover or Work Package				
	Hydraulic press drip tray	PSSC-021-015	Engineered Components	Turnover or Work Package				
	Hydraulic unit drip tray	PSSC-021-016	Engineered Components	Turnover or Work Package				
	Hydraulic unit level detection	PSSC-021-017	Engineered Components	Turnover or Work Package				

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	Operator presence during LCT press operation	PSSC-021-018	Admin	Procedure(s)				
	Penetration seals, removable panels, plugs	PSSC-021-019	Engineered Components	Turnover or Work Package				
	Personnel fire doors	PSSC-021-020	Engineered Components	Turnover or Work Package				
	Process fire doors/fire locks	PSSC-021-021	Engineered Components	Turnover or Work Package				
	Supply air fire dampers	PSSC-021-022	Engineered Components	Turnover or Work Package				
	Vertical secondary containment in process rooms B-119 and B-121	PSSC-021-023	Engineered Components	Turnover or Work Package				
	Walls, floors and ceilings	PSSC-021-024	Structure	Closed to PSSC-036-005, PSSC-016-001 and PSSC-016-002		**Complete** Scope included in PSSC-036-005, PSSC-016-001 and PSSC-016-002	N/A	
PSSC-022 Fire Detection and Suppression	Fire detection system	PSSC-022-001	Engineered Components	Turnover Package				
	Fire suppression system	PSSC-022-002	Engineered Components	Turnover Package				
PSSC-023 Fluid Transport Systems	Administrative control of transfers from drip trays or process vessels to prevent overflow	PSSC-023-001	Admin	Procedure(s)				

Table 17-1: PSSC Completion (not including criticality IROFS)								
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	Administrative control for high process variable alarms	PSSC-023-002	Admin	Procedure(s)				
	Administrative control on DCS flow to defined vessels	PSSC-023-003	Admin	Procedure(s)				
	Administrative control to maintain adequate buffer space in the KWD HAW unit to complete Pu and solvent flushing activities during shutdown	PSSC-023-004	Admin	Closed to PSSC-045-006		**Complete** Scope included in PSSC-045-006	N/A	
	Component drains and overflows	PSSC-023-005	Engineered Components	Closed to PSSC-003-008		**Complete** Scope included in PSSC-003-008	N/A	
	Process flow controls	PSSC-023-006	Engineered Components	Closed to PSSC-007-012		**Complete** Scope included in PSSC-007-012	N/A	
	Process level controls	PSSC-023-007	Engineered Components	Turnover or Work Package				
	Process pressure controls	PSSC-023-008	Engineered Components	Turnover or Work Package				
	Process temperature controls	PSSC-023-009	Engineered Components	Closed to PSSC-045-038		**Complete** Scope included in PSSC-045-038	N/A	
	Process vessels and pipes	PSSC-023-010	Engineered Components	Closed to PSSC-041-001		**Complete** Scope included in PSSC-041-001		

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	SPS confinement boundary	PSSC-023-011	Engineered Components	Turnover or Work Package				
	FPW confinement boundary	PSSC-023-012	Engineered Components	Turnover or Work Package				
	Pressure monitors for detection of primary-secondary interface failures	PSSC-023-013	Engineered Components	Closed to PSSC-003-009		**Complete** Scope in included in PSSC-003-009	N/A	
PSSC-024 Glovebox	Glovebox	PSSC-024-001	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
PSSC-025 Glovebox Pressure Controls	Glovebox	PSSC-025-001	Engineered Components	Closed to PSSC-024-001		**Complete** Scope included in PSSC-024-001	N/A	
	Glovebox dump valves	PSSC-025-002	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	Glovebox vacuum breaker valve (backdraft damper)	PSSC-025-003	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	LTP Equalization Line restricting Orifice	PSSC-025-004	Engineered Components	Turnover or Work Package				
	LTP pressure controls	PSSC-025-005	Engineered Components	Turnover or Work Package				

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	NTP pressure controls	PSSC-025-006	Engineered Components	Turnover or Work Package				
	NTP/LTP/LLP pneumatic transfer system pressure boundary	PSSC-025-007	Engineered Components	Turnover or Work Package				
	PFE/PFF offgas cooling water piping pressure boundary	PSSC-025-008	Engineered Components	Turnover or Work Package				
	Restricting orifice on high pressure gas lines	PSSC-025-009	Engineered Components	Turnover or Work Package				
PSSC-026 Guide Sleeves	Electrolyzer insulators and isolators	PSSC-026-001	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			.
PSSC-027	Administrative - Chemical safety controls	PSSC-027-001	Admin	Procedure(s)				
PSSC-028 Instrument Air System (Scavenging Air) IAS	Administrative control on detection of and recovery from loss of instrument air	PSSC-028-001	Admin	Procedure(s)				
	AP process tank isolation	PSSC-028-002	Engineered Components	Closed to PSSC-007-006		**Complete** Scope included in PSSC-007-006	N/A	
	Automatic sampling controls	PSSC-028-003	Engineered Components	Closed to PSSC-007-007		**Complete** Scope included in PSSC-007-007	N/A	

