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028	WVNS-SAR-002	2		06/22/89	EXISTING PLANT AND OPERATIONS VOLUME II

NOTE

-ATTACHED IS ADDENDUM #1, REV. 3 TO WVNS-SAR-002, REV. 2.
-INSERT ADDENDUM #1 DIRECTLY AFTER SECTION B.9.0.
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RECORD OF REVISION

PROCEDURE

If there are changes to the procedure, the revision number increases by one. These changes are indicated by a bar in the right margin of the body.

Example:

The bar in the margin indicates a change. |

Rev. No.	Description of Changes	Revision On Page(s)	Dated
0	Original Issue	All	
1	Revise text to correct inaccurate descriptions of facilities and operations, update dose calculations for annual meteorology data, and adjust source terms to reference year 1987	Various changes indicated in margins	05/85
2	Per ECN #2890. Revise text to incorporate requirements for HLW sampling and resulting accident analysis	83, 84, 192 213, 214	06/89
3	Per ECN #4288 - Issue of Addendum #1 which includes WVNS-SAR-011 DOE Approved DW:92:1367	Addendum #1	12/23/92

WV-1807A, Rev. 0
-COM005174:SAR-2

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LIST OF ACRONYMS AND ABBREVIATIONS

ALARA	As Low As Reasonably Achievable
amp	Ampere
ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing Material
BRP	Big Rock Point
BWR	Boiling Water Reactor
C	Centigrade
CAM	Continuous Air Monitor
CEDE	Committed Effective Dose Equivalent
cfm	Cubic Feet per Minute
Ci	Curie
cm	Centimeter
cpm	Counts per Minute
CUP	Cask Unloading Pool
DCF	Dose Conversion Factor
DF	Decontamination Factor
DOE	Department of Energy
DOP	Dioctyl Phthalate
DOT	Department of Transportation
EDE	Effective Dose Equivalent
EPA	Environmental Protection Agency
FRS	Fuel Receiving and Storage
FSP	Fuel Storage Pool
g	Gram
g	Gravitational Acceleration
G.M.	Geiger Mueller
GP	General Plant
gpm	Gallons per Minute
HEPA	High Efficiency Particulate Air
HIC	High Integrity Container
HVAC	Heating Ventilation and Air Conditioning

LIST OF ACRONYMS AND ABBREVIATIONS
(Continued)

I.D.	Inner Diameter
ID	Idaho Operations Office
IWP	Industrial Work Permit
kg	Kilogram
kw	Kilowatt
L	Liter
LLL	Lawrence Livermore Laboratory
LLWT	Low-Level Waste Treatment
m	Meter
m	Milli, prefix for 10^{-3}
m/s	Meters per Second
MCC	Motor Control Center
mph	Miles per Hour
mR	Milliroentgen
mrem	Millirem
MT	Metric Ton
MTU	Metric Ton Uranium
MWD	Mega-watt Day
NFS	Nuclear Fuel Services
NRC	Nuclear Regulatory Commission
OSR	Operational Safety Requirement
P&ID	Piping and Instrumentation Diagram
PWR	Pressurized Water Reactor
psf	pounds per square foot
PVU	Portable Ventilation Unit
QA	Quality Assurance
QM	Quality Management

LIST OF ACRONYMS AND ABBREVIATIONS
(Continued)

REG	Robert E. Ginna
rh	Relative Humidity
RWP	Radiation Work Permit
SAI	Science Applications, Incorporated
SAR	Safety Analysis Report
SNF	Spent Nuclear Fuel
SOP	Standard Operating Procedure
SSE	Safe Shutdown Earthquake
TN	Trans Nuclear
TR	Technical Requirement
TRU	Transuranic
UBC	Uniform Building Code
USAEC	United States Atomic Energy Commission
v	Volt
VEC	Vent Exhaust Cell
v/o	Volume Percent
w/o	Weight Percent
WTA	Water Treatment Area
WVDP	West Valley Demonstration Project
WVNS	West Valley Nuclear Services Co., Inc.
α	alpha
β	beta
γ	gamma
μ	micro, prefix for 10^{-6}

LIST OF ACRONYMS AND ABBREVIATIONS
(Concluded)

Instrument Nomenclature

A	Alarm
AI	Alarm Indicator
F	Flow
FG	Flow Gauge
FI	Flow Indicator
L	Level
LI	Level Indicator
P	Pressure
PDI	Pressure Differential Indicator
PG	Pressure Gauge
PI	Pressure Indicator
T	Temperature
TG	Temperature Gauge

B.1.0 INTRODUCTION AND GENERAL DESCRIPTION OF THE FACILITY

B.1.1 INTRODUCTION

This addendum to Safety Analysis Report (SAR) WVNS-SAR-002, Volume II, Part B, Fuel Receiving and Storage (FRS), was prepared to meet the requirements of the U.S. Department of Energy (DOE) Order DOE-5481.1B (U.S. DOE, 1986), Idaho Operations Office (ID) Order ID-5481.1A (U.S. DOE-ID, 1989), and West Valley Nuclear Services Co., Inc. (WVNS) Policy and Procedure WV-906, Rev. 8 (WVNS, 1990). Further introductory information relating to the WVDP Act, ancillary tasks and supporting activities that must be accomplished may be obtained from Section A.1.1 of Volume I, Project Overview and General Information, of the Project SAR (WVNS-SAR-001).

B.1.2 GENERAL PLANT DESCRIPTION

The WVDP site is located in a rural setting approximately 50 km (30 mi) south of Buffalo, New York, at an average elevation of 400 m (1,300 ft) on New York State's western plateau. The plant facilities used by the WVDP occupy approximately 63 hectares (156 acres) of chain-link fenced area within a 1,350 hectares (3,300 acres) reservation that constitutes the Western New York Nuclear Service Center (WNYNSC). 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 site, but no human habitation, hunting, fishing or public access is permitted on the WNYNSC.

The Fuel Receiving and Storage (FRS) Facility located adjacent to the original fuel reprocessing plant at the West Valley Demonstration Project (WVDP) contains two interconnected pools: the Fuel Storage Pool (FSP) and the Cask Unloading Pool (CUP). The original analyses which established the critically safe design for the FRS facility were done in the early 1960s. The results of these analyses are reported in the 1964 Nuclear Fuel Services (NFS) Safety Analysis Report. In the

mid-1970s criticality analyses were performed and documented to support a more compact storage configuration (NFS, 1973). These analyses represent a very conservative assessment of the present storage configuration and, by comparison with the present configuration, demonstrate that the fuel currently stored in the pool is a subcritical array with a k_{eff} less than 0.95, and will remain subcritical in the event of all postulated credible accidents. Subsequent calculations performed in 1983 have shown that the k_{eff} limit of 0.95 will not be exceeded for intended operations and postulated accident scenarios (WVNS, 1983).

Seismic analysis of the facility has indicated that the pool and canister racks will survive the Safe Shutdown Earthquake (intensity VII-VIII) with an epicenter 37 km (23 miles) from the site (SAI, 1981). (SSE as used in this document is equivalent to the design basis earthquake [DBE]). The information presented in these analyses confirms that a single fuel assembly remains subcritical under normal and accident conditions during transport from its position in the fuel storage racks to the CUP and from the CUP into an approved shipping cask.

Throughout this document "FRS building" refers to the structure containing the fuel storage pool, cask unloading pool and water treatment area, while "FRS facility" refers to the FRS building and associated support systems, including the Recirculation Ventilation System and the Radwaste Treatment System.

B.1.3 GENERAL PROCESS DESCRIPTION

A simplified description of the overall WVDP activities is presented in Section A.1.3 of Volume I.

The FRS facility was designed for transfer and storage of irradiated fuel assemblies in a manner that assures that the array remains subcritical. These assemblies are stored in canisters located within racks designed to maintain the fuel subcritical by controlling the geometry of the array. Storage configuration and fuel transfer operations are controlled by Operational Safety

Requirement OSR / GP-11. During decommissioning activities in the 1980s, approximately 75 percent of the canister storage racks were removed and 625 of the 750 assemblies stored in the pool were returned to the owner utility companies. At present, 85 boiling water reactor (BWR) assemblies and 40 pressurized water reactor (PWR) assemblies remain in the pool. The United States Nuclear Regulatory Commission (NRC) has approved certification of special shipping casks for rail shipment of these fuel assemblies. As of October 1991, DOE-HQ has stated that the assemblies will not be shipped until 1994 or later. Additional fuel assemblies will not be received.

B.1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

Section A.1.4 of Volume I identifies the agents and contractors responsible for implementing the WVDP. The relationships between WVNS and agents and contractors is illustrated in Figure A.1.4-1 of Volume I.

B.1.5 STRUCTURE OF THIS SAR

The concept of this modular SAR is explained in Section A.1.5 of Volume I. The structure of the SAR is patterned after NRC Regulatory Guide 3.26. See the Table of Contents, which reflects this pattern.

An evaluation of the safety of all operations associated with the FRS Radwaste Treatment System has been documented in this addendum. This evaluation includes all analyses formerly covered by WVNS-SAR-011, "Transfer of High Integrity Containers".

B.1.6 REFERENCES

NFS, 1973 Nuclear Fuel Services, Inc., "Nuclear Safety Evaluation of Arrays of Light Water Reactor Fuel in the Fuel Storage Pool," J. R. Clark to L. C. Rouse, USAEC, August, 1973.

WVNS, 1983 West Valley Demonstration Project, "Safety Assessment Document for the West Valley Demonstration Project Fuel Receiving and Storage Area, West Valley, New York," September, 1983.

SAI, 1981 Science Applications, Inc., NUREG/CR-2236, "Seismic Resistance Capacity Evaluation of Spent Fuel storage Racks and Fuel at West Valley, New York," December, 1981.

B.2.0 SUMMARY SAFETY ANALYSIS

B.2.1 SITE ANALYSIS

B.2.1.1 NATURAL PHENOMENA

See Section A.2.1 of Volume I.

B.2.1.2 SITE CHARACTERISTICS AFFECTING THE SAFETY ANALYSIS

The components of the Fuel Receiving and Storage facility associated with handling of radioactive material and processing of radioactive wastes are contained within the FRS and Radwaste Process Buildings. The Recirculation Ventilation System and the Radwaste Treatment System provide support for the FRS facility and are in buildings in the north FRS yard adjacent to the FRS building. Wastes generated in the FRS water treatment system are transferred via a 5 cm (2-inch) steel pipe to a High Integrity Container (HIC) in the Radwaste Process Building. It is expected that accidents associated with the FRS facility involving the release of radioactive contamination would occur within these buildings or in the connection between the FRS building and the Radwaste Process Building. Although these buildings were not designed as contamination containment structures, they do provide a level of contamination control. Radioactive contamination released in accidents occurring within these buildings would eventually be exhausted to the main plant stack or treated at the Low-Level Waste Treatment (LLWT) System. Section 9.2.4 evaluates the drop of a high integrity container (HIC) in the north FRS yard. This evaluation analyzes the bounding accident in the FRS facility occurring outside of FRS buildings. For the accidents analyzed in this report it was found that the dose to the maximally exposed off-site individual was found to be well below limits set forth in DOE Order 5400.5, "Radiation Protection of the Public and the Environment."

The FRS building is situated adjacent to the main plant at an elevation well above potential flood levels. Other natural phenomena (earthquakes, tornadoes,

high winds and snow loading) and site characteristics were determined not to affect the conclusions presented herein. The dose to the maximally exposed off-site individual in the complete absence of engineered barriers was also found to be well below DOE limits (DOE 5400.5).

B.2.1.3 EFFECT OF NEARBY INDUSTRIAL, TRANSPORTATION AND MILITARY FACILITIES

There are no nearby industrial, transportation, or military facilities that affect the safe interim storage of spent nuclear fuel assemblies in the FRS. Onsite activities were analyzed and were found not to affect the conclusions presented on the SAR. See Section A.2.1.3 of Volume I for further discussion.

B.2.2 RADIOLOGICAL IMPACT OF NORMAL OPERATIONS

Both on-site and off-site dose assessments were performed in order to determine the radiological impact of normal operations (Section B.8.0). The dose to on-site personnel is due to work in areas of elevated gamma radiation levels relative to background. The annual collective occupational dose was estimated to be 2 person-rem (Section B.8.4).

To calculate off-site doses, two pathways were considered: air discharges (radionuclides exhausted to the main plant stack from the FRS building during normal operations) and normal liquid releases (Section B.8.6). An atmospheric dispersion code (AIRDOS-PC) was used to estimate the airborne transport and uptake of radioactive particulates. The resulting concentrations were coupled with models to estimate dose to off-site individuals from routine operational activities. The dose to the maximally exposed off-site individual was estimated to be 8.8E-05 mrem per year for airborne releases and 2.8E-03 mrem for liquid releases (WVDP-065).

B.2.3 RADIOLOGICAL IMPACT FROM ABNORMAL OPERATIONS

Abnormal operations are events which could occur from malfunctions of systems or operator error. Abnormal events are only of consequence when they affect systems in the FRS facility which process, control, or confine radioactivity. Abnormal events considered in this analysis (Section B.9.1) are of little consequence and are not predicted to result in a release of radioactive or hazardous material.

B.2.4 ACCIDENTS

Doses to an individual result from exposure to, or ingestion of, radioactively contaminated material. The FRS facility contains sources of radioactively contaminated materials that have the potential for incurring doses to both on-site and off-site individuals. These sources include irradiated fuel elements, spent pool filter media and ion exchange resins and contaminated storage pool water. Six bounding accidents associated with operation of the FRS facility were analyzed (Section B.9.2).

The first two accidents considered the effects of the most destructive natural disasters: a tornado and an earthquake. A tornado with wind speeds of 90 m/s (200 mph) is not considered to be a credible initiating event due to the recurrence interval of approximately 2 million years. Analysis of an earthquake indicated that on-site and off-site exposures may occur during to the release of fission gas from the rupture of all of the spent fuel assemblies. Maximum doses of 2.1 rem on-site and 42 mrem off-site were calculated.

The third accident involves the drop of a fuel assembly in the Cask Unloading Pool resulting in a release of fuel material to the pool water due to cladding failure. This accident resulted in doses of 17 mrem on-site and 0.34 mrem off-site.

The fourth accident examines the drop and subsequent rupture of a High Integrity Container (HIC) in the north FRS yard. The total on-site dose was calculated to be 38 mrem and the off-site dose was calculated to be 0.5 mrem.

The fifth accident involves the spill of spent fuel pool filter media during transfer to the Radwaste Process Building. Radioactivity released from this accident was not expected to result in any off-site dose; however, a direct dose of 62 mrem for an individual involved with the spill and 170 mrem for a cleanup crew individual was calculated.

The final accident involves the drop of a shipping cask in the CUP. No radioactivity was believed to be released from this accident and therefore no exposures would be incurred.

B.2.5 CONCLUSIONS

This safety analysis report was prepared to meet the requirements of DOE Order 5480.5, "Safety of Nuclear Facilities"; DOE/ID Order 5480.5A "Safety of Nuclear Facilities"; DOE/ID Order 5481.1A, "Safety Analysis and Review System"; and WVNS Policy and Procedure WV-906, "Safety Review Program." Analysis indicates that the facility as designed can be operated safely with no doses* of consequence to off-site individuals. Conservative assumptions lead to an estimated normal worker collective dose of 2 person-rem/yr, in keeping with the ALARA philosophy. Calculated doses to off-site persons were determined for both normal and accident conditions. In both cases, the doses are well within the requirements of DOE Order 5400.5.

* As used in this section and throughout this SAR, dose is defined as:
- Dose equivalent for skin contamination events;
- Effective Dose Equivalent (EDE) for external radiation exposures; and
- 50-year Committed Effective Dose Equivalent (CEDE) for ingestion or inhalation of radionuclides.

B.2.6 REFERENCES

ID 5481.1A U.S. Department of Energy, Idaho Operations Office, Order DOE 5481.1A, "Safety Analysis and Review System," April, 1989.

ID 5480.5A U.S. Department of Energy, Idaho Operations Office, Order DOE 5480.5A, "Safety of Nuclear Facilities," August, 1990.

DOE 5400.5 U.S. Department of Energy, Order 5400.5, "Radiation Protection of the Public and the Environment," June, 1990.

DOE 5480.5 U.S. Department of Energy, Order 5480.5, "Safety of Nuclear Facilities," September, 1990.

AIRDOS-PC United States Environmental Protection Agency, "User's Guide for AIRDOS-PC, Version 3.0," EPA/520/6-89-035, December, 1989.

WVDP-065 Faillace, E. R., J. J. Prowse, and Y. Yuan, "Radiological parameters for assessment of West Valley Demonstration Project Activities," Rev. 2, October, 1990.

WVNS Policy and Procedure WV-906, "Safety Review Program," June, 1990.

TABLE D.2.4-1

SUMMARY OF FRS NON-NATURAL PHENOMENA ACCIDENT ANALYSES

Accident	On-site CEDE (cSv or rem)	Off-site CEDE (Csv or rem)	Detection Method	Possible Cause	Corrective Action
Drop of a fuel assembly in the CUP	N.A.	0.00034	Visual and area radiation detector	Hoist or grapple failure and fuel cladding breach	Filtration of CUP by water treatment system; recovery and storage of dropped assembly
Drop of HIC	0.029	0.00045	Visual	Crane failure	Contain spill and decontaminate area
Waste transfer line rupture	0.097	N.A.	Visual	Line rupture	Discontinue transfer; decontaminate area

B.3.0 SITE CHARACTERISTICS

Site characteristics potentially affecting the WVDP as a whole are detailed in Volume I, Section A.3. Site characteristics affecting the FRS are described below.

B.3.1 GEOGRAPHY AND DEMOGRAPHY OF WVDP ENVIRONS

See Section A.3.1 of Volume I for a general site description. There is no effect on FRS operations from geography and demography of WVDP environs.

B.3.2 NEARBY INDUSTRIAL, TRANSPORTATION, AND MILITARY FACILITIES

See Section A.3.2 of Volume I for a general site discussion. There is no effect on FRS operations as a result of these site characteristics.

B.3.3 METEOROLOGY

See Section A.3.3 of Volume I for a general discussion of meteorology around the WVDP.

B.3.4 SURFACE HYDROLOGY

See Section A.3.4 of Volume I for a general discussion of surface hydrology around the WVDP. Surface hydrology characteristics did not affect the FRS design.

B.3.5 SUBSURFACE HYDROLOGY

See Section A.3.5 of Volume I for a general discussion of subsurface hydrology around the WVDP.

B.3.6 GEOLOGY AND SEISMOLOGY

See Section A.3.6 of Volume I for a general site description.

B.3.7 ECOLOGICAL CHARACTERIZATION OF WESTERN NEW YORK NUCLEAR SERVICES CENTER

See Section A.3.7 of Volume I for a general discussion. There is no effect on the FRS operation as a result of site ecology.

B.4.0 PRINCIPAL DESIGN CRITERIA

The Fuel Receiving and Storage Area was designed and constructed to the applicable building codes in effect at the time (1965). The facility was not constructed to the requirements of DOE 6430.1A; however additions or modifications of the facility shall comply with Order 6430.1A and the associated editions of the references therein. The WVPO has concurred that existing facilities, such as the FRS, do not need to meet the requirements of DOE Order 6430.1A (Bixby, 1989). An evaluation of deviations from current DOE requirements in the FRS facility is given in Table B.4.0-1.

B.4.1 PURPOSE OF THE FUEL RECEIVING AND STORAGE FACILITY

B.4.1.1 FUEL CHARACTERISTICS

Physical and radiological characteristics of the fuel stored in the pool are provided in Tables B.4.1-1 through B.4.1-4.

B.4.1.2 FUEL RECEIVING AND STORAGE FACILITY BYPRODUCTS

Operation of the FRS facility will result in the generation of approximately 0.7 m^3 (25 ft^3) of spent filter media and 0.7 m^3 (25 ft^3) of spent ion exchange resin annually.

B.4.1.3 FACILITY FUNCTIONS

The Fuel Receiving and Storage Area was designed for the receipt, storage and handling of spent nuclear fuel assemblies. Facilities are provided for water treatment and building ventilation. The initial capacity of the fuel storage pool was 924 assemblies. Spent fuel 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 22 canisters. Decommissioning activities in 1987 resulted in the removal of 31 of the racks.

Currently 125 spent nuclear fuel assemblies are stored in the pool. The facility's primary function is the interim protective, custodial, safe storage of the 125 spent nuclear fuel assemblies. The FRS is not intended to process or utilize the nuclear material.

Solid waste generated during the operation of the FRS facility is stored in High Integrity Containers (HICs) and shielded 208 L (55 gal) drums. Liquid wastes are processed at the Low-Level Waste Treatment Facility. Temporary storage for full HICs is provided in shielded casks in the north FRS yard. Shielded 208 L drums are stored in the LAG Storage facility prior to permanent disposal. Utilities for the FRS buildings are provided through the site utility supply system described in Section B.5.4.2 of Volume II.

B.4.2 STRUCTURAL AND MECHANICAL SAFETY CRITERIA

The FRS building was constructed as part of the original NFS facility and was designed to meet all applicable building codes in effect at the time (1965). These codes and safety criteria are listed below:

Wind	UBC 25 psf Base
Earthquake	UBC Zone 3
Snow Load	40 psf

Material of construction of the building is ASTM A36 steel. Additional design specifications are listed in Table B.4.2-1.

B.4.3 SAFETY PROTECTION SYSTEMS

B.4.3.1 GENERAL

The FRS facility has been designed for safe operation. Criticality control and confinement of radioactivity are the primary safety concerns. Specific safety protection systems are described in the following subsections.

B.4.3.2 PROTECTION BY MULTIPLE CONFINEMENT BARRIERS AND SYSTEMS

The following is a breakdown of the FRS into its component parts and a discussion of the various barriers applicable to each.

- Spent Fuel - the primary barrier is the spent fuel cladding; the secondary barrier is the water in the pool; the tertiary barrier is the fuel pool structure (walls).

- Spent Fuel Pool - The primary barrier is the structure (walls and floor) of the pool.

- Cask Unloading Pool - The primary barrier is the structure (walls and floor) of the unloading pool.

- Water Treatment System - The primary barrier consists of the water treatment vessels and the interconnecting piping; the secondary barrier is the FRS building (walls, floor and roof structure).

- Ventilation System - The primary barrier for the ventilation system is the sheet metal ductwork and component enclosures. The secondary barrier is the FRS building (walls, floor and roof structure).

- Radwaste Treatment System - The primary barrier consists of the vessels and interconnecting piping and the High Integrity containers. The secondary barrier is the building structure, and the berm formed by the sump and the 18 cm (7 inches) high curb on the building slab.

- See Section 4.3 for design specification references

B.4.3.3 PROTECTION BY EQUIPMENT AND INSTRUMENTATION SELECTION

Procurement of new equipment and instrumentation for operation of the FRS facility has been done in compliance with WVNS's Quality Assurance Program which is described in Section A.12.0 of Volume I. The quality levels of the individual components for the FRS facility are given in Table B.12.3-1. Current requirements for new equipment and modification to existing equipment are given in Section B.12.0.

Existing equipment and instrumentation is subjected to inspection and testing commensurate with its intended use.

B.4.3.4 NUCLEAR CRITICALITY SAFETY

B.4.3.4.1 Methods of Criticality Protection

The potential for a nuclear criticality accident exists in the handling and storage of irradiated nuclear fuel; therefore, several features were incorporated into the FRS facility design in order to prevent a criticality during FRS operations in accordance with DOE 5480.5. 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 including Operational Safety Requirements (OSRs) and Standard Operating Procedures (SOPs) have also been implemented to maintain the criticality potential as low as practicable.

B.4.3.5 RADIOLOGICAL PROTECTION

B.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 DOE 5480.11.

The FRS is considered a radiation area with areas of removable contamination. Access to the FRS is provided to authorized individuals through key card portal control.

B.4.3.5.2 Shielding

The major sources of radioactivity in the FRS facility are the spent fuel assemblies, loaded pool filter and ion exchange media in the water treatment vessels and High Integrity Containers. Intense gamma radiation from the spent fuel assemblies is adequately shielded by the water in the fuel storage pool. A sufficient quantity of water (3.4 m [11 ft]) is maintained above the fuel assemblies to reduce surface exposure levels to below 1 mR/hr above background. Water in the pool will also provide the shielding necessary to attenuate the high radiation exposure rates associated with water treatment vessels. Shielding from spent filter media is provided by concrete and steel shield containers that reduce contact exposure levels to below 100 mR/hr.

B.4.3.5.3 Radiation Alarm Systems

Continuous air monitors have been provided in the FRS facility to detect airborne contamination. An area radiation detector that alarms at 20 mR/hr above background is present on the service bridge during fuel handling (DOE 5480.5).

B.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 building, however, does maintain the following fire fighting equipment as a minimum (DOE 5480.7):

- A fire hose located in the north aisle. Water to the station is provided through the main plant fire water supply system,

- ABC-type fire extinguishers located throughout the building. The FRS facility does not process substances with an explosive potential.

A formal fire hazards analysis has not been performed for the FRS. Current plans are to complete all site required fire hazards analyses by December 1992.

B.4.3.7 RADIOACTIVE WASTE HANDLING AND STORAGE

Radioactive waste is created as a byproduct of water treatment operations in the FRS facility. Wastes include spent filter media in the pool filter and loaded ion exchange resin. These wastes are handled remotely to reduce occupational doses (DOE 5480.5 and 5480.11). Wastes are slurried to a shielded storage container in an area adjacent to the FRS building for temporary storage. The new water treatment system will utilize the current shielded storage container (HIC) for storage of the spent ion exchange media; however, spent filter cartridges from this system will be stored in shielded 208 L drums in the LAG storage facility.

Spent fuel cartridge filters from the new water treatment system will be removed to a preshielded disposal drum. The shielding material is preformed concrete having a wall thickness of 13 cm (5 in) and a bottom thickness of 6 cm (2.5 in), and steel for the top of the disposal drum. The shield thicknesses have been

chosen such that a contact exposure rate in the exterior of the disposal drum will not exceed 100 mR/hr (Dooley, 1992).

B.4.3.8 INDUSTRIAL AND CHEMICAL SAFETY

The administrative controls concerning industrial and chemical safety are found in the WVNS Industrial Hygiene and Safety Manual (WVDP-011) and DOE 5480.10. Processes in the FRS facility do not require the use of hazardous chemicals, therefore accidents involving chemicals are not likely.

B.4.4 CLASSIFICATION OF STRUCTURES, COMPONENTS AND SYSTEMS

The dose to the maximally exposed off-site individual as a result of accidents does not exceed 500 mrem (see Section B.9.2), therefore none of the structures, systems or components require a safety class designation of A or B. Safety classification of structures, components and systems in the FRS facility are given in Table B.4.3-1.

The WVDP Safety Classification System, which complies with DOE Order 6430.1A, "General Design Criteria," consists of three safety classes listed in decreasing order of importance: Safety Class A, Safety Class B, and Safety Class C. Class N is not important to safety in comparison to Safety Classes A, B, and C. These four classes are defined as:

Safety Class A - Structures, systems, and components whose failure to function as designed could cause an off-site effective dose equivalent in excess of 25 cSv (25 rem).

Safety Class B - Structures, systems, and components whose failure to function could cause an off-site effective dose equivalent in excess of 0.5 cSv (0.5 rem).

Safety Class C - Structures, systems, and components whose failure to function could cause and on-site effective dose equivalent in excess of 3 cSv (3 rem).

Safety Class N - Structures, systems, and components not important to safety (specifically, radiological safety as defined above).

Any new equipment installed or modified will be classified per QM-3, Rev. 6.

B.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. Future decommissioning activities will be facilitated by the use of sumps in the water treatment area and Radwaste Process Building to collect decontamination solution. In addition, all process equipment and piping has been designed to be remotely flushed. Decommissioning activities are expected to be performed in accordance with DOE 6430.1A.

B.4.6 REFERENCES

DOE 5400.5 U.S. Department of Energy, Order 5400.5 Chg.1, "Radiation Protection of the Public and the Environment," June, 1990.

DOE 5480.5 U.S. Department of Energy, Order 5480.5, "Safety of Nuclear Facilities," September, 1990.

DOE 5480.7 U.S. Department of Energy, Order 5480.7, "Fire Protection," November, 1987.

DOE 5480.10 U.S. Department of Energy, Order 5480.10, "Contractor Industrial Hygiene Program," June, 1985.

DOE 5480.11 U.S. Department of Energy, Order 5480.11, "Radiation Protection for Occupational Workers," June, 1990.

DOE 6430.1A U.S. Department of Energy, Order 6430.1A, "DOE General Design Criteria," April, 1989.

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

Gates, 1992 "FRS-SAR, TRG Review," memo to J. J. Prowse, dated March 6, 1992, FB:92:0070.

WVDP-010 WVDP Radiological Controls Manual, Rev. 4, 1990.

WVDP-011 WVNS Industrial Hygiene and Safety Manual, Rev. 11, 1989.

Wolniewicz, 1992 "Calculation of Decay Heat Generated by the Irradiated Fuel Stored in the FRS Fuel Storage Pool," memo to S. R. Reeves dated February 27, 1992, FB:92:0055.

TABLE B.4.0-1

DOE Order 6430.1A Evaluation

6430.1A Section	REQUIREMENT	EVALUATION
0111-99.0.1	Many buildings are subject to future additional ceiling-roof equipment loadings. In planning and designing buildings for special facilities, consideration shall be given to providing for a future 10 to 20 psf additional structural loading.	The FRS was designed to a 40 psf snow loading with no considerations made for additional loading. Due to the current mission of the facility future loadings are not expected.
0200-99.0.6	Studies shall be performed to determine site features such as ground failure under dynamic loading, surface faulting, liquefaction, vibratory ground motion, and site amplification that could influence the design or operation of the facility.	The basin that forms the fuel storage pool was designed for hydrostatic lateral pressures and thermal gradients with little consideration for seismic loads (Gates, 1992). Section B.9.2.2 considers the impact to the facility due to a DBE earthquake.

TABLE B.4.0-1

DOE Order 6430.1A Evaluation
(Continued)

6430.1A Section	REQUIREMENT	EVALUATION
1320-3	Storage racks shall be designed as safety class items and shall [maintain] their integrity during and following a DBA.	Based on all available design documents, the fuel storage racks, as originally 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. Based on these assumptions and the fact that corrosion and other forms of deterioration could have taken place during the life of the facility, it must be concluded that the rack system has less than the required capacity to resist an earthquake of 0.2g, but may possess sufficient capacity to resist the DBE of 0.1g (Gates, 1992). Section B.9.2.2 analyzes the impact of storage rack collapse. This is the bounding accident resulting from a rack failure.
1320-4	The cooling water system for a water pool type IFMSF shall perform its required functions during normal and anticipated operating conditions, and shall be capable of limiting the maximum pool temperature to 110°F.	There is insufficient decay heat generated by the fuel stored in the pool to raise the temperature of the pool water to 110°F (Wolniewicz, 1992). The existing cooling system has been taken out-of-service.

TABLE B.4.0-1

DOE Order 6430.1A Evaluation
(Continued)

6430.1A Section	REQUIREMENT	EVALUATION
1320-4	A system shall be incorporated into the design that can detect leakage from stored IFM in the event of a cladding or canning failure that could allow the escape of fission products and other radioactive material greater than specified limits. The storage facility shall contain provisions that allow the temporary storage of a leaky assembly.	As stated in Section 4.1.3 the current mission of the FRS facility is the interim safe storage of the 125 spent nuclear fuel assemblies. The FRS will not receive future fuel shipments. Minimal handling activities associated with the storage of fuel in the pool therefore will not aggravate current conditions or induce failures in the stored assemblies. Small quantities of fission products and other radioactive material that leak from hairline fractures or pinholes in the stored fuel can be adequately removed by the water treatment system to maintain the concentrations of radioactive material in the pool below acceptable levels.
1320-5.1	In general, the primary confinement shall be the IFM cladding or canning. Secondary confinement shall be established by the facility buildings that enclose the dry storage area and/or the storage pool and auxiliary systems.	Confinement provided by the FRS building is not necessary. Section B.8.6.4 of the SAR states that routine airborne activity levels in the building are below detection limit values.

TABLE B.4.0-1

DOE Order 6430.1A Evaluation
(Continued)

6430.1A Section	REQUIREMENT	EVALUATION
1320-5.4	Generally, the IFMSF building for a water pool type facility need not be protected from tornado missiles or missiles from other external sources (e.g., explosions on nearby transportation routes), but shall be designed to prevent massive collapse of building structures or the dropping of heavy objects onto the stored IFM as a result of building structural failures	An evaluation of the seismic capacity of the FRS building found that the building had apparently been designed in conformance with the 1964 UBC lateral force provisions (Gates, 1992). In lieu of verification of conformance to these requirements, Section 9.2.2 of the SAR has evaluated the impact of structure failure. The analysis evaluates the impact of the loss of the barrier provided by the structure and rupture of the assemblies in the pool. These events were found insufficient to initiate radioactive releases offsite in excess of the limits set forth in DOE 5400.5.
1320-7	The facility design shall include features that will facilitate decontamination for future decommissioning. For water pool type facilities, the design professional shall consider providing the pool liner with a leakage collection system that will allow leakage detection and limit absorption of contaminated pool water by concrete structures.	The fuel storage pool has not been designed with special features to facilitate future decontamination, nor has it been provided with a leak detection system.

TABLE B.4.0-1

DOE Order 6430.1A Evaluation
(Continued)

6430.1A Section	REQUIREMENT	EVALUATION
1530-99.0	An assessment shall be made early in the design or modification to determine the facility structures, systems, and components that shall be protected against the effects of a DBF and explosion. A fire protection engineer or person knowledgeable in applying the principles of fire protection shall develop the fire protection system. To maximize the protection against fire, the system shall contain an appropriate integration of fire prevention, detection, and suppression features.	The FRS does not meet the general requirements of complying with the "improved risk" fire protection requirements as stated in DOE 5480.7. Fire and explosion protection equipment that has been provided in the FRS is given in Section B.4.3.6.
1550-99.0.1	Ventilation system balancing shall be specified to ensure that the building air pressure is always negative with respect to the outside atmosphere.	Pressure in the FRS will be at equilibrium with the outside atmosphere during those periods when the north or south roll up doors are open to allow access to the facility. There are no vestibules or other systems to control air flow when the roll-up doors are open. Section 8.6.2 of the SAR has shown however that negative pressure relative to the outside need not be maintained due to the low airborne radioactivity concentrations in the facility.

TABLE B.4.0-1

DOE Order 6430.1A Evaluation
(Continued)

6430.1A Section	REQUIREMENT	EVALUATION
1550-99.0.3	The sources and characteristics of radioactive material in off-gas systems shall be identified. The design of an off-gas system shall be commensurate with the characteristics of the radioactive material in the off-gas and the risk associated with its release as an effluent.	Off-gas from vessels in the current water treatment system are ventilated through the main plant ventilation system. While the sources and characteristics of radioactive material in this system has not be identified, it can be inferred from the waste in the ventilated tanks. These wastes have been characterized and a summary of the characterization determinations is given in Table B.8.2-1. These wastes do not contain volatile constituents nor do they possess undesirable chemical characteristics (i.e. extreme values of pH). These off-gases therefore can be adequately processed by the main plant ventilation exhaust system without affecting the integrity of the system. The new water treatment system will be located entirely underwater and therefore will not require a vessel off-gas system.
1660-99.0.1	For safety class items that require electric power to perform their safety functions, the design shall provide safety class emergency electric power systems (AC, DC, and their distribution systems). The design shall define the type, capacity, performance characteristics, and features of the safety class electric systems, including generator and batteries, required to meet safety class system needs.	CAMs in the FRS are not provided with a continuous, uninterruptible power supply but are connected to the site backup power system.

