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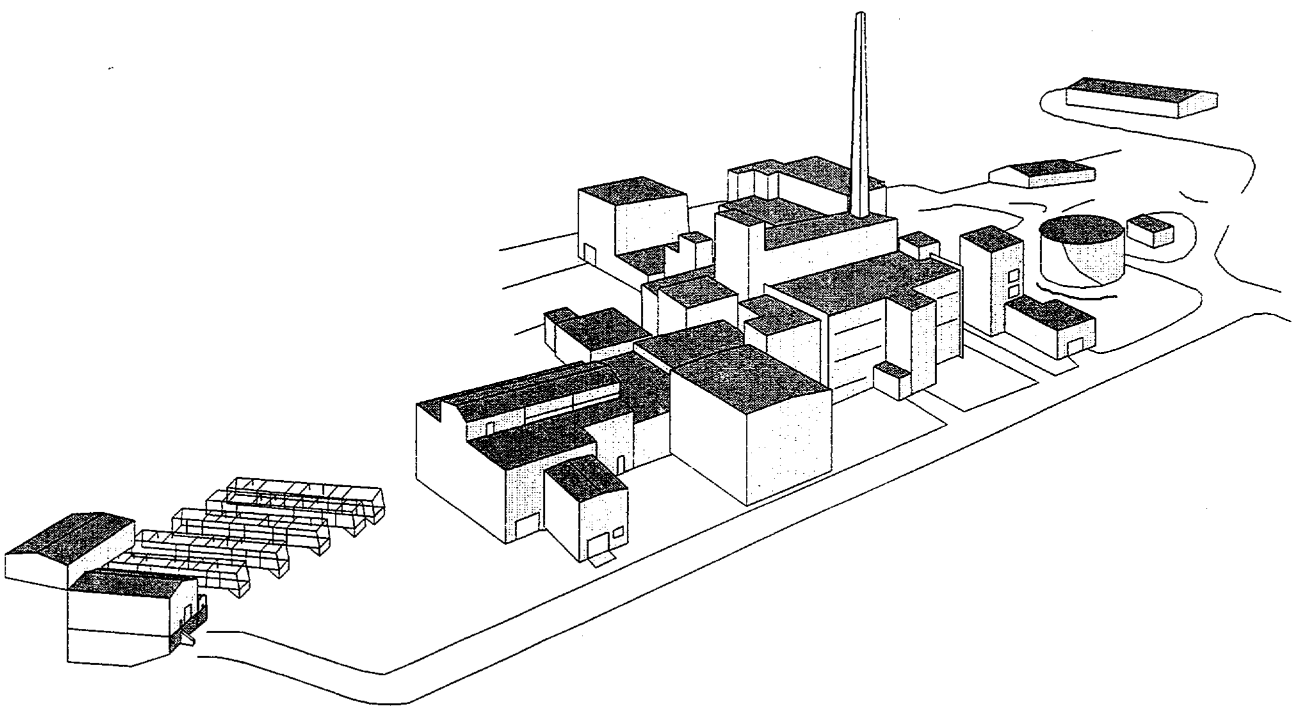
SAFETY ANALYSIS REPORT FOR FUEL RECEIVING AND STORAGE FACILITY



WVNS-SAR-012

West Valley Demonstration Project

West Valley, New York 14171

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West Valley Demonstration Project

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SAFETY ANALYSIS REPORT FOR

FUEL RECEIVING AND STORAGE FACILITY

J. C. WOLNIEWICZ

DAMES & MOORE FOR
WVNS SPENT FUEL PROJECT
MARCH 31, 2000

APPROVED HAZARD CATEGORIZATION: 2

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WVNS RECORD OF REVISION

DOCUMENT

If there are changes to the controlled document, the revision number increases by one. Indicate changes by one of the following:

- Placing a vertical black line in the margin adjacent to sentence or paragraph that was revised.
- Placing the words GENERAL REVISION at the beginning of the text.
- Placing either FC#> or PC#> (whichever applies) in the left-hand margin at the beginning of the paragraph or section where the field/page change has been made AND placing a vertical black line in the margin adjacent to the actual change.
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Example:

The vertical line in the margin indicates a change. |

FC1> The FC#> in the margin along with the vertical line (redline) indicates a change. |

Rev. No.	Description of Changes	Revision On Page(s)	Dated
0	Original Issue	All	05/28/97
1	ECN#11457 represents a routine annual update of WVNS-SAR-012, as required by DOE Order 5480.23	All	05/29/98
2	ECN#11978 represents a routine annual update of WVNS-SAR-012, as required By DOE Order 5480.23. Added information and conclusions from various FHAs, including information on lightning protection. Updated information with respect to changes in the Nuclear Criticality Safety Program. All references to standard operating procedures have been replaced with appropriate vocabulary.	All	03/31/99
3	ECN #12554 represents a routine annual update of WVNS-SAR-012, as required by DOE Order 5480.23. Standardized the acronyms and abbreviations list. Revised fuel characteristics presented in Tables 4.1-1 through 4.1-3 to reflect the latest available information. Revised description of the cask crane in Section 5.2.3.3.2 to reflect increased load capability	All	03/31/00

WVNS RECORD OF REVISION CONTINUATION FORM

Rev. No.	Description of Changes	Revision On Page(s)	Dated
3	of the cask crane due to hardware modifications and analyses, which were supported by a representative of the original crane manufacturer. Substantially revised the heating and ventilation systems discussion in Section 5.4.1 and Figure 5.4-1 to show actual current systems' configuration and use. Reflected the current situation as regards the need to install poison rods in the REG fuel as discussed in Section 6.2.1. Updated the estimated annual doses to the maximum exposed offsite individual from routine liquid and airborne releases. Revised Tables 7.2-1 and 9.2-2 to more accurately reflect laboratory analysis of radiological characteristics of the contents of high integrity containers "B" and "E." Added summary statement and reference regarding double contingency analysis in Section 8.7.4.3. Cited new or proper "WV-" and "WVDP-" documents as appropriate throughout.		

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ACRONYMS AND ABBREVIATIONS

Å	Ångstrom (10^{-8} centimeter)
A&PC	Analytical and Process Chemistry
AA	Atomic Absorption
AAC	Assembly Area Coordinator
AADT	Average Annual Daily Traffic
ABA	Authorization Basis Addendum
ACC	Ashford Community Center
ACFM	Absolute Cubic Feet Per Minute
ACGIH	American Conference of Governmental Industrial Hygienists
ACI	American Concrete Institute
A/E	Architect/Engineer
AEA	Atomic Energy Act
AEC	Atomic Energy Commission
AED	Assistant Emergency Director
AEDE	Annual Effective Dose Equivalent
AEOC	Alternate Emergency Operations Center
AES	Atomic Emission Spectrophotometer
AIHA	American Industrial Hygiene Association
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
ALARA	As Low As Reasonably Achievable
ALI	Annual Limit of Intake
ALS	Advanced Life Saving
AMCA	Air Movement and Control Association
AMS	Aerial Measurement System
AMS	Alarm Monitoring Station
ANC	Analytical Cell
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOC	Ashford Office Complex
APOC	Abnormal Pump Operating Condition
AR-OG	Acid Recovery - Off-Gas
ARC	Acid Recovery Cell
ARF	Airborne Release Fraction
ARI	Air-Conditioning and Refrigeration Institute
ARM	Area Radiation Monitor
ARPR	Acid Recovery Pump Room
ARR	Airborne Release Rate
ASCE	American Society of Civil Engineers
ASER	Annual Site Environmental Report
ASHRAE	American Society of Heating, Refrigeration, and Air-Conditioning Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
AU	Alfred University
AWS	American Welding Society

ACRONYMS AND ABBREVIATIONS (Continued)

B&P	Buffalo & Pittsburgh
BDAT	Best Demonstrated Available Technology
BDB	Beyond Design Basis
BDBE	Beyond Design Basis Earthquake
BNFL	British Nuclear Fuels Limited
BNL	Brookhaven National Laboratory
Bq	Becquerel
BRP	Big Rock Point
BSW	Bulk Storage Warehouse
BWR	Boiling Water Reactor
c	Centi, prefix for 10^{-2}
C	Coulomb
CAM	Continuous Air Monitor
CAS	Criticality Alarm System
cc	Cubic Centimeter
CC	Communications Coordinator
CCB	Cold Chemical Building
CCDS	Cold Chemical Delivery System
CCR	Chemical Crane Room
CCS	Chilled Water System
CCSR	Cold Chemical Scale Room
CCSS	Cold Chemical Sump Station
CCTV	Closed-Circuit Television
CDDS	Computer Data Display System
CDS	Criticality Detection System
CEC	Cation Exchange Capacity
CEDE	Committed Effective Dose Equivalent
cfm	Cubic feet per minute
CFMT	Concentrator Feed Make-up Tank
CFR	Code of Federal Regulations
cfs	Cubic feet per second
CGA	Compressed Gas Association
CHT	Condensate Hold Tank
Ci	Curie
CLCW	Closed-Loop Cooling Water
cm	Centimeter
CMAA	Crane Manufacturers Association of America
CMP	Construction Management Procedure
CMR	Crane Maintenance Room
COA	Chemical Operating Aisle
CPC	Chemical Process Cell
CPC-WSA	Chemical Process Cell Waste Storage Area
cpm	Counts per minute
CR	Control Room
CRM	Community Relations Manager
CRT	Cathode Ray Tube
Cs	Cesium
CSDM	Cognizant System Design Manager

ACRONYMS AND ABBREVIATIONS (Continued)

CSE	Criticality Safety Engineer
CSE	Cognizant System Engineer
CSER	Confined Space Entry Rescue
CSPF	Container Sorting and Packaging Facility
CSR	Confined Space Rescue
CSRF	Contact Size Reduction Facility
CSS	Cement Solidification System
cSv	centi-Sievert
CTS	Component Test Stand
CUA	Catholic University of America
CUP	Cask Unloading Pool
Cv	Column Volume
CVA	Chemical Viewing Aisle
CW	Cooling Tower Water
CWTP	Commercial Waste Treatment System
CY	Calendar Year
D&D	Decontamination and Decommissioning
D&M	Dames & Moore
DAC	Derived Air Concentration
DAS	Data Acquisition System
DB	Dry Bulb
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DBT	Design Basis Tornado
DBW	Design Basis Wind
DC	Drum Cell
DCF	Dose Conversion Factor
DCG	Derived Concentration Guide
DCS	Distributed Control System
DEAR	Department of Energy Acquisition Regulation
DF	Decontamination Factor
DGR	Diesel Generator Room
DOE	Department of Energy
DOE-EM	Department of Energy - Environmental Management
DOE-HQ	Department of Energy - Headquarters
DOE-HQ-EOC	Department of Energy - Headquarters - Emergency Operations Center
DOE-ID	Department of Energy - Idaho
DOE-OCRWM	Department of Energy - Office of Civilian Radioactive Waste Management
DOE-OH	Department of Energy - Ohio Field Office
DOE-PD	Department of Energy - Project Director
DOE-WV	Department of Energy - West Valley Area Office
DOE-WVDP	Department of Energy - West Valley Demonstration Project
DOELAP	Department of Energy Laboratory Accreditation Program
DOP	Dioctylphthalate
DOSR	DOE On-Site Representative
DOT	Department of Transportation
DP	Differential Pressure
dpm	Disintegrations per minute
DR	Data Recorder

ACRONYMS AND ABBREVIATIONS (Continued)

DR	Damage Ratio
DVP	Developmental Procedure
DWPF	Defense Waste Processing Facility
DWS	Demineralized Water System
E-Spec	Equipment Specification
EA&SRP	Engineering Administration & Safety Review Program
EBA	Evaluation Basis Accident
EBE	Evaluation Basis Earthquake
ECN	Engineering Change Notice
ECO	Environmental Control Officer
ED	Emergency Director
EDE	Effective Dose Equivalent
EDR	Equipment Decontamination Room
EDRVA	Equipment Decontamination Room Viewing Aisle
EDS	Electrical Power Distribution
EG	Evaluation Guideline
EHS	Employee Health Services
EID	Environmental Information Document
EIP	Emergency Implementing Procedure
EIS	Environmental Impact Statement
EMC	Emergency Management Coordinator
EMOA	East Mechanical Operating Aisle
EMP	Emergency Management Procedure
EMRT	Emergency Medical Response Team
EMT	Emergency Medical Technician
EMT	Environmental Monitoring Team
EMU	Emergency Medical Unit
EOC	Emergency Operation Center
EP	Engineering Procedure
EPA	Environmental Protection Agency
EPD	Elevation Plant Datum
EPI	Emergency Prediction Information
EPIcode	Emergency Protection Information Code
EPRI	Electric Power Research Institute
EPZ	Emergency Protection Zone
ERO	Emergency Response Organization
ERPG	Emergency Response Planning Guideline
ES&H	Environmental, Safety, and Health
ESA	Endangered Species Act
ESH&QA	Environmental, Safety, Health, and Quality Assurance
ESQA&LO	Environmental, Safety, Quality Assurance, and Laboratory Operations
FACTS	Functional and Checklist Testing of Systems
FBC	Fire Brigade Chief
FBR	Fluidized Bed Reactor
FFCA	Federal Facility Compliance Act
FHA	Fire Hazards Analysis
FM	Factory Mutual
fpm	Feet per minute

ACRONYMS AND ABBREVIATIONS (Continued)

fps	Feet per second
FRI	Feed Reduction Index
FRS	Fuel Receiving and Storage
FSAR	Final Safety Analysis Report
FSFCA	Federal and State Facility Compliance Act
FSP	Fuel Storage Pool
ft	Feet
FWCA	Fish and Wildlife Coordination Act
g	Gram
g	Gravitational Acceleration Constant
G	Giga, prefix for 10^9
GAC	Granular Activated Carbon
gal	Gallon
GC	Gas Chromatograph
GCR	General Purpose Cell Crane Room
GCS	Gravelly Clayey Soils
GE	General Electric
GET	General Employee Training
GFE	Government Furnished Equipment
gM	Gravelly mud
GM	Geometric Mean
GM	Geiger-Mueller
GOA	General Purpose Cell Operating Aisle
GOALS	General Office Automated Logging System
GOCO	Government-Owned, Contractor-Operated
GPC	General Purpose Cell
gpd	Gallons per day
GPLI	General Purpose LAN Interface
gpm	Gallons per minute
GRS	General Record Schedule
G_s	Specific gravity
GTAW	Gas Tungsten Arc Welding
h	Hour
ha	Hectare
HAC	Hot Acid Cell
HAF	Hot Acid Feed
HAPR	Hot Acid Pump Room
HAZMAT	Hazardous Materials
HAZWOPER	Hazardous Waste Operations and Emergency Response
HDC	High Density Concrete
HEC	Head End Cells
HEME	High Efficiency Mist Eliminator
HEPA	High Efficiency Particulate Air
HEV	Head End Ventilation
HFE	Human Factors Engineering
HIC	High Integrity Container
HLDS	High-Level Drainage System
HLW	High-Level Waste

ACRONYMS AND ABBREVIATIONS (Continued)

HLWIS	High-Level Waste Interim Storage
HLWISA	High-Level Waste Interim Storage Area
HLWTS	High-Level Waste Transfer System
hp	Horsepower
HPGe	Hyperpure Germanium
HPLC	High Performance Liquid Chromatography
HPS	High Pressure Sodium
HRA	Human Reliability Analysis
HRM	Human Resources Manager
HV	Heating and Ventilation
HVAC	Heating, Ventilation, and Air Conditioning
HVOS	Heating, Ventilation Operating Station
HWSF	Hazardous Waste Storage Facility
i.d.	Inner Diameter
I&C	Instrumentation and Control
IA	Instrument Air
IC	Incident Commander
ICEA	Insulated Cable Engineers Association
ICP	Inductively Coupled Plasma
ICR	Instrument Calibration Recall
ICRP	International Commission on Radiological Protection
ID	Idaho
IDLH	Immediately Dangerous to Life and Health
IEEE	Institute of Electrical and Electronics Engineers
IES	Illuminating Engineering Society
IH&S	Industrial Hygiene and Safety
ILDS	Infrared Level Detection System
in	Inch
INEL	Idaho National Engineering Laboratory
INEEL	Idaho National Environmental and Engineering Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
IRTS	Integrated Radwaste Treatment System
ISM	Integrated Safety Management
ISMS	Integrated Safety Management System
IV&V	Independent Validation and Verification
IWP	Industrial Work Permit
IWSF	Interim Waste Storage Facility
IX	Ion Exchange
JIC	Joint Information Center
JTG	Joint Test Group
k	Neutron Multiplication Factor
k	Kilo, prefix for 10^3
K_d	Partition Coefficient
k_{eff}	Effective Neutron Multiplication Factor
kg	Kilogram
K_h	Horizontal hydraulic conductivity
kN	Kilo-Newton

ACRONYMS AND ABBREVIATIONS (Continued)

kPa	Kilo-Pascal
kPag	Kilo-Pascal gauge
kph	Kilometer per hour
kV	Kilo-Volt
K_v	Vertical hydraulic conductivity
kVA	Kilovolt-ampere
kW	kilo-Watt
L	Liter
LAH	Level Alarm High
LAN	Local Area Network
LANL	Los Alamos National Laboratory
LAP	Laboratory Accreditation Program
LAP	Lower Annealing Point
LASL	Los Alamos Scientific Laboratory
lb	Pound
LCO	Limiting Condition for Operation
lfpm	Linear feet per minute
LFR	Live Fire Range
LI	Level Indicate
LIMS	Laboratory Information Management System
LLDS	Low-Level Drainage System
LLL	Lawrence Livermore Laboratory
LLNL	Lawrence Livermore National Laboratory
LLRW	Low-Level Radioactive Waste
LLW	Low-Level Waste
LLW2	Low-Level Waste Treatment Replacement Facility
LLWTF	Low-Level Waste Treatment Facility
LLWTS	Low-Level Waste Treatment System
LM	Liaison Manager
LMITCO	Lockheed-Martin Idaho Technologies Corporation
LOS	Level of Service
LOVS	Loss of Voltage Signal
LPF	Leak Path Factor
LPG	Liquid Propane Gas
lpm	Liters per minute
LPM	Liters per minute
LPS	Liquid Pretreatment System
LR	Level Record
LSA	Lag Storage Area
LUNR	Land Use and Natural Resources
LWA	Lower Warm Aisle
LWC	Liquid Waste Cell
LWTS	Liquid Waste Treatment System
LXA	Lower Extraction Aisle
m	Meter
m	Milli, prefix for 10^{-3}
m/s	Meters per second
M	Mega, prefix for 10^{6}

ACRONYMS AND ABBREVIATIONS (Continued)

M&O	Maintenance and Operations
M&O	Management and Operating
M&TE	Maintenance and Test Equipment
MAR	Material at Risk
m_b	Earthquake Magnitude
MBtu	Mega-British Thermal Units
MC	Miniature Cell
MCC	Materials Characterization Center
MCC	Motor Control Center
MCE	Maximum Credible Earthquake
mCi	milli-Curie
MEOSI	Maximally Exposed Off-Site Individual
MeV	Mega-electron Volt
MFHT	Melter Feed Hold Tank
mG	Muddy gravels
mi	Mile
MMI	Modified Mercalli Intensity
M&O	Management and Operating
MOA	Mechanical Operating Aisle
MOI	Maximally Exposed Off-Site Individual
mol	Mole
MOU	Memorandum of Understanding
MPag	Mega-Pascal gauge
MPC	Maximum Permissible Concentration
MPFL	Maximum Possible Fire Loss
mph	Miles per hour
MPO	Main Plant Operator
MPOSS	Main Plant Operations Shift Supervisor
mR/hr	Milli-Roentgen per hour
MRC	Master Records Center
mrem	Millirem
MRR	Manipulator Repair Room
MSDS	Material Safety Data Sheet
msG	Muddy Sandy Gravels
MSM	Master-Slave Manipulator
mSv	milli-Sievert
MT	Metric Ton
MTIHM	Metric Tons Initial Heavy Metal
MTU	Metric Tons Uranium
MUF	Material-Unaccounted-For
MW	Mega-Watt
MWD	Mega-Watt-Day
n	Nano, prefix for 10^{-9}
Na	Sodium
NAA	North Analytical Aisle
NAD	Nuclear Accident Dosimeter
NARA	National Archives and Records Administration
NDA	NRC-Licensed Disposal Area

ACRONYMS AND ABBREVIATIONS (Continued)

NDA-LPS	NRC-Licensed Disposal Area - Liquid Pretreatment System
n_e	Effective porosity
NEC	National Electric Code
NEMA	National Electrical Manufacturers Association
NEPA	National Environmental Policy Act
NESHAP	National Emission Standard for Hazardous Air Pollutants
NFPA	National Fire Protection Association
NFS	Nuclear Fuel Services, Inc.
NGVD	National Geodetic Vertical Datum
NIOSH	National Institute of Occupational Safety and Health
NIST	National Institute of Standards and Technology
NMC	News Media Center
NMPC	Niagara Mohawk Power Corporation
NOAA	National Oceanic and Atmospheric Administration
NP	North Plateau
NPH	Natural Phenomena Hazard
NPPS	North Plateau Pump System
NPPTS	North Plateau Pump and Treatment System
NQA	Nuclear Quality Assurance
NR	Nonconformance Report
NRC	Nuclear Regulatory Commission
NRRPT	National Registry of Radiation Protection Technology
NWS	National Weather Service
NY	New York
NYCRR	New York Code of Rules and Regulations
NYS	New York State
NYSDEC	New York State Department of Environmental Conservation
NYSDOH	New York State Department of Health
NYSERDA	New York State Energy Research and Development Authority
NYSGS	New York State Geological Survey
o.d.	Outer Diameter
OAAM	Operational Accident Assessment Manager
OAM	Operational Assessment Manager
OB	Office Building
OBE	Operating Basis Earthquake
OEP	On-Site Evaluation Point
OGA	Off-Gas Aisle
OGBR	Off-Gas Blower Room
OGC	Off-Gas Cell
OGMR	Off-Gas Monitoring Room
OGTS	Off Gas Treatment System
OH	DOE, Ohio Field Office
OH/WVDP	Ohio Field Office, West Valley Demonstration Project
OJT	On-the-Job Training
OM	Operations Manager
OOS	Out-of-Service
ORNL	Oak Ridge National Laboratory
ORR	Operational Readiness Review
ORRB	Operational Readiness Review Board

ACRONYMS AND ABBREVIATIONS (Continued)

ORT	Operations Response Team
OSC	Operations Support Center
OSHA	Occupational Safety and Health Act
OSHA	Occupational Safety and Health Administration
OSR	Operational Safety Requirement
oz	Ounce
p	Pico, prefix for 10^{-12}
P	Peta, prefix for 10^{15}
P&ID	Piping and Instrument Diagram
Pa	Pascal
PA	Project Appraisals
PAG	Protective Action Guideline
PAH	Pressure Alarm High
PBT	Performance-Based Training
PC	Partition Coefficient
PCB	Polychlorinated Biphenyl
PCDOCS	Personal Computer Document Organization and Control Software
pcf	Pounds per cubic foot
PCH	Pressure Control High
PCM	Personal Contamination Monitor
PCR	Process Chemical Room
PD	Project Director
PDAH	Pressure Differential Alarm High
PDAL	Pressure Differential Alarm Low
PDCH	Pressure Differential Control High
PDCL	Pressure Differential Control Low
PDR	Pressure Differential Record
PEL	Permissible Exposure Limit
PF	Personnel Frisker
PGA	Peak Ground Acceleration
PGSC	Pasquill-Gifford Stability Class
PHA	Process Hazards Analysis
PHA	Product Handling Area
PID	Public Information Director
PLC	Programmable Logic Controller
PM	Preventive Maintenance
PMC	Process Mechanical Cell
PMCR	Process Mechanical Cell Crane Room
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
PMP	Project Management Plan
PNL	Pacific Northwest Laboratory
PNNL	Pacific Northwest National Laboratory
PPB	Parts Per Billion
PPC	Product Purification Cell
ppm	Parts Per Million
PPM	Parts Per Million
PPS	Product Packaging and Shipping
PRC	Pressure Record Control

ACRONYMS AND ABBREVIATIONS (Continued)

PRM	Process Radiation Monitor
PSAR	Preliminary Safety Analysis Report
psf	Pound per square foot
psi	Pound per square inch
psig	Pound per square inch gauge
PSO	Plant Systems Operations
PSO	Plant Systems Operator
PSR	Process Safety Requirement
Pu	Plutonium
PVC	Polyvinyl chloride
PVS	Permanent Ventilation System
PVU	Portable Ventilation Unit
PWR	Pressurized Water Reactor
PWS	Potable Water System
QA	Quality Assurance
QA/QC	Quality Assurance/Quality Control
QAP	Quality Assurance Program
QAPP	Quality Assurance Program Plan
QAPD	Quality Assurance Program Description
QARD	Quality Assurance Requirements Document
QCN	Qualification Change Notice
QM	Quality Management
R	Roentgen
R/hr	Roentgen per hour
R&S	Radiation and Safety
R&SC	Radiation and Safety Committee
RAP	Radiological Assistance Plan
RCO	Radiological Controls Operations
RCOS	Radiological Controls Operations Supervisor
RCRA	Resource Conservation and Recovery Act
RCT	Radiological Control Technician
RCTC	Radiological Control Team Commander
RCTL	Radiation Control Team Leader
REAAM	Radiological and Environmental Accident Assessment Manager
REAM	Radiological and Environmental Assessment Manager
REG	Robert E. Ginna
rem	Roentgen Equivalent Man
RER	Ram Equipment Room
RESL	Radiological and Environmental Sciences Laboratory
RF	Respirable Fraction
RID	Records Inventory and Disposition Schedule
RMW	Radioactive Mixed Waste
RP	Radiation Protection
rpm	Revolutions per minute
RPM	Revolutions Per Minute
RPM	Radiation Protection Manager
Rt	Route
RTS	Radwaste Treatment System

ACRONYMS AND ABBREVIATIONS (Continued)

RWI	Radiological Worker I
RWII	Radiological Worker II
RWP	Radiation Work Permit
s	Second
S&EA	Safety and Environmental Assessment
SA&I	Safety Analysis and Integration
SAA	Satellite Accumulation Area
SAI	Science Applications International
SAR	Safety Analysis Report
SBS	Submerged Bed Scrubber
SCBA	Self-Contained Breathing Apparatus
scfm	Standard cubic feet per minute
SCR	Selective Catalytic Reduction
SCS	Soil Conservation Service
SCSSCs	Safety-Class Structures, Systems, and Components
SDA	New York State-Licensed Disposal Area
SEAM	Safety and Environmental Assessment Manager
sec	Second
SER	Site Environmental Report
SFCM	Slurry-Fed Ceramic Melter
SFPE	Society of Fire Protection Engineers
SFR	Secondary Filter Room
SGN	Society Generale pour les Techniques Nouvelles
SGR	Switch Gear Room
SI	International System of Units
SIP	Special Instruction Procedure
slpm	Standard liter per minute
SM	Security Manager
SMACNA	Sheet Metal and Air Conditioning Contractors National Association
SMS	Sludge Mobilization System
SMT	Slurry Mix Tank
SMWS	Sludge Mobilization and Wash System
SNF	Spent Nuclear Fuel
SNL	Sandia National Lab
SNM	Special Nuclear Material
SO	Security Officer
SOG	Seismic Owner's Group
SOP	Standard Operating Procedure
SPDES	State Pollutant Discharge Elimination System
SPO	Security Police Officer
Sr	Strontium
SR	Surveillance Requirement
SRE	Search and Reentry
SRL	Savannah River Laboratory
SRR	Scrap Removal Room
SRSS	Square-root-of-the-sum-of-the-squares
SS	Stainless Steel
SSC	Sample Storage Cell
SSCs	Systems, Structures, and Components

ACRONYMS AND ABBREVIATIONS (Continued)

SSE	Safe Shutdown Earthquake
SSS	Security Shift Supervisor
SSS	Slurry Sample System
SSWMU	Super Solid Waste Management Unit
STC	Sample Transfer Cell
STD	Standard
STP	Standard Temperature and Pressure
STS	Supernatant Treatment System
Sv	Sievert
SVS	Scale Vitrification System
SWC	Surge Withstand Capability
SWMU	Solid Waste Management Unit
T	Tera, prefix for 10^{12}
TBP	Tri-butyl phosphate
TE	Test Exception
TEDE	Total Effective Dose Equivalent
TEEL	Temporary Emergency Exposure Limit
Ti	Titanium
TID	Tamper-Indicating Device
TIG	Tungsten Inert Gas
TIP	Test Implementation Plan
TIP	Test In-Place
TIP	Test Instruction Procedure
TLD	Thermoluminescent Dosimeter
TLV	Threshold Limit Value
TN	Transnuclear, Inc.
TPC	Test Procedure Change
TPL	Test Plan
TR	Technical Requirement
TRG	Technical Review Group
TRMS	Training Records Management System
TRR	Test Results Report
TRU	Transuranic
TSB	Test and Storage Building
TSC	Technical Support Center
TSCS	Technical Support Center Staff
TSD	Technical Support Document
TSR	Technical Safety Requirement
TVS	Temporary Ventilation System
UA	Utility Air
UAP	Upper Annealing Point
UBC	Uniform Building Code
UCRL	University of California Research Laboratory
UDF	Unit Dose Factor
UL	Underwriters Laboratories, Inc.
ULO	Uranium Load Out
UPC	Uranium Product Cell
UPS	Uninterruptible Power Supply

ACRONYMS AND ABBREVIATIONS (Continued)

UR	Utility Room
USDOE	U. S. Department of Energy
USDOI	U. S. Department of the Interior
USDOL	U. S. Department of Labor
USDOT	U. S. Department of Transportation
USEPA	U. S. Environmental Protection Agency
USGS	U. S. Geological Survey
USNRC	U. S. Nuclear Regulatory Commission
USQ	Unreviewed Safety Question
USQD	Unreviewed Safety Question Determination
UWA	Upper Warm Aisle
UWS	Utility Water Supply
UXA	Upper Extraction Aisle
V	Volt
VA	Volt-Ampere
VAC	Volt Alternating Current
VDC	Volt Direct Current
V&S	Ventilation and Service Building
VEC	Ventilation Exhaust Cell
VF	Vitrification Facility
VFFCP	Vitrification Facility Fire Control Panel
VIV	Variable Inlet Vane
VL	Vitrification Liaison
VOG	Vessel Off-Gas
VOSS	Vitrification Operations Shift Supervisor
VPP	Voluntary Protection Program
VS	Vitrification System
VSR	Ventilation Supply Room
VTF	Vitrification Test Facility
VWR	Ventilation Wash Room
W	Watt
WAPS	Waste Acceptance Product Specifications
WC	Water Column
WCC	Warning Communications Center
WCCC	Warning Communications Center Communicator
WDC	Waste Dispensing Cell
WDV	Waste Dispensing Vessel
WGES	Westinghouse Government Environmental Services
WHC	Westinghouse Hanford Company
WHSE	Warehouse
WIPP	Waste Isolation Pilot Plant
WMO	Waste Management Operations
WMO	Westinghouse Maintenance Operation
WMOA	West Mechanical Operating Aisle
WNYNSC	Western New York Nuclear Service Center
WO	Work Order
WQR	Waste Qualification Report
WRPA	Waste Reduction and Packaging Area

ACRONYMS & ABBREVIATIONS (Concluded)

wt%	Weight percent
WTF	Waste Tank Farm
WTFVS	Waste Tank Farm Ventilation System
WVDP	West Valley Demonstration Project
WVNS	West Valley Nuclear Services Company
WVPP	West Valley Policies and Procedures
WVVHC	West Valley Volunteer Hose Company
XC-1	Extraction Cell 1
XC-2	Extraction Cell 2
XC-3	Extraction Cell 3
XCR	Extraction Chemical Room
XSA	Extraction Sample Aisle
y	Year
Y_d	Dry density
YOY	Young of Year
yr	Year
Y2K	Year 2000
°C	Degrees Celsius
°F	Degrees Fahrenheit
μ	Micro, prefix for 10^{-6}
χ/Q	Relative concentration

1.0 INTRODUCTION AND GENERAL DESCRIPTION OF THE FRS FACILITY

1.1 Introduction

The West Valley Fuel Receiving and Storage (FRS) facility was constructed between 1963 and 1966 by Bechtel Corporation for Nuclear Fuel Services, Inc. (NFS) as part of the original fuel reprocessing facility. The FRS facility provided for the receipt, storage, and handling of spent nuclear fuel (SNF) assemblies. Fuel receipt for reprocessing in the Main Plant began in May of 1965, and continued until November 1971 when the plant was shutdown for facility expansion. Additional shipments of 750 spent nuclear fuel (SNF) assemblies were received between February, 1973 and December, 1975, in anticipation of facility restart which never occurred. Of these 750 SNF assemblies, 625 assemblies were returned and 125 assemblies remain in storage in the fuel storage pool. The design, construction and operation of the original FRS facility and reprocessing facility was the subject of a U.S. Atomic Energy Commission (AEC) approved Final Safety Analysis Report (FSAR) (Nuclear Fuel Services, Inc., 1964).

In 1980, Congress passed the West Valley Demonstration Project Act (U.S. Congress, 1980), directing the U.S. Department of Energy (DOE) to carry out a high-level waste (HLW) management demonstration project at the site (without taking title to the facilities or the wastes) to demonstrate solidification techniques for preparing the HLW for disposal. Through a contractual agreement with New York State, DOE operates the Project in conjunction with the New York State Energy Research and Development Authority (NYSERDA). DOE and NYSERDA contracted with Westinghouse Electric Company to manage the Project through a wholly-owned Westinghouse subsidiary, West Valley Nuclear Services Company (WVNS). Westinghouse Electric Company was acquired in March 1999 by Morrison Knudsen Corporation and BNFL Inc. (a U.S. subsidiary of British Nuclear Fuels).

The current inventory of the FRS fuel storage pool includes 40 Pressurized Water Reactor (PWR) SNF assemblies from the Robert E. Ginna (REG) nuclear power plant and 85 Boiling Water Reactor (BWR) SNF assemblies from the Big Rock Point (BRP) nuclear power plant. Through a 1984 contractual agreement between NFS and the Department of Energy, DOE assumed title of these assemblies for use in a shipping and storage demonstration program. In support of these programs, NRC-licensed shipping casks specific to each fuel type have been fabricated for shipment of the remaining 125 SNF assemblies. Additional SNF assemblies will not be received or stored at the FRS facility.

This SAR documents the safety assessment of the storage and handling of SNF in the FRS facility and was prepared to meet the requirements of Department of Energy Order 5480.23, *Nuclear Safety Analysis Reports*, and WVNS Policy and Procedure WV-365, *Preparation of WVDP Safety Documents*. Introductory information relating to the WVDP

Act and ancillary tasks and supporting activities that must be accomplished in fulfillment of the Act is discussed in Section A.1.0 of WVNS-SAR-001, *Project Overview and General Information.*

1.2 Fuel Receiving and Storage Facility Description

The WVDP site, shown in Figure 5.1-1, occupies approximately 220 acres of chain-link fenced area within the approximately 3,345 acre reservation that constitutes the Western New York Nuclear Service Center (WNYNSC), located about 55 km (35 mi) south of Buffalo, New York, in rural Cattaraugus County. The communities of West Valley, Riceville, Ashford Hollow and the village of Springville are located within 8 km (5 mi) of the plant. Several roads and one railway pass through the WNYNSC, but no human habitation is permitted on the WNYNSC.

The FRS facility is comprised of three primary areas: the FRS Building (commonly referred to simply as the FRS), the Radwaste Process Building (also referred to as the Hittman Building), and the Recirculation Ventilation Building. The FRS Building and associated structures are located on the north plateau of the WVDP, adjacent to the Main Plant, as shown in Figure 5.1-1. The FRS Building, which serves as a weather structure for the Fuel Storage Pool (FSP), Cask Unloading Pool (CUP), and associated fuel and cask handling equipment, is located on the east side of the Main Plant. The Radwaste Process Building houses the shielded containers that provide temporary storage for the loaded ion exchange resin that is discharged from the fuel pool demineralizer unit. The Recirculation Ventilation Building houses components of the Recirculation Ventilation System, which provides the heating, ventilation and air conditioning (HVAC) for the FRS Building. A small building on the south side of the FRS facility serves as a change room and office area for operations personnel.

1.3 Activities Description

The FRS facility houses equipment for the handling, storage, and shipment of SNF assemblies. This equipment includes cranes and hoists to move canisters, SNF assemblies, and shipping casks, as necessary, within the FRS Building. SNF assemblies are stored in canisters on storage racks in the FSP.

This SAR has been written to assess the impacts of normal operations and accident conditions associated with (1) the storage of SNF, (2) SNF handling operations, including those required to load the assemblies into a cask, and (3) preparation of a loaded cask for shipment. Other safety analyses assess the safety of casks to be used for the off-site transport of the SNF assemblies. Nuclear Regulatory Commission (NRC) -approved safety analysis reports for the SNF shipping casks (WVDP-228 and WVDP-229) have been prepared to document the analyses and tests that demonstrate that the casks comply with applicable requirements of Title 10, Code of Federal Regulations, Part 71, *Packaging of Radioactive Material for Transport.* Many of the activities associated with cask preparation for transport are discussed in Section 6.2.1 of this SAR as well as in WVDP-228 and WVDP-229.

DOE-approved safety documentation for the receipt, and subsequent handling and storage of fuel from the WVDP has been developed at the Test Area North of the Idaho National Environmental and Engineering Laboratory. Volume I of DOE/EIS-0203-F, *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement*, provides "(a) an introduction and overview to DOE's spent nuclear fuel management program throughout the nation, (b) the purpose and need for action to manage spent nuclear fuel, (c) management alternatives that are under consideration, (d) the affected environment, and (e) potential environmental consequences that may be caused by the implementation of each alternative." Volume I addresses "impacts to worker safety, public health, the environment, and socioeconomic factors related to transporting, receiving, stabilizing, and storing DOE and naval spent nuclear fuel, as well as special-case commercial fuels under DOE responsibility."

On May 30, 1995, DOE issued a Record of Decision for the subject EIS. The Record of Decision states that non-aluminum clad fuels, such as that stored at the WVDP, will be transferred to the Idaho National Environmental and Engineering Laboratory.

Significant activities within the scope of this SAR include:

- Custodial oversight for the storage of the 125 SNF assemblies currently in storage in the fuel storage pool;

- SNF assembly handling;

- Operations required to: 1) inspect and prepare the SNF assemblies for loading into the casks, 2)load the SNF assemblies into casks, 3) prepare the casks for transport from the WVDP (in conjunction with the associated cask SAR), 4) stage the loaded casks for off-site transport;

- Operation of the fuel pool Submerged Water Filtration System; and

- Filtration system waste handling, including High Integrity Container (HIC) transfer operations.

Fuel storage and handling operations are described in Sections 6.2 of this SAR. Fuel pool filtration system and HIC transfer operations are described in Section 5.3. Operational interfaces between activities within the scope of this safety analysis report and those within the scope of the NRC-approved shipping cask SARs are indicated in Section 6.2. Criticality concerns associated with cask loading have been evaluated in the shipping cask SARs. Additional issues associated with the prevention of an inadvertent criticality are summarized in Section 8.7 of this SAR.

Activities in support of the WVDP that are unrelated to fuel storage and handling are also conducted in the FRS building. These activities, which include handling of low activity ion exchange resins and sampling or inspection of low activity waste packages from the WVDP Lag Storage Facility, involve only minor amounts of radioactive or hazardous materials. The safety of these activities is ensured through compliance with administrative controls specified in WVDP-010, *WVDP Radiological Controls Manual*, and WVDP-011, *WVNS Industrial Hygiene and Safety Manual*.

1.4 Identification of Agents and Contractors

Section A.1.4 of WVNS-SAR-001, *Project Overview and General Information*, identifies the agents and contractors responsible for implementing the WVDP. The relationships among WVNS and its agents and contractors are illustrated in Figure A.1.4-1 of WVNS-SAR-001.

1.5 Hazard Categorization

DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*, provides a uniform methodology for determining a facility's hazard category. As stated in DOE-STD-1027-92, the hazard category is determined by consideration of the total inventory of radioactive material in a given facility and the consequences of an unmitigated release, and the potential for an inadvertent criticality. Using the hazard category criteria given in the Standard it has been determined that the FRS facility represents a category 2 hazard based on the fact that the FRS contains greater than 1,000 Ci of mixed fission products.

1.6 Structure of the Safety Analysis Report

The Department of Energy employs safety analyses of its nuclear facilities as the principal safety basis for decisions to authorize the design, construction, or operation of these facilities. In support of the development of consistent safety documentation throughout the DOE complex, the Department has issued DOE Order 5480.23, *Nuclear Safety Analysis Reports*, to provide the requirements for the development of safety analyses that establish and evaluate the adequacy of the safety bases of the facilities.

This Safety Analysis Report has therefore been developed to the requirements of Order 5480.23. Specifically, this SAR has been written to the guidance provided in DOE Standard DOE-STD-3009-94 which was developed by the Department to provide more detailed direction and thereby assist contractors in providing analyses consistent with the intent of the Order. Because the Order does not require a specific format for nuclear safety analysis reports, the format of this SAR corresponds to the format set forth in NRC Regulatory Guide 3.26, *Standard Format and Content of Safety Analysis Reports for Fuel Reprocessing Plants*. A listing of DOE Order 5480.23 topics and the chapter(s) in this SAR that address those topics is provided in Table 1.6-1.

Detailed documentation of site characteristics and Project administrative programs is given in WVNS-SAR-001, *Project Overview and General Information*.

REFERENCES FOR CHAPTER 1

Nuclear Fuel Services, Inc. 1964. *Final Safety Analysis Report: Spent Fuel Reprocessing Plant*. Nuclear Regulatory Commission Docket 50-201.

U.S. Congress. October 1, 1980. *An Act to Authorize the Department of Energy to Carry Out a High-Level Liquid Nuclear Waste Management Demonstration Project at the Western New York Nuclear Service Center in West Valley, New York*. Public Law 96-368 [S.2443]. Congressional Record, Vol. 126.

U.S. Department of Energy. September, 1997. DOE Standard DOE-STD-1027-92, Change 1: *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*. Washington, D.C.: U.S. Department of Energy.

__________. July, 1994. DOE Standard DOE-STD-3009-94: *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*. Washington, D.C.: U.S. Department of Energy.

__________. April 30, 1992. Change 1 (March 10, 1994.) DOE Order 5480.23: *Nuclear Safety Analysis Reports*. Washington, D.C.: U.S. Department of Energy.

__________. April 1995. DOE/EIS-0203-F: *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement*. Washington, D.C.: U.S. Department of Energy.

__________. *Packaging of Radioactive Material for Transport*, 10 CFR 71.

U.S. Nuclear Regulatory Commission. February 1975. *Standard Format and Content of Safety Analysis Reports for Fuel Reprocessing Plants*. Regulatory Guide 3.26.

West Valley Nuclear Services Co., Inc. WV-365: *Preparation of WVDP Safety Documents*. (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WVDP-010: *WVDP Radiological Controls Manual* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WVDP-011: *Industrial Hygiene and Safety Manual* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WVDP-228. *TN-BRP Spent Fuel Package Safety Analysis Report for Transport*. (Latest Revision.) Transnuclear, Inc.

REFERENCES FOR CHAPTER 1 (Concluded)

__________. WVDP-229. *TN-REG Spent Fuel Package Safety Analysis Report for Transport.* (Latest Revision.) Transnuclear, Inc.

__________. Safety Analysis Report WVNS-SAR-001: *Project Overview and General Information.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

TABLE 1.6-1

LOCATION OF DOE ORDER 5480.23 REQUIRED INFORMATION

DOE 5480.23 - 8.b(3) Topics	THIS SAR (Reg. Guide 3.26 Chapters)
a) Executive Summary	1.0 Introduction and General Description of the FRS Facility 2.0 Summary Safety Analysis
b) Applicable statutes, rules, regulations, and departmental orders	Each Chapter, as appropriate
c) Site characteristics	3.0 Site Characteristics
d) Facility description and operation, including design of principal structures, components, all systems, engineering-safety features, and processes	4.0 Principal Design Criteria 5.0 Facility Design 6.0 Process Systems
e) Hazard analysis and classification of the facility	1.0 Introduction 9.0 Hazard and Accident Analysis
f) Principal health & safety criteria	8.0 Hazards Protection
g) Radioactive and hazardous material waste management	7.0 Waste Confinement and Management
h) Inadvertent criticality protection	8.0 Hazards Protection
I) Radiation protection	8.0 Hazards Protection
j) Hazardous material protection	8.0 Hazards Protection
k) Analysis of normal, abnormal, and accident conditions, including design basis accidents, assessment of risks, consideration of natural and man-made external events, assessment of contributory and casual events, mechanisms, and phenomena, and evaluation of the need for an analysis of beyond-design-basis accidents; however, the SAR is to exclude acts of sabotage and other malevolent acts since these actions are covered under security protection of the facility.	9.0 Hazard and Accident Analysis
l) Management, organization, and institutional safety provisions	10.0 Conduct of Operations
m) Procedures and training	10.0 Conduct of Operations
n) Human factors	Each Chapter, as appropriate
o) Initial testing, in-service surveillance, and maintenance	10.0 Conduct of Operations
p) Derivation of TSRs	11.0 Derivation of Technical Safety Requirements
q) Operational Safety	10.0 Conduct of Operations
r) Quality Assurance	12.0 Quality Assurance
s) Emergency Preparedness	10.0 Conduct of Operations
t) Provisions for decontamination and decommissioning	10.0 Conduct of Operations
u) Applicable facility design codes and standards	Each Chapter, as appropriate

2.0 SUMMARY SAFETY ANALYSIS

A summary of the safety analyses performed for the Fuel Receiving and Storage (FRS) facility is presented in this chapter. In all of the accidents analyzed in this Safety Analysis Report (SAR), no credit was taken for any preventive or mitigative design features to reduce the risk of accidents analyzed to an acceptable level. All consequences from accidents analyzed are well below the evaluation guidelines specified in Section 9.1.3. Doses from routine operations are well below the limits established in Title 10, Code of Federal Regulations, Part 835. Additional details on these analyses and supporting systems analyses can be found in Chapters 8 and 9 of this SAR. The evaluation guidelines for radiological accidents are given in Figures 9.1-2 and 9.1-3. For the purposes of evaluating potential unreviewed safety questions, these risks represent the authorization basis risks for activities conducted in the FRS facility.