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	Emergency scavenging air system	PSSC-028-004	Engineered Subsystem	Turnover or Work Package				
	Mass controls	PSSC-028-005	Engineered Components	Turnover or Work Package				
	Process level controls	PSSC-028-006	Engineered Components	Closed to PSSC-007-013		**Complete** Scope included in PSSC-007-013	N/A	
	Process sampling controls (administrative)	PSSC-028-007	Admin	Closed to PSSC-007-014		**Complete** Scope included in PSSC-007-014	N/A	
	Radiation detection controls	PSSC-028-008	Engineered Components	Closed to PSSC-007-016		**Complete** Scope included in PSSC-007-016	N/A	
PSSC-029 Laboratory Material Controls	These safety functions were not required to be credited in the ISA in order to meet 10 CFR 70.61.	PSSC-029-001	N/A	N/A		**Complete** DCS-NRC-000425 NRC ML 16147A096	N/A	NRC-DCS-000765 NRC ML 16257A016
PSSC-030 Maintenance Activity Controls	Administrative controls on electrolyzer maintenance activities	PSSC-030-001	Admin	OLM Specification				
PSSC-031 Material Handling Controls	Administrative - Automated rod insertion machine setup	PSSC-031-001	Admin	Procedure(s)				
	Administrative - Material handling controls - container	PSSC-031-002	Admin	Procedure(s)				

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	Administrative - Material handling controls (final HEPA filters)	PSSC-031-003	Admin	Procedure(s)				
	Administrative - Material handling controls (primary confinement)	PSSC-031-004	Admin	Procedure(s)				
	Administrative - Respiratory protection during bag in/bag out activity	PSSC-031-005	Admin	Closed to PSSC-020-004		**Complete** Scope included in PSSC-020-004	N/A	
	Administrative - Restriction on use of polyethylene DDDS bin	PSSC-031-006	Admin	OLM Specification				
	Position Sensors	PSSC-031-007	Engineered Components	Closed to PSSC-032-007		**Complete** Scope included in PSSC-032-007	N/A	
	Facility Worker Evacuation in Response to Observation of Event	PSSC-031-008	Admin	Procedure(s)				
	Keyed switches	PSSC-031-009	Engineered Components	Turnover or Work Package				
PSSC-032 Material Handling Equipment	Administrative - Removable design feature placement	PSSC-032-001	Admin	OLM Specification				
	Anti-reversing device	PSSC-032-002	Engineered Components	Turnover or Work Package				
	Connection Module Scale	PSSC-032-003	Engineered Components	Turnover or Work Package				

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	Helical gearbox	PSSC-032-004	Engineered Components	Turnover or Work Package				
	Hydraulic damping cylinder	PSSC-032-005	Engineered Components	Turnover or Work Package				
	Safety programmable logic controller (SPLC)	PSSC-032-006	Engineered Components	Turnover or Work Package				
	Position Sensors	PSSC-032-007	Engineered Components	Turnover or Work Package				
	Torque Sensors	PSSC-032-008	Engineered Components	Turnover or Work Package				
	Speed Sensors	PSSC-032-009	Engineered Components	Turnover or Work Package				
	Shaft Brakes	PSSC-032-010	Engineered Components	Turnover or Work Package				
PSSC-033 Material Maintenance and Surveillance Programs	Glovebox	PSSC-033-001	Engineered Components	Closed to PSSC-024-001		**Complete** Scope included in PSSC-024-001	N/A	
	Process vessels and pipes	PSSC-033-002	Engineered Components	Closed to PSSC-041-001		**Complete** Scope included in PSSC-041-001	N/A	
	VHD system	PSSC-033-003	Engineered System	Closed to PSSC-006-015		**Complete** Scope included in PSSC-006-015	N/A	
PSSC-034 MFFF Tornado Dampers	Tornado dampers	PSSC-034-001	Engineered Components	Turnover or Work Package				
	VHD lab fan shutdown	PSSC-034-002	Admin	Procedure(s)	Contains IROFS not required			