TABLE B.4.0-1

DOE Order 6430.1A Evaluation
(Concluded)

6430.1A Section	REQUIREMENT	EVALUATION
1320-4	...consider the criteria provided in 10 CFR 72 and the following guides for applicability to IFMSFs: R.G. 3.49 and R.G. 3.54.	10 CFR 72 requires that pool water level equipment must be provided to alarm in a continuously manned location if the water level in the storage pool falls below a predetermined level. The FRS does not have this alarm. The impact of complete loss of water in the pool has been bound by the accident evaluated in Section B.9.2.

TABLE B.4.1-1

CHARACTERISTICS OF FUEL STORED IN THE POOL

Reactor/Type # of Assemblies	MTU	Spent Nuclear Fuel Assembly Size (mm)	Average Burn-up MWD/MTU	Reactor Discharge Date	Date Received at West Valley
Ginna/PWR 40	15.06	198 × 198 × 4064 (7.8" × 7.8" × 160")	10,000- 21,000	3/71 5/72	2/73-6/73 "
Big Rock Point/BWR 85	10.50	165 × 165 × 2134 (6.5" × 6.5" × 84")	11,300	6/68 4/69 2/71 3/72 4/73 5/74	2/73-9/73 11/74 " " " "

TABLE B.4.1-2

BRP FUEL ASSEMBLY PARAMETERS

ASSEMBLY TYPE	ARRAY	PELLET OD (in)	CLAD THICK (in)	ROD OD (in)	PITCH (in)	MASS U (kg)	# OF ASSEMBLY
B	11x11	0.373/0.275	0.034	0.449/0.344	0.577	132	2
C	11x11	0.373/0.275	0.034	0.449/.0344	0.577	121	4
E	9x9	0.471	0.040	0.562	0.707	141	51
F	9x9	0.471	0.040	0.562	0.707	137	15
D	7x7	*0.620	0.040	0.700	0.921	133	4
D	8x8	*0.500	0.035	0.570	0.807	113	2
D(EG)	9x9	0.471	0.040	0.562	0.707	136	4
EP	9x9	#0.471	0.040	0.562	0.707	118 5.33 (Pu)	3

NOTES:
* Assumed Value with no diametrical gap.
# Assumed value based on other 9x9 assemblies.

TABLE B.4.1-3

REG FUEL ASSEMBLY PHYSICAL CHARACTERISTICS

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 (% 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
"Control" Rod: Outer Stainless Steel Tube: OD (in) - ID (in) -	 0.432 0.393
"Control" Rod: Inner Stainless Steel Tube: OD (in) - ID (in) -	 0.312 0.268

TABLE B.4.1-4

POISON ROD AND FILLER ROD CHARACTERISTICS

Filler Rod:	
Diameter (in) -	0.370 (min)
Length (in) -	141.75
Material (in) -	Aluminum Alloy
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)
B^{10} (atom %) -	19.6 (min)
Total Boron (wt %) -	73 (min)
Total B + C (wt %) -	98 (min)

TABLE B.4.2-1
COMPONENT AND SYSTEM DESIGN SPECIFICATIONS

System/Component	Drawing/Specification
FRS Building	Drawing DS15-A-5
Fuel Pool Filter and Demineralizer	Specification V-11
Fuel Pool Rack 1M-1	Specification M-3
Fuel Storage Canister	Drawing D-2529-2
Cask Unloading Crane 1V-1	Specification V-4
Fuel Canister Crane 1V-2 and Service Bridge 1V-3	Specification V-5
Fuel Storage Pool	Drawing 1A-Q-3 (Figs. B.5.2-1, -2, -3)
Cask Unloading Pool	Drawing 1A-Q-3 (Figs. B.5.2-1, -2, -3)
Ventilation System Blower (1K-1)	Figure B.7.4-1
Recirculation Ventilation System	Figures B.5.2-6, B.7.4-1
Radwaste Treatment System	U.S. NRC Reg. Guide 1.143 (Fig. B.5.2-4, -5)
Water Treatment System (in pit)	Specification V-11
New Water Treatment System (in pool)	DOE 6430.1A
CAMS/Radiation Monitors	DOE 5480.11
Cranes and Hoists	Specification V-4
Canister/Service Bridge	Specification V-5
Storage Racks	Specification M-3 (Fig. B.5.2-7

TABLE B.4.3-1

SAFETY CLASSIFICATION AND QUALITY LEVEL OF STRUCTURES, COMPONENTS AND SYSTEMS

System	Safety Class	Quality Level
Fuel Storage Pool	C	C
Cask Unloading Pool	C	C
Ventilation System	C	C
Radwaste Treatment System (Including High Integrity Containers)	C	C
Heat Exchanger	N	N
Water Treatment System	C	C
Cranes and Hoists	N	N*
Canister/Service Bridge	N	N*
Lift Rack	N	N*
Clearwell Bucket	N	N
Decon Pump and Stall	N	N
Storage Racks	N	N*
CAMS/Rad Monitors	C	C
Security System	N	N*
Utilities	N	N

* Quality Assurance/Control procedures and inspections apply to these systems.

B.5.0 FUEL RECEIVING AND STORAGE FACILITIES DESIGN

B.5.1 SUMMARY DESCRIPTION

B.5.1.1 LOCATION AND FACILITY LAYOUT

The Fuel Receiving and Storage Building is located on the east side of the main process building. The Radwaste Process Building houses the equipment for the Radwaste Treatment System, including the shield containers and support equipment for storing spent pool filter media and ion exchange resin. The Recirculation Ventilation Building houses equipment for FRS building ventilation including fans, filters, heaters, chiller and controls. A small building on the south side of the FRS facility serves as a change room and office area for operations personnel. The FRS facility is shown in Figure B.5.1-1.

B.5.1.2 PRINCIPLE FEATURES

The FRS facility contains equipment for the storage and shipment of spent nuclear fuel and for the maintenance of water quality in the fuel storage pool. The principal features include:

- Cask decontamination area
- Cask and fuel handling equipment
- Cask unloading pool
- Fuel storage pool
- Water treatment system
- Spent pool filter media and ion exchange resin storage equipment
- Recirculation ventilation system

The FRS building is serviced by three ventilation systems: the main plant exhaust system, a recirculation system, and an exhaust blower.

B.5.2 FUEL RECEIVING AND STORAGE BUILDING

The fuel storage pool was designed to fulfill the requirements of the Uniform Building Code for Zone III and subsequent analysis has shown that the pool will survive the Safe Shutdown Earthquake (SAI, 1981).

Facility design criteria are outlined in Section A.4.2 of Volume I.

B.5.2.2 BUILDING LAYOUT

B.5.2.2.1 FRS Building Plan

FRS building plans and equipment layout are shown in Figures B.5.2-1 through B.5.2-3. Figure B.5.2-1 gives the building details. Figures B.5.2-2 and B.5.2-3 give the arrangement of the equipment in the building. The Radwaste Process Building contains the equipment for dewatering and storing spent pool filter media and ion exchange resins. Building and equipment details are shown in Figures B.5.2-4 and B.5.2-5. The Recirculation Ventilation Building is also located in the north FRS yard. This building contains the equipment for providing the heating, ventilation and air conditioning (HVAC) requirements for the FRS facility. See Figure B.5.2-6 for HVAC details.

Fuel Receiving and Storage equipment is summarized in Table B.5.2-1.

B.5.2.2.2 Confinement Features

The FRS building was not designed for the containment of radioactive contamination but does provide a level of containment: ventilation systems ensure that levels of airborne contamination are minimized. No evaluation has been made to quantify the degree of confinement provided by the FRS building nor has credit been taken for this confinement in routine release calculations of accident analyses except where containment would be more conservative (e.g. Section B.9.2.3).

Building air is recirculated through a bank of high-efficiency particulate air (HEPA) filters to remove airborne particulates. All equipment for maintaining pool water quality including vessels and associated piping is located in a concrete pit (water treatment area). Spills resulting from line ruptures or vessel failures would be contained in the pit and be collected in a sump located in the northeast corner. New water treatment system vessels will be located entirely within the fuel storage pool. An air operated diaphragm pump used in ion exchanger resin handling will be located in the water treatment area. Therefore any spills associated with the new water treatment system will be contained in either the fuel storage pool or the water treatment area pit.

The Radwaste Process Building provides for the containment of radioactive contamination as well. The foundation perimeter is bermed and a sump located in the southwest corner of the building provides for spill collection. The berm is sufficiently high to provide containment for a volume of liquid equal to 150 percent of the volume of a 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 or to the on-line HIC. Ventilation in the Radwaste Process Building is provided by a portable HEPA-filter ventilation unit (PVU) which exhausts to the return duct of the recirculation ventilation system. The ventilation unit provides ventilation for the High Integrity Container as well and is normally in operation only during transfers of waste to the on-line HIC.

B.5.2.3 FRS FACILITY DESCRIPTION

The FRS facility has been designed for the handling and storage of spent nuclear fuel assemblies. Major areas and equipment include:

- Cask Decontamination Area
- Cask and Fuel Handling Equipment
- Cask Unloading Pool
- Fuel Storage Pool

- Water Treatment System
- Radwaste Treatment System

B.5.2.3.1 Function of FRS Areas and Equipment

Cask Decontamination Area

The cask decontamination area has been designed to facilitate the decontamination of shipping casks for CUP operations and for shipping preparation. An annular, elevator-type platform allows for radiation surveys and manual decontamination, if necessary.

Cask and Fuel Handling Equipment

Cask and fuel handling equipment was designed for the safe handling of fuel shipping casks and the safe transfer and movement of fuel assemblies.

Cask Unloading Pool

The Cask Unloading Pool ensures that operations personnel are adequately shielded during transfer of spent nuclear fuel assemblies from storage canisters to a shipping cask.

Fuel Storage Pool

The Fuel Storage Pool provides shielding from irradiated fuel and ensures that stored assemblies are maintained in a critically safe geometry.

Water Treatment System

The FRS Water Treatment System has been designed to provide a level of water quality that ensures visual clarity for underwater operations and that degradation of spent nuclear fuel assemblies is minimized. Floor drains and

sumps in the FRS facility drain to the site interceptors; therefore, pool water radioactivity contamination levels are maintained below interceptor discharge limits given in the LLWT standard operating procedures through mechanical filtration and ion exchange. Temperatures in the storage pool are currently below 20°C (68°F). Due to decreased heat generated by decay of fuel in the pool and a decreased fuel inventory, the cooler, designed for the removal of decay heat, is no longer needed and has been placed out of service.

The new water treatment system is expected to maintain pool water quality within the same operational limits as those currently in place.

Radwaste Treatment System

The Radwaste Treatment System provides containment and shielding of spent pool filter media and ion exchange resins that reduce the external contact exposure rate to ≤2 mR/hr.

B.5.2.3.2 Component Description

B.5.2.3.2.1 Cask Decontamination Area

The Cask Decontamination Area consists of an enclosed, curbed stall approximately 4.3 m by 4.3 m by 9 m high (14-feet × 14-feet × 29½-feet high) located at the east end of the FRS building (see Figure B.5.2-3). The stall is equipped with a sliding door and roof to permit the 90 MT (100-ton) crane to position a cask vertically inside. The shipping cask is positioned within an annular, elevator-type platform that allows the operator to inspect and decontaminate the cask prior to shipping. A 20 cm (8-inch) 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 ducted 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 area was added as part of the NFS modification program.

B.5.2.3.2.2 Cask and Fuel Handling Equipment

Cask and fuel handling equipment in the FRS facility includes a 90 MT (100-ton) crane with two 4.5 MT (5-ton) auxiliary hoists, cask lifting yokes, a clearwell bucket and clearwell bucket lifting yoke, a cask lid bail and grapple, assorted fuel grapples, the CUP service bridge, a 900 kg (1-ton) fuel transfer hoist, fuel storage canisters, a canister lift rack and a canister bridge equipped with a canister crane.

The 90 MT (100-ton) cask unloading crane is mounted on a high bridge running east-west at the east end of the FRS building and services the decontamination stall, cask unloading pool and water treatment area. The crane has a trolley-mounted 90 MT (100-ton) hoist with a north-south travel and two smaller trolley-mounted 4.5 MT (5-ton) hoists also moving in the north-south direction. The cask unloading crane is used to transfer the fuel shipping casks to the cask decontamination stall and the CUP. The crane, with its two auxiliary hoists, is also capable of serving the water treatment area and does extend partially over the fuel storage pool. Since the crane is necessary for installation of the water-tight gate between the CUP and the fuel storage pool, it is not possible to mechanically block it from moving over the eastern portion of the fuel storage pool. However, the fuel racks have been removed from the east end of the pool. As a result, it is impossible for an object to drop from the hoist on to fuel assemblies stored in the pool. The 90 MT (100-ton) crane and the 4.5 MT (5-ton) auxiliary hoists are capable of either manual or remote operation.

The clearwell bucket is used in the CUP and serves to isolate casks from CUP water. The bucket is cylindrical and measures 1.6 m (5 feet 4 inches) I.D. by 5.3 m (17 feet 6 inches) high. A specially designed lid seals the opening between the top of the bucket and the top of the cask to prevent contaminated CUP water from contacting the outside of the cask. The annular space between the cask and the bucket is filled with clean demineralized water and is kept pressurized when the device is sealed by means of a head pot located above the bucket. The clearwell bucket is positioned on a platform at the 4.9 m (16-foot)

level of the CUP for cask insertion or removal and is lowered to the 13.4 m (44-foot) level for fuel removal. In order to minimize contamination of the crane equipment, a special lifting strongback is used by the 90 MT (100-ton) crane to lift or lower the bucket. Special casks for the off-site shipment of fuel remaining in the pool have been designed and have received certificate of compliance from the NRC. The shipping cask designed for transport of BWR fuel is discussed in Section 5.1 of SAR-012 (WVNS-SAR-012) and the cask designed for transport of PWR fuel is discussed in Section 5.1 of SAR-015 (WVNS-SAR-015). These casks have been designed for direct placement in the CUP and will not require the use of the clearwell bucket.

A small 900 kg (1-ton) electrically driven fuel transfer hoist is mounted on the service bridge and is controlled from that point. The hoist has a boom that swivels with a hand tiller which can be locked in position. The fuel hoist, in combination with a grapple, lifts an individual spent nuclear fuel (SNF) assembly from a canister located in the CUP and lowers it into a cask for shipment.

The canister crane is mounted on a trolley that is on the canister bridge and is used to lift canisters and move them in a north-south direction in the racks or into the CUP. East-west movement is accomplished by moving the canister bridge. The canister crane has a 1.8 MT (2-ton) capacity. This hoist uses a telescoping shaft with the grappling mechanism designed to mate with the canister lifting lugs. Grappler design precludes lifting a fuel assembly from its canister and lifting tool design precludes lifting a canister more than 15 cm (6-inches) off the rack.

The canister lift rack is installed on the west wall of the CUP. The lift rack is positioned at the 8.8 m (29-foot) depth for receipt of assemblies transferred from the fuel storage pool and is lowered to the bottom of the 13.4 m (44-foot) level of the CUP for transfer of assemblies to a shipping cask. The lift rack design includes a mechanical stop which precludes raising the fuel assembly closer than 3.4 m (11-feet) from the water surface. The lift rack is capable of holding four fuel canisters with the same spacing as that employed in the fuel

storage rack design. The travel of the lifting tool has been limited (both electrically and mechanically) to ensure that an assembly cannot be lifted to a depth of less than 3.4 m (11 ft) which ensures the dose rate at the surface of the pool is less than 1 mrem/h.

B.5.2.3.2.3 Cask Unloading Pool

The CUP is located east of the fuel storage pool and provides an area for the removal of fuel assemblies from canisters and placement into shipping casks. The CUP is 7.9 m (26-feet) long by 7.3 m (24-feet) wide and is sectioned to depths of 8.8 m (29-feet) and 13.4 m (45-feet). The deeper section provides the necessary shielding during removal of the irradiated fuel assemblies up to 4.9 m (16-feet) in length from a canister sitting on the bottom of the CUP. The shelf area maintains a platform that supports the clearwell bucket for cask transfer operations.

The CUP is lined with stainless steel, 2 mm (14 gauge) on the walls and 5 mm (3/16-inch) thick on the floor. This liner provides physical protection to the concrete vault from abrasion and will facilitate decontamination of the area.

There is a removable watertight gate which serves to isolate the CUP from the fuel storage pool so that the CUP can be drained without draining the pool. A rack on the north wall provides for storage of the gate.

B.5.2.3.2.4 Fuel Storage Pool

The fuel storage pool is a concrete reinforced structure 23 m (75-feet) long by 12 m (40-feet) wide and 8.8 m (29-feet) deep. The concrete floor of the pool is 1 m (3-feet) thick, and the outside walls are 1 m (3-feet) thick at the base, tapering to 0.3 m (1 foot) at the top. A system of 35 mm (1-3/8 inch) thick reinforcing rods are embedded in the pool structure and connected by expansion joints. The pool was approved by the United States Atomic Energy Commission

(USAEC) for use as meeting the Safe Shutdown Earthquake guidelines and all other USAEC safety requirements in effect at the time of construction.

The geology underneath 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 silty till layer (Lavery till) which 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.

Borehole drillings (62 DMB-16, 62 DMB-20, 74 DMB-40) from the area around the fuel storage pool provide information about the soil's physical characteristics. Studies conducted by the USGS (Yager, 1987) characterized the movement of groundwater through the sand and gravel sequence, and silty till. These studies described the permeability characteristics of the sand and gravel based on in situ field measurements. Falling-head and rising-head well tests evaluated the permeability of the ground by measuring the rate at which water entered or exited wells near the processing facility. Using this data Yager constructed groundwater flow models which map the flow of groundwater beneath this area.

The fuel storage racks 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 restricting the number of canisters in each row to 22 canisters. 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-inches), a canister cannot be lifted and moved over the top of the array. There is a 1.2 m (4-foot) wide aisle between the south end of the racks and the pool wall to allow movement of the canisters. When the canisters are emplaced on the storage racks, there is 3.5 m (11.5-feet) of shielding 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 (see Figure B.5.2-7). This main support beam is fastened to the north wall of the

fuel pool and is also under-supported by six beams which are bolted to the FRS floor and spacers located between the support beams to 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).

Each fastener bolt is made and approved to withstand at least the design basis earthquake (0.12 g force). Lawrence Livermore Laboratory, in an independent evaluation, indicated that the integrity would be maintained up to 0.16 g fcrce (LLL, 1978). The racks have been analyzed and found able to withstand the Safe Shutdown Earthquake (SAI, 1981).

The fuel canisters were manufactured using aluminum 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 slide on the storage rack and prevent rotation 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 (20.25-inches) center-to-center spacing and at least 19 cm (7.5-inches) edge-to-edge between canisters in the same row. Between rows the spacing is 53 cm (21.0-inches) center-to-center and 19 cm (7.5-inches) 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 (20.25-inches) outside diameter, 32 cm (12.5-inches) inside diameter, and up to 4.9 m (16-feet) long, can hold fuel assemblies as large as 4.9 m (16-feet) in length and 22 cm × 22 cm

(8.75-inches × 8.75-inches) 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.

In addition to housing the fuel, the fuel storage pool will also contain the new water treatment system. This system will be located at the northeast corner of the pool adjacent to the current water treatment system pit. This arrangement is illustrated in Figure B.5.2-4 and Figure B.5.2-4a.

B.5.2.3.2.5 Water Treatment System

The fuel pool water treatment system was designed to maintain pool water purity and to moderate water temperature. This system consists of a filter precoat tank, heat exchanger, circulator pump, leaf filter, and ion exchanger. During operation of the fuel pool water treatment system, water from the surface of the pool is withdrawn via a weir to a 32 L/s (500 gpm) recirculating pump which pumps the water through a particulate leaf filter and returns it to various areas of the pool. A side stream may be sent to either the pool ion exchanger and/or the pool cooler. Removal of water through the weir increases system efficiency by providing a skimming action that removes small particulates from the surface of the pool water.

The 32 L/s (500 gpm) leaf filter, which utilizes replaceable filter aid (precoat) on fixed leaf screens, and the 6 L/s (100 gpm) nonregenerative ion exchanger are housed in a shield vault within the water treatment area. Spent filter aid and/or spent resin are slurried to a settling tank which is also located in the vault. Settling tank supernate is pumped back to the weir and the settled solids are air transferred to a High Integrity Container in the Radwaste Process Building. New filter aid is added via the precoat mix tank. An alternate procedure allows for direct addition of filter aid via the weir. The ion exchanger is charged with 0.34 m^3 (12 ft^3) of resin via the precoat mix tank.

The fuel pool recirculation pump is normally in continuous operation except when discharging and recharging of the filter aid.

The water treatment system also includes a pool cooler to maintain the temperature of the water in the pool below 27°C (80°F). The original system, with a capacity of 0.18 MW (600,000 BTU/hr), was replaced with a dual evaporative cooling system with a total cooling capacity of 3.5 MW (1.2E+07 BTU/hr) as part of the NFS modification program. Due to current decay heat generation by the fuel in the pool and a decreased fuel inventory, the fuel pool water temperature remains below 20°C (68°F) and therefore the coolers are no longer needed and have been placed out of service.

The new water treatment system is designed to maintain pool water purity through mechanical filtration and ion exchange. This system has two main components: a mechanical underwater filter unit and an ion exchange process vessel assembly (Figure B.5.2-4b).

The filter unit is constructed of 304 stainless steel and has four filter cartridge housings and a center pump housing welded onto a 61 cm x 76 cm (24 inches x 30 inches) base plate. Each of the four filter housings are connected at the bottom to the pump housing tube, which acts as the inlet suction plenum to the pump. These housing tubes are open at the top to take suction from the general area.

The filter unit employs 15 cm O.D. x 76 cm high (6 inches O.D. x 30 inches high) cylindrical filters constructed of pleated paper with stainless steel mesh containment. These filters are located within the filter cartridge housing with one filter per housing. Filters are secured in the housings through pump suction and the filters own weight; no mechanical devices are used. The total available filter area is 16.0 sq. meters (172 sq. ft). The pore size of the filters have been selected to optimize water filtration capability while maximizing filter run time. Filter elements are mated to the no-flow maximum pressure capacity of the pump so that the elements, even when totally plugged, will not rupture.

A tube-type motor and pump assembly provide the suction for filter operations. The motor operates on 480v, 20 amp, 3 phase electrical service and is secured in its housing in a manner similar to that of the filters. The pump is designed to provide 260 gpm continuous service.

The unit is approximately 200 cm (80 inches) high overall and weighs approximately 500 lbs., with the pump installed.

A geiger-mueller probe attached to the filter body gives a remote readout of real time filter dose rates which serves as a criteria for filter changeout.

The ion exchange unit is operated independently of the mechanical filtration unit. Material of construction is 304 stainless steel. The unit has a total capacity of 0.85 m^3 (30 ft^3) and is designed to be sluiced out and reloaded under water (Figure B.5.2-4c). Resin is maintained in the vessel through the use of media retention elements with a secondary barrier to resin migration provided by Johnson screens in the event of retention element failure. A pump and motor unit similar to that used for the filter provides the suction for ion exchange operations. The pump is sized for a continuous flow rate of 50 gpm. This vessel also maintains an attached G.M. probe for remote readout of real time dose rates (Figure B.5.2-4d)

B.5.2.3.2.6 Radwaste Treatment System

The Radwaste Treatment System contains two High Integrity Containers; an on-line unit and a standby unit. The on-line unit is connected to the FRS water treatment system by a 5 cm (2-inch) stainless steel pipe from the settling tank (35143) in the FRS water treatment area. Inside the Radwaste Process Building the pipe is converted to a 5 cm (2-inch) reinforced rubber line that terminates at the on-line HIC. The design pressure of this reinforced rubber transfer line is 90 psi.

Waste is transferred from the settling tank to a High Integrity Container which is constructed from high density cross-linked polyethylene and outfitted with an engineered dewatering system. Figure B.5.2-8 shows a simplified drawing of a HIC. Individual HICs are housed in two concentric shields. The inner shield is constructed of steel-reinforced concrete with a thickness of 36 cm (14-inches). The outer shield is constructed of carbon steel and has a thickness of 5 cm (2-inches).

Flow to the HIC is accomplished by a two-way air operated inlet valve. This valve automatically closes upon high-level indication from level alarms which are shown in Figure B.5.2-5. Any possible overflow from filling the HIC is caught in a 55 gallon drum connected to the HIC vent. A high-level alarm in this 55 gallon drum will also automatically close the air operated two-way inlet valve.

B.5.3 <u>FRS SUPPORT SYSTEMS</u>

B.5.3.1 VENTILATION

The FRS HVAC system (fully described in Section B.5.4.1) is designed to maintain air quality in the FRS building and to maintain the building under a slight negative pressure. The facility is serviced by three separate systems:

1. The <u>Recirculation Ventilation System</u> provides the HVAC requirements by recirculating air through heating and cooling coils to maintain a constant relative humidity and indoor air temperature, and by HEPA filtration of air to remove entrained particulates. This is a recirculation system only and does not exhaust air to the environment.

2. The <u>Main Plant Ventilation System</u> exhausts air from the cask decontamination stall and maintains the water treatment area vessels under negative pressure. This exhaust air is filtered by the main plant ventilation system HEPA filters and is exhausted through the main plant stack.

3. An Exhaust Blower (1K-1) provides increased 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.

B.5.3.2 MONITORING SYSTEMS

The FRS facility has a radiation monitoring system consisting of area radiation detectors, three CAMs and a radiation monitor to measure levels of airborne contamination and area exposure levels in the building. Monitors are equipped with an alarm which sounds locally to alert operators to hazardous or potentially hazardous conditions.

B.5.3.3 STANDBY POWER

Main plant auxiliary power is supplied to the FRS building to provide electricity for lighting and area radiation detectors in the event of a line power failure. Standby power supply is fully discussed in Section B.5.4.2 of Volume II. A backup battery in each unit ensures that area radiation detector operation is continuously supported until standby power is available. Standby power can be restored to the area radiation detectors within several minutes. The backup battery will power the alarms for up to 11 hours.

B.5.4 DESCRIPTION OF SERVICE AND UTILITY SYSTEMS

B.5.4.1 FRS BUILDING HEATING AND VENTILATION SYSTEM

The FRS building ventilation system was designed to minimize airborne contamination and to provide the building cooling and heating requirements. The FRS facility utilizes three independent ventilation systems to meet these requirements (see Figure B.7.4-1). The recirculation ventilation system routes air through cooling and heating coils, a HEPA filter bank and a reheat coil

before returning it to the FRS building. The FRS building exhaust blower withdraws an additional 1.9 m^3/s (4,000 cfm) to provide increased negative pressure within the building. Finally, a small amount of air is withdrawn from the water treatment area settling tank and the cask decontamination shroud to the main plant ventilation washer where it is subsequently exhausted to the main plant stack.

B.5.4.1.1 Major Components and Operating Characteristics

Recirculation Ventilation System

The components of the recirculation ventilation system include a cooler system, HEPA filters, identical recirculation fans, heating and reheating coil, makeup air roughing filter, ductwork and controls. The system is designed to maintain a temperature of 21°C (70°F) to 27°C (80°F) with a relative humidity of approximately 70 percent in the FRS building. During routine operation the system delivers 4 m^3/s (8,500 cfm) of filtered, conditioned air to the FRS building.

Recirculation Ventilation System Cooler

The ventilation cooler system is a closed loop recirculating system consisting of a recirculating pump, cooling coils, and a chiller. Make-up water is added automatically, controlled by a pressure regulator in the water supply line. An expansion tank absorbs expansions and contractions due to temperature changes.

The chiller in the cooling system is cooled by a freon cooling loop consisting of three compressors and three fan-cooled condensers located on the roof of the ventilation building. When the temperature exceeds 21°C (70°F) (wet bulb), the compressors start sequentially every 30 seconds and, when signaled by the controller, drop out of operation in a last on - first off sequence. A sequence switch provides capability to reverse the sequence, relieving the work load on the number one compressor.

The chiller provides water to a set of cooling coils which removes excess moisture in the return air. This prevents excessive differential pressure across the HEPA filters due to moisture loading and maintains a relative humidity of approximately 70 percent in the FRS building. Water removed in the process is collected in a drip pan and is routed back to the fuel pool weir.

Filters

Air from the FRS building (4 m^3/s (8,500 cfm)) is filtered through four banks of three HEPA 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. HEPA filters are DOP tested per DOE regulations to ensure mass removal efficiencies greater than 99.95% for particulates 0.3 microns and larger.

Recirculation Fans

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 on-line during normal operations with the other maintained in standby.

Heating and Reheating Coils

Air passing through the cooling coils is heated prior to entering the HEPA filters in order to raise the temperature of the air above the dew point. The heating coils are steam supplied and automatically controlled to maintain a constant plenum air temperature. A steam supplied reheat coil ensures that air returning to the FRS building is maintained within a constant temperature range of 21°C (70°F) to 27°C (80°F).

B.5.4.1.2 Safety Considerations

The recirculation ventilation system is not necessary for the safety of the facility as airborne contamination in the facility is below derived airborne contamination guidelines. However, to protect the performance of the system, the HEPA 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.

B.5.4.2 ELECTRICAL

Facility primary, secondary, and standby power are described in Section B.5.4.2 of Volume II of the SAR.

Power to equipment in the FRS building is distributed through a motor control center (MCC) in the south operating aisle. Equipment in the Radwaste Treatment System and the Recirculation Ventilation System is supplied through an MCC in the Recirculation Ventilation Building.

B.5.4.3 COMPRESSED AIR

Utility air and instrument air are required in the FRS facility to operate instruments and to support operations in the water treatment area and Radwaste Treatment System. The FRS facility is connected to the Process Building compressed air supply which is described in Section B.5.4.2 of Volume II of the SAR.

B.5.4.4 STEAM SUPPLY AND DISTRIBUTION

The FRS facility is connected to the Process Building steam supply which is discussed in Section B.5.4.2 of Volume II.

B.5.4.5 WATER SUPPLY

Demineralized water is used in the FRS facility for maintaining the level in the fuel storage pool and in pressurizing the clearwell bucket. The FRS building is connected to the Process Building water supply. See Section B.5.4.2 of Volume II.

B.5.4.6 COOLING WATER

Fuel elements in the pool do not generate sufficient decay heat to necessitate a cooling water system. The fuel storage pool water cooler has been placed out of service.

B.5.4.7 SEWAGE TREATMENT

There are no sanitary facilities in the FRS building. Operators use facilities in the Main Process Building.

B.5.4.8 SAFETY COMMUNICATIONS AND ALARMS

The Fuel Receiving and Storage facility has been equipped with detectors that alert operators in the facility to a elevated area radiation levels and to high airborne radioactivity levels in the building. The FRS facility has also been equipped with alarms to enunciate high water treatment area sump levels and ventilation pressure. These alarms sound at the east mechanical operating aisle alarm panel. An alarm in the main plant control room alerts operations personnel to an alarm on this panel.

During emergency conditions, internal communications are provided by telephones and radios, external communications is facilitated by the use of the plant telephone system. The plant all page system is used to supplement these.

B.5.4.9 FIRE PROTECTION SYSTEMS

The FRS facility fire protection system maintains a fire station in the north operating aisle which includes a 3.8 cm (1½-inch) hose connected to the site fire water supply loop. ABC-type fire extinguishers are also located throughout the building. Fire protection in the Radwaste Process Building and Recirculation Ventilation System Building is provided by fire extinguishers as well.

B.5.4.10 MAINTENANCE SYSTEMS

The FRS facility was designed as a contact maintenance facility. Due to high contact exposure rates, the pool filter, ion exchanger and settling tank have been located in a concrete-shielded vault in the water treatment area; however, associated valving has been located in accessible areas. As much as practical, valves and pumps are separated from high radiation fields by distance and/or shield walls. All equipment and piping is remotely drainable and flushable to reduce radiation levels prior to maintenance activities. New water treatment system equipment will be stored in the pool and will be remotely maintainable.

B.5.4.11 COLD CHEMICAL SYSTEMS

Not applicable to this SAR.

B.5.5 REFERENCES

SAI, 1981 Science Applications, Inc., NUREG/CR-2236, "Seismic Resistance Capacity Evaluation of Spent Fuel Storage Racks and Fuel at West Valley, New York," December, 1981.

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.

WVDP-022 WVNS Emergency Plan and Procedures Manual, Rev. 5, 1991.

WVNS-SAR-012 W. P. Duggan, "Safety Analysis for TN-BRP Cask Operations," Rev. 0, August 1989.

WVNS-SAR-015 S. A. MacVean, "Safety Analysis for TN-REG Cask Operations," Rev. 0, October, 1989.

Yager, R. M. 1985. Simulation of Groundwater Flow Near the Nuclear-Fuel Reprocessing Facility at the Western New York Nuclear Service Center, Cattaraugus County, New York. United States Geological Survey Water-Resources Investigations Report 85-4308.

TABLE B.5.2-1

SAFETY CLASSIFICATION AND SEISMIC QUALIFICATION

Equipment No.	Description	Function	Safety Class	Levels of Seismic Qualification	Risk in Earthquake	Comments
ID-3	Precoat Mixing Tank	Water Treatment	C	N/A	None	Not required to function after an earthquake
1E-1	Fuel Pool Cooler	Heat Exchanger	N	N/A	None	No longer in service.
1G-2	Fuel Pool Circulating Pump	Water Treatment	C	N/A	None	No longer in service
1G-5	Resin Transfer and Sump Pump	Water Treatment	C	N/A	None	No longer in service.
1M-1	Fuel Storage Rack	Fuel storage Racks for support of canisters and fuel assemblies under water in pool.	N	Reviewed for 0.2g PGA by analysis.	Yes	Not tested, collapse
1M-2	Fuel Pool Gate	Fuel Storage Pool Gates	C	N/A	None	Used to close off section of the fuel storage pool, not in service, not part of the pool barrier system.
1M-6	Cask Unloading Pool Liftrack	Lift Rack System	N	N/A	None	Not in service as part of the fuel storage process.
1M-9	Fuel Pool Gate Storage Rack	Fuel Storage Pool Gates	C	N/A	None	In use to store pool gate.

TABLE B.5.2-1

SAFETY CLASSIFICATION AND SEISMIC QUALIFICATION

(Continued)

Equipment No.	Description	Function	Safety Class	Levels of Seismic Qualification	Risk in Earthquake	Comments
1K-1	Exhaust Blower	Ventilation System	C	N/A	None	Exterior to FRS Building and is not required to function following earthquake.
1T-1	Fuel Pool Filter	Water Treatment System	C	N/A	None	Not required to function after an earthquake. Located in the dry well or pool adjacent to the fuel pool.
1V-1	Cask Unloading Crane	Cranes and Hoists	N	None	None	Large overhead crane on building rails does not extend over section of fuel pool where canisters are stored. Very low potential for crane derailment under a 0.1 to 0.2g event, based on past earthquake observations with similar construction.