2.1 Site Analysis

2.1.1 Natural Phenomena

Severe natural phenomena considerations in facility design at the West Valley Demonstration Project (WVDP) include tornadoes, tornado-generated missiles, earthquakes, and snow loadings. Information regarding natural phenomena that can affect the safety of operations at the WVDP is provided in Chapter 3 of WVNS-SAR-001. Characteristics of design basis wind, tornadoes, flood, tornado-generated missiles, earthquakes, and snow loadings at the WVDP are provided in Sections A.4.2.1 through A.4.2.6 of WVNS-SAR-001.

2.1.2 Site Characteristics Affecting the Safety Analysis

This SAR assesses the hazards associated with the FRS facility. Activities in the FRS facility involve the handling, storage, and shipment of spent nuclear fuel (SNF) assemblies. Chapter 4 of WVNS-SAR-001 provides a discussion of principal engineering design criteria and design bases for severe natural phenomena (e.g., earthquakes, tornadoes, high straight winds, floods, and snow loading) for WVDP structures, systems, and components (SSCs). Chapter 9 of this SAR evaluates the accident-related risk due to severe natural phenomena. The results show that in the event of a beyond evaluation basis earthquake that leads to the failure of all 125 SNF assemblies, only minor quantities of radioactive materials are released to the environment. Analyses indicate that such an earthquake would need to be of a magnitude greater than the currently specified WVDP design basis earthquake of 0.1g. Table 4.2-2 of this SAR provides a summary of fuel pool seismic studies.

The recurrence interval for the WVDP design basis tornado (DBT) has been documented in WVNS-SAR-001 as being one million years, thereby placing the frequency of occurrence of the DBT in the "incredible" range. Consequently, only the impacts of tornadoes with a greater recurrence frequency (and therefore less severe characteristics) have been evaluated in Chapter 9.

Other site-specific loads (e.g., high straight winds and snow loading) are bounded by more controlling loads and their associated margins of safety. The FRS facility is located at an elevation well above potential flooding. The site's topographic setting renders the likelihood of major flooding not credible, and local run-off and flooding is adequately accommodated by natural and man-made drainage systems in and around the WVDP.

2.1.3 Effect of Nearby Industrial, Transportation and Military Facilities

Nearby industrial, transportation, and military facilities are not considered to pose significant risks to WVDP activities due to the distance of these facilities from the site and the nature of the operations at these facilities. See Section A.2.1.3 of WVNS-SAR-001 for a further discussion of this topic.

2.2 Radiological Impact of Normal Operations

Both on-site and off-site dose assessments have been performed in Chapter 8 of this SAR to determine the radiological impact of normal operations. Occupational exposures are minimized at the WVDP through strict adherence to as low as reasonably achievable (ALARA) principles. Estimated annual FRS worker occupational exposure is below federally allowed dose limits and WVNS administrative control dose limits. See Section 8.4 for additional information.

As discussed in Section 5.4.1, air ventilated from the cask decontamination stall and former water treatment area is combined with air ventilated through exhaust blower 1K-1 and is discharged to the environment through the Main Plant stack. Liquids that are generated by FRS facility operations are processed through the site Low-Level Waste Treatment Replacement Facility (LLW2) before discharge to the environment. Hence, an airborne pathway and liquid pathway must be considered in calculating off-site doses. Section 8.6 states that the total effective dose equivalent (TEDE) to the maximally exposed off-site individual in 1998 was calculated to be 3.4E-02 mrem/yr (3.4E-07 Sv/yr) for airborne discharges from all stacks evaluated (which includes the Main Plant stack). In regard to liquid releases from the LLW2, Section 8.6 states that the dose to the maximally exposed off-site individual for liquid discharges from all WVDP sources in 1998 was approximately 7.4E-03 mrem/yr (7.4E-08 Sv/yr). Releases from the FRS facility are fractional contributors to these doses.

2.3 Impacts From Abnormal Operations

Abnormal operations are events that could occur as a result of malfunctions in facility systems or as a result of operator error. Abnormal events are only of consequence when they affect systems or components that process, control, or confine radioactive or hazardous materials. Abnormal events considered in this analysis have a potential for a range of consequences; however, they are not expected to present a significant risk. Qualitative risk estimates from abnormal operations of the FRS facility are provided in the Process Hazards Analysis (PHA) found in Table 9.1-1.

2.4 Accidents

Doses to an individual result from exposure to radioactive material. The FRS facility contains sources of radioactivity that have the potential for incurring doses to both on-site and off-site individuals. These sources include the 40 pressurized water reactor (PWR) SNF assemblies and the 85 boiling water reactor (BWR) SNF assemblies in storage in the Fuel Storage Pool (FSP), and the loaded ion exchange resin and used filter cartridges from the fuel pool Submerged Water Filtration System.

Four accidents associated with operation of the FRS facility have been analyzed in Section 9.2. Bounding accident evaluations described in Section 9.2.2 calculate the maximum credible consequences for operational accidents involving the significant sources of hazards in the FRS facility (i.e., SNF and water filtration system wastes). The first accident considers the effect of dropping a fuel assembly or fuel-loaded canister in the Cask Unloading Pool (CUP). This accident would result in an off-site TEDE to the maximally exposed individual of 6.1 mrem (6.1E-05 Sv/yr), as described in Section 9.2.2.1. Analyses in section 9.2.2.2 examine the consequences of dropping a high integrity container (HIC) loaded with contaminated ion exchange resin, resulting in catastrophic failure of the HIC. This accident would result in an off-site dose to the maximally exposed individual of 2.7 mrem (2.7E-05 Sv/yr). The consequences of an inadvertent criticality in the CUP have been evaluated in Section 9.2.2.3. This analysis determines that the dose to the maximally exposed off-site individual following a criticality involving 8 spent nuclear fuel assemblies would be 353 mrem (3.53E-03 Sv/yr).

An analysis in Section 9.2.3.1 determines the on-site and off-site consequences of releases from the FRS following a beyond evaluation basis seismic event. The analysis assumes that a beyond evaluation basis earthquake leads to damage of (and hence radioisotope release from) all 125 SNF assemblies. This natural phenomena-initiated event would result in an off-site TEDE of 768 mrem (7.68E-03 Sv/yr).

FRS facility systems and operational activities do not require the use of bulk quantities of hazardous chemicals. Small quantities of some reagents and cleaning solutions may be used periodically for various cleaning, analytical chemistry, or maintenance activities. Hence, the risk from accidents involving hazardous chemicals has not been evaluated in Chapter 9.

2.5 Conclusions

A summary of the radiological consequence assessments performed in this SAR is provided in Table 2.5-1. Consequences were determined at distances of 640 m and 1050 m. These distances correspond to the location of the on-site evaluation point and the location of the site boundary for the sector exhibiting the highest radiological material concentration, respectively. The consequences of all accidents analyzed are within the evaluation guidelines provided in Section 9.1.3.

The failure of all 125 SNF assemblies results in a TEDE to the maximally exposed off-site individual of 768 mrem. This represents the bounding accident for radiological releases. Calculated doses to off-site persons were determined for both normal and accident conditions. Routine doses to off-site individuals are well within the requirements of DOE Order 5400.5.

REFERENCES FOR CHAPTER 2

U.S. Department of Energy. February 8, 1990. Change 2 (January 7, 1993). DOE Order 5400.5: *Radiation Protection of the Public and the Environment*. Washington, D.C.: U.S. Department of Energy

__________. *Occupational Radiation Protection*, 10 CFR 835.

West Valley Nuclear Services Co., Inc. Safety Analysis Report WVNS-SAR-001: *Project Overview and General Information*. (Latest Revision.) West Valley Nuclear Services Co., Inc.

TABLE 2.5-1

SUMMARY OF FRS ACCIDENT CONSEQUENCES

Accident Scenario	Maximum Off-Site Dose (rem)	Maximum On-Site Dose (rem)	Evaluation Guideline Level
Dropping of a Fuel Assembly in the FRS	6.14E-03	1.49E-02	On-site - 25 rem Off-site - 5 rem
Dropping of a Loaded High Integrity Container	2.97E-03	6.27E-03	On-site - 100 rem Off-site - 25 rem
Inadvertent Criticality in the FRS	3.53E-01	8.57E-01	On-site - 100 rem Off-site - 25 rem
Failure of 125 SNF Assemblies Due to Seismic Event	7.68E-01	1.86E+00	On-site - Natural Phenomena, N/A Off-site - 25 rem

3.0 SITE CHARACTERISTICS

Site characteristics associated with the West Valley Demonstration Project (WVDP) are provided in WVNS-SAR-001, Project Overview and General Information, Section A.3.0.

3.1 Geography and Demography of WVDP Environs

WVNS-SAR-001, Section A.3.1, contains a comprehensive description of the geographic and demographic features of the WVDP and surrounding areas. Neither geography or demography affected the original design or current operation of the FRS.

3.2 Nearby Industrial, Transportation, and Military Facilities

A detailed discussion of the effects on the WVDP from these nearby sources is provided in WVNS-SAR-001, Section A.3.2. There are no direct effects on the FRS from these facilities.

3.3 Meteorology

Section A.3.3 of WVNS-SAR-001 provides information regarding meteorological conditions at the WVDP. The impacts of severe natural phenomena on the FRS facility are addressed in Chapter 9.

3.4 Surface Hydrology

Section A.3.4 of WVNS-SAR-001 provides a general discussion of the surface hydrological conditions at the WVDP. Specific surface hydrological conditions were not found to affect the conclusions of the analyses provided in Chapter 9.

3.5 Subsurface Hydrology

Section A.3.5 of WVNS-SAR-001 provides a general discussion of the subsurface hydrological conditions at the WVDP. Specific subsurface hydrological conditions were not found to affect the conclusions of the analyses provided in Chapter 9.

3.6 Geology and Seismology

The geology underlying the fuel pool consists of two basic layers. The top layer, composed of relatively loose sand and gravel, extends from the ground surface to roughly 25 feet below grade. Underlying the sand and gravel, there exists a variously pebbly, silty clay till layer (Lavery till) ranging in thickness from 40 to 110 ft. The silty clay till is typically dense, compact, and moist, and is of low permeability.

Prior to Main Plant/FRS facility construction, soil investigations were conducted by Dames & Moore (Dames & Moore, May 8, 1963) to determine the general soil conditions at the site and to obtain soil data directly relevant to foundation design and construction. Based on their analysis of soil borings taken at the site, Dames & Moore recommended that the main process area be pile supported. Piles selected for foundation support by Bechtel were 12-BP-53 steel H-piles driven into the compact glacial till soil stratum which underlies the site and consists of a mixture of sand, gravel, silt and clay. In all, 476 piles were driven to elevations between 32 and 42 feet, (Plant Datum) beneath the Main Plant complex. Elevation 100 feet, Plant Datum, corresponds to the northwest corner of the Chemical Process Cell foundation and is approximately ground level). Pile load tests and pile driving criteria developed by Dames & Moore are summarized in a report to Bechtel Corporation (Dames & Moore, July 19, 1963).

Section A.3.6 of WVNS-SAR-001 provides a complete discussion of the geology and seismology of the WVDP. The risks from certain severe natural phenomena to the FRS facility are assessed in the analyses presented in Chapter 9.

REFERENCES FOR CHAPTER 3

Dames & Moore. May 8, 1963. *Site Investigation: Proposed Spent Nuclear Fuel Reprocessing Plant Near Springville, New York For Nuclear Fuel Services, Inc.*

__________. July 19, 1963. *Report of Consultation: Pile Load Tests and Pile Driving Criteria Spent Nuclear Fuel Reprocessing Plant Springville, New York Nuclear Fuel Services, Inc.*

__________. Safety Analysis Report WVNS-SAR-001: *Project Overview and General Information.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

4.0 PRINCIPAL DESIGN CRITERIA

Current design-related facility safety requirements pertaining to nuclear safety design, criticality safety, fire protection, and natural phenomena hazards mitigation for non-reactor nuclear facilities are specified in Department of Energy (DOE) Order 420.1 (U.S. Department of Energy, October 13, 1995). A number of the West Valley Demonstration Project (WVDP) facilities, including the Fuel Receiving and Storage (FRS) facility, pre-date the Project and DOE's presence at the site. The FRS facility was constructed by Nuclear Fuel Services (NFS) according to United States Atomic Energy Commission (AEC) license CSF-1 and design criteria in effect at the time. These criteria, which were originally documented in the AEC-approved Final Safety Analysis Report (FSAR) for the fuel reprocessing plant (NFS, 1964), are given in Table 4.2-1. OH/WVDP has concurred that in some cases pre-existing facilities, such as the FRS facility, do not meet all of the current design criteria, but are nonetheless judged to meet the Project's current needs (Bixby, W.W., July 17, 1989). Significant additions or modifications to the FRS facility since 1989, such as the fuel pool Submerged Water Filtration System, comply with requirements of DOE Order 6430.1A (U.S. Department of Energy, April 6, 1989) and the references contained therein. DOE Orders 420.1 and 430.1A (U.S. Department of Energy, October 14, 1998) effectively cancel DOE Order 6430.1A, though DOE Order 6430.1A is not currently an "archived" Order.

4.1 Purpose of the Fuel Receiving and Storage Facility

The FRS facility is designed for the storage and handling of spent nuclear fuel (SNF) assemblies. Fuel in the FRS is stored in canisters on racks in the fuel storage pool located in the FRS Building. Currently 85 boiling water reactor (BWR) and 40 pressurized water reactor (PWR) fuel assemblies are in storage in a critically safe array in the fuel storage pool. Handling equipment and a rail line into the building allow for either over-the-road or rail shipment of SNF assemblies.

4.1.1 Fuel Characteristics

Fuel currently in the FRS has been in storage for over 25 years. Consequently, a significant degree of post reactor decay of the fuel has occurred. Physical and radiological characteristics of the SNF stored in the pool are provided in Tables 4.1-1 through 4.1-3.

4.1.2 Fuel Receiving and Storage Facility Products and By-Products

The primary function of the FRS is the custodial storage of the remaining 125 spent nuclear fuel assemblies at the site and therefore there are no products from this facility. By-products of FRS operations, including loaded pool filter elements and ion exchange resins, and miscellaneous liquid and solid wastes, result from operation of the fuel storage pool skimmer and Submerged Water Filtration System and normal maintenance and operations activities. Handling and storage of wastes generated in the FRS is discussed in Chapter 7 of this SAR.

4.1.3 Facility Functions

The FRS facility was designed for the storage and handling of SNF assemblies. Pool water quality is maintained by the fuel pool Submerged Water Filtration System, while FRS Building air quality and temperature is maintained primarily by the Recirculation Ventilation System. The initial capacity of the fuel storage pool was 882 assemblies. SNF is stored in canisters arranged on storage racks. The original pool configuration consisted of 42 racks arranged in a north-south orientation, each rack having a capacity of 21 canisters. Facility modification activities in 1987 resulted in the removal of 31 of these storage racks, thereby leaving the 11 racks that presently exist. Currently, 125 SNF assemblies are stored in the pool. Table 4.1-5 presents the location of fuel elements and fuel canisters as of March 2000. However, any alternative arrangement of the fuel canisters (with or without a SNF assembly) properly positioned on the storage racks is permitted. The eleven racks that remain are numbered 32 through 42. Each rack, or row, provides 21 storage locations, designated A through X (letters I, O, and Q are not used).

The FRS facility does not process or otherwise utilize the existing stored nuclear material and no additional receipts of fuel at the WVDP are planned.

4.2 Structural and Mechanical Safety Criteria

The FRS Building was constructed as part of the original NFS facility, which was designed for NFS by Bechtel in 1963 as a conventional chemical process plant to conventional seismic standards. Structural design codes referenced by the Bechtel design specifications are given in Table 4.2-1.

Specific FRS design criteria have not been relied upon in Section 9.2 in demonstrating that the consequences of all credible, bounding accidents within the FRS are below the evaluation guidelines specified in Section 9.1.3.

4.2.1 Wind Loadings

See Section A.4.2.1 of WVNS-SAR-001 for a discussion of the characteristics of the design basis wind loadings in place at the WVDP.

4.2.2 Tornado Loadings

Following the decision to cease reprocessing of SNF, several tornado and wind studies were performed for the NRC to determine a design basis magnitude for a tornado at the NFS site. Lawrence Livermore National Laboratory (LLNL) sponsored these studies as part of a larger DOE-funded study. The LLNL study (Fujita, T.T., 1981) reviews fastest mile-per-hour wind probabilities for the West Valley site. The results of this study were compared to an earlier study that he had performed for the NRC and the work of Simiu, et al. Another study was performed by McDonald (McDonald, J.R., July, 1981) for the same LLNL/DOE program. This study examined both tornado and straight wind probabilities. Characteristics of the WVDP design basis tornado are based on these studies and are given in Section A.4.2.2 of WVNS-SAR-001.

4.2.3 Flood Design

See Section A.4.2.3 of WVNS-SAR-001 for a discussion of flood protection requirements at the WVDP.

4.2.4 Missile Protection

See Section A.4.2.4 of WVNS-SAR-001 for a discussion of characteristics of tornado-induced missiles used at the WVDP.

4.2.5 Seismic Design

The design basis earthquake (DBE) employed at the WVDP has been selected based on probabilistic assessments of earthquake exposure using the graded approach of DOE-STD-1023-95. (See Sections A.3.6.1 and A.4.2.5 of WVNS-SAR-001, and DOE-STD-1020-94 and DOE-STD-1021-93 for related information.) This event corresponds to a peak horizontal ground acceleration of 0.1 g, with a vertical component of two-thirds the horizontal (i.e., 0.067 g) and an annual recurrence frequency of 5E-4. The peak ground acceleration response spectra for the WVDP site, derived using DOE-STD-1024-92 guidelines, results in a peak ground acceleration anchored (highest frequency) at 0.078g. The design basis earthquake peak ground acceleration response spectra, anchored at 0.1g, was developed using the standard Design Response Spectra and associated damping values given in NRC Regulatory Guides 1.60 (USNRC, 1973a) and 1.61 (USNRC, 1973b). This results in a design basis earthquake response spectra that conservatively envelopes the response spectra evaluated using the DOE Standard methodology over the entire range of frequencies.

The current design basis seismic criteria will be applied to new facilities and major modifications to existing facilities at the WVDP. These criteria, however, were not used in the design of the original NFS facilities, including the FRS. At the time of

FRS construction (1964) no specific seismic standards had been established for nuclear fuel reprocessing facilities. In lieu of these standards the facility was designed to meet requirements of Uniform Building Code (UBC) Seismic Zone 3 specifications (International Conference of Building Officials, 1961). The UBC is a static method of analysis appropriate for non-critical facilities.

To assess the seismic safety of the Fuel Receiving and Storage facility, several investigations have been conducted on the structural integrity of the FRS Building, pool, and fuel storage racks by various investigators. These analyses include:

1. Blaw-Knox (1972): the structural integrity of the FRS building and pool were evaluated for tornado and earthquake loading.

2. Lawrence Livermore Laboratory (1978): a structural analysis of the Fuel Storage Pool was performed using a three-dimensional finite element model to study operating loads including thermal gradients, seismic induced loads, and impact load from a dropped fuel cask. This report was reviewed by the NRC (1979 & 1982).

3. SAI (1981): performed a seismic evaluation of the storage racks for the spent fuel cells in the pool.

4. Dames & Moore (1995): seismic evaluation of the FRS/Main Plant masonary wall interface.

Results of these analyses are presented in Table 4.2-2. The analytical results can be compared to an evaluation basis earthquake (EBE) for the FRS, which, for this SAR, has the magnitude and characteristics of the current WVDP DBE identified above (0.1g).

(Note: *Evaluation Basis* is the terminology the Department of Energy uses for describing the conditions against which an existing facility is evaluated when those conditions, such as seismic design, are not adequately described in the original facility design basis. For assessing the safety of current FRS facilities, the characteristics of evaluation basis criteria are the same as current design basis criteria for construction of new facilities at the WVDP. These criteria are specified in Section A.4.2 of WVNS-SAR-001.)

4.2.6 Snow Loading

See Section A.4.2.6 of WVNS-SAR-001 for a discussion of estimated snow loadings used at the WVDP.

4.2.7 Process- and Equipment-Derived Loads

The parameters used to establish process and equipment loads for the FRS are not fully specified in the historical record of the site. Design considerations for new facilities and modifications to existing facilities will include all feasible load combinations, including process- and equipment-derived loads, in accordance with applicable building and design codes.

4.2.8 Combined Load Criteria

Parameters used to establish the combined load design of the FRS are not fully specified in the historical site record. However, as with process- and equipment-derived loads, it may be assumed that conservative values were factored into the original design, based on performance of the systems.

4.2.9 Subsurface Hydrostatic Loadings

Design criteria for subsurface hydrostatic loadings on FRS structures below grade are not specified in the historical site record; however, these loadings have been calculated in evaluation of the structural integrity of the pool with the pool in a drained condition (Dames & Moore, 1995). Section A.4.2.9 of WVNS-SAR-001 provides a discussion of subsurface hydrostatic loadings for developing design criteria for new facilities at the WVDP.

4.2.10 Temperature Design Loadings

The WVDP has a freeze protection program in place to prevent damage to existing equipment and facilities due to cold weather (WVDP-183, *WVDP Freeze Protection Plan*). Requirements for freeze protection are incorporated into new designs. Facilities are equipped with heating systems and the fuel storage area of the FRS Building is insulated to maintain inside temperatures above freezing. The FRS Building foundations and buried utilities are placed below the frost line of 1 m (3 ft).

4.3 Safety Protection Systems

4.3.1 General

The FRS facility has been designed for the safe storage and handling of a wide range of spent nuclear fuels. Criticality control, confinement of radioactive contamination, and control of worker radiation exposure are the primary safety concerns.

4.3.2 Protection Through Defense-in-Depth

The design and operation of FRS facilities provides defense-in-depth for public and worker safety during normal, off-normal, and accident conditions. Implementation of the defense-in-depth philosophy ensures that layers of defense are provided against the release of radiological and hazardous materials such that no one layer by itself is completely relied upon. The primary layers of defense for the FRS are given below in order of relative importance:

- Confinement barriers
- Personnel training
- Administrative planning and controls

Details of FRS facility design and operations are discussed in Chapters 4, 5, and 6 of this SAR, while personnel training and administrative controls are discussed in Chapters 8, 10, 11, and 12. Elements of these design features and administrative controls, as they relate to defense-in-depth, are discussed below.

4.3.2.1 Confinement Barriers

Confinement barriers are provided in the FRS facility to prevent the uncontrolled release of radioactive materials into the environment. The first confinement barrier for the SNF is provided by the fuel cladding. However, it should be noted that the cladding on the majority of both types of fuel stored (100 to 104 of the 125 assemblies) show indications of leaks as determined by at-reactor sipping tests). Control of water quality in the fuel storage pool ensures that degradation of this barrier is minimized. A second barrier to the release of radioactive materials to the environment from the SNF is the shielding water in the storage pool. Radioactive material released from the fuel into the pool would disperse and be handled by the fuel pool Submerged Water Filtration System. A third barrier to the release of radioactive material from the spent fuel is the concrete walls and floor of the fuel storage pool and cask unloading pool. A final barrier is provided by the silty clay till layer (Lavery till) underneath the pool structure which is a mixture of very fine grained heterogeneous clay and silt containing minor amounts of sand and stones. The silty clay till is typically dense, compact, and moist, and is of low permeability.

Confinement barriers for the radioactively contaminated wastes generated in the Submerged Water Filtration System have also been provided. Resin wastes from the treatment system are transferred to a polyethylene high integrity container (HIC) located in the adjacent Radwaste Treatment Building. This container provides the primary barrier to release of this material. Concentric concrete and steel radiation shields around the HIC provide a second barrier to an unplanned release. The

Radwaste Treatment Building has been provided with an 18 cm (7 in) high curb that completely surrounds the perimeter of the pad on which the storage container is located. This berm, and an associated sump, constitute a third confinement barrier. The metal structure of the Radwaste Treatment Building would afford some degree of confinement in the event of an airborne release of radioactive contamination.

Confinement for loaded resin contained in full HICs is provided by the high integrity container itself and by the outer concrete storage containers (Surepaks) which are located in the North FRS yard.

4.3.2.2 Personnel Training

Qualification standards and training requirements are established for all FRS operations positions. Operators are qualified in accordance with documented performance-based training programs. This training includes responsibilities and actions during emergency situations. Periodic emergency drills are performed, with follow-on critiques, to gain experience and confidence and to ensure that personnel are ready to respond to accident situations. Fuel handlers and supervisors are certified in accordance with DOE Order 5480.20A, *Personnel Selection, Qualification, and Training Requirements for DOE Nuclear Facilities.*

4.3.2.3 Administrative Planning and Controls

FRS facility operations are accomplished through a clearly defined organizational structure with well defined responsibilities. Operations are conducted in accordance with a protocol that has been established both procedurally and through training. Operational and maintenance activities are controlled through the use of West Valley Nuclear Services Co. (WVNS) procedures that implement applicable DOE Orders. WVNS systematically integrates safety into management and work practices at all levels so that missions are accomplished while protecting the public, the worker, and the environment. This integration is accomplished by implementing an Integrated Safety Management System (ISMS), which is described in WVDP-310, *WVDP Safety Management System (SMS) Description.* The DOE has developed seven guiding principles to provide the focus for implementing an ISMS. While these principles guide the implementation of an ISMS, five core functions define its make-up. These functions comprise a cycle of activities which, although different in detail, are the same for activities on a program or site level and a facility and work task level.

The WVDP *Industrial Hygiene and Safety Manual* (WVDP-011) establishes the policies used to control chemical and industrial hazards for all West Valley operations. Safety is ensured through facility and equipment design, protective clothing and equipment selection, personnel training, and administrative controls.

The *WVDP Radiological Controls Manual* (WVDP-010) establishes the control organization, staffing and training requirements, performance goals, control zones and associated levels, posting and labeling requirements, and other administrative control requirements associated with work in radiation and contamination areas. Operations within radiologically contaminated areas require the use of work control practices to maintain exposure as low as reasonably achievable (ALARA). These practices include the use of radiation work permits, pre-job briefings, personnel protective equipment and clothing, and dosimetry.

The WVDP uses Process Safety Requirements (PSRs) to reduce worker risk and focus attention on those systems under the direct control of the operator that are important to the safe operation of FRS activities. These requirements, found in WVDP-218, *Process Safety Requirements*, define limiting conditions for operation, surveillance requirements, and actions, and provide the associated bases for systems and/or components under the direct control of the operator. Process Safety Requirements are identified per the OH/WVDP approved radiological, nonradiological, and worker risk-reduction criteria defined in WV-365, *Preparation of WVDP Safety Documents*, and are implemented through procedures. Procedure WV-365 specifies the approval authority for a PSR, which may be WVNS or OH/WVDP, depending upon the criterion or criteria that necessitated the requirement.

4.3.3 Protection by Equipment and Instrumentation Selection

Procurement of new equipment and instrumentation for operation of the FRS facility is done in compliance with WVNS's Quality Assurance Program which is described in Chapter 12 of WVNS-SAR-001. Existing equipment and instrumentation is subjected to inspection and testing commensurate with its intended use. Safety Class and Quality Level designations of the individual components of the FRS facility, Main Plant, and support facilities are given in WVDP-204, *WVDP Quality List (Q-List)*. See Section 5.4.8 of this SAR for a discussion of site-wide and FRS facility-specific communications and alarms.

4.3.4 Nuclear Criticality Safety

The potential for a nuclear criticality accident exists in the handling and storage of SNF. Hence, several features were incorporated into the FRS facility design, and several operational practices are followed to prevent an inadvertent criticality. Equipment used in the transfer and handling of fuel is provided with stops and limit switches to prevent unsafe conditions. Fuel is maintained in subcritical geometries through storage in rack-mounted canisters provided with spacers that ensure adequate spacing between adjacent assemblies. Administrative controls regarding the movement and storage location of SNF assemblies also exist to prevent an inadvertent criticality.

A detailed discussion of the engineered features and administrative controls that ensure criticality safety during storage and handling operations in the FRS is contained in Section 8.7 of this SAR.

4.3.5 Radiological Protection

Operations and maintenance activities at the WVDP are performed in accordance with WVDP-010, *Radiological Controls Manual*, which is based on the requirements of Title 10, Code of Federal Regulations, Part 835. Shielding (walls, windows, water, etc.), confinement and containment structures as well as administrative controls (e.g., procedures, training, etc.) are used as necessary to maintain radiation doses to occupationally exposed personnel ALARA. Personnel protective equipment (e.g., anti-C's and respiratory protection) is worn when required by radiological conditions, as prescribed by WVDP-010. In addition, system decontamination and flushing may be performed when contact maintenance is required.

4.3.5.1 Access Control

Area access in the FRS facility is dictated by the requirements of the *WVDP Radiological Controls Manual* (WVDP-010) and 10 CFR 835. The FRS is posted as a radiological buffer area. Access to the FRS is provided to authorized individuals through key card portal control.

4.3.5.2 Shielding

SNF assemblies in the fuel storage pool represents the greatest source of radiation in the FRS facility. Engineered features in the FRS have been provided to ensure that a sufficient quantity of water is maintained above the fuel assemblies to reduce surface exposure levels to below 1 mrem/hr above background. Original design requirements for mechanical stops and limit switches on fuel handling equipment ensure that at least 3.4 m (11 ft) of shielding water is maintained by restricting the upward movement of the fuel. Although these physical limits remain in place, the radiation level of the fuel currently in storage is much lower than that of the original design basis fuel due to the significant amount of post reactor decay that the fuel has undergone. Consequently, a lesser amount of shielding water is sufficient to attenuate the radiation levels to achieve the dose rate requirement specified above.

A second source of radiation in the FRS is the waste that is generated during the operation of the fuel pool Submerged Water Filtration System. Loaded pleated-paper filters removed from the pool filter are placed in 208 L (55-gal) drums. Currently no shielding is required, although shielding could be used if needed to ensure that external contact dose rates are below 100 mrem/hr. Loaded resin from the underwater ion exchange unit is sluiced to a HIC in the Radwaste Treatment Building. The estimated contact dose rate for the unshielded HIC is 3 to 15 rem/hour on the side. A 36 cm (14 in) steel-reinforced inner concrete shield and a 5 cm (2 in) outer carbon steel shield surround the HIC to attenuate the radiation from a full HIC to less than 5 mrem/hr in the Radwaste Process Building. A full HIC that has been removed from its shield container in the Radwaste Process Building is transferred to a concrete storage cask (Surepak) with cover and polyethylene liner that is used for storage of a loaded HIC.

4.3.5.3 Radiation Alarm Systems

Continuous air monitors are provided in the FRS facility to detect airborne contamination. During fuel storage, at least one area radiation detector is provided to alert workers to unusual/upset conditions. An area radiation detector that alarms at 20 mrem/hr above background shall be present on the service bridge during fuel handling.

4.3.6 Fire and Explosion Protection

Flammable materials are stored in approved flammable storage lockers in the FRS Building, thereby minimizing the fire potential. The FRS facility does not process substances with an explosive potential.

The FRS facility and supporting facilities have fire suppression systems commensurate with requirements specified in DOE Order 420.1. A fire station, which includes a 3.8 cm (1-1/2 in.) hose connected to the site fire water supply loop, is located in the north operating aisle of the FRS Building. (See Section B.5.3.1 of WVNS-SAR-002 for a discussion of site fire protection water supplies.) ABC-type fire extinguishers are also located throughout the building. Fire protection in the Radwaste Process Building and Recirculation Ventilation Building is provided by fire extinguishers as well.

4.3.7 Radioactive Waste Handling and Storage

Loaded resin and used filter cartridges generated during the operation of the fuel pool Submerged Water Filtration System and skimmer are stored in HICs and 208 L (55-gal) drums, respectively. Filter cartridges and resin are replaced through remote operations. Storage for full HICs is provided in shielded casks in the north FRS yard. Drums containing used filter cartridges are stored in the Lag Storage facility prior to permanent disposal. Liquid wastes are processed at the Low-Level Waste Treatment Replacement Facility (LLW2). The LLW2 is described in Section B.7.5 of WVNS-SAR-002.

4.3.8 Industrial and Chemical Safety

Administrative controls concerning industrial and chemical safety are found in the *WVNS Industrial Hygiene and Safety Manual* (WVDP-011) which is based on DOE Order 440.1A, *Worker Protection Management for DOE Federal and Contractor Employees.* Processes in the FRS facility do not require the use of hazardous chemicals.

4.4 Classification of Structures, Systems, and Components

Systems, structures, or components required to mitigate the off-site consequences of accidents below the evaluation guidelines given in Section 9.1.3 are designated as safety class systems, structures or components. As demonstrated in Section 9.2, no credit has been taken for FRS equipment or facilities in the evaluation of the consequences of facility accidents. All off-site consequences of the evaluated (bounding) accidents are below the evaluation guidelines. The FRS facility therefore contains no systems, structures or components required to be designated as safety class, as defined by DOE 5480.23. In addition, no equipment is required to maintain the on-site doses below the evaluation guideline levels specified in Section 9.1.3 and therefore there are no safety significant systems, structures, or components in the FRS.

Safety Class and Quality Level designations are provided in WVDP-204, *WVDP Quality List (Q-List)*, for the individual components of the FRS facility. Retrofitting of pre-existing equipment to Safety Classes and Quality Levels to meet the requirements of the current Quality Management (QM) Manuals is not required.

4.5 Decommissioning

The FRS facility was designed to facilitate eventual decontamination and decommissioning activities. NFS drained and decontaminated the fuel storage pool in 1973 prior to anticipated facility expansion, and in 1987 approximately 75 percent of the storage racks in the pool were removed and size reduced. Most FRS facility equipment and piping was designed to be remotely flushed, thereby minimizing doses to workers during removal and size reduction activities. Decommissioning activities will be performed in accordance with Department of Energy requirements that are applicable at the time of FRS facility decommissioning.

REFERENCES FOR CHAPTER 4

Bixby, Willis W. July 17, 1989. *DOE Order 6430.1A.* Letter to R.A. Thomas CBL:010:89-0902:89:01 (DW:89:0365).

Blaw-Knox Chemical Plants, Inc. December 12. 1972. *Evaluation of the Fuel Receiving and Storage Structure for Tornado and Earthquake Forces.*

Dames & Moore. July 25, 1995. *Report - Seismic Integrity Review, Fuel Receiving and Storage Facility, West Valley, New York, for West Valley Nuclear Services Company, Inc.* W. E. Gates. Dames & Moore Job Number 10805-967-004.

Fujita, T. Theodore. 1981. *Tornado and High Winds Hazards at Western New York State Nuclear Service Center.* West Valley, New York.

International Conference of Building Officials. 1961 Edition. *Uniform Building Code.*

Kennedy, R.P. et al. June 1990. *Design and Evaluation Guidelines for Department of Energy Facilities Subjected to Natural Phenomena Hazards.* LLNL Report UCRL-15910.

LLL, 1978 Lawrence Livermore Laboratory, *Structural Analyses of the Fuel Receiving Station Pool at the Nuclear Fuel Service Reprocessing Plant, West Valley, New York,* (UCRL-52575), May, 1978.

McDonald, James R. 1981. *Assessment of Tornado and Straight Wind Hazard Probabilities at the Western New York State Nuclear Service Center, West Valley, New York.*

Nuclear Fuel Services, Inc. 1964. *Final Safety Analysis Report: Spent Fuel Reprocessing Plant.* Nuclear Regulatory Commission Docket 50-201.

SAI. December, 1981. *Seismic Resistance Capacity Evaluation of Spent Fuel Storage Racks and Fuel.* SAI Report SAI-148-026. NUREG/CR-2236.

Simiu E., M.J. Cangery, J.L. Filliben. 1979. *Extreme Wind Speeds at 129 Stations in the Continental United States.* Science 118, National Bureau of Standards, pp. 314.

U.S. Nuclear Regulatory Commission. 1973. Regulatory Guide 1.60, *Design Response Spectra for Seismic Design of Nuclear Power Plants.*

__________. 1973. Regulatory Guide 1.61, *Damping Values for Seismic Design of Nuclear Power Plants.*

REFERENCES FOR CHAPTER 4 (Continued)

__________. 1979. L.C. Rouse Letter to NFS, Docket 50-201, May 14.

__________. January, 1982. *Safety Evaluation of the Dormant West Valley Reprocessing Facility*. Docket 50-201.

U.S. Department of Energy. April 30, 1992. Change 1 (March 10, 1994.) DOE Order 5480.23: *Nuclear Safety Analysis Reports*. Washington, D.C.: U.S. Department of Energy.

__________. November 15, 1994. DOE Order 5480.20A: *Personnel Selection, Qualification, and Training Requirements for DOE Nuclear Facilities*. Washington, D.C.: U.S. Department of Energy.

__________. October 13, 1995. Change 2 (October 24, 1996). DOE O 420.1: *Facility Safety*. Washington, D.C.: U.S. Department of Energy.

__________. October 14, 1998. DOE O 430.1A: *Life Cycle Asset Management*. Washington, D.C.: U.S. Department of Energy.

__________. March 27, 1998. DOE O 440.1A: *Worker Protection Management for DOE Federal and Contractor Employees*. Washington, D.C.: U.S. Department of Energy.

__________. April 6, 1989. DOE Order 6430.1A: *General Design Criteria*. Washington, D.C.: U.S. Department of Energy.

__________. July, 1994. DOE Standard DOE-STD-3009-94: *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*. Washington, D.C.: U.S. Department of Energy.

__________. 1994. DOE Standard DOE-STD-1020-94: *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*. Washington, D.C.: U.S. Department of Energy.

__________. 1993. DOE Standard DOE-STD-1021-93: *Natural Phenomena Hazards Performance Categorization Guidelines for Structures, Systems, and Components*. Washington, D.C.: U.S. Department of Energy.

__________. 1992. DOE Standard DOE-STD-1024-92: *Guidelines for Use of Probabilistic Seismic Hazard Curves at Department of Energy Sites*. Washington, D.C.: U.S. Department of Energy.

__________. *Occupational Radiation Protection*, 10 CFR 835.

REFERENCES FOR CHAPTER 4 (Concluded)

West Valley Nuclear Services Co., Inc. WV-365: *Preparation of WVDP Safety Documents.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WV-365: *Preparation of WVDP Safety Documents.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WVDP-010: *WVDP Radiological Controls Manual* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WVDP-011: *Industrial Hygiene and Safety Manual* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WVDP-183: *WVDP Freeze Protection Plan* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WVDP-204: *WVDP Quality List (Q-List)* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WVDP-218: *Process Safety Requirements* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WVDP-310: *WVDP Safety Management System (SMS)Description* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. Safety Analysis Report WVNS-SAR-001: *Project Overview and General Information.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. Safety Analysis Report WVNS-SAR-002: *Safety Analysis Report for Low-Level Waste Processing and Support Activities.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

TABLE 4.1-1
CHARACTERISTICS OF FUEL STORED IN FRS

Fuel Type	Dimension (l x w)[1]	Initial Enrichment (w/o U-235)	Assembly Burn-up average (range) (MWD per MTIHM)	Clad Type	Reactor Discharge Dates (# of assemblies)
REG PWR[2]	161.4" x 7.77"	3.473	10,117 (5,592 to 14,293)	Zr-4	3/71 (12) 5/72 (28)
BRP-B	82.2" x 6.52"	3.089 (avg.) (2.6 to 4.2)	20,218 (20,189 to 20,247)	Zr-2	3/72 (2)
BRP-C	82.2" x 6.52"	3.627 (avg.) (2.9 to 5.2)	24,094 (22,970 to 24,997)	Zr-2	3/72 (4)
BRP-D 7x7	81.8" x 6.52"	2.939 (avg.) (0.22 to 5.6)	1,643 (1,596 to 1,690)	Zr-2	6/68 (4)
BRP-D 8x8	81.8" x 6.52"	2.853 (avg.) (0.22 to 5.6)	4,546 (2,065 to 7,027)	Zr-2	6/68 (1) 4/69 (1)
BRP-E[3]	84.2" x 6.52"	2.979 (avg.) (2.35 to 3.55)	11,892 (10,049 to 13,792)	Zr-2	2/71 (5) 3/72 (13) 4/73 (12) 5/74 (3)
BRP-EG[4]	84.2" x 6.52"	3.523 (avg.) (2.5 to 4.5)	13,476 (5,502 to 18,362)	Zr-2	3/72 (18)
BRP-F[5]	84.2" x 6.52"	3.523 (avg.) (2.5 to 4.5)	11,163 (6,930 to 15,554)	Zr-2	4/73 (6) 5/74 (9)
BRP-MEG	84.2" x 6.52"	3.513 (avg.) (2.5 to 4.5)	12,446 (9,464 to 13,685)	Zr-2 Zr-3Nb-1Sn	3/72 (1) 4/73 (3)
BRP-EP	84.2" x 6.52"	1.62 - 9.07 w/o Pu[6]	17,859 (16,988 to 19,275)	Zr-2	4/73 (2) 5/74 (1)

1. The width dimension represents the measure of one side of a square array.
2. Seven fuel assemblies with clad damage.
3. One fuel assembly with clad damage.
4. Two of the assemblies each have two mixed oxide rods within them; two of the assemblies each have two rods from F assemblies in them.
5. Two of the assemblies each have two mixed oxide rods within them.
6. BWR EP-type fuel assemblies are mixed oxide assemblies with rods of various w/o Pu, with 90% of the Pu fissile. The Pu was blended with natural uranium.

TABLE 4.1-2
BWR FUEL ASSEMBLY PARAMETERS

BRP ASSEMBLY TYPE	ARRAY	POWDER OR PELLET OUTER DIAMETER (in)	CLAD THICK (in)	ROD OUTER DIAMETER (in)	ROD PITCH (in)	MASS U[1] (kg)	NUMBER OF ASSEMBLIES
B	11x11	0.373/0.275[2]	0.034	0.49/0.344[2]	0.577	132.9	2
C	11x11	0.373/0.275[3]	0.025 0.034	0.449/.0344[3]	0.577	133.1 121.8	2 2
E	9x9	0.471	0.040	0.562	0.707	141.2	33
F	9x9	0.471	0.040	0.562	0.707	141.2	15
D	7x7	0.620/.607[4]	0.040	0.700	0.921	142.8/139.4[4]	4
D	8x8	0.500/.488[5]	0.035	0.570	0.807	122.7/118.4[5]	2
MEG	9x9	0.471	0.040	0.562	0.707	141.2	4
EP	9x9	0.471	0.040	0.562	0.707	118 5.33 (Pu)	3
EG	9x9	0.471	.040	0.562	0.707	141.2	18
						Total	85

NOTES:
1. Calculated values. Do not reflect pellet dishing or variations in pellet density and length.
2. Three rods in each corner.
3. Two rods in each corner
4. Two assemblies contain pellet and powder fuel. (First number is for powder, second number is for pellets.)
5. One assembly contains pellet and powder fuel. (First number is for powder, second number is for pellets.)

TABLE 4.1-3
PWR FUEL ASSEMBLY PHYSICAL CHARACTERISTICS

Number of Assemblies in FRS	40
Rod Array	14 x 14
Rods Per Assembly	179
Instrument/Control Rod Guide Tubes	17
Rod Pitch (in)	0.556
Rod O.D. (in)	0.422
Clad Thickness (in)	0.024
UO_2 Pellet Diameter (in)	0.367
Assembly Length (in)	160.875
Length of Control Rod (in)	161.380
Active Fuel Length (in)	144
Initial Uranium Loading (kg)	382.18
Initial Enrichment (wt% U-235)	3.473
Guide Tubes: Quantity - Material - OD (in) - ID (in) -	 16 stainless steel 0.5375 0.5075
Instrument Tube: OD (in) - ID (in) -	 0.422 0.3455
Burnable Poison Rod: Outer Stainless Steel Tube*: OD (in) - ID (in) -	 0.432 0.393
Burnable Poison Rod: Inner Stainless Steel Tube*: OD (in) - ID (in) -	 0.312 0.268
Control Rod: Stainless Steel Tube**: OD (in) - ID (in) -	 0.431 0.393

*Concentric tubes have annular borosilicate glass (pyrex) between them.
**Contains 80% Ag, 15% Indium, 5% Cadmium.

TABLE 4.1-5
LOCATION OF 125 FUEL ASSEMBLIES (MARCH 2000)[@]

Loc	Row																					
	42		41		40		39		38		37		36		35		34		33		32	
	Assm	Can	PWR Assm	Can	PWR Assm	Can	PWR Assm	Can	BWR Assm	Can	BWR Assm	Can	BWR Assm	Can	BWR Assm	Can	BWR Assm	Can	BWR Assm	Can	Assm	Can
A	----	----	C-32	515	C-10	501	C-30	534	D-50	367	CE-63	350	CE-52	343	B-04	359	CE-50	334	CE-33	335	Empty	No #
B	Debris	454	C-14	514	C-28	527	C-19	540	D-52	370	CE-59	349	CF-18	374	CE-01	326	D-60	480	CE-17	328	Empty	No #
C	Debris	479	C-01	513	C-36	530	C-12	523	CF-24	321	CC-10	355	CF-12	373	CE-10	325			CE-23	327	Empty	Illegible
D	Debris	481	C-07	508	C-27	531	C-34	529	D-55	363	CC-25	365	CF-01	375	CE-16	331			CE-22	323	Empty	No #
E	Debris	625	C-05	509	C-03	537	C-23	539	D-54	366	B-16	360	CE-35	342	CE-42	336			CE-29	330	Empty	Illegible
F	Empty	No #	C-11	506	C-17	521			D-53	368	CC-14	356	CF-06	314	CEP-3	85			CE-11	324	Empty	114
G	LT**	1143	C-24	733	C-16	538			CF-19	371	CC-39	364	Empty	518	CE-56	344			CE-31	329	Empty	No #
H	LT**	1144	C-06	723	C-21	535			CE-71	389	CE-60	361			CF-03	87			CE-03	333	TZ1 *	No #
J			C-08	660	C-18	512			CE-75	384	CE-58	348			CEP-2	322			CE-24	332	Empty	23
K			C-04	505	C-33	528			CE-70	386	CE-57	347			CE-81	388			CE-36	337	Empty	24
L			C-39	504	C-26	526			CE-67	383	CE-53	345			CF-14	316			CE-51	341	Empty	30
M			C-02	503	C-25	511			CE-83	385	CE-73	352			CEP-1	372			CE-32	340	Empty	36
N			C-09	502	C-22	536			CE-85	381	CE-64	351			CE-84	380			CE-41	339	Empty	151
P			C-20	654	C-29	525			CE-66	382	CE-54	346			CE-82	313			CE-37	338	Empty	5
R					C-38	500			D-63	484	CF-42	318			CE-79	315						
S					C-31	519			CF-23	86	CE-87	353			CF-25	33						
T					C-15	520			CF-26	89	CE-77	358			CE-62	483						
U					C-35	664			CF-35	88	CE-86	354			CE-74	378						
V					C-37	517			D-61	482	CE-76	357			CE-61	377						
W					C-40	516			CF-02	376	D-51	369			CF-13	319						
X					C-13	524			D-62	379	CE-69	362			CE-80	317						
No. of Assemblies:			14		21		5		21		21		6		21		2		14		Total: 125	

*Empty storage canister for Thoria Rods
**Load Test Canister

[@]For Information Only (Not subject to USQD Procedure upon fuel movement.)