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
					for initial operations			
PSSC-035 Missile Barriers	Missile barriers	PSSC-035-001	Engineered Components	Turnover or Work Package				
PSSC-036 MOX Fuel Fabrication Building Structure	Administrative - Combustible material loading controls	PSSC-036-001	Admin	Closed to PSSC-008-001		**Complete** Scope included in PSSC-008-001	N/A	
	Administrative - Facility worker action for self-protection	PSSC-036-002	Admin	Closed to PSSC-019-002		**Complete** Scope included in PSSC-019-002	N/A	
	Administrative - Transient ignition source controls	PSSC-036-003	Admin	Closed to PSSC-008-004		**Complete** Scope included in PSSC-008-004	N/A	
	Fire dampers	PSSC-036-004	Engineered Components	Closed to PSSC-021-005		**Complete** Scope included in PSSC-021-005	N/A	
	MOX Fuel Fabrication Building structure (BMF)	PSSC-036-005	Structure	Turnover or Work Package				
	Penetration seals, removable panels, plugs	PSSC-036-006	Engineered Components	Closed to PSSC-021-019		**Complete** Scope included in PSSC-021-019	N/A	
	Personnel fire doors	PSSC-036-007	Engineered Components	Closed to PSSC-021-020		**Complete** Scope included in PSSC-021-020	N/A	

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	Structures (BMF, BEG, UEF)	PSSC-036-008	Structure	Closed to PSSC-036-005, PSSC-016-001 and PSSC-016-002		**Complete** Scope included in PSSC-036-005, PSSC-016-001, and PSSC-016-002	N/A	
	Walls, floors and ceilings	PSSC-036-009	Structure	Closed to PSSC-036-005, PSSC-016-001 and PSSC-016-002		**Complete** Scope included in PSSC-036-005, PSSC-016-001, and PSSC-016-002	N/A	
PSSC-037 MOX Fuel Transport Cask	MOX fresh fuel package administrative control	PSSC-037-001	Admin	OLM Specification				
PSSC-038 Offgas Treatment System (KWG)	Administrative Controls related to KWG Operation	PSSC-038-001	Admin	Procedure(s)				
	Administrative controls related to venting for KDD electrolyzers	PSSC-038-002	Admin	Procedure(s)				
	Process vessel offgas venting	PSSC-038-003	Engineered Components	Turnover or Work Package				
PSSC-039 PTFE Insulator	Electrolyzer insulators and isolators	PSSC-039-001	Engineered Components	Closed to PSSC-026-001		**Complete** DCS-NRC-000456 NRC ML 17339A201	N/A	

Table 17-1: PSSC Completion (not including criticality IROFS)

PSSC	ISAS IROFS	PSSC Control Group Tracking	IROFS Type	Proposed 10CFR70.23(a)(8) Completion Basis	Comments	MOX Services PSSC Completion	Remaining Completion Items	NRC Acceptance
PSSC-040 Pressure Vessel Controls	No IROFS identified for this event in ISAS	PSSC-040-001	N/A	N/A				
PSSC-041 Process Cells	Process vessels and pipes	PSSC-041-001	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			.
	Process cell drip tray	PSSC-041-002	Engineered Components	Turnover or Work Package				
	Process cell exhaust system (POE)	PSSC-041-003	Engineered System (POE)	Closed to PSSC-044-003		**Complete** Scope included in PSSC-044-003	N/A	
PSSC-042 Process Cell Entry Controls	Administrative - Process cell entry controls	PSSC-042-001	Admin	OLM Specification				
PSSC-043 Process Cell Fire Prevention Features	Administrative - Combustible material loading controls	PSSC-043-001	Admin	Closed to PSSC-008-001		**Complete** Scope included in PSSC-008-001	N/A	
	Administrative - Process cell entry controls	PSSC-043-002	Admin	Closed to PSSC-042-001		**Complete** Scope included in PSSC-042-001	N/A	
	Administrative - Transient/in situ ignition source controls	PSSC-043-003	Admin	OLM Specification				
	BMF electrical grounding system for	PSSC-043-004	Engineered Components	Turnover or Work Package				

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	pipes, tanks, and vessels within the process cells							
	Process cell maintenance controls	PSSC-043-005	Admin	Procedure(s)				
	Welded process vessels and pipes	PSSC-043-006	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
PSSC-044 Process Cell Exhaust System (POE)	Administrative Controls Related to KWG Operation	PSSC-044-001	Admin	Closed to PSSC-038-001		**Complete** Scope included in PSSC-038-001	N/A	
	AP process offgas treatment unit (KWG)	PSSC-044-002	Engineered System	Closed to PSSC-038-003		**Complete** Scope included in PSSC-038-003	N/A	
	Process cell exhaust system (POE)	PSSC-044-003	Engineered System	Turnover Package				
PSSC-045 Process Safety Control Subsystem	Administrative - Drip tray sampling controls	PSSC-045-001	Admin	Closed to PSSC-007-002		**Complete** Scope included in PSSC-007-002	N/A	
	Administrative - Facility worker evacuation in response to alarms	PSSC-045-002	Admin	Closed to PSSC-019-003		**Complete** Scope included in PSSC-019-003	N/A	
	Administrative control - Premix Ar-H gas supply	PSSC-045-003	Admin	Procedure(s)				
	Administrative control of process flow	PSSC-045-004	Admin	Procedure(s)				