TABLE B.5.2-1

SAFETY CLASSIFICATION AND SEISMIC QUALIFICATION

(Continued)

Equipment No.	Description	Function	Safety Class	Levels of Seismic Qualification	Risk in Earthquake	Comments
IV-2	Fuel Pool Canister Crane	Canister Service Bridge	N	None	Yes	Parked directly over the submerged fuel storage racks, canisters and fuel assemblies. Bridge could be dislodged from rail on top of pool walls, collapse into the pool on the canister racks should not cause total rack collapse. Some nuclear materials might be freed from fuel assemblies in the pool of covering water. This risk could be mitigated by parking the fuel pool canister crane at the east end of the pool away from all fuel storage racks.

TABLE B.5.2-1

SAFETY CLASSIFICATION AND SEISMIC QUALIFICATION

(Continued)

Equipment No.	Description	Function	Safety Class	Levels of Seismic Qualification	Risk in Earthquake	Comments
IV-3	Fuel Pool Service Bridge	Canister Service Bridge	N	None	None	Parked on rails on top of pool wall at east end. Could fall off rail, in extreme earthquake; however, electronic controls (stops) prevent it from traveling 75 feet to the west end of the fuel storage pool where the racks and fuel canisters are located.
IV-6	Fuel Assembly Storage Canister	Canister/rack storage protective shield around fuel assemblies during storage	N	None (?)	Yes	Canisters have not been directly analyzed for effects of impact if they were to fall from the storage racks in an earthquake. If the canisters were to break apart as a result of collapse of the racks, the estimated radiological consequences are still within acceptable limits for life safety.

TABLE B.5.2-1

SAFETY CLASSIFICATION AND SEISMIC QUALIFICATION

(Concluded)

Equipment No.	Description	Function	Safety Class	Levels of Seismic Qualification	Risk in Earthquake	Comments
IV-12	Fuel Pool Demineralizer	Water Treatment System	C	N/A	None	Not required to function after an earthquake.
IV-15	Motor Driven Winch (for lift rack)	Lift Rack	N	N/A	None	Not in service as part of the fuel storage process.
IV-24	Railroad Door (North)	Entry	C	N/A	None	
IV-25	Railroad Door (South)	Entry	C	N/A	None	
35141	Bucket Head Tank	Cask Decontamination while in CUP	N	N/A	None	Not in service.
35143	Spent Filter Media Settling Tank	Water Treatment System	C	N/A	None	
50061	Fuel Handling Hoist	Cranes and Hoists	N	N/A	None	
50062	Shroud (Clearwell) Bucket	Clearwell Bucket	N	N/A	None	
50065	Cask Decontamination Stall	Decon Pump and Stall	N	N/A	None	
Recirculation Ventilation System Equipment			C	N/A	None	

B.6.0 PROCESS SYSTEMS

B.6.1 PROCESS DESCRIPTION

B.6.1.1 NARRATIVE DESCRIPTION

The Fuel Receiving and Storage Facility maintains equipment for the handling, storage and shipment of spent nuclear fuel assemblies. Primary process systems in the FRS facility are the fuel handling system, the water treatment system and the Radwaste Treatment System.

The water treatment system maintains the pool water quality (e.g., clarity and radioactivity). Fission products and TRUs slowly leach into the pool water from the 7 failed SNF assemblies and the pool bottom as a result of former activities in the CUP and FSP. The Radwaste Treatment System containerizes the loaded resins from the water treatment system.

The HIC, the HIC shield and the Radwaste Process Building are ventilated through a Portable Ventilation Unit (PVU). The PVU is an enclosed stainless steel unit comprised of a prefilter, HEPA filter, variable damper, blower and motor. Instrumentation is provided to indicate the differential pressure across the filter unit (i.e. prefilter and HEPA filter). The unit blower capacity is 1000 cfm.

B.6.1.1.1 Fuel Handling System

Spent nuclear fuel assemblies are stored in canisters on racks in the fuel storage pool. A shipping cask is placed in the CUP prior to fuel handling for shipment. Precautions are taken to minimize cask contamination from pool water by using specially coated casks or through the use of a clearwell bucket. The specially coated casks are placed directly in the CUP at the 13.4 m (44-foot) level through the use of the 90 MT (100-ton) crane and lifting strongback.

Uncoated casks are placed in a clearwell bucket that is positioned on a platform at the 4.9 m (16-foot) level of the CUP. The clearwell bucket is then lowered to the 13.4 m (44-foot) level of the CUP using the 90 MT (100-ton) crane and lifting strongback. Removal of fuel for shipping is accomplished by first placing the storage pool bridge over the canister containing the fuel to be shipped. The canister crane is mounted on a trolley on this bridge. Because the canister crane is prevented from raising the canister more than 13 cm (6-inches), 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 rack and along the storage pool south wall to the CUP. The assembly is then placed in the canister lift rack which holds up to four canisters in the same geometry as in the fuel storage pool racks. The lift rack is lowered from the 8.8 m (29-foot) level to the 13.4 m (44-foot) level of the CUP and the CUP service bridge is then brought into position over a canister. The fuel is removed from the canister to a shipping cask using an electrically powered hoist mounted on the bridge. After transfer of fuel to the cask is complete, the cask is lifted out of the pool to the cask decontamination area using the 90 MT (100-ton) crane. If the cask had been stored in the clearwell bucket while in the CUP, the bucket is first raised to a platform at the 4.9 m (16-foot) level where the cask is then removed and placed in the decontamination stall. The cask is then surveyed, and decontaminated if necessary, prior to shipping.

B.6.1.1.2 FRS Pool Water Treatment System

The FRS pool water treatment system is required to maintain the quality of the water in the pool. The water treatment system is designed for continuous operation. The purpose of the water treatment system is to maintain water clarity and to minimize pool radioactivity. Water clarity must be maintained to allow visual confirmation of the fuel at a depth of 3.4 m (11 feet) for special nuclear material accountability. Pool radioactivity is minimized to maintain operational exposure ALARA. The system consists of a filter precoat tank, heat exchanger, circulator pump, leaf filter and ion exchanger. The water treatment system P&ID is shown in Figure B.6.1-1. During operation of the system, water

from the surface of the pool is withdrawn via a weir to a 32 L/s (500 gpm) recirculating pump which pumps the water through a leaf filter and returns it to various areas of the pool. A side stream may be sent to the pool ion exchanger. Spent pool filter media is removed through a process known as hydroscraping. High pressure water and movable blades are passed across the surface of the filter leaf effectively removing the media from the leaves. The waste media is drained to the water treatment area settling tank and new filter aid is added via the precoat mix tank. An alternative procedure allows direct addition of filter aid via the weir. The fuel pool recirculation pump (1G-2) is normally in continuous operation except when discharging and recharging of filter aid.

The ion exchanger utilizes nonregenerative resin which is replaced on a routine schedule or when required to maintain pool water quality. Spent resin is removed from the ion exchanger by draining to the settling tank. After the resin is removed the ion exchange vessel is flushed and the flush water is transferred to the settling tank. The ion exchanger is then charged with 0.34 m^3 (12-ft^3) of resin via the precoat mix tank.

The settling tank, located in the water treatment area vault, has a conical bottom for accepting settled waste media. Wastes transferred to the tank are held for a sufficient period of time to allow for settling of particulates. After settling is complete, the supernate is removed and a small amount of water is backwashed through the bottom of the cone to loosen the waste. The tank is then air pressurized and the waste is transferred to the on-line HIC in the Radwaste Process Building.

The new water treatment system consists of a filter/pump unit, an ion exchange/pump unit and waste handling equipment. During operation of the system, water will be withdrawn in separate streams from the weir and the pool cavity to the filter unit recirculating pump at a rate of 16.4 L/m (260 gpm) [16.6 L/m (310 gpm) total unit flow]. The ion exchanger has been designed to maintain chemistry and isotopically decontaminate pool water and operates at a processing rate of 3.2 L/m (50 gpm). All normal operating functions, including filter

changeout and resin sluicing and reloading, will be accomplished underwater in the pool.

The filter unit employs a cylindrical pleated-paper filter cartridge. Spent filter cartridges will be removed from the housing in the filter unit to a filter transfer can which will facilitate transfer of the filters from the filter unit to a preshielded storage container. The transfer can is a single-use container having four storage locations to accommodate the full complement of filters in the filter unit. Three new filters will be loaded into the can with the fourth position free to accept the first spent filter. The transfer can will then be lowered into a position adjacent to the filter unit. A spent filter will be placed in the vacant transfer can tube and will be replaced with a new filter. This procedure will continue until three new filters have been loaded into the filter unit. With the transfer can full, the operator will lower the fourth filter into the pool and into the remaining vacant position in the filter unit. An overhead crane will then be used to move the loaded transfer can from the fuel storage pool to the CUP where a filter sample will be taken prior to placement of the transfer can in the shielded storage container. This storage container will have a pitched bottom to facilitate dewatering of the filter cartridges. This water will be pumped to the CUP through the use of an air operated diaphragm pump. This process is illustrated in Figures B.6.1-3, -4, -5, and -6.

The ion exchange unit will withdraw water directly from the pool cavity, pumps it through the resin bed in a downflow cycle and return it to the pool. An organic resin, in combination with a selective ion exchanger, will be used in the ion exchange vessel to maintain pool water chemistry and to achieve isotopic decontamination.

Spent resin will be removed from the ion exchanger through the use of an air operated diaphragm pump which slurries the resin from the ion exchange vessel to the on-line HIC in the Radwaste Process Building. The ion exchanger will be charged with 30 ft^3 of resin by pumping the resin directly from its shipping container to the ion exchange vessel using the air operated diaphragm pump.

Geiger-Mueller counters on both the filter unit and ion exchanger will monitor the radiation levels of the filter and resin respectively. These radiation levels will then be correlated to an isotopic inventory that will be used as a criterion for media changeout.

B.6.1.1.3 Radwaste Treatment System

The Radwaste Treatment System contains two HICs: an on-line unit and a standby unit. The on-line unit is connected to the FRS water treatment system by a 5 cm (2-inch) stainless steel pipe from the settling tank (35143) in the FRS water treatment area. In the Radwaste Process Building the pipe is converted to a 5 cm (2-inch) reinforced rubber line that terminates at the on-line HIC. The design pressure of this reinforced rubber transfer line is 90 psi.

Waste is transferred from the settling tank to the on-line HIC by pressurizing the settling tank (35143) to 10-15 psi. Transfer is achieved pneumatically without the aid of a pumping system.

The High Integrity Containers are constructed fr… high density cross-linked polyethylene and outfitted with an engineered dewatering system. Individual HICs are housed in two concentric shields. The inner shield is constructed of steel-reinforced concrete with a thickness of 36 cm (14-inches). The outer shield is constructed of carbon steel and has a thickness of 5 cm (2-inches).

The HIC is fitted inside with a porous pipe to withdraw water … om the slurry. Figure B.5.2-8 shows a simplified drawing of the HIC. FRS facility wastes are dewatered in two stages. The bulk of the water is removed in the first stage with an air-operated pump suction applied from a pump skid. The design pressure of this 2.5 cm (1 inch) reinforced rubber suction line is 40 psi. The final dewatering (resulting in ≤1 v/o of free liquid) is accomplished by application of a vacuum through the vacuum hose. The dewatered liquid is routed to a floor drain in the FRS building where it flows by gravity to the interceptor and is subsequently treated at the LLWTF. Figure B.5.2-5 shows a piping and

instrumentation diagram (P&ID) of the system. Two HICs are shown in the building layout. One HIC is maintained on-line for processing and the other is a spare. Each holds approximately 3.4 m^3 (120 ft^3) of dewatered waste. Waste is transferred in stages to the on-line container. After the container is filled, the alternate HIC is manually put on-line. The full HIC is then dewatered for the final tir , and a full port closure is installed to seal it. The full HIC is removed from the Radwaste Process Building for temporary storage in a concrete shield container located in the FRS yard north of the Radwaste Process Building. The full HIC is then replaced with a new High Integrity Container.

The filled HIC is stored in the FRS yard in a heavy-walled (36 cm thick (14-inch)) concrete shield container located within the FRS controlled area. Weather-tight storage is provided inside the shield container which has an epoxy paint finish to prevent in-leakage of moisture. The lid has a tongue and groove seal with the body and the shield container contains a 19 mm (3/4-inch) polyethylene liner to further isolate HIC contents from the environment. The area around the shield container is routinely surveyed to ensure that site radiological controls are met according to the WVDP Radiological Controls Manual (WVDP-010).

A small sump pump is provided to handle any liquid spilled in the building. The sump contents will be sampled to determine radioactive material concentrations in the collected liquid. If gross beta concentrations are found to be greater than the interceptor discharge limits given in LLWT standard operating procedures, it will be routed back to the HIC. Any spill occurring during waste transfer will also be returned to the HIC.

B.6.1.2 FLOW SHEETS

The transport of fuel in the FRS facility is illustrated in Figure B.6.1-2.

B.6.1.3 IDENTIFICATION OF ITEMS FOR SAFETY ANALYSIS CONCERN

Operation of the FRS facility involves the handling of spent nuclear fuel assemblies and the transfer of solid radioactive wastes to a storage container. The primary items of safety analysis concern are:

- criticality prevention;
- maintaining occupational doses As Low As Reasonably Achievable (ALARA); and
- ensuring the confinement of radioactive material.

Measures for criticality prevention in the FRS facility are listed in the following section. The major mechanisms for ensuring the containment of contamination in the FRS building are the main plant ventilation system and the fuel storage pool structure. The main plan ventilation system maintains a slight negative pressure on process vessels relative to surrounding areas to ensure confinement of contamination from these sources. Radioactively contaminated water in the storage pool is confined by the fuel pool structure.

B.6.1.3.1 Criticality Prevention

Prevention of criticality in the fuel pool is provided by engineered and administrative controls. Canisters provide storage for fuel assemblies in a uniform configuration and canister inner dimensions prevent the inadvertent placement of a fuel assembly adjacent to another assembly already in the canister. These canisters are stored on racks that have been designed to provide sufficient spacing to ensure a critically safe geometry. Fuel canisters are provided with collars to ensure that sufficient spacing is maintained between adjacent canisters on the same rack. Slots in the collar engage with rails on the storage rack so that fuel storage is possible only in an approved, predetermined orientation. If the canister is not placed correctly on the rack

the canister grapple will not release the canister. The fuel assembly lift rack in the CUP maintains the same canister spacing as the storage racks in the fuel storage pool

Administrative controls including Operational Safety Requirements and Standard Operating Procedures supplement the engineered controls to ensure that criticality safety is maintained in the FRS facility. OSRs that ensure criticality safety in the FRS are identified in Section B.11.0 of this Addendum.

B.6.1.3.2 Chemical Safety

There are no chemical hazards introduced with the operation of this system.

B.6.1.3.3 Maintenance

The FRS facility has been designed as a contact maintenance facility. Occupational exposures are maintained ALARA through the use of shielding, remotely operated equipment, engineered barriers and approved operating procedures. Proper safety precautions for maintenance activities are specified in Radiation Work Permits (RWPs) and Industrial Work Permits (IWPs) as required by WVNS Policies and Procedures and the Radiological Controls Manual (WVDP-010).

B.6.2 MECHANICAL PROCESS SYSTEMS

B.6.2.1 SYSTEM DESCRIPTION

Spent nuclear fuel assemblies are stored in canisters on storage racks in the fuel storage pool. The stored fuel arrangement is shown in Figure B.6.2-1. The fuel storage pool 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 work platform and hand rails and has the canister crane mounted on it. The fuel pool service bridge, including the canister crane, are

illustrated in Figure B.6.2-2. Operation is from a control box on the bridge and there are two east-west speeds of 3 cm/s (6 ft/min) and 13 cm/s (25 ft/min) and an inching speed of 0.25 cm/s (6-inches/min).

The canister crane is mounted on the fuel storage pool bridge. It is used to lift canisters and move them in a north-south direction in the racks (last-in, first-out). East-west movement is accomplished by the fuel storage pool bridge. The canister crane has a 1.8 MT (2-ton) capacity and a hoist speed of 1 cm/s (2-feet/min). The vertical travel is limited to 15 cm (6-inches) in the pick-up position and 33 cm (13-inches) in the released position. Travel speed of the north-south trolley is 3 cm/s (6-feet/min) and 13 cm/s (25-feet/min) and an inching speed of 0.25 cm/s (6-inches/min). The canisters are lifted by means of a grapple on the end of the canister hoist boom which extends vertically downward from the bridge 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 crane. 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 engage the storage rack (see Figure B.6.2-3). The canister crane grapple which engages the lifting lugs will index about 30° before it is physically stopped by the crane housing. If the grapple engages the canister lifting lugs in any other than the correct angular position, it cannot rotate the canister enough to engage the support lugs with the storage rack. The grapple is raised 13 cm (6-inches) 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.

Canisters to be removed for shipment are moved from the canister storage racks in the pool to a lift rack in the cask unloading pool. The canister lift rack is mounted on the west wall of the CUP and can hold up to four fuel canisters in a straight row oriented north-south. The rack has a vertical travel of 4.9 m (16-feet) and is moved by means of an electric hoist. At the lower end of the

rack travel the bottom of the canisters are at the 13.4 m (44-foot) depth to facilitate fuel removal to a shipping cask with the fuel hoist. When the rack is raised to its upper level the tops of the canisters are at the same level as when they rest on the storage racks and are available to be picked up with the canister crane.

Fuel is removed from the storage canisters by means of an electrically driven hoist mounted on a service bridge. The CUP service bridge is similar to the fuel storage pool bridge and is positioned on the same rails. The bridge has a walkway with hand rails suitable for personnel to work from it, and is motorized with the controls on the bridge rail. The fuel hoist is mounted on the CUP service bridge and movement of fuel in the CUP is performed from the bridge. The CUP service bridge does not normally run out over the storage pool because of interference between the hoist boom and the lower roof line over the pool. The hoist boom can be removed if the CUP service bridge is needed over the storage pool.

The 900 kg (1-ton) electrically driven fuel hoist is mounted on the CUP service bridge and is controlled from that point. The fuel hoist, in combination with a grapple serves to lift spent nuclear fuel from canisters in the canister lift rack for placement in a shipping cask. Various long handled grappling tools are used to engage the fuel elements and lift them free of the canister. These grapples are usually specific for each fuel type and again also serve to keep the hoist parts in the CUP when not in use to eliminate the spread of contaminated pool water. They can be removed and decontaminated for repair or adjustment.

Prior to fuel transfer to the CUP, a shipping cask is placed in the clearwell bucket and staged at the 13.4 m (44-foot) level. The bucket is normally stored on a platform at the 4.9 m (16-foot) level and serves to isolate casks from the CUP water. The bucket is cylindrical and measures 1.6 m (5-feet 4-inches) I.D. by 5.3 m (17-feet 6-inches) high. A specially designed lid seals the opening between the top of the bucket and the top of the cask to prevent contaminated CUP water from contacting the outside of the cask. The lid is lifted by means of a

hand operated jack screw device and swings aside to permit insertion or removal of a cask. The annular space between the cask and bucket is filled with clean demineralized water and kept pressurized when the device is sealed by means of a 0.98 m^3 (260 gal) head pot located about 1.5 m (5-feet) above the bucket and connected through a rubber hose to the bucket.

The clearwell bucket is positioned on a platform on the 4.9 m (16-foot) level of the CUP for cask insertion or removal and is lowered to the 13.4 m (44-foot) level for fuel placement. Movement of the bucket and cask is done through a lifting yoke and the 90 MT (100-ton) crane. Use of the clearwell bucket makes a more rapid cask turn-around possible by reducing decontamination time for the cask exterior after it has been loaded. The clearwell bucket also minimizes the phenomena of contamination leaching out of the cask surface with time. Alternatively, a specially coated cask may be placed directly in the CUP at the 13.4 m (44-foot) level for fuel placement.

A special lifting strongback attaches to two trunnions on the clearwell bucket or specially coated shipping cask and is used by the 90 MT (100-ton) crane to lift or lower the bucket between the 4.9 m (16-foot) and 13.4 m (44-foot) levels of the CUP. This lifting strongback is stored on a rack on the CUP wall when not in use and saves having to immerse the crane hook, block, and cable into the contaminated CUP water so that when the cask is lifted out of the clearwell bucket there is no dripping of CUP water on the floor and the cask exterior.

Shipping casks are handled by means of the 90 MT (100-ton) cask crane. The cask crane cannot extend over the fuel storage pool which has a lower roof line. There is no way that this hoist can drop anything on the stored fuel assemblies. Redundant rigging is therefore not required for crane operations with this pool configuration. The total lift available is approximately 26 m (85-feet), of which 11.6 m (38-feet) is above the floor of the cask unloading area.

The crane is normally operated from a remote hand held control box but also has a pendant control. Each movement of the 90 MT (100-ton) crane has a 5-step

variable speed in either direction. The 4.5 MT (5-ton) hoists have two speed motions.

B.6.2.2 SAFETY FEATURES

Spent nuclear fuel assemblies are stored in canisters in the fuel storage pool. These canisters have been designed with spacing collars that provide a minimum of 19 cm (7.5-inches) of water between fuel contained in adjacent canisters which effectively prevent sufficient neutron interaction between assemblies and thereby prevent an inadvertent criticality. A minimum depth of shielding water is maintained above fuel stored in the fuel storage pool through engineered features including anti-syphon protection and administrative controls.

Safe conditions are maintained during handling operations through the use of engineered controls. There are seven limit switches restricting travel of the canister grapple hoist mechanism. Mechanical and electrical stops prevent the lift rack from being raised to a height that reduces shielding above the fuel elements to less than 3.4 m (11 ft) which ensure the exposure rate is less than 1 mR/h at the water surface. The grapple lower position restricts the upward movement of the grapple in either the pickup or release mode. The grapple upper/pickup position restricts the upward movement of the grapple and canister so that the fuel assembly will not be raised to a height that reduces the shielding provided by the pool water to an unacceptable level. The grapple upper/released position switch restricts upward movement of the grapple but allows it to raise high enough to clear fuel assemblies in the storage racks. The hoist arm has four directional limit switches, any of which will turn off the bridge drive motor if the hoist arm contacts an obstruction while traveling in any of the directions.

The canister lift rack has been designed to maintain assemblies in the same geometry as fuel stored on the fuel storage racks. A mechanical stop prevents the lift rack from being raised to a height that reduces shielding to an unacceptable level. 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

The CUP service bridge has also been provided with limit switches to prevent unsafe conditions. Elevation and overload limit switches ensure safe operation of the fuel transfer hoist. The fuel hoist is designed to allow lifting fuel assemblies only and is precluded by lifting tool design from allowing access to the fuel storage racks.

Fuel integrity was visually assessed. This required handling the fuel with the canister service bridge and hoist. Pool water activity levels are monitored weekly to ensure that radioactivity levels in the pool are maintained below interceptor discharge limits of 5E-3 μCi/ml gross beta.

B.6.3 SAFETY RELATED INSTRUMENTATION

The FRS building area radiation monitors are capable of detecting gamma ray exposure rates in the range $0.1\text{-}10^4$ mR/hr. CAMs employed in the FRS are alpha/beta monitoring systems designed for readout of 10 to 100,000 cpm. Calibration of these instruments is conducted on established schedules in accordance with Project Quality Assurance procedures.

An area radiation monitor set to alarm at 20 mR/hr above background is also placed on the service bridge during fuel handling operations.

B.6.4 REFERENCES

B.J. Connors, M.P. Golden, P.J. Valenti, J.J. Winkel, Topical Report DOE/NE/44139-37, "Spent Nuclear Fuel Removal Program at the West Valley Demonstration Project," March, 1987.

WVNS Policies and Procedures Manual.

WVDP-010 WVDP Radiological Controls Manual, Rev. 4, 1990.

B.7.0 WASTE CONFINEMENT AND MANAGEMENT

B.7.1 WASTE MANAGEMENT CRITERIA

Criteria for the management and reduction of radioactive wastes generated in the FRS facility are described in the WVDP Annual Waste Management Plan (WVDP-019) and the WVDP Waste Minimization Plan (WVDP-087). The guiding principles followed in the preparation of these plans are:

- Protect the worker, public health, and the environment;
- Minimize occupational doses;
- Minimize waste generation;
- Provide as much flexibility as possible in designed facilities to accommodate future uncertainties (e.g., liquid and solid waste volumes, storage, process equipment, etc.);
- Segregate uncontaminated from contaminated waste as early as possible to minimize additional storage, disposal, and transportation requirements;
- Conform to applicable Department of Energy (DOE) Orders and guidance from other regulations provided by the Department of Transportation (DOT), Environmental Protection Agency (EPA), Nuclear Regulatory Commission (NRC), and New York State;
- Minimize costs.

B.7.2 RADIOLOGICAL WASTES

Liquid and solid radioactive wastes are generated from activities and processes in the FRS facility. The fuel pool water treatment system generates approximately 3 m^3 (100 ft^3) of solid waste annually. These wastes include dewatered spent ion exchange resins and filter media and have been classified as transuranic (TRU) waste (McVay, 1987). Temporary storage is provided by High Integrity Containers placed in shielded concrete storage containers in the north FRS yard. Sources of low-level solid radioactive waste include spent HEPA filters from the recirculation ventilation system and discarded anticontamination clothing. These wastes are handled and disposed of according to existing procedures. Liquid wastes generated in the FRS facility, including floor drain effluents, sump wastes, excess pool water, water from cask decontamination activities and liquids generated during sludge dewatering are directed to the Low-Level Waste Treatment Facility (LLWTF) for processing.

B.7.3 NONRADIOLOGICAL WASTES

No significant quantities of nonradiological wastes are generated during normal FRS operations.

B.7.4 OFF-GAS TREATMENT AND VENTILATION

B.7.4.1 OPERATING CHARACTERISTICS

The FRS building is serviced by three independent ventilation systems during normal operations: The Recirculation Ventilation System, the Exhaust Blower and the Main Plant Ventilation System. Figure B.7.4-1 "FRS Building HVAC System" schematically depicts these systems.

The Recirculation Ventilation System provides temperature (70 - 80°F) and humidity (≤70% rh) control for the FRS building. Air flows across the Fuel Storage Pool and is collected in ducting along the north aisle. In addition, a

small amount of air (1000 cfm) is drawn into the system from the east end of the FRS Building. The recirculation air is then ducted to the Equipment Building where it is dehumidified by passing through cooling coils. It then passes through heating coils to raise the temperature of the stream above the dew point prior to entering the HEPA filters. Infiltration air (600 cfm) is drawn into the system from the Equipment Building to maintain the Building under a slight negative pressure. A bank of HEPA filters removes particulate matter. Outside make-up air (1400 cfm) is drawn through a roughing filter and into the system at a point between the HEPA filter bank and the fans. The total recirculation system air is then drawn through the fan(s) and is forced through a reheat coil (for final temperature control) and ducted to the FRS Building where it is discharged along the south operating aisle. The system is operated with only one fan when conditions are such that this will satisfy the FRS building temperature and humidity requirements. When single fan operation is insufficient, the second fan is brought on-line. Differential pressure across the filter bank is monitored and alarms are provided to alert operations personnel to abnormal differential pressure occurrences. Spent HEPA filters are removed and disposed of in accordance with existing procedures.

The Exhaust Blower 1K-1 vents 1.9 m^3/s (4000 cfm) of unfiltered air from the south aisle to the main process building stack; however, the air is continuously sampled and monitored by main plant stack monitors.

The Main Plant Ventilation System vents 0.5 m^3/s (1000 cfm) of air from the cask decontamination stall to the main plant ventilation system HEPA filter bank 15T-49 or 15T-49A, the main system fan, and then discharged through the main stack. This system also provides a slight negative pressure on water treatment system vessels.

All vessels associated with the new water treatment system will be located under water in the fuel storage pool and, therefore, will not be connected to an off-gas system.

B.7.4.2 SAFETY CRITERIA AND ASSURANCE

HEPA filters remove adequate radioactive particulate matter to ensure that effluent airborne radioactivity levels are maintained within safe levels and in accordance with DOE Order 5400.5. HEPA filters used in the recirculation ventilation system are tested annually to ensure mass removal efficiencies greater than 99.95% for particulates 0.3 μm and larger.

B.7.5 LIQUID WASTE TREATMENT AND RETENTION

Sources of liquid waste in the FRS facility are listed in Section B.7.2. All liquid wastes generated in the FRS building drain to the site interceptors for subsequent treatment at the LLWTF. Liquid wastes collected in the Radwaste Process Building sump are sampled and analyzed. Liquids with activity levels greater than the interceptor discharge limits given in LLWT standard operating procedures are returned to the on-line HIC; those with activities less than this level are pumped to a floor drain in the FRS building for treatment at the LLWTF. Water removed from air passing through the recirculation ventilation system is collected in a drip pan and returned to the fuel storage pool via the weir.

B.7.6 LIQUID WASTE SOLIDIFICATION

Liquid wastes generated in the FRS facility are not solidified.

B.7.7 SOLID WASTES

Solid wastes generated by the operation of the FRS facility are identified in Section B.7.2. Handling of such wastes is in accordance with Standard Operating Procedures and procedures contained within the WVDP Radiological Controls Manual (WVDP-010) and the WVNS Policies and Procedures Manual.

B.7.8 REFERENCES

DOE Order 5400.5 U.S. Department of Energy, Order 5400.5, Radiation Protection of the Public and the Environment," June, 1990.

WVDP-019 WVDP Annual Waste Management Plan, Rev. 9, 1991.

WVDP-010 WVDP Radiological Controls Manual, Rev. 4, 1990.

WVDP-087 WVDP Waste Minimization Plan, Rev. 0, 1991.

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

WVNS Policies and Procedures Manual

B.8.0 RADIATION PROTECTION

B.8.1 ASSURING THAT RADIATION EXPOSURES ARE AS LOW AS REASONABLY ACHIEVABLE (ALARA)

B.8.1.1 POLICY CONSIDERATIONS

WVNS radiological protection policies are outlined in the WVDP Radiological Controls Manual (WVDP-010).

B.8.1.2 DESIGN CONSIDERATIONS

The control of contamination and the minimization of radiation exposure are the primary elements of the ALARA philosophy. The incorporation of this philosophy in FRS facility design includes:

- Engineered controls, including mechanical stops on crane equipment which ensure an amount of water above fuel elements to provide sufficient shielding;

- Valving and instrumentation is located remotely as much as possible to reduce occupational doses;

- Liquid spills drain to sumps and floor drains where they can be collected for further processing;

- Special cask handling equipment to prevent the spread of contaminated pool water onto the hoist equipment and within the FRS building;

- Vessels containing highly radioactive materials are sufficiently shielded to reduce contact exposure rates to acceptable levels;

- Contamination of building air is minimized through filtering in the FRS building recirculation ventilation system and the main plant ventilation system;

- Airborne contamination from process vessels and the cask decontamination area is controlled through the main plant ventilation wash system;

These design features provide a high degree of reliability in confinement of contamination and radiation protection.

B.8.1.3 OPERATIONAL CONSIDERATIONS

The methods and procedures for FRS operation were developed to ensure that occupational doses are maintained ALARA. These methods allow remote operations and viewing.

Periodic visual assessment of the fuel is made to identify fuel which has been physically damaged or distorted to the point that it could affect the ability to load the fuel into shipping casks. Inspection has been made of known or suspected failed fuel using the NRC failed fuel definition (i.e. fuel assemblies or fuel rods with cladding defects greater than pinholes and hairline cracks). Pool water is used to provide shielding from the fuel assemblies during these operations.

The Cask Decontamination Area is an enclosed, curbed stall within the FRS building. Casks bearing levels of surface contamination greater than the levels allowable for off-site transport are decontaminated in this area.

Spent filter media and ion exchange resin are removed from their respective vessels to a settling tank where excess water is decanted prior to transfer to a High Integrity Container. Media transfer operations are performed using remotely located valving and instrumentation. Filter cartridges used in the new

water treatment system will be transferred to a shielded storage drum. Routine analysis of pool water samples ensures that water treatment system equipment is performing efficiently and that the concentration of radioactivity in the water remains below the discharge limit of the plant interceptors.

B.8.2 RADIATION SOURCES

B.8.2.1 CONTAINED SOURCES

The primary sources of radiation in the FRS facility are the irradiated fuel assemblies and spent filter media and ion exchange resin. Loaded filter media and ion exchange resins are found in both water treatment area vessels and in High Integrity Containers in the Radwaste Process Building and north FRS yard. Radioactive characteristics of HIC contents, including typical isotopic composition and curie content are listed in Table B.8.2-1. The fuel storage pool currently contains 125 spent nuclear fuel assemblies. This inventory consists of 85 spent BWR assemblies and 40 spent PWR assemblies. Radiological characteristics of PWR and BWR fuel are given in Tables B.8.2-2 through B.8.2-5.

B.8.2.2 AIRBORNE RADIOACTIVE MATERIAL SOURCES

Potential sources of contamination in the FRS facility are found in water treatment vessels and equipment, High Integrity Containers and in the fuel storage pool. Leaking of radioactive material from equipment in the water treatment area (primarily pumps and pump seals) has contributed to increased airborne radioactivity concentrations in the lower level of the pit; however, contamination levels in this area are controlled by routine spray-down of the walls. In addition, the elevated airborne concentrations are located in the lower level of the pit where air circulation is not sufficient to produce significant migration of contamination. Water treatment vessels and High Integrity Containers provide a high degree of confinement for the radioactive materials that they contain. Contamination in the fuel pool is essentially bound

in the water matrix. Therefore, sources of airborne contamination in routinely occupied areas of the FRS facility are minimal.

B.8.3 RADIATION PROTECTION DESIGN FEATURES

B.8.3.1 FACILITY DESIGN FEATURES

Several facility design features ensure that occupational exposures are maintained ALARA. These features include shielding, ventilation, and equipment for remote handling of radioactive materials.

Spent nuclear fuel handling is performed entirely in the FRS pool. Operation of the pool water treatment system is by use of remote valving and instrumentation to the greatest extent possible.

B.8.3.2 SHIELDING

Water treatment vessels containing highly radioactive materials are contained in a shielded vault. Vault construction is of steel-reinforced concrete with a thickness of 30 cm (1 foot).

Water treatment byproducts are stored in a High Integrity Container in the Radwaste Process Building. Shielding for a HIC is provided by two concentric shielding vessels. The inner shield container is of reinforced concrete and has a thickness of 36 cm (14-inches) and the outer shield container is of carbon steel with a thickness of 5 cm (2-inches). When the HIC becomes full it is removed from the Radwaste Process Building to a shielded storage container in the north FRS yard. These storage containers are constructed of steel-reinforced concrete with a thickness of 38 cm (15-inches).

Pool water is used for the shielding of stored fuel assemblies. Sufficient water above the fuel assemblies is maintained to ensure that exposure rates at the surface do not exceed 1 mR/hr above background. Spent cartridge filters from the

new water treatment system will be removed to a preshielded disposal drum. The shielding material is preformed concrete having a wall thickness of 13 cm (5 in) and a bottom thickness of 6 cm (2.5 in), and steel for the top of the disposal drum. The shield thicknesses have been chosen such that a contact exposure rate on the exterior of the disposal drum will not exceed 100 mR/h (Dooley, 1992).