TABLE 4.2-1

FUEL RECEIVING AND STORAGE FACILITY DESIGN CRITERIA

Structural Design Codes [1]	
NYS	New York State Building Code, 1961 Edition.
UBC	Pacific Coast Building Officials Conference Uniform Building Code, 1961 Edition.
ACI	American Concrete Institute Building Code Requirements for Reinforced Concrete, ACI 318-56.
AWS	American Welding Society Standard Code for Arc and Gas Welding in Building Construction, AWS D1.0-46.
AISC	Specification for Design, Fabrication, and Erection of Structural Steel for Buildings, 1961, 5th Edition.
Design Loads [1]	
Snow Load	40 psf on roof areas
Wind Load	100 mph wind
Earthquake	UBC Zone 3 (1961)
Construction Materials [1]	
Masonry Walls	Concrete block
Concrete Walls [2]	3,000 psi concrete
Reinforcing Steel [2]	60,000 psi (yield strength)
Structural Steel	ASTM A 36
Standard Bolts	ASTM A 307, Grade B

(1) - As referenced by Bechtel in the design specifications for the original reprocessing plant.

(2) - Specific to the FRS.

TABLE 4.2-2

SUMMARY OF FRS SEISMIC ANALYSES RESULTS

	Blaw-Knox 1972[1]	LLL 1978[2]	SAI 1981[3]	Dames & Moore 1995[4]
Earthquake Criteria OBE DBE	 0.06 g 0.12 g	 --- 0.2 g	 --- 0.19	 --- 0.1 g
Tornado Criteria Wind Pressure	 200 mph 1 psi	 --- ---	 --- ---	 160 mph 0.35 psi
Fuel Pool Full Empty	 OK ---	 0.16 g ---	 --- ---	 --- ---
Canister Racks	OK	---	0.19 g	---
FRS/Process Building Interface	---	---	---	0.15 g

1 This analysis was a preliminary evaluation for 0.12g peak ground acceleration seismic loads using simple analysis models.

2 This analysis determined the ground acceleration which would result in structural failure of the fuel storage pool.

3 This analysis determined the ground acceleration which would result in structural failure of the fuel storage racks.

4 This analysis determined the ground acceleration which would result in structural failure of the FRS/Process Building interface.

5.0 FACILITY DESIGN

5.1 Summary Description

5.1.1 Location and Facility Layout

The Fuel Receiving and Storage (FRS) facility is located within the Western New York Nuclear Service Center (WNYNSC). The primary buildings and outdoor structures that comprise the FRS facility, and their relationship to the Main Plant (which is discussed in WVNS-SAR-002) are shown in Figure 5.1-1. The FRS Building is located on the east side of the Main Plant. The Radwaste Process Building (or Hittman Building) houses equipment for the Radwaste Process System, including the shielded containers and support equipment for storing loaded ion exchange resin. The Recirculation Ventilation Building houses major components of the Recirculation Ventilation System, which services the FRS Building. A small building on the south side of the FRS facility serves as a change room and office area for operations personnel.

5.1.2 Principal Features

5.1.2.1 Site Boundary

The boundary of the WNYNSC is shown in Figure 5.1-2. This boundary encompasses approximately 3,345 acres and is irregular in shape. The site encloses the entire downstream portion of Buttermilk Creek to its confluence with Cattaraugus Creek. The perimeter of this entire area is enclosed within a 3-strand barbed wire fence.

5.1.2.2 Property Protection Area

The Property Protection Area is comprised of approximately 220 acres located near the center of the WNYNSC. This area is enclosed by a 2.4 m (8 ft) high chain link fence topped with three strands of barbed wire. Nearly all the Project facilities are located within this area. Access to this area is controlled by the Project security force.

5.1.2.3 Site Utility Supplies and Systems

Site utilities are located and controlled from a Utility Room (UR) adjacent to (and on the south side of) the Main Plant. Electrical feed to the UR is routed overhead from an on-site substation. Water to the site is provided from two man-made on-site reservoirs located approximately 1.3 km (0.8 mi) southwest of the plant. Water from the northernmost reservoir is pumped via a buried 20 cm (8 in) diameter pipe to the UR. The southern reservoir is maintained as a backup to the primary supply. Natural gas is routed to the site via a 15 cm (6 in) high pressure gas line and is regulated and metered at the UR. The sanitary sewage treatment facility, or Waste Water Treatment Facility, is located south of the Main Plant.

5.1.2.4 Principal FRS Features

The FRS facility is comprised of structures and equipment for the storage and shipment of spent nuclear fuel (SNF) and for the maintenance of water quality in the Fuel Storage Pool (FSP) and Cask Unloading Pool (CUP). Principal features include:

- Fuel storage pool
- Cask unloading pool
- Cask loading area
- Cask, canister, and fuel handling equipment
- Cask decontamination stall
- Loaded resin transfer and storage equipment
- Fuel Pool Submerged Water Filtration System
- Water Treatment Area
- Recirculation Ventilation System

5.2 Fuel Receiving and Storage Facility

5.2.1 Structural Specifications

The FRS Building was constructed in accordance with various criteria in effect at the time (e.g., 1961 New York State Building Code, 1956 American Concrete Institute [ACI] Code 318, and the 1961 Uniform Building Code). The FSP was designed to fulfill the seismic requirements of the 1961 Uniform Building Code for Zone 3.

5.2.2 Layout of the Fuel Receiving and Storage Facility

5.2.2.1 Building Plan and Sections

A plan view of the FRS Building and equipment layout within the building are shown in Figures 5.2-1 through 5.2-3. The Radwaste Process Building contains the equipment for dewatering and storing spent ion exchange resin. Radwaste Process Building plan and elevation views are provided in Figure 5.2-4. The Recirculation Ventilation Building is also located in the north FRS yard. This building contains the equipment that provides the majority of the heating, ventilation and air conditioning (HVAC) requirements.

5.2.2.2 Confinement Features

The FRS Building was not designed for radioactive contamination confinement; however, filters in the Recirculation Ventilation System would substantially decrease the amount of airborne radioactive materials in the FRS Building should an upset event lead to elevated levels of airborne contamination. The FRS Building (i.e., walls, floor and roof structure) would also provide some measure of confinement in the event of an airborne release. No evaluation has been made to quantify the degree of confinement provided by the FRS Building and HVAC systems, nor has credit been taken for this confinement in the accident analyses provided in this Safety Analysis Report (SAR).

Key components of the fuel pool Submerged Water Filtration System are located entirely within the FSP. An air operated diaphragm pump used in ion exchanger resin handling is located in a concrete pit (formerly the water treatment area) that is located adjacent to the fuel pool. Any spills associated with the fuel pool Submerged Water Filtration System would be contained in either the FSP or the adjacent concrete pit.

The Radwaste Process Building is equipped with provisions for the confinement of radioactive materials. The foundation perimeter is curbed and a sump located in the southwest corner of the building provides for spill collection. The curb is sufficiently high to provide containment for a volume of liquid equal to 150 percent of the volume of a high integrity container (HIC). Sump contents may be pumped to the site interceptors (a basin used for the collection and batch sampling of plant liquid effluents) via a floor drain in the FRS Building.

5.2.3 Fuel Receiving and Storage Building

The FRS Building, which serves as a weather structure for the primary FRS components, is a steel-framed structure with exterior steel siding. Major components of the FRS Facility located within the FRS Building include the FSP where spent fuel assemblies are stored under 3.4 m (11 ft 6 in) of water that provides cooling and radiation shielding; the CUP where the spent fuel assemblies are loaded under water to the shipping cask from the storage canisters; the Submerged Water Filtration System which provides treatment of the water in the FSP and CUP; and Cask Loading Area which serves as the staging, decontamination and loading area for fuel shipping casks. These areas are discussed in the following sections. A pit area located within the

main pool wall boundaries, adjacent to the CUP and to the northeast corner of the FSP was the location of the original fuel pool water treatment equipment. This equipment is no longer used and is out-of-service. Plan and section drawings of the FRS Building are shown in Figures 5.2-1 through 5.2-3.
Assessments of the FRS Building have been performed to determine the response of the structure under the seismic loads of an evaluation basis earthquake (EBE) of 0.1 g peak ground acceleration. As indicated in section 5.2.3.3.3, it is not expected that the steel-framed weather structure would fail catastrophically under the forces of an earthquake of this magnitude. The interface of the FRS Building and the Main Plant building is a 20 cm (8 in) masonry block wall that is part of the original Main Plant building. This wall, and its capacity to withstand evaluation basis seismic accelerations, was deemed to be a potential vulnerability for the spent fuel assemblies that remain in the Fuel Storage Pool.

In response to this perceived vulnerability, Dames & Moore performed an assessment to evaluate its integrity under EBE loadings (Dames & Moore, 1995). The results of this analysis determined that at earthquake ground accelerations of 0.15 g (1.5 x EBE) small pieces of spalled masonry, as well as a few of the broken blocks adjacent to the FRS roof framing members, may become dislodged and fall to the floor adjacent to the FRS pool. However, it was not anticipated that large sections of the wall would break up and fall. Therefore, based on the analysis, and past engineering experience with similar construction in earthquakes, it was concluded that the masonry wall does not pose a threat to the SNF assemblies stored in the FRS pool for an EBE of 0.1 g peak ground acceleration.

5.2.3.1 Fuel Storage Pool Description

The FSP, shown in Figure 5.2-1, is located at the west end of the FRS Building. The fuel pool is a single walled, unlined, reinforced concrete structure approximately 23 m (75 ft) long by 12 m (40 ft) wide and 8.8 m (29 ft) deep. The concrete floor of the pool is 1 m (3 ft 3 in) thick, and the outside walls are 1.1 m (3 ft 6 in) thick at the base, tapering to 0.45 m (1 ft 6 in) at grade. Above grade, the pool walls are 1.1 m (3 ft 6 in) high and 0.3 m (1 ft) in width.

Fuel storage in the FSP is on aluminum alloy racks. Original construction of the fuel pool provided for 42 storage racks; however, 31 of these racks were removed in 1987, leaving 11 racks to meet current storage requirements. Handling of fuel in the FSP is performed through the use of a canister grapple that is attached to a movable bridge. This fuel pool canister bridge is capable of servicing the entire FSP and has been designed to transport canisters from the FSP to the CUP. The United States Atomic Energy Commission (USAEC) determined that the fuel pool met all applicable safety requirements at the time of construction (1965), and approved the pool for use in the same year.

5.2.3.1.1 Function

The FSP provides storage for the 125 SNF assemblies remaining at the WVDP. Of these 125 assemblies, 40 are pressurized water reactor (PWR) fuel assemblies from the Robert E. Ginna (REG) nuclear power plant while 85 assemblies are from the boiling water reactor (BWR) at the Big Rock Point (BRP) nuclear power plant. Storage of all assemblies in the pool is in aluminum alloy canisters that are placed on storage racks. Canisters with fuel assemblies are oriented in rows on the storage racks under 3.4 m (11 ft 6 in) of water, which provides shielding from the high radiation of the assemblies. Storage of the fuel within canisters on the storage rack ensures that a critically safe array is maintained.

In addition to housing the fuel, the FSP also houses the fuel pool Submerged Water Filtration System. This system is located in the northeast corner of the pool, as illustrated in Figure 5.2-1.

5.2.3.1.2 Components

Fuel Storage Racks

The fuel storage racks, shown in Figure 5.2-5, are an array of aluminum alloy beams and columns bolted to both the north wall and floor of the storage pool, such that there are 11 rows running north-south. Each storage rack has the capacity for 22 canisters; however, only 21 storage spaces per rack are used. Space is provided to store a total of 242 canisters. Since vertical travel of the canisters within the storage pool is limited to 15 cm (6 in), a canister cannot be lifted and moved over the top of the array. There is a 1.2 m (4 ft) wide aisle between the south end of the racks and the pool wall to allow movement of the canisters. When the canisters are placed on the storage racks, there is 3.5 m (11 ft 6 in) of water above the fuel.

The fuel storage racks are constructed of aluminum alloy 6061-T6. The main support is constructed of three extruded beams bolted in six sections. This main support beam is fastened to the north wall of the fuel pool and is also under-supported by six beams that are bolted to the FRS floor. Spacers located between the support beams prevent distortion of the support structure. The spacers are bolted to the under-supports and the north-most spacer is bolted to the north wall. The top extruded beam has projections to allow the canister rings to slide on the rack without rotation after they are set in place. The rack is leveled using leveling plates located under the rack supports. Each fastener bolt is made from the 6061-T6 aluminum alloy and has a yield tension of 2,800 kg/cm^2 (40,000 psi) with a shear strength of 2,100 kg/cm^2 (30,000 psi).

Fuel Canisters

The fuel canisters, shown in Figure 5.2-6, were manufactured using 6061-T6 aluminum alloy and are designed to maintain the fuel stored in the pool subcritical by geometry control of the storage array. The canister support ring is grooved to engage the storage rack and ensure proper placement of the canisters. A lifting ring is provided at the top of each canister for movement of the canister. The racks and canisters in their storage configuration provide at least 51 cm (20 in) center-to-center spacing, 30 cm (12 in) face-to-face spacing and at least 19 cm (7-1/2 in) edge-to-edge between fuel assemblies in the same row. Between rows the spacing between assemblies is 53 cm (21 in) center-to-center, 32 cm (13 in) face-to-face, and at least 21 cm (8-1/2 in) edge-to-edge. The canister lifting lugs are designed to assure positive and correct latching with the grapple prior to movement of a canister. This design assures correct angular orientation for placing fuel canisters into the racks and assures positive latching of the grapple to the canister to prevent dropping canisters during movement.

The aluminum canisters, 51 cm (20 in) maximum outside diameter at base, support, and lifting rings, 32 cm (12-1/2 in) inside diameter, and up to 4.9 m (16 ft 1 in) long, can hold fuel assemblies as large as 4.9 m (16 ft) in length and 21 cm × 21 cm (8-1/4 in × 8-1/4 in) in cross section. Two of the smaller BWR assemblies can be stored in each canister. Assuming one PWR assembly (equivalent to 0.45 MTU) per canister and 11 rows of canisters, the current nominal storage capacity of the pool is 109 MTU.

Fuel Pool Canister Crane

The fuel pool canister crane, which is shown in Figure 5.2-7, has a 1.8 MT (2 ton) capacity, and includes a bridge, a trolley mounted on rails that are attached to the bridge, and a hoist and grapple that are transported north and south by the trolley. The canister crane bridge spans the width of the storage pool and runs east-west on rails mounted on the tops of the north and south pool walls. The bridge can travel the full east-west length of the storage pool and part way over the CUP. The bridge has a working platform and hand rails and may be operated at one of two speeds. Limit switches on the bridge prevent damage to the bridge and associated drive motor by stopping the motor before the bridge hits the fuel pool service bridge or the end of the rails. Limit switches on the trolley stop its drive motor when the trolley approaches either the north or south end of the bridge. Seven limit switches on the hoist and grapple mechanism restrict travel and prevent damage to equipment and canisters.

The canisters are lifted by means of a grapple on the end of the canister hoist which extends vertically downward from the trolley. The grapple is positioned over the center of the fuel canister by movement of the bridge and trolley. There are indexes for each canister location to locate the crane trolley and bridge. The grapple is lowered into position using the hoist. The operating handle is manually turned clockwise to engage the canister for pickup. The canister has three lifting lugs spaced 120° apart near the top of the canister which the grapple engages and a support ring which engages the storage rack. The canister crane grapple that engages the lifting lugs moves about 30° before it is physically stopped by the crane housing. The grapple is raised 15 cm (6 in) and the canister is ready to be transported. After the canister is moved to the designated location, a reverse procedure is used to disengage the grapple from the canister.

5.2.3.1.3 Design Bases and Safety Assurance

The FRS facility was constructed as part of the original NFS facility and was designed to meet applicable building codes in effect at that time (e.g., 1961 New York State Building Code, 1956 ACI Code 318, and the 1961 Uniform Building Code). The seismic loads of the facility were determined using seismic Zone 3 of the 1961 UBC.

The FSP has been extensively evaluated over the course of its operational history to determine its seismic capacity and structural integrity. A dynamic analysis of a 0.12g earthquake that included modeling of the interaction between canisters, water in the pool, the pool walls, and the soil concluded that factored load combinations did not exceed pool wall or mat capacity (Blaw-Knox Chemical Plants Inc., 1972). Lawrence Livermore Laboratory, in an independent evaluation, performed a static analysis with the pool full of water that indicated that the integrity of the pool would be maintained up to 0.16g force (LLL, 1978). Analysis indicates that the racks can withstand up to a 0.19g earthquake (SAI, 1981).

Dames & Moore performed an engineering review to assess the capacity of the fuel pool walls under combined loading (Dames & Moore, 1994). This analysis considered the following loads:

- Dead Load
- Live load - racks and SNF assemblies
- Soil pressure
- Hydrostatic pressure - internal pool water and external ground water
- Seismic loads - 0.1g site design basis earthquake

The review confirmed the results of previous analyses that showed that the pool, when filled with water, has adequate strength for the design loads, including seismic. The vertical reinforcing in the cantilever walls is adequate to support the loads. The horizontal reinforcement meets the minimum requirements of ACI-318 to control (but not eliminate) cracking.

An area of concern related to the structural integrity of the FSP is the presence of cracks in above grade sections of the pool wall. Over the life span of the pool, starting with construction, a series of thin vertical cracks have formed in the pool walls above the grade floor slab. These cracks have been patched as they developed to control seepage. The possibility that these cracks may be indicative of a gradual deterioration in the pool's structural integrity, ultimately resulting in catastrophic collapse of the pool, was evaluated by Dames & Moore and the possible causes and implications of cracks along the north, east, and south walls of the fuel pool was assessed (Dames & Moore, 1994). The cracks had widths on the order of hundredths of an inch.

Based on evidence gained through field observation, construction records, and crack mapping, as well as the operating history of the pool, Dames & Moore concluded that the cracks (excluding construction joints) in the top 1.2 m (4 ft) of the pool were formed during the pool construction and the early stages of pool usage when creep and shrinkage of the concrete, which ceased long ago, combined to induce shortening of the pool wall. The high operating temperature of the pool may also have aggravated the cracking. It is therefore presumed that the existing cracks open and close because of the constant agitation that the wall experiences due to variations in thermal gradients and possibly due to seasonal variations in ground water level and not as a result of some new loading condition or deteriorating property of the structural walls.

As further confirmation that significant loss of pool water is not occurring through cracks in the pool wall, a blocked evaporation test of the FRS spent fuel pool was performed. The maximum unaccounted loss of water from the pool over a 3-month period was less than 34 L (9 gal) per day (WVNS, 1994). This small volume of water may be accounted for through evaporation at the vapor barrier boundaries and instrumentation recording accuracy. For all practical purposes, leakage from the pool is virtually zero.

There currently are no signs that the pool structure itself is undergoing deterioration in the form of rusting of reinforcing bars, spalling of concrete cover from the bars, differential settlement or deflection of the walls such as out of plane bowing. The cracks have minimal separation, are oriented perpendicular to the direction of primary loading on the walls and thus appear to have no bearing on the structural integrity of the walls. The structure appears to be in good condition. In general, the pool structure has been maintained in an excellent condition since 1982.

Based on conclusions provided in the referenced evaluations, catastrophic structural collapse of the pool walls is highly improbable. Catastrophic collapse in the pool structure could only occur should the vertical reinforcing bars at the base of the walls corrode away (Dames & Moore, 1994). Since these bars are encased in concrete as a protective barrier and are under water (which was treated to remove impurities), and are thus prevented from being exposed to free oxygen for oxidation processes, the possibility of the material deterioration and catastrophic structural failure is highly improbable.

Fuel storage racks in the FSP were designed and constructed as part of the original NFS reprocessing facility. Based on available design documents, the fuel storage racks, as designed, were not intended to resist lateral earthquake forces. The 1961 UBC provided no specific provisions for the design of parts or portions of a building such as racks under seismic loading. Standard welding procedures specified in the ASME codes were used for fabrication welding of the racks. Anchor bolts were designed and installed with minimal inspection. Oxidation of the anchor bolts in the pool water as well as in the concrete may have taken place over the life span of the pool.

Analysis, as part of the seismic qualification of the rack system, was conducted by SAI as documented in a 1981 report. This analysis indicated that under ideal design conditions (e.g., without corrosion and related deterioration), the rack system could survive peak ground accelerations on the order of 0.2 g at ultimate strength (e.g., incipient failure of critical bolted connections). The analysis by SAI neglected to assess the capacity of the anchor bolts in the concrete. Limited information on the configuration of the bolts and their depth of embedment would judgmentally preclude their capacity in the concrete exceeding the capacity of the bolts above the concrete. In all probability, the bolt capacity within the concrete is less than the ultimate capacity above the concrete interface.

A series of approximations were made in idealizing the canister-rack-pool-system under earthquake in a linear elastic dynamic model. These approximations in some cases were conservative, and in other cases were admittedly non-conservative. Results of SAI's analysis provide a measure of assurance that the rack system has sufficient capacity to resist an evaluation basis earthquake peak ground acceleration of 0.1g.

The fuel pool canister crane is situated on rails on top of the fuel pool walls. There is no restraint to prevent the crane from jumping off the rail in the event of major ground shaking. However, for the DBE at the WVDP, there is a very remote chance that the crane would be dislodged. If it were to be dislocated from the rail in the event of an earthquake, there is a good possibility that it would be caught by ladders provided for personnel access to the service bridge and would not fall on the canisters. Furthermore, because of its light weight and short fall distance, the impact on the canisters would not cause excessive damage (Dames & Moore, 1992a).

A summary of the results of seismic analyses that have been performed on FRS facilities is given in Table 4.2-2.

5.2.3.2 Cask Unloading Pool Description

The CUP is a concrete basin located east of the FSP and is 7.9 m (26 ft) long by 7.3 m (24 ft) wide and is sectioned to depths of 8.8 m (29 ft) and 13.4 m (44 ft). The deeper section provides the necessary shielding during removal of the irradiated fuel assemblies up to 4.9 m (16 ft) in length from a canister positioned near the bottom of the CUP. The CUP is lined with stainless steel, 2 mm (14 gauge) on the walls and 5 mm (3/16 in) thick on the floor. This liner provides physical protection of the concrete vault from abrasion during cask placement.

Fuel handling in the CUP is accomplished through the use of a fuel pool service bridge and a canister lift rack, as well as a 90 MT (100 ton) cask handling crane which services both the CUP and the cask loading area. A removable watertight gate that serves to isolate the CUP from the FSP is provided in the event that the pools need to be physically isolated from each other. This gate is stored on a rack on the north wall of the CUP.

5.2.3.2.1 Function

The CUP serves as the interface between the FSP and the transportation/loading area. Fuel located in the FSP that is to be loaded into a cask for shipping is first transported to the CUP where it is staged on the canister lift rack. From the lift rack fuel assemblies may then be placed into a shipping cask which has been placed at the 13.4 m (44 ft) level of the CUP. When the cask is fully loaded, it is moved from the CUP to the decontamination stall in the loading area where it is further prepared for shipping. The design of the CUP ensures that operations personnel are adequately shielded during transfer of SNF assemblies from storage canisters to a shipping cask.

5.2.3.2.2 Components

The primary components of the CUP include the canister lift rack, the fuel pool service bridge, the cask crane, and the fuel pool gate. The cask crane is described in section 5.2.3.3.

Canister Lift Rack

The canister lift rack is an elevator-type device mounted on the west wall of the CUP and can hold up to four fuel canisters in a straight row oriented north-south. Canisters containing fuel assemblies to be removed for shipment are moved from the canister storage racks in the FSP to the lift rack in the CUP. The rack has a vertical travel of 4.9 m (16 ft) and is moved by means of an electric winch located in the former water treatment area pit north of the CUP. At the lower end of the rack travel the bottom of the canisters are 23 cm (9 in) above the 13.4 m (44 ft) depth to facilitate fuel removal to a shipping cask. When the rack is raised to its upper level, the tops of the canisters are at the same level as the storage racks in the FSP and may be handled with the fuel pool canister crane. Limit switches restrict both upward and downward movement of the canister lift rack. Additionally, a mechanical stop prevents the lift rack from being raised to a height that reduces shielding to less than 3.3 m (11 ft) of water. Design of the lift rack precludes the bottom of the canisters from contacting the CUP floor even in the event of catastrophic failure of the cable.

Fuel Pool Service Bridge

Fuel handling in the CUP is performed through the use of equipment located on the fuel pool service bridge. The fuel pool service bridge spans the width of the storage pool and operates on the same rails as the fuel pool canister crane bridge. The bridge is controlled through the use of a portable push button control unit and can travel the full east-west length of the CUP. The bridge has a working platform and hand rails and may be operated at one of two speeds. Fuel in the CUP is removed from storage canisters by means of an electrically driven hoist mounted on the fuel pool service bridge. The fuel hoist has a capacity of 900 kg (1 ton). The hoist has a jib that swivels using a hand tiller that can be locked in position. North and south movements are controlled manually by releasing the hand tiller and rotating the fuel hoist the desired amount. Operation of the fuel hoist is also controlled by a portable push button control unit.

The fuel pool service bridge does not normally travel over the storage pool because of interference between the jib hoist and the lower roof level over the FSP. Westward movement of the service bridge is restricted by a limit switch to prevent damage to the west wall of the high bay by the hoist jib. The hoist jib can be removed if the service bridge is needed over the storage pool, and the subject limit switch bypassed.

The fuel jib hoist, in combination with a grapple, serves to lift SNF from canisters in the canister lift rack for placement in a shipping cask. Operation of the fuel hoist is restricted by a pair of limit switches that are activated by either exceeding the lift capacity, or reaching the maximum upper position of the cable. The hoist is also equipped with a mechanical stop to prevent lifting a fuel assembly to undesired levels within the pool when used in conjunction with the proper fuel grapple. Long handled grappling tools are used to engage the fuel assemblies and lift them free of their respective canisters. The grapples are specific for each fuel type, thereby ensuring positive attachment to a given SNF assembly. Grapples can be removed and decontaminated for repair or adjustment.

Fuel Pool Gate

A stainless steel fuel pool gate is provided to separate the CUP from the FSP in the event that the two pools need to be physically isolated from each other. The gate is tapered from top to bottom and a hard rubber seal (J-seal) on the "low water" side that is compressed by an inflatable bladder on the opposite side of the gate. The fuel pool gate is stored on a rack on the north wall of the CUP. The south 4.5 MT (5 ton) auxiliary crane on the cask crane is used to install and remove the fuel pool gate.

5.2.3.2.3 Design Bases and Safety Assurance

The design and construction of the CUP is similar to that of the FSP. Conclusions of the assessments described in Section 5.2.3.1.3 are also valid for the CUP.

The fuel pool service bridge rolls on a rail situated on top of the fuel pool walls. There is no restraint to prevent the bridge from jumping off the rail in the event of major ground shaking; however, for the DBE at the WVDP, there is a very remote chance that the crane could be dislodged. Since it is normally located at the east end of the FSP and has limit switches that prevent it from traveling down the tracks to the west end of the pool where the fuel canisters are stored, even if it were to fall off its rail, the bridge would not come in contact with the fuel canisters (Dames & Moore, 1992a).

A summary of the results of seismic analyses that have been performed on FRS facilities is given in Table 4.2-2.

Fuel handling equipment in the CUP is provided with limit switches to prevent damage to handling equipment and fuel assemblies. A limit switch on the jib hoist ensures that fuel assemblies that cannot be lifted from the canisters are not damaged. Mechanical stops on lifting equipment ensure that fuel cannot be raised to a height that results in unacceptable surface exposure rates.

5.2.3.3 Cask Loading Area Description

The cask loading area, consisting of the cask decontamination stall and the railroad area, is located to the east of the FSP and CUP and has been provided as a staging area for shipping vehicles. The area can accommodate equipment for either over-the-road or rail transport. The cask loading area is serviced by the cask crane.

5.2.3.3.1 Function

The cask loading area provides a sheltered enclosure for cask shipping equipment preparation, cask decontamination, and cask handling, i.e., loading a cask onto or unloading a cask from a transport vehicle.

5.2.3.3.2 Components

Cask Decontamination Stall

The cask decontamination stall is an aluminum structure located at the east end of the FRS Building, adjacent to the railroad area. The stall facilitates the inspection of cask internals prior to the installation of a cask into the CUP, and facilitates the decontamination of a cask after removal from the CUP. It is equipped with a sliding door and roof to permit the cask crane to position a cask vertically inside. The shipping cask is positioned within an annular, elevator-type platform that allows an operator to inspect, radiologically survey, and decontaminate the cask prior to shipping. Drains are provided to route any decontamination water to the site Low-Level Waste Treatment Replacement Facility (LLW2). A 20 cm (8 in) duct at the top of the decontamination stall is capable of withdrawing air at a rate of 0.5 m^3/s (1,000 cfm) (10 air changes per hour) from the stall interior. This air is routed to the main plant ventilation washer in the reprocessing plant, filtered through the main exhaust filters, and released to the plant stack. The cask decontamination stall was added as part of the NFS modification program, and has been modified to accommodate the larger shipping casks anticipated for use in future shipments (i.e., a Transnuclear Incorporated Big Rock Point [TN-BRP] cask or Transnuclear Incorporated Robert E. Ginna [TN-REG] cask).

Cask Crane

Shipping casks in the FRS are handled by means of the cask crane. The weight of a fully fuel loaded and drained REG shipping cask is approximately 97 MT (107 tons). Authorization for fully loaded cask handling in support of fuel shipping was obtained in 2000 from the representative of the original crane manufacturer per ASME B30.2b Section 2-3.2.1.1 for Planned Engineered Lifts (Weiss, T.G., 2000). Subsequently, the FRS cask crane has been load tested to 110 MT (121 tons).

The cask crane cannot extend over the FSP which has a lower roof line. Hence, it is not possible to drop a load being carried by the cask crane onto the stored fuel assemblies. Redundant rigging is therefore not required. The total lift available is approximately 26 m (85 ft), of which 11.6 m (38 ft) is above the floor of the cask unloading area.

The cask crane and two 4.5 MT (5 ton) auxiliary cranes are normally operated remotely by radio control. They can also be operated from separate control pendants that can perform the same operations as the radio control, except that bridge movement cannot be controlled from the auxiliary crane control pendants. Each motion control lever associated with the cask crane provides a 5-step variable speed in any direction (i.e., east and west for the bridge, north and south for the trolley, and up and down for the hook). The two 4.5 MT (5 ton) auxiliary cranes are supported by the cask crane bridge. Both auxiliary trolleys are single speed. Both auxiliary hoists are two speed.

5.2.3.3.3 Design Bases and Safety Assurance

Equipment and structures in the cask loading area have been designed to withstand the effects of a design basis earthquake (0.1 g) and have been provided with engineered features to reduce the risk of cask handling accidents.

The cask unloading crane rolls on topside rails along the crane girders in the FRS Building. It is possible that in the event of a major earthquake, this configuration would permit the crane to bounce free and fall to the floor below. Based on an evaluation by Dames & Moore, it was determined that, based on previous experience, this type of failure does not occur under ground accelerations on the order of 0.1 g to 0.2 g. Instead, the crane rails warp and become misaligned under the large lateral loads induced by the crane girder. Even if the crane were to break free and fall, it would not fall in the area where the fuel canisters are stored under the low bay at the far west end of the building. Thus, seismic risk of crane girder collapse on the fuel assemblies is non-existent (Dames & Moore, 1992b).

Furthermore, if the crane girder were to collapse in an earthquake, the greatest impact it would have on the basic building structure would be to tear out a line of diagonal seismic bracing. Even with a loss of all the seismic bracing in the building, light steel mill buildings with metal siding have continued to remain upright in earthquakes three to four times the DBE postulated for the WVDP. Thus, the crane girder could not induce partial or total building collapse as a result of its falling from the support rail (Dames & Moore, 1992b).

The cask loading crane could potentially fall on the walls of the fuel pool resulting in cracking and leakage. If such a scenario were to occur, the level of the pool could be lowered to ground level and further leakage from the pool would be minimal. This might result in an immediate reduction in pool depth of approximately 0.9 m (3 ft), leaving 2.4 m (8 ft) of water covering the fuel canisters in their storage racks as shielding. (Based on the current inventory of SNF, this would result in an exposure rate on the fuel pool service bridge of less than 10 mrem/hr.)

A summary of the results of seismic analyses that have been performed on FRS facilities is given in Table 4.2-2.

Engineered controls have been provided to prevent damage to the shipping casks and crane equipment. Four bridge motor emergency stop buttons are located in operating areas. Limit switches on the bridge disconnect bridge motor power when the bridge reaches either end of the track. Though there are no limit switches associated with trolley operation, the trolley is physically limited from over-travel by stops at the ends of the track. The upward travel of the crane hook is controlled by a limit switch using a trip weight. The upward and downward travel of the crane hook is also controlled by a limit switch geared to the hoist drum. There are no limit switches associated with the operation of the 4.5 MT (5 ton) trolleys; however, stops on the rail prevent them from running off of the rail. Each of the 4.5 MT (5 ton) hoists have one upper and one lower limit switch. The upper limit switch is controlled by a gear reduction directly from the drum to a lead screw. The lower limit switch is geared to the hoist drum and turns off the hoist motor when the hook is at the lowest point.

5.3 FRS Support Systems

The FRS facility has been provided with support equipment for pool water filtration and fire protection. These systems are described below.

5.3.1 Fuel Pool Water Filtration System

The fuel pool Submerged Water Filtration System has been designed to provide a level of water quality that ensures visual clarity for underwater operations and that ensures that degradation of SNF assemblies is minimized. Operational requirements for the water filtration system are based on Low-Level Waste Treatment System interceptor limits. Floor drains and sumps in the FRS facility drain to the site interceptors; therefore, pool water radioactivity levels are maintained below interceptor activity limits. The fuel pool Submerged Water Filtration System was placed on-line in 1994, and replaces the original pool water filtration equipment. Original fuel pool water treatment equipment is located in a pit area adjacent to the CUP, at the northeast corner of the FSP. This equipment is no longer used.

Fuel currently in the FSP has been in storage (not necessarily at the WVDP) for more than 25 years, which has resulted in significant post-reactor cooling (see Table 4.1-1 for reactor discharge dates for the assemblies). The estimated total thermal load of the fuel inventory is 8,800 watts, based on a 20 year decay period for BWR fuel and a ten year decay period for PWR fuel (Dames & Moore, 1992a). Due to the decreased heat generated by decay of fuel in the pool and a low fuel inventory, no pool water cooling function is required. Pool water temperature is adequately maintained through thermal convection with the atmosphere and thermal conduction with the ground.

The fuel pool submerged water filtration system is comprised of an underwater filter unit and an underwater demineralizer unit as indicated in Figure 5.2-3. Each unit has its own motor and pump assembly, allowing either unit to be operated independently of the other.

The underwater filter unit is designed to operate for extended periods with minimal maintenance. The filter unit has four pleated-paper cartridge filter elements. Each filter element is enclosed within a stainless steel housing. The bases of the four filter housings are mounted on the pump housing, which provides an inlet suction plenum to the pump. The motor and pump are located at the center of the filter assembly. Water from the pool is drawn downward through the four filter elements, up through the pump, and discharges at a nominal rate ranging from 570 to 1,100 LPM (150 - 290 gpm). For special cleaning purposes, a cover can be installed on any of the filter housings so that a hose may be attached to the cover to take suction from a localized area in the pool or from the surface. Loaded filters are removed and replaced remotely while the filter housing unit is underwater. Filter elements are replaced when an administrative dose limit is reached, or when flow rate is reduced to less than 570 LPM (150 gpm), or when the Spent Fuel Shipping and Main Plant Operations Manager directs the filters to be replaced.

Resin is maintained in the underwater demineralizer unit through the use of Johnson screens located on the inlet and outlet of the unit, while media retention elements provide a secondary barrier to resin migration. The demineralizer bed capacity is 0.85 m^3 (30 ft^3); however, the vessel normally contains about 0.68 to 0.79 m^3 (24 to 28 ft^3) of resin. The unit is designed so that resin may be remotely sluiced out and reloaded under water. The associated motor and pump assembly is located at the top of the demineralizer vessel. Water is drawn into the demineralizer unit through a nozzle in the top, flows down uniformly through the resin bed, up through an internal line into the pump and discharges at a nominal rate that ranges from 76 to 450 LPM (20 to 120 gpm). Resin is replaced when an administrative dose limit is reached, or when the Spent Fuel Shipping Operations Manager directs the resin to be replaced.

The pool is also equipped with a self-contained floating skimmer to remove surface debris. The skimmer is 91 cm (36 in) in diameter, 218 cm (86 in) high, and weighs approximately 118 kg (260 lbs). It is powered by a 2.5 hp submerged pump and motor. It utilizes one filter element which is identical to the elements in the underwater filter unit. Normal flow rates range from 189 to 1135 LPM (50 to 300 gpm). The filter element is replaced when the flow rate through the skimmer is reduced to less than 189 LPM (50 gpm), or when the Spent Fuel Shipping Operations Manager directs the filter to be replaced.

5.3.2 Radwaste Process System

The Radwaste Process System is housed in the Radwaste Process Building (also referred to as the Hittman Building), located in the yard area north of the FRS Building, as shown in Figure 5.1-1. The building is steel-framed, with steel siding and roofing. The center section of the roof is removable to allow access to steel and concrete shields that house HICs used to store loaded resins from the fuel pool Submerged Water Filtration System (see Figure 5.3-1). The concrete floor slab on which the concrete shield rests is provided with an integral curb and sump to collect spills that could occur in the building.

The Radwaste Process System, shown in Figure 5.3-2, provides a means to transfer loaded resin from the underwater demineralizer unit to a high-integrity container located in the Radwaste Process Building. An air operated valve in the system is designed to terminate the transfer of loaded resin when the volume of waste transferred to the on-line HIC reaches a predetermined level, or if liquid is detected in the overflow drum connected to the on-line HIC. The system is also used to dewater the loaded resin after placement within a HIC.

Two high integrity containers are located in the Radwaste Process Building with one HIC on-line at all times. Each HIC is housed in a shield structure that is comprised of two concentric shields. The inner shield is constructed of steel-reinforced concrete with a thickness of 36 cm (14 in). The outer shield is constructed of carbon steel and has a thickness of 5 cm (2 in). Each HIC has an approximate volume of 3.54 m^3 (125 ft^3). A full HIC weighs approximately 3,200 kg (7,000 lbs), and holds approximately 2.83 m^3 (100 ft^3) of waste with up to 1 percent by volume of free liquid. Based on surveys of existing containers, a full HIC has a contact dose rate between 3 to 15 rem/hour on the side. HIC contents are radiologically classified based upon results of sample analyses.

HICs are constructed from high density cross-linked polyethylene and outfitted internally with a "single layer well point underdrain" to support the dewatering process. (Dewatering denotes the removal of bulk liquid from a slurry by the use of pump suction on a filtering network, which in this instance is the "single layer well point underdrain" located within a HIC.) Water from the dewatering process is discharged to a floor drain in the FRS, where it is routed to the site Low-Level Waste Treatment System interceptors.

When the on-line HIC has been filled, it is dewatered for the final time, sampled, and sealed through installation of a fill port closure. The full HIC is lifted out of its process shield through the Hittman Building roof through the use of a crane and placed in a polyethylene-lined concrete shield container (Surepak) for storage.

Lifting of a HIC is considered to be a "critical lift", which is subject to the requirements of WVDP-082, *Hoisting and Rigging Manual*.

Because the pool water is relatively pure, replacement of demineralizer resin is expected to occur on a relatively infrequent basis (e.g., once every 8 to 12 months). Each replacement of the demineralizer resin uses up about one-third of the working volume of a HIC. Hence, the two HICs which are presently installed in the Hittman Building will likely provide sufficient capacity for spent resin until the planned shipping date of the SNF (i.e., after 2001). Should these two HICs not provide sufficient capacity, the north FRS yard has sufficient area available to accommodate additional Surepaks (i.e., at least two) for the storage of loaded HICs.

5.3.3 <u>Fire Protection</u>

A fire hose station is located in the north operating aisle of the FRS Building that includes a 3.8 cm (1.5 inch) hose connected to the site fire water supply loop. (Section B.5.3.1 of WVNS-SAR-002 provides a discussion of the site fire protection water supply.) ABC-type fire extinguishers are also located throughout the building. Fire extinguishers are provided for incipient stage fire fighting.

NFPA 780, "Standard for the Installation of Lightning Protection Systems," (NFPA 1997), states that "Strike termination devices shall not be required for those parts of a structure located within a zone of protection." A strike termination device is "a component of a lightning protection system that is intended to intercept lightning flashes and connect them to a path to ground." Strike termination devices include air terminals (i.e., lightning rods), metal masts, permanent metal parts of structures in some instances, and overhead ground wires installed in catenary lightning protection systems. A zone of protection is "the space adjacent to a lightning protection system that is substantially immune to direct lightning flashes." The Main Plant stack is considered to serve as a strike termination device and to provide a zone of protection for the FRS facility. NFPA 780 states that "The zone of protection shall form a cone having an apex at the highest point of the strike termination device, with walls forming approximately a 45-degree or 63-degree angle from the vertical." Hence, the FRS facility is within the Main Plant stack's zone of protection, and therefore does not require a strike termination device.

5.4 <u>Description of Service and Utility Systems</u>

5.4.1 <u>FRS Building Heating and Ventilation Systems</u>

The FRS HVAC systems are designed to maintain air quality in the FRS Building and to maintain the building under a slight negative pressure. FRS Building ventilation systems and capacities are illustrated in Figure 5.4-1.

The Recirculation Ventilation System provides the HVAC requirements by recirculating approximately 7.1 m^3/s (15,000 cfm) of air through heating or cooling coils while adding approximately 0.9 m^3/s (2000 cfm) of makeup air, and by filtration of air to remove entrained particulates. This is a recirculation system only and does not exhaust air to the environment.

The Main Plant Ventilation System exhausts air from the cask decontamination stall and former water treatment equipment through a 20 cm (8 in) duct. This air is subsequently HEPA filtered by the Main Plant Ventilation System and is discharged through the Main Plant stack.

An Exhaust Blower (1K-1) provides negative pressure in the FRS Building by exhausting air from the south aisle pool area to the Main Plant stack. This exhaust air stream is unfiltered; however, the air passing through the stack is continuously monitored by plant stack air monitors.

The Recirculation Ventilation System includes high efficiency filters, redundant recirculation fans, a reheating coil, a makeup air roughing filter, ductwork and controls. Two identical recirculation fans provide airflow through the system. Each fan is powered by a 20 h.p. motor and has a rated flow capacity of 4 m^3/s (8,500 cfm). Only one fan is usually on-line with the other maintained in standby, though both may be run simultaneously.

Air from the FRS Building is filtered through four banks of three high efficiency filters. The filter banks are located in a building in the north FRS yard. Filter housings are dampered so that individual filters may be removed and replaced without taking the entire system off-line. Plenums have been installed upstream and downstream of the filter housing to ensure uniform filter loading. A steam supplied reheat coil is provided to warm air returning to the FRS Building. Filter pressure differential is monitored and indicated on a panel in the Recirculation Ventilation Building. An alarm located in the East Mechanical Operating Aisle alerts operations personnel to an abnormal differential pressure situation.

5.4.2 Electrical

Electric power for the West Valley Demonstration Project (WVDP) is supplied from a 34.5 kV Niagara Mohawk Power Corporation loop system. Electricity from a 34.5 kV line is routed through a fused disconnect switch to the 2500 kVA transformer at the Main Plant, which delivers power to a 480V, three phase bus via a 4,000 amp main breaker in the Switchgear Room. From the 480V, three phase bus, power flows to ten main circuit breakers which, in turn, supply subpanels through underground cables and conduits. Power to equipment in the FRS Building is distributed through a subpanel located in the south operating aisle. Equipment in the Radwaste Process System and the Recirculation Ventilation System is supplied through a subpanel located in the Recirculation Ventilation Building.

Three phase, 60 Hz backup power is produced at 480V by a 625 kVA diesel-driven generator located in the Utility Room. Diesel fuel is supplied to the engine from a 1,000 liter (275 gallon) day tank in the Utility Room which is enough for eight hours of operation. Additional fuel is supplied to the backup generator from a 38,000 liter (10,000 gallon) above ground tank sufficient for a period of at least five days. Backup power is supplied to the FRS Building to provide electricity for lighting and area radiation detectors in the event of a line power failure. Other FRS facility loads can receive power (if deemed necessary) by manual actions. A battery in each area radiation detector ensures that their operation is continuously supported until backup AC power is available.