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	Administrative control to limit pump operation	PSSC-045-005	Admin	Procedure(s)				
	Administrative control to maintain adequate buffer space in the KWD HAW unit to complete Pu and solvent flushing activities during shutdown	PSSC-045-006	Admin	OLM Specification				
	Administrative control to preload settler	PSSC-045-007	Admin	Procedure(s)				
	Administrative control to prevent drying	PSSC-045-008	Admin	Procedure(s)				
	Administrative control to station a firewatch	PSSC-045-009	Admin	OLM Specification				
	Administrative flushing control	PSSC-045-010	Admin	Procedure(s)				
	AP process tank isolation	PSSC-045-011	Engineered Components	Closed to PSSC-007-006		**Complete** Scope included in PSSC-007-006	N/A	
	Area hydrogen monitors	PSSC-045-012	Engineered Components	Turnover or Work Package				
	Automatic sampling controls	PSSC-045-013	Engineered Components	Closed to PSSC-007-007		**Complete** Scope included in PSSC-007-007	N/A	
	Catholyte flow monitor	PSSC-045-014	Engineered Components	Turnover or Work Package				
	Density controls for the detection of HTP/TBP	PSSC-045-015	Engineered Components	Closed to PSSC-007-009		**Complete** Scope included in PSSC-007-009	N/A	

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	Drip tray level transmitters and associated alarms	PSSC-045-016	Engineered Components	Turnover or Work Package				
	Drip tray sampling controls	PSSC-045-017	Admin	Procedure(s)				
	Emergency scavenging air system	PSSC-045-018	Engineered Components	Closed to PSSC-028-004		**Complete** Scope included in PSSC-028-004	N/A	
	Fire detection and suppression system	PSSC-045-019	Engineered Components	Closed to PSSC-022-001 and PSSC-022-002		**Complete** Scope included in PSSC-022-001 and PSSC-022-002	N/A	
	Gas pad hydrogen analyzers	PSSC-045-020	Engineered Components	Turnover or Work Package				
	Gas pad hydrogen/argon mix isolation valves	PSSC-045-021	Engineered Components	Turnover or Work Package				
	Glovebox	PSSC-045-022	Engineered Components	Closed to PSSC-024-001		**Complete** Scope included in PSSC-024-001	N/A	
	Glovebox differential pressure instrumentation and alarms	PSSC-045-023	Engineered Components	Closed to PSSC-006-006		**Complete** Scope included in PSSC-06-006	N/A	
	Glovebox dump valves	PSSC-045-024	Engineered Components	Closed to PSSC-025-002		**Complete** Scope included in PSSC-025-002	N/A	
	Humidifier mixer level controls	PSSC-045-025	Engineered Components	Turnover or Work Package				
	Double walled process pipe	PSSC-045-026	Engineered Components	Closed to PSSC-010-006		**Complete**	N/A	

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
						Scope included in PSSC-010-006		
	Inert gas purge supply valves	PSSC-045-027	Engineered Components	Turnover or Work Package				
	Inert gas purge system	PSSC-045-028	Engineered Components	Turnover or Work Package				
	Laboratory furnace chiller pressure relief valves	PSSC-045-029	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations.			
	Pressure monitors for detection of primary-secondary interface failures	PSSC-045-030	Engineered Components	Closed to PSSC-003-009		**Complete** Scope included in PSSC-003-009	N/A	
	Normality controller	PSSC-045-031	Engineered Components	Turnover or Work Package				
	Offgas glovebox isolation valves	PSSC-045-032	Engineered Components	Turnover or Work Package				
	Pneumatic transfer exhauster controls	PSSC-045-033	Engineered Components	Turnover or Work Package				
	Process cell drip trays	PSSC-045-034	Engineered Components	Closed to PSSC-041-002		**Complete** Scope included in PSSC-041-002	N/A	
	Process cell fire prevention features	PSSC-045-035	Engineered Components	Closed to PSSC-043-001 through PSSC-043-006		**Complete** Scope included in PSSC-043-001 through PSSC-043-006	N/A	