B.8.3.3 VENTILATION

The ventilation in the FRS building was designed to minimize airborne contamination levels inside the building. The recirculation ventilation system contains HEPA filters for the removal of airborne particulates entrained in FRS building air. After filtration, this air is recirculated to the FRS building. HEPA filters used for radionuclide removal are DOP tested to establish that the installed filters provide a collection efficiency of at least 99.95 percent for particulates 0.3 microns in diameter and larger.

The recirculation ventilation system and the exhaust blower are not supplied by plant standby power; however, the main plant exhaust blower is operational during power outages and this system provides a slight negative pressure on water treatment vessels, thereby preventing the spread of airborne contamination.

System performance is routinely monitored utilizing four continuous air samplers and three continuous air monitors (CAMs), and through the use of filter bank differential pressure instrumentation that indicates excessive HEPA filter loading or filter failure.

B.8.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

To ensure compliance with TR/GP-19, "Area Radiation Detector Requirements During Fuel Handling and Storage", an area radiation detector system is operational at all times. Monitors in this system are operationally checked and calibrated routinely. The system is equipped with failure alarm circuitry to alert area operations personnel should AC power be lost and the unit be operating on DC

power only. This feature is provided with a delay to allow for power transients while avoiding false alarms. These radiation detectors are capable of detecting gamma ray exposure rates in the range 0.1-10^4 mR/hr.
In addition, three CAMs sample the building air for airborne alpha/beta contamination; an alarm sounds if elevated levels of contamination are detected. These alpha/beta monitoring systems are designed for readout of 10 to 100,000 cpm. Calibration of these instruments is conducted on established schedules in accordance with Project Quality Assurance procedures.

An area radiation monitor set to alarm at 20 mR/hr above background is also placed on the service bridge during fuel handling operations.

B.8.4 ESTIMATED ON-SITE COLLECTIVE DOSE

Occupational exposures in the FRS facility result primarily from operations in the water treatment area. The total calculated annual dose from operations in this facility is approximately 2 person-rem. Contributions from individual groups to this dose is given in Table B.8.4-1.

B.8.5 HEALTH PHYSICS PROGRAM FOR THE FRS FACILITY

The FRS facility is operated in compliance with the requirements of DOE 5480.11 as implemented by the WVDP Radiological Controls Manual (WVDP-010). These requirements are applicable to all Project activities. The health physics program for the Project is discussed in Section A.8.5 of Volume I.

B.8.6 ESTIMATED OFF-SITE DOSE

B.8.6.1 EFFLUENT MONITORING PROGRAMS

Effluents from FRS facilities are combined with effluents from other WVDP facilities and monitored according to established programs discussed in Section B.8.6.1 of Volume I.

B.8.6.2 ESTIMATED DOSE - AIR RELEASES

Air is exhausted from the FRS building via the exhaust blower (1K-1) located in the main plant vent exhaust cell (VEC). This air is mixed with plant air and exhausted to the main plant stack. It can be assumed that because air is exhausted from the FRS building unfiltered, airborne radioactivity concentrations present in the building are representative of those exhausted from the building. Four weekly building air samples are collected and analyzed for gross α and gross β activity. Levels of airborne activity in the FRS building are routinely below the detection limits of 1.0×10^{-14} μCi/mL for gross α and 4×10^{-13} μCi/mL for gross β (Gessner, 1989). Based on the assumptions that Sr-90 and Pu-239 exist at detection limit concentrations and are exhausted at the rate of 1.9 m^3/s (4,000 cfm), it is calculated that 24 μCi Sr-90 and 3.6 μCi Pu-239 are released annually resulting in an annual effective dose equivalent to the maximally exposed off-site individual of 8.8×10^{-5} mrem as determined by use of the AIRDOS-PC computer code.

B.8.6.3 ESTIMATED DOSE - LIQUID RELEASES

Liquid discharges from the FRS facility result from the dewatering of High Integrity Containers and the pumpdown of the water treatment area sump. These effluents are discharged to the site interceptors and subsequently treated at the LLWTF.

The dose to the maximally exposed off-site individual is estimated to be [illegible] $\times 10^{-3}$ mrem (WVDP-065). Assumptions used in the calculation of this dose [illegible] given in Table B.8.6-1.

B.8.7 REFERENCES

DOE 5480.11 U.S. Department of Energy, Order 5480.11, "Radiation Protection for Occupational Workers," June, 1990.

WVDP-010 WVDP Radiological Controls Manual, Rev. 4, 1990.

Gessner 1989 R. F. Gessner, "Fuel Receiving and Storage (FRS) Area Ventilation," memo to R. A. Humphrey, R&S Committee dated June 22, 1989, DA:89:0079.

AIRDOS-PC United States Environmental Protection Agency, "User's Guide for AIRDOS-PC, Version 3.0," EPA/520/6-89-035, December, 1989.

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

WVDP-065 Faillace, E. R., J. J. Prowse, and Y, Yuan, "Radiological parameters for assessment of West Valley Demonstration Project Activities," Rev. 2, October, 1990.

TABLE B.8.2-1
RADIOACTIVE CHARACTERISTICS OF HIC CONTENTS

Parameter	HIC "A" Concentration ($\mu Ci/g$)[a]	HIC "B" Concentration ($\mu Ci/g$)[b]
Gross α	6.65E-02	3.05E-01
Gross β	3.37E+01	8.10E+01
Sr-90	1.43E+00	6.80E-01
Cs-137	2.24E+01	6.54E+01
Co-60	N/A[c]	1.63E+00
Am-241	<2.95E-02	5.63E-02
Cs-134	N/A	1.41E+00
Total Pu	8.70E-02	9.69E-02
Total U[d]	N/A	3.63E-05

(a) Average of top, middle and bottom concentrations in HIC "A"

(b) Average of sample results for "B" HIC (McVay, 1987, Attachment 1)

(c) N/A = not available

(d) Units are (g Uranium / g sludge)

TABLE B.8.2-2

INVENTORY OF BWR FUEL IN THE FUEL STORAGE POOL

Assembly No.	Elemental Uranium (g)	Elemental Plutonium (g)	Burnup (MWD)
B-04	120,501	837.11	20,292
B-16	127,840	893.73	20,126
CC-10	113,497	830.27	22,683
CC-14	116,879	810.42	20,844
CC-25	109,494	790.81	22,233
CC-39	116,710	829.82	12,695
CE-01	133,613	681.96	12,271
CE-03	133,752	682.82	12,275
CE-10	133,913	621.16	10,627
CE-11	133,620	629.22	10,877
CE-16	134,034	622.31	10,643
CE-17	133,903	659.93	11,652
CE-22	134,087	654.72	11,491
CE-23	134,120	629	10,809
CE-24	133,745	691.82	12,515
CE-29	134,240	559.35	9,119
CE-31	134,016	616.17	10,483
CE-32	134,138	583.27	9,662
CE-33	133,807	610.21	10,351
CE-35	134,059	613.64	10,411
CE-36	134,060	597.15	9,979
CE-37	134,327	656.96	11,519
CE-41	132,923	601.2	10,218
CE-42	132,894	616.04	10,617
CE-50	135,429	588.23	9,225
CE-51	134,349	728.51	12,579
CE-52	134,232	722.27	12,438
CE-53	131,279	593.95	8,238
CE-54	134,288	724.88	12,496
CE-56	134,149	778.68	13,942
CE-57	134,534	720.63	12,356
CE-58	134,368	746.45	13,030
CE-59	135,130	681.1	13,144
CE-60	134,011	777.7	13,937
CE-61	133,569	855.66	16,390
CE-62	133,334	862.49	16,662
CE-63	133,905	761.07	13,500
CE-64	134,133	739.56	12,887
CE-66	134,791	693.3	11,635
CE-67	134,608	705.12	11,956
CE-69	134,076	567.85	8,928
CE-70	132,451	484.83	7,346
CE-71	135,028	695.8	11,667
CE-73	135,407	637.13	10,216
CE-74	131,360	777.52	14,360
CE-75	131,540	716.77	12,669
CE-76	136,505	365.99	4,993
CE-77	131,678	591.73	9,637
CE-79	130,675	830.18	15,901
CE-80	130,952	810.36	15,204
CE-81	134,781	783.86	14,067
CE-82	131,143	802.54	14,961
CE-83	135,079	736.06	12,669
CE-84	134,701	779.89	13,966
CE-85	135,140	725.16	12,388
CE-86	136,009	566.08	8,727
CE-87	135,525	624.71	9,954
CEP-1	14,236	4587.05	15,416
CEP-2	14,194	4114.11	15,712
CEP-3	14,167	4494.78	17,352
CF-01	135,134	456.51	6,654
CF-02	134,389	633.12	10,418
CF-03	130,289	753.96	13,677
CF-06	134,818	552.02	8,146
CF-12	135,276	437.18	6,289
CF-13	133,762	739.37	12,700
CF-14	130,224	762.49	13,815
CF-18	134,505	560.06	8,748
CF-19	135,104	450.61	6,547
CF-23	133,917	739.56	12,695
CF-24	135,384	437.53	6,289
CF-25	133,753	744.43	12,787
CF-26	134,501	583.45	8,947
CF-35	134,252	655.33	10,576
CF-42	134,295	680.44	11,135
D-50	56,720	316.36	6,377
D-51	59,946	116.56	1,874
D-52	70,359	114.44	1,449
D-53	78,736	121.89	1,448
D-54	70,084	120.27	1,534
D-55	72,679	122.66	1,533
D-60	127,048	505.66	8,588
D-61	127,872	662.35	12,333
D-62	129,469	673.53	12,418
D-63	122,816	617.91	11,837

TABLE B.8.2-3

INVENTORY OF PWR FUEL IN THE FUEL STORAGE POOL

Assembly No.	Elemental Uranium (g)	Elemental Plutonium (g)	Burnup (MWD)
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	97,75
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

TABLE B.8.2-4

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

Isotope	Initial Fuel Content (Ci)[b]	Decayed Fuel Content (Ci)[c]
Fe-55	1,271	18
Co-60	3,738	456
Ni-59	2	2
Ni-63	315	279
Kr-85	4,561	1,622
Sr-90	36,070	24,650
Y-90	37,620	24,660
Sb-125	3,243	60
Te-125m	165	4
Cs-134	33,980	157
Cs-137	39,670	27,410
Ba-137m	37,610	25,930
Pm-147	86,560	1,344
Sm-151	168	152
Eu-154	1,727	405
Eu-155	1,123	159
Pu-238	237	218
Pu-239	69	70
Pu-240	74	74
Pu-241	10,990	5,089
Am-241	5	199
Cm-244	11	6

(a) ORIGEN results for BWR Fuel - Power - 25.9 MW; 12,423 MWD burnup

(b) Isotopic content of fuel at reactor discharge

(c) Isotopic content of fuel decayed 16 years

TABLE B.8.2-5

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

Isotope	Initial Fuel Content (Ci)[b]	Decayed Fuel Content (Ci)[c]
H-3	533	229
Fe-55	2,000	37
Co-60	6,400	892
Ni-63	565	505
Kr-85	11,300	4,320
Sr-90	77,600	53,600
Y-90	80,700	53,600
Tc-99	14	14
Ru-106	545,000	18
Rh-106	740,000	18
Sb-125	8,707	188
I-129	0.04	0.04
Cs-134	246,000	1,540
Cs-137	108,000	76,200
Ba-137m	101,000	71,300
Pm-147	102,000	2,060
Sm-151	1,250	1,110
Eu-154	6,990	3,650
Eu-155	7,480	24
Np-239	18,500,000	18
Pu-238	2,720	2,600
Pu-239	318	323
Pu-240	477	479
Pu-241	105,000	51,500
Am-241	86	1,860
Am-242m	9	9
Am-242	63,400	9
Am-243	18	18
Cm-242	33,400	7
Cm-244	2,440	1,380

(a) ORIGEN results for PWR Fuel - 3.3 w/o U-235 enriched PWR fuel; 30 kw/kg specific power; 33,000 MWD burnup

(b) Isotopic content of fuel at reactor discharge

(c) Isotopic content of fuel decayed 15 years

TABLE B.8.4-1

ESTIMATED ANNUAL OCCUPATIONAL EXPOSURE FROM ACTIVITIES IN THE FUEL RECEIVING AND STORAGE FACILITY[a]

Department	Number in Group	Average Dose Rate (mrem/hr)	Man-hr per Group	Total Person-mrem
Engineering	4	1.3	45	60
Maintenance	5	5.3	15	80
Operations	8	2.4	748	1800
Radiation & Safety	1	3.3	12	40
			Total -	1980

(a) Exposure estimates are derived from the WVNS ALARA budget and are based on prior operating experience.

TABLE B.8.6-1

ANNUAL DOSE TO THE MAXIMALLY EXPOSED OFF-SITE INDIVIDUAL
DURING NORMAL OPERATIONS (LIQUID RELEASES)

Isotope	Activity Released Off-site (Ci)	Dose Conversion Factor (DCF) (mrem/Ci)	Dose to Maximally Exposed Off-site Individual (mrem)
Cs-137	1.5E-04	6.1	9.1E-04
Am-241	2.5E-04	7.57	1.9E-03
		Total	2.8E-03

Dose is calculated as the product of the dose conversion factor (from WVDP-065) and the released activity.

For this analysis it was assumed that the volume released from HIC dewatering activities is 24,000 L (6,300 gallons) per year and the volume of water from sump pumpout is 5,700 L (1,500 gallons) per year. This represents a total volume of 29,700 L (7,800 gallons) discharged from the facility having the gross α and gross β concentrations given below.

The pool water gross β activity is assumed to be entirely Cs-137 at a concentration equal to the interceptor discharge limit of 5.0E-03 μCi/mL. A decontamination factor of 1,000 for removal of cesium at the LLWTF has been used.

The pool water α concentration is inferred from HIC sludge gross α concentrations. It is assumed that the pool water filtration rate is 32 L/s (500 gpm) with a removal efficiency of 1 percent. The pool water α concentration is calculated to be 8.4E-06 μCi/mL and is assumed to consist entirely of Am-241.

B.9.0 ACCIDENT SAFETY ANALYSIS

Assumptions for calculating radiation doses to workers and the public are presented in WVDP-065, "Radiological Parameters for Assessment of West Valley Demonstration Project Activities." In brief, the maximally exposed individual during an accident is assumed to be at the point on the perimeter where the largest airborne concentration of radioactivity would occur. The radioactivity concentration is based on the maximum sector χ/Q. The external and internal dose conversion factors are taken from DOE/EH-0070, "External Dose Rate Conversion Factors for Calculation of Dose to the Public," and DOE/EH-0071, "Internal Dose Conversion for Calculation of Dose to the Public." These dose factors are consistent with DOE Order 5480.11.

Unit dose conversion factors are from a unit release (e.g., Sv/Bq [rem/Ci]) of each isotope, as discussed in WVDP-065, and are used to calculate the total committed effective dose equivalent (CEDE). The total CEDE from a release is determined by summing the component doses received from each isotope in the source term. The dose conversion factors are in accordance with DOE Order 5480.11.

B.9.1 ABNORMAL OPERATIONS

Abnormal events are events which could occur from the malfunctioning of systems or operator error. Abnormal events are only of potential consequence when they affect those systems in the FRS facility which process, control or confine radioactivity.

B.9.1.1 LOSS OF FUEL STORAGE POOL WATER

If the water were completely lost from the FRS pool, reactivity of the array of spent fuel assemblies would decrease due to loss of moderation. Because of the low decay heat generation rate, clad melting and fuel slumping is not credible.

A complete loss of coolant could only occur in one of two ways:

1. Evaporation
2. Failure of the pool wall.

Both mechanisms are protected against by design. Evaporation is not a credible means of loss of pool water because of the ready availability of makeup water, low evaporation rate, and the large volume of water in the FRS pool (greater than 3,000,000 liters (800,000 gallons)).

An investigation into the impact of an earthquake on the pool integrity was completed by the Lawrence Livermore Laboratory (LLL, 1978). Livermore concluded that structural distress could occur at accelerations above 16 percent of gravity in the pool region in the upper east corner of the north wall of the storage basin. It further concludes that a strong earthquake could cause some cracking in this general area (north wall of FSP). It is likely that some degree of cracking would take place with an earthquake characterized by 0.2 g peak horizontal ground acceleration. Simple water additions, such as those currently performed to replace evaporative losses, would maintain acceptable water levels. An analysis (Shearer, 1991), in which a hypothetical earthquake resulting in total loss of water from the FSP was used to calcualte the dose rate at the service bridge. The results show that the dose rate on the service bridge would be approximately 16.5 mC/kg-h (64 R/h). Therefore, water in the FSP is only necessary to maintain occupational exposure to ALARA.

B.9.1.2 LOSS OF SPACING AS A RESULT OF NATURAL PHENOMENA

During a postulated disruptive event, it is conceivable that four loaded canisters could be knocked off the canister liftrack and fall to the floor of the CUP. It is unlikely that this would produce a critical array because of the canister support rings which would continue to provide spacing between assemblies. Further, the probability of such natural phenomena is extremely low. However, a postulated accident scenario resulting in three loaded PWR canisters

dropped to the floor of the pool was modeled (WVNS, 1983, Attachment 8). The model used three canistered assemblies with a radial separation of 30 cm (1 foot) of water placed on a concrete floor and having full water moderation with concrete reflection on one side and water reflection on three sides. The fuel was not assumed to be damaged. The calculated k_{eff} is 0.8502 ± 0.00484, thus resulting in a $k_{eff} + 2\sigma$ value of 0.8599.

B.9.1.3 LOSS OF SPACING RESULTING FROM IMPROPER CRANE OPERATION

Mechanical or electrical controls are not provided to preclude lifting a fuel assembly from a canister and lowering it in position next to another loaded canister. This action would result in a fuel-to-fuel separation of about 15 cm (6-inches) which could be significant for PWR fuels.

The accidental placement of an uncanistered 15 × 15 base case unirradiated assembly in close radial contact with a canistered PWR assembly in the lift rack has been evaluated (WVNS, 1983, Attachment 8). The system was fully water moderated and reflected and included the influence of the aluminum canister shell. The resultant $k_{eff} + 2\sigma$ value for the system is 0.874.

Calculations performed by NFS in 1973 (NFS, 1973) determined the reactivity of canistered BWR assemblies in terms of their equivalence to PWR fuel reactivity. In that evaluation it was found that two BWR assemblies in the same canister are less reactive than a single PWR assembly. Therefore, placing an BWR assembly adjacent to a single BWR assembly in a canister would be subcritical by a larger margin.

B.9.1.4 LOSS OF SPACING RESULTING FROM IMPROPER HANDLING OF AN ASSEMBLY

It has been shown (WVNS, 1983, Attachment 5) that a single assembly is always subcritical in any geometry. During fuel transfers from the lift rack to a cask, an assembly could be placed on a loaded canister in the lift rack. A canistered PWR fuel assembly on top of a loaded canister represents a less reactive geometry than that described in Section B.9.1.3. It has been calculated (WVNS, 1983,

Attachment 8) that the reactivity for two PWR's in this configuration is $k_{eff} + 2\sigma = 0.8599$. This is effectively the same as a single assembly, indicating little interaction between the assemblies. The system will be less reactive for BWR assemblies.

B.9.1.5 NEW WATER TREATMENT SYSTEM SEAL FAILURE

The new water treatment system will be located entirely within the fuel storage pool and it is possible that failure of a seal or line could result in the release of ion exchange media into the fuel pool. Should this event occur, the effected equipment would be removed and repaired. This work would be performed using remote handling techniques.

B.9.1.6 CUP SEAL FAILURE

A stainless steel gate with inflatable rubber seal exists to isolate the CUP from the fuel storage pool in the event of a leak in the CUP. It is assumed that while the gate is in place and the CUP is empty the seal fails allowing the water to enter the CUP with a corresponding decrease in water height above the fuel in the fuel storage pool. Assuming normal water levels in the fuel storage pool and no water in the CUP, a decrease in water shielding over the fuel of 1.9 m would occur. A calculation using ISOSHLD version 1.6 has indicated that this would result in an increase in exposure rate directly over the fuel of less than 2.9 μC/kg-h (10 mR/hr). Given that the water level would decrease slowly as it leaked through the failed seal and that operators are not routinely positioned directly over the fuel, only minimal exposures would be incurred. Following the attainment of equilibrium water levels in the CUP and FSP, the pool water level could be restored, the gate removed, and the seal repaired.

B.9.2 ACCIDENT CONDITIONS

B.9.2.1 SEVERE TORNADO

Tornado occurrence is a very infrequent weather related event in the area of the WVDP. NRC in 1974 estimated the recurrence interval of the design-basis tornado as 10 million years (NRC, 1982). A more recent estimate (NRC, 1982) places the recurrence interval for that size tornado in the West Valley area at about 1 billion years or more. The recurrence interval for a tornado with maximum wind speeds of 90 m/s (200 mph) is estimated at 2 million years. The NRC in their safety evaluation report (NRC, 1982) considers a tornado strike with winds greater than 90 m/s (200 mph) to be incredible. However, an analysis was performed which indicated that the FRS Building would withstand the effect of such a tornado.

Because of the extremely low recurrence frequency and short duration of continued storage, the probability of a tornado strike is very remote for the duration of fuel storage at the site. An event of this magnitude is considered incredible based on its probability of occurrence.

B.9.2.2 EARTHQUAKE

Nuclear Fuel Services, Inc. originally recognized that seismic activity could be expected in the West Valley area and designed all structures according to the Uniform Building Code for Zone 3 (area of possible major damage). However, the NRC later concluded that the UBC was not an adequate basis for evaluating the structures and that a new design should be based upon a method which considered the dynamic effects which would result from peak horizontal acceleration of 0.2 gravity. The required pool modifications were not made because NFS terminated operations after this requirement was developed.

Subsequently, Science Applications, Inc. (SAI) was contracted to prepare a seismic-structural analysis of the existing racks in the FSP. In the analysis

SAI determined the threshold ground acceleration above which potential yielding, permanent deformation or collapse, and/or excessive deformation could occur to the storage rack (SAI, 1981). In 1987 decommissioning activities conducted by WVNS resulted in the removal of approximately 75 percent of the fuel storage racks in the pool. This represents a rack configuration different from the one in place at the time that SAI conducted its analysis. However, in their analyses SAI concluded that the single rack case would best represent the actual behavior of the system in a seismic event due to the high improbability of the canisters to provide a positive tie or connection between the racks. It was therefore determined that removal of racks in the storage pool would not have an effect on the results of the SAI analysis (Westinghouse, 1987).

The NRC has developed information on seismic recurrence intervals for the site which are equivalent to predicting the probability of the occurrence of a single earthquake as a function of its magnitude. The recurrence frequency for a 0.2 g event has been estimated by the NRC to be 2×10^{-4} per year (once in 5,000 years) (NRC, 1982). The SAI analysis indicated that the anchor bolts at the north wall could reach yield stress at about 0.19 g. The results also determined that the anchor bolt at the north wall connection might break at a threshold of about 0.22 g. A probable scenario of events following the north wall bolt failure is that the anchor bolts connecting the rack columns to the pool floor will fail causing the rack to deflect until it impacts the south wall. This event could cause a few canisters to become dislodged from the rack and fall to the pool floor causing crushing of the canisters. It is expected that in the event of canister crushing there will be damage to the fuel in the canister with a corresponding release of radioactive material to the storage pool. Source terms and exposures for this scenario are similar to those resulting from the drop of a single fuel assembly (Accident 3, Tables B.9.2-1 and -2, times 125 assemblies) in the pool. Because of the difficulty in justifying the seismic qualifications for the fuel pool and canister racks, a complete failure of all 125 assemblies was assumed resulting in an on-site dose of 2.1 rem and an off-site dose of 42 mrem.

The Lawrence Livermore Laboratory (LLL) determined that the occurrence of a 0.2 g event would also result in structural distress in the pool region in the upper east corner of the north wall of the storage pool. LLL determined that this distress could cause cracking that would result in a leak in the region and that leakage, if it occurred, would occur above the soil and into the building enclosing the FRS pool. It is likely that failure of the north east wall of the FSP would result in the flooding of the entire water treatment area; however, this would not result in a decrease of storage pool water level that would significantly increase area exposure rates.

B.9.2.3 DROP OF FUEL ASSEMBLY

The handling of spent nuclear fuel assemblies is necessary 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 rigging that prevents 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 and it must be assumed that the fuel cladding would be breached.

The accident was evaluated using the isotopic content of spent PWR fuel (ORIGEN, 1973) as it represents the fuel with the greatest fission product activity per assembly. The source term was calculated by decaying this fuel from November, 1975 (the date fuel was last received) to January 1992. The source term for this accident was calculated assuming a release of 1 percent of the assembly nonvolatile fission products and a release of 30 percent of the fission gas (ANSI, 1981) to the pool. The off-site source term was based on a 2 hour release of unfiltered building air. It is assumed that the air is fully saturated with contaminated water vapor and exhausted to the plant stack at a rate of 1.9 m^3/s (4,000 cfm). In addition, it is assumed that 100 percent of the liberated gas

is released to the environment. The accident source term is given in Table B.9.2-1.

The dose to an operator on the CUP service bridge would result from external exposure to radioactive material released to the cask unloading pool water. Technical Requirement GP-19 specifies that an area radiation detector set to alarm at 20 mR/hr above background be present on the bridge during fuel handling so that it is reasonable that an operator would leave the service bridge within one minute. External radiation exposure from nonvolatile solids released to the CUP is based on a uniformly distributed source in a slab of finite thickness and has been calculated using ISOSHLD. The cask unloading pool source term is given in Table B.9.2-1. The release of a small volume (approximately 1 L) of Kr-85 gas associated with the failure of the fuel cladding is also assumed. Exposure rates from Kr-85 are difficult to estimate due to the fact that the released gas will not become homogeneously distributed in the short period that the operator is in the area. Calculations were made for exposure from a plane source (the surface of the water in the CUP) and for immersion in a gas cloud. Both analyses assume that the activity is instantaneously released and is uniformly distributed within the respective geometry. Although krypton is a gas, exposure from a plane source is reasonable due to the short time period involved, the small quantity of gas released, the relatively high density of krypton compared to air and the small linear air velocity across the surface of the water in the CUP. An immersion dose was calculated assuming that the krypton expands into a hemisphere with diameter equal to the lateral dimension of the cask unloading pool (7.9 m) and centered over the CUP. The resulting concentration is multiplied by a dose conversion factor to obtain an immersion dose. A correction factor of 0.009 (USAEC, 1968) must be applied to this dose to account for immersion in a finite cloud (the dose conversion factor assumes immersion in a semi-infinite cloud). For the purpose of this calculation it is assumed that shielding is not provided by the service bridge and that the CUP water is not cleaned by the water treatment system. In addition, credit is not taken for operation of the recirculation ventilation system. An exposure rate of 7 mR/min on the CUP service bridge due to nonvolatile fission product release has been calculated.

The exposure rate for krypton as a plane source on the CUP surface would be 10 mR/min. Immersion in the gas cloud corresponds to a dose rate of 2 mrem/min. Total dose to the operator for this accident is calculated as the sum of the exposure from activity in the CUP and exposure to Kr-85. Assuming exposure to krypton is from a plane source on the CUP surface results in a total exposure of 17 mR for the minute that it takes the operator to leave the service bridge. The total dose is calculated to be 9 mrem when immersion in a krypton gas cloud is assumed.

The total dose to the maximally exposed off-site individual due to a dropped fuel assembly is estimated to be 3.4E-01 mrem. Off-site dose estimates are outlined in Table B.9.2-2.

The heat generation of the fuel assemblies currently stored in the pool has been calculated to be 8,800 watts (Wolniewicz, 1992). Due to the assembly storage configuration, convective 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.

B.9.2.4 DROP OF A HIGH INTEGRITY CONTAINER

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-feet). Prior to lifting the HIC free of its shield it is lifted approximately 5 to 8 cm (2-3 inches) 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.

The HIC and its associated lifting ring are designed to withstand an abrupt lift force of 3 g (i.e. 3x gravitational acceleration) 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-feet) 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 the HIC and it is assumed that the HIC would rupture upon impact, releasing its entire contents to the ground. The volume of sludge in a full HIC is 3.4 m^3 (120 ft^3) with 1 volume percent free liquid (water). For this analysis it has been assumed that source activity is homogeneously distributed in a cylinder with a radius of 2 m (6-feet) and that it takes 10 minutes for personnel to leave the spill area. Individuals located 6 m (20-feet) from the spill edge would be exposed to a radiation field of approximately 52 mrem/hr resulting in a dose of 9 mrem. Estimates for exposure were calculated using ISOSHLD. In addition to the direct radiation exposure, personnel in the area would receive an estimated inhalation dose of 29 mrem due to breathing contaminated air. Contributions of specific nuclides to the inhalation dose are given in Table B.9.2-3.

In addition to on-site exposures, there would be an off-site dose due to the inhalation of contaminated water vapor. It is calculated that the maximally exposed off-site individual would receive a committed effective dose equivalent of 0.5 mrem. Dose commitments are outlined in Table B.9.2-4.

This hazard is controlled by conducting the lift in compliance with the requirements given in WVDP-082, "WVDP Hoisting and Rigging Manual," for performance of high consequence lifts.

B.9.2.5 RUPTURE OF THE WASTE TRANSFER LINE

There are two possible scenarios for the rupture of the waste transfer line: (1) rupture of the waste transfer line inside the plant or Radwaste Process Building, or (2) rupture of the waste transfer line between the plant and the Radwaste Process Building. Should a break occur between the plant connection and the Radwaste Process Building isolation valve, the Radwaste Process Building operator must notify the plant operator to terminate the waste transfer. Depending on the work location in the plant and communications with the plant operator, it could take two to three minutes to secure the flow. This would also be the case if the

break occurred inside the Radwaste Process Building upstream of the isolation valve. If the break is downstream of the isolation valve, flow will be terminated much quicker. Waste spilled inside the buildings will be handled by the area sump.

It is expected that the source term would be greatest for the rupture of the transfer line inside the Radwaste Process Building upstream of the isolation valve. Operators are positioned inside their respective buildings (FRS building or Radwaste Process Building) during transfer operations so that a rupture of the line between the buildings would be less obvious and require a greater response time than if the break occurred inside. However, material released from a break in the line between the buildings would be contained in the transfer line pipe chase. Material spilled inside the FRS building would be contained entirely in the water treatment area and could easily be handled by the area sump. Waste spilled in the Radwaste Process Building would be within the operator work area. It is expected that the greatest exposure for an accident of this type would result from the rupture of the transfer line in the Radwaste Process Building. Exposure calculations were therefore performed only for the rupture of the transfer line inside the Radwaste Process Building upstream of the isolation valve.

For this analysis it is assumed that slurry material is transferred at a rate of 3.2 L/s (50 gpm) and that it takes 4 minutes to terminate the spill, resulting in a release of approximately 760 L (200 gallons) of slurry to the building. It is further assumed that the operator remains in the building for a period of 2 minutes, resulting in an inhalation dose of 0.1 mrem due to contamination released to the air through evaporation. A dose of 62 mrem for direct exposure to the spill has been calculated through the use of ISOSHLD. The assumptions and doses from specific isotopes are given Table B.9.2-5.

In addition to the doses received by the operators in the building at the time of the line rupture, there would be an exposure related to spill remediation. A 30 minute cleanup would result in a dose of 170 mrem for an individual cleanup crew member as shown in Table B.9.2-5. Shielding by equipment in the building

or the building exterior has not been assumed, resulting in a conservative dose estimate.

The Radwaste Process Building is an enclosed structure with ventilation provided by a HEPA-filtered portable ventilation unit. It is not expected that there would be an airborne release to the environs for this accident. In addition, the building perimeter is bermed so that liquids released in the spill would be entirely contained in the building. It is therefore expected that releases to the environment due to this accident would not be significantly different than those given in Section B.8.6 for normal operations.

B.9.2.6 SHIPPING CASK DROP

WVNS safety analyses for TN-BRP and TN-REG cask operations (WVNS-SAR-012 and WVNS-SAR-015 respectively) address the issue of a TN shipping cask drop into the cask unloading pool. An analysis of the drop of a cask contained in the clearwell bucket was not addressed in the above safety analyses because TN shipping casks have been designed for direct placement into the CUP. An incident involving the drop of a TN cask would represent an upper bound for this accident due to the large cask weight (approximately 100 MT (110-tons)).

B.9.3 REFERENCES

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TABLE B.9.2-1

SOURCE TERM DUE TO THE ACCIDENTAL DROP
OF A FUEL ASSEMBLY IN THE CUP

Isotope	Current PWR Inventory (Ci)[a]	CUP Source Term (Ci)	CUP Water Activity (Ci/L)[b]	To Environs (Ci)[c]
Co-60	7.65E+02	7.65E+00	1.20E-05	2.85E-06
Ni-63	5.01E+02	5.01E+00	7.87E-06	1.86E-06
Sr-90	5.21E+04	5.21E+02	8.19E-04	1.94E-04
Y-90	5.19E+04	5.19E+02	8.16E-04	1.93E-04
Sb-125	1.40E+02	1.40E+00	2.19E-06	5.19E-07
Cs-134	1.04E+03	1.04E+01	1.63E-05	3.86E-06
Cs-137	7.42E+04	7.42E+02	1.17E-03	2.76E-04
Ba-137m	6.94E+04	6.94E+02	1.09E-03	2.58E-04
Pm-147	1.52E+03	1.52E+01	2.39E-05	5.66E-06
Sm-151	1.10E+03	1.10E+01	1.73E-05	4.09E-06
Eu-154	3.47E+03	3.47E+01	5.46E-05	1.29E-05
Pu-238	2.59E+03	2.59E+01	4.07E-05	9.64E-06
Pu-239	3.23E+02	3.23E+00	5.08E-06	1.20E-06
Pu-240	4.79E+02	4.79E+00	7.53E-06	1.78E-06
Pu-241	4.87E+04	4.87E+02	7.66E-04	1.81E-04
Am-241	2.36E+03	2.36E+01	3.71E-05	8.79E-06
Cm-244	1.32E+03	1.32E+01	2.08E-05	4.91E-06
H-3	2.14E+02	6.43E+01	1.01E-04	2.39E-02
Kr-85	4.01E+03	1.20E+03	0.00E+00	1.20E+03

(a) ORIGEN results for PWR Fuel - 3.3 w/o U-235 enriched PWR fuel; 30 kw/kg specific power; 33,000 MWD burnup. Fuel decayed from November, 1975 to January, 1992. Inventory based on one metric ton uranium charged to the reactor (1 MTU). Average PWR assembly inventory is 0.45 MTU.

(b) Volume of water in the CUP is 636,000 L (168,000 gallons).

(c) Total activity released to the environs is based on a decontamination factor (DF) of 1000 for nonvolatile solids and a DF of 1.0 for H-3 and Kr-85. It is assumed that 240 L (63 gallons) of contaminated water vapor is released directly to the environment due to evaporation from the pool surface.