5.4.3 Compressed Air

Utility air and instrument air are used in the FRS facility to operate instruments and to support operations in the fuel pool Submerged Water Filtration System and Radwaste Process System. The FRS facility is connected to the Main Plant compressed air supply. Four compressors are supplied for plant air systems: a 300 hp steam turbine-driven compressor; a 350 hp electric centrifugal compressor; and two 200 hp screw compressors. All compressors are of non-lubricated design. Carbon monoxide monitors are installed to ensure air is of suitable quality for breathing to support manned entry to areas of elevated airborne radioactive contamination.

Instrument air is provided from the utility air system using an air dryer and a pressure reducing valve to reduce the air pressure to 380 kPa (55 psi).

5.4.4 Steam Generation and Distribution

The steam generation and distribution system is comprised of two natural gas fueled fire-tube boilers with a 15,658 kg/hr (34,520 lb/hr) combined steam generating capacity. Number 2 diesel fuel oil can be used as an alternate fuel source in the event of an interruption in the gas supply. Each boiler is designed to provide the full steady-state steam demand requirements. Therefore, one boiler is normally in standby. Intermittent batch demand will be satisfied in all instances except for the simultaneous operation of the Concentrator Feed Make-up Tank in the Vitrification Facility and the LWTS evaporator in the peak winter months. At these times, the intermittent steam demand is met by operating the standby boiler unit. Return condensate is collected in a condensate receiver where it is continuously sampled for radioactivity. It may then be returned to the boiler water makeup system or pumped to the interceptor. A radiation monitor is provided on the condensate return lines to the receivers. Steam is used to support FRS HVAC and building heating functions.

5.4.5 Water Supply

The FRS facility is connected to the Main Plant water supply, which originates from two man-made, interconnected lakes created by the construction of two dams near the south end of the site. From the water in these lakes, the WVDP derives utility, potable, and demineralized water. The Demineralized Water System normally produces about 1 L/s (16 gpm) of demineralized water and may produce 2 L/s (32 gpm) maximum makeup to the 68,000 L (18,000 gal) demineralized water storage tank. Demineralized water is used in the FRS Building for maintaining the water level in the FSP, and for equipment (e.g., shipping cask) decontamination purposes. For additional information regarding the WVDP water supply, see Section B.5.4.5 of WVNS-SAR-002.

5.4.6 Natural Gas Supply and Distribution

The FRS facility does not require the use of natural gas and there are no natural gas lines passing through the facility. The natural gas supply and distribution system for the WVDP is fully described in Section B.5.4.6 of WVNS-SAR-002.

5.4.7 Waste Water Treatment Facility

There are no sanitary facilities in the FRS Building. Operators use facilities located in the Main Plant. The Site Waste Water Treatment Facility is described in Section B.5.4.7 of WVNS-SAR-002.

5.4.8 Safety Communications and Alarms

5.4.8.1 Safety Communications

A paging system is available from site telephones to notify WVDP personnel of an abnormal or emergency condition. When the appropriate number is dialed a distinct alarm is annunciated through the site paging system speakers. The alarm is then followed by an announcement of the type and location of the emergency.

On-site communications systems include telephones, pagers and radios. The WVDP radio network consists of nets A and B. Net A is assigned to Security and net B is assigned to Operations, Radiation Protection, the Emergency Operation Center and Security. The Project also maintains a radio link with the Cattaraugus County Sheriff's Department and the West Valley Volunteer Hose Company (WVVHC) which can be used to request assistance or as a source of information.

5.4.8.2 Alarms

The FRS facility has a Radiation Monitoring System consisting of at least one area radiation detector and continuous air monitors (CAMs). These devices alert operators to hazardous or potentially hazardous conditions.

5.4.9 Maintenance Systems

The FRS facility was designed as a contact maintenance facility. Due to potentially high contact exposure rates, underwater filter unit cartridges and loaded ion exchange resin are remotely replaced. As much as practical, valves and pumps are separated from high radiation fields by distance and/or shielding materials. All equipment and piping is remotely drainable and flushable to reduce radiation levels prior to maintenance.

5.4.10 Cold Chemical Systems

There are no cold chemical systems associated with FRS facility operations.

REFERENCES FOR CHAPTER 5

Blaw-Knox Chemical Plants, Inc. December 12. 1972. *Evaluation of the Fuel Receiving and Storage Structure for Tornado and Earthquake Forces.*

Dames & Moore. February 27, 1992a. *Calculation of Decay Heat Generated by the Irradiated Fuel Stored in the FRS Fuel Storage Pool.* J. C. Wolniewicz to S. R. Reeves. Memo FB:92:0055.

__________. 1992b. Response to DOE-TRG Comments 05-03 and 09-06 contained in RCR #SAR-TRG-WV-FR-004, W. E. Gates, October 14, 1992.

__________. 1994. *Report, FRS Pool Wall Integrity Review, West Valley, New York, for West Valley Nuclear Services Company, Inc.*, Job No. 10805-868-023, September 28, 1994.

__________. 1995. *Report, Seismic Integrity Review, Fuel Receiving and Storage Facility, West Valley, New York, for West Valley Nuclear Services Company, Inc.*, Job No. 10805-967-004, July 25, 1995.

LLL, 1978 Lawrence Livermore Laboratory, *Structural Analyses of the Fuel Receiving Station Pool at the Nuclear Fuel Service Reprocessing Plant, West Valley, New York,* (UCRL-52575), May, 1978.

NFPA. 1997. NFPA 780: Standard for the Installation of Lightning Protection Systems. National Fire Protection Association.

SAI. December, 1981. *Seismic Resistance Capacity Evaluation of Spent Fuel Storage Racks and Fuel.* SAI Report SAI-148-026. NUREG/CR-2236.

U.S. Department of Energy. 1994. *Plan of Action to Resolve Spent Nuclear Fuel Vulnerabilities, Phase III,* October, 1994.

Weiss, T.G., January 28, 2000. *Completion of Milestone SF-1, Section B.1.* Letter to B.A. Mazurowski (WD:2000:0086).

West Valley Nuclear Services Co., Inc. WVDP-082: *Hoisting and Rigging Manual.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. Safety Analysis Report WVNS-SAR-002: *Safety Analysis Report for Low-Level Waste Processing and Support Activities.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

REFERENCES FOR CHAPTER 5 (Concluded)

__________. 1994. *Report on Spent Fuel Storage Pool Water Loss Tests 11/1/92 to 1/16/94*, R. F. Zalenski, March 1994.

SR12-511.DWG

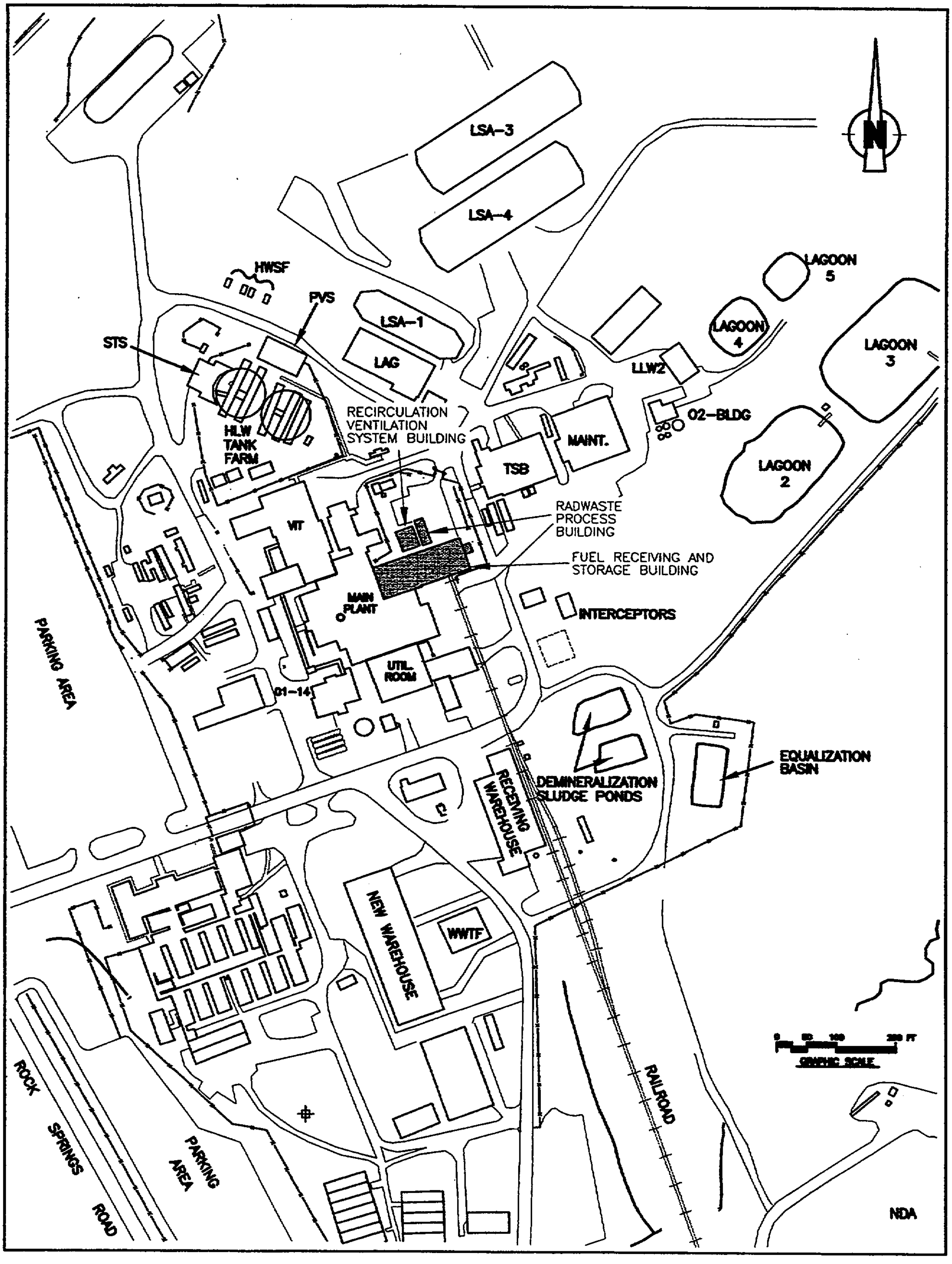


Figure 5.1-1 Location of the Fuel Receiving and Storage Facilities.

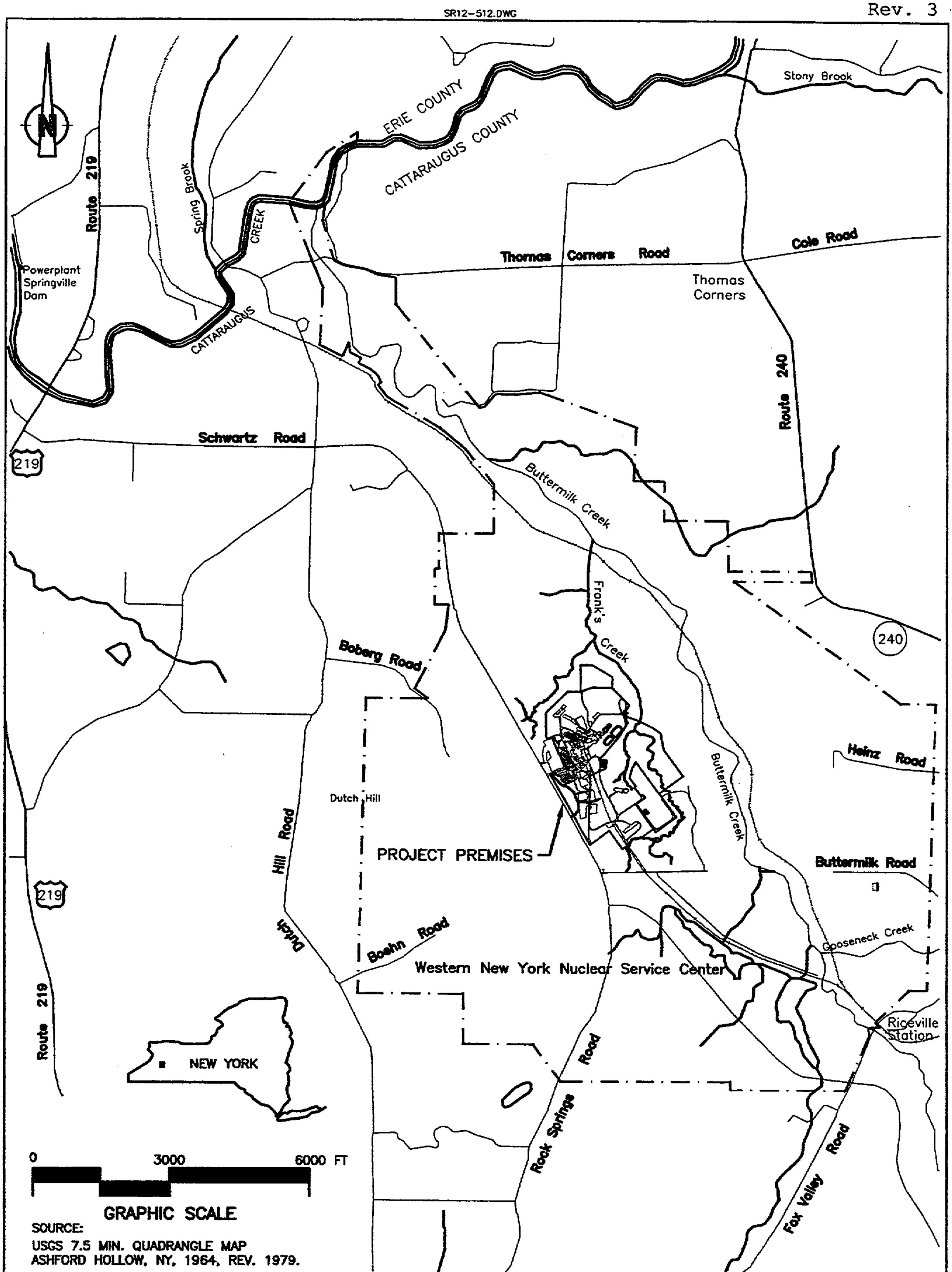


Figure 5.1-2 Location of West Valley Demonstration Project Site

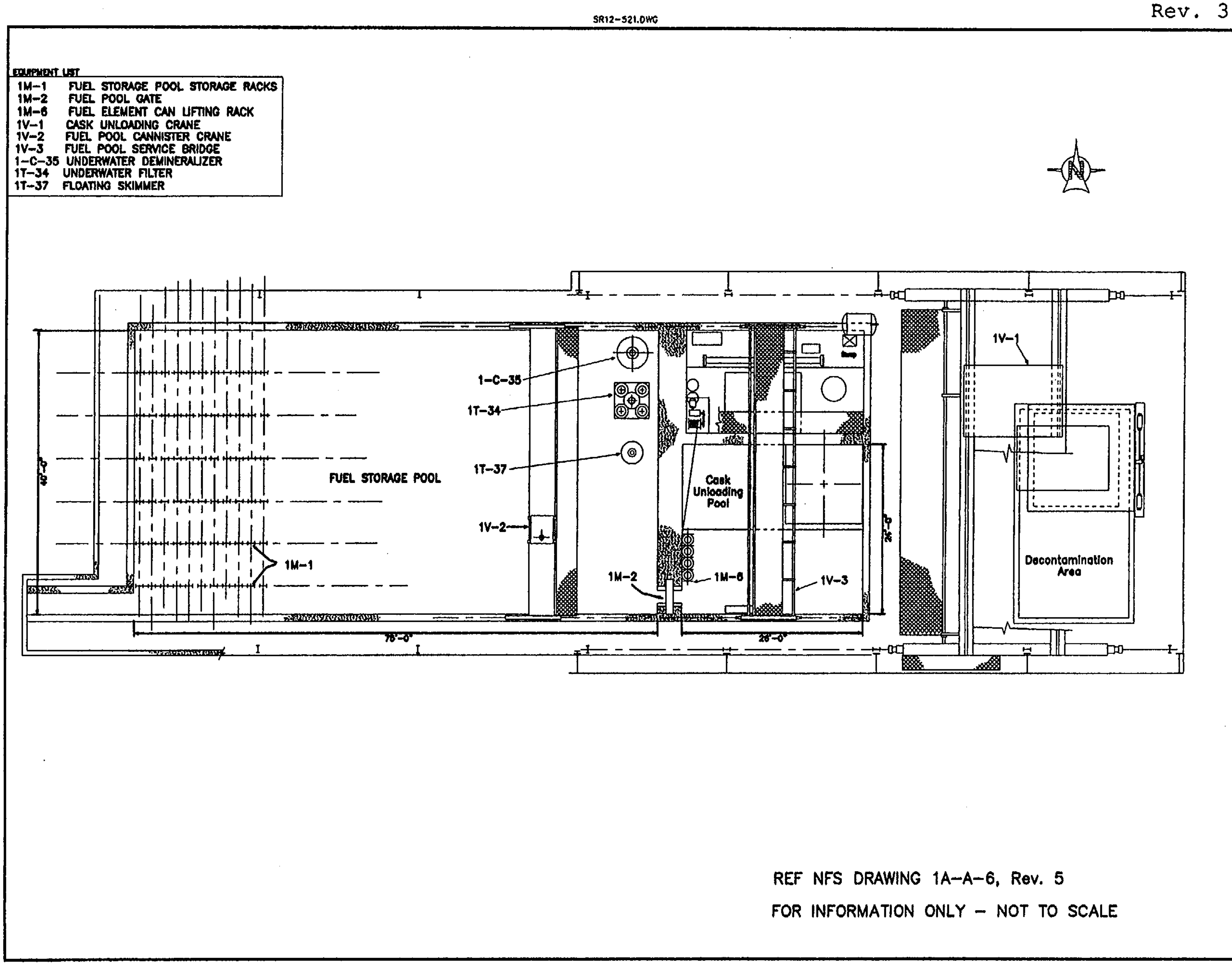
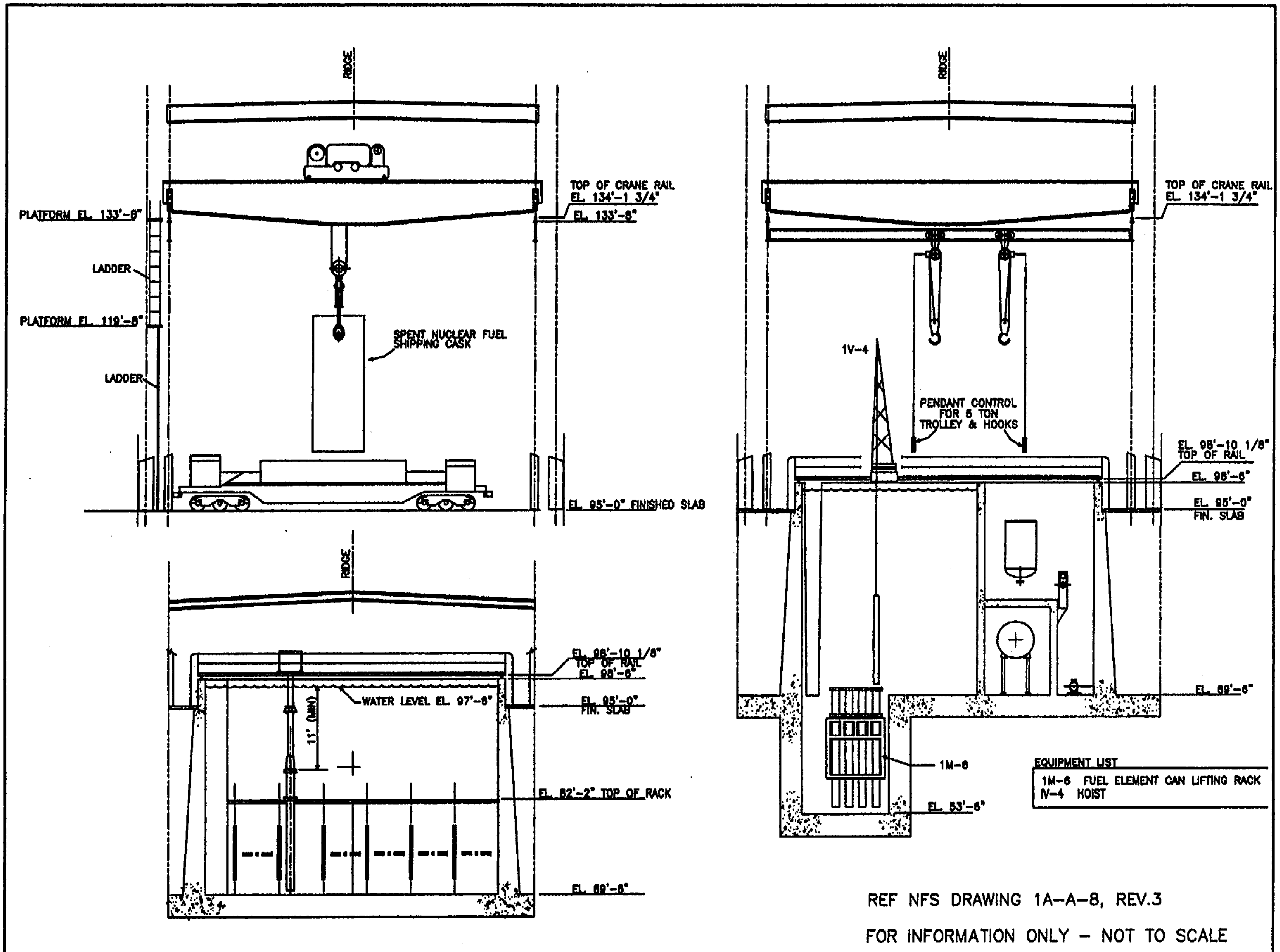


Figure 5.2-1 Fuel Receiving and Storage Building Plan

SR12-522.DWG



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FOR INFORMATION ONLY – NOT TO SCALE

Figure 5.2–2 General Arrangement Fuel Receiving and Storage Area Sections

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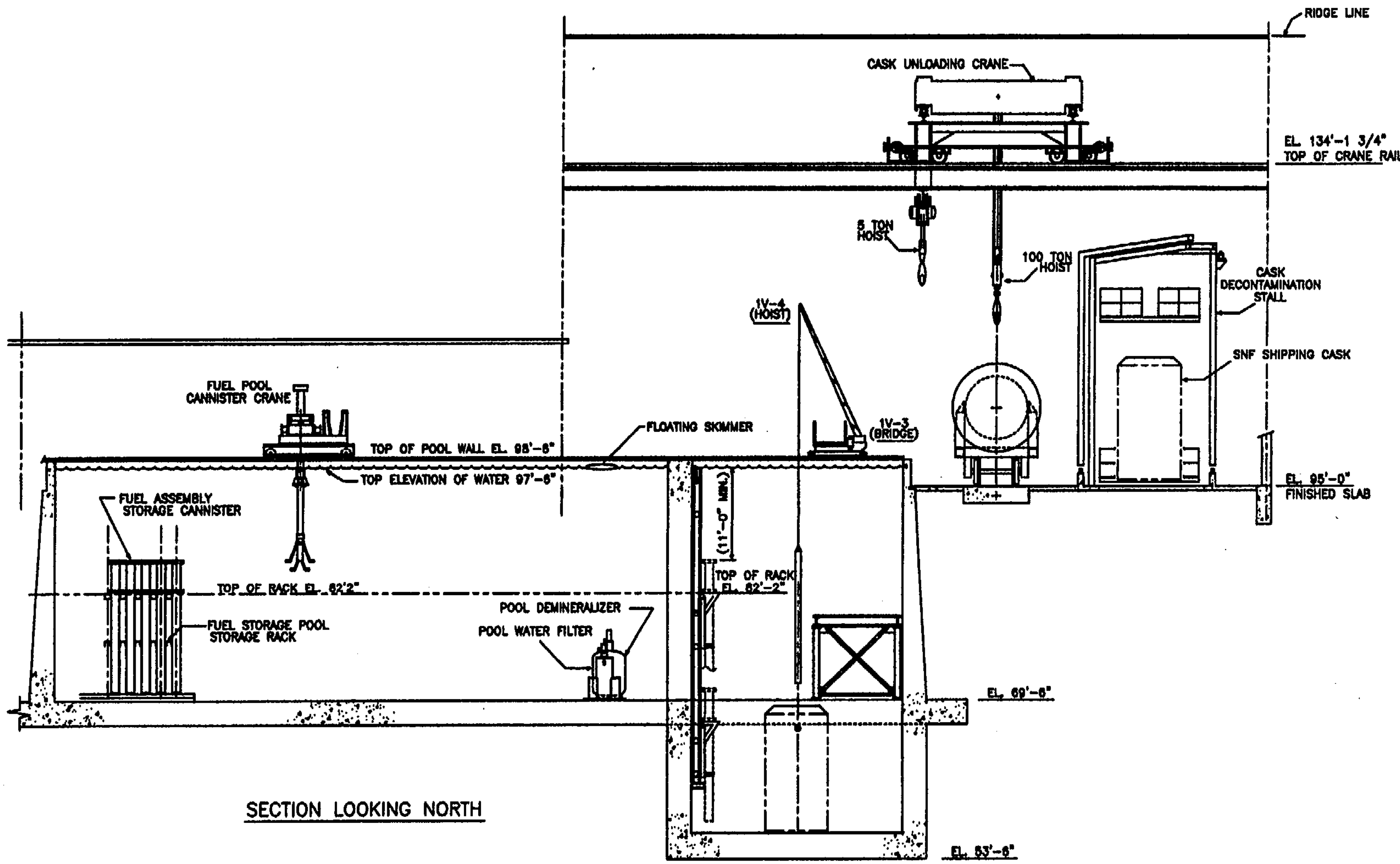
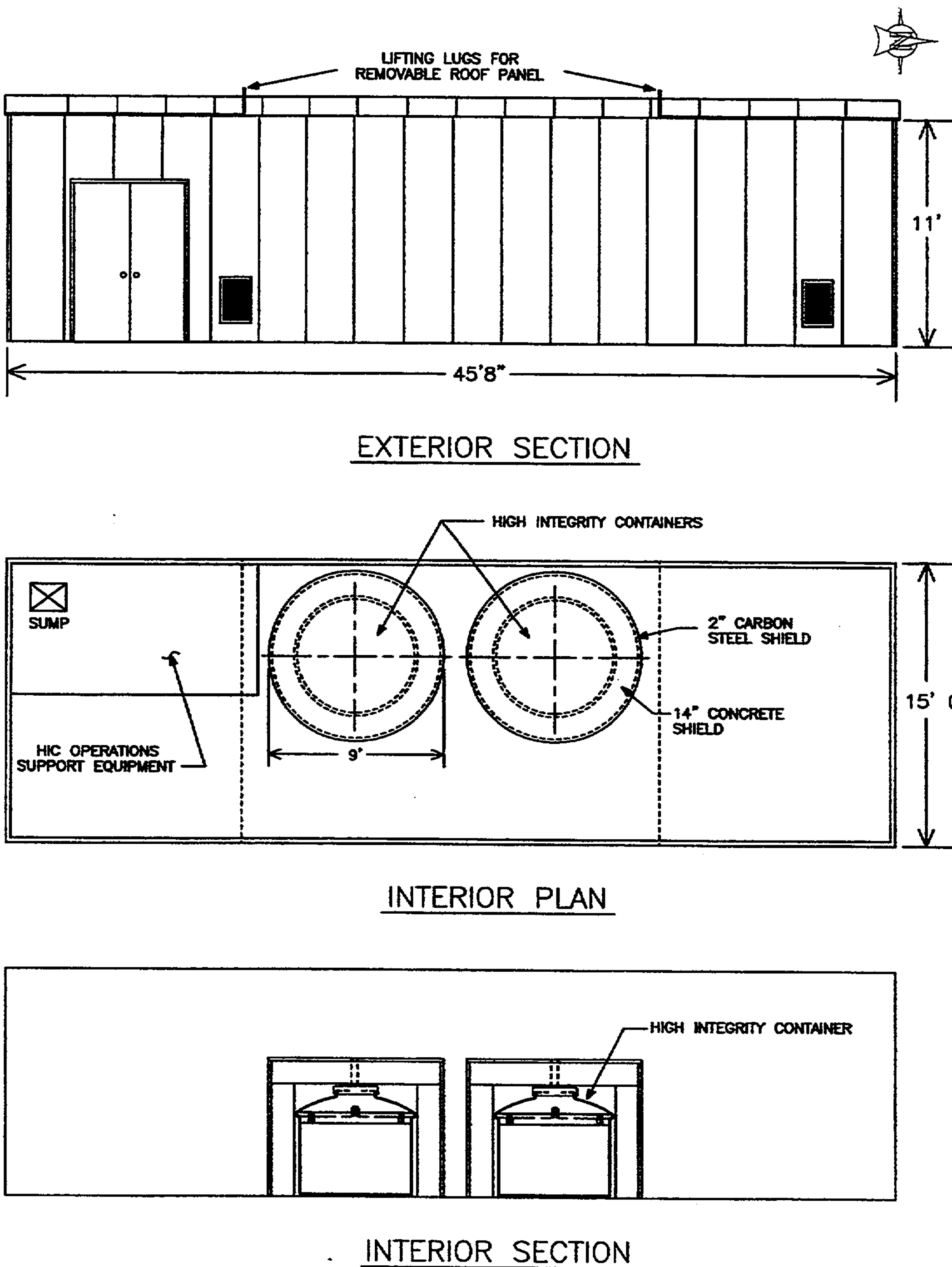