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	Process level controls	PSSC-045-036	Engineered Components	Closed to PSSC-023-007		**Complete** Scope included in PSSC-023-007	N/A	
	Process sampling controls (administrative)	PSSC-045-037	Admin	Closed to PSSC-007-014		**Complete** Scope included in PSSC-007-014	N/A	
	Process temperature controls	PSSC-045-038	Engineered Components	Turnover or Work Package				
	Process Vessel Off-Gas Venting	PSSC-045-039	Engineered Components	Closed to PSSC-038-003		**Complete** Scope included in PSSC-038-003	N/A	
	Process vessels and pipes	PSSC-045-040	Engineered Components	Closed to PSSC-041-001		**Complete** Scope included in PSSC-041-001	N/A	
	Reagent sampling controls (administrative)	PSSC-045-041	Admin	Closed to PSSC-007-017		**Complete** Scope included in PSSC-007-017	N/A	
	Reagent tank isolation	PSSC-045-042	Engineered Components	Closed to PSSC-007-018		**Complete** Scope included in PSSC-007-018	N/A	
	Seismic monitoring and trip system	PSSC-045-043	Engineered System/ Components	Closed to PSSC-046-001		**Complete** Scope included in PSSC-046-001	N/A	
	Sintering furnace airlock Ar-H supply valve limit switches	PSSC-045-044	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	Sintering furnace airlock doors closed limit switches	PSSC-045-045	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	Sintering furnace airlock furnace equalization valves limit switches	PSSC-045-046	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	Sintering furnace airlock glovebox equalization valves limit switches	PSSC-045-047	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	Sintering furnace airlock pressure interlocks	PSSC-045-048	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	Sintering furnace airlock vacuum valve limit switches	PSSC-045-049	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	Sintering furnace Ar-H supply valves limit switches	PSSC-045-050	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	Sintering furnace exhaust bypass valves	PSSC-045-051	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	Sintering furnace gas supply isolation valves	PSSC-045-052	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	Sintering furnace humidifier restricting orifice	PSSC-045-053	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	Sintering furnace humidifier restricting orifice bypass controls	PSSC-045-054	Admin	Procedure(s)	Contains IROFS not required for initial operations			
	Sintering furnace offgas glovebox HVAC flow controls	PSSC-045-055	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	Sintering furnace oxygen monitors	PSSC-045-056	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	Sintering furnace safety controllers	PSSC-045-057	Engineered Components	Closed to PSSC-048-008		**Complete** Scope included in PSSC-048-008	N/A	
	Sintering furnace shell	PSSC-045-058	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			.
	Slab settler	PSSC-045-059	Engineered Components	Turnover or Work Package				
	HVAC Temperature Controls	PSSC-045-060	Engineered Components	Turnover or Work Package				
	VHD system	PSSC-045-061	Engineered System	Closed to PSSC-006-015		**Complete** Scope included in PSSC-006-015	N/A	
PSSC-046 Seismic Monitoring Sys & Assoc Seismic Isolation Valves	SMT System (Train A and B)	PSSC-046-001	Engineered System/ Components	Turnover or Work Package				
PSSC-047 Sintered Silicon Nitride Barrier	This safety function was not required to be credited in the ISA in order to meet 10 CFR 70.61.	PSSC-047-001	N/A	N/A		**Complete** DCS-NRC-000427 NRC ML 16217A034	N/A	NRC-DCS-000771 NRC ML 16294A166

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
PSSC-048 Sintering Furnace (PFE/PFF)	Furnace airlock temperature controls	PSSC-048-001	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	Furnace shell	PSSC-048-002	Engineered Components	Closed to PSSC-045-058		**Complete** Scope included in PSSC-045-058	N/A	
	Sintering furnace airlock doors closed limit switches	PSSC-048-003	Engineered Components	Closed to PSSC-045-045		**Complete** Scope included in PSSC-045-045	N/A	
	Sintering furnace Ar-H gas flow orifice	PSSC-048-004	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	Sintering furnace cooling loop temperature controls	PSSC-048-005	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	Sintering furnace loss of cooling controls	PSSC-048-006	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	Sintering furnace resistive heating shutdown controls	PSSC-048-007	Engineered Components	Turnover or Work Package	Contains IROFS not required			