TABLE B.9.2-2

DOSE TO MAXIMALLY EXPOSED OFF-SITE INDIVIDUAL
DUE TO THE ACCIDENTAL DROP OF A FUEL ASSEMBLY IN THE CUP

C.E.D.E.[a] Due to Release of Nonvolatile Fission Products

Nuclide[b]	Source Term (Ci)	Elevated Unit Dose Factor (mrem/Ci)[c]	C.E.D.E (mrem)
Sr-90	1.94E-04	2.9E+01	5.6E-03
Cs-137	2.76E-04	7.2E-01	2.0E-04
Pu-238	9.64E-06	1.0E+04	9.6E-02
Pu-239	1.20E-06	1.1E+04	1.3E-02
Pu-240	1.78E-06	1.1E+04	2.0E-02
Pu-241	1.81E-04	2.2E+02	4.0E-02
Am-241	8.79E-06	1.2E+04	1.1E-01
Cm-244	4.91E-06	6.0E+03	2.9E-02
		Total C.E.D.E. =	3.1E-01 mrem

E.D.E.[a] Due to Immersion in Fission Gas Cloud

Nuclide[b]	Source Term (Ci)	Relative Dispersion[c] (sec/m^3)	Dose Conversion Factor[d] ($mrem\text{-}m^3/Ci\text{-}sec$)	E.D.E. (mrem)
Kr-85	1.20E+03	6.72E-05	3.55E-01	2.86E-02

Total Effective Dose to the Maximally Exposed Off-site Individual is Equal to the Sum of the C.E.D.E. + E.D.E = 3.4E-01 mrem.

(a) See Section B.2.5 for explanation of dose descriptions
(b) Radionuclides contributing more than 0.1% of dose
(c) Values from WVDP-065, 1990
(d) Values from DOE/EH-0070, 1988

TABLE B.9.2-3

INHALATION DOSE TO ON-SITE PERSONNEL
DUE TO THE DROP OF A HIC

Nuclide	Sludge Activity (Ci)	Water Activity Conc. (μCi/mL)[b]	Partition Coefficient	Activity Released to Atmosphere in 10 min (Ci)[c]	DCF for Inhalation (mrem/Ci)[d]	Inhalation Dose to Individual Spill Area (mrem)[e]
Co-60	5.22E+00	1.63E+00	1000	3.12E-06	1.50E+08	8.66E-02
Sr-90	3.37E+00	1.05E+00	1000	2.02E-06	1.31E+09	4.89E-01
Cs-134	4.51E+00	1.41E+00	1000	2.70E-06	4.70E+07	2.35E-02
Cs-137	1.41E+02	4.39E+01	1000	8.42E-05	3.20E+07	4.98E-01
Pu-238[a]	2.25E-01	7.02E-02	1000	1.35E-07	4.60E+11	1.14E+01
Pu-239[a]	2.80E-02	8.76E-03	1000	1.68E-08	5.10E+11	1.58E+00
Pu-240[a]	4.15E-02	1.30E-02	1000	2.49E-08	5.10E+11	2.34E+00
Pu-241[a]	4.22E+00	1.32E+00	1000	2.53E-06	1.00E+10	4.67E+00
Am-241	1.37E-01	4.29E-02	1000	8.23E-08	5.20E+11	7.90E+00
					Total	2.90E+01

(a) Pu concentrations based on average of Pu concentrations in HIC "A" and HIC "B". Characterization of HIC contents reports only Total Pu concentration which represents only α-emitting isotopes (Pu-241 not included). Plutonium ratios used in this table calculated using the reported Total Pu value and the ratios of Pu in stored fuel.

(b) Average of values for HIC "A" and HIC "B" from waste characterization report. Mass of sludge in HIC is 3,200 kg (7,000 lb) (from HIC characterization report (McVay, 1987)).

(c) Evaporation of 2 L (0.5 gallon) of water assumed. Water vapor activity concentration 0.001 of concentration in sludge.

(d) Reference DOE/EH-0071, 1988.

(e) Inhalation dose assumes individual breathes contaminated air for a period of 10 minutes. Concentration of contamination in the air calculated by dividing the total activity released to the atmosphere by the volume of air in a hemisphere of radius 8 m (24-feet). (It is assumed that the individual is confined within a radius of 8 m (24-feet) from the spill center.) The breathing rate of Reference Man (0.33 L/s) is also assumed (Radiological Health Handbook, 1970).

TABLE B.9.2-4

DOSE TO THE MAXIMALLY EXPOSED OFF-SITE INDIVIDUAL
DUE TO THE DROP OF A HIC

Nuclide	Source Term (Ci)[a]	Unit Dose Factor for Ground Release (mrem/Ci)[b]	C.E.D.E.[c] (mrem)
Co-60	3.75E-05	3.6E+01	1.3E-03
Sr-90	2.42E-05	3.1E+02	7.5E-03
Cs-134	3.24E-05	1.1E+01	3.6E-04
Cs-137	1.01E-03	7.6E+00	7.7E-03
Pu-238	1.61E-06	1.1E+05	1.8E-01
Pu-239	2.02E-07	1.2E+05	2.4E-02
Pu-240	2.99E-07	1.2E+05	3.6E-02
Pu-241	3.03E-05	2.4E+03	7.3E-02
Am-241	9.87E-07	1.2E+05	1.2E-01
		Total C.E.D.E. =	4.5E-01

(a) Source term assumes a partition coefficient of 1000 with evaporation of 23 L (6 gallons) of contaminated water from the sludge spill.

(b) Reference WVDP-065, 1990.

(c) See Section B.2.5 for explanation of dose

TABLE B.9.2-5

EXPOSURE TO ON-SITE PERSONNEL
FROM THE RUPTURE OF THE WASTE TRANSFER LINE

Dose to Personnel in Hittman Building

Nuclide	Water Conc. (μCi/mL)	Total Activity Spilled in 2 minutes (Ci)	Activity Released to Building Air in 2 minutes (Ci)	Activity Concentration in Building Air (Ci/L)	Dose Conversion Factor[a] (rem/μCi)	Inhalation Dose to Person in Building[b] (mrem)
Co-60	1.63E+00	6.17E-01	9.78E-09	4.89E-14	1.50E+08	2.90E-04
Sr-90	1.05E+00	3.97E-01	6.30E-09	3.15E-14	1.31E+09	1.63E-03
Cs-134	1.41E+00	5.34E-01	8.46E-09	4.23E-14	4.70E+07	7.87E-05
Cs-137	4.39E+01	1.66E+01	2.63E-07	1.32E-12	3.20E+07	1.67E-03
Am-241	4.29E-02	1.62E-02	2.57E-10	1.29E-15	5.20E+11	2.65E-02
Pu-238	7.02E-02	2.66E-02	4.21E-10	2.10E-15	4.60E+11	3.83E-02
Pu-239	8.76E-03	3.31E-03	5.25E-11	2.63E-16	5.10E+11	5.31E-03
Pu-240	1.30E-02	4.91E-03	7.79E-11	3.89E-16	5.10E+11	7.86E-03
Pu-241	1.32E+00	4.99E-01	7.91E-09	3.96E-14	1.00E+10	1.57E-02
					Total =	9.74E-02

Dose to Cleanup Personnel Due to Direct Radiation

Nuclide	Spill Conc (μCi/mL)	Total Spill Activity (Ci)	Specific γ Emission[c] (R-m^2/Ci-hr)	Exposure Rate to Cleanup Crew Member (mR/hr)	Total Exposure for 30 min Cleanup (mR)
Co-60	1.63E+00	1.23E+00	1.32E+00	4.1E+01	2.1E+01
Cs-134	1.41E+00	1.07E+00	8.70E-01	2.3E+01	1.2E+01
Ba-137m	4.39E+01	3.32E+01	3.30E-01	2.7E+02	1.4E+02
			Totals	3.4E+02	1.7E+02

(a) Dose conversion factors from WVDP-065

(b) Inhalation dose calculated using breathing rate of 0.33 L/s (Radiological Health Handbook, 1970)

(c) Specific γ Emission for nuclides from Radiological Health Handbook, 1970

B.10.0 CONDUCT OF OPERATIONS

B.10.1 ORGANIZATIONAL STRUCTURE

See section A.10.1 of Volume I provides the basic organization from which the FRS operational organization is derived. The Site Projects Manager is designated as the FRS facilities manager. The Site and Fuel Projects Manager provides engineering support. The Site and Fuel Projects Manager reports to the Site Projects Manager. The Site Projects Manager reports to the Site and Vitrification Engineering Manager. The Main Plant Operations Manager provides operations support. The Main Plant Operations Manager reports to the Plant Operations Manager. The Plant Operations Manager reports to the Site Operations Manager. The Site and Vitrification Engineering Manager and the Site Operations Manager reports to the President and General Manager of WVNS. All other support for operation of the facility is provided through the standard responsibility matrix for WVNS.

The scope of this SAR is limited to fuel storage. Receipt of new fuel would not be allowed under terms of the "COOPERATIVE AGREEMENT between the UNITED STATES DEPARTMENT of ENERGY and NEW YORK STATE ENERGY RESEARCH AND DEVELOPMENT AUTHORITY on the WESTERN NEW YORK NUCLEAR SERVICE CENTER at WEST VALLEY, NEW YORK", Paragraph 4.02. For operation of the storage facility, two qualified fuel handlers are normally retained on staff to ensure that one fuel handler is available. See Section B.10.3 for training and qualification levels.

B.10.2 PREOPERATIONAL TESTING AND OPERATION

The Fuel Receiving and Storage facility began operations in 1966 as part of the original Nuclear Fuel Services plant. Prior to plant startup, preoperational functional checkouts of all major equipment and systems were performed by both NFS and Bechtel.

B.10.3 TRAINING PROGRAMS

B.10.3.1 INTRODUCTION

B.10.3.1.1 Scope

The primary purpose of the FRS facility was to ensure the safe handling and storage of irradiated fuel assemblies received for processing at the former NFS facility. One hundred and twenty-five spent nuclear fuel assemblies are presently stored in the pool awaiting transfer to an interim storage facility.

Qualification programs have been developed that meet the requirements of applicable DOE Orders and operators are trained accordingly in the critically

B.10.3.1.2 Program Objective

The overall objective of the qualification program is to provide qualified personnel to safely operate the FRS facility in accordance with DOE Order 5480.5, "Safety of Nuclear Facilities".

B.10.3.1.3 Qualification Program Curriculum Overview

The course curriculum for the qualification of operators in the FRS facility is designed to include:

- Radiation Worker training per the Radiological Controls Manual (WVDP-010);
- System and subsystem features and operations description;
- System and subsystem components, including:
  - Instrumentation and Controls;
  - Alarms and Response Procedures;

- Emergency Operating Situations including criticality response;
- Operational Safety Requirements;
- Health physics, including radiological and environmental monitoring;
- Industrial health and safety;
- Ancillary systems, pumps, cranes, sumps, etc.

B.10.3.2 PERSONNEL SELECTION

B.10.3.2.1 Entry Requirements

The personnel selected for FRS Operations shall possess a high school education or equivalent with courses in algebra, chemistry and/or physics.

B.10.3.2.2 Prerequisites

Operator candidates shall have training equivalent to appropriate sections of Plant Systems Operator "B" qualification as a minimum, and shall be trained in the procedures and requirements of the equipment which they will operate prior to equipment operation. The extent of training necessary is determined by the requirements of DOE Order 5480.5. In addition, operators qualifying as fuel handlers shall have additional training relevant to fuel handling including criticality and administrative controls.

B.10.3.2.3 Training Resources

The WVNS Training Department has the responsibility for overall coordination and documentation of the qualification program. The department employs the expertise of Operations Supervisors and Cognizant Engineers to provide classroom training

on the basic theory, process concepts, subsystems, components and procedures that compose the FRS process. This training is supplemented by the use of vendor prepared materials related to basic function of valves, pumps, instruments, cranes and other vendor materials specific to the FRS facility. Further training reference material including material on administrative controls, procedures, and Operational Safety Requirements are provided. Training Department personnel also provide instruction, tutorial activities, operator qualification guidance and training material preparation including training aids and lecture videotaping. On-the-job training is required prior to operator qualification.

B.10.3.3 OPERATOR QUALIFICATION PROGRAM CONTENT

B.10.3.3.1 <u>Program Goals</u>

The FRS Operator Qualification Program ensures a safe and efficient operation of the facility.

The operator, at the completion of the training/qualification program, shall be able to:

- Demonstrate the knowledge and perform the normal modes of operation for the FRS facility using established Standard Operating Procedures in accordance with Operational Safety Requirements.

- Detect and respond to abnormal or emergency conditions using the instrumentation available and through visual monitoring of the components.

- Operate the facility safely in accordance with DOE Order 5480.5, "Safety of Nuclear Facilities".

- Maintain and update their knowledge and skills as required to remain qualified.

B.10.3.3.2 Training Content Outline

Operation of the FRS facility is conducted by qualified operations personnel. Training of operators for FRS facilities is incorporated into the Operator Qualification curriculum. The content and depth of coverage for the FRS Operations Qualification Program satisfies the specific needs for knowledge in FRS process areas. Personnel qualification includes sufficient on-the-job training to ensure a minimum level of competence in operation of equipment for which they are responsible. Operator qualification shall be determined by acceptable completion of written examinations, walk through exams and oral boards when appropriate.

The following is an outline of the curriculum for the FRS Operator Qualification Program.

I. ADMINISTRATIVE
- 1.0 Nuclear Materials Safeguards

II. RADIOLOGICAL AND NUCLEAR SAFETY
- 1.0 Radiation Worker Safety
- 2.0 Respiratory Protection
- 3.0 Nuclear Criticality Safety

III. INDUSTRIAL HEALTH AND SAFETY
- 1.0 Plant Industrial Safety
- 2.0 Hazardous Material Safety
- 3.0 Industrial Fire Safety
- 4.0 WVNS Lock and Tag Procedure

IV. GENERAL KNOWLEDGE

V. PLANT SAFETY
- 1.0 Emergency Preparedness

2.0	Plant Monitors
3.0	Operational Safety and Technical Requirements
VI.	FUEL RECEIVING AND STORAGE
1.0	FRS Ventilation System
2.0	Water Treatment System
3.0	90 MT (100-Ton) Cask Unloading Crane
4.0	Fuel Pool Canister Crane
5.0	Service Bridge
6.0	Removal and Emplacement of Pool Gate
7.0	FRS Accountability
8.0	FRS CUP Lift Rack
9.0	Cask Operations
VII.	EMERGENCY PREPAREDNESS
1.0	WVDP Emergency Plans and Procedures
2.0	Plant Emergency Shutdown Procedures
3.0	Emergency Spill Training

In addition to the training and qualifications required of FRS facility qualified operators, persons with the qualification of Fuel Handler are trained in the following:

- Operational Safety Requirement - OSR/GP-7 - "Criticality Safety for Liquid Transfer";

- Operational Safety Requirement - OSR/GP-11 - "Fuel Assembly Storage and Handling Requirements";

- Operational Safety Requirement - OSR/GP-12 - "Fissile Material Limits and Requirements for Waste Packages";

- Operational Safety Requirement - OSR/GP-18 - "Type B Shipping Cask License Compliance Documentation";

- Technical Requirement - TR/GP-19 - "Area Radiation Detector Requirements During Fuel Handling and Storage";

- Standard Operating Procedures Applicable to Fuel Shipout.

B.10.3.3.3 Documentation

The WVNS Training Department has the responsibility for maintaining current records on the status of trained personnel and a schedule for upgrading training and requalification of personnel. The following is the outline for documentation of WVNS operator qualification.

- The qualification program shall be documented in sufficient detail to permit independent evaluation of the scope of the training program. Procedures specific to training are found in WVNS Procedure WV-538 and the Training Policies and Procedures Manual.

- A written training signature list (e.g., qualification standard) is kept for each trainee, which covers all significant steps of training. The on-the-job-training qualification sign-off checklist is more detailed to assure proper instruction and practice (under close supervision) on every aspect of FRS operations.

- Verification of training is documented annually.

B.10.3.3.4 Requalification

DOE Order 5480.5 states that retraining and reexamination shall be required at least annually on all procedures for handling abnormal nuclear facility conditions and emergency situations relative to the employee's assigned

responsibilities and at least every 2 years on all other subjects in which the fissionable material handler, operator, or supervisor is expected to be proficient.

The requalification program is designed to review changes to procedures and equipment and subject matter not reinforced by direct use (e.g., fundamentals and operation of seldom used equipment and procedures.) The requalification training program maintains current knowledge and skills of the operator. The requalification program includes, but is not limited to:

- Equipment and Plant Modifications;
- Safety Analysis Reports (SARs) and OSRs;
- Changes in Operating Procedures;
- Unusual occurrences, accidents, or near misses which occur locally or elsewhere, if appropriate;
- Changing sources of radioactivity, criticality potentials or other potential environmental hazards;
- New outlooks or methods regarding the ALARA concept;
- Safety (fire, personnel injury, etc.).

B.10.4 NORMAL FUEL RECEIVING AND STORAGE FACILITY OPERATIONS

B.10.4.1 FUEL RECEIVING AND STORAGE FACILITY PROCEDURES

The FRS facility is operated using procedures prepared, reviewed, and approved per the requirements of the WVDP Policy and Procedures Manual. Procedures have been written to cover all aspects of FRS facility operation including but not limited to:

Cask Unloading Crane
Operation of the Fuel Pool Canister Crane
Operation of the Fuel Pool Service Bridge and Fuel Hoist
Emplacement and Removal of the Fuel Pool Gate

Fuel Pool Water System

FRS Accountability

Operation of the Cask Unloading Pool Lift Rack

Operation of the FRS Cask Decon Stall, the High Pressure Pump and the Diesel Engine

FRS Filter Media and Resin Disposal and Replacement

Operation of the FRS Ventilation System

Operation of the FRS Dewatering System

Visual Assessment of BRP Spent Fuel

Visual Assessment of REG Spent Fuel

TN-BRP and TN-REG Cask Painting Procedure

TN-BRP and TN-REG Cask Lifting Procedure

Replacement of High Integrity Containers

Inventorying Nuclear Materials in Nuclear Material Control Areas

B.10.4.2 PLANT RECORDS

See Section A.10.4.2 of Volume 1.

B.10.5 EMERGENCY PLANNING

See Section A.10.5 of Volume 1.

B.10.6 DECOMMISSIONING

See Section A.10.6 of Volume 1.

B.10.7 REFERENCES

DOE 5480.5 U.S. Department of Energy, Order 5480.5, "Safety of Nuclear Facilities," September, 1986.

WVDP-010 WVDP Radiological Controls Manual, Rev. 4, 1990.

Operational Safety Requirement OSR/GP-11, "Storage Canister Spacing and Loading".

WVNS Training Policy and Procedures Manual

WVNS Policy and Procedure WV-538, "Personnel Indoctrination and Training," Rev. 5, 1990.

WVDP Policy and Procedure Manual.

B.11.0 OPERATIONAL SAFETY REQUIREMENTS

The OSR section of this volume has been deleted in its entirety and incorporated in to Volume VI of the SAR.

The following OSRs apply to FRS operations:

OSR/GP-7	"Criticality Safety for Liquid Transfer"
OSR/GP-11	"Fuel Assembly Storage and Handling Requirements"
OSR/GP-18	"Type B Shipping Cask License Compliance Documentation"

In addition, the following Technical Requirements (TRs) apply to or affect FRS Operations

TR/GP-19	"Area Radiation Detector Requirements During Fuel Handling and Storage"

B.12.0 QUALITY ASSURANCE

The Quality Assurance (QA) Program for the Fuel Receiving and Storage Area was prepared and is implemented to meet the requirements of the U.S. Department of Energy Orders DOE-4700.1 and 5700.6B, Idaho Operations Office Orders 4700.1 and 5700.6C, West Valley Nuclear Services Co., Inc. Quality Management Program and Quality Assurance Program Requirements, and the American Society of Mechanical Engineers Quality Assurance Program Requirements for Nuclear Facilities (NQA-1). As a preexisting facility, t[illegible] not subject to the requirements of DOE Order 6430.1A. However, additions or modifications of the facility shall comply with Order 6430.1A and the associated editions of the references therein.

B.12.1 ORGANIZATION

Section A.1.4 of Volume I identifies the agents and contractors responsible for implementing the WVDP Act. The relationships between WVNS, agents and contractors is illustrated in Figure A.1.4-1.

Section A.12.1 of Volume I describes the organizational structures, functional responsibilities, levels of authority, and line of communication for activities affecting quality at the WVDP. The overall responsibility for QA during design, procurement, construction, installation, modifications, operation, and testing rests with WVNS. The QA Program is implemented through the combined efforts of WVNS and its contractors and the United States DOE.

B.12.2 QUALITY ASSURANCE PROGRAM

Section A.12.2 of Volume I supports WVNS' commitment to quality by reviewing the QA Program policies and requirements as they relate to facility structures, systems, and components. The QA Program meets the requirements of NQA-1 and all applicable DOE Orders.

B.12.3 QUALITY ASSURANCE IMPLEMENTATION

Section A.12.3 of Volume I describes the Quality Level and Service Classification Systems used by WVNS to implement the QA program.

Quality Levels and Service Classifications for FRS facility components, structures, and systems are listed in Table B.12.3-1.

B.12.4 REFERENCES

DOE 4700.1 U.S. Department of Energy, Order DOE 4700.1, "Project Management System," March, 1987.

DOE 5700.6B U.S. Department of Energy, Order DOE 5700.6B, "Quality Assurance," September, 1986.

ID 4700.1 U.S. Department of Energy, Idaho Operations Office, Order DOE 4700.1, "Project Management System," November, 1988.

ID 5700.6C U.S. Department of Energy, Idaho Operations Office, Order DOE 5700.6C, "Quality Assurance," April, 1989.

NQA-1 American Society of Mechanical Engineers, ASME NQA-1-1989, "Quality Assurance Program Requirements for Nuclear Facilities."

QM West Valley Nuclear Services Co., Inc., Quality Management Program, December 1991.

WVDP ACT Public Law 96-368, October 1, 1980

TABLE B.12.3-1

QUALITY LEVELS AND DESIGN CRITERIA
FOR FUEL RECEIVING AND STORAGE
COMPONENTS, STRUCTURES, AND SYSTEMS

Non-Safety Related Systems	Quality Level	Design Criteria	Figure Reference
Fuel Storage Pool	C	Drawing 1A-Q-3	Fig B.5.2-1,-2,-3
Cask Unloading Pool	C	Drawing 1A-Q-3	Fig B.5.2-1,-2,-3
Ventilation System (blower 1K-1)	C		Fig B.7.4-1
Recirculation Ventilation System	C		Fig B.5.2-6,B.7.4-1
Radwaste Treatment System	C	US NRC R.G. 1.143	Fig B.5.2-4, -5
Water Treatment System (in pit)	C	Specification V-11	
Water Treatment System (in pool)	C	DOE 6430.1A	
CAMS/Rad Monitors	C	DOE 5480.11	
Cranes and Hoists	N	Specification V-4	
Canister/Service Bridge	N	Specification V-5	
Lift Rack	N		
Clearwell Bucket	N		
Decon Pump and Stall	N		
Storage Racks	N	Specification M-3	Fig B.5.2-7
Heat Exchanger	N		
Security System	N		
Utilities	N		

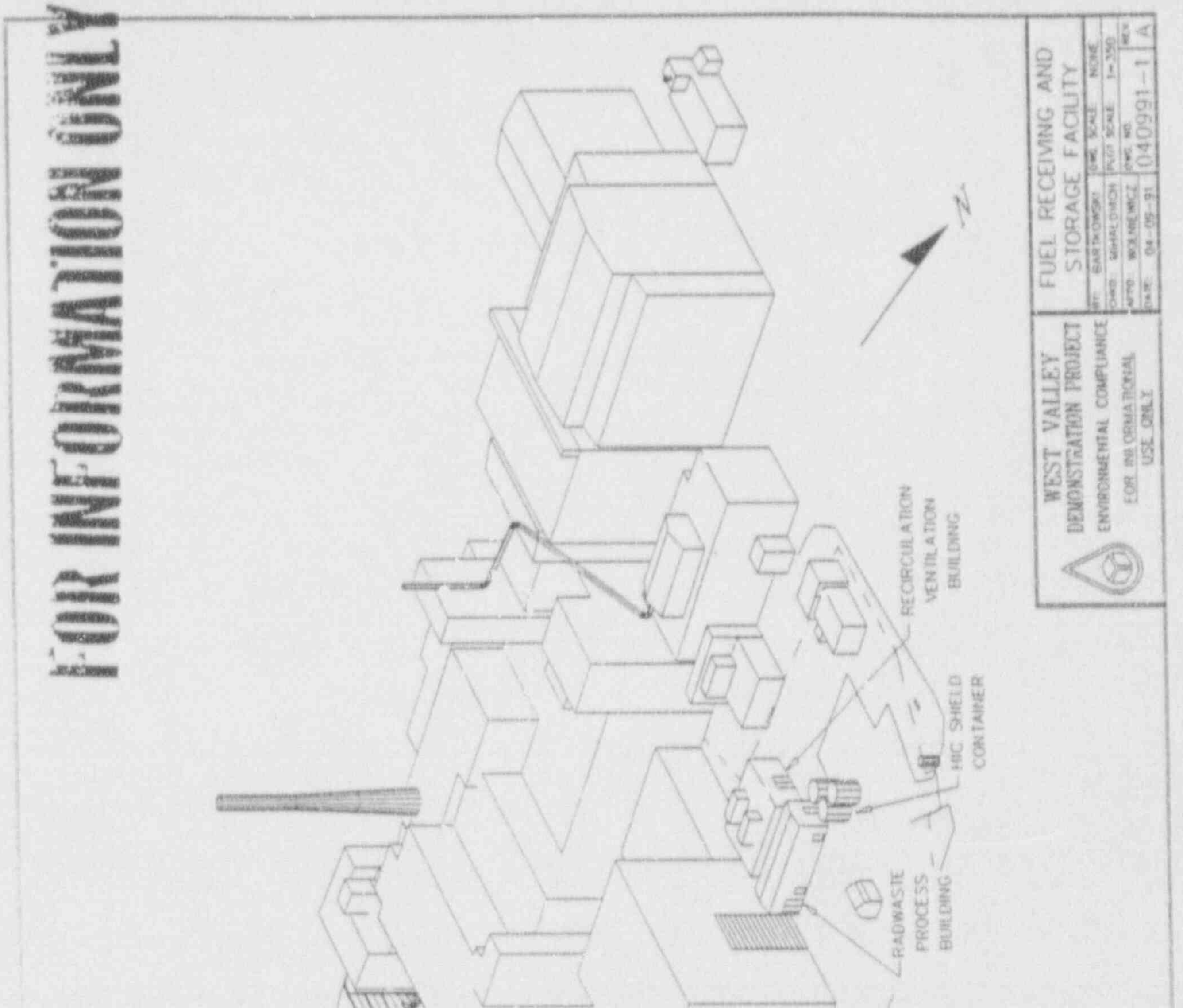
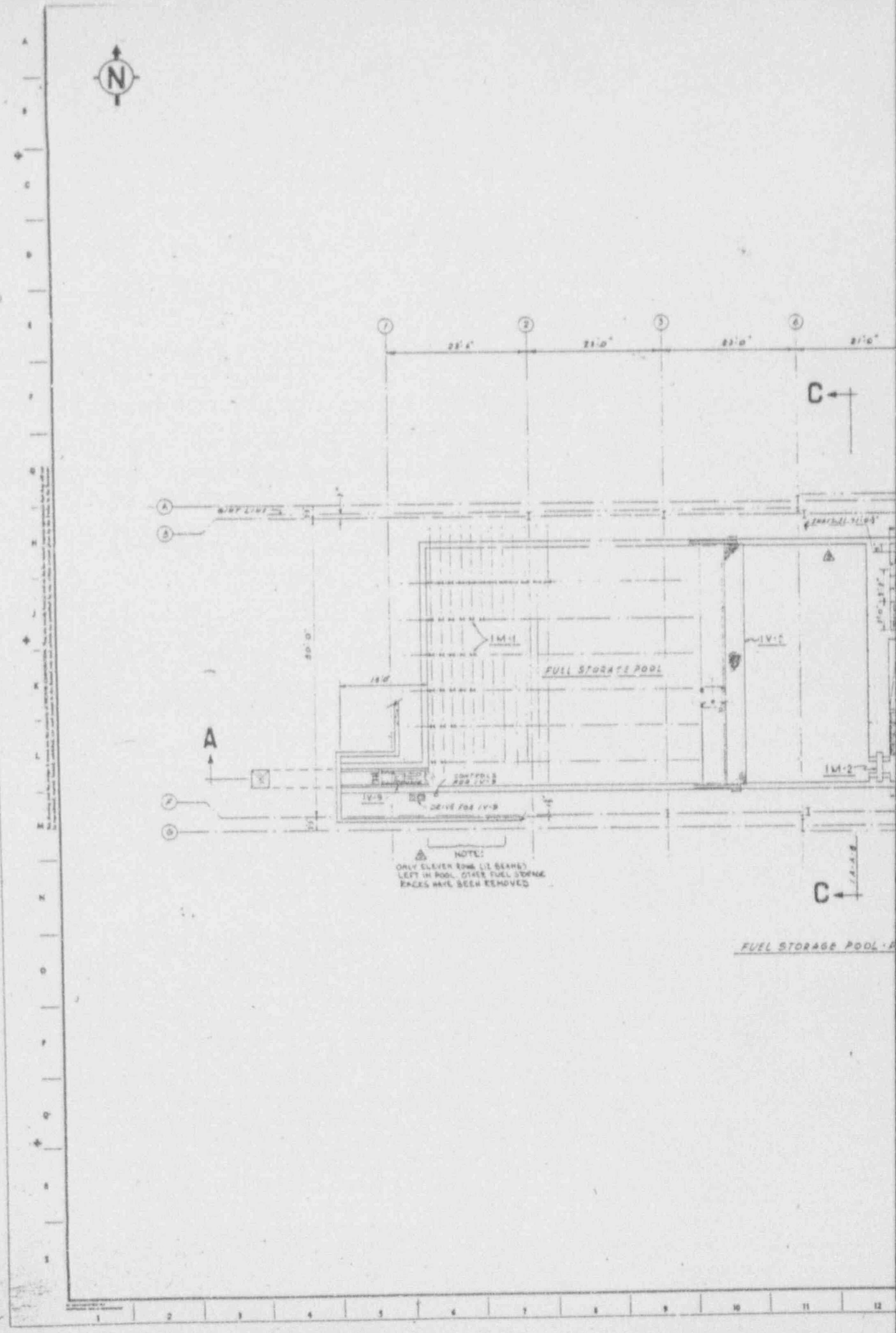

RECIRCULATION
VENTILATION
BUILDING
HIC SHIELD
CONTAINER
RADWASTE
PROCESS
BUILDING
FRS
BUILDING
N
WEST VALLEY
DEMONSTRATION PROJECT
ENVIRONMENTAL COMPLIANCE
FOR INFORMATIONAL
USE ONLY
FUEL RECEIVING AND
STORAGE FACILITY
040991-1
A

Figure B.5.1-1



N
C
A
IM-1
FUEL STORAGE POOL
IV-2
IM-2
IV-9
DRIVE FOR IV-9
NOTE:
ONLY ELEVEN ROWS (12 BEAMS) LEFT IN POOL. OTHER FUEL STORAGE RACKS HAVE BEEN REMOVED
FUEL STORAGE POOL

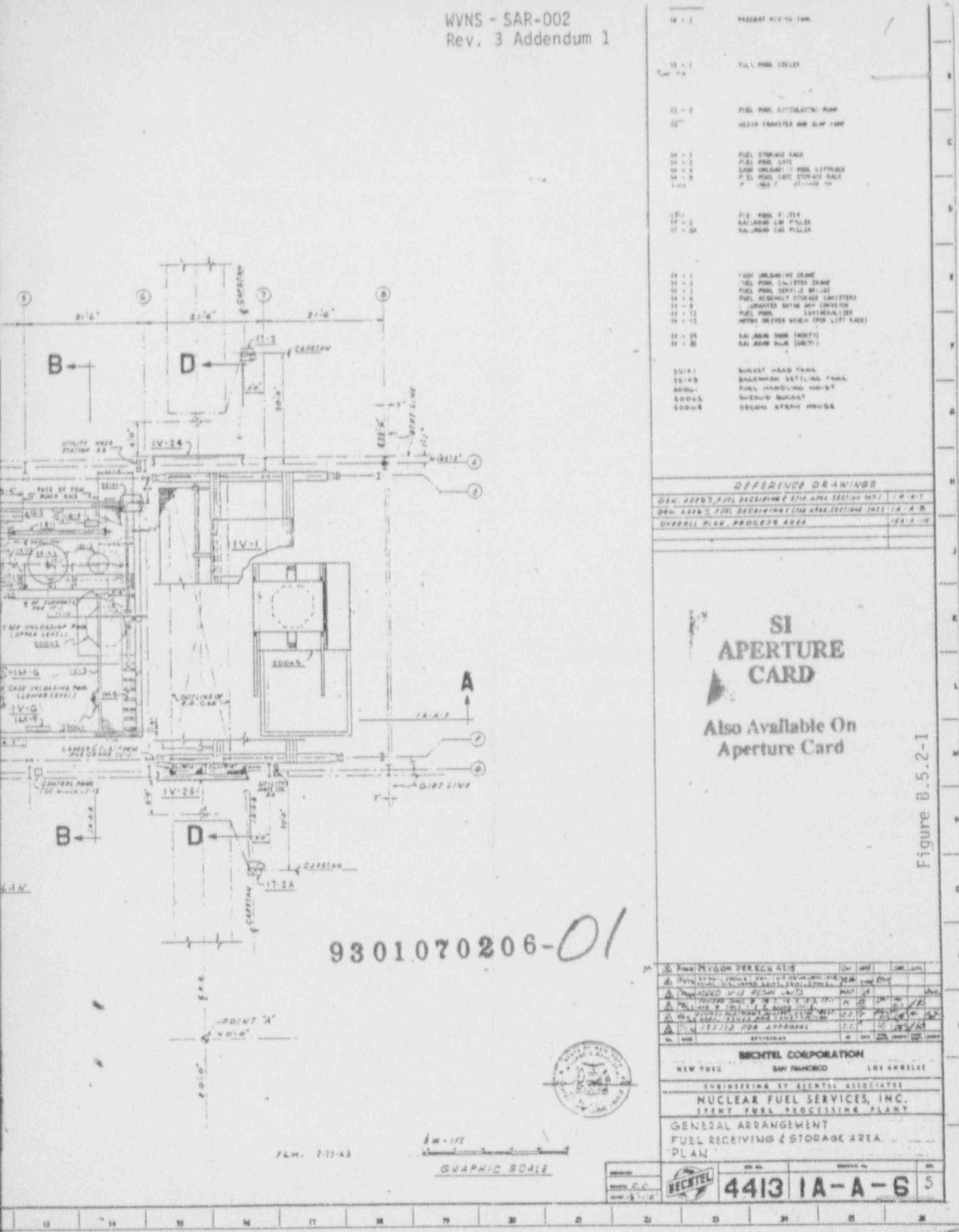
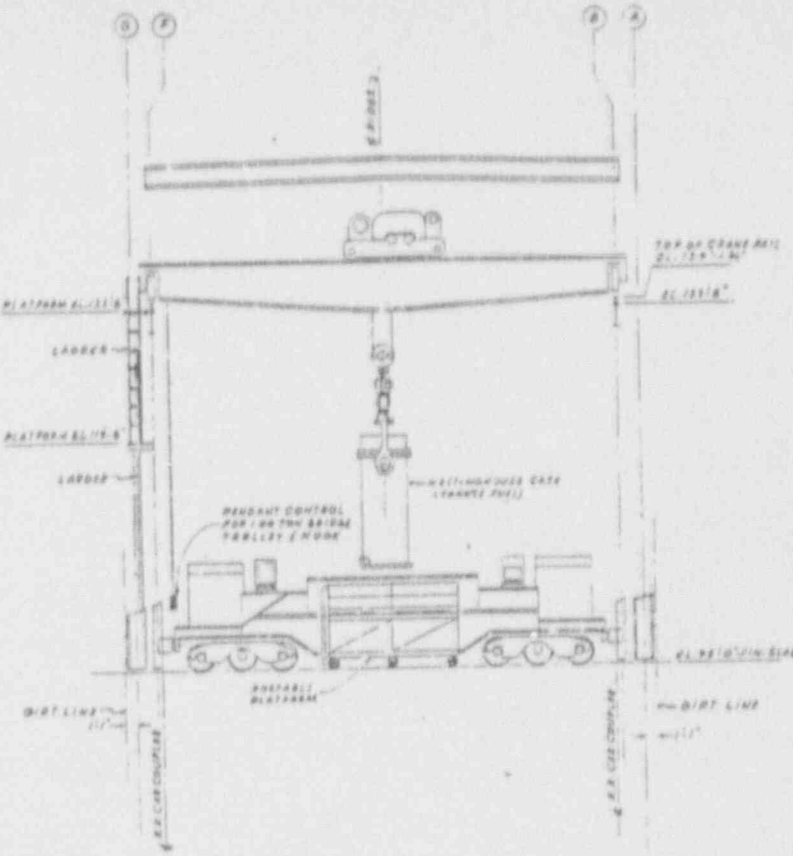
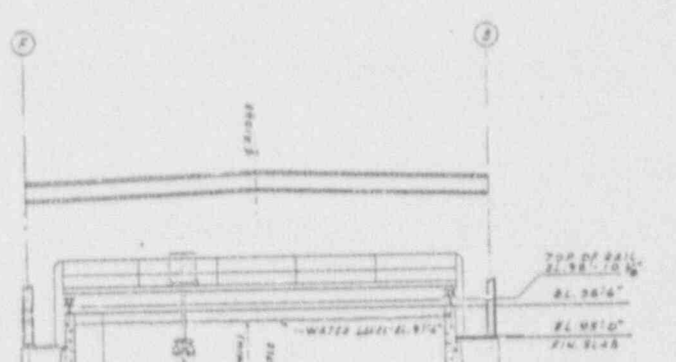