REF NFS DRAWING 2401-217-16, REV.1

FOR INFORMATION ONLY – NOT TO SCALE

Figure 5.2–3 General Arrangement Fuel Receiving and Storage Area Sections

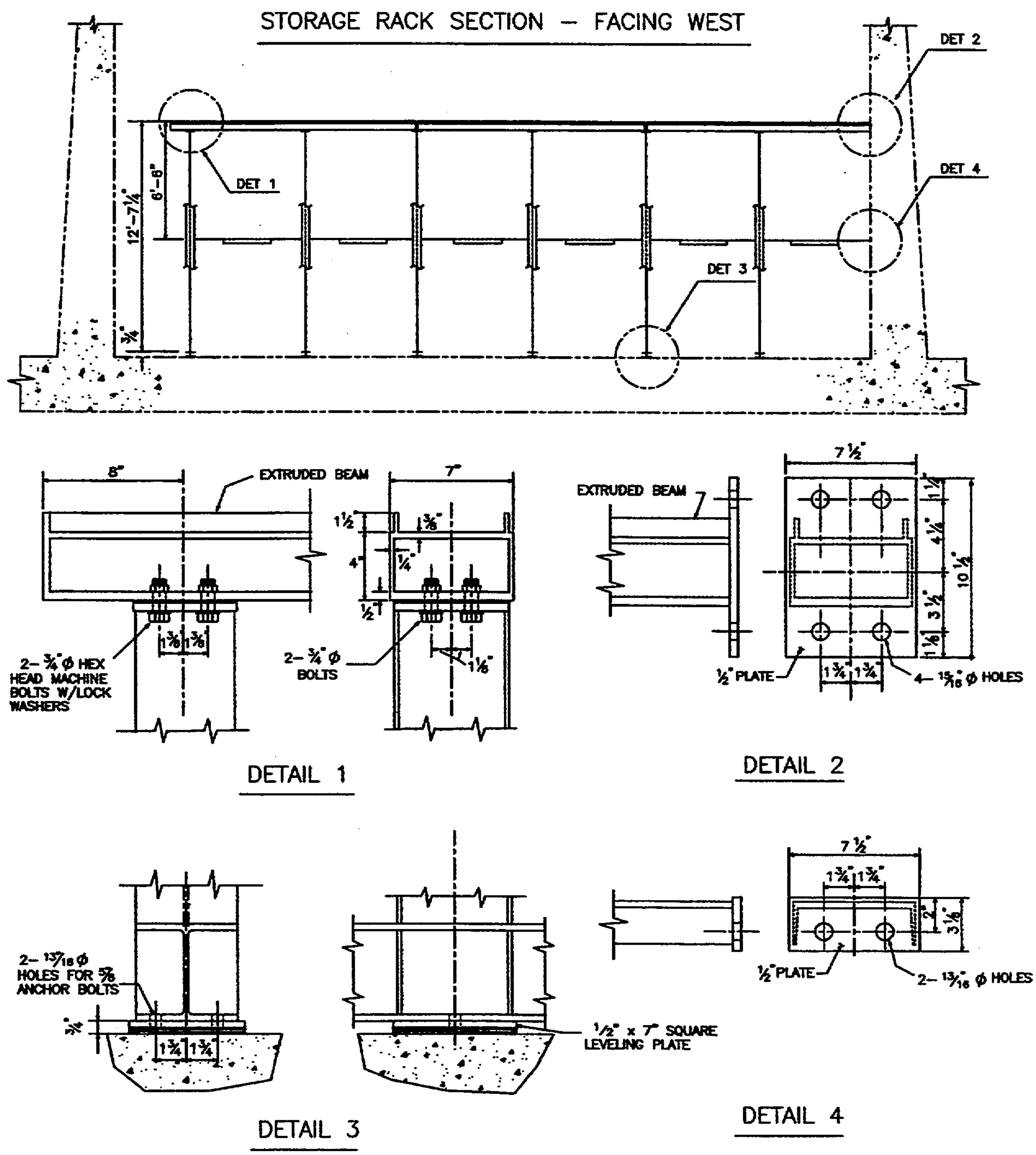
SR12-524.DWG



REF. WVNS DRAWING 900D-4381, Rev. 0
FOR INFORMATION ONLY – NOT TO SCALE

Figure 5.2–4 Radwaste Process Building Plan and Sections

SR12-525.DWG



Ref. NFS Drawing 1A–M–7, Rev. 4

FOR INFORMATION ONLY – NOT TO SCALE

Figure 5.2–5 Fuel Storage Rack 1M–1

SR12-526.DWG

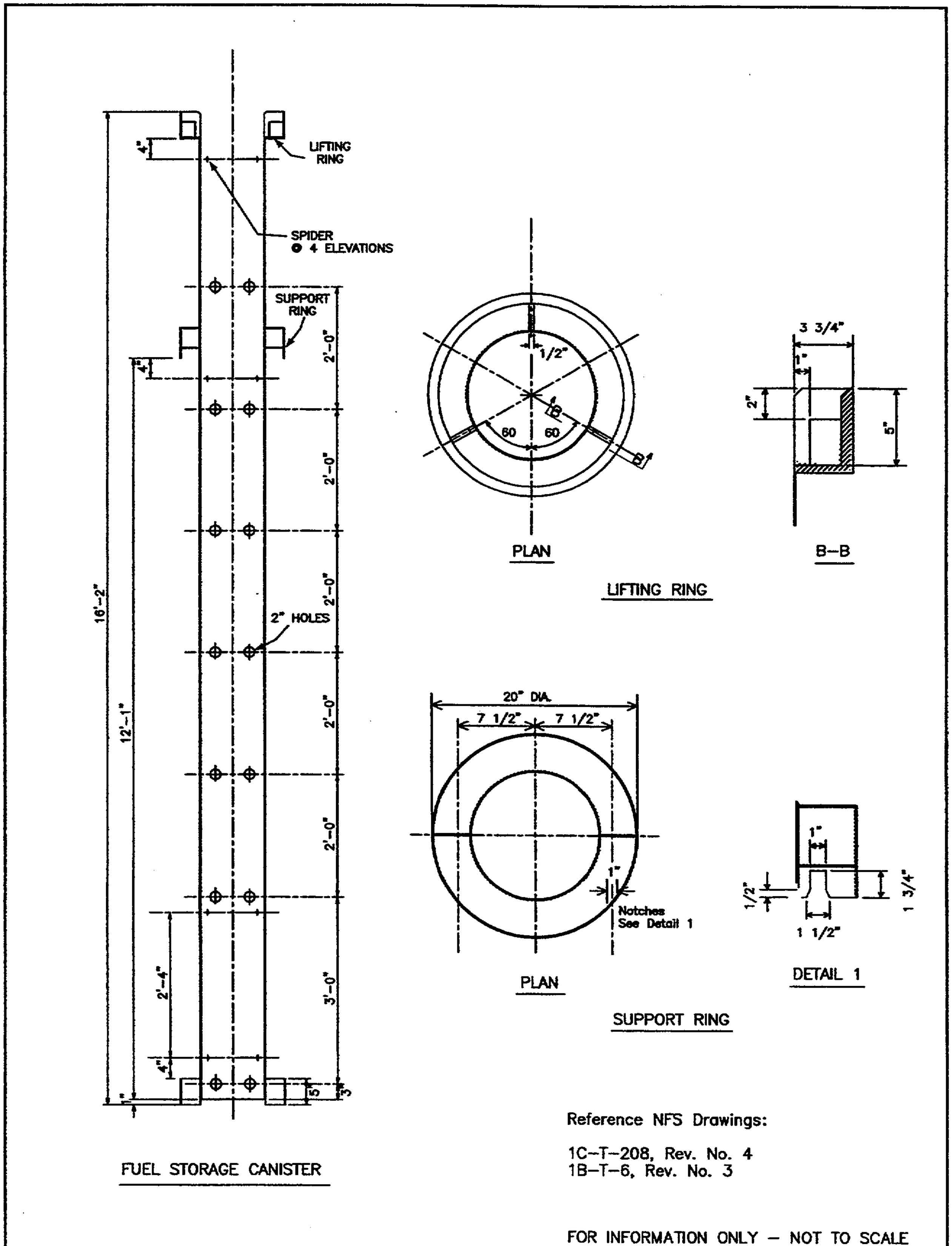


Figure 5.2–6 FRS Fuel Storage Canister

SR12-527.DWG

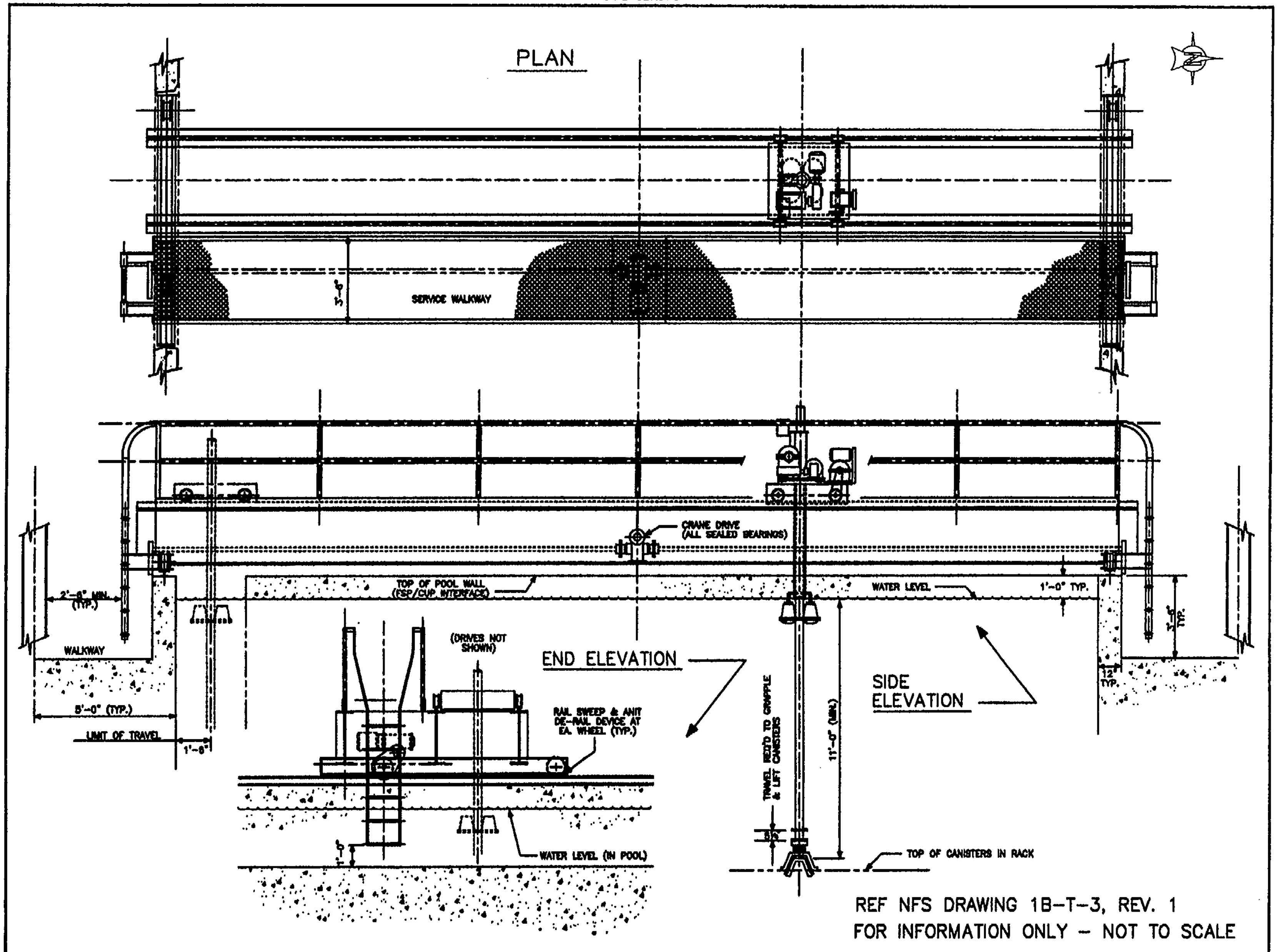


Figure 5.2-7 Fuel Pool Canister Crane 1V-2

SR12-531.DWG

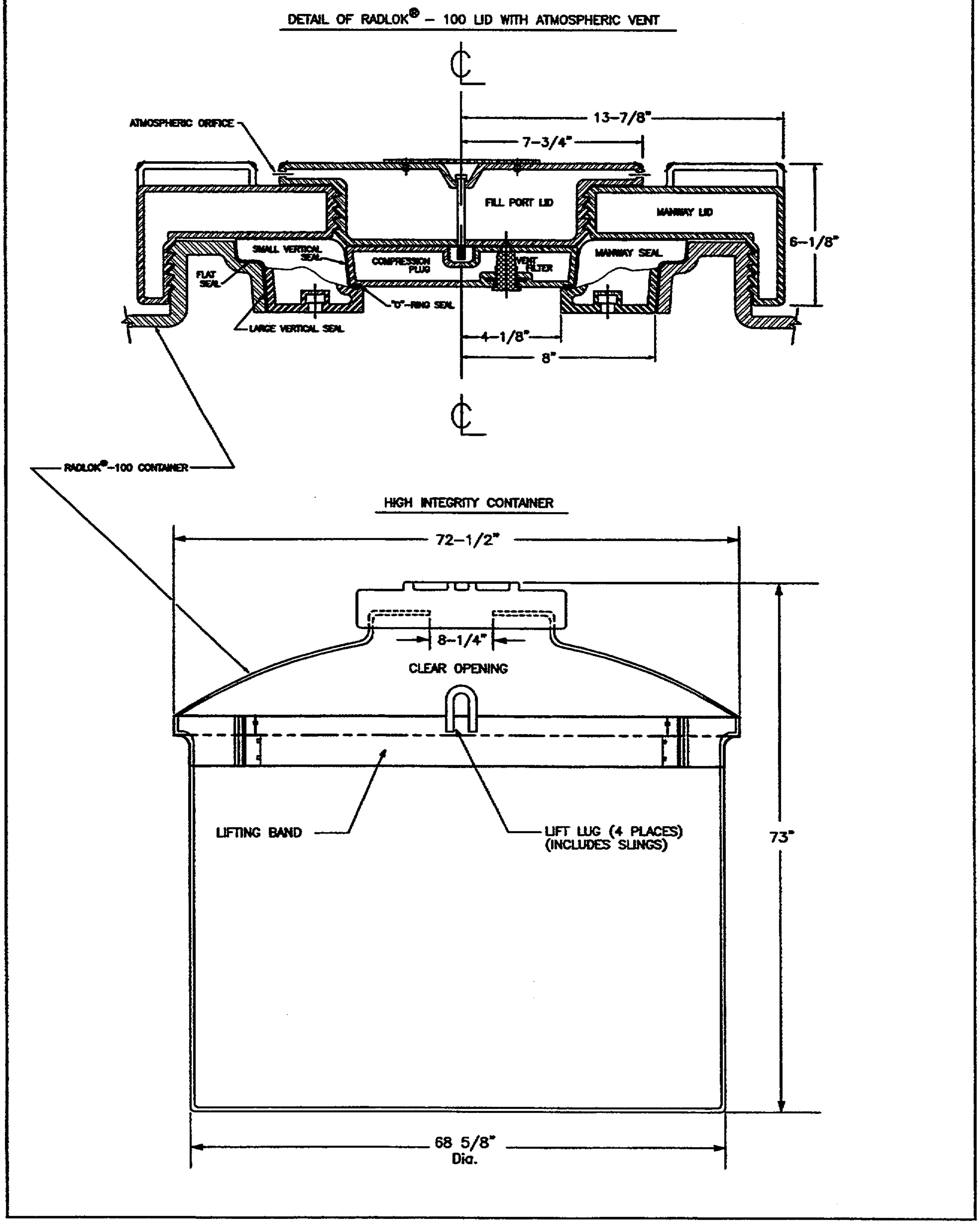


Figure 5.3-1 High Integrity Container

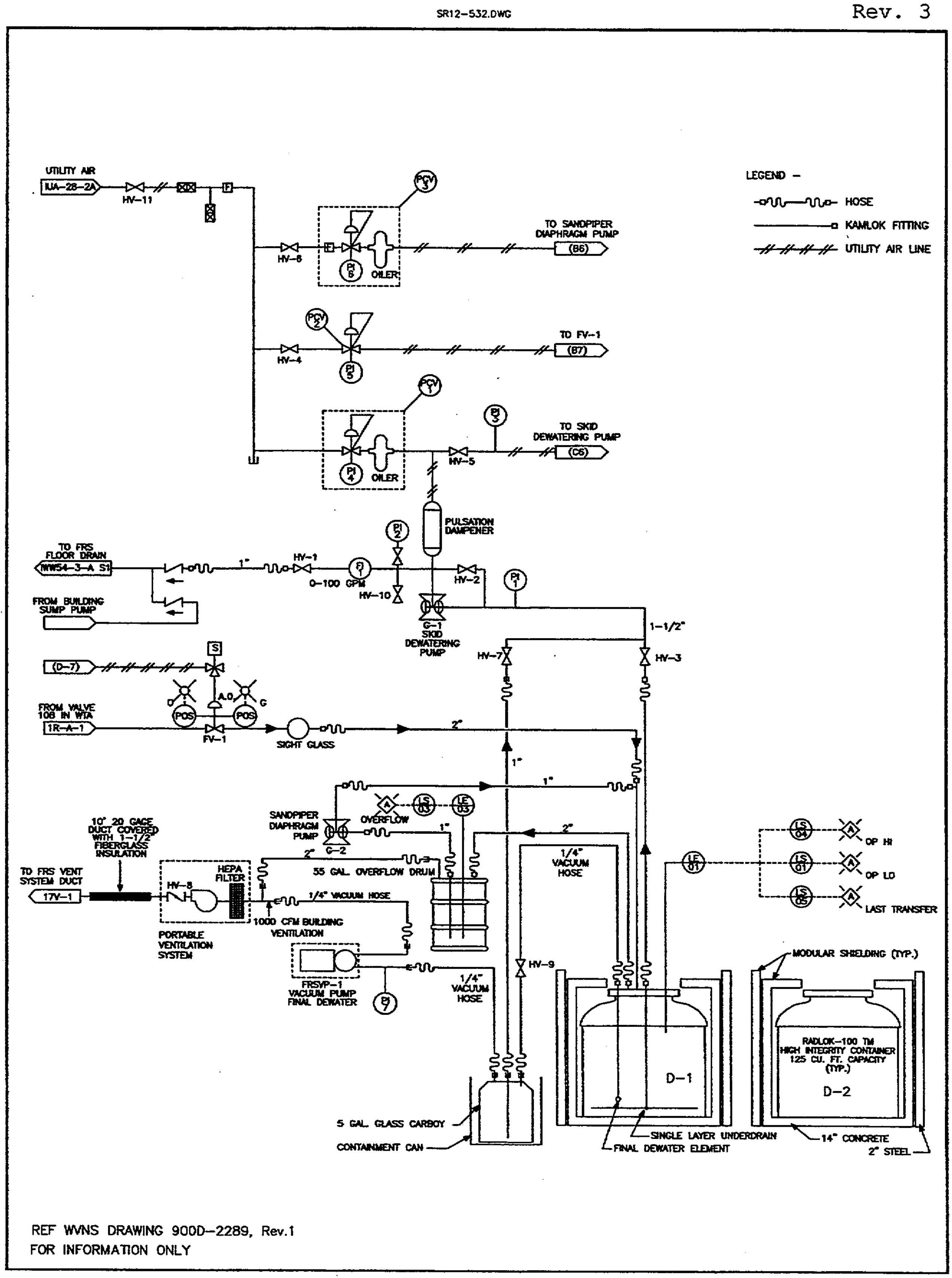


Figure 5.3-2 Radwaste Process System Piping and Instrument Diagram

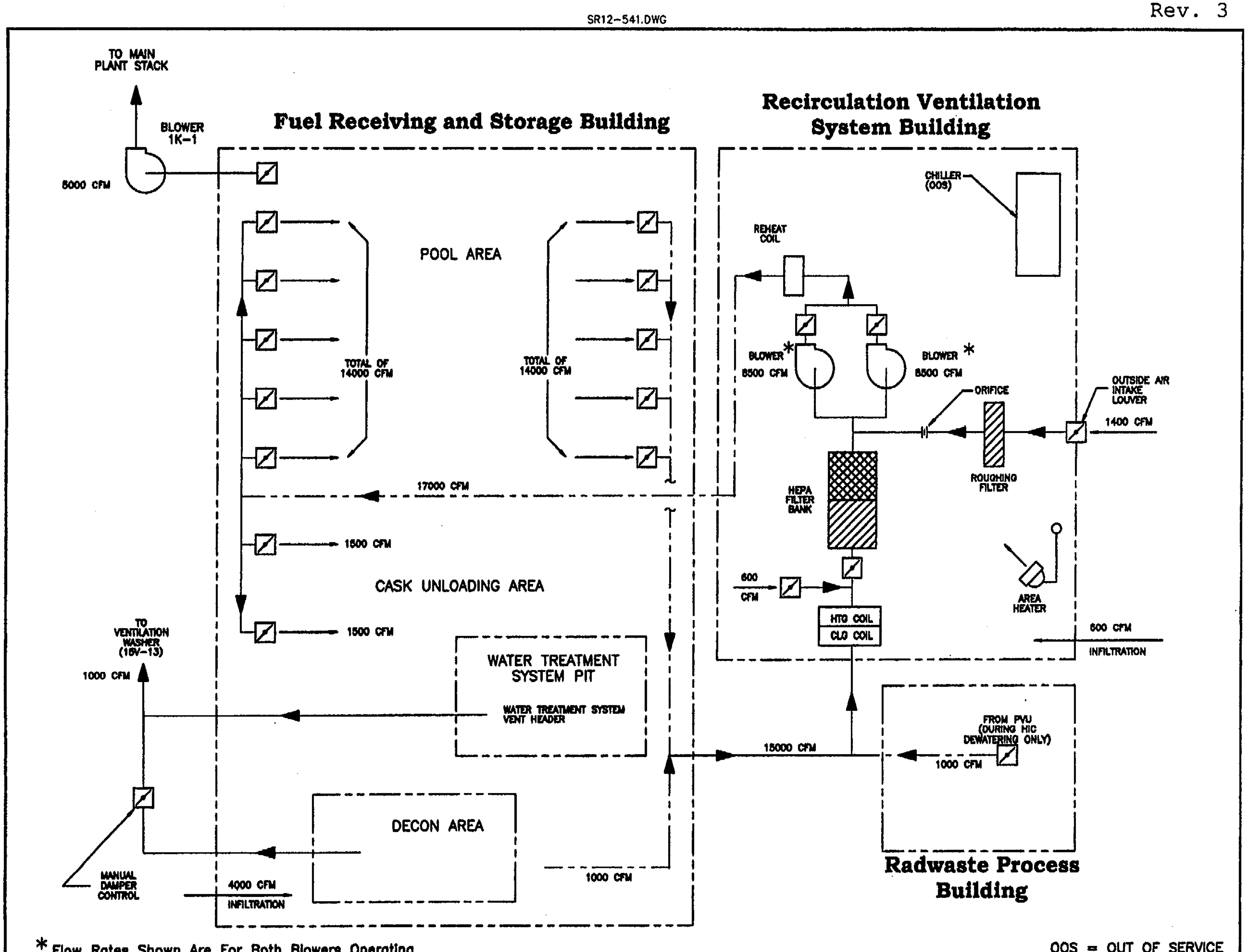


Figure 5.4-1 Fuel Receiving and Storage Facility Ventilation Flow

6.0 PROCESS SYSTEMS

6.1 Process Description

The Fuel Receiving and Storage (FRS) facility houses equipment for the handling, storage, and shipment of spent nuclear fuel (SNF) assemblies. This equipment includes cranes and hoists to move canisters, SNF assemblies, and shipping casks as necessary within the FRS Building. SNF assemblies are typically stored in canisters on storage racks in the fuel storage pool (FSP).

The fuel pool Submerged Water Filtration System and Radwaste Process System support safe storage of the SNF assemblies. The fuel pool Submerged Water Filtration System and Radwaste Process System are discussed in Section 5.3 of this Safety Analysis Report (SAR).

6.1.1 Overview of Fuel Handling for Shipout

SNF assemblies are stored in canisters on racks in the FSP. A shipping cask is placed in the Cask Unloading Pool (CUP) prior to fuel handling for shipment. An NRC-approved shipping cask is placed at the 13.4 m (44 ft) level of the CUP through the use of the 90 MT (100 ton) crane, a lifting beam with 4 m (13 ft) lift arms, and a beam extension. Removal of SNF assemblies from the racks for shipping is accomplished by first placing the fuel pool canister crane bridge and trolley over the canister containing the fuel to be shipped. After the canister has been engaged by the crane it is moved out of the rack and along the storage pool south wall to the CUP. The canister is then placed in the canister lift rack which holds up to four canisters in the same geometry and at the same elevation as in the fuel storage pool racks. The lift rack is lowered from the loading position level to the lower level of the CUP and the fuel hoist (located on the fuel pool service bridge) or specially rigged 5-ton hoist is then brought into position over a canister. The SNF assembly is removed from the canister to a shipping cask (already located in the CUP) using the fuel hoist or specially rigged 5-ton hoist. After transfer of the SNF assemblies to the cask is complete and the lid installed, the cask is lifted out of the pool to the cask decontamination stall using the cask crane. Operational anomalies may necessitate that the loaded cask be placed back in the CUP, the lid removed, adjustments performed and the lid reinstalled. The cask is then surveyed and decontaminated prior to loading on a transport vehicle. The loaded transport vehicle may be removed to the rail siding for final preparations prior to shipping.

6.1.2 Flowsheets

The movement of fuel in the FRS Building to support shipout is illustrated in Figure 6.1-1.

6.1.3 Identification of Items for Safety Analysis Concern

Nuclear Regulatory Commission-approved safety analysis reports for shipping casks currently anticipated for use (WVDP-228 and WVDP-229) address the impacts of normal and accident conditions associated with the use of the casks for shipment of SNF. Hence, this SAR does not evaluate accident scenarios or expected doses to exposed populations during fuel shipment. Evaluations of expected dose rates have been reviewed to ensure that loaded casks do not present an undue radiological hazard to FRS workers.

Operation of the FRS facility involves the storage and handling of SNF assemblies, and the transfer of solid radioactive wastes to appropriate storage containers and locations. The primary items of safety analysis concern are:

- Worker protection from direct radiation and confinement of radioactive material;
- Avoiding nuclear criticality accidents;
- Minimizing the risk of accidents through adherence to established policies and procedures.

6.1.3.1 Radiation Protection

Protection from direct radiation is achieved through shielding, work planning, and remote handling of highly radioactive materials. Confinement barriers and systems (i.e., spent fuel cladding, pool water which is filtered by the fuel pool Submerged Water Filtration System, fuel pool walls and floor, and the silty till layer underneath the pool structure) greatly minimize the likelihood of an uncontrolled release of radioactive materials. Criteria for these systems and barriers are summarized in Section 4.3.2.

6.1.3.2 Prevention of Inadvertent Criticality

Prevention of inadvertent criticality in the FRS is achieved through engineered features and administrative controls. Fuel canisters and storage racks ensure that fuel assemblies are stored in a uniform configuration in the FSP and canister lift rack. Spacing collars on the canister ensure that a sufficient amount of spacing is maintained between adjacent canisters on storage racks. Slots in the collar, which engage with rails on the storage rack, in conjunction with the canister inserts, ensure that fuel storage is possible only in an approved, predetermined orientation. The canister grapple will not release the canister if it is not placed correctly on the storage rack and physical limitations of the grapple prevent a fuel assembly from being raised above another assembly. The fuel assembly lift rack in the CUP maintains the same canister spacing as the storage racks in the FSP. Shipping casks are provided with inserts (as appropriate) and spacers (as appropriate) that maintain the fuel assemblies in a specific geometric configuration. Administrative controls for fuel handling supplement the engineered features to ensure that an inadvertent criticality does not occur within the FRS facility.

6.1.3.3 Management, Organization, and Institutional Safety Provisions

All personnel at the WVDP receive extensive training in safety aspects associated with their responsibilities. Operations involving radioactive or hazardous materials are conducted in a manner consistent with the requirements of Title 10, Code of Federal Regulations, Part 835, and DOE Orders 420.1 and 440.1A. Additionally, an overall safety culture has been developed at the WVDP through a comprehensive implementation of the principles of the DOE Conduct of Operations philosophy as given in DOE Order 5480.19. The implementation of DOE Order 5480.19 at the WVDP, as given in WVDP-106, is summarized in Chapter A.10 of WVNS-SAR-001. Training of operations personnel is conducted per the requirements of DOE Order 5480.20A, which is also discussed in Chapter A.10 of WVNS-SAR-001.

6.2 Fuel Handling

Typical activities associated with shipping cask receipt inspection, handling, preparation, transfer, loading, lid installation, decontamination, and final preparation for off-site shipment are addressed in the shipping cask safety analysis reports. Detailed descriptions of these operations, which are contained in the shipping cask SARs, are summarized in the discussion of fuel shipping operations given below. This safety analysis report provides additional assessments for storage canister handling in the FSP and canister and fuel handling in the CUP.

6.2.1 Description

Transport vehicles used to carry shipping casks to and from the FRS Building enter the building through roll-up doors on the north and south side of the building. Rail tracks, located between the CUP and cask decontamination stall, run north-south through the FRS Building. Hence, shipping casks can be transported either over-the-road or via rail car. It is anticipated that the shipping casks will be transported via rail car.

Prior to cask loading, an inspection of the cask will be performed to check for damage or irregularities. A lift beam, which will be attached to the hook of the cask crane, will engage the lifting trunnions on the side of the cask and the cask will be placed into the cask decontamination stall. Once in the stall the cask lid shall be removed and placed in the loading area of the FRS. When the lid has been removed, the configuration of cask internals (e.g., periphery inserts and fuel replacement inserts) shall be verified to be consistent with the configuration specified in the associated NRC-approved shipping cask SAR (WVDP-228 or WVDP-229). It is essential that the inserts are in the correct configuration as accident and criticality analyses in the cask SARs assume a specific configuration.

When it has been determined that the periphery and fuel replacement inserts are in the correct configuration, the cask is filled with demineralized water and the cask is lifted from the decontamination stall and moved into position over the cask unloading pool where it is then lowered onto the CUP shelf at the 8.8 m (29 ft) level. An extension is then installed onto the lifting beam and the cask is lowered to the 13.4 m (44 ft) level of the CUP. No fuel is permitted in the canister lift rack when a shipping cask is being transferred into or out of the CUP.

Once the shipping cask has been placed at the 13.4 m (44 ft) level of the CUP, SNF assemblies may be transferred from the FSP to the CUP. To transfer fuel assemblies from the FSP, the fuel pool canister crane bridge and trolley are first positioned over the canister containing the fuel to be shipped. Because the canister crane is prevented from raising the canister more than 15 cm (6 in) while in the "pick-up" position, canisters are removed from the racks in a last-on, first-off basis. After the canister has been engaged by the crane it is moved out of the storage rack and along the storage pool south wall to the CUP. The canister is then placed in the canister lift rack which holds up to four canisters in the same geometry as in the FSP racks.

Criticality analyses to be submitted to the Nuclear Regulatory Commission for the full load safety analysis report for the PWR spent nuclear fuel shipping cask have determined that a portion of the PWR assemblies in storage at the WVDP require modification prior to shipment. These modifications involve insertion of poison rods (see Table 6.2-1 for characteristics) into the assemblies to reduce the reactivity of the assemblies under accident (i.e., flooding) conditions. Poison rod installation activities required to satisfy the assumptions of the criticality assessment contained in the PWR shipping cask SAR (WVDP-229) may include:

- Removal of the burnable poison rod assembly, if present, with specially designed handling cage and grapple;
- Cleaning of guide tubes with specially designed vacuum and brush devices;
- Gauging the inside diameter and length of guide tubes with customized device;
- Installing the poison rods (as appropriate) using rod grapple; and

- Reinstalling (as appropriate) the burnable poison rod assembly, if previously removed.

These activities are not needed for half load shipments. Most of the REG fuel contains burnable poison rod assemblies (29 are 12-rod, 8 are 8-rod). The diameter of the rods is 1.10 cm (0.432 in) and they contain annular borosilicate glass pellets. The rod material is stainless steel. One REG assembly contains one REG control rod assembly. The absorber material is 80% silver, 15% indium, and 5% cadmium.

Following all necessary fuel assembly preparation, the lift rack is lowered from the loading level to the lower level of the CUP and the fuel hoist (located on the fuel pool service bridge) or specially rigged 5-ton hoist is then brought into position over a canister. The fuel is removed from the canister to the shipping cask using the fuel hoist or specially rigged 5-ton hoist. At any time, no more than one SNF assembly of any type can be handled outside of its canister or licensed shipping container in the CUP. Additionally, no more than one fuel type is permitted to be present in the CUP at any time. The maximum number of SNF assemblies permitted in a shipping cask at a time is limited to that quantity permitted by the license for that container. During BWR fuel assembly loading, fuel spacers (for Type B, C, and D SNF assemblies) and dummy fuel assemblies are installed as necessary to comply with requirements specified in the NRC-approved SAR for BWR SNF shipping casks (WVDP-228).

When transfer of SNF assemblies to the shipping cask is complete, i.e., specific fuel assemblies loaded into specified cask locations per the applicable SAR (either WVDP-228 or WVDP-229), and the cask lid is in place, four special underwater bolts are used to secure the cask lid prior to removal of the cask from the CUP. The cask is then lifted to the 8.8 m (29 ft) level of the CUP and the beam extension is removed. The top of the cask is then lifted a few inches above the surface of the water and the cask cavity is drained prior to transfer of the cask to the decontamination stall.

Draining of the cask cavity is accomplished through the use of a drain pump attached to a fill/drain connector located near the bottom of the cask. Drain pump discharge is directed to the CUP, FSP, or water treatment area vessel.

Once the cask is loaded, several activities to prepare the shipping cask for shipping will be performed in the decontamination stall. These activities are based on operational considerations and guidelines contained in the NRC-approved SAR for the cask to be shipped and include:

- Decontamination and radiological survey of the cask

- Installation of all lid bolts
- Vacuum drying of the lid gasket interspace
- Vacuum drying of the cask cavity
- Pressure rise leakage rate test of the cask cavity; the upper gas sampling port transport plug seal; the thermocouple and pressure port fittings; the lower gas sampling port transport cover seal; and for assembly verification of the cask closure lid and lid penetration covers
- Nitrogen filling of the cask
- Installation of various test plugs.

A vacuum pump is needed to accomplish some of the activities noted above. The discharge from this vacuum pump is filtered as necessary prior to discharge to the environment to ensure releases are less than those allowed in DOE Order 5400.5. Operational anomalies may necessitate that the loaded cask be placed back in the CUP, the lid removed, adjustments performed and the lid reinstalled.

Upon completion of preparatory activities, the cask is moved from the decontamination stall to the transport vehicle. Prior to cask release for shipment, the loaded transport vehicle may be moved out of the FRS on the rail (inside of the site boundary), where the trunnion tie-downs, front impact limiter spacer, front and rear impact limiters, impact limiter attachment tie rods, security seals, and appropriate Department of Transportation labels and placards will be installed. Final radiation and contamination surveys are also performed.

6.2.2 Safety Features

Safe conditions are maintained during handling operations through administrative controls that restrict fuel movements and locations based on (1) the quantity, type, location, and movement of SNF assemblies occurring at any given time, and/or (2) other activities that may be occurring within the FRS Building at any given time (e.g., installation of the fuel pool gate, placement of a shipping cask into the CUP, etc.). Safe conditions are also maintained during handling operations through the use of several limit switches and/or mechanical stops on key equipment as discussed in Chapter 5.

Continuous air monitors are provided in the FRS facility to detect airborne contamination. During fuel storage, at least one area radiation detector shall be operating so that workers can be made aware of unusual/upset conditions. An area radiation detector that alarms at 20 mrem/hr above background shall be present on the service bridge during fuel handling.

6.3 Sampling-Analytical

Water samples from the FSP are collected monthly and analyzed for the water quality parameters of pH, conductivity, Cs-137 gamma activity, and gross beta activity. Quarterly samples are analyzed for chlorides, nitrates, nitrites, sulfates, gross alpha activity, gross beta activity, and Cs-137 gamma activity. The purpose of this sampling and analysis is to verify the performance of the fuel pool Submerged Water Filtration System in maintaining water quality parameters within desired ranges. Additionally, possible fuel failures can be detected by gross alpha activity measurements. An "outlet" sample is taken from the demineralizer outlet sample line and an "inlet" sample is taken approximately three feet below the pool surface.

Prior to the replacement of a full High Integrity Container (HIC), samples are taken from the top, middle, and bottom so that the waste within the HIC may be classified. A rod, several feet in length, is inserted through a penetration located on top of the HIC. Attached to the bottom end of the rod is a hollow, slotted, right circular cone that, when twisted in the proper direction, collects a sample. The samples are sent to an analytical laboratory where the activity per unit mass is determined for gross alpha and beta activity, total plutonium, and select radioactive isotopes.

WVDP analytical capabilities are described in Section B.6.7.2 of WVNS-SAR-002.

REFERENCES FOR CHAPTER 6

U.S. Department of Energy. February 8, 1990. Change 2 (January 7, 1993). DOE Order 5400.5: *Radiation Protection of the Public and the Environment.* Washington, D.C.: U.S. Department of Energy.

__________. July 9, 1990. Change 1 (May 19, 1992.) DOE Order 5480.19: *Conduct of Operations Requirements for DOE Facilities.* Washington, D.C.: U.S. Department of Energy.

__________. November 15, 1994. DOE Order 5480.20A: *Personnel Selection, Qualification, and Training Requirements for DOE Nuclear Facilities.* Washington, D.C.: U.S. Department of Energy.

__________. October 13, 1995. Change 2 (October 24, 1996). DOE Order 420.1: *Facility Safety.* Washington, D.C.: U.S. Department of Energy.

__________. March 27, 1998. DOE Order 440.1A: *Worker Protection Management for DOE Federal and Contractor Employees.* Washington, D.C.: U.S. Department of Energy.

__________. *Occupational Radiation Protection,* 10 CFR 835.

West Valley Nuclear Services Co., Inc. WVDP-011: *Industrial Hygiene and Safety Manual* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WVDP-106: *Westinghouse Conduct of Operations.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WVDP-228. *TN-BRP Spent Fuel Package Safety Analysis Report for Transport.* (Latest Revision.) Transnuclear, Inc.

__________. WVDP-229. *TN-REG Spent Fuel Package Safety Analysis Report for Transport.* (Latest Revision.) Transnuclear, Inc.

__________. Safety Analysis Report WVNS-SAR-001: *Project Overview and General Information.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. Safety Analysis Report WVNS-SAR-002: *Safety Analysis Report for Low-Level Waste Processing and Support Activities.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

TABLE 6.2-1

PWR POISON ROD CHARACTERISTICS

Poison Rod:	
Stainless steel tube:	
OD (in) -	0.426 (min)
Wall (in) -	0.013 (min)
B_4C Pellet:	
Diameter (in) -	0.388 (min)
Pellet Stack (in) -	141.50
Density (g/cm^3) -	1.72 (min)
^{10}B (atom %) -	19.6 (min)
Total Boron (wt %) -	73 (min)
Total B + C (wt %) -	98 (min)

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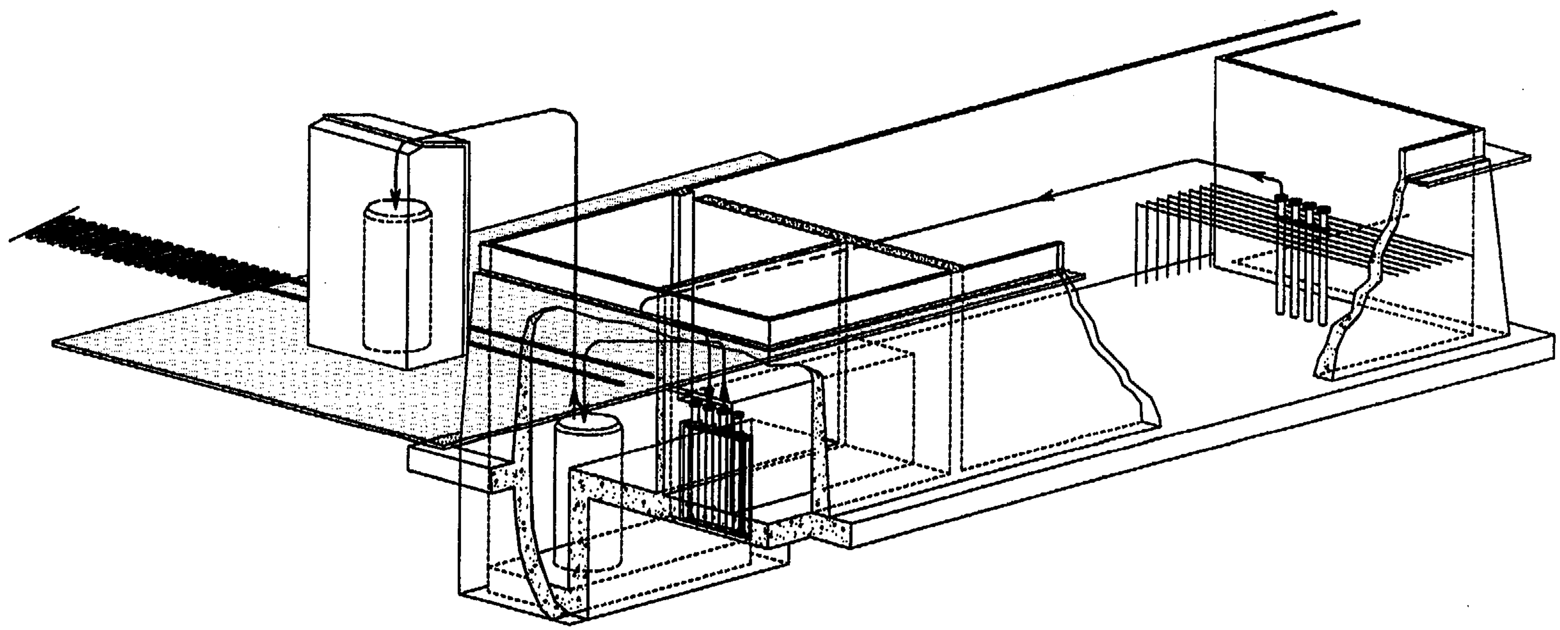

Figure 6.1-1 FRS Flow Diagram

7.0 WASTE CONFINEMENT AND MANAGEMENT

7.1 Waste Management Criteria

Radioactive wastes resulting from Fuel Receiving and Storage (FRS) facility operations include gaseous, liquid, and solid low-level wastes, and transuranic (TRU) waste. Nonhazardous, nonradioactive wastes are also generated in relatively small quantities. (FRS facility systems and operational activities do not require the use of hazardous chemicals. Small quantities of reagents or cleaning solutions may be used periodically for various cleaning, analytical chemistry, or maintenance activities.) Waste handling and processing facilities have been designed to ensure environmental effluent releases are maintained well within discharge guidelines given in DOE Order 5400.5, *Radiation Protection of the Public and the Environment*, and 10 CFR 61, *Licensing Requirements for Land Disposal of Radioactive Waste*.

The West Valley Demonstration Project (WVDP) has developed comprehensive waste management plans to ensure that radioactive, hazardous, mixed, and industrial wastes are handled and stored in compliance with applicable state and federal regulations. A summary of WVDP waste management plans is given in Table B.7.1-1 of WVNS-SAR-002.

7.2 Solid and Liquid Radiological Wastes

The fuel pool Submerged Water Filtration System generates solid radioactive waste in the form of contaminated filter cartridges and loaded ion exchange resin. Filter cartridges and resin are replaced with a frequency established by criteria given in Section 5.3.1 of this SAR. Used filter cartridges are stored in 208 L (55 gal) drums with storage for up to four used filter cartridges provided per drum. High Integrity Containers (HICs) are used for the storage of loaded ion exchange resin (and filter media from the former pool water filter, which has been placed out-of-service). Wastes contained in the first two HICs filled (designated as HICs "A" and "B") have been classified as TRU waste (McVay, 1987). HIC "C" is classified as mixed waste, while HICs "D" and "E" have been classified as low-level waste. Radiological characteristics of HIC contents are given in Table 7.2-1.

Other sources of low-level solid radioactive waste include discarded anticontamination clothing and other personal protective equipment, and items used to support maintenance and decontamination efforts. These wastes, and 208 L (55 gal) drums that contain contaminated Submerged Water Filtration System filter cartridges, are size reduced and/or compacted (if allowable per established criteria) and stored in facilities described in Chapter 7 of WVNS-SAR-002.

Potential sources of radioactive liquid wastes include floor drain and sump effluents, excess pool water, water from cask decontamination activities, and liquid removed during dewatering of the on-line HIC. Liquid from all of these sources is directed to the Low-Level Waste Treatment Replacement Facility (LLW2) for processing. The LLW2 is described in Section B.7.5 of WVNS-SAR-002.

7.3 Nonradiological Wastes

As previously stated, FRS facility operations do not require the use of hazardous chemicals. Small quantities of reagents and cleaning solutions may be used periodically for various cleaning, analytical chemistry, or maintenance activities. Handling of these wastes is performed in accordance with the site Hazardous Waste Management Program (West Valley Demonstration Project, WV-996).

Maintenance and miscellaneous activities generate some nonradiological, nonhazardous wastes (e.g., office trash, packing materials, scrap equipment, sewage, etc.). Nonhazardous, nonradioactive solid wastes are disposed of off-site at a licensed landfill facility. Liquid effluents are regulated by the New York State Department of Environmental Conservation (NYSDEC) for nonradiological parameters.

7.4 Ventilation

No significant sources of airborne contamination exist in FRS facilities. Cask decontamination activities have the potential to generate a small amount of airborne contamination and therefore the cask decontamination stall is ventilated by the Main Plant ventilation system to ensure that airborne releases are controlled and to ensure that air from this area is filtered prior to release to the environment. Air from the former water treatment system equipment continues to be ventilated to the Main Plant ventilation system to maintain confinement of contamination contained within system vessels.

Transfer of resin to the on-line high integrity container in the Radwaste Process Building also presents the potential for airborne contamination release. Consequently, ventilation ducting has been provided on HICs to ensure that off-gases released during resin transfer are passed through the FRS recirculation ventilation filters to remove any effluent contamination.

7.5 Liquid Waste Treatment and Retention

Liquid effluents from the FRS facility are processed by the LLW2 prior to discharge to the environment. The LLW2 is described in Section B.7.5 of WVNS-SAR-002.

7.6 Liquid Waste Solidification

Solidification of byproduct liquid waste is not performed at the WVDP.

7.7 Solid Wastes

Temporary (lag) storage and treatment of solid wastes generated during operation of the FRS facility is provided by the site Lag Storage facilities. These facilities are described in Section B.7.7 of WVNS-SAR-002.

REFERENCES FOR CHAPTER 7

Code of Federal Regulations. Title 10, Part 61: *Licensing Requirements for Land Disposal of Radioactive Waste.*

McVay, Charles. 1987. DD/O Department. *Classification of "B" HIC.* Memo to T. Hughes dated September 16, 1987. EH:87:0089.

U.S. Department of Energy. February 8, 1990. Change 2 (January 7, 1993). DOE Order 5400.5: *Radiation Protection of the Public and the Environment.* Washington, D.C.: U.S. Department of Energy.

West Valley Nuclear Services Co., Inc. 1986. Analytical Request Form: 86-1730.

__________. 1987. Analytical Request Form: 87-1113.

__________. 1987. Analytical Request Form: 87-3131.

__________. 1991. Analytical Request Form: 9101655.

__________. 1992. Analytical Request Form: 9203040.

__________. 1996. Analytical Request Form: 96-0485.

__________. 1993. Analytical and Environmental Lab. *Adjusted Plutonium Results.* Memo to distribution dated April 1, 1993. IH:93:0044.

__________. WV-996: *WVDP Hazardous Waste Management Program.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. Safety Analysis Report WVNS-SAR-002: *Safety Analysis Report for Low-Level Waste Processing and Support Activities.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

TABLE 7.2-1

RADIOLOGICAL CHARACTERISTICS OF HIC CONTENTS

Parameter	HIC "A"(1) (µCi/g)	HIC "B"(2) (µCi/g)	HIC "C"(3) (µCi/g)	HIC "D"(4) (µCi/g)	HIC "E"(5) (µCi/g)	Total Activity(6) (Ci)
Gross α	6.65E-02	3.03E-01	3.49E-02	4.10E-02	6.20E-02	1.62E+00
Gross β	3.37E+01	8.10E+01	3.03E+01	4.24E+01	2.67E+01	6.85E+02
Co-60	-----	1.63E+00	8.37E-02	1.26E-02	3.58E-03	6.04E+00
Sr-90	1.43E+00	6.80E-01	3.36E-01	4.32E-01	7.16E-02	9.43E+00
Cs-137	2.24E+01	6.52E+01	2.97E+01	2.92E+01	2.57E+01	5.51E+02
Pu-238	-----	-----	3.13E-03	3.95E-04	8.74E-05	9.27E-02
Pu-239/240	-----	-----	1.74E-02	2.88E-03	5.90E-04	5.77E-01
Am-241	2.95E-02	5.63E-02	1.56E-02	3.63E-03	1.05E-02	3.70E-01
Total Pu	8.70E-02	9.69E-02	2.05E-02	3.26E-03	6.76E-04	6.67E-01

Notes:

Concentrations based on average of top, middle, bottom samples.

(1) - Ref: Analytical Request Form No. 87-3131.

(2) - Based on average of top, middle, bottom samples, and analysis from separate sample. Ref: Analytical Request Form Nos. 86-1730 & 87-1113.

(3) - Ref: Analytical Request Form Nos. 9101655, 9101656 & 9101657.

(4) - Ref: Analytical Request Form Nos. 9203040,9203041 & 9203042 (as amended by memo# IH:93:0044 [WVNS, 1993]).

(5) - Ref: Analytical Request Form Nos. 96-0485, 96-0486 & 96-0487.

(6) - Based on the sum of activity from all HICs and an assumed mass of sludge of 3.2E+6 g per HIC. Total Co-60 activity based on an estimated concentration of Co-60 in HIC "A". Estimate determined from Cs-137 concentration in HIC "A" and average of Co-60:Cs-137 ratios in HICs "B" through "E". Total Pu-238 and Pu-239/240 activities based on estimated concentrations of Pu-238 and Pu-239/240 in HICs "A" and "B". Estimates determined from Total Pu concentration in respective HIC (A or B) and average of Pu isotopic: Total Pu ratios in HICs "C" through "E".

8.0 HAZARDS PROTECTION

8.1 Assuring that Occupational Hazards Exposures are ALARA

8.1.1 Policy Considerations

A formal documented program directed toward maintaining personnel radiation doses as low as reasonably achievable (ALARA) has been established in West Valley Nuclear Services, Inc. (WVNS) Policy and Procedure WV-984, *ALARA Program.* The ALARA program is based on requirements set forth in Title 10, Code of Federal Regulations, Part 835, and Department of Energy (DOE) Order 5400.5, *Radiation Protection of the Public and the Environment*. The radiation protection program and the ALARA program site-specific requirements are outlined in WVDP-010, *Radiological Controls Manual*, WVDP-076, *Environmental Protection Implementation Plan*, and WVDP-163, *ALARA Program Plan*. Departmental procedures are used to provide more detailed instructions for workers and technical personnel. A discussion and summary of the ALARA program is provided in WVNS-SAR-001, *Project Overview and General Information*.

In addition to radiation protection programs, the WVDP has established a comprehensive industrial hygiene and safety program for the identification, assessment and monitoring of nonradiological hazards. Administration of the industrial hygiene and safety program is through WVDP-011, *WVDP Industrial Hygiene and Safety Manual*, which incorporates the guidance of DOE Order 440.1A, *Worker Protection Management for DOE Federal and Contractor Employees*, as well as DOE adopted Occupational Safety and Health Administration (OSHA) standards 29 CFR 1910 and 29 CFR 1926.

8.1.2 Design Considerations

A prime consideration in maintaining exposure to radioactive materials ALARA is ensuring that adequate confinement of these materials is maintained. Design features of the Fuel Receiving and Storage (FRS) facility that safeguard against the release of radioactive materials include:

- Engineered features, including limit switches and/or mechanical stops on equipment, that ensure an adequate amount of water above fuel assemblies to provide sufficient shielding;

- Remote valving and instrumentation for vessels and components containing radioactive materials;

- A spill control system that includes floor drains and sumps where spills can be collected for further processing;

- Special cask handling equipment that prevents the spread of contaminated pool water onto the hoist equipment and into the FRS Building;

- Shielding of vessels containing highly radioactive materials to reduce exposure rates to acceptable levels;

- High efficiency ventilation filtration systems for FRS Building air and HEPA filtration systems for the cask decontamination stall;

8.1.3 Operational Considerations

In addition to considerations incorporated in facility design, administrative procedures and controls are necessary to ensure that personnel hazards exposures are maintained ALARA. Administrative and procedural control is maintained in accordance with WVDP-011 and WVDP-010. Site operations personnel are fully trained in elements of the radiological control and industrial hygiene programs, as discussed in Chapter A.10 of WVNS-SAR-001.

8.2 Sources of Hazards

Radiological hazards in the FRS facility include the 40 pressurized water reactor (PWR) spent nuclear fuel (SNF) assemblies and 85 boiling water reactor (BWR) SNF assemblies in storage in the fuel storage pool (FSP) and the contaminated ion exchange resin and filter cartridges in the Submerged Water Filtration System equipment or high integrity containers. Radiological characteristics of high integrity container (HIC) contents, including typical isotopic composition and curie content, are listed in Table 7.2-1. Radiological characteristics of PWR and BWR fuel are given in Tables 8.2-1 through 8.2-4.

FRS facility systems and operational activities do not require the use of hazardous chemicals. Small quantities of reagents and cleaning solutions may be used periodically for various cleaning, analytical chemistry, or maintenance activities.

8.3 Hazard Protection Design Features

8.3.1 Radiation Protection Design Features

8.3.1.1 FRS Facility Design Features

Radiation protection features incorporated in the design of FRS facility structures, systems, and components are provided to maintain radiation exposures to members of the general public and work force ALARA. Radiation exposures in the FRS are controlled through design features that provide adequate shielding from all sources of radiation, remote operations and maintenance capabilities, proper ventilation, and effluent control. These physical design features, plus strict adherence to the operational requirements given in WVDP-010, *Radiological Controls Manual*, provide effective radiation control.

8.3.1.2 Shielding

Shielding has been provided in FRS facilities to reduce radiation dose rates to acceptable levels under normal operating conditions. Areas where shielding is not sufficient to reduce radiation levels below the level for uncontrolled access, as required by 10 CFR 835, are posted as Radiation Areas, High Radiation Areas, or Very High Radiation Areas. At the West Valley Demonstration Project (WVDP), areas where a worker can receive greater than 100 mrem (1E-3 Sv) in one year, under full-time occupancy, are posted as Radiological Buffer Areas since personnel dosimetry and monitoring is required by 10 CFR 835 at these levels.

Primary shielding for the SNF in the FRS is provided by the water in the FSP and the cask unloading pool (CUP). Design features of fuel handling equipment, including mechanical stops and limit switches, were provided to meet the shielding requirements of the original design basis fuel (see Table B.8.2-2 of WVNS-SAR-002), which specified that at least 3.35 m (11 ft) of shielding water be maintained above the stored assemblies. This amount of shielding was required to ensure that surface exposure rates did not exceed 1 mrem/hr (1E-5 Sv/hr) above background. Although the original amount of shielding water is maintained in the pool, a significantly lower level is required to achieve the same design basis surface exposure rates due to the low current inventory of the pool (relative to the original design capacity) and the age of the fuel in the pool.

Shielding from the radiation associated with the loaded resin generated in the fuel pool Submerged Water Filtration System is provided by concentric steel and concrete shields that surround the HICs in which the loaded resin is stored. The inner shield is constructed of steel-reinforced concrete with a thickness of 36 cm (14 in). The outer shield is constructed of carbon steel and has a thickness of 5 cm (2 in). A full, unshielded HIC has an estimated contact dose rate of 3 rem/hour to 15 rem/hour (1.5E-1 Sv/hr). A full HIC is lifted out of these concentric shields after removal of the Radwaste Process Building roof and placed in a Surepak located in the north FRS yard for temporary storage. A Surepak is constructed of steel-reinforced concrete with a thickness of 38 cm (15 in).

Contaminated pleated-paper filter cartridges from the fuel pool water filtration system are manually raised from the underwater filter unit and placed into a waste drum that is provided with sufficient shielding to ensure that the contact exposure rate of a full drum does not exceed 100 mrem/hr (1E-3 Sv/hr) (currently no shielding is required). Distance, time, and shielding considerations are integrated into handling operations to ensure that occupational exposures are kept ALARA.

When maintenance is required on contaminated equipment or when decontamination activities require personnel to work in elevated exposure rate areas, supplemental shielding may be used to shield workers from the radiation source and reduce exposure rate levels. Prior to initiation of work activities, the area is surveyed with an exposure rate meter to assure the effectiveness of the additional shielding with stay times established on the Radiation Work Permit (RWP).

8.3.1.3 Ventilation

Ventilation in the FRS facility is provided by three independent systems: the recirculation ventilation system, the Main Plant ventilation system, and an exhaust blower (1K-1). These systems are described in Section 5.4. There are no significant sources of airborne contamination in the FRS and airborne radioactivity levels in the facility are well below the derived air concentrations (DACs) specified in 10 CFR 835.

8.3.1.4 Radiation and Airborne Radioactivity Monitoring Instrumentation

Continuous air monitors are provided in the FRS facility to detect airborne contamination. During fuel storage, at least one area radiation detector is operating so that workers can be made aware of unusual/upset conditions. An area radiation detector that alarms at 20 mrem/hr (2E-4 Sv/hr) above background shall be present on the service bridge during fuel handling.

Radiological monitoring instrumentation used at the WVDP is calibrated in accordance with American National Standards Institute (ANSI) N323-1978. Most radiation detection equipment is calibrated on a six month cycle; however, some instruments are calibrated annually depending on the frequency and type of use, and whether calibration is performed off-site by a service vendor. Stack exhaust monitors and their calibration are discussed in Section B.8.6 of WVNS-SAR-002.

DOE Order 5480.4 and 10 CFR 835 require that monitoring instrumentation comply with the requirements set forth in the applicable American National Standard. The WVDP has implemented these requirements in site service manuals and operating procedures. Audits, appraisals, and surveillances are conducted by external and internal groups at the WVDP to ensure compliance with DOE Orders and DOE-prescribed standards.

Requirements for air monitoring programs are specified in 10 CFR 835. Additional requirements are set forth in DOE-prescribed standards ANSI 13.1 and ANSI 13.6. Air monitoring samples are taken in select locations in the FRS facility to detect and evaluate airborne radioactive material. Data obtained by air monitoring is used for assessing the control of airborne radioactive material in the workplace. The WVDP has incorporated the general guidance for placement of air monitors provided in ANSI 13.1 into the air monitoring program.

8.4 Estimated Collective On-Site Dose Assessment

Activities associated with the FRS facility include the underwater movement of SNF assemblies, cask-related operations, the replacement of loaded resin and used filter cartridges, and support activities that include analytical chemistry, radiological control monitoring, routine maintenance activities, and facility surveillance by security and safety personnel. Worker whole body exposure estimates are calculated as part of the WVDP ALARA program. Dosimetry data for work groups involved with supporting current fuel storage operations indicate that the combined occupational dose for all individuals in all associated work groups is less than 30 mrem/month (3E-4 Sv/month).

It is expected that activities associated with the cask loading of SNF assemblies for transport off-site will result in higher occupational doses than those received during present custodial activities due to the nature of the respective activities. Dose estimates from fuel shipping activities in 1983 indicated that a total of 46 man rem was received during the cask loading operations of 625 SNF assemblies in storage at the time (WVNS, 1987). Scaling this exposure to the 125 assemblies currently in storage results in a projected dose of approximately 9 man-rem for cask loading operations. It is reasonable that the total dose for this activity should be lower due to the greater post-reactor decay time that this fuel has experienced.

A program of air particulate monitoring is in place for the FRS facility to ensure airborne radioactive material levels in routinely occupied areas are well within acceptable limits. This is accomplished by drawing facility air at a constant rate through glass fiber filters. WVNS-SAR-002 states that radiological analyses of these types of filters located in the Main Plant and Waste Processing Facilities indicate typical airborne radioactivity concentrations of 1E-15 µCi/mL gross alpha and 1E-15 to 1E-14 µCi/mL gross beta with occasional gross beta concentrations of 1E-13 µCi/mL measured in certain areas. With this data, WVNS-SAR-002 performs a conservative analysis to estimate the annual inhalation dose per worker. The results yield an annual estimated inhalation dose per worker of 1.6 mrem (1.6E-5 Sv). Annual doses to FRS facility workers due to airborne activity are assumed to be comparable or significantly less.

8.5 Hazards Protection Programs

8.5.1 Integrated Safety Management System

WVNS systematically integrates safety into management and work practices at all levels so that missions are accomplished while protecting the public, the worker, and the environment. This integration is accomplished by implementing an Integrated Safety Management System (ISMS). The DOE has developed seven guiding principles to provide the focus for implementing an ISMS. While these principles guide the implementation of an ISMS, five core functions define its make-up. These functions comprise a cycle of activities which, although different in detail, are the same for activities on a program or site level and a facility and work task level.

8.5.2 WVDP Health Physics Program

A formally documented health physics program for the WVDP has been established in WVNS Policy and Procedure WV-905, *Radiological Protection.* The health physics program is based on requirements set forth in 10 CFR 835 and DOE Order 5400.5. At the WVDP, the health physics program's site-specific requirements are promulgated in WVDP-010. The FRS facility is operated in compliance with the requirements given in WVDP-010. The health physics program for the Project is discussed and summarized in Section A.8.5 of WVNS-SAR-001.

8.5.3 Industrial Hygiene and Safety Program

A comprehensive industrial hygiene and safety program based on the requirements of DOE Order 440.1A, *Workers Protection Management for DOE Federal and Contractor Employees,* and OSHA standards 29 CFR 1910 and 29 CFR 1926 has been developed for the WVDP. Administration of this program is through procedures developed from WVDP-011.

An industrial hygiene program addresses many topics in addition to hazardous chemicals. These include illumination, noise, temperature, confined space entry, nonionizing radiation, and sanitary requirements.