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
					for initial operations			
	Sintering furnace safety controllers	PSSC-048-008	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
PSSC-049 Sintering Furnace Pressure Controls	Sintering furnace airlocks pressure relief valves	PSSC-049-001	Engineered Components	Work Package	Contains IROFS not required for initial operations			
	Sintering furnace pressure relief valves	PSSC-049-002	Engineered Components	Work Package	Contains IROFS not required for initial operations			
PSSC-050 Supply Air System (HSA)	HVAC supply air (HSA) Emergency Supply system	PSSC-050-001	Engineered System (HSA - subsystem)	Turnover Package				
PSSC-051 Transfer Container	Conveyance cart administrative control	PSSC-051-001	Admin	OLM Specification				
	HDE system	PSSC-051-002	Engineered System (HDE)	Closed to PSSC-005-001		**Complete** Scope included in PSSC-005-001	N/A	
	Stainless steel DDS (transfer container)	PSSC-051-003	Engineered Components	Procurement Specification				

Table 17-1: PSSC Completion (not including criticality IROFS)								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	Transfer container administrative control	PSSC-051-004	Admin	OLM Specification				
PSSC-052 Waste Containers	Waste containers	PSSC-052-001	Engineered Components	OLM specification or Procurement Specification				
PSSC-053 Waste Transfer Line	High alpha liquid waste transfer line	PSSC-053-001	Engineered Components	Turnover or Work Package				
	Protection of the High Alpha Liquid Waste Transfer Line	PSSC-053-002	Admin	OLM Specification				

Table 17-2: PSSC Completion – Criticality Engineered IROFS								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
PSSC-009 Criticality Controls	DCE Criticality IROFS	PSSC-009-001	Engineered Components	Turnover or Work Package				
	DCM Criticality IROFS	PSSC-009-002	Engineered Components	Turnover or Work Package				
	DCP Criticality IROFS	PSSC-009-003	Engineered Components	Turnover or Work Package				
	DCS Criticality IROFS	PSSC-009-004	Engineered Components	Turnover or Work Package				
	EEC Criticality IROFS	PSSC-009-005	Engineered Components	Closed to PSSC-012-002 and PSSC-12-003		**Complete** Scope included in PSSC-012-002 and PSSC-012-003	N/A	
	GDE Criticality IROFS	PSSC-009-006	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations.			.
	GNO Criticality IROFS	PSSC-009-007	Engineered Components	Turnover or Work Package				
	HSA Criticality IROFS	PSSC-009-008	Engineered Components	Turnover or Work Package				
	HWS Criticality IROFS	PSSC-009-009	Engineered Components	Turnover or Work Package				
	IAS Criticality IROFS	PSSC-009-010	Engineered Components	Turnover or Work Package				
	KCA Criticality IROFS	PSSC-009-011	Engineered Components	Turnover or Work Package				

Table 17-2: PSSC Completion – Criticality Engineered IROFS								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	KCB Criticality IROFS	PSSC-009-012	Engineered Components	Turnover or Work Package				
	KCC Criticality IROFS	PSSC-009-013	Engineered Components	Turnover or Work Package				
	KCD Criticality IROFS	PSSC-009-014	Engineered Components	Turnover or Work Package				
	KDA Criticality IROFS	PSSC-009-015	Engineered Components	Turnover or Work Package				
	KDB Criticality IROFS	PSSC-009-016	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	KDD Criticality IROFS	PSSC-009-017	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	KDM Criticality IROFS	PSSC-009-018	Engineered Components	Turnover or Work Package				
	KKJ Criticality IROFS	PSSC-009-019	Engineered Components	Turnover or Work Package				
	KLA Criticality IROFS	PSSC-009-020	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	KLB Criticality IROFS	PSSC-009-021	Engineered Components	Turnover or Work Package	Contains IROFS not			

Table 17-2: PSSC Completion – Criticality Engineered IROFS								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
					required for initial operations			
	KLC Criticality IROFS	PSSC-009-022	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	KLD Criticality IROFS	PSSC-009-023	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	KLE Criticality IROFS	PSSC-009-024	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	KLF Criticality IROFS	PSSC-009-025	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	KLH Criticality IROFS	PSSC-009-026	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			

Table 17-2: PSSC Completion – Criticality Engineered IROFS

PSSC	ISAS IROFS	PSSC Control Group Tracking	IROFS Type	Proposed 10CFR70.23(a)(8) Completion Basis	Comments	MOX Services PSSC Completion	Remaining Completion Items	NRC Acceptance
	KLI Criticality IROFS	PSSC-009-027	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			.
	KLJ Criticality IROFS	PSSC-009-028	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	KLK Criticality IROFS	PSSC-009-029	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	KLL Criticality IROFS	PSSC-009-030	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	KLN Criticality IROFS	PSSC-009-031	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	KPA Criticality IROFS	PSSC-009-032	Engineered Components	Turnover or Work Package				
	KPB Criticality IROFS	PSSC-009-033	Engineered Components	Turnover or Work Package				