8.6 Estimated Collective Off-Site Dose Assessment

8.6.1 Effluent and Environmental Monitoring Program

A comprehensive environmental monitoring program is in place at the WVDP to monitor site activities and their possible impact to the environment. Details concerning this program can be found in Section A.8.6.1 of WVNS-SAR-001.

8.6.1.1 Gas Effluent Monitoring

Air drawn from the cask decontamination stall and air discharged from exhaust blower 1K-1 is combined with Main Plant ventilation exhaust gases and is discharged to the environment through the Main Plant stack. (Air from the cask decontamination stall is HEPA filtered prior to release to the environment.) Main stack effluent monitoring is discussed in Section B.8.6.1.1 of WVNS-SAR-002.

8.6.1.2 Liquid Effluent Monitoring

Liquids that are generated during FRS facility operations are processed through the Low Level Waste Treatment Replacement Facility (LLW2) before discharge to the environment. The monitoring of LLW2 discharges is described and evaluated in Section B.8.6 of WVNS-SAR-002.

8.6.2 Analysis of Multiple Contribution

Contributions to off-site dose due to other nearby nuclear facilities is given in Section A.8.6.2 of WVNS-SAR-001.

8.6.3 Estimated Exposures from Airborne Releases

As previously noted, air drawn from the cask decontamination stall and air discharged from exhaust blower 1K-1 is discharged to the environment through the Main Plant stack. As reported in Section B.8.6.3 of WVNS-SAR-002, the total effective dose equivalent (TEDE) to the maximally exposed off-site individual (MEOSI) in 1998 was calculated to be 3.4E-02 mrem/yr (3.4E-07 Sv/yr) for airborne discharges from all stacks. The MEOSI is located at approximately 1800 m (5900 ft) northwest of the Main Plant stack.

8.6.4 Estimated Exposures from Liquid Releases

Effluents from the FRS facility are combined with other Project liquid effluents and treated at the LLW2 prior to discharge to the environment. As reported in Section B.8.6.4 of WVNS-SAR-002, the MEOSI TEDE from all liquid effluents from the WVDP in 1998 was calculated to be 7.4E-03 mrem/yr (7.4E-8 Sv/yr).

8.7 Prevention of Inadvertent Criticality

The PWR and BWR fuel assemblies stored in the FRS were designed to sustain criticality when placed in a suitable configuration with other fuel assemblies. Consequently, operations involving the handling and storage of SNF assemblies must be evaluated for criticality safety. Analyses demonstrate that expected configurations (i.e., configurations associated with normal operations) do not support criticality. Credible accident-induced or inadvertently-created configurations are examined in analyses referenced in Section 8.7.3 of this SAR.

Criticality safety in the FRS Building is achieved through engineered features (such as the design of canisters that ensure that sufficient spacing is provided between SNF assemblies to prevent significant neutron interaction with adjacent assemblies), and application of strict administrative controls. These controls restrict fuel movement and location based on (1) the quantity, type, location, and movement of SNF assemblies occurring at any given time, and/or (2) other activities that may be occurring within the FRS Building at any given time (e.g., installation of the fuel pool gate, placement of a shipping cask into the CUP, etc.). Safe conditions are also maintained during handling operations through the use of several limit switches and/or mechanical stops on key equipment as discussed in Section 5.2. See Section B.8.7.5 of WVNS-SAR-002 for a discussion of the Criticality Protection Program at the WVDP.

8.7.1 Introduction

Operations involving the handling and storage of SNF assemblies are evaluated for criticality safety. Criticality safety in the FRS is achieved through engineered and administrative controls. Evaluations referenced in Section 8.7.3 have shown that there currently is no credible potential for an inadvertent criticality during the storage of fuel in either the FSP or CUP or during the handling of fuel for shipping cask loading.

8.7.2 Requirements

Criticality safety at the WVDP is maintained through the management policy established in WV-923, *Nuclear Criticality Safety,* and adherence to the requirements set forth in WVDP-162, *WVDP Nuclear Criticality Safety Program Manual.* WVDP-162 implements the requirements of DOE O 420.1, Attachment 2, Contractor Requirements Document, *Facility Safety,* and incorporates the elements of the following mandatory American National Standards of the American Nuclear Society (ANSI/ANS) pertaining to nuclear criticality safety:

- ANSI/ANS-8.1-1983, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors,* (with paragraphs 4.2.2 and 4.2.3, and paragraph 3.3 modified as directed in Section 4.3.2.d of DOE O 420.1, Attachment 2);

- ANSI/ANS-8.3-1986, *Criticality Accident Alarm System,* (with paragraphs 4.1.2, 4.2.1 and 4.2.2 modified as directed in Section 4.3.2.c of DOE O 420.1, Attachment 2);

- ANSI/ANS-8.5-1986, *Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material;*

- ANSI/ANS-8.6-1983,R88, *Safety in Conducting Subcritical Neutron-Multiplication Measurements in Situ,* (with paragraph 5.3 modified as directed in DOE O 420.1, Attachment 2);

- ANSI/ANS-8.7-1975,R87, *Guide for Nuclear Criticality Safety in the Storage of Fissile Materials,* (with paragraph 5.2 modified as directed in Section 4.3.3.c of DOE O 420.1, Attachment 2);

- ANSI/ANS-8.9-1987, *Nuclear Criticality Safety Criteria for Steel-Pipe Intersections Containing Aqueous Solutions of Fissile Materials;*

- ANSI/ANS-8.10-1983,R88, *Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement;*

- ANSI/ANS-8.12-1987,R93, *Nuclear Criticality Control and Safety of Plutonium-Uranium Fuel Mixtures Outside Reactors;*

- ANSI/ANS-8.15-1981,R87, *Nuclear Criticality Control of Special Actinide Elements;*

- ANSI/ANS-8.17-1984,R89, *Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors,*(with paragraph 4.3 modified as directed in Section 4.3.2.g of DOE O 420.1, Attachment 2);

- ANSI/ANS-8.19-1984,R89, *Administrative Practices for Nuclear Criticality Safety;*

- ANSI/ANS-8.21-1995, *Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors.*

Notification, investigation, and reporting requirements are in accordance with DOE Orders 232.1A, *Occurrence Reporting and Processing of Operations Information,* and 231.1, *Environment, Safety and Health Reporting.*

8.7.3 <u>Criticality Concerns</u>

The FRS has been designed for the storage and handling of SNF assemblies. These assemblies contain fissionable material in excess of that required to support a criticality if placed in an optimum array and therefore represent a criticality concern. To address this concern, the FRS has been provided with engineered and administrative controls to prevent an inadvertent criticality. The results of criticality safety analyses that have been performed by several organizations to assess the criticality safety of fuel stored in the FRS FSP conclude that a criticality in the FRS is incredible during all normal storage and handling conditions and under all credible inadvertent handling and accident conditions. These analyses are described below.

Original criticality safety criteria for the design of fuel storage canisters appear to be based on information provided by utilities, as no independent criticality calculations for design of this equipment appear in the original Nuclear Fuel Services (NFS) Safety Analysis Report (SAR) for fuel storage (NFS, 1964). An analysis was performed by NFS in 1973, however, in support of a request to the AEC for a revision to the existing Technical Specifications that would allow for a more dense packing of fuel assemblies in the storage pool (WVNS, 1983, Attachment 3). In order to ensure that a conservative assessment was performed that would bound the types of fuel that were anticipated to be reprocessed in the Main Plant, NFS selected the Westinghouse 15x15 PWR assembly and double batched General Electric 7x7 BWR assemblies for the analysis. These assemblies represented the most reactive of the two types of light water reactor fuel to be stored in the FRS. The more reactive of these assemblies (i.e., the 15x15 PWR) is larger and possesses a higher U-235 enrichment than assemblies currently in storage (4 wt% U-235 versus 3.48 wt% U-325), and thus present a conservative bound for arrays of fuel currently in storage.

Calculations were performed to determine the maximum k_{eff} for arrays of each fuel type under varying conditions of U-235 enrichment and fuel-to-moderator ratio. All calculations were performed assuming unirradiated fuel. Figures 8.7-1 and 8.7-2 illustrate the arrays of fuel in a standard storage configuration (i.e., faces parallel) as well as in an optimum rotated configuration (i.e., faces rotated 45°) that were evaluated. It was found that even with a face-to-face separation of 20.3 cm (8 in), a $k_{eff} + 2\sigma$ of 0.95 is not exceeded. The results of these analyses are summarized in Table 8.7-1.

A second criticality analysis of fuel in the FSP was prepared by WVNS in 1983 (WVNS, 1983, Attachment 8), in support of the off-site shipment of 625 of the 750 fuel assemblies in storage at the time. This analysis evaluated the reactivity of unirradiated PWR and BWR assemblies, and combinations of these assemblies, under both normal and accident configurations. Illustrations of the fuel in the abnormal or accident configurations evaluated in this assessment are provided in Figures 8.7-3 and 8.7-4. As in the earlier NFS assessment, the 15x15 Westinghouse PWR assembly and the 6x6 Dresden-I BWR assembly were selected for analysis. The results of these analyses, which are summarized in Table 8.7-2, also concluded that a $k_{eff} + 2\sigma$ of 0.95 would not be exceeded under all normal and credible accident conditions.

For the present analysis, additional criticality analyses have been performed by Square Y Consultants (Yuan, Yuchien, 1996) to further evaluate the reactivity of PWR and BWR assemblies under both normal and accident conditions. This analysis evaluated the reactivity of PWR and BWR fuel assemblies in array configurations, as well as under configurations anticipated during handling of fuel for shipment. For consistency with the criticality assessments contained in the NRC-approved SARs for SNF shipping casks, the characteristics of the fuel assumed in the Square Y assessment are the same as those used in the cask SARs (WVDP-228, Rev. 5 and WVDP-229, Rev. 6).

As part of the assessment, a parametric analysis was performed to determine the reactivity of pairs of uncanistered assemblies as a function of assembly separation in order to determine the distance between assemblies that coincided with optimum reactivity. It was found through this analysis that the k_{eff} of two adjacent PWR fuel assemblies will not exceed unity, even under optimum conditions of spacing, configuration (i.e., faces parallel), and burnup (i.e., unirradiated fuel). In order to determine the minimum parameters that are necessary for criticality, a separate analysis was performed that determined that a linear array of at least three PWR assemblies is required for criticality. This configuration is shown in Figure 8.7-5.

The analysis by Square Y also included an assessment to determine the reactivity of fuel assemblies under credible accident conditions. The assessment found that for the most reactive credible accident configuration, which is shown in Figure 8.7-6, a $k_{eff} + 2\sigma$ of 0.95 is not exceeded. A summary of the conclusions contained in the Square Y report is given in Table 8.7-3.

Evaluation of the criticality safety of BWR and PWR fuel while in storage in the NRC-approved shipping casks has been provided in the NRC-approved safety analysis reports for these casks (WVDP-228 and WVDP-229, respectively). These analyses evaluate the criticality safety of the SNF while in normal transportation configurations and under credible and incredible accident conditions. Assessments of inadvertent fuel configurations due to accident conditions described in these criticality analyses bound any credible inadvertent loading configuration resulting from operator error.

The analyses referenced in this section indicate that a single assembly is subcritical under all normal operating and storage conditions and that the storage array is subcritical by a large margin. The analyses by NFS and Square Y indicate that a significantly denser packing of fuel in the FSP could be achieved without exceeding a $k_{eff} + 2\sigma$ of 0.95. Evaluations referenced in this section have also demonstrated that credible accident-induced or inadvertently-created configurations of fuel assemblies in the FSP, CUP, or shipping cask will not result in a critical condition. It is possible that if a credible accident scenario in which fuel slumping or crushing could be postulated, a more reactive configuration could result.

However, this vulnerability has been extensively evaluated in the reports referenced in Section 5.2 and no credible mechanism for fuel crushing has been identified.

8.7.4 Criticality Controls

Engineering and administrative controls are provided for FRS facilities and operations to ensure that the occurrence of an inadvertent criticality is prevented throughout the course of normal activities and accident conditions. Administrative controls for the prevention of an inadvertent criticality at the WVDP are developed through the guidelines given in WVDP-162 and the references contained therein.

8.7.4.1 Engineering Controls

Equipment in the FRS has been designed to ensure that a subcritical array is maintained in the FSP and CUP. Fuel stored in the FRS is contained in storage canisters that have been provided with spacing collars that ensure that a minimum distance of 19 cm (7.5 in.) between fuel assemblies is maintained. These spacing collars have been provided with slots that engage a channel on the storage rack to ensure that canisters maintain a specific orientation to the support beam while on the storage rack. Inserts may exist in a given canister that facilitate the centering of the fuel assembly in the canister and that restrict rotation of the assembly in the canister.

The grapple on the fuel pool canister crane has been provided with mechanical stops to limit the height to which fuel canisters can be raised. While the original intention of this design was to provide assurance that canisters could not be raised to a level that would result in unacceptable surface exposure rates, it also prevents a full canister from being raised above another full canister. As indicated in Section 8.7.3, a fuel assembly above another fuel assembly does not result in a $k_{eff} + 2\sigma > 0.95$.

Equipment within the CUP has also been provided with features to prevent criticality. The canister lift rack in the CUP has been designed to maintain fuel in canisters in the same configuration as in the storage racks in the FSP. The canister lift rack has also been provided with mechanical stops to ensure that even in the event of a catastrophic failure of the lift rack cable, the fuel canisters would not contact the CUP floor.

8.7.4.2 Administrative Controls

Administrative controls developed through the guidelines and requirements given in WVDP-162 along with the engineered design features provide the means for criticality control in FRS facilities. These controls ensure that activities which require the storage, processing or handling of fissile or fissionable materials are performed in a manner that provides an acceptable margin to safety for the prevention of an inadvertent criticality.

Accessible areas of WVDP facilities for which administrative controls must be maintained to preclude an inadvertent criticality as a result of the form, quantity or concentration of stored fissile or fissionable material are designated as a criticality control zone. Criticality control zones are posted to indicate a definite boundary and provide a means of accounting for and controlling fissionable material inventory in the designated location. Both the FSP and the CUP have been designated as criticality control zones.

8.7.4.3 Application of Double Contingency

The FRS has been designed for the storage and handling of SNF. Operations in the FRS will not alter the physical characteristics or dimensions of the fuel. Therefore, characteristics of the fuel such as mass, enrichment, and geometry are fixed. In addition, water shielding in the FSP requires that moderation and reflection must be considered to be optimum. Consequently, maintaining a safe spacing of the assemblies in the storage array is the primary means for providing an assurance of subcriticality in the FRS.

The double contingency principle is satisfied in the FRS through the engineering and administrative controls described in Sections 8.7.4.1 and 8.7.4.2. In order for a criticality to occur in the FRS, at least three fuel assemblies would need to be brought into close proximity to each other. Fuel in the FRS is stored in canisters that maintain safe spacing. The canister crane in the FSP is not capable of handling fuel and the fuel hoist on the service bridge cannot be brought over the FSP. It is therefore not possible to handle more than one uncanistered assembly at a time. In summary, the double contingency principle is satisfied by having two controls, one passive engineered and the other administrative, over a single parameter, i.e., the spacing between fuel assemblies (Lazzaro, J.A., 1999).

8.7.5 Criticality Protection Program

Criticality safety at the WVDP is implemented through the requirements of WVDP-162, *Nuclear Criticality Safety Program Plan.* Subsections of this section provide general information regarding the WVDP criticality safety program with added detail for features of the program which apply specifically to facilities and operations within the scope of this SAR.

8.7.5.1 Criticality Safety Organization

Administration of the criticality safety program at the WVDP is through the WVNS Safety Analysis and Integration (SA&I) Department. The SA&I Manager is responsible for monitoring and implementing nuclear criticality safety requirements, assisting operating management in developing programs and plans for maintaining nuclear criticality safety of the plant by regular evaluations and assessments in work areas. The SA&I Manager is responsible for developing and maintaining the criticality safety program manual and for concurring with the establishment and abolishment of criticality control zones and for criticality control zone management. Additional responsibilities of the SA&I manager are listed in WVDP-162 and WV-923.

A Criticality Safety Engineer (CSE) is responsible for establishing and abolishing criticality control zones and their operating limits and is responsible for performing nuclear criticality safety evaluations for activities conducted at the WVDP. In addition, a CSE provides programmatic evaluation to ensure that fissionable materials are packaged in a manner that protects worker health and safety and the environment and to ensure that nuclear criticality safety evaluations are performed to identify potential accumulations of fissionable material during production, storage, transport and handling. A CSE is responsible for developing controls for fissionable material accumulations to reduce the risk of accidental criticality.

8.7.5.2 Criticality Safety Plans and Procedures

Operations at the WVDP where nuclear criticality safety is a consideration are governed by written plans and procedures for initial planned operations and for subsequent modifications that may affect reactivity. Documented plans and procedures are provided for storing, processing and handling of fissionable materials. Modifications to these plans and procedures are subject to an Unreviewed Safety Question Determination to assess any potential impact to the approved authorization basis.

Accessible areas of facilities that contain significant quantities of fissionable material or that provide for storage, processing or handling of fissionable materials which require administrative controls to preclude an inadvertent criticality are designated as a Criticality Control Zone. These zones are prominently identified with criticality control zone signs posted at all anticipated avenues of approach with clearly marked boundaries.

8.7.5.3 Criticality Safety Training

A criticality safety training program has been developed at the WVDP in accordance with the requirement of DOE Order 5480.20A. Criticality safety training is given to individuals who operate, maintain, and/or supervise activities in areas where significant quantities of fissionable materials are stored or handled. Elements of the training program require that each individual receive instruction in nuclear criticality safety including a summary of criticality accident history and nuclear criticality theory, normal procedures, radiation control practices, configuration control, criticality control zones, procedural compliance, and individual responsibility.

8.7.5.4 Determination of Operational Nuclear Criticality Limits

Operational nuclear criticality limits at the WVDP are developed based upon considerations of approved nuclear criticality safety evaluations. At the WVDP these evaluations are primarily performed using the KENO-V.a code and various cross section data provided by the Radiation Shielding Information and Computation Center (RSICC) at Oak Ridge National Laboratory. Prior to use at the WVDP, the KENO-V.a code is verified on each computing platform on which it will be used following standard site computer code verification procedures. Verification and validation guidance and information related to KENO-V.a are provided in NUREG/CR-6483, *Guide to Verification and Validation of the SCALE-4 Criticality Safety Software*. NUREG/CR-6483 concludes that for low-enriched U-235 systems there is an average bias that ranges from approximately -0.01 to +0.01 Δk depending on the system being analyzed. The results for highly enriched U-235 systems indicate an average bias ranging from -0.02 to +0.025 Δk depending on the system being analyzed. The results for U-233 systems indicate an average bias ranging from -0.02 to +0.045 Δk and for Pu-239 systems, a range of approximately +0.01 to +0.035 Δk, depending on the system being analyzed with many individual systems calculating nearly unbiased.

Safety margins for all calculations performed for WVDP activities and systems are established such that the calculated effective neutron multiplication factors, including all computational uncertainties for a unit, array of units, or systems containing fissionable material is less than 0.95, within a 95 percent probability and 95 percent confidence level (i.e., $k_{eff} + 2\sigma \leq 0.95$, where σ is the uncertainty associated with the method of calculation).

Analyses used for the development of operational limits are reviewed by the WVDP Radiation and Safety Committee in accordance with WV-906 and WV-923. Furthermore these analyses are independently reviewed by individuals whose education and experience meet or exceed the requirements of a criticality safety engineer.

8.7.5.5 Criticality Safety Inspection/Audits

The WVDP SA&I Manager is responsible for ensuring that independent appraisals are performed in accordance with WV-121. Appraisals review and evaluate nuclear criticality safety against DOE orders, federal and management requirements, Technical Safety Appraisal criteria listed in DOE/EH-0135 or latest DOE requirements, as well as good and best management practices.

8.7.5.6 Criticality Infraction Reporting and Follow-Up

Occurrence reporting requirements dictated by DOE O 232.1A, *Occurrence Reporting and Processing of Operations Information*, are implemented at the WVDP through WVNS Manual WVDP-242, *Event Investigation and Reporting Manual*. This manual establishes a system for determining, evaluating, reporting, and correcting occurrences.

As prescribed in the procedure, the Facility Manager is responsible for evaluating and categorizing occurrences, including criticality infractions, and completes oral notification per DOE requirements when determined applicable. Furthermore, the Facility Manager is responsible for ensuring that the corrective actions proposed and implemented as a result of an occurrence are adequate, and approves the closeout of identified corrective action items resulting from occurrences in areas for which they are responsible.

8.7.6 Criticality Instrumentation

DOE Order 420.1 requires that in those cases where the mass of fissionable material exceeds the limits established in paragraph 4.2.1 of ANSI/ANS-8.3, but the probability of occurrence is determined to be less than 10^{-6} per year, neither a criticality alarm system (CAS) nor a criticality detection system is required. DOE Order 420.1 further states that neither a CAS nor a criticality detection system is required to be installed underwater when fissionable material is handled or stored beneath water shielding that is adequate to protect personnel. Also, neither a CAS nor a criticality detection system is required for fissionable material during shipment of fissionable material packaged in approved shipping containers.

Analyses referenced in Section 8.7.3 have determined that a criticality in the FRS is not possible under all normal operations and credible (i.e., $>10^{-6}$ events per year) accident conditions. In addition, shipping of SNF in the FRS will be via an NRC-approved shipping cask. For these reasons, no criticality alarm systems or criticality detection systems are installed in the FRS facility.

8.8 Fire Protection

DOE Order 420.1 states that Fire Hazard Analysis (FHA) documents shall be developed for "all nuclear facilities, significant new facilities, and facilities that represent unique or significant fire safety risks." The subject Order also states that FHAs shall be developed using a graded approach. WVNS-FHA-011, "Fire Hazard

Analysis Main Process Plant," contains the required FHA for the FRS facility. WVNS-FHA-011 documents that there are no "requirements" (i.e., actions required to correct fire protection deficiencies as regards compliance with mandatory fire protection requirements), and concludes that the facilities that are evaluated in WVNS-FHA-011 meet the Life Safety requirements for special-purpose industrial occupancies and DOE property loss requirements. It is noted that WVNS-FHA-011 states that "The FRS is classified as a non-occupancy area," and that the estimated heat of combustion is "negligible as there is a limited amount of combustibles located within the FRS."

The WVDP Fire and Explosion Protection Program is discussed in Section A.4.3.6 of WVNS-SAR-001.

REFERENCES FOR CHAPTER 8

American National Standards Institute. ANSI/ANS 8.1-1983: *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors.* American National Standards Institute. New York, New York.

__________. ANSI/ANS 8.3-1986: *Criticality Accident Alarm System.* American National Standards Institute. New York, New York.

__________. ANSI/ANS 8.5-1986: *Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material.* American National Standards Institute. New York, New York.

__________. ANSI/ANS 8.6-1983: *Safety in Conducting Subcritical Neutron-Multiplication Measurements in Situ.* American National Standards Institute. New York, New York.

__________. ANSI/ANS 8.7-1975 (R1982): *Guide for Nuclear Criticality Safety in the Storage of Fissile Materials.* American National Standards Institute. New York, New York.

__________. ANSI/ANS 8.9-1987: *Nuclear Criticality Safety Criteria for Steel-Pipe Intersections Containing Aqueous Solutions of Fissile Materials.* American National Standards Institute. New York, New York.

__________. ANSI/ANS 8.10-1983: *Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement.* American National Standards Institute. New York, New York.

__________. ANSI/ANS 8.12-1987: *Nuclear Criticality Control and Safety of Plutonium-Uranium Fuel Mixtures Outside Reactors.* American National Standards Institute. New York, New York.

__________. ANSI/ANS 8.15-1981: *Nuclear Criticality Control of Special Actinide Elements.* American National Standards Institute. New York, New York.

__________. ANSI/ANS 8.17-1984: *Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors.* American National Standards Institute. New York, New York.

__________. ANSI/ANS 8.19-1984: *Administrative Practices for Nuclear Criticality Safety.* American National Standards Institute. New York, New York.

REFERENCES FOR CHAPTER 8 (Continued)

__________. ANSI/ANS 8.21-1995: *Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors.* American National Standards Institute. New York, New York.

__________. ANSI 13.1-1969: *Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities.* American National Standards Institute, New York, New York.

__________. ANSI 13.6-1966 (R1972): *Practice for Occupational Radiation Exposure Records Systems.* American National Standards Institute, New York, New York.

__________. ANSI N323-1978: *Radiation Protection Instrumentation Test and Calibration.* The Institute of Electrical and Electronic Engineers, Inc., New York, New York.

Nuclear Fuel Services, Inc. 1964. *Final Safety Analysis Report: Spent Fuel Reprocessing Plant.* Nuclear Regulatory Commission Docket 50-201.

__________. 1973. *Final Safety Analysis Report: Nuclear Fuel Services, Inc. Reprocessing Plant, West Valley, New York.* Nuclear Regulatory Commission Docket 50-201.

Lazzaro, J.A. June 29, 1999. *Closure of an Action Item Resulting from a Self-Assessment of WVDP-162, "Nuclear Criticality Safety Program Manual."* Memo to L.J. Chilson (FD:99:0049).

ORIGEN. 1973. *ORIGEN - The ORNL Isotope Generation and Depletion Code.* ORNL-4628. M. J. Bell. Oak Ridge National Laboratory, 1973.

U.S. Department of Energy. July 21, 1997. DOE Order 232.1A: *Occurrence Reporting and Processing of Operations Information.* Washington, D.C.: U.S. Department of Energy.

__________. February 8, 1990. Change 2 (January 7, 1993). DOE Order 5400.5: *Radiation Protection of the Public and the Environment.* Washington, D.C.: U.S. Department of Energy.

__________. May 15, 1984. Change 4 (January 7, 1993). DOE Order 5480.4: *Environmental Protection, Safety, & Health Protection Standards.* Washington, D.C.: U.S. Department of Energy.

REFERENCES FOR CHAPTER 8 (Continued)

__________. November 15, 1994. DOE Order 5480.20A: *Personnel Selection, Qualification, and Training Requirements for DOE Nuclear Facilities.* Washington, D.C.: U.S. Department of Energy.

__________. October 13, 1995. Change 2 (October 24, 1996). DOE Order 420.1: *Facility Safety.* Washington, D.C.: U.S. Department of Energy.

__________. March 27, 1998. DOE O 440.1A: *Worker Protection Management for DOE Federal and Contractor Employees.* Washington, D.C.: U.S. Department of Energy.

__________. DOE O 231.1: *Environmental, Safety and Health Reporting.* Washington, D.C.: U.S. Department of Energy.

__________. June, 1990. DOE/EH-0135: *Performance Objectives and Criteria for Technical Safety Appraisals at Department of Energy Facilities and Sites.* Washington, D.C.: U.S. Department of Energy Office of Environmental, Safety, and Health (DOE-EH).

__________. *Occupational Radiation Protection,* 10 CFR 835.

U.S. Department of Labor. *Occupational Safety and Health Standards,* 29 CFR 1910.

__________. *Safety and Health Regulations for Construction,* 29 CFR 1926.

U.S. Nuclear Regulatory Commission. December 1996. NUREG/CR-6483: *Guide to Verification and Validation of the SCALE-4 Criticality Safety Software.*

Weiss, T.G. January 28, 1999. *Plutonium Distribution in Three Big Rock Point EP Type Assemblies in the FRS Fuel Pool.* Letter to File. (EN:99:0002)

West Valley Nuclear Services Co., Inc. WV-121: *Self-Assessment Program.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WV-905: *Radiological Protection.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WV-906: *Radiation and Safety Committee.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

REFERENCES FOR CHAPTER 8 (Continued)

__________. WV-923: *Nuclear Criticality Safety.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WV-984: *ALARA Program.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WV-987: *Occurrence Investigation and Reporting.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WVDP-010: *WVDP Radiological Controls Manual* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WVDP-011: *Industrial Hygiene and Safety Manual* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WVDP-076: *Environmental Protection Implementation Plan.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WVDP-162: *Nuclear Criticality Safety Program Plan.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WVDP-163: *ALARA Program Plan.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WVDP-228. *TN-BRP Spent Fuel Package Safety Analysis Report for Transport.* (Latest Revision.) Transnuclear, Inc.

__________. WVDP-229. *TN-REG Spent Fuel Package Safety Analysis Report for Transport.* (Latest Revision.) Transnuclear, Inc.

__________. WVDP-242: *Event Investigation and Reporting Manual.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WVDP-310: *WVDP Safety Management System (SMS) Description May 1998.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WVNS-FHA-011: *Fire Hazard Analysis Main Process Plant.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. Safety Analysis Report WVNS-SAR-001: *Project Overview and General Information.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

REFERENCES FOR CHAPTER 8 (Concluded)

__________. Safety Analysis Report WVNS-SAR-002: *Safety Analysis Report for Low-Level Waste Processing and Support Activities.* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. September 23, 1983. *Safety Assessment Document for the West Valley Demonstration Project Fuel Receiving and Storage Area, West Valley, New York.* Revision 1.

__________. 1987. *Spent Nuclear Fuel Removal Program at the West Valley Demonstration Project.* Topical Report DOE/NE/44139-37. March, 1987.

Yuan, Yuchien. April 22, 1996. *Nuclear Criticality Safety Evaluation: Spent Fuel Storage and Handling at the WVDP.* Square Y Consultants, Orchard Park, New York.

TABLE 8.2-1

CHARACTERISTICS OF THE 85 BWR SPENT NUCLEAR FUEL ASSEMBLIES IN THE FUEL STORAGE POOL(1)

Assembly No.	Elemental Uranium (g)	Elemental Plutonium (g)	Burnup (MWD/MTU)	Assembly No.	Elemental Uranium (g)	Elemental Plutonium (g)	Burnup (MWD/MTU)
B-04	120,501	837.11	20,247	CE-73	135,407	637.13	11,258
B-16	127,840	893.73	20,189	CE-74	131,360	777.52	15,825
CC-10	113,497	830.27	24,997	CE-75	131,540	716.77	13,961
CC-14	116,879	810.42	22,970	CE-76	136,505	365.99	5,502
CC-25	109,494	790.81	24,501	CE-77	131,678	631.18	10,620
CC-39	116,710	829.82	23,908	CE-79	130,675	830.18	17,667
CE-01	133,613	681.96	13,523	CE-80	130,952	810.36	16,906
CE-03	133,752	682.82	13,527	CE-81	134,781	783.86	15,502
CE-10	133,913	621.14	11,050	CE-82	131,143	839.93	16,630
CE-11	133,620	629.22	11,986	CE-83	135,079	736.06	13,961
CE-16	134,034	622.31	11,729	CE-84	134,701	779.89	15,391
CE-17	133,903	659.93	12,841	CE-85	135,140	725.16	13,652
CE-22	134,087	654.72	11,912	CE-86	136,009	566.08	9,617
CE-23	134,120	629.00	11,912	CE-87	135,525	624.71	10,969
CE-24	133,745	691.82	13,792	CEP-1	14,236	4587.05	16,988
CE-29	134,240	559.35	10,049	CEP-2	14,194	4114.11	17,315
CE-31	134,016	616.17	11,552	CEP-3	14,167	4494.78	19,275
CE-32	134,138	583.27	10,648	CF-01	135,134	456.51	7,333
CE-33	133,807	610.21	11,407	CF-02	134,389	633.12	11,481
CE-35	134,059	613.64	11,473	CF-03	130,289	791.90	15,267
CE-36	134,060	597.15	10,997	CF-06	134,818	552.02	9,407
CE-37	134,327	656.96	12,694	CF-12	135,276	437.18	6,930
CE-41	132,923	601.20	11,260	CF-13	133,762	739.37	14,209
CE-42	132,894	616.04	11,700	CF-14	131,948	811.04	15,554
CE-50	135,429	588.23	10,166	CF-18	134,505	560.06	9,640
CE-51	134,349	728.51	13,862	CF-19	135,104	450.61	7,215
CE-52	134,232	722.27	13,707	CF-23	133,917	739.56	14,193
CE-53	131,279	593.95	10,700	CF-24	135,384	437.53	6,930
CE-54	134,288	724.88	13,771	CF-25	133,753	744.43	14,339
CE-56	137,673	786.75	15,364	CF-26	134,501	583.45	10,227
CE-57	134,534	720.63	13,616	CF-35	134,252	655.33	12,051
CE-58	134,368	746.45	14,359	CF-42	134,295	680.44	12,672
CE-59	135,130	681.10	14,485	D-50	56,720	316.36	7,027
CE-60	134,011	777.70	15,359	D-51	59,946	116.56	2,065
CE-61	133,569	855.66	18,062	D-52	70,359	114.44	1,597
CE-62	133,334	862.49	18,362	D-53	78,736	121.89	1,596
CE-63	133,905	761.07	14,877	D-54	70,084	120.27	1,690
CE-64	134,133	739.56	14,201	D-55	72,679	122.66	1,689
CE-66	134,791	693.30	12,822	D-60	127,048	505.66	9,464
CE-67	134,608	705.12	13,176	D-61	127,872	662.35	13,591
CE-69	134,076	567.85	9,389	D-62	129,469	673.53	13,685
CE-70	132,451	484.83	8,095	D-63	122,816	617.91	13,044
CE-71	135,028	695.80	12,857				

Note: (1) Values at discharge

TABLE 8.2-2

CHARACTERISTICS OF THE OF 40 PWR SPENT NUCLEAR FUEL ASSEMBLIES IN THE FUEL STORAGE POOL[1]

Assembly No.	Elemental Uranium (g)	Elemental Plutonium (g)	Burnup (MWD/MTU)
C01	377,246	1,483	8,712
C02	377,353	1,455	8,516
C03	376,449	1,693	10,195
C04	377,251	1,482	8,703
C05	377,320	1,464	8,577
C06	377,414	1,438	8,403
C07	377,328	1,461	8,561
C08	377,167	1,504	8,858
C09	378,965	1,005	5,592
C10	376,347	1,719	10,385
C11	377,190	1,498	8,815
C12	374,589	2,146	13,748
C13	374,506	2,164	13,909
C14	376,522	1,674	10,059
C15	376,456	1,691	10,182
C16	374,461	2,175	13,998
C17	374,578	2,148	13,770
C18	376,674	1,635	9,774
C19	374,581	2,147	13,763
C20	378,794	1,054	5,899
C21	376,607	1,652	9,899
C22	375,269	1,986	12,432
C23	374,310	2,209	14,293
C24	378,699	1,082	6,069
C25	376,673	1,635	9,775
C26	376,673	1,635	9,776
C27	376,623	1,648	9,869
C28	376,409	1,703	10,269
C29	376,381	1,710	10,321
C30	375,922	1,827	11,188
C31	374,541	2,157	13,842
C32	376,367	1,714	10,348
C33	376,313	1,728	10,450
C34	376,785	1,606	9,568
C35	376,683	1,633	9,758
C36	376,653	1,640	9,812
C37	376,756	1,613	9,622
C38	376,248	1,745	10,572
C39	378,807	1,050	5,875
C40	376,272	1,738	10,526

Note: (1) Values at discharge

TABLE 8.2-3

RADIOACTIVE CHARACTERISTICS OF BWR FUEL[1]
(PER METRIC TON U CHARGED TO REACTOR)

Nuclide	Initial Fuel Activity[2] (Ci)	Decayed Fuel Activity[3] (Ci)
Fe-55	1.72E+03	4.89E+00
Co-60	5.11E+03	2.83E+02
Ni-63	4.39E+02	3.72E+02
Kr-85	5.75E+03	1.39E+03
Sr-90	4.57E+04	2.70E+04
Y-90	4.81E+04	2.71E+04
Ru-106	1.37E+05	3.87E-02
Sb-125	5.44E+03	2.24E+01
Cs-134	5.94E+04	3.65E+01
Cs-137	5.12E+04	3.08E+04
Ba-137m	4.85E+04	2.91E+04
Pm-147	8.92E+04	2.83E+02
Sm-151	1.76E+02	1.52E+02
Eu-154	3.18E+03	5.39E+02
Eu-155	1.86E+03	8.60E+01
Pu-238	5.62E+02	4.94E+02
Pu-239	7.50E+01	7.66E+01
Pu-240	1.04E+02	1.04E+02
Pu-241	1.98E+04	6.86E+03
Am-241	1.04E+01	4.31E+02
Cm-244	5.67E+01	2.45E+01

Notes:

[1] BWR Fuel: 3.0 w/o U-235; burnup 16,111 MWD/MTU; specific power - 25.9 MW/MTU

[2] Isotopic content of fuel at reactor discharge

[3] Isotopic content of fuel decayed 22 years to 1996

TABLE 8.2-4

RADIOACTIVE CHARACTERISTICS OF PWR FUEL[1]
(PER METRIC TON U CHARGED TO REACTOR)

Nuclide	Initial Fuel Activity[2] (Ci)	Decayed Fuel Activity[3] (Ci)
H-3	8.00E+02	2.08E+02
Fe-55	5.28E+03	8.80E+00
Co-60	7.49E+03	3.19E+02
Ni-63	6.60E+02	5.51E+02
Kr-85	9.49E+03	2.01E+03
Sr-90	7.47E+04	4.22E+04
Y-90	8.06E+04	4.22E+04
Tc-99	1.31E+01	1.31E+01
Ru-106	5.00E+05	3.40E-02
Sb-125	1.49E+04	3.71E+01
I-129	3.07E-02	3.10E-02
Cs-134	1.45E+05	4.54E+01
Cs-137	1.03E+05	5.94E+04
Ba-137m	9.81E+04	5.62E+04
Pm-147	1.30E+05	2.43E+02
Sm-151	3.52E+02	2.99E+02
Eu-154	9.95E+03	1.44E+03
Eu-155	6.12E+03	2.14E+02
Pu-238	2.12E+03	1.96E+03
Pu-239	3.05E+02	3.11E+02
Pu-240	5.10E+02	5.11E+02
Pu-241	1.19E+05	3.74E+04
Am-241	1.07E+02	2.75E+03
Am-243	1.42E+01	1.42E+01
Cm-244	1.50E+03	5.99E+02

Notes:

[1] PWR Fuel: 3.3 w/o U-235; burnup 33,000 MWD/MTU; specific power - 30 MW/MTU

[2] Isotopic content of fuel at reactor discharge

[3] Isotopic content of fuel decayed 24 years to 1996

TABLE 8.7-1

NFS CRITICALITY CALCULATION RESULTS
(WVNS, 1983 Attachment 3)

Reference BWR Fuel:

Dresden-I 6 x 6
Dimension: 4.20 x 4.20 x 144 in.
Enrichment: As stated below
H/U Ratio: As stated below

Reference PWR Fuel:

Westinghouse 15 x 15
Dimension: 8.575 x 8.575 x 144 in.
Enrichment: As stated below
H/U Ratio: As stated below

Model	Keff + 2σ
Two BWR Assemblies per Canister	
100 x 100 canister array 4 wt% U-235 H/U = 5.29	0.819 + 0.028
Single Canister 4 wt% U-235 H/U = 5.29	0.803 + 0.015
Single PWR Assembly per Canister	
100 x 100 canister array w/assy faces parallel 4 wt% U-235 H/U = 6.26	0.907 + 0.018
100 x 100 canister array w/assy faces parallel 4 wt% U-235 H/U = 5.29	0.913 + 0.020
100 x 100 canister array w/assy faces parallel 4 wt% U-235 H/U = 4.44	0.878 + 0.016
100 x 100 canister array w/assy faces parallel 4 wt% U-235 H/U = 4.12	0.861 + 0.018
Single Canister 4 wt% U-235 H/U = 5.29	0.884 + 0.019
63 x 63 canister array w/corner interaction 3.5 wt% U-235 H/U = 6.41	0.910 + 0.013
100 x 100 canister array w/assy faces parallel 3.5 wt% U-235 H/U = 6.41	0.903 + 0.017
Single Assembly 3.5 wt% U-235 H/U = 6.41	0.884 + 0.018

TABLE 8.7-2

WVNS CRITICALITY CALCULATION RESULTS
(WVNS, 1983 Attachment 8)

Reference BWR Fuel:		Reference PWR Fuel:	
Dresden-I	6 x 6	Westinghouse	15 x 15
Dimension:	4.27 x 4.27 x 109 in.	Dimension:	8.576 x 8.576 x 144 in.
Enrichment:	2.34 wt% U-235	Enrichment:	4.0 wt% U-235
H/U Ratio:	Not stated	H/U Ratio:	5.29

Model	Keff + 2σ
Normal Storage & Handling Operation	
One PWR Canister (15 x 15) PWR Assembly 4 wt% U-235 Enrichment in Canister	0.849752 + 0.009752
Four (4) PWR Canisters in Lift Rack 15 x 15 PWR Assembly 4.0 wt% U-235 Enrichment	0.84265 + 0.01034
PWR Canister and three Dresden-I Canisters 2 x 2 Array (20 Inch Pitch)	0.8502 + 0.0092
Accident Conditions	
One PWR Canister on Top of one PWR Canister Axially (15 x 15 Assembly, 4 wt% U-235 Enrichment)	0.850028 + 0.010
Three (3) PWR Assemblies on the Pool Floor in the CUP (Concrete and Water Reflection)	0.85017 + 0.00968
One PWR Assembly Next to a PWR Canister (Assembly in Contact with Loaded Canister)	0.86669 + 0.007532

TABLE 8.7-3

SQUARE-Y CRITICALITY CALCULATION RESULTS
(Yuan, Yuchien, 1996)

Westinghouse 14 x 14
Rod OD: 1.072 cm
Enrichment: 3.48 wt% U-235

Big Rock Point 9 x 9
Rod OD: 1.427 cm
Clad Thickness: 0.1016 cm
Pellet OD: 1.196 cm
Fuel Rod Pitch: 1.796 cm
Enrichment: 2.5 - 4.5 wt% U-235

Big Rock Point 9 x 9 (Type EP)
Rod OD: 1.427 cm
Clad Thickness: 0.1016 cm
Pellet OD: 1.196 cm
Fuel Rod Pitch: 1.796 cm
Enrichment: 4.33 wt% Pu-239

Case No.	Array	Fuel Type/Burnup (MWD/MTU)	Fuel Assembly Edge-to-Edge Spacing (cm)	Keff + 2σ
Normal Storage Cases				
1	Infinite (all-c)	PWR / 0	31.7	0.8236 + 0.00437
2	Infinite (all-c)	PWR / 10,000	31.7	0.7779 + 0.00411
3	Infinite (all-c)	BWR / 0	35.3	0.6951 + 0.00444
4	Infinite (all-c)	BWR-EP / 0	35.3	0.7757 + 0.0026[1]
Normal Transfer Cases				
5	1 x 1 (u)	PWR / 0	---	0.8138 + 0.00447
6	1 x 1 (u)	PWR / 10,000	---	0.7729 + 0.00437
7	4 x 1 (c-c-c-c)	PWR / 0	31.7	0.8171 + 0.00476
"Safe" Spacing Cases				
8	Infinite (all-c)	PWR / 0	10.1	0.95
9	Infinite (all-c)	PWR / 10,000	8.53	0.95
10	4 x 1 (c-c-c-c)	PWR / 0	5.39	0.95
Accident-Induced or Inadvertently-Created Configurations				
11	4 x 1 (c-u-c-c)	PWR / 0	---	0.8316 + 0.00418
12	2 x 1 (u-u)	PWR / 0	1.75[os]	0.9521 + 0.00423
13	2 x 1 (u-u)	PWR / 10,000	1.84[os]	0.9028 + 0.00478
14	3 x 1 (u-u-u)	PWR / 0	1.56[os]	1.001 + 0.00478

1. Reanalyzed by LMITCO (Weiss, 1999)
u: uncanistered
c: canistered: only spacing provided by canister is considered, aluminum shell of the canister is not.
os: optimum spacing