Table 17-2: PSSC Completion – Criticality Engineered IROFS

PSSC	ISAS IROFS	PSSC Control Group Tracking	IROFS Type	Proposed 10CFR70.23(a)(8) Completion Basis	Comments	MOX Services PSSC Completion	Remaining Completion Items	NRC Acceptance
	KPC Criticality IROFS	PSSC-009-034	Engineered Components	Turnover or Work Package				
	KPG Criticality IROFS	PSSC-009-035	Engineered Components	Turnover or Work Package				
	KWG Criticality IROFS	PSSC-009-036	Engineered Components	Turnover or Work Package				
	KWS Criticality IROFS	PSSC-009-037	Engineered Components	Turnover or Work Package				
	LAC Criticality IROFS	PSSC-009-038	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	LBT Criticality IROFS	PSSC-009-039	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	LCP Criticality IROFS	PSSC-009-040	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	LCT Criticality IROFS	PSSC-009-041	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			.

Table 17-2: PSSC Completion – Criticality Engineered IROFS								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	LDS Criticality IROFS	PSSC-009-042	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	LFT Criticality IROFS	PSSC-009-043	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	LFX Criticality IROFS	PSSC-009-044	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	LGF Criticality IROFS	PSSC-009-045	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	LLJ Criticality IROFS	PSSC-009-046	Engineered Components	Turnover or Work Package				
	LLP Criticality IROFS	PSSC-009-047	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	LME Criticality IROFS	PSSC-009-048	Engineered Components	Turnover or Work Package	Contains IROFS not			

Table 17-2: PSSC Completion – Criticality Engineered IROFS								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
					required for initial operations			
	LPG Criticality IROFS	PSSC-009-049	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	LPS Criticality IROFS	PSSC-009-050	Engineered Components	Turnover or Work Package				
	LRD Criticality IROFS	PSSC-009-051	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	LSR Criticality IROFS	PSSC-009-052	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	LTP Criticality IROFS	PSSC-009-053	Engineered Components	Turnover or Work Package				
	NBX Criticality IROFS	PSSC-009-054	Engineered Components	Turnover or Work Package				
	NBY Criticality IROFS	PSSC-009-055	Engineered Components	Turnover or Work Package				
	NCR Criticality IROFS	PSSC-009-056	Engineered Components	Turnover or Work Package				

Table 17-2: PSSC Completion – Criticality Engineered IROFS								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	NDD Criticality IROFS	PSSC-009-057	Engineered Components	Turnover or Work Package				
	NDP Criticality IROFS	PSSC-009-058	Engineered Components	Turnover or Work Package				
	NDS Criticality IROFS	PSSC-009-059	Engineered Components	Turnover or Work Package				
	NNJ Criticality IROFS	PSSC-009-060	Engineered Components	Turnover or Work Package				
	NPG Criticality IROFS	PSSC-009-061	Engineered Components	Turnover or Work Package				
	NPH Criticality IROFS	PSSC-009-062	Engineered Components	Turnover or Work Package				
	NTM Criticality IROFS	PSSC-009-063	Engineered Components	Turnover or Work Package				
	NTP Criticality IROFS	PSSC-009-064	Engineered Components	Turnover or Work Package				
	NXR Criticality IROFS	PSSC-009-065	Engineered Components	Turnover or Work Package				
	PFE Criticality IROFS	PSSC-009-066	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	PFF Criticality IROFS	PSSC-009-067	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			

Table 17-2: PSSC Completion – Criticality Engineered IROFS								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	PPJ Criticality IROFS	PSSC-009-068	Engineered Components	Turnover or Work Package				
	PRE Criticality IROFS	PSSC-009-069	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	PRF Criticality IROFS	PSSC-009-070	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	Process Vessels & Pipes in Process Cells Criticality IROFS	PSSC-009-071	Engineered Components	Closed to PSSC-041-001		**Complete** Scope included in PSSC-041-001	N/A	
	PSE Criticality IROFS	PSSC-009-072	Engineered Components	Turnover or Work Package				
	PSF Criticality IROFS	PSSC-009-073	Engineered Components	Turnover or Work Package				
	PSI Criticality IROFS	PSSC-009-074	Engineered Components	Turnover or Work Package				
	PSJ Criticality IROFS	PSSC-009-075	Engineered Components	Turnover or Work Package				
	RCA Criticality IROFS	PSSC-009-076	Engineered Components	Turnover or Work Package				
	RNA Criticality IROFS	PSSC-009-077	Engineered Components	Turnover or Work Package				