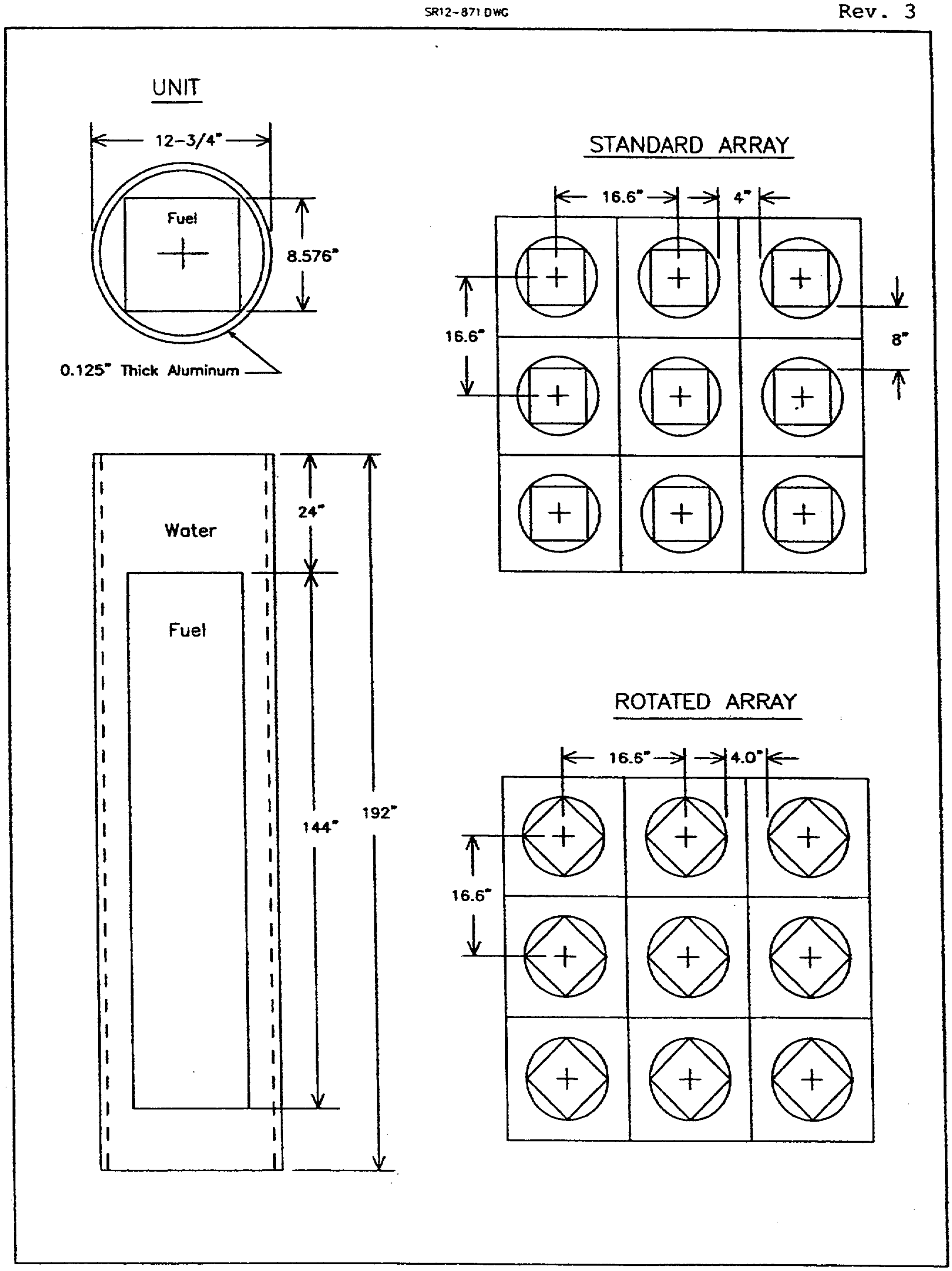

Figure 8.7-1 NFS Criticality Assessment – PWR Fuel

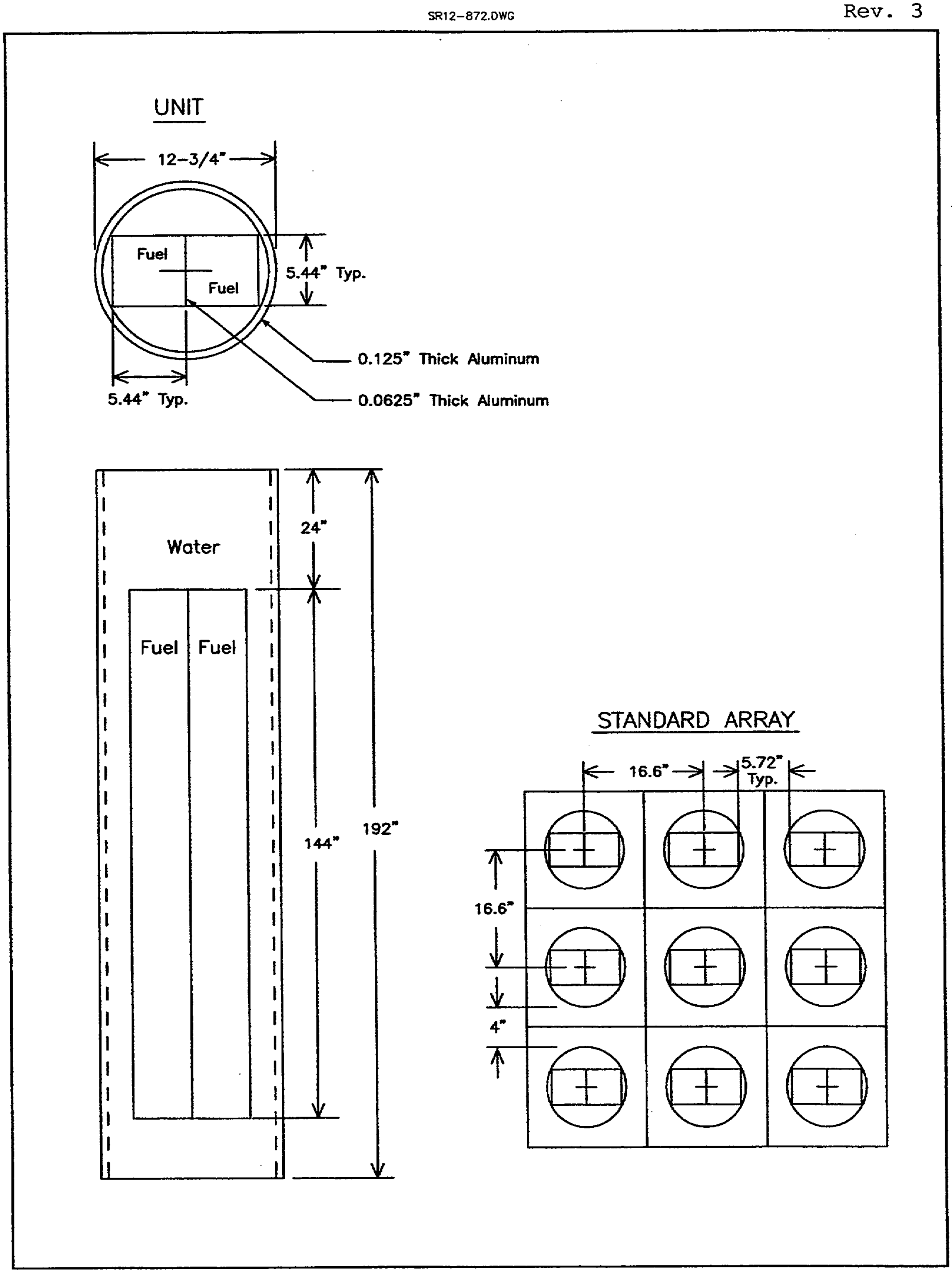

Figure 8.7–2 NFS Criticality Assessment – BWR Fuel

SR12-873.DWG

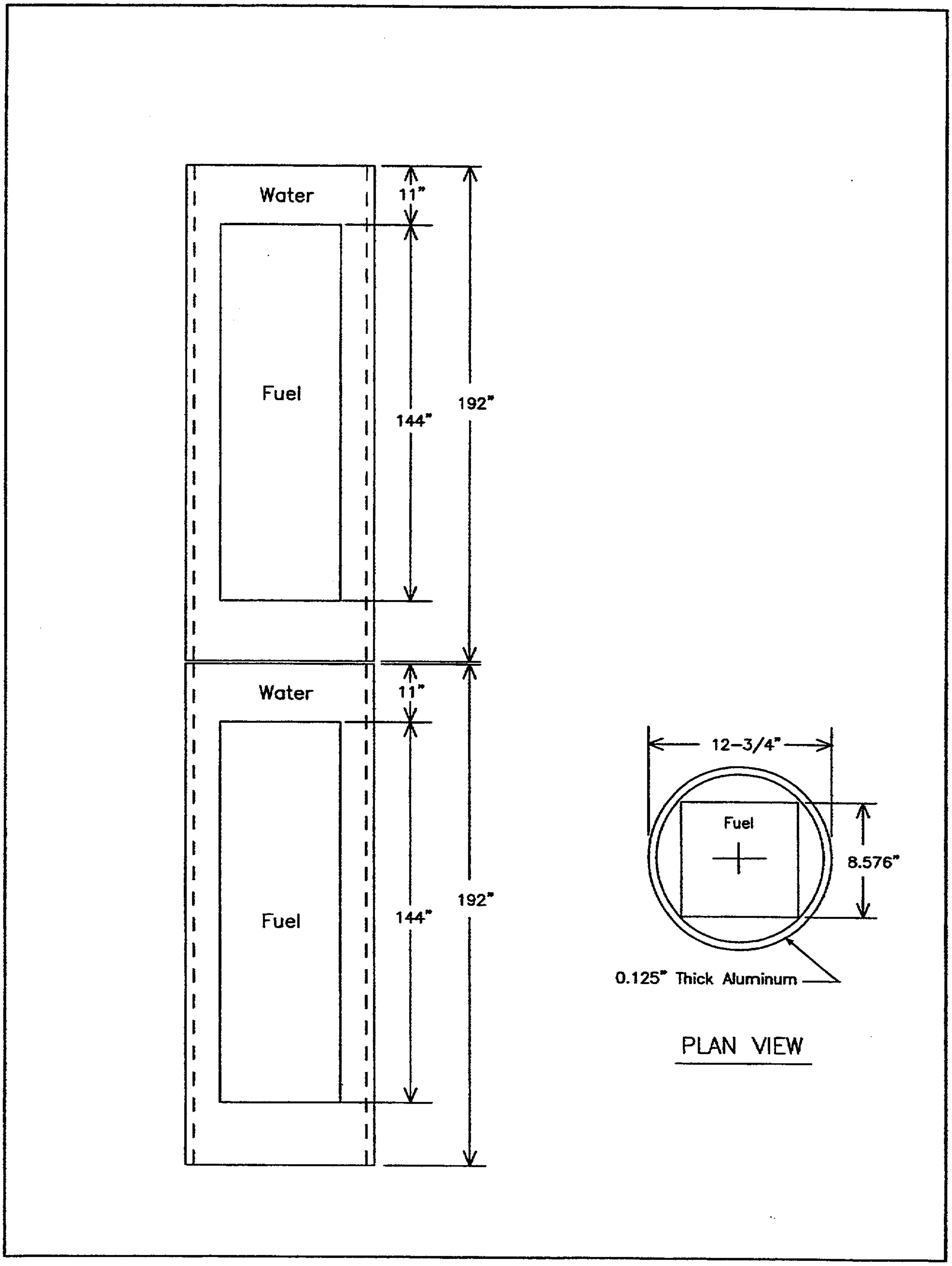

Figure 8.7-3 PWR Canister on PWR Canister

SR12-874.DWG

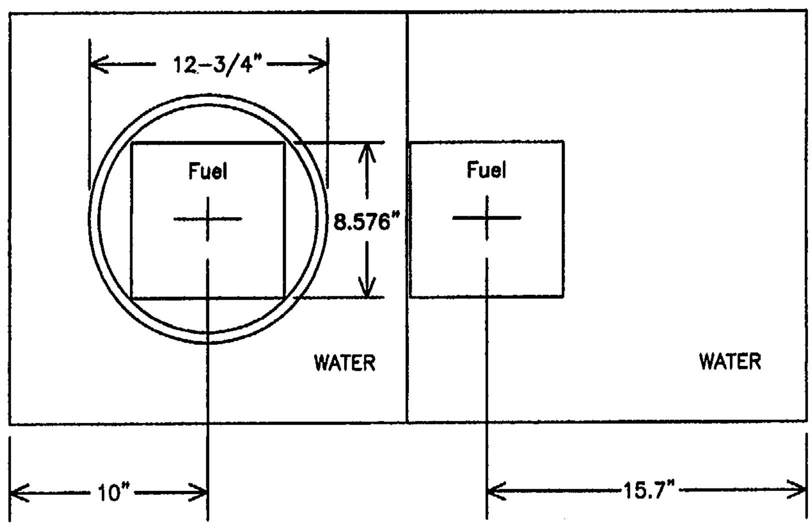

Figure 8.7-4 Canistered PWR Assembly next to Uncanistered PWR Assembly

SR12-875.DWG

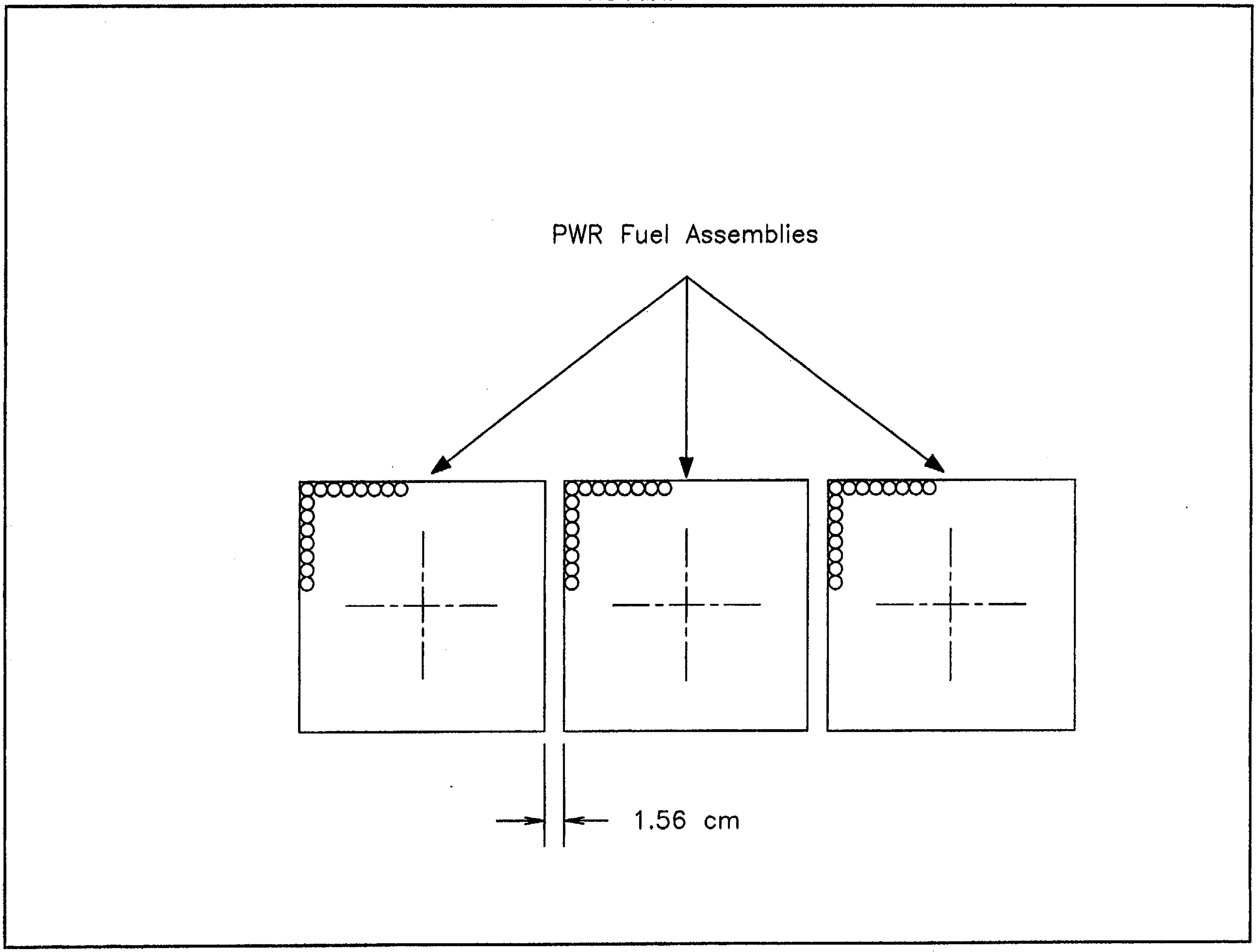

Figure 8.7–5 Minimum Conditions for Critical Configuration

SR12-876.DWG

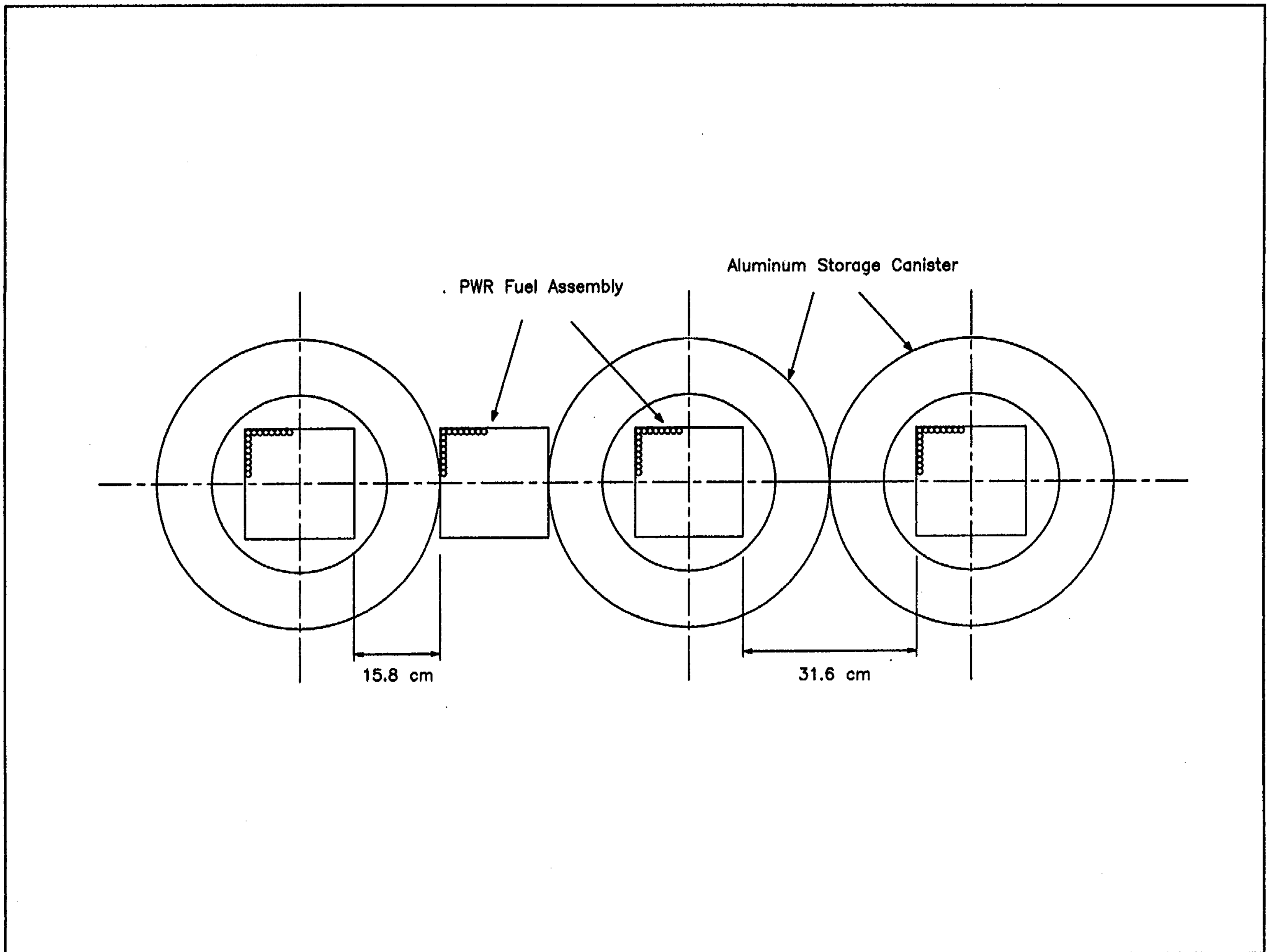

Figure 8.7-6 Worst Case Credible Accident Configuration

9.0 HAZARD AND ACCIDENT ANALYSES

9.1 Hazard Analysis

The systematic analysis of hazards associated with the Fuel Receiving and Storage (FRS) facility has been accomplished in this Safety Analysis Report (SAR) through the completion of a Process Hazards Analysis (PHA), which is presented in Table 9.1-1. Historical precedents at other chemical and nuclear facilities have demonstrated that a PHA can serve as a comprehensive and cost effective risk assessment technique, and can provide the basis for the entire safety analysis of a given facility or activity. The PHA provided in this SAR is intended to provide a qualitative analysis of the potential sources of hazards and mitigative features associated with operations at the FRS facility. Information gained through this analysis is then used in selecting accidents to be further analyzed more rigorously in Section 9.2, and in grading facility and process descriptions provided throughout this SAR.

9.1.1 Methodology

A PHA has been developed for the FRS facility that is consistent with hazard analysis guidelines provided in DOE-STD-3009-94. A hazard should be understood as "a source of danger, with the potential to cause illness, injury, or death to personnel." A PHA consists of two primary steps, namely, hazard identification and hazard evaluation.

9.1.1.1 Hazard Identification

The process of accomplishing the PHA is to identify the hazards in terms of quantity, form and location, potential initiating events, and other events which could result in an undesirable consequence. To ensure that a comprehensive, systematic analysis was performed, information was obtained from several sources. Primary among these sources were current FRS facility safety documents that identify and evaluate the risks of significant hazards. Additional information for the PHA has been obtained from diagrams of systems that support FRS facility operations, FRS-specific procedures, and miscellaneous documents associated with the fuel loading of NRC-approved spent nuclear fuel (SNF) shipping casks. Several current safety documents associated with fuel movement operations at the Idaho Nuclear Technology and Engineering Center (INTEC) (which is located at the Idaho National Environmental and Engineering Laboratory) were also examined.

9.1.1.2 Hazard Evaluation

Evaluation of hazards for the Process Hazards Analysis required the qualitative assessment of event consequences and frequencies. Qualitative consequence and frequency classifications used in Table 9.1-1 are as follows:

Qualitative Consequence Classification:

Negligible	Negligible on-site and off-site impact on people or the environs.
Low	Minor on-site and negligible off-site impact on the people or environs.
Moderate	Major on-site impact on people or the environs; only minor off-site impacts.
High	Major on-site and off-site impacts on the people and the environs.

Qualitative Frequency Classification:

Anticipated	($10^{-1} \geq p > 10^{-2}$) Incidents that may occur once or more during the lifetime of the facility.
Unlikely	($10^{-2} \geq p > 10^{-4}$) Accidents that may occur at some time during the lifetime of the facility.
Extremely Unlikely	($10^{-4} \geq p > 10^{-6}$) Accidents that will probably not occur during the life cycle of the facility.
Incredible	($10^{-6} \geq p$) Accidents that are not credible.

(p is the probability of a given event per year)

For each event in Table 9.1-1, a Risk Factor (RF) has been developed that is based on the consequence and frequency for the event. The value of the Risk Factor is determined from a three-by-three frequency- and consequence-ranking matrix, shown in Figure 9.1-1. Events having either an on-site or an off-site consequence but with frequencies of occurrence less than or equal to 1E-6 per year (i.e., incredible events) were assigned a risk factor of "I".

9.1.2 <u>Hazard Analysis Results</u>

9.1.2.1 Hazard Identification

The process of accomplishing the PHA is to identify the hazards in terms of quantity, form and location, potential initiating events, and other events that could result in an undesirable consequence. Due to the relatively low complexity of the FRS facility, hazard identification is straightforward. FRS facility systems and activities do not require the use of hazardous chemicals; consequently, radiological hazards dominate the FRS facility PHA. SNF assemblies and wastes from the fuel pool Submerged Water Filtration System represent the major hazards in the FRS. Industrial (nonradiological) hazards also exist in association with activities performed at the FRS facility.

Accidents with significant off-site consequences result when upsets occur in systems or operations that involve both significant quantities of radioactive or nonradioactive hazardous materials and large sources of energy. Operations in the FRS, including SNF storage and handling and pool water filtration, involve highly radioactive materials that are in a stable, (chemically) unreactive, solid matrix. Although there are few sources of energy in the facility to disperse radioactivity in the FRS, gravitational potential energy does represent a source of energy threatening these hazards. Consequently, accidents within the FRS facility are not capable of generating releases with significant on-site or off-site consequences.

9.1.2.2 Hazard Classification

The hazard classification for the FRS facility is presented in Section 1.5.

9.1.2.3 Hazard Evaluation

9.1.2.3.1 Summary of Significant Worker-Safety Features

While FRS worker hazards protection is provided by engineered facility features (e.g., limit switches, pool water filtration system, area radiation detectors, etc.), the most significant facility worker-safety features, namely fuel pool water and High Integrity Container (HIC) shielding enclosures, are passive in nature. Therefore, the primary operational worker-safety features identified in the hazards analysis are administrative controls. Specifically, worker protection from radiological hazards is controlled through the requirements of the WVDP Radiological Controls Manual (WVDP-010), while worker protection from nonradiological hazards is controlled through the requirements of the WVNS Industrial Hygiene and Safety Manual (WVDP-011). Additionally, safe conditions are maintained during fuel handling operations through the use of numerous administrative controls that restrict fuel movements and locations based on (1) the quantity, type, and location of SNF assemblies at any given time, and/or (2) other activities that may be occurring within the FRS Building at any given time (e.g., installation of the fuel pool gate, placement of a shipping cask into the cask unloading pool [CUP], etc.).

9.1.2.3.2 Accident Selection

The identification of accidents presenting the greatest risk to on-site individuals and the off-site public is one of the primary goals of the PHA. Accidents selected for more rigorous evaluation in Section 9.2 were those accidents with the largest Risk Factors (i.e., those accidents with a Risk Factor greater than or equal to 3). Consistent with this rationale, the following accidents were selected for further evaluation: drop of a loaded fuel canister or fuel assembly during handling for cask loading; failure of all 125 SNF assemblies in the storage pool due to a beyond evaluation basis seismic event; and catastrophic failure of a high integrity container due to a failure of the lifting equipment during handling for transfer from the shield container in the Radwaste Process Building to the shielded Surepak container in the north FRS yard.

As stated previously, SNF and fuel pool Submerged Water Filtration System wastes represent the significant sources of hazards in the facility. Accidents selected for evaluation in Section 9.2 represent bounding scenarios for upset conditions involving these hazards. Although dropping of a single SNF assembly most likely represents the bounding accident for this hazard, the evaluation of the consequences due to the failure of all 125 SNF assemblies was performed as a result of an inability to adequately establish the present yield capacity of the storage rack anchor bolts.

9.1.3 WVDP Evaluation Guidelines (EGs)

To facilitate the development of safety analysis evaluation guidelines for hazards associated with WVDP facilities, two distinctions have been made. These distinctions are as follows:

1) Whether the event (accident) is manmade or caused by natural phenomena; and
2) Whether the population at risk is the public or on-site workers.

These distinctions lead to four different combinations for which an evaluation guideline is required. This section establishes evaluation guidelines for these four situations.

For manmade accidents with either internal or external initiators, radiological evaluation basis accidents (EBAs) are compared to EGs over the frequency spectrum of 0.1 to 1E-06 events per year.

<u>Public Radiological EG</u>: Manmade EBAs shall not cause doses to the maximally exposed off-site individual (MOI) greater than: (1) 0.5 rem for accidents with estimated frequencies <0.1 per year but ≥1E-02 per year; (2) 5 rem for

accidents with estimated frequencies <1E-02 per year but ≥1E-04 per year; and (3) 25 rem for accidents with estimated frequencies <1E-04 per year but >1E-06 per year. Manmade EBAs with estimated frequencies ≤1E-06 per year are not considered credible. These EGs are depicted graphically in Figure 9.1-2.

On-Site Radiological EG: Manmade EBAs shall not result in calculated doses at the on-site evaluation point (OEP) (640 meters) greater than: (1) 5 rem (5E-2 Sv) for accidents with estimated frequencies <0.1 per year but ≥1E-02 per year; (2) 25 rem (2.5E-1 Sv) for accidents with estimated frequencies <1E-02 per year but ≥1E-04 per year; and (3) 100 rem (1 Sv) for accidents with estimated frequencies of <1E-04 per year but >1E-06 per year. Manmade EBAs with estimated frequencies ≤1E-06 per year are not considered credible. These EGs are depicted graphically in Figure 9.1-3.

Natural phenomena-induced EBAs with initiating frequencies defined by applicable design criteria and/or authorization basis documents are compared against the following EGs.

Public Radiological EG: Natural phenomena induced EBAs shall not cause doses to the MOI greater than 25 rem (2.5E-01 Sv).

On-Site Radiological EG: On-site numerical EGs shall not be required for safety assurance in the analysis of accidents induced by natural phenomena. Severe natural phenomena present hazards to on-site personnel that are dominated by non-radiological concerns. If the natural phenomena resistance capabilities for structures, systems, and components are exceeded, then the consequences of the natural phenomenon itself pose a greater risk to worker health and safety than any exposure to radioactive material released by the event.

9.2 Accident Analyses

9.2.1 Methodology

Accident analyses are performed through the use of established and accepted references and computer codes. Computer codes used in accident analyses are verified per approved procedures prior to use. Analyses to evaluate the consequences of airborne radiological releases utilize source terms developed from guidance given in DOE-HDBK-3010-94, "Airborne Release Fractions/Rates and Respirable Fractions for Non-Reactor Nuclear Facilities"; dispersion factors for stability class D with wind speed 4.5 m/s and stability class F with wind speed 1 m/s, and site specific dispersion factors calculated using the PAVAN computer codes; and radiological dose conversion factors given in DOE/EH-0070, "External Dose-Rate Conversion Factors for Calculation of Dose to the Public," and DOE/EH-0071, "Internal Dose Conversion Factors for Calculation of Dose to the Public."

Site-specific dispersion factors (χ/Q values) are calculated using the PAVAN computer code which implements the guidance provided in NRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequences Assessments at Nuclear Power Plants." The χ/Q calculations are based on the theory that material released to the atmosphere will be normally distributed (Gaussian) about the plume center-line. A straight-line trajectory is assumed between the point of release and all distances for which the χ/Q values are calculated.

The PAVAN program uses meteorological data in the form of joint frequency distributions of hourly averages of wind direction and wind speed by atmospheric stability class. Wind direction is distributed into 16 sectors (N, NNE, NE,...) and atmospheric stability is distributed into 7 classes (A-G). For each of 16 downwind sectors, the program calculates χ/Q values for each combination of wind speed and atmospheric stability at two distances. The χ/Q values calculated for each sector are then ordered from greatest to smallest and an associated cumulative frequency distribution is derived based on the frequency distribution of wind speed and stabilities for that sector. The smallest χ/Q value in the distribution will have a corresponding cumulative frequency equal to the wind-direction frequency for that sector. The program then determines for each sector an upper envelope curve based on these data such that no plotted point is above the curve. From this upper envelope the χ/Q value which is equaled or exceeded 0.5% of the total time is obtained. The maximum 0.5% χ/Q value from the 16 sectors becomes the maximum sector χ/Q value.

9.2.1.1 Initiating Event Summary

Initiating event summaries have not been provided for accident evaluations in this SAR as all assessments deterministically assume the occurrence of a particular accident event, with no regard for the mechanisms or chains of events necessary to arrive at the analyzed event.

9.2.1.2 Scenario Development

Accident scenarios have been provided in sufficient detail to support the evaluation of source terms utilized in the calculations. Scenario developments deterministically assume the occurrence of a particular accident event, with no regard for the probability of mechanisms or chains of events necessary to arrive at the analysis event.

9.2.1.3 Source Term Analysis

To bound the consequences of accidents analyzed, source terms used in this SAR are based on either the entire inventory of material at risk (MAR) (as in the case of the catastrophic failure of a HIC due to dropping), or upon conservative damage ratios (DRs) and/or airborne release fractions (ARFs) (as in the case of dropping a SNF assembly). (The DR is the fraction of the MAR actually impacted by the accident-generated conditions.) To account for the fact that some releases involve particulate matter that is not entirely respirable, a factor representing the respirable fraction (RF) of released material is applied. For all analyses, leakpath factors (LPFs) as described in DOE-HDBK-3010-94, are conservatively assumed to be equal to 1.0. Radiological source terms are calculated as MAR x DR x ARF x RF x LPF.

9.2.1.4 Consequence Analysis

Consequences of radiological accidents in this SAR are calculated for individuals at an On-site Evaluation Point (OEP) located 640 m from the center of the accident release and for off-site individuals located at the site boundary (1050 m) in the sector having the maximum concentration of radioactivity. The magnitude of radiological consequences is calculated by multiplying the accident source term by unit dose factors. Only nuclides contributing greater than 0.1% of the total dose are reported. Unit dose factors for airborne releases are calculated using site-specific dispersion values (χ/Q) and external and internal dose conversion factors given in DOE/EH-0070 and DOE/EH-0071, respectively. Unit dose factors assume a two hour exposure time for affected individuals (consistent with guidance provided in NRC Regulatory Guide 1.145) and an individual breathing rate of 0.333 L/s (ICRP 23, 1975). Consequences are calculated for several meteorological conditions: stability class "D", wind speed 4.5 m/s; stability class "F", wind speed 1 m/s; and site-specific 95% meteorology.

9.2.1.5 Comparison to Guidelines

Guidelines utilized for the comparison to accident analysis consequences are given in Section 9.1.3. Guidelines for radiological consequences due to operating and natural phenomena accidents are provided. Maximum acceptable consequences for radiological accidents are given in Figures 9.1-2 and 9.1-3. For the purposes of performing an Unreviewed Safety Question Determination, Figures 9.1-2 and 9.1-3 present the authorization basis risk for activities conducted in the FRS facility.

9.2.2 Operational Accidents

Operational accidents are those events having internal initiators, such as fires, explosions, spills, or inadvertent nuclear criticality. Consequences of these accidents are evaluated against guidelines given in Section 9.1.3 based on the probability of occurrence.

9.2.2.1 Drop of a Loaded Canister or Fuel Assembly

9.2.2.1.1 Scenario Development

The handling of SNF assemblies is necessary for cask loading and for periodic visual assessment to evaluate fuel integrity. Canisters containing assemblies are transferred from the fuel storage pool to the CUP. In the CUP, assemblies are raised from the canister using the bridge-mounted 900 kg (1 ton) fuel hoist with special extension grapples. The hoist is provided with limit mechanisms that prevent the fuel assembly from being raised to unsafe levels. Although the fuel assembly grapple is latched to the fuel, the fuel is not lanyarded to the grapple, and it is possible that the fuel assembly could be dropped during handling. It is assumed that some of the fuel cladding would be breached in such a scenario. Dropping of a loaded fuel canister is considered to present a similar radiological hazard.

9.2.2.1.2 Source Term Analysis

The source term for this accident assessment is based on the isotopic content of spent pressurized water reactor (PWR) fuel (ORIGEN, 1973) as it represents the fuel with the greatest fission product activity per assembly. Table 8.2-4 gives the isotopic content of PWR fuel assemblies per metric ton of uranium charged to the reactor. PWR fuel assemblies in storage in the FRS were designed with an initial uranium content of 0.382 MT. Consequently, the material-at-risk for this accident was calculated by multiplying the third column of Table 8.2-4 (24-year decayed PWR fuel) by 0.382.

The accident scenario assumes radioactive material release to the fuel pool occurs as a result of the crushing of a dropped fuel assembly. DOE-HDBK-3010-94 (DOE, 1994) provides bounding values for the fraction of material that would become airborne due to impact stress as a function of fall height, and provides a bounding value of the fraction of the airborne material that would be of respirable size (i.e. ARF and RF values). For this assessment it is assumed that the fuel assembly falls a distance of 4.88 m (16 ft), corresponding to a respirable release fraction (ARF x RF) of 1.05E-04. Multiplication of the isotopic content of a fuel assembly by this fraction gives the source term activity for the pool. To determine the environmental source term, the fuel pool source term value is multiplied by a fraction value (ARF x RF = 2E-03) appropriate for release of contaminants due to evaporation from a contaminated pool of water. It has been assumed in the analysis that the total inventory of H-3 and Kr-85 in a dropped assembly is released to the atmosphere.

9.2.2.1.3 Analysis of Results

Table 9.2-1 presents the dose at the on-site evaluation point and to the maximally exposed off-site individual from the dropping of a PWR fuel assembly in the CUP for various meteorological conditions. The maximum total effective dose equivalent (TEDE) at the on-site evaluation point has been calculated to be 15 mrem, as shown in Table 9.2-1. The maximum TEDE received by an off-site individual has been calculated to be 6.1 mrem.

9.2.2.1.4 Comparison to Guidelines

Section 9.1.3 defines the means by which safety assurance is shown by providing numerical criteria against which to judge the results of the accident analyses.

Radiological evaluation guidelines given in Section 9.1.3 state that total effective dose equivalent to the maximally exposed off-site individual due to an operational accident shall not be greater than 0.5 rem (5E-03 Sv) for accidents with estimated frequencies < 0.1 event per year but ≥ 0.01 event per year; 5 rem (5E-02 Sv) for accidents with estimated frequencies < 1E-2 event per year but ≥ 1E-4 event per year; and 25 rem (2.5E-01 Sv) for accidents with estimated frequencies < 1E-4 event per year but ≥ 1E-6 event per year.

For the on-site evaluation point, the dose limit is 5 rem (5E-02 Sv) TEDE for events that have a frequency range of 0.1 to 0.01 per year, 25 rem (2.5E-01 Sv) TEDE for events that have a frequency range of 1E-2 to 1E-4 per year, and 100 rem (1 Sv) TEDE for those events that have a frequency less than 1E-4 per year.

The dose to the maximally exposed off-site individual (6.1 mrem (6.1E-05 Sv) TEDE), and the dose to a receptor at the on-site evaluation point (15 mrem (1.5E-04 Sv) TEDE), due to the dropping of a fuel assembly are below the radiological dose acceptance criteria specified in Section 9.1.3.

9.2.2.2 Drop of a Loaded High Integrity Container

9.2.2.2.1 Scenario Development

A full HIC is lifted out of its process shield through an opening in the Radwaste Process Building roof to a height of approximately 4.6 m (15 ft). Prior to lifting the HIC free of its shield, it is lifted approximately 5 to 8 cm (2 to 3 in) and held for five minutes. If the load doesn't slump or drop during this time, the HIC is placed into a shielded storage container in the north FRS yard.

A high integrity container and its associated lifting ring are designed to withstand an abrupt lift force of 3g with a full payload of 4500 kg (10,000 lb) and have been shown to successfully withstand drops onto compacted sand from 7.6 m (25 ft) on both a top corner and bottom corner without splitting open while fully loaded. However, it is possible that failure of any portion of the lifting or rigging equipment could result in the drop of a HIC and it is assumed that the HIC would rupture upon impact, releasing its entire contents to the ground.

9.2.2.2.2 Source Term Analysis

The radioactive inventory of a full HIC is based on the radiological characteristics of sludge and resin contained in HIC "B". (HIC "B" exhibits the highest concentration of activity of the five HICs that have been filled since the use of HICs began in 1986.) The inventory is calculated using HIC "B" characterization data (WVNS, 1987) and an assumed resin mass of 3,200 kg (7,000 lb). DOE-HDBK-3010-94 gives a bounding fraction of respirable material that would be released from the free fall spill of a slurry of 4E-5 (ARF = 5E-5, RF = 0.8). Multiplication of the HIC activity by this fraction gives the source term activity.

9.2.2.2.3 Analysis of Results

Table 9.2-2 presents the dose at the on-site evaluation point and to the maximally exposed off-site individual from the drop of a loaded HIC for various meteorological conditions. The maximum TEDE at the on-site evaluation point has been calculated to be 6.3 mrem (6.3E-05 Sv), as shown in Table 9.2-2. The maximum TEDE received by an off-site individual has been calculated to be 3.0 mrem (3.0E-05 Sv).

9.2.2.2.4 Comparison to Guidelines

Section 9.1.3 defines the means by which safety assurance is shown by providing numerical criteria against which to judge the results of the accident analyses. The dose to the maximally exposed off-site individual (3.0 mrem (3.0E-05 Sv) TEDE), and the dose to a receptor at the on-site evaluation point (6.3 mrem (6.3E-05 Sv) TEDE), due to the dropping and catastrophic failure of a loaded HIC are below the radiological dose acceptance criteria specified in Section 9.1.3.

9.2.2.3 Inadvertent Criticality in the FRS

9.2.2.3.1 Scenario Development

Shipping of SNF from the FRS requires that fuel assemblies be transferred from their storage positions in the Fuel Storage Pool to the lift rack in the CUP and subsequently loaded one at a time into the cask. Nevertheless, it is assumed that an operator, while in the process of handling fuel for shipping, removes several fuel assemblies from their storage canisters and stages them at the lower level of the CUP to facilitate loading in the shipping cask. It is assumed that the arrangement of two full lift rack loads (i.e., eight SNF assemblies) is placed in the CUP in such a manner that an inadvertent criticality occurs.

9.2.2.3.2 Source Term Analysis

The particulate source term for this accident assessment is based on the isotopic content of undecayed irradiated PWR fuel given in Table 8.2-4. (Criticality assessments referenced in Section 8.7 have conservatively assumed unirradiated fuel for determining the k_{eff} of various fuel combinations. This accident assessment conservatively utilizes the radionuclide inventory of spent fuel for determining off-site dose consequences due to the higher concentration of toxic transuranic nuclides in spent fuel.) The particulate fission product source term is calculated using the release fractions of respirable materials from a moderated and reflected solid following an inadvertent criticality recommended in Table 6-10 of DOE-HDBK-3010-94 (DOE, 1994). Multiplication of the isotopic content of eight PWR fuel assemblies by the release fractions indicated in DOE-HDBK-3010-94 gives the atmospheric source term.

The fission gas source term for the assessment is based on the source term for spent fuel solutions given in Table 6-7 of DOE-HDBK-3010-94. It is not expected that the magnitude of the release for intact fuel assemblies would be as great as that for solutions due to the lower surface to volume ratios in fuel. Consequently, it is assumed that 10% of the fuel softens due to the heat generated, thus allowing noncondensible gases and radioiodine in that fraction to be released, consistent with the recommendations given in DOE-HDBK-3010-94. It is assumed that all of the fission gas released to the moderator is subsequently released to the atmosphere.

9.2.2.3.3 Analysis of Results

Table 9.2-3 presents the dose at the on-site evaluation point and to the maximally exposed off-site individual from an inadvertent criticality in the CUP for various meteorological conditions. The maximum total effective dose equivalent (TEDE) at the on-site evaluation point has been calculated to be 857 mrem (8.57E-03 Sv), as shown in Table 9.2-3. The maximum TEDE received by an off-site individual has been calculated to be 353 mrem.

9.2.2.3.4 Comparison to Guidelines

Section 9.1.3 defines the means by which safety assurance is shown by providing numerical criteria against which to judge the results of the accident analyses.

Radiological evaluation guidelines given in Section 9.1.3 states that total effective dose equivalent to the maximally exposed off-site individual due to an operational accident shall not be greater than 0.5 rem (5E-03 Sv) for accidents with estimated frequencies < 0.1 event per year but ≥ 0.01 event per year; 5 rem (5E-02 Sv) for accidents with estimated frequencies < 1E-2 event per year but ≥ 1E-4 event per year; and 25 rem (2.5E-01 Sv) for accidents with estimated frequencies < 1E-4 event per year by ≥ 1E-6 event per year.

For the on-site evaluation point, the dose limit is 5 rem (5E-02 Sv) TEDE for events that have a frequency range of 0.1 to 0.01 per year, 25 rem (2.5E-01 Sv) TEDE for events that have a frequency range of 1E-2 to 1E-4 per year, and 100 rem (1 Sv) TEDE for those events that have a frequency less than 1E-4 per year.

The dose to the maximally exposed off-site individual (353 mrem TEDE) and the dose to a receptor at the on-site evaluation point (857 mrem (8.57E-03 Sv) TEDE) due to an inadvertent criticality in the FRS are below the radiological dose acceptance criteria specified in Section 9.1.3 for all frequency categories. (Under the incredible circumstances that all 125 spent nuclear fuel assemblies are involved in an inadvertent criticality, the doses to the maximally exposed off-site individual and to the receptor at the on-site evaluation point are 6.2 rem (6.2E-2 Sv) and 15 rem (1.5E-01 Sv), respectively.)

9.2.3 Natural Phenomena Events

Natural phenomena accidents are those events having external, natural initiators, such as earthquakes, tornadoes, and floods. Consequences of these accidents are evaluated against guidelines given in Section 9.1.3, independent of the probability of occurrence.

9.2.3.1 Beyond Evaluation Basis Seismic Event

9.2.3.1.1 Scenario Development

Several analyses to determine the structural integrity of FRS facilities and equipment following seismic events of varying magnitudes have been performed over the history of fuel pool operation. The results of these analyses, which have been summarized in Chapters 4 and 5 of this SAR, conclude that structural integrity of the FRS facility would be maintained under the conditions evaluated. Deterministic evaluations to assess the consequences of a complete loss of shielding water in the storage pool, including heat generation and exposure rate calculations, have also been performed.

Post reactor cooling of the fuel has resulted in a significant decrease in the thermal power of the fuel. In 1992, a calculation of the heat generation rate determined that the total thermal output of the 125 stored assemblies was approximately 8,800 watts (Wolniewicz, 1992). Due to the assembly storage configuration, convective air currents would be sufficient to remove the heat produced. Therefore, even in the event of a complete loss of water in the pool, sufficient heat would not be generated to significantly affect the pool structure.

A separate analysis to estimate the dose rate on the service bridge from the total loss of pool water due to an earthquake (Shearer, 1991) was also performed. Results of this assessment show that the dose rate on the service bridge due to loss of shielding water would be approximately 64 rem/hr (6.4E-01 Sv/hr). Therefore, the primary purpose of pool water is to maintain occupational radiation exposure as low as reasonably achievable (ALARA).

Complete loss of shielding water in the fuel storage pool and cask unloading pool is considered to be extremely unlikely. The silty till layer (Lavery till) underlying the pool structure is a mixture of very fine grained heterogeneous clay and silt containing minor amounts of sand and stones. The silty till is typically dense, compact, and moist, and is of low permeability.

The analysis in Section 9.2.2 which evaluates the dropping of a single SNF assembly most likely represents the bounding accident for a beyond design basis seismic event. Nevertheless, an evaluation to determine the consequences of the failure of all 125 SNF assemblies was performed due to an inability to adequately establish the present yield capacity of the storage rack anchor bolts. The storage rack and its associated hardware have been in service since initial fuel reprocessing operations began in the mid 1960's and it is possible that some degree of corrosion of this equipment might have occurred. An engineering evaluation of the storage rack and its associated hardware has determined that it is suitable for normal operations; however, based on current information, it is not possible to determine its capacity under beyond evaluation basis seismic induced stresses. The deterministic assumption has therefore been made that under these stresses the anchor bolts fail, allowing all 125 SNF assemblies to fall to the floor of the fuel storage pool.

9.2.3.1.2 Source Term Analysis

The source term for this assessment is 125 times the source term given in Section 9.2.2.1.2 for a single failed fuel assembly. The source term given in Section 9.2.2.1.2 is based on the isotopic content of spent pressurized water reactor (PWR) fuel (ORIGEN, 1973) as it represents the fuel with the greatest fission product activity per assembly.

9.2.3.1.3 Analysis of Results

Table 9.2-1 presents the dose at the on-site evaluation point and to the maximally exposed off-site individual from failure of all 125 SNF assemblies in the fuel storage pool for various meteorological conditions. (This analysis conservatively models all 125 fuel assemblies in the FSP as PWR assemblies.) The maximum total effective dose equivalent (TEDE) at the on-site evaluation point has been calculated to be 1.86 rem (1.86E-02 Sv), as shown in Table 9.2-1. The maximum TEDE received by an off-site individual has been calculated to be 768 mrem (7.68E-03 Sv).