Table 17-2: PSSC Completion – Criticality Engineered IROFS								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	RUN Criticality IROFS	PSSC-009-078	Engineered Components	Turnover or Work Package				
	SMK Criticality IROFS	PSSC-009-079	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	STK Criticality IROFS	PSSC-009-080	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	TAS Criticality IROFS	PSSC-009-081	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			
	VHD Criticality IROFS	PSSC-009-082	Engineered Components	Turnover or Work Package				
	Miscellaneous Criticality IROFS	PSSC-009-083	Engineered Components	Turnover or Work Package	Contains IROFS not required for initial operations			

Table 17-3: PSSC Completion – Criticality Administrative IROFS								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
PSSC-009 Criticality Controls	DCE Admin Controls	PSSC-009-500	Administrative	Procedure(s)				
	DCP Admin Controls	PSSC-009-501	Administrative	Procedure(s)				
	GDE Admin Controls	PSSC-009-502	Administrative	Procedure(s)				
	GME Admin Controls	PSSC-009-503	Administrative	Procedure(s)				
	KCA Admin Controls	PSSC-009-504	Administrative	Procedure(s)				
	KCB Admin Controls	PSSC-009-505	Administrative	Procedure(s)				
	KCC Admin Controls	PSSC-009-506	Administrative	Procedure(s)				
	KCD Admin Controls	PSSC-009-507	Administrative	Procedure(s)				
	KDA Admin Controls	PSSC-009-508	Administrative	Procedure(s)				
	KDB Admin Controls	PSSC-009-509	Administrative	Procedure(s)				
	KDD Admin Controls	PSSC-009-510	Administrative	Procedure(s)				
	KKJ Admin Controls	PSSC-009-511	Administrative	Procedure(s)				

Table 17-3: PSSC Completion – Criticality Administrative IROFS								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	KKJ Aux Admin Controls	PSSC-009-512	Administrative	Procedure(s)				
	KPA Admin Controls	PSSC-009-513	Administrative	Procedure(s)				
	LCT Admin Controls	PSSC-009-514	Administrative	Procedure(s)				
	LGF Admin Controls	PSSC-009-515	Administrative	Procedure(s)				
	LLJ Admin Controls	PSSC-009-516	Administrative	Procedure(s)				
	NBX Admin Controls	PSSC-009-517	Administrative	Procedure(s)				
	NCR Admin Controls	PSSC-009-518	Administrative	Procedure(s)				
	NDD Admin Controls	PSSC-009-519	Administrative	Procedure(s)				
	NDP Admin Controls	PSSC-009-520	Administrative	Procedure(s)				
	NDS Admin Controls	PSSC-009-521	Administrative	Procedure(s)				
	NPG Admin Controls	PSSC-009-522	Administrative	Procedure(s)				
	NTM Admin Controls	PSSC-009-523	Administrative	Procedure(s)				
	NXR Admin Controls	PSSC-009-524	Administrative	Procedure(s)				
	PAD Admin Controls	PSSC-009-525	Administrative	Procedure(s)				

Table 17-3: PSSC Completion – Criticality Administrative IROFS								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	PAR Admin Controls	PSSC-009-526	Administrative	Procedure(s)				
	PFE Admin Controls	PSSC-009-527	Administrative	Procedure(s)				
	PML Admin Controls	PSSC-009-528	Administrative	Procedure(s)				
	PPJ Admin Controls	PSSC-009-529	Administrative	Procedure(s)				
	PQE Admin Controls	PSSC-009-530	Administrative	Procedure(s)				
	PRE Admin Controls	PSSC-009-531	Administrative	Procedure(s)				
	PSE Admin Controls	PSSC-009-532	Administrative	Procedure(s)				
	PTE Admin Controls	PSSC-009-533	Administrative	Procedure(s)				
	RCA Admin Controls	PSSC-009-534	Administrative	Procedure(s)				
	SDK Admin Controls	PSSC-009-535	Administrative	Procedure(s)				
	STK Admin Controls	PSSC-009-536	Administrative	Procedure(s)				
	TAS Admin Controls	PSSC-009-537	Administrative	Procedure(s)				
	TGM Admin Controls	PSSC-009-538	Administrative	Procedure(s)				
	TTJ Admin Controls	PSSC-009-539	Administrative	Procedure(s)				

Table 17-3: PSSC Completion – Criticality Administrative IROFS								
PSSC	**ISAS IROFS**	**PSSC Control Group Tracking**	**IROFS Type**	**Proposed 10CFR70.23(a)(8) Completion Basis**	**Comments**	**MOX Services PSSC Completion**	**Remaining Completion Items**	**NRC Acceptance**
	TXE Admin Controls	PSSC-009-540	Administrative	Procedure(s)				
	VDQ Admin Controls	PSSC-009-541	Administrative	Procedure(s)				
	VDT Admin Controls	PSSC-009-542	Administrative	Procedure(s)				