9.2.3.1.4 Comparison to Guidelines

Section 9.1.3 defines the means by which safety assurance is shown by providing numerical criteria against which to judge the results of the accident analyses. The radiological dose acceptance criteria (25 rem TEDE) for the maximally exposed off-site individual specified in Section 9.1.3, for a natural phenomena event, is independent of frequency. On-site numerical dose evaluation guidelines are not required for safety assurance in accident analyses for natural phenomena.

The dose to the maximally exposed off-site individual due to failure of all 125 SNF assemblies (768 mrem TEDE) is well below the radiological dose acceptance criteria specified in Section 9.1.3 for natural phenomena events. Though on-site numerical dose evaluation guidelines are not required for safety assurance in accident analyses for natural phenomena, it is nevertheless noted than the on-site evaluation point dose of 1.86 rem (1.86E-02 Sv) is much less that the 5 rem (5E-02 Sv) allotted for man-made accidents with estimated frequencies <0.1 per year but ≥1E-02 per year.

9.2.4 <u>Radiological Accident Analysis Summary</u>

A summary of the radiological consequence assessments performed in this SAR is provided in Table 9.2-4. All accidents analyzed are within the evaluation guidelines given in Section 9.1.3. The failure of all 125 SNF assemblies results in a total effective dose equivalent to the maximally exposed off-site individual of 768 mrem (7.68E-03 Sv). This represents the bounding accident for radiological releases.

REFERENCES FOR CHAPTER 9

ICRP 23, *Reference Man: Anatomical, Physiological and Metabolic Characteristics*, International Commission on Radiological Protection, 1975.

McVay, Charles. 1987. DD/O Department. *Classification of "B" HIC*. Memo to T. Hughes dated September 16, 1987. EH:87:0089.

Nuclear Fuel Services, Inc. 1973. *Nuclear Safety Evaluation of Arrays of Light Water Reactor Fuel in the Fuel Storage Pool*. J. R. Clark to L. C. Rouse, United States Atomic Energy Commission. August 1973.

ORIGEN. 1973. *ORIGEN - The ORNL Isotope Generation and Depletion Code*. ORNL-4628. M. J. Bell. Oak Ridge National Laboratory, 1973.

Pacific Northwest National Laboratory. November 1982. *PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations*. NUREG/CR-2858, PNL-4413.

Shearer, Todd W. 1991. *FRS Pool Loss of Water Dose Rate Analysis*. FB:91:0227.

U.S. Department of Energy. July, 1994. DOE Standard DOE-STD-3009-94: *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*. Washington, D.C.: U.S. Department of Energy.

__________. December, 1994. DOE Handbook DOE-HDBK-3010-94: *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*. Washington, D.C.: U.S. Department of Energy.

__________. July, 1988. DOE/EH-0070: *External Dose-Rate Conversion Factors for Calculation of Dose to the Public*. Washington, D.C.: U.S. Department of Energy.

__________. July, 1988. DOE/EH-0071: *Internal Dose Conversion Factors for Calculation of Dose to the Public*. Washington, D.C.: U.S. Department of Energy.

U.S. Nuclear Regulatory Commission. November, 1982. Regulatory Guide 1.145: *Atmospheric Dispersion Models for Potential Accident Consequences Assessments at Nuclear Power Plants*. Revision 1.

West Valley Nuclear Services Co., Inc. 1994. *Report on Spent Fuel Storage Pool Water Loss Tests 11/1/92 to 1/16/94*, R. F. Zalenski, March 1994.

REFERENCES FOR CHAPTER 9 (Concluded)

__________. September 23, 1983. *Safety Assessment Document for the West Valley Demonstration Project Fuel Receiving and Storage Area, West Valley, New York.* Revision 1.

__________. WVDP-010: *WVDP Radiological Controls Manual* (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WVDP-011: *Industrial Hygiene and Safety Manual* (Latest Revision.) West Valley Nuclear Services Co., Inc.

Wolniewicz, Joseph C. March, 1993. *Estimation of Activity in the Former Nuclear Fuel Services Reprocessing Plant.* CN:93:0015.

__________. 1992. *Calculation of Decay Heat Generated by the Irradiated Fuel Stored in the FRS Fuel Storage Pool.* Memo to S. Reeves 2/27/92. FB:920055.

TABLE 9.1-1

PROCESS HAZARDS ANALYSIS FOR THE FRS FACILITY

Hazard	Event	Preventive and Mitigative Features	Consequences[1]	Frequency	Risk Factor[2]
Spent Nuclear Fuel	1) Major leakage of pool water and/or release of SNF materials from cladding due to beyond EBA seismic event 2) Major leakage of pool water and/or release of SNF materials due to design basis tornado 3) Inadvertent criticality 4) Fire involving spent nuclear fuel 5) Substantial pool water level decrease (i.e., losses exceed makeup capabilities for extended period of time) not caused by natural phenomena (e.g., seal failure of fuel pool gate while CUP is empty, inadvertent syphoning of pool water while sluicing loaded resin, dropping of pool gate cracks pool wall) 6) Handling mishap of loaded cask (e.g., cask bridge failure, crane hook cabling break, cask trunnion failure, limit switch failure) 7) Handling mishap of loaded canister (e.g., canister crane hoist break, grapple malfunction, operator error, canister lift rack failure/malfunction, limit switch failure) 8) Handling mishap of fuel assembly (e.g., fuel hoist or jib break, grapple malfunction, operator error, limit switch failure)	- Analysis that the fuel pool walls possess adequate strength under combined loads (dead, live, soil, hydrostatic, and 0.1g WVDP design basis earthquake) - Administrative controls on fuel operations - Analyses that demonstrate that postulated irregular fuel configurations would not support criticality - Area radiation detector(s) and continuous air monitors to alert workers to abnormal radiological conditions - Minimal combustible materials in the FRS Building - FRS fire protection equipment, and WVDP fire department services - Low permeability of soil around/underneath of pool - Syphon break hole on the sluice out line located 18 inches below normal pool water level - Formal operating procedures for every major activity or piece of equipment - Established hoisting and rigging procedures, including load testing - Installation of fuel pool gate, should CUP be damaged/leaking - Limit switches and/or mechanical stops on bridges, trolleys, and hoists (as discussed in Section 5.2 of this SAR) - Design conservatisms for lifting equipment (typically minimum safety factor of 5, based on ultimate strength of material) - Per 10 CFR 71, trunnions should have a safety factor of 3 against yielding - Personnel must be cleared from "areas where the crane hook or carried equipment will be traveling" - Lid must be installed on cask after installation of fuel and secured by four lid bolts prior to cask movement	1) Moderate 2) High 3) Moderate 4) High 5) Low 6) Low 7) Moderate 8) Moderate	1) Unlikely 2) Incredible 3) Incredible 4) Incredible 5) Unlikely 6) Extremely unlikely 7) Unlikely 8) Unlikely	1) 5 2) I 3) I 4) I 5) 2 6) 1 7) 5 8) 5

Hazard	Event	Preventive and Mitigative Features	Consequences[1]	Frequency	Risk Factor[2]
High activity in pool water	1) Breach in demineralizer unit 2) Gross mishandling (or rinsing off) of used filter cartridges while in (or over) pool 3) Releases from fuel element(s) beyond fuel pool water filtration system capacity to cleanup	- Maintenance program - Inlet and outlet of the demineralizer unit are protected with Johnson screens (and secondary retention elements should Johnson screens fail) that prevent the escape of resin into the pool - Established/written criteria for replacement of resin, filter cartridges, and a HIC - Avoiding jarring movements with fuel assemblies and canisters containing fuel	1) Low 2) Low 3) Low	1) Unlikely 2) Unlikely 3) Unlikely	1) 2 2) 2 3) 2
Airborne contamination	1) Unavailability/failure of Main Plant Ventilation System (which services the cask decontamination stall) during cask decon 2) Unavailability/failure of the portable ventilation unit during transfer of loaded resin to on-line HIC, or during dewatering 3) Damage to spent nuclear fuel assembly(ies) (See Note 3) 4) Component or pipe integrity failure during transfer of loaded resin 5) Breaching of all HEPA filters in the Recirculation Ventilation System 6) Tornado of magnitude less than the design basis tornado, but nevertheless capable of causing upsets in the FRS facility	- Maintenance program - Redundant blower/filter trains in Main Plant Ventilation System - Backup power for Main Plant Vent System - Portable ventilation unit required to be operating during dewatering (which is performed prior to and during resin transfers to on-line HIC) - Avoiding jarring movements with fuel assemblies and canisters containing fuel - Use of formal procedures during fuel movement operations - Monitoring of differential pressure across HEPA filters	1) Low 2) Low 3) Moderate 4) Low 5) Low 6) Low	1) Extremely unlikely 2) Extremely unlikely 3) Unlikely 4) Unlikely 5) Extremely unlikely 6) Unlikely	1) 1 2) 1 3) 5 4) 2 5) 1 6) 2

TABLE 9.1-1 (Concluded)

Hazard	Event	Preventive and Mitigative Features	Consequences[1]	Frequency	Risk Factor[2]
Loaded resin	1) Spill due to component failure or misalignment, or pipe or connection integrity failure, during loaded resin transfer 2) Overfilling of a HIC and connected overflow drum 3) HIC integrity failure (e.g., leak) 4) Failure of HIC due to dropping 5) Unnecessary external gamma radiation exposure 6) Loss of HIC shielding 7) Explosion due to H_2 generation in full HIC 8) Overpressurization of HIC due to H_2 generation	- Maintenance program - System walkdown/inspection prior to loaded resin transfers - Independent verification of valve lineup - All hose connections wrapped in plastic bags, and secured with tape - HIC level indication - Auto-termination of transfers if high-level detected in HIC or liquid detected in HIC overflow drum - HIC enclosed in shielding structure - Radwaste Building perimeter bermed - Rugged construction of a HIC - Use of Surepaks for temporary storage - Established hoisting and rigging procedures, including load testing - Design conservatisms for lifting equipment (typically minimum safety factor of 5, based on ultimate strength of material) - Resin replacement performed remotely - ALARA practices - A distance of 10 feet or more is maintained from resin transfer hose during sluice out of loaded resin - Vent in HIC - No ignition sources at HIC - HIC resins dewatered	1) Low 2) Low 3) Low 4) Moderate 5) Low 6) Low 7) Moderate 8) Low	1) Unlikely 2) Extremely unlikely 3) Extremely unlikely 4) Extremely unlikely 5) Unlikely 6) Unlikely 7) Incredible 8) Incredible	1) 2 2) 1 3) 1 4) 3 5) 2 6) 2 7) I 8) I
Used filter cartridges	1) Improper handling and/or storage after removal from the pool gives unnecessary external gamma radiation exposure to workers 2) Radiological equipment malfunctions and/or operator errors provide incorrect (low) dose rate reading of used filter cartridges	- Maintenance program and instrument calibration program - Used filter cartridges are stored in 55 gallon steel drums - Used filter cartridge replacement performed mostly remotely - ALARA practices	1) Low 2) Low	1) Unlikely 2) Extremely unlikely	1) 2 2) 1

Notes:

1. Column entries refer to postulated <u>radiological</u> consequences.
2. See Section 9.1.1.2 for an explanation of Risk Factor.
3. Approximately 6 SNF assemblies exhibit minor cladding failure. Due to the limited nature of the damage, engineering judgment is that handling of these assemblies (during cask loading or otherwise) will not lead to airborne contamination. If airborne contamination should occur, the consequences are considered to be bounded by the analysis of the dropping of a SNF assembly.

TABLE 9.2-1
DROP OF A FUEL ASSEMBLY IN THE FRS

Assumptions:

ARF x RF x DR x LPF (Pool Release) 1.05E-04 - See Note [1] Below
ARF x RF x DR x LPF (Atmospheric Release) 2.00E-03 - See Note [1] Below
Release Height 60 m (Elevated Release)

Receptor Location				640 m	640 m	640 m	1050 m	1050 m	1700 m	
Stability Class, Wind Speed				D, 4.5 m/s	F, 1 m/s	95%	D, 4.5 m/s	F, 1 m/s	95%	
Dispersion (X/Q)				1.59E-06 s/m³	2.70E-11 s/m³	1.63E-04 s/m³	5.54E-06 s/m³	1.03E-07 s/m³	6.72E-05 s/m³	
Nuclide	Decayed PWR Activity (Ci)	Pool Source Term (Ci)	Atmospheric Source Term (Ci)	On-Site Dose (rem)	On-Site Dose (rem)	On-Site Dose (rem)	Off-Site Dose (rem)	Off-Site Dose (rem)	Off-Site Dose (rem)	Percent Dose Contribution
Am-241	1.05E+03	1.10E-01	2.21E-04	6.08E-05	1.03E-09	6.23E-03	2.12E-04	3.94E-06	2.57E-03	41.8%
Pu-238	7.47E+02	7.85E-02	1.57E-04	3.82E-05	6.49E-10	3.92E-03	1.33E-04	2.48E-06	1.62E-03	26.3%
Pu-241	1.43E+04	1.50E+00	3.00E-03	1.59E-05	2.70E-10	1.63E-03	5.53E-05	1.03E-06	6.71E-04	10.9%
Pu-240	1.95E+02	2.05E-02	4.10E-05	1.11E-05	1.88E-10	1.14E-03	3.86E-05	7.17E-07	4.68E-04	7.6%
Cm-244	2.29E+02	2.40E-02	4.80E-05	6.87E-06	1.17E-10	7.04E-04	2.39E-05	4.45E-07	2.90E-04	4.7%
Pu-239	1.19E+02	1.25E-02	2.49E-05	6.73E-06	1.14E-10	6.90E-04	2.35E-05	4.36E-07	2.84E-04	4.6%
H-3	7.94E+01	7.94E+01	7.94E+01	2.65E-06	4.50E-11	2.72E-04	9.23E-06	1.72E-07	1.12E-04	1.8%
Sr-90	1.61E+04	1.69E+00	3.38E-03	2.34E-06	3.98E-11	2.40E-04	8.16E-06	1.52E-07	9.90E-05	1.6%
Kr-85	7.68E+02	7.68E+02	7.68E+02	4.34E-07	7.36E-12	4.45E-05	1.51E-06	2.81E-08	1.83E-05	0.3%
Am-243	5.41E+00	5.68E-04	1.14E-06	3.13E-07	5.31E-12	3.21E-05	1.09E-06	2.03E-08	1.32E-05	0.2%
Cs-137	2.27E+04	2.38E+00	4.77E-03	8.22E-08	1.40E-12	8.43E-06	2.86E-07	5.33E-09	3.47E-06	0.1%
			Total	1.45E-04	2.47E-09	1.49E-02	5.07E-04	9.42E-06	6.14E-03	100%
				1.82E-02	3.09E-07	1.86E+00	6.33E-02	1.18E-03	7.68E-01	

Notes:

[1] - Based on values given in DOE-HDBK-3010-94

[2] - Based on nuclides expected to be present in spent nuclear fuel in storage in the FRS. Nuclides given here represent those that contribute greater than 0.1% of the TEDE.

[3] - Activity based on PWR fuel assemblies having 0.382 MTU per assembly.

TABLE 9.2-2
DROP OF A HIGH INTEGRITY CONTAINER

Assumptions:

ARF x RF x DR x LPF	4.00E-05 - see note [1] below
Release Height	0 m (ground level release)
Mass of sludge in HIC	3200 kg

Receptor Location				640 m	640 m	640 m	1050 m	1050 m	2350 m	
Stability Class, Wind Speed				D, 4.5 m/s	F, 1 m/s	95%	D, 4.5 m/s	F, 1 m/s	95%	
Dispersion (χ/Q)				6.35E-05 s/m³	1.49E-03 s/m³	7.26E-04 s/m³	2.85E-05 s/m³	6.85E-04 s/m³	7.07E-04 s/m³	
Nuclide	Conc. of Activity in HIC "B" (μCi/g)	Total Activity in HIC "B" (Ci)	Source Term (Ci)	On-Site Dose (rem)	On-Site Dose (rem)	On-Site Dose (rem)	Off-Site Dose (rem)	Off-Site Dose (rem)	Off-Site Dose (rem)	Percent of Dose
Am-241	5.63E-02	1.80E-01	7.21E-06	7.93E-05	1.86E-03	9.06E-04	3.56E-05	8.55E-04	8.83E-04	29.7%
Pu-239*	4.76E-02	1.52E-01	6.10E-06	6.58E-05	1.54E-03	7.52E-04	2.95E-05	7.09E-04	7.32E-04	24.6%
Pu-240*	3.63E-02	1.16E-01	4.65E-06	5.01E-05	1.18E-03	5.73E-04	2.25E-05	5.41E-04	5.58E-04	18.8%
Pu-241**	1.79E+00	5.74E+00	2.29E-04	4.85E-05	1.14E-03	5.55E-04	2.18E-05	5.23E-04	5.40E-04	18.2%
Pu-238	1.34E-02	4.27E-02	1.71E-06	1.66E-05	3.90E-04	1.90E-04	7.46E-06	1.79E-04	1.85E-04	6.2%
Cs-137	6.52E+01	2.09E+02	8.35E-03	5.65E-06	1.33E-04	6.46E-05	2.54E-06	6.09E-05	6.29E-05	2.1%
Co-60	1.63E+00	5.21E+00	2.08E-04	6.61E-07	1.55E-05	7.55E-06	2.97E-07	7.13E-06	7.36E-06	0.2%
Sr-90	6.80E-01	2.18E+00	8.70E-05	4.23E-07	9.93E-06	4.84E-06	1.90E-07	4.56E-06	4.71E-06	0.2%
Cs-134	1.41E+00	4.51E+00	1.80E-04	1.79E-07	4.21E-06	2.05E-06	8.05E-08	1.93E-06	2.00E-06	0.1%
			Total TEDE	2.67E-04	6.27E-03	3.05E-03	1.20E-04	2.88E-03	2.97E-03	100%

* Pu-239 and Pu-240 isotopic proportions are based on SNF distribution as reported in Wolniewicz, 1993.
** Pu-241 estimated by Pu-239: Pu-241 ratio reported in Wolniewicz, 1993.
[1] - Based on values given in DOE-HDBK-3010-94.

TABLE 9.2-3
INADVERTENT CRITICALITY IN THE FRS

Assumptions:	
ARF x RF x DR x LPF (particulate)	Ref. Table 6-10, DOE-HDBK-3010-94
ARF x RF x DR x LPF (fission gas)	10% of Total Source Term, Table 6-7, DOE-HDBK-3010-94
Release Height	60 m

Receptor Location		640 m	640 m	640 m	1050 m	1050 m	1700 m	
Stability Class, Wind Speed		D, 4.5 m/s	F, 1 m/s	95%	D, 4.5 m/s	F, 1 m/s	95%	
Dispersion (χ/Q)		1.59E-06 s/m³	2.70E-11 s/m³	1.63E-04 s/m³	5.54E-06 s/m³	1.03E-07 s/m³	6.72E-05 s/m³	
Nuclide	Source Term [Ci]	On-Site Dose [rem]	On-Site Dose [rem]	On-Site Dose [rem]	Off-Site Dose [rem]	Off-Site Dose [rem]	Off-Site Dose [rem]	Percent Dose Contribution
Sr-90	3.42E+00	2.37E-03	4.03E-08	2.43E-01	8.26E-03	1.54E-04	1.00E-01	28.4%
Kr-89	4.10E+03	2.11E-03	3.58E-08	2.16E-01	7.34E-03	1.36E-04	8.90E-02	25.2%
Cs-134	4.43E+01	1.12E-03	1.90E-08	1.15E-01	3.90E-03	7.25E-05	4.73E-02	13.4%
Cs-137	3.16E+01	5.45E-04	9.26E-09	5.59E-02	1.90E-03	3.53E-05	2.30E-02	6.5%
Pu-241	7.25E-02	3.84E-04	6.52E-09	3.94E-02	1.34E-03	2.49E-05	1.62E-02	4.6%
Ru-106	1.53E+00	3.56E-04	6.05E-09	3.65E-02	1.24E-03	2.31E-05	1.50E-02	4.3%
Xe-138	1.10E+03	3.48E-04	5.91E-09	3.57E-02	1.21E-03	2.25E-05	1.47E-02	4.2%
Pu-238	1.29E-03	3.15E-04	5.35E-09	3.23E-02	1.10E-03	2.04E-05	1.33E-02	3.8%
Xe-137	4.90E+03	2.36E-04	4.01E-09	2.42E-02	8.23E-04	1.53E-05	9.98E-03	2.8%
Cm-244	1.38E-03	1.97E-04	3.34E-09	2.02E-02	6.85E-04	1.27E-05	8.31E-03	2.4%
Pu-240	3.12E-04	8.42E-05	1.43E-09	8.63E-03	2.93E-04	5.45E-06	3.56E-03	1.0%
Pu-239	1.87E-04	5.04E-05	8.55E-10	5.16E-03	1.76E-04	3.26E-06	2.13E-03	0.6%
H-3	1.22E+03	4.08E-05	6.92E-10	4.18E-03	1.42E-04	2.64E-06	1.72E-03	0.5%
Kr-88	6.60E+01	3.76E-05	6.38E-10	3.85E-03	1.31E-04	2.43E-06	1.59E-03	0.4%
I-134	4.80E+01	3.61E-05	6.14E-10	3.71E-03	1.26E-04	2.34E-06	1.53E-03	0.4%
Xe-135m	3.30E+02	3.57E-05	6.07E-10	3.66E-03	1.25E-04	2.31E-06	1.51E-03	0.4%
Am-241	9.79E-05	2.70E-05	4.58E-10	2.76E-03	9.39E-05	1.75E-06	1.14E-03	0.3%
Kr-87	1.00E+02	2.25E-05	3.82E-10	2.31E-03	7.84E-05	1.46E-06	9.52E-04	0.3%
I-135	1.20E+01	1.20E-05	2.04E-10	1.23E-03	4.19E-05	7.79E-07	5.08E-04	0.1%
I-133	3.50E+00	1.05E-05	1.79E-10	1.08E-03	3.68E-05	6.83E-07	4.46E-04	0.1%
Co-60	3.43E-01	5.68E-06	9.65E-11	5.83E-04	1.98E-05	3.68E-07	2.40E-04	0.1%
	Total TEDE	8.36E-03	1.42E-07	8.57E-01	2.91E-02	5.41E-04	3.53E-01	99.8%

Notes:

[1] - Nuclides not contributing at least 0.1% to the TEDE not reported in table.

[2] - Source term activity based on PWR fuel having 0.382 MTU per assembly.

TABLE 9.2-4

SUMMARY OF FRS ACCIDENT CONSEQUENCES

Accident Scenario	Maximum Off-Site Dose (rem)	Maximum On-Site Dose (rem)	Evaluation Guideline Level
Dropping of a Fuel Assembly in the FRS	6.14E-03	1.49E-02	On-site - 25 rem Off-site - 5 rem
Dropping of a Loaded High Integrity Container	2.97E-03	6.27E-03	On-site - 100 rem Off-site - 25 rem
Inadvertent Criticality in the FRS	3.53E-01	8.57E-01	On-site - 100 rem Off-site - 25 rem
Failure of 125 SNF Assemblies Due to Seismic Event	7.68E-01	1.86E+00	On-site - Natural Phenomena, N/A Off-site - 25 rem

SR12-911.DWG

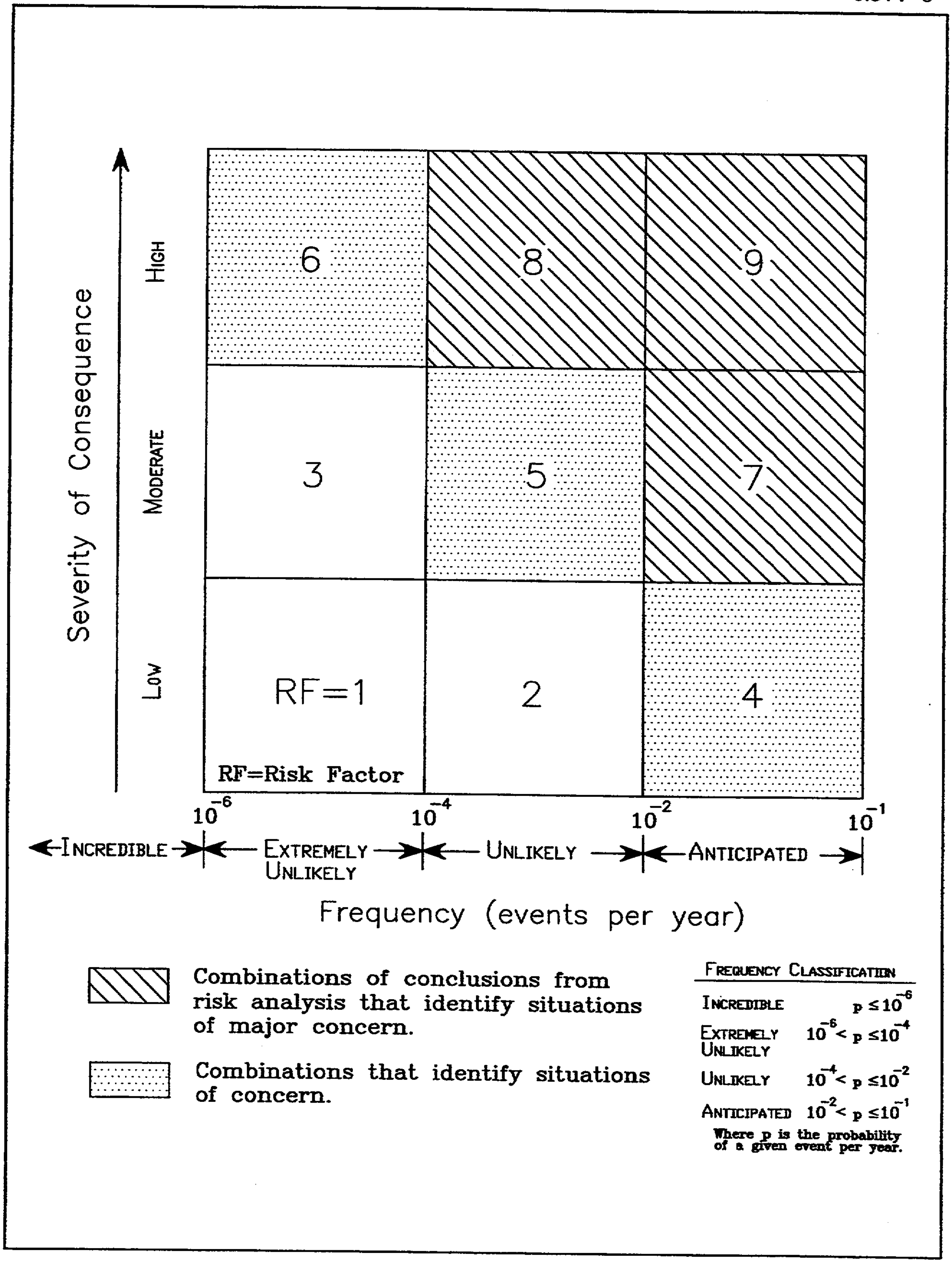

Figure 9.1–1 Process Hazards Analysis Risk Bins

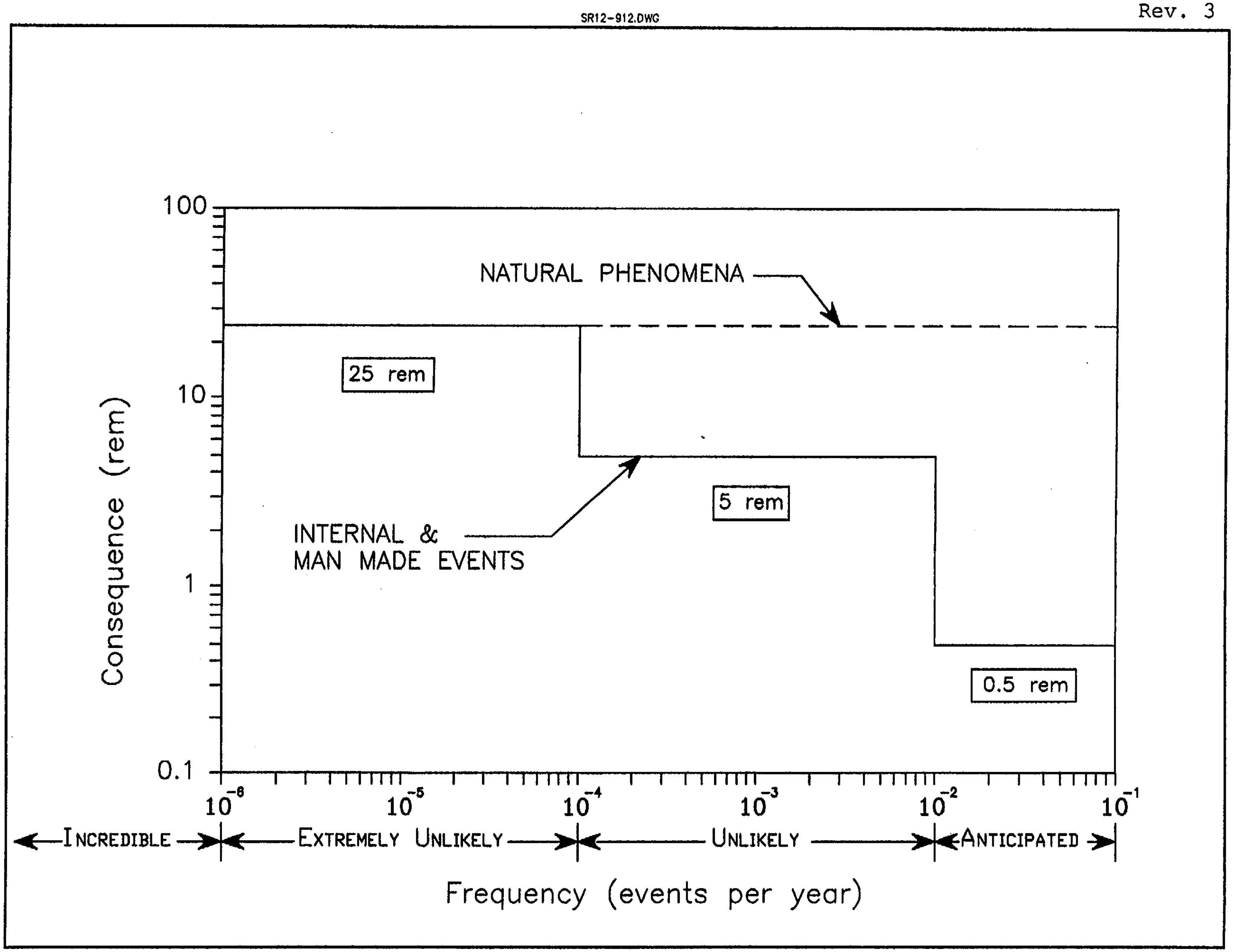

Figure 9.1–2 Evaluation Guidelines for Off–site Evaluation Point

SR12-913.DWG

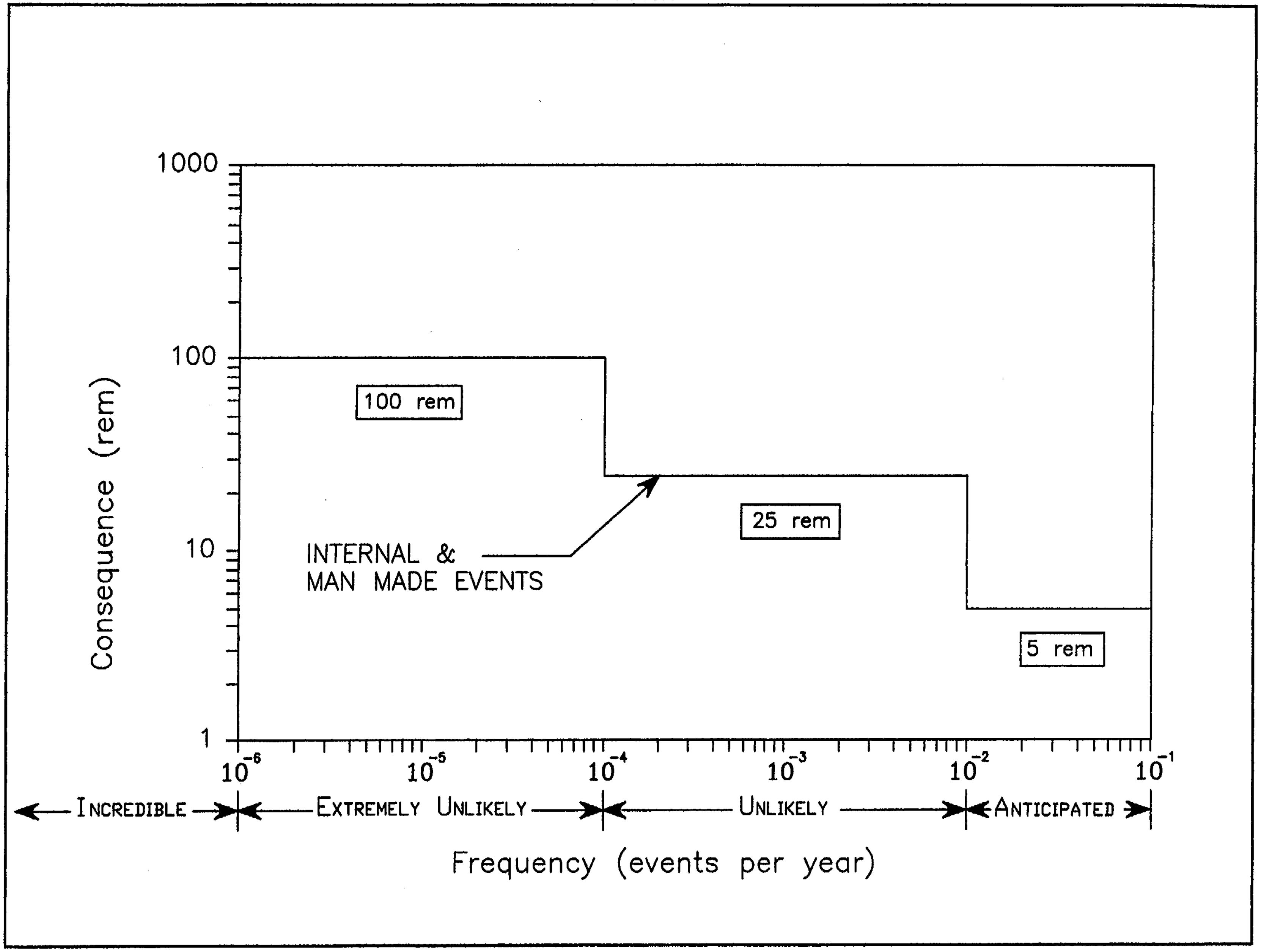

Figure 9.1-3 Evaluation Guildlines for On-site Evaluation Point

10.0 CONDUCT OF OPERATIONS

The WVDP Conduct of Operations program is presented in detail in Chapter A.10.0 of WVNS-SAR-001, *Project Overview and General Information* (WVNS).

10.1 Management, Organization, and Institutional Safety Provisions

10.1.1 Organizational Structure

Engineering support for the FRS is provided by the Spent Fuel Engineering Project and operations support is provided by Spent Fuel Shipping Operations.

The overall WVDP organizational structure is presented in Sections A.10.1 and A.10.2 of WVNS-SAR-001.

10.1.2 Organizational Responsibilities

WVDP organizational responsibilities are discussed in Sections A.10.1 through A.10.4 of WVNS-SAR-001.

10.1.3 Staffing and Qualifications

WVDP staffing and qualifications are discussed in Section A.10.1 of WVNS-SAR-001.

10.1.4 Safety Management Policies and Programs

Safety performance assessment, configuration and document control, event reporting, and safety culture are discussed in Section A.10.4.2 of WVNS-SAR-001.

10.2 Procedures and Training

10.2.1 Procedures

The development and maintenance of procedures is discussed in Section A.10.4.1 of WVNS-SAR-001.

10.2.2 Training

A description of the WVNS training program is presented in Section A.10.3 of WVNS-SAR-001. The training program for fuel handlers ensures that they are certified as appropriate in accordance with requirements contained in DOE Order 5480.20A, *Personnel Selection, Qualification and Training Requirements for DOE Nuclear Facilities* (U.S. DOE, November 15, 1994).

10.3 Initial Testing, In-Service Surveillance, and Maintenance

10.3.1 Initial Testing Program

The FRS Facility began operations in 1966 as part of the original Nuclear Fuel Services reprocessing efforts. Prior to startup, preoperational functional checkouts of major equipment and systems were performed by both NFS and Bechtel.

10.3.2 In-Service Surveillance and Maintenance Program

A complete description of the WVDP In-Service Surveillance and Maintenance Program is presented in Section A.10.4.3 of WVNS-SAR-001.

10.4 Operational Safety

10.4.1 Conduct of Operations

The WVDP Conduct of Operations Program is discussed in Section A.10.4.4 of WVNS-SAR-001.

10.4.2 Fire Protection

The WVDP Fire Protection Program is discussed in Section A.4.3.6 of WVNS-SAR-001.

10.5 Emergency Preparedness Program

The WVDP Emergency Preparedness Program is presented in detail in Section A.10.5 of WVNS-SAR-001.

10.6 Decontamination and Decommissioning

Though extensive decontamination of the Main Plant building has already been conducted in support of WVDP activities, final decontamination and decommissioning (D&D) plans are dependent on facility closure plans which are yet to be determined. Facility design features which will facilitate final D&D have been described in Section B.4.5. Safety analyses and Unreviewed Safety Question Determinations (USQDs) associated with site D&D activities will be performed as appropriate.

The WVDP Decommissioning Program is also discussed in Section A.10.6 of WVNS-SAR-001.

REFERENCES FOR CHAPTER 10.0

U.S. Department of Energy. November 15, 1994. DOE Order 5480.20A, *Personnel Selection, Qualification and Training Requirements for DOE Nuclear Facilities*.

West Valley Nuclear Services Co., Inc. Safety Analysis Report WVNS-SAR-001: *Project Overview and General Information*. (Latest Revision.)

fuel handling supplement the engineered features to ensure that an inadvertent criticality does not occur within the FRS facility.

Worker protection at the WVDP is achieved through administration of DOE-required radiological protection, occupational safety and health programs. In its discussion of worker safety, DOE Order 5480.22 acknowledges that "The impact from the release of hazardous materials is also reduced through industrial hygiene and radiation protection oversight (e.g., monitoring of worker exposures, use of personnel protective equipment [PPE] and emergency evacuation planning), as well as the use of TSRs." This statement indicates that formal measures other than TSRs are recognized by the DOE as being acceptable for ensuring worker safety. DOE-STD-3009-94 reinforces this position, stating: "It is important to develop TSRs judiciously. TSRs should not be used as a vehicle to cover the many procedural and programmatic controls inherent in any operation." Consistent with relevant DOE Orders and federal and state regulations with which WVNS is currently contractually obligated to comply, the control of the levels of hazardous and radioactive materials to which workers may, at any time, be exposed, is addressed in WVDP radiological protection, occupational safety and health programs. Furthermore, worker exposure to hazardous materials and/or conditions is regulated under the provisions of the Occupational Safety and Health Act administered by the Occupational Safety and Health Administration (OSHA).

The existing authorization basis documents at the WVDP recognize the measures provided by existing site programs for protecting the health and safety of workers. In this regard, TSR administrative controls would not further contribute to worker safety at the WVDP.

In consideration of the above discussion, no TSR administrative controls are required for facilities or activities within the scope of this SAR.

11.4 Interface With TSRs From Other Facilities

There are no TSRs from other facilities that interface with the facilities within the scope of this SAR.

REFERENCES FOR CHAPTER 11

U.S. Department of Energy. February 25, 1992. Change 1 (September 15, 1992.) DOE Order 5480.22: *Technical Safety Requirements*. Washington, D.C.: U.S. Department of Energy.

__________. April 30, 1992. Change 1 (March 10, 1994.) DOE Order 5480.23: *Nuclear Safety Analysis Reports*. Washington, D.C.: U.S. Department of Energy.

__________. July, 1994. DOE Standard DOE-STD-3009-94: *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*. Washington, D.C.: U.S. Department of Energy.

West Valley Nuclear Services Co., Inc. WVDP-082: *Hoisting and Rigging Manual*. (Latest Revision.) West Valley Nuclear Services Co., Inc.

__________. WVDP-162: *Nuclear Criticality Safety Program Plan*. (Latest Revision.) West Valley Nuclear Services Co., Inc.

11.0 TECHNICAL SAFETY REQUIREMENTS

11.1 Introduction

The objective of this chapter is to provide information that will satisfy the requirements of DOE Order 5480.23, *Nuclear Safety Analysis Reports*, Section 8.b.(3)(p), *Derivation of TSRs*. This chapter is intended to link the accident analyses, through descriptions of the Safety Class structures, systems, and components (SSCs) to the Technical Safety Requirements (TSRs). The TSR document, as stated in DOE Order 5480.22, *Technical Safety Requirements*, is intended to constitute an agreement or contract between DOE and WVNS regarding the safe operation of the WVDP facilities.

Safety Class SSCs are those structures, systems, or components whose preventive and/or mitigative functions are necessary to maintain the consequence of an accident below the off-site evaluation guidelines provided in Section 9.1.3. Because the accidents analyzed in Chapter 9 do not rely on protective or mitigative features to maintain dose consequences below the evaluation guidelines, no TSRs are required for the activities addressed in this SAR.

11.2 Requirements

This SAR meets the requirements in DOE Orders 5480.23 and 5480.22, with respect to TSRs. There are no TSRs associated with activities covered by this SAR.

11.3 TSR Input

There are no enveloping Evaluation Basis Accidents that exceed the Evaluation Guidelines. There are no active Safety Class SSCs in facilities within the scope of this SAR, nor are there any Safety Class SSCs which are under the direct control of operators of facilities within the scope of this SAR.

11.3.1 Safety Limits and Limiting Conditions for Operation

There are no evaluation basis accidents which require active Safety Class SSCs nor Safety Class SSCs under the direct control of operators of facilities within the scope of this SAR to mitigate the consequences or prevent the occurrence to meet the Evaluation Guidelines (EGs). Initial accident conditions under the direct control of the operator have been analyzed at the maximum credible worst-case conditions (e.g., maximum vessel inventory, maximum concentration).

Therefore, no TSR Safety Limits or TSR Limiting Conditions for Operation (LCO) are required for facilities or activities within the scope of this SAR. The WVDP has initiated Process Safety Requirements (PSRs) which contain PSR LCOs as well as associated Surveillance Requirements (see Sections 4.3.2.3 and 11.3.3).

11.3.2 Design Features

The primary safety features in the Fuel Receiving and Storage (FRS) facility are the fuel storage racks, fuel storage canisters and the pool shielding water. FRS facility design features are described in Chapters 4, 5, and 6 of this SAR.

11.3.3 Administrative Controls

Administrative Controls are the provisions relating to organization and management, procedures, record keeping, reviews, and audits necessary to ensure safe operation of the facility.

Technical Safety Requirements are not based upon maintaining worker exposures below some acceptable level following an uncontrolled release of hazardous material or inadvertent criticality; rather the risk to workers is reduced through the reduction of the likelihood and potential impact of such events. Because of the necessary and inherent presence of hazardous and radioactive materials at WVDP nuclear facilities and the workers' proximity to these materials, it is impractical to reduce worker risk to an insignificant level through TSRs. The consequences of occupational exposures resulting from the release of hazardous and radioactive materials at the WVDP is reduced through the implementation of industrial hygiene and radiation protection programs which have been developed consistent with guidance given in relevant DOE Orders.

Engineered and administrative controls are provided for FRS facilities and operations to ensure that the occurrence of an inadvertent criticality or other operational mishap (e.g., dropping of a load from a crane) is prevented. Administrative controls for the prevention of an inadvertent criticality at the WVDP are developed through the guidelines given in WVDP-162 and the references contained therein. Hoisting and rigging activities associated with FRS facility operations are subject to the requirements of WVDP-082, *Hoisting and Rigging Manual*, with details of such activities provided in procedures as appropriate.

Safe conditions are maintained during fuel handling operations through the use of administrative controls that restrict fuel movements and locations based on (1) the quantity, type, and location of SNF assemblies at any given time, and/or (2) other activities that may be occurring within the FRS Building at any given time (e.g., installation of the fuel pool gate, placement of a shipping cask into the cask unloading pool [CUP], etc.). These controls are reflected in various existing procedures and/or Process Safety Requirements. These administrative controls for

12.0 QUALITY ASSURANCE

The Quality Assurance Program (QAP) at the WVDP is implemented on a site-wide basis and is applied in compliance with the QA Rule, 10 CFR 830.120, *Quality Assurance Requirements*. Definition and description of the WVNS QAP is provided by the DOE-approved WVNS document WVDP-111, *Quality Assurance Program* (WVNS).

The Quality Assurance Program provides guidance for determining the graded applicability of quality assurance standards to items, systems, or services. The FRS facility structures, systems, and components that are covered by the QAP are graded and identified by quality level, which is based upon safety, environmental, health, and other programmatic considerations. The assigned list, methodology for classification, and rationale for establishment of quality levels are contained in WVDP-204, *WVDP Quality List (Q-List)* (WVNS). With activities clearly identified by quality level, existing WVNS procedures and practices provide a mechanism and process for graded quality assurance. Criteria for quality level designations are provided in Section A.12.3 of WVNS-SAR-001.

The WVNS Quality Assurance Program is presented in Chapter A.12.0 of WVNS-SAR-001, *Project Overview and General Information* (WVNS).

REFERENCES FOR CHAPTER 12.0

U.S. Department of Energy. Quality Assurance Requirements. 10 CFR 830.120.

West Valley Nuclear Services Co., Inc. WVDP-111: Quality Assurance Program Plan. (Latest Revision.)

__________. WVDP-204: WVDP Quality List (Q-List) (Latest Revision.)

__________. Safety Analysis Report WVNS-SAR-001: *Project Overview and General Information*. (Latest Revision.)