REFERENCE DRAWINGS
OVERALL PLAN, PROCESS AREA
B
D
A
IV-24
IV-1
IV-25
IT-2
IT-2A
CAPSTAN
GIRT LINE
SI
APERTURE
CARD
Also Available On
Aperture Card
9301070206-01
BECHTEL CORPORATION
NEW YORK
SAN FRANCISCO
LOS ANGELES
ENGINEERING BY BECHTEL ASSOCIATES
NUCLEAR FUEL SERVICES, INC.
SPENT FUEL PROCESSING PLANT
GENERAL ARRANGEMENT
FUEL RECEIVING & STORAGE AREA
PLAN
4413
1A-A-6
5
GRAPHIC SCALE

Figure B.5.2-1



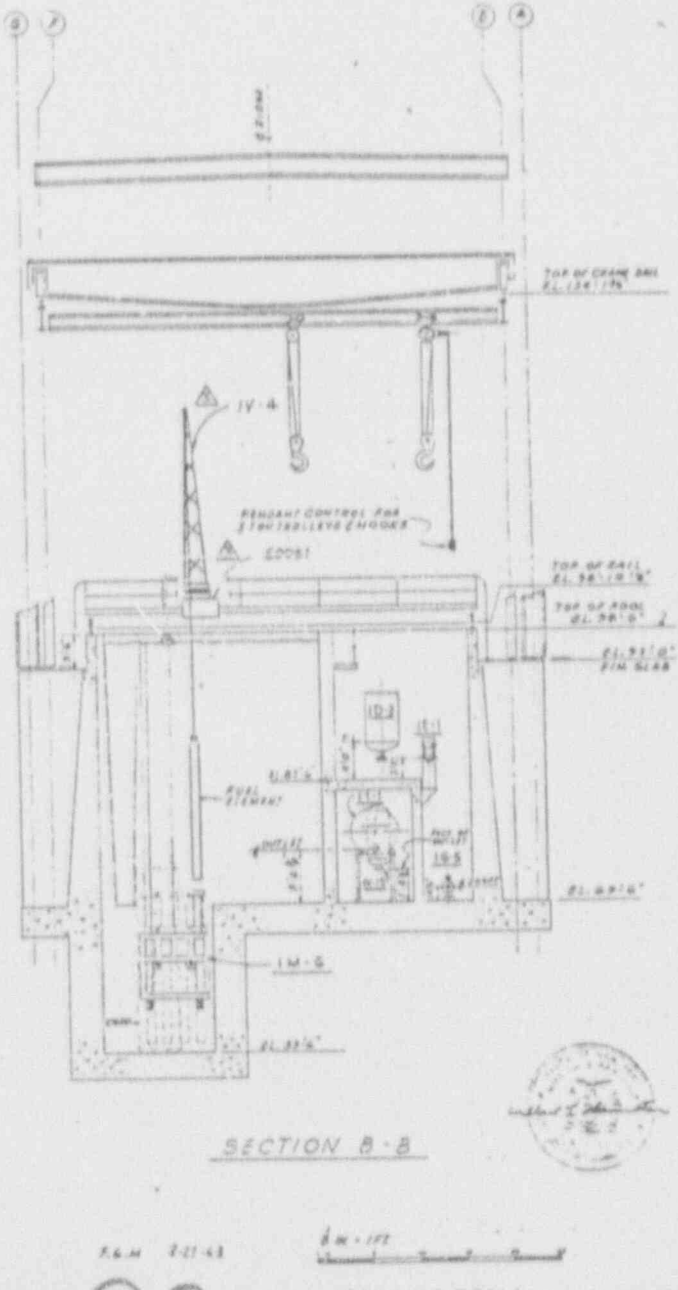
SECTION D-D



SECTION C-C

SI
APERTURE
CARD

Also Available On
Aperture Card



SECTION B-B

GRAPHIC SCALE

Figure B.5.2-2

REFERENCE DRAWINGS

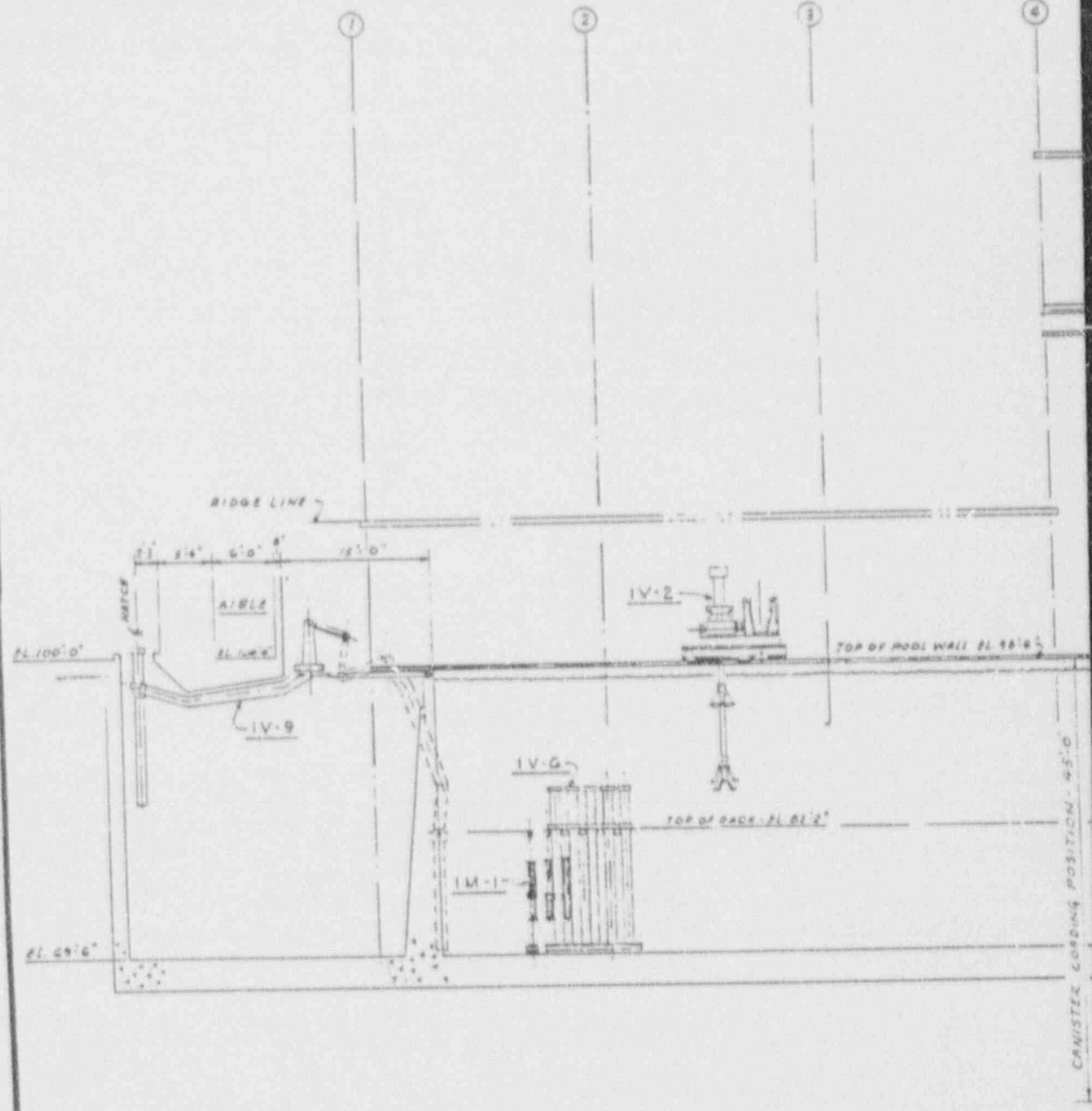
BECHTEL CORPORATION
NEW YORK SAN FRANCISCO LOS ANGELES

ENGINEERING BY BECHTEL ASSOCIATES
NUCLEAR FUEL SERVICES, INC.
SPENT FUEL PROCESSING PLANT

GENERAL ARRANGEMENT
FUEL RECEIVING & STORAGE AREA
SECTIONS SHEET 2

4413 IA-A-8 3

A B C D E F G H



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REV.	DESCRIPTION	MADE CHKD DATE	REV.	DESCRIPTION	MADE CHKD DATE

FORM NO. E-399-REV. 7/71 "D"

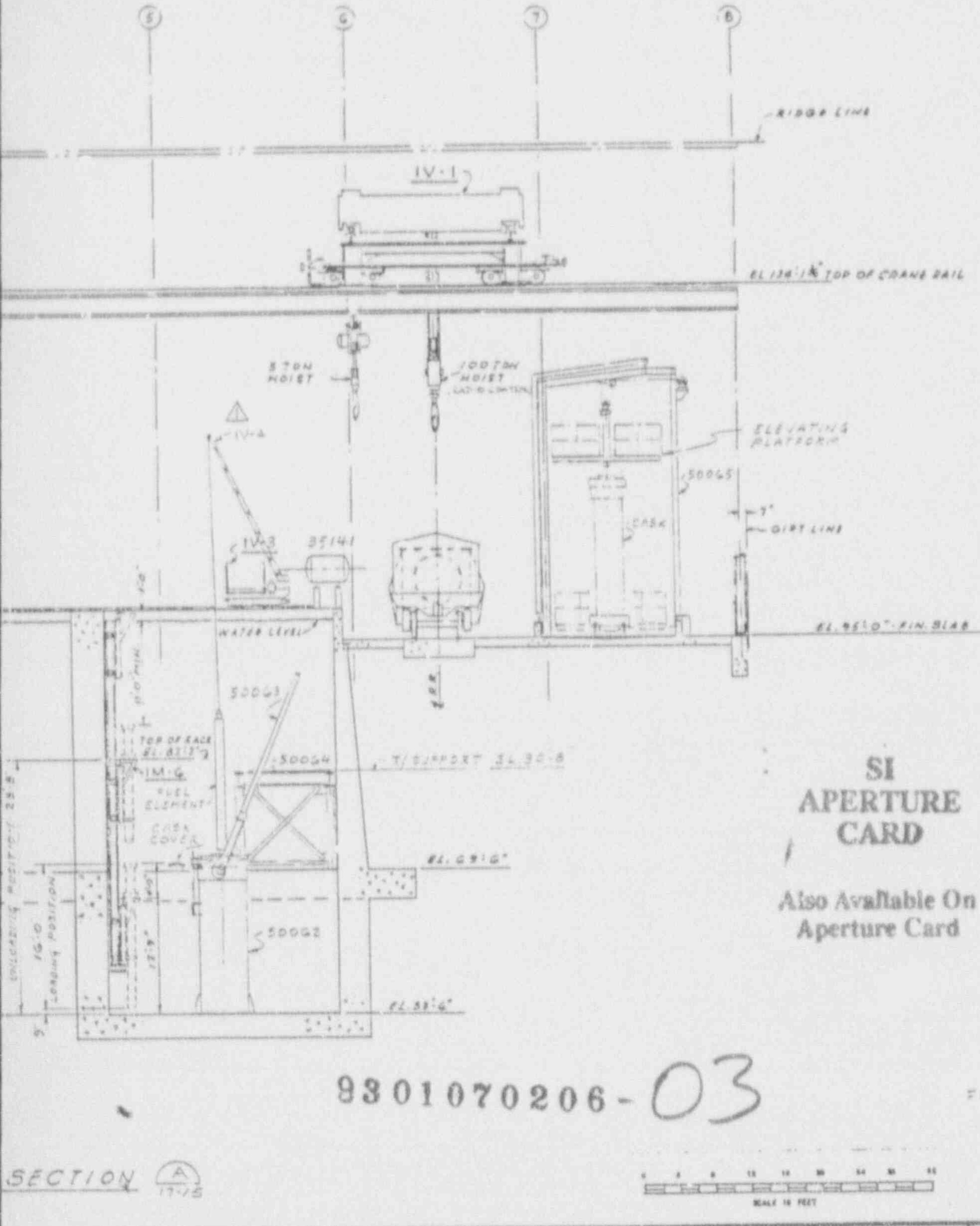


Figure B.5.2-3

SI APERTURE CARD

Also Available On Aperture Card

9301070206-03

FIGURE III-2-2

DESIGNED		
DRAWN	ZIENTARSKI	1-26-73
CHECKED		
APPROVED		1-30-73
APPROVED		
APPROVED		

REV. | RELEASED FOR | APP. DATE

TITLE: F.R.S. MODIFICATION AREA #17 FUEL RECEIVING & STORAGE AREA LAYOUT SECTION "A"

FOR: N.F.S. INC W. VALLEY N.Y.

SCALE: 1/8" = 1'-0"

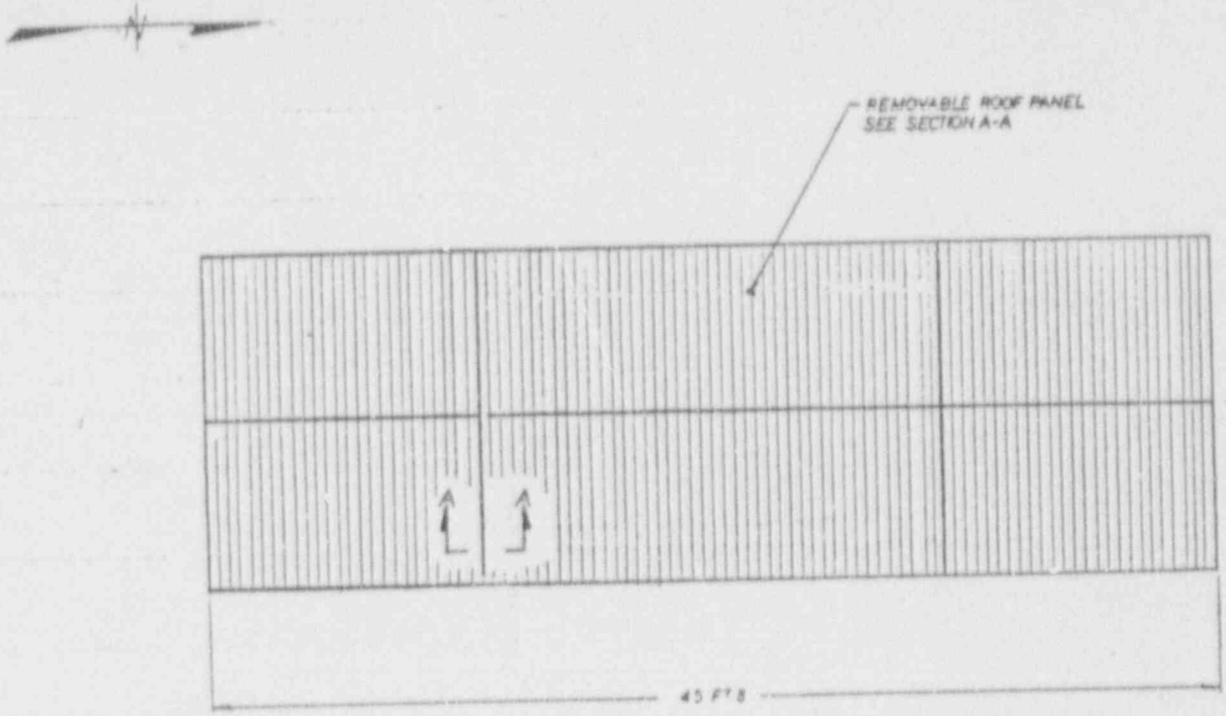
BLAW-KNOX CHEMICAL PLANTS, INC.
PITTSBURGH, PENNSYLVANIA
ENGINEERS - CONSTRUCTORS

DRAWING NUMBER: 2401-217-16

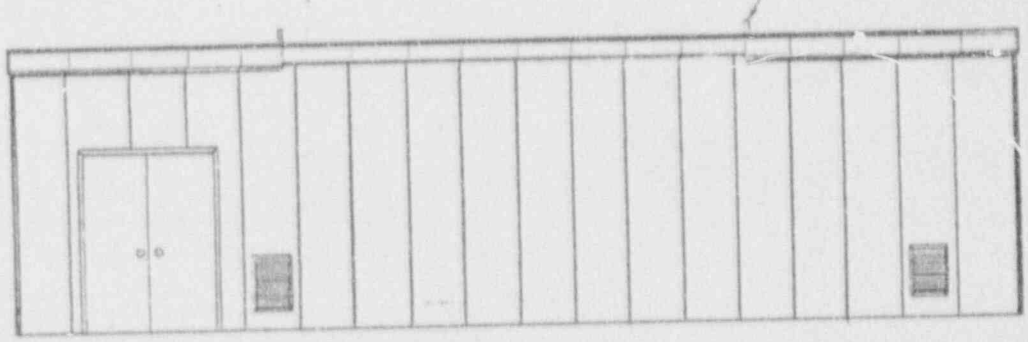
REV. 1

DO NOT SCALE THIS DRAWING. IT MAY BE REDUCED FROM ORIGINAL SIZE.

PRINTS OF THIS DRAWING ON PINK PAPER ARE "PRELIMINARY" AND SHALL NOT BE USED FOR MANUFACTURING OR CONSTRUCTION PURPOSES.

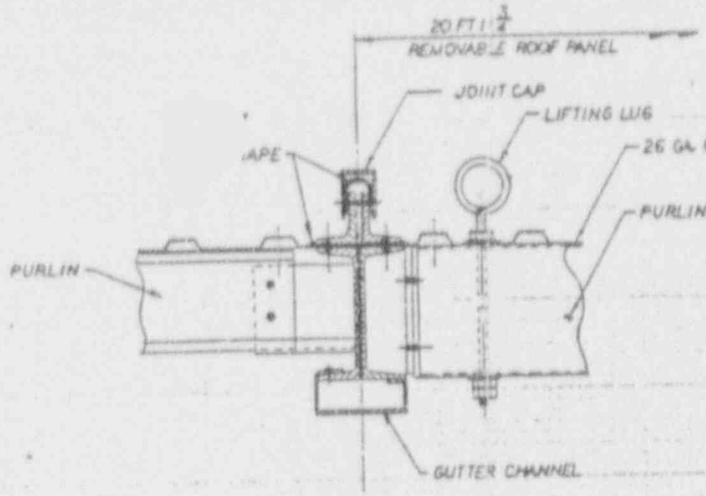


PLAN

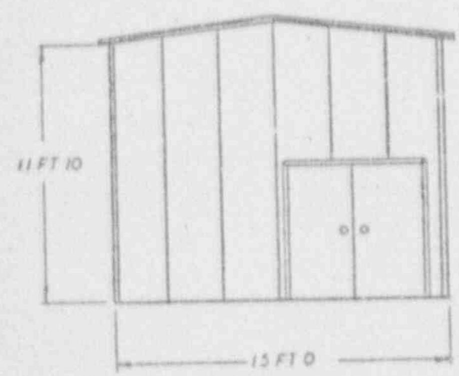


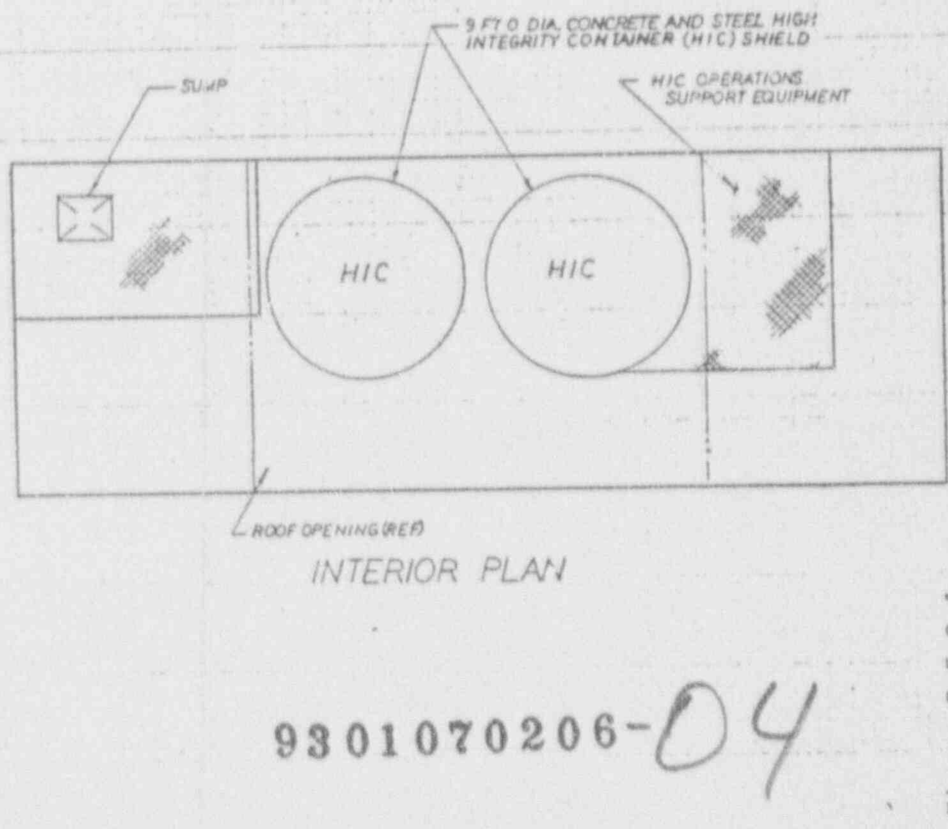
ELEVATION

REVISIONS		
NO.	DESCRIPTION	APVD
0	ER 20.46	



SECTION A-A (C7)
SCALE: NONE





INTERIOR PLAN

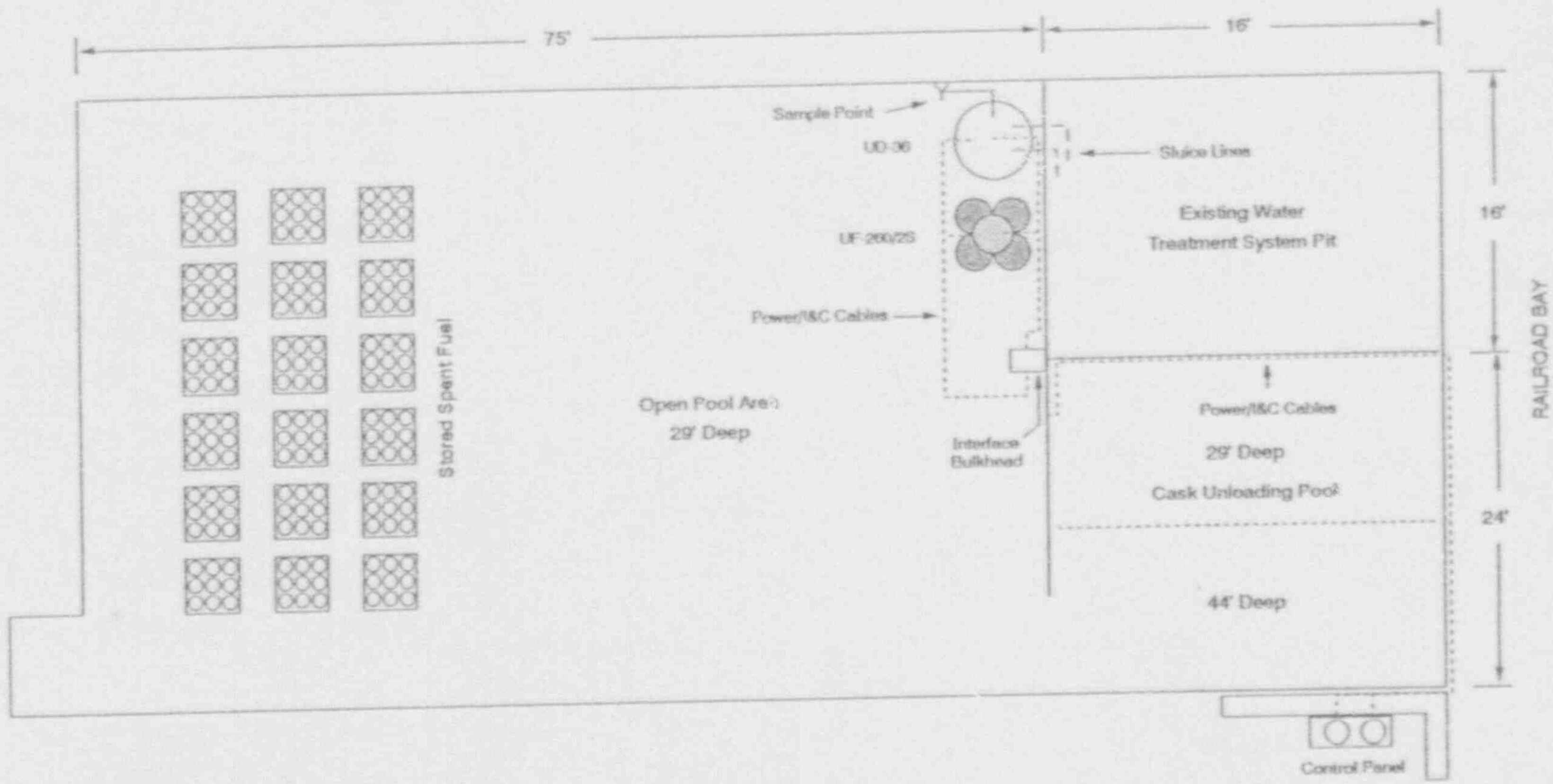
9301070206-04

Figure B.5.2-4

QTY REQD	ITEM NO.	NOMENCLATURE OR DESCRIPTION	PART OR IDENTIFYING NO.	MATERIAL OR SPEC

PARTS LIST

UNLESS OTHERWISE SPECIFIED	DRAFTER C WEIGHT	DATE 3-14-91	A/E CONTRACT NO.	CONT IDENT NO.
ALL DIMENSIONS IN FT & INCHES	CHECKER	DATE 3-25-91	FOR West Valley Nuclear Services Co.,Inc. West Valley, New York	
TOLERANCES-DO NOT SCALE	ENGR.	DATE 4-3-91		
2 PL DEC ± N/A; 3 PL DEC ± N/A; ANGLES ± N/A			FRS RADWASTE PROCESS BULDING	
FRACTIONS ± 1/4	COG/SUPE APPVL	DATE 4/4/91		
D-SPEC NO. 1848	CLIENT APPVL	DATE	DWG SIZE D; SCALE 1/4" = 1 FT 0; WEIGHT	DWG NO. 9000-4381; REV 0
NEXT ASSEMBLY	QA APPVL	DATE	A/E DWG NO.	SHEET 1 OF 1
	APPVL ... for JRH	DATE 4-5-91		



PO #______ AR #______

[✓] APPROVED
[ ] APPROVED WITH COMMENTS
[ ] DISAPPROVED
[ ] RECEIPT ACKNOWLEDGED

SIGNED ______
COGNIZANT ENGINEER

4	Revised per customer comments.	3/13/92	
3	Revised per customer comments.	1/13/92	
2	Revised per customer comments.	11/16/91	
1	Revised per customer comments.	10/23/92	

DIVERSIFIED TECHNOLOGIES
103 DIXON DRIVE CHESTERTOWN, MD 21620

FRS WATER TREATMENT SYSTEM

DATE: 07/02/91	PROJECT: WVNS	DRWN: C.E.J.
SCALE: None		CK'D:
TOL: None	DRW'G: DT-700	APP'D:

Figure B.5.2-4a

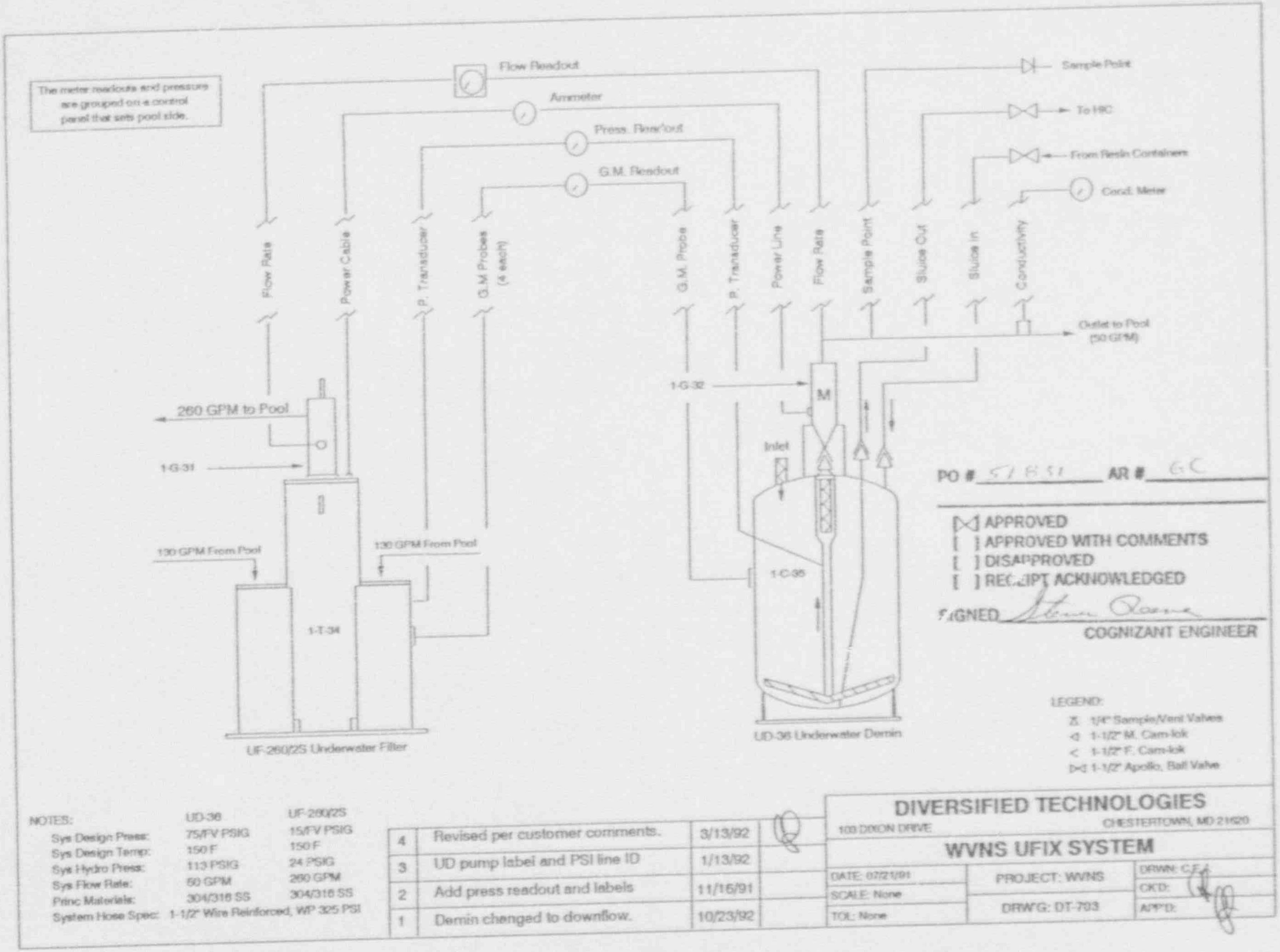


Figure B.5.2-4b

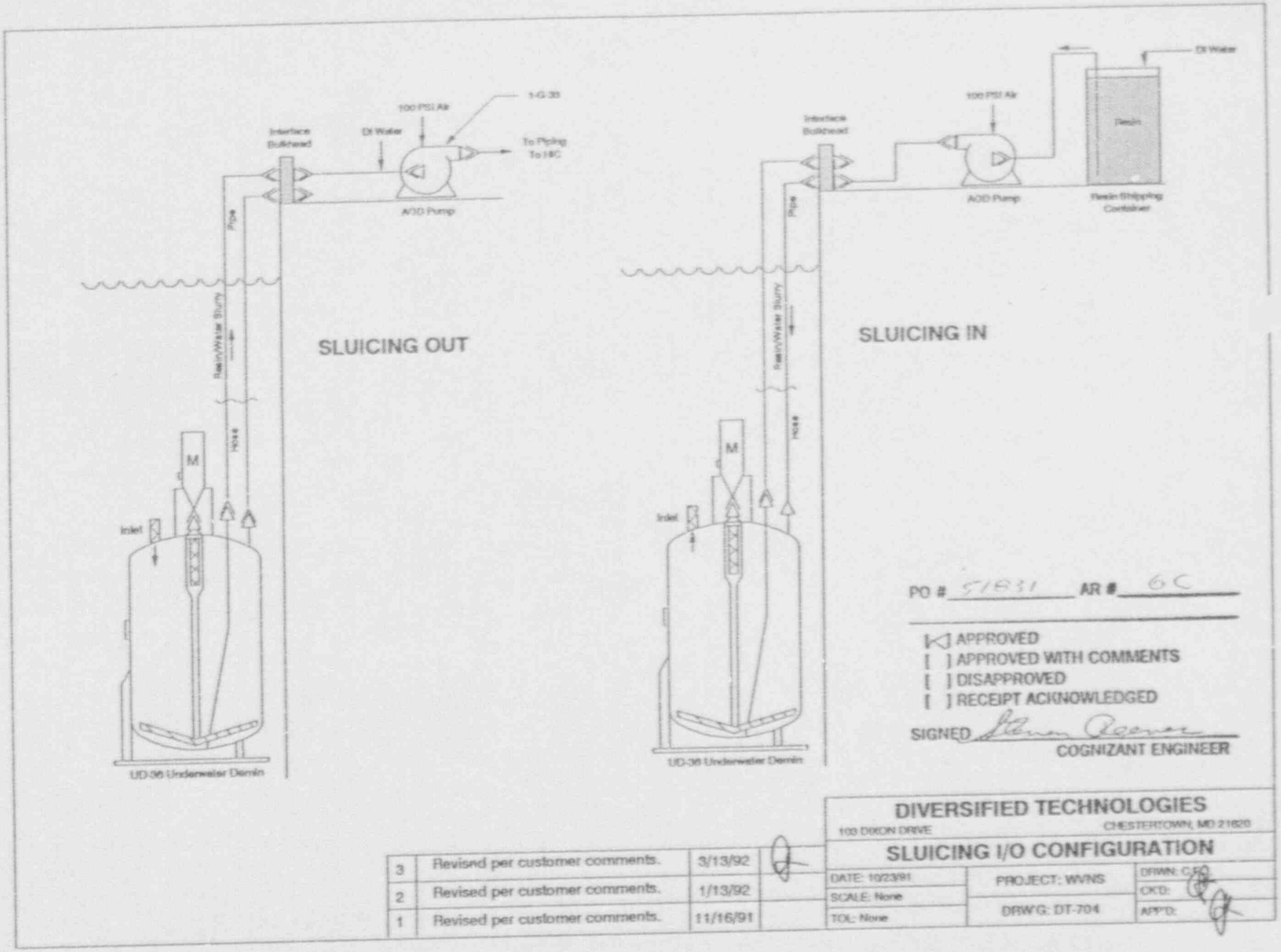
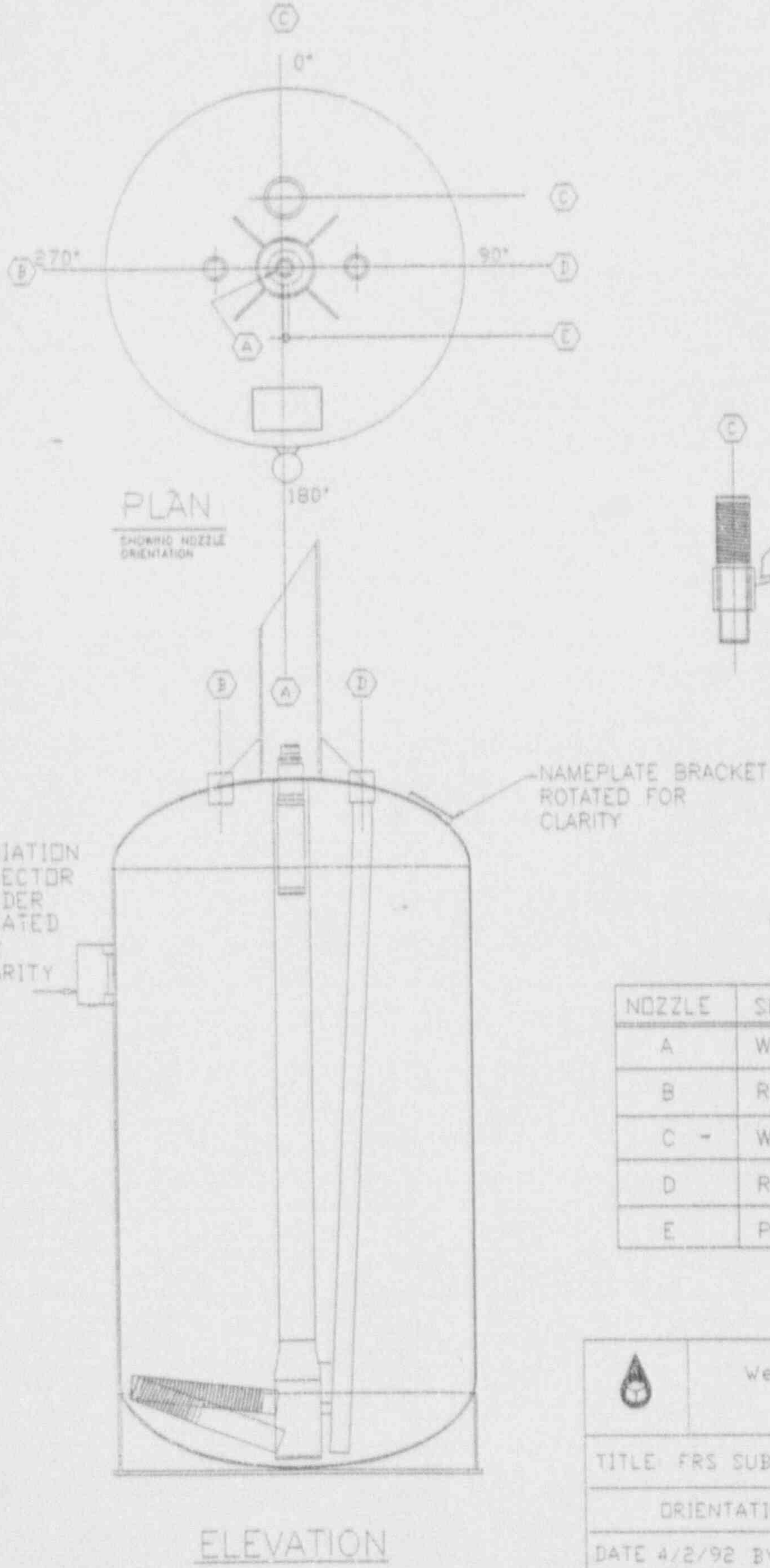
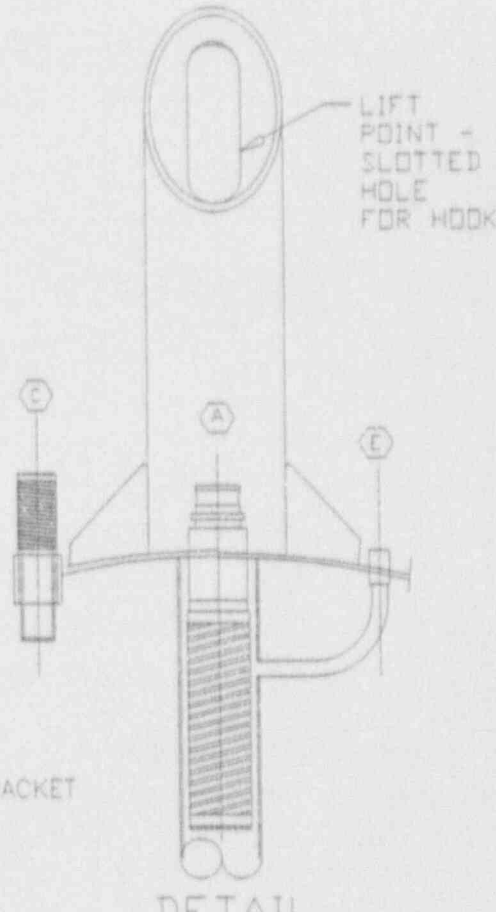


Figure B.5.2-4c





NOZZLE	SERVICE	SIZE (in.)
A	WATER OUT	1 1/2
B	RESIN IN	1 1/2
C -	WATER IN	3
D	RESIN OUT	1 1/2
E	PRESSURE TAP	3/8

West Valley Nuclear Services Co., Inc.
West Valley N.Y.

TITLE: FRS SUBMERGED DEMINERALIZER

ORIENTATION OF NOZZLES ON HEAD

DATE 4/2/92 BY: S.R. Sloan CK'D: K.J. Cotter

SKETCH NO: SK-SRR-040292-1 SHEET 1 OF 1

Figure B.5.2-4d

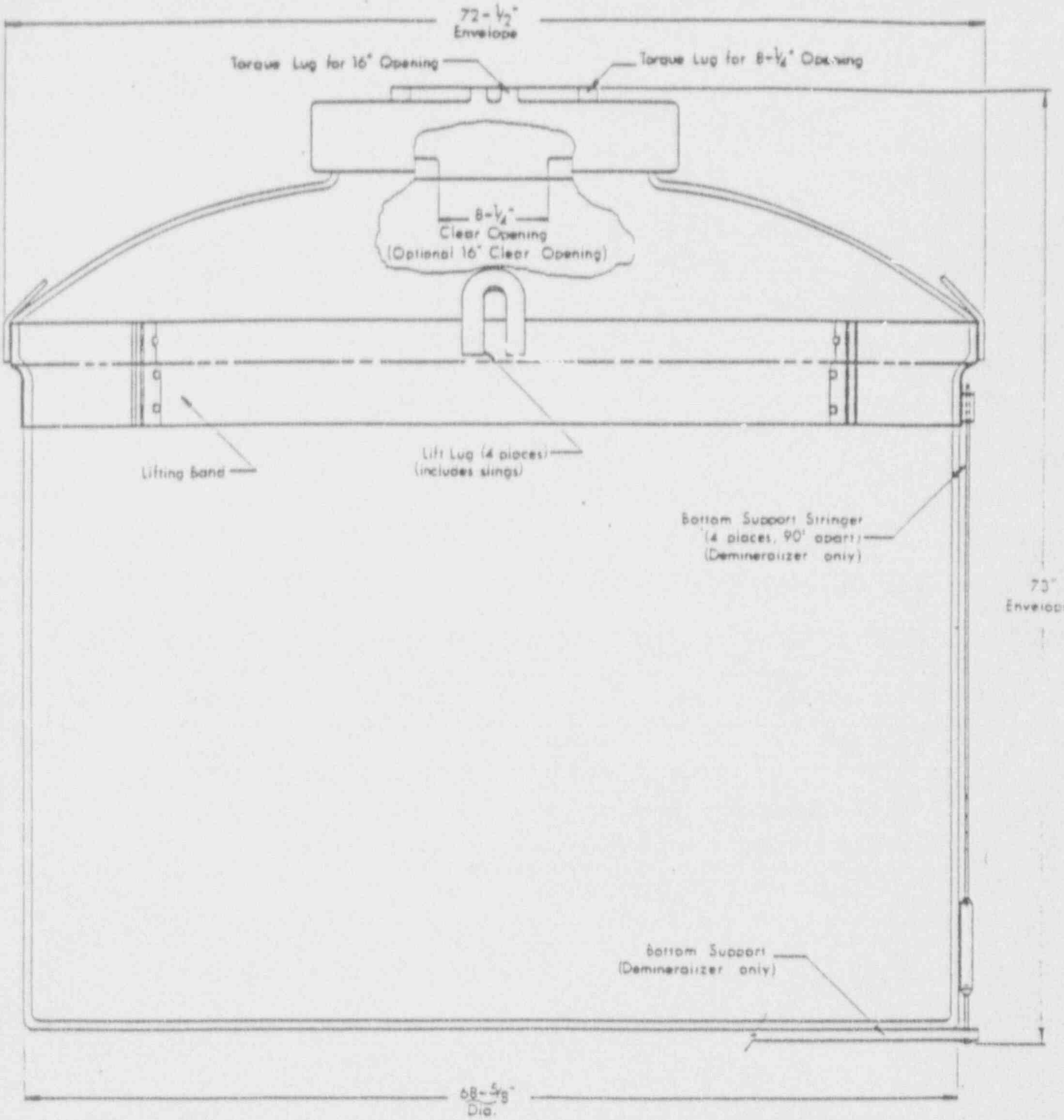
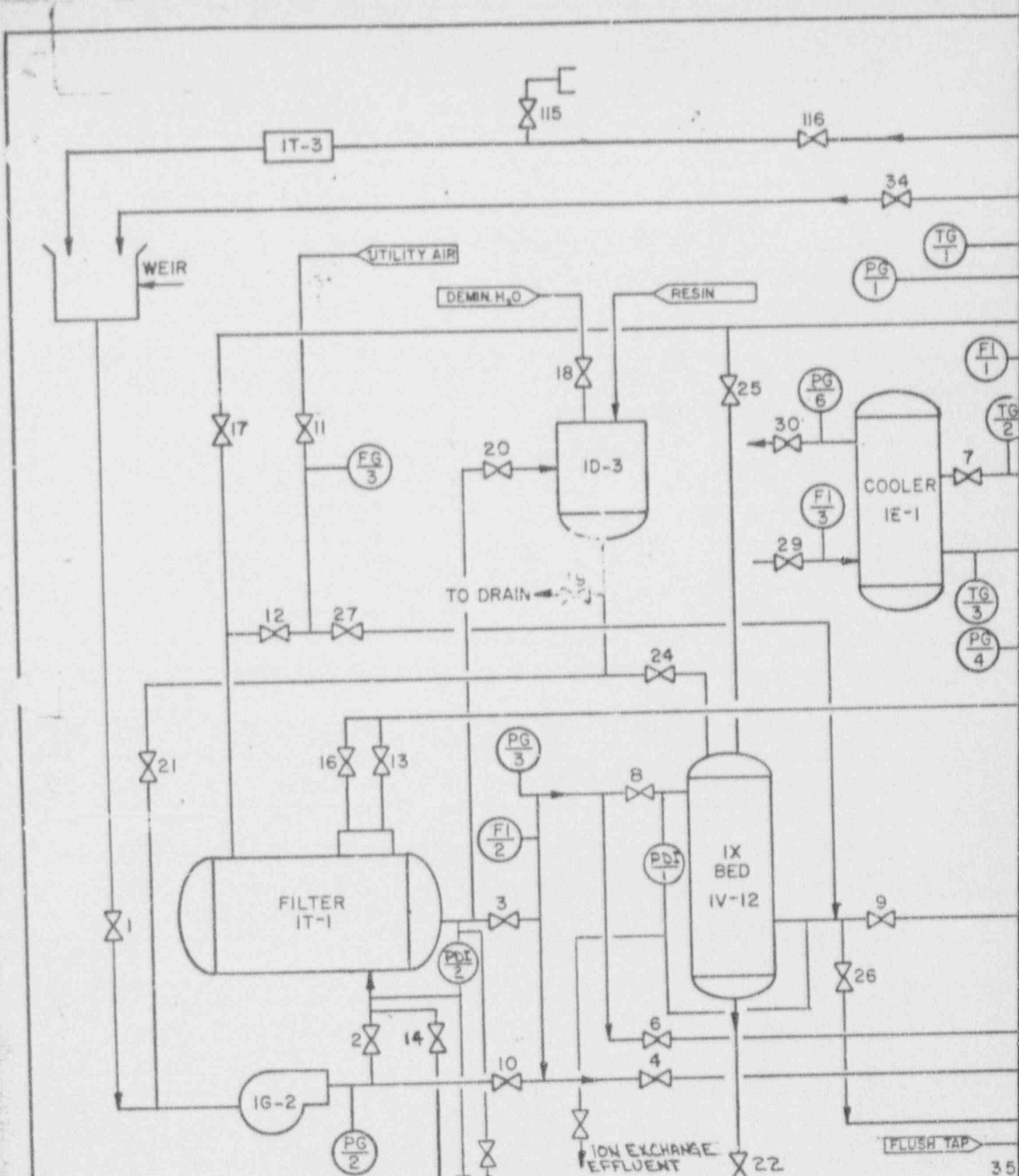


Figure B.5.2-9

HITTMAN RADLOK™-100
HIGH INTEGRITY CONTAINER

FOR INFORMATION ONLY



FRS WATER TREATMENT AREA PIPING

900B-130 L.R.P. - 8/10/83
CK'D BY LEK DATE 8/10/83
APP'D BY PB DATE 8/10/83
REV. 2 APP'D PB DATE 9/22/83

REV	DESCRIPTION	APPD	DATE
2	REV PER ECN 2940	HMV	5/3/89

CK-KMQ

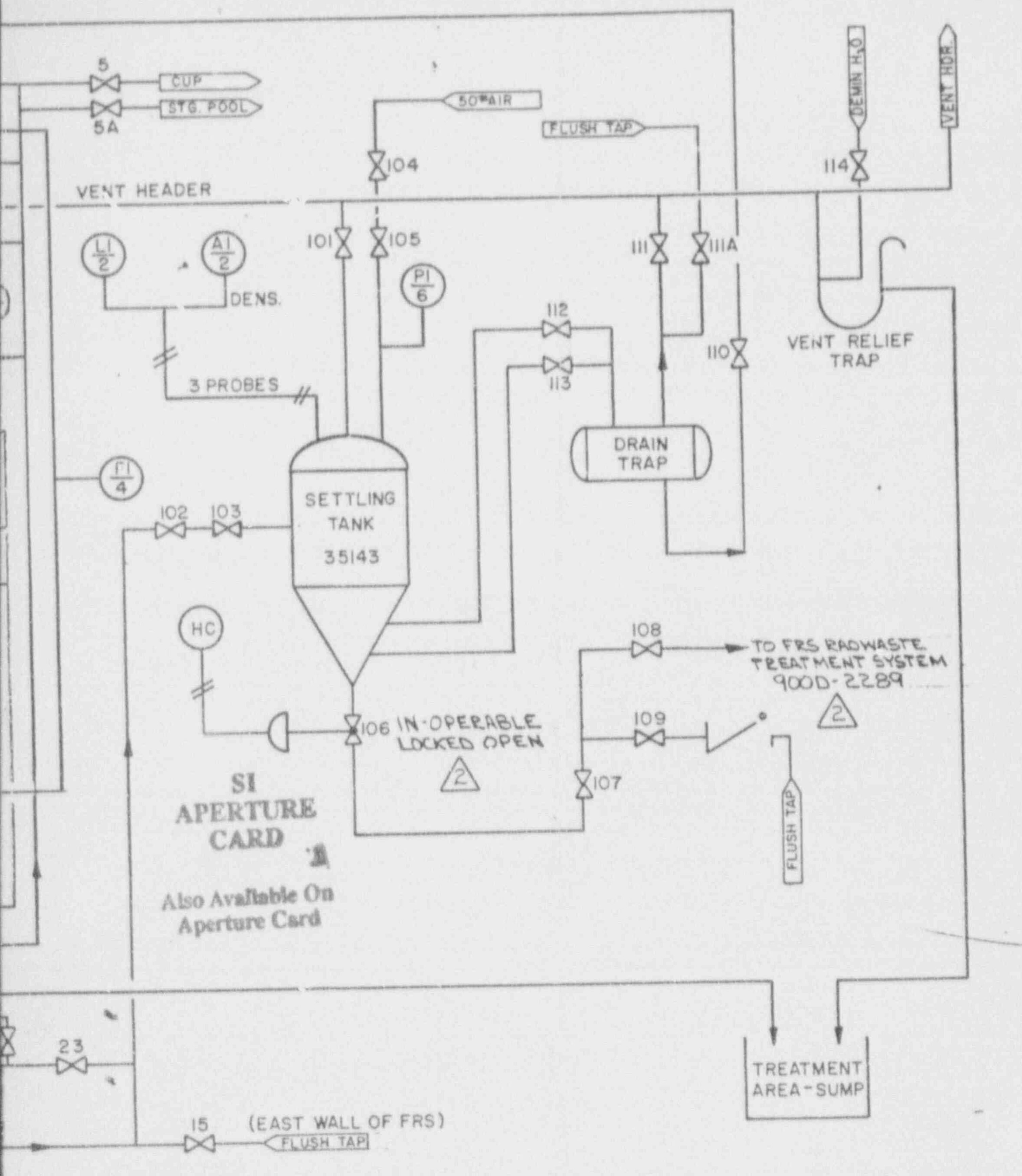
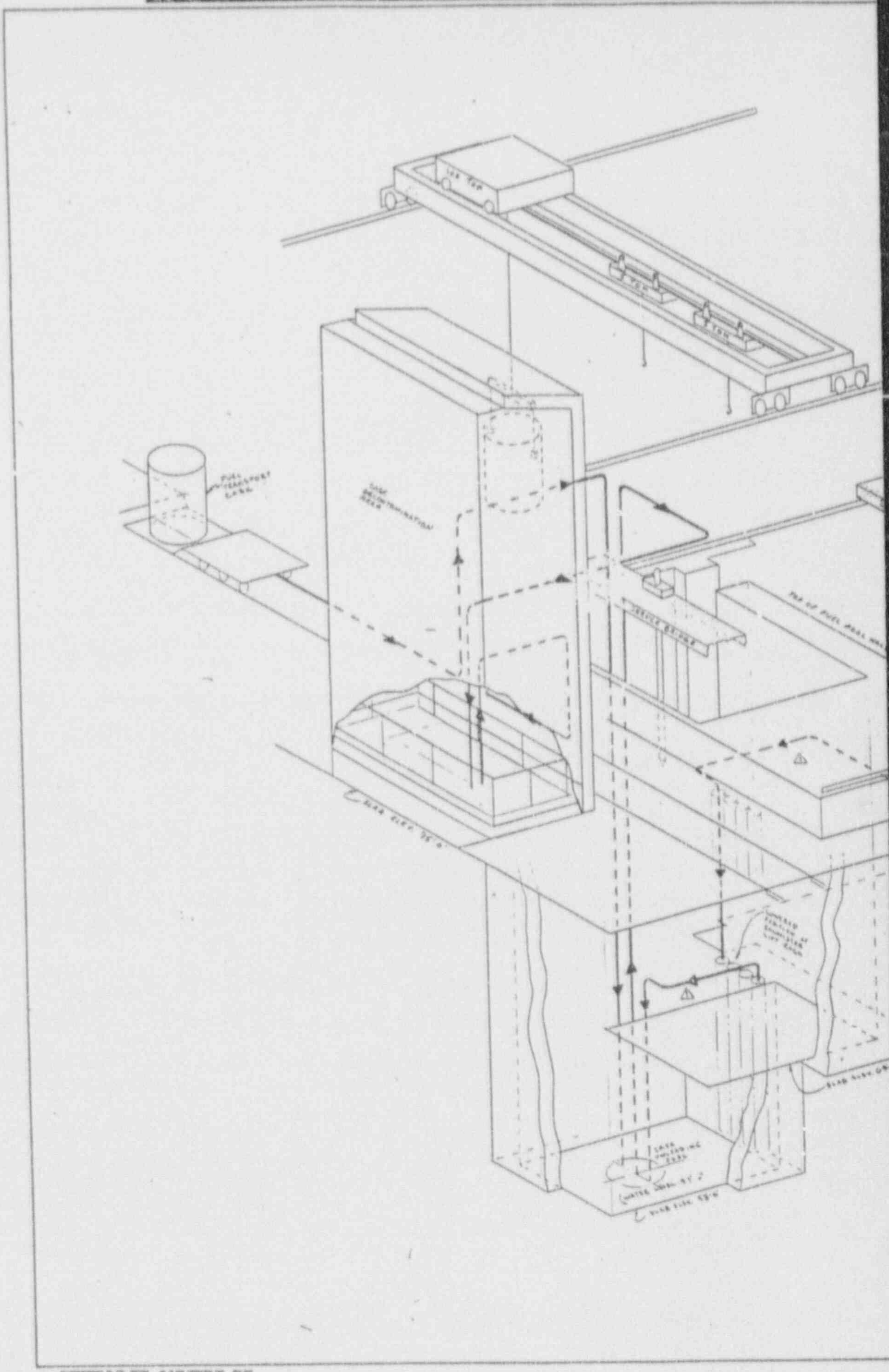


Figure B.6.1-1

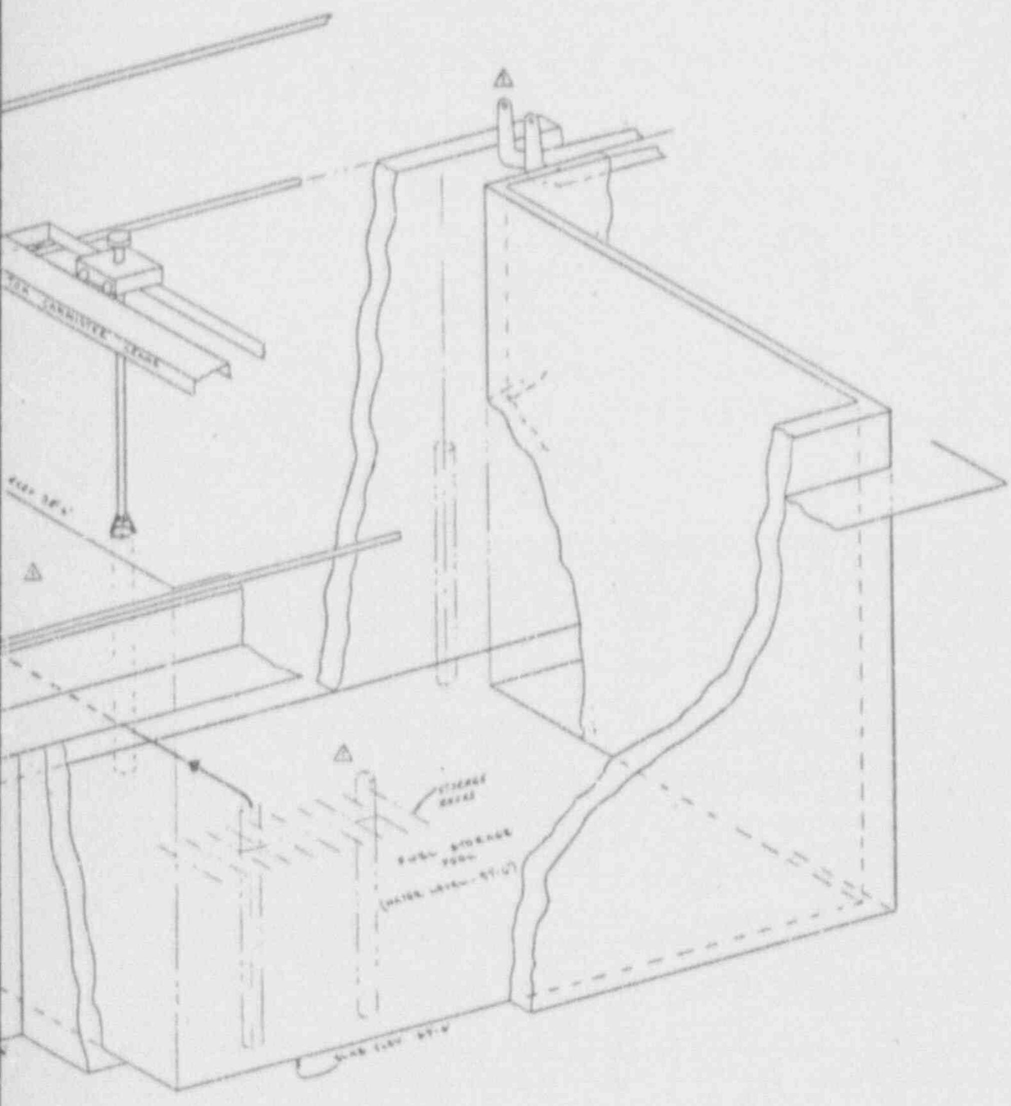
SI
APERTURE
CARD

Also Available On
Aperture Card

9301070206-05



E D C B A



SI APERTURE CARD

Also Available On Aperture Card

FIGURE IV-2-1 FRS FLOW DIAGRAM

NUCLEAR FUEL SERVICES, INC.
WEST VALLEY, NEW YORK

FRS FLOW DIAGRAM

1 A-A-1121

9301070206-06

Figure B.6.1-2

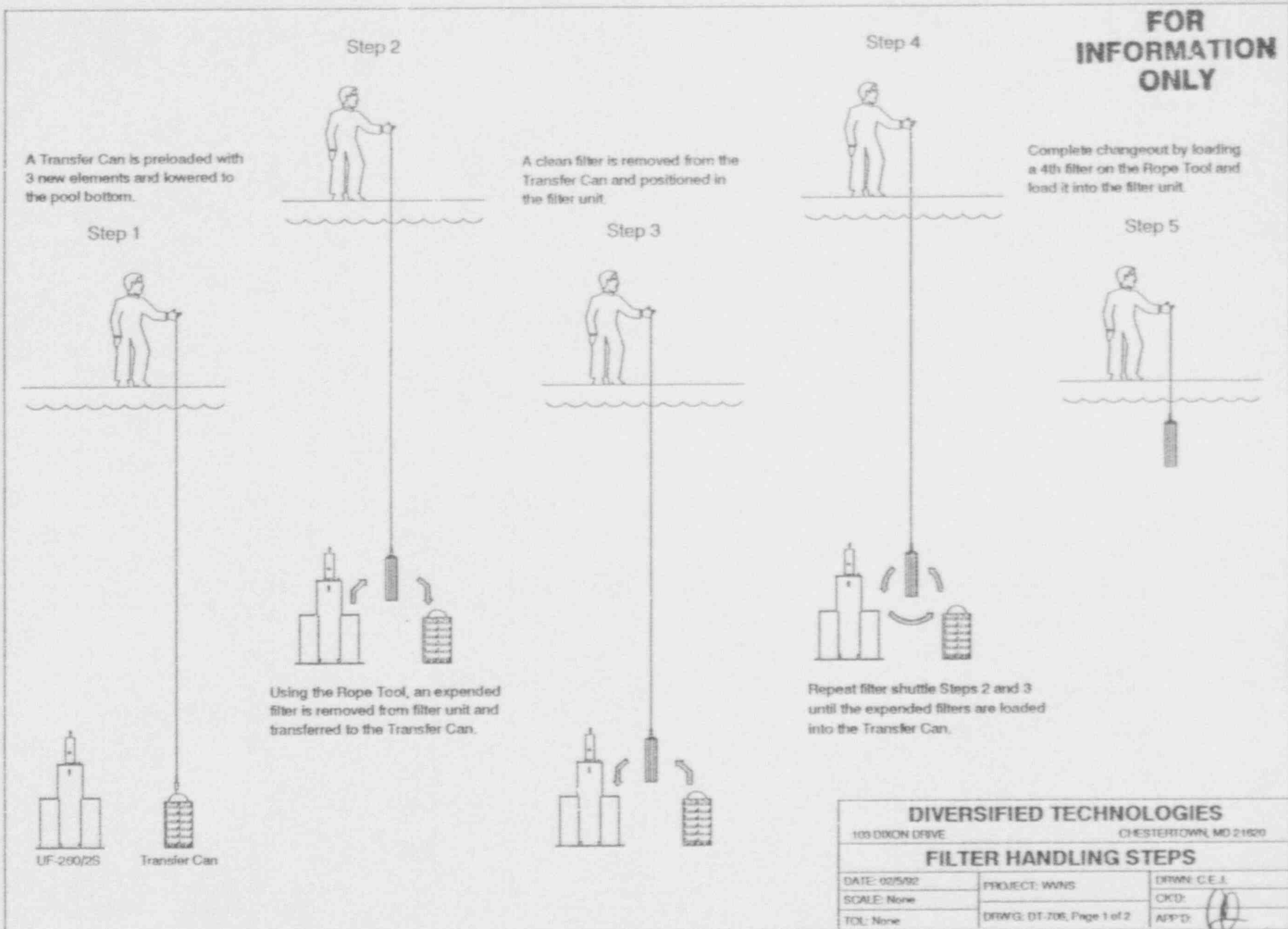

FOR INFORMATION ONLY
A Transfer Can is preloaded with 3 new elements and lowered to the pool bottom.
Step 1
UF-260/2S
Transfer Can
Step 2
Using the Rope Tool, an expended filter is removed from filter unit and transferred to the Transfer Can.
A clean filter is removed from the Transfer Can and positioned in the filter unit.
Step 3
Step 4
Repeat filter shuttle Steps 2 and 3 until the expended filters are loaded into the Transfer Can.
Complete changeout by loading a 4th filter on the Rope Tool and load it into the filter unit.
Step 5
DIVERSIFIED TECHNOLOGIES
103 DIXON DRIVE
CHESTERTOWN, MD 21620
FILTER HANDLING STEPS
DATE: 02/5/92
SCALE: None
TOL: None
PROJECT: WVNS
DRWG: DT-706, Page 1 of 2
DRWN: C.E.J.
CKD:
APP'D:

Figure B.6.1-3

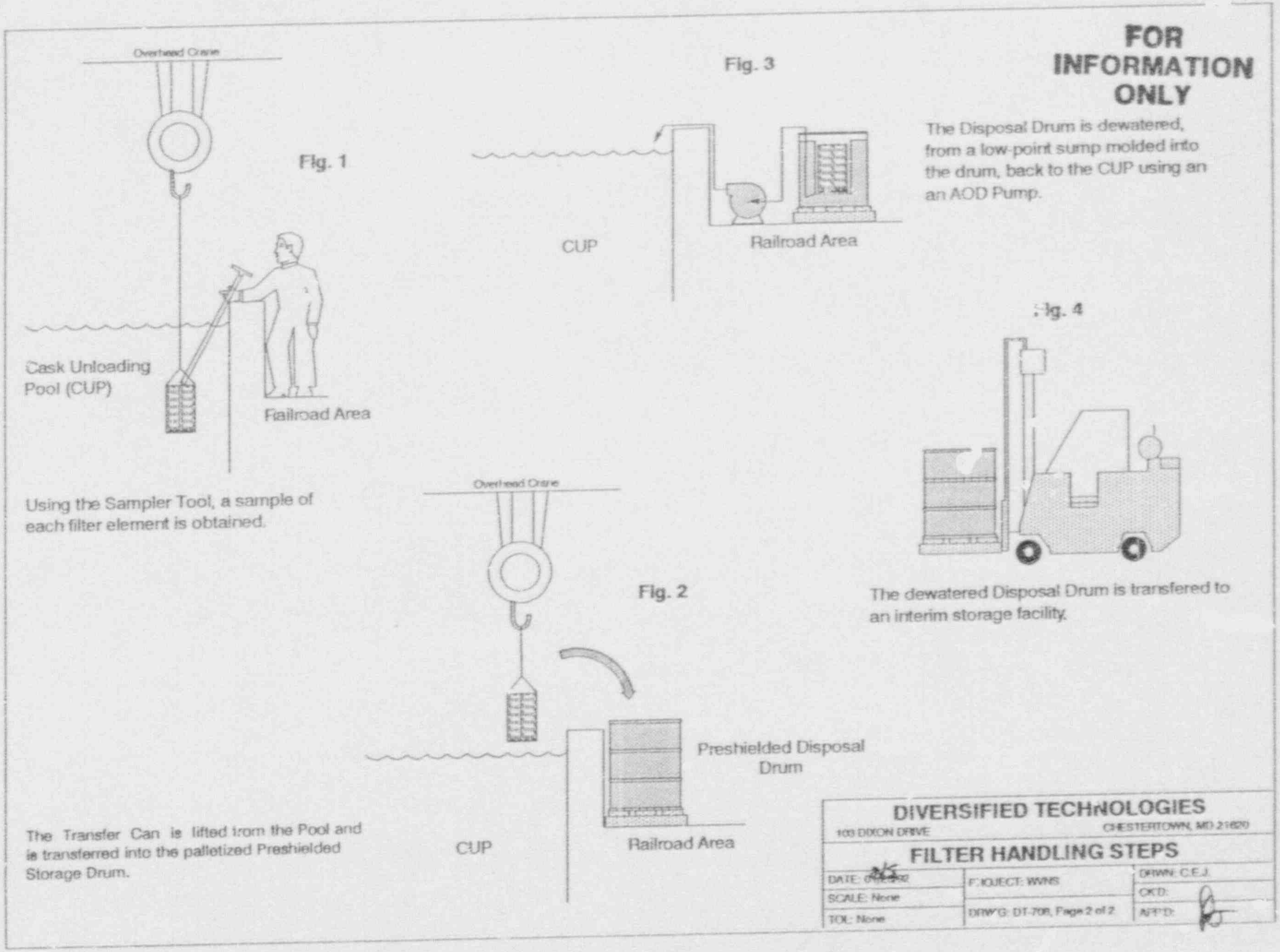


Figure B.6.1-4

WVNS-SAR-002
Rev. 3 Addendum 1

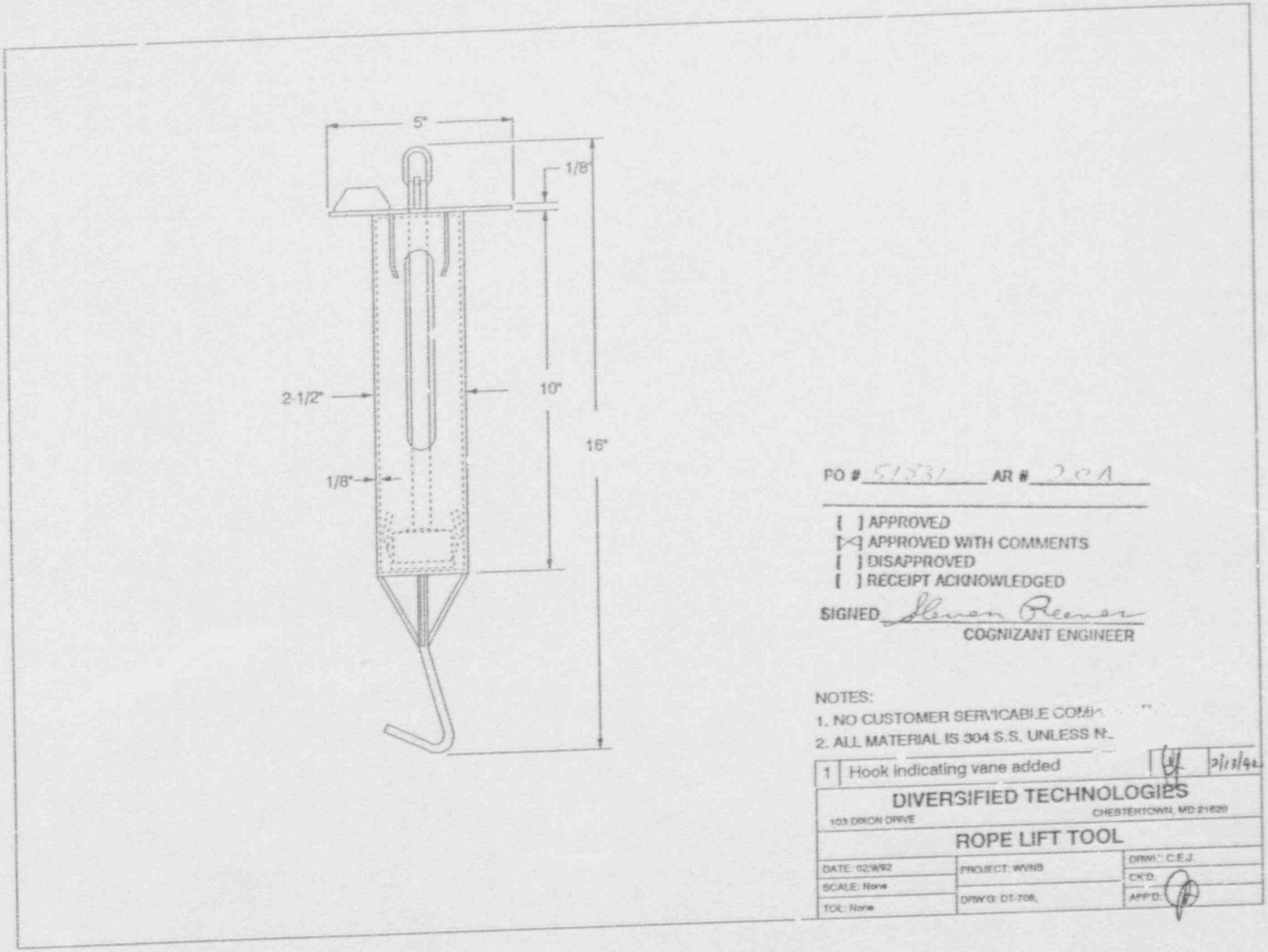

5"
1/8"
2-1/2"
10"
16"
1/8"
PO # 51831 AR # 201
[ ] APPROVED
[X] APPROVED WITH COMMENTS
[ ] DISAPPROVED
[ ] RECEIPT ACKNOWLEDGED
SIGNED Steven Reeves
COGNIZANT ENGINEER
NOTES:
1. NO CUSTOMER SERVICABLE COM
2. ALL MATERIAL IS 304 S.S. UNLESS N
1 Hook indicating vane added 3/13/92
DIVERSIFIED TECHNOLOGIES
103 DIXON DRIVE
CHESTERTOWN, MD 21620
ROPE LIFT TOOL
DATE: 02/9/92
SCALE: None
TOL: None
PROJECT: WVNS
DRW'G: DT-708
DRW: C.E.J.
CK'D:
APP'D:

Figure B.6.1-5

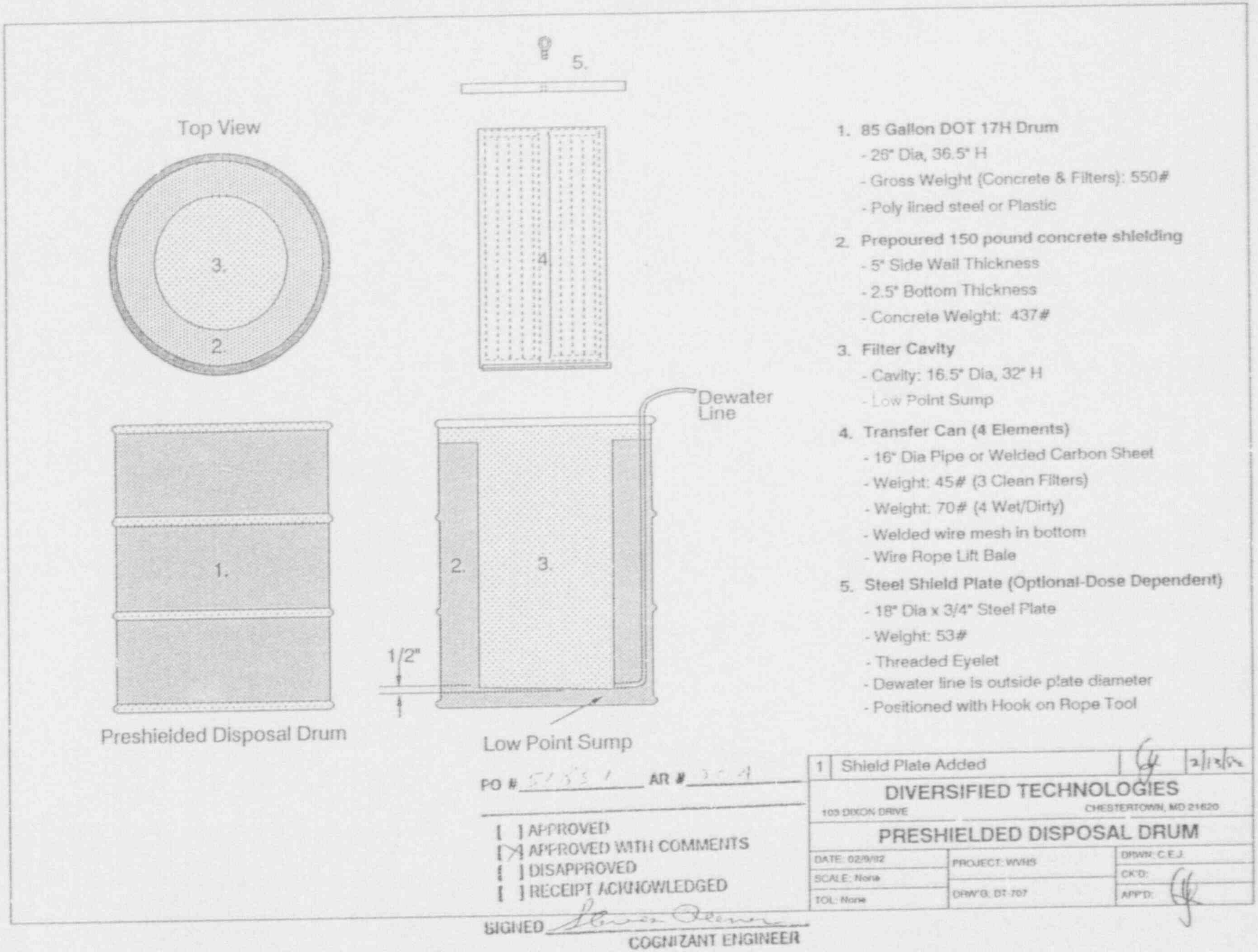

5.
Top View
3.
2.
4
Dewater Line
1.
2.
3.
1/2"
Preshielded Disposal Drum
Low Point Sump
1. 85 Gallon DOT 17H Drum
- 26" Dia, 36.5" H
- Gross Weight (Concrete & Filters): 550#
- Poly lined steel or Plastic
2. Prepoured 150 pound concrete shielding
- 5" Side Wall Thickness
- 2.5" Bottom Thickness
- Concrete Weight: 437#
3. Filter Cavity
- Cavity: 16.5" Dia, 32" H
- Low Point Sump
4. Transfer Can (4 Elements)
- 16" Dia Pipe or Welded Carbon Sheet
- Weight: 45# (3 Clean Filters)
- Weight: 70# (4 Wet/Dirty)
- Welded wire mesh in bottom
- Wire Rope Lift Bale
5. Steel Shield Plate (Optional-Dose Dependent)
- 18" Dia x 3/4" Steel Plate
- Weight: 53#
- Threaded Eyelet
- Dewater line is outside plate diameter
- Positioned with Hook on Rope Tool
1 Shield Plate Added
DIVERSIFIED TECHNOLOGIES
103 DIXON DRIVE
CHESTERTOWN, MD 21620
PRESHIELDED DISPOSAL DRUM
DATE: 02/8/02
PROJECT: WVNS
DRWN: C.E.J.
SCALE: None
CK'D:
TOL: None
DRW'G: DT-707
APP'D:
PO #
AR #
[ ] APPROVED
[X] APPROVED WITH COMMENTS
[ ] DISAPPROVED
[ ] RECEIPT ACKNOWLEDGED
SIGNED
COGNIZANT ENGINEER

Figure B.6.1-6

FRS POOL PHYSICAL INVENTORY

TYPICAL ARRANGEMENT

42	41	40	39	38	37	36	35	34	33	32	31	30	29	28	27	26	25	24	23	22	21	20	19	18	17	16	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1	
	R	R	R	C	C	C	C	C	C	X																																A
	R	R	R	C	C	C	C	C	C	X																																B
	R	R	R	C	C	C	C		C	X																																C
	R	R	R	C	C	C	C		C	X																																D
	R	R	R	C	C	C	C		C	X																																E
	R	R		C	C	C	C		C	X																																F
	R	R		C	C	X	C		C	X																																G
	R	R		C	C		C		C	TZ1																																H
	R	R		C	C		C		C	X																																J
	R	R		C	C		C		C	X																																K
	R	R		C	C		C		C	X																																L
	R	R		C	C		C		C	X																																M
	R	R		C	C		C		C	X																																N
	R	R		C	C		C		C	X																																P
		R		C	C		C			X																																R
		R		C	C		C			X																																S
		R		C	C		C			X																																T
		R		C	C		C			X																																U
		R		C	C		C			X																																V
		R		C	C		C			LT																																W
		R		C	C		C			LT																																X
																																										Y

---------- CANISTER TRANSFER AISLE ----------

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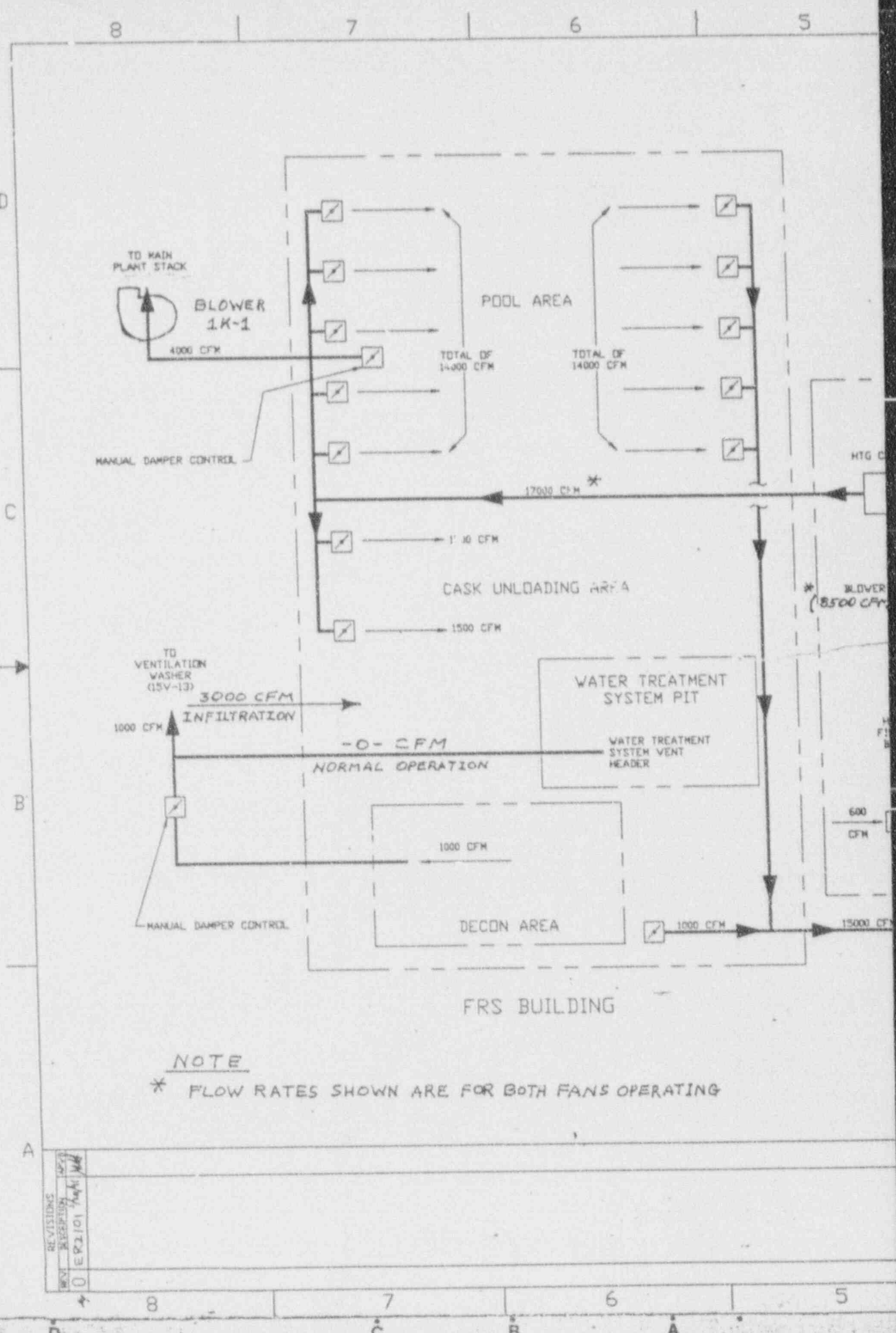
FUEL IDENTIFICATION LEGEND:

R = Rochester Gas & Electric - Ginna
C = Consumer's Power - Big Rock Point
TZ1 = Empty Thoria Rod Canister
x = Empty Canister
LT = Load Test Canister

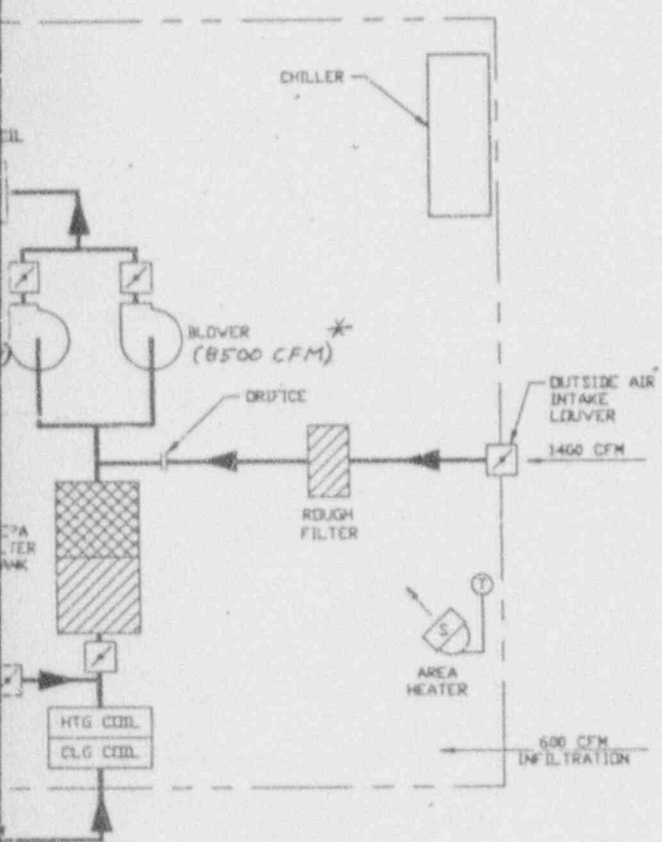
COMMENTS:

FOR INFORMATION ONLY

Figure B.6.2-1



8
7
6
5
D
C
B
A
TO MAIN PLANT STACK
BLOWER 1K-1
4000 CFM
POOL AREA
TOTAL OF 14000 CFM
TOTAL OF 14000 CFM
MANUAL DAMPER CONTROL
HTG C
17000 CFM *
CASK UNLOADING AREA
1500 CFM
BLOWER
(8500 CFM
TO VENTILATION WASHER (15V-13)
3000 CFM INFILTRATION
1000 CFM
-0- CFM
NORMAL OPERATION
WATER TREATMENT SYSTEM PIT
WATER TREATMENT SYSTEM VENT HEADER
600 CFM
1000 CFM
MANUAL DAMPER CONTROL
DECON AREA
1000 CFM
15000 CFM
FRS BUILDING
NOTE
* FLOW RATES SHOWN ARE FOR BOTH FANS OPERATING
REVISIONS
0
ER2101



SI APERTURE CARD

Also Available On Aperture Card

9301070206-07

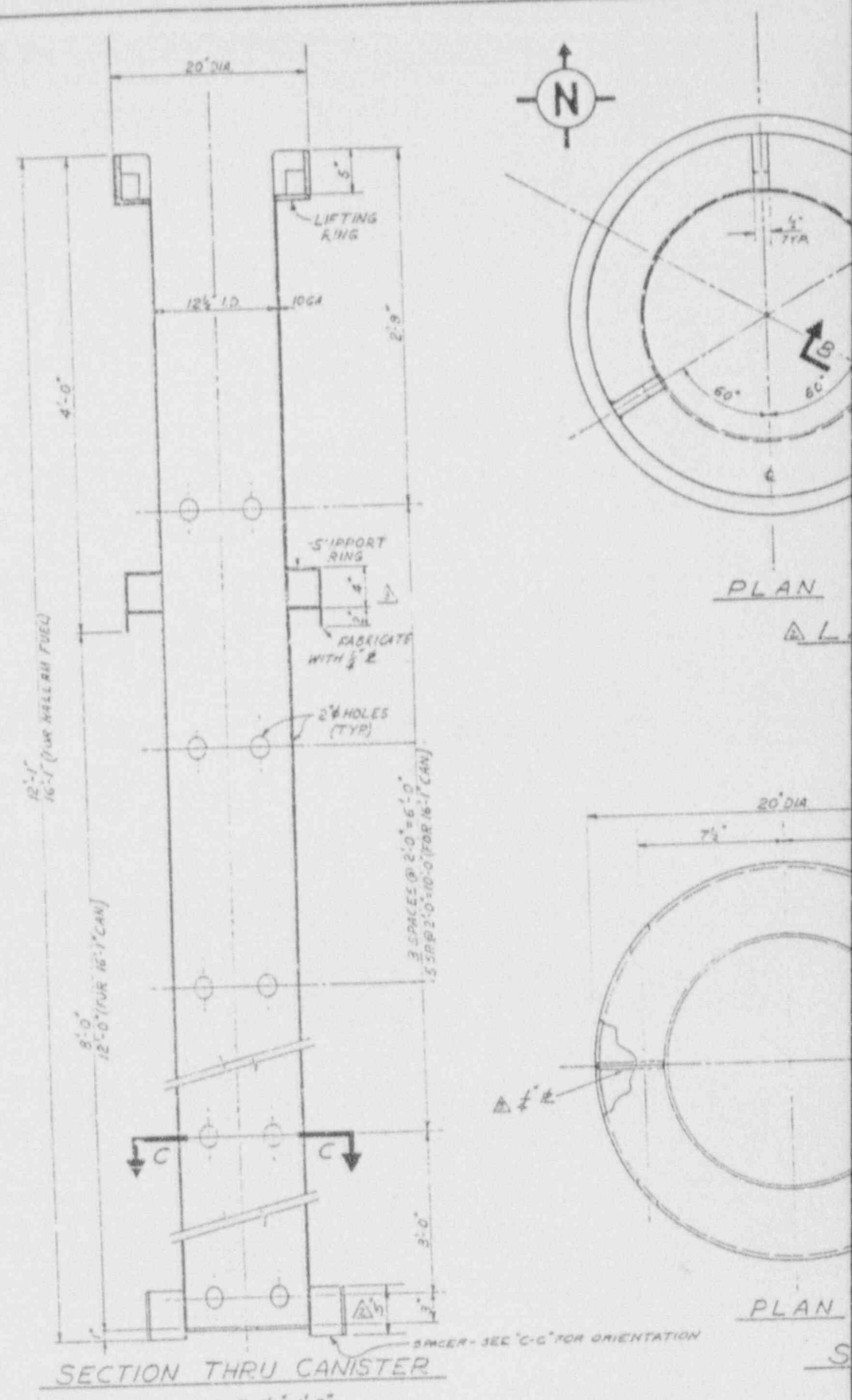
QTY REQD	ITEM NO.	NOMENCLATURE OR DESCRIPTION	PART OR IDENTIFYING NO.	MATERIAL OR SPEC

PARTS LIST

UNLESS OTHERWISE SPECIFIED	DRAFTER V. KOCH, DATE 5/3/91	FOR West Valley Nuclear Services Co.,Inc. West Valley, New York
ALL DIM. IN INCHES		
TOLERANCES-DO NOT SCALE		
± N/A, ± N/A, ± N/A; FRACTIONS ± N/A		FRS VENTILATION SYSTEM DIAGRAM
NEXT ASSEMBLY 900D-2610 900D-2611		DWG SIZE D; SCALE NONE; DWG NO. 900D-4250; REV 0; SHEET 1 OF 1

CAD DRAWING-DO NOT REVISE THIS ORIGINAL

Figure B.7.4



SECTION THRU CANISTER

SCALE: 1½" = 1'-0"

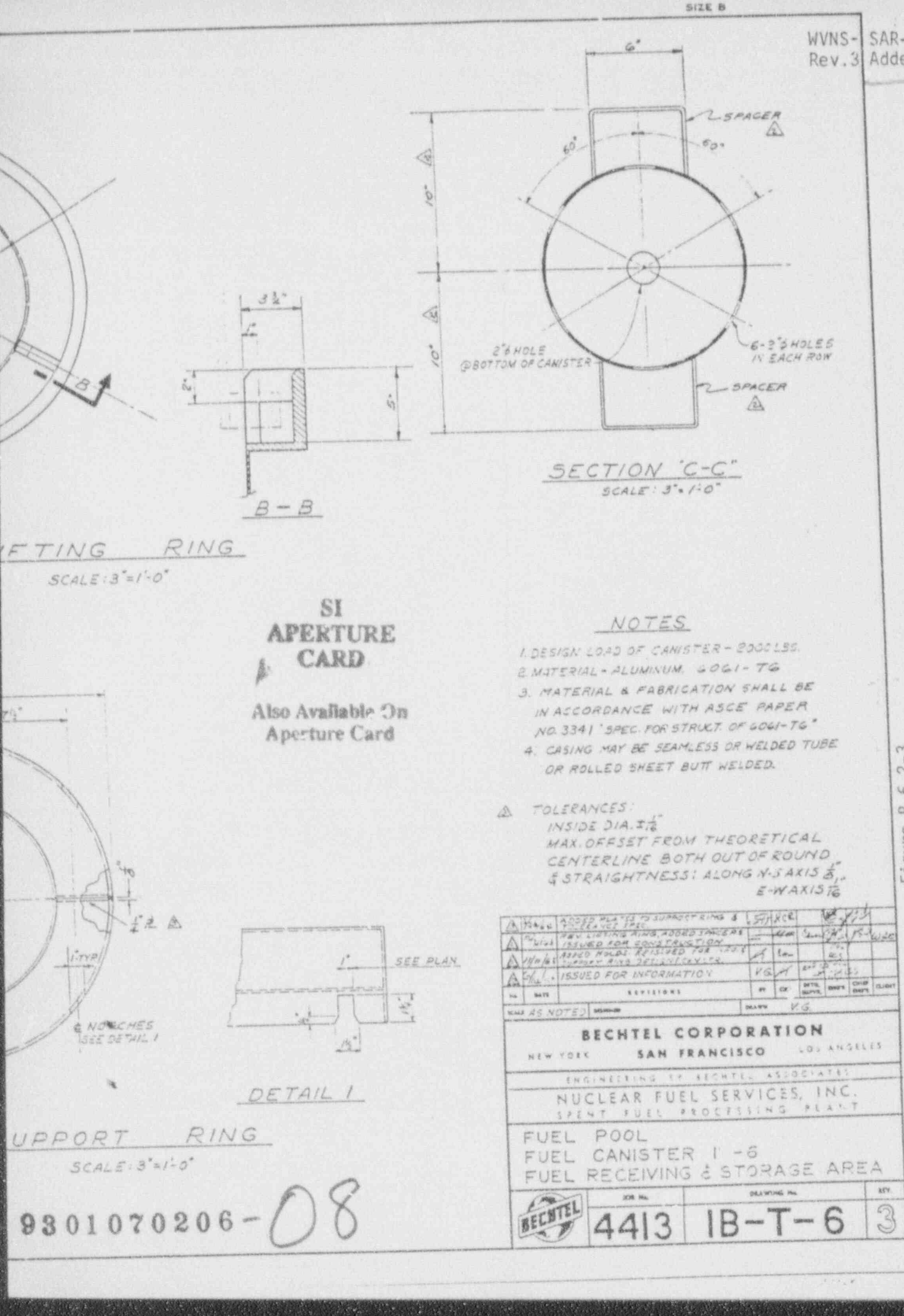
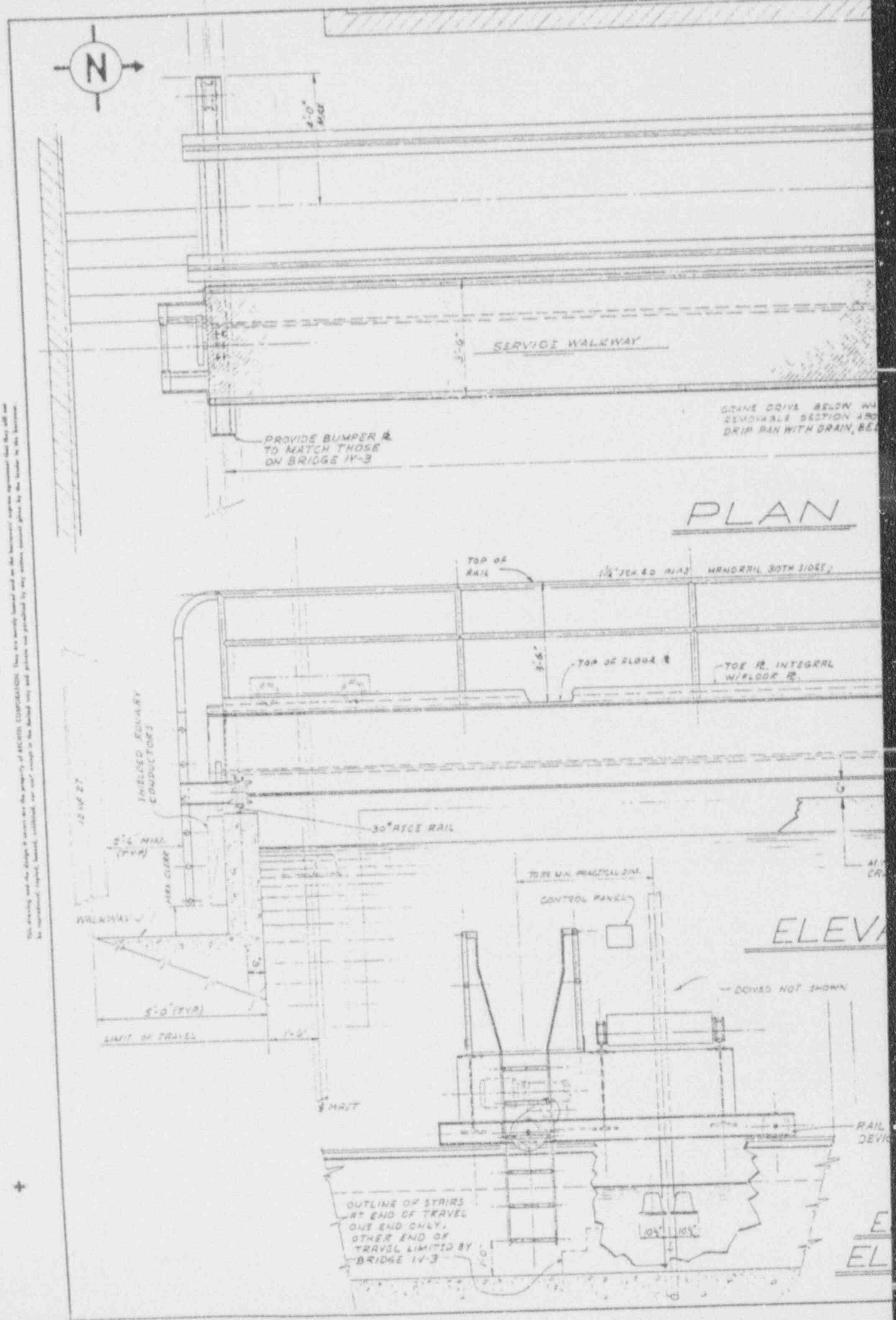
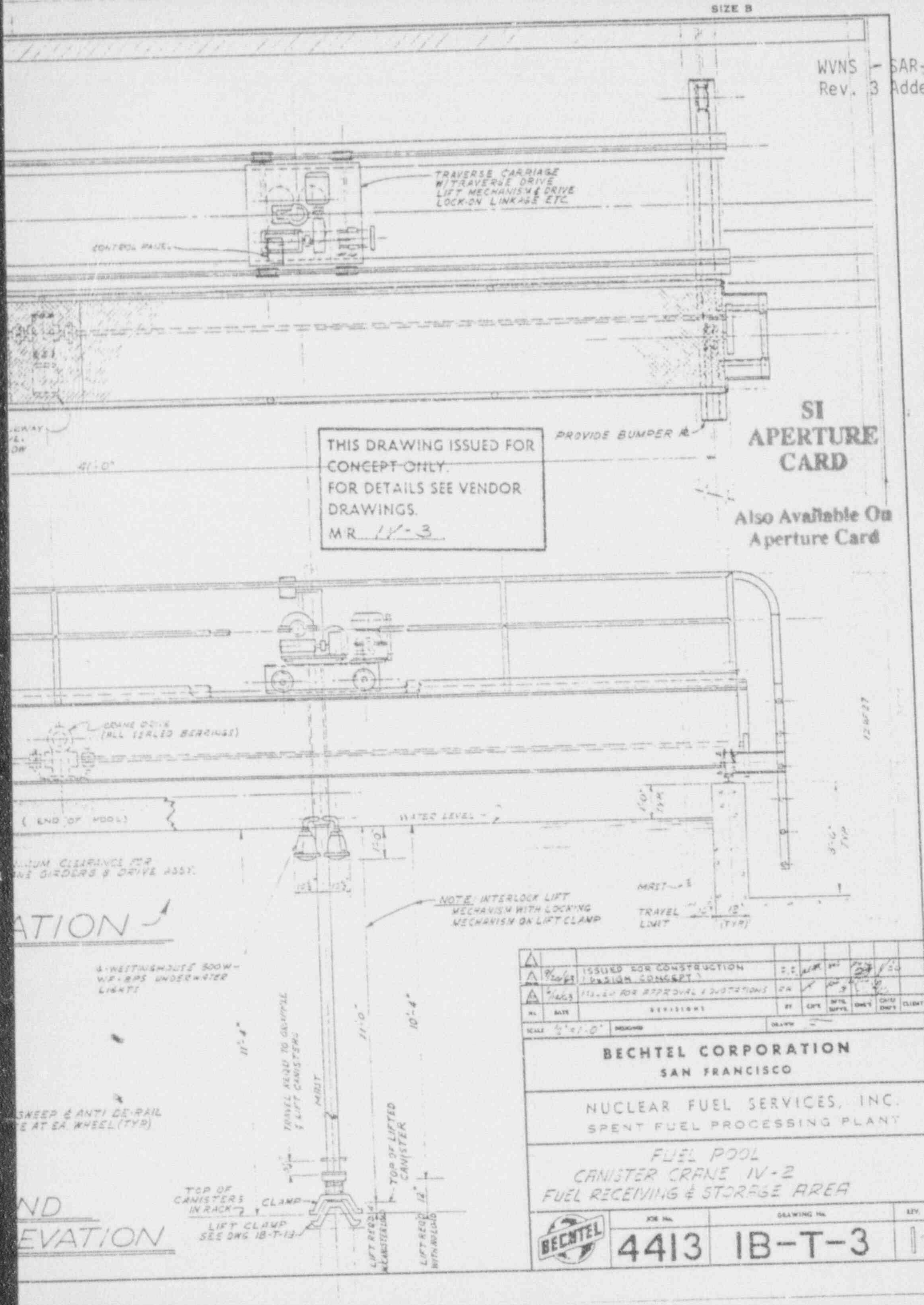


Figure B.6.2-3

9301070206-08


N
SERVICE WALKWAY
PROVIDE BUMPER PL
TO MATCH THOSE
ON BRIDGE IV-3
PLAN
TOP OF RAIL
TOP OF FLOOR PL
TOE PL INTEGRAL W/FLOOR PL
SHIELDED RUNWAY CONDUCTORS
30# ASCE RAIL
WALKWAY
CONTROL PANEL
DRIVES NOT SHOWN
5'-0" (TYP)
LIMIT OF TRAVEL
OUTLINE OF STAIRS
AT END OF TRAVEL
OUR END ONLY.
OTHER END OF
TRAVEL LIMITED BY
BRIDGE IV-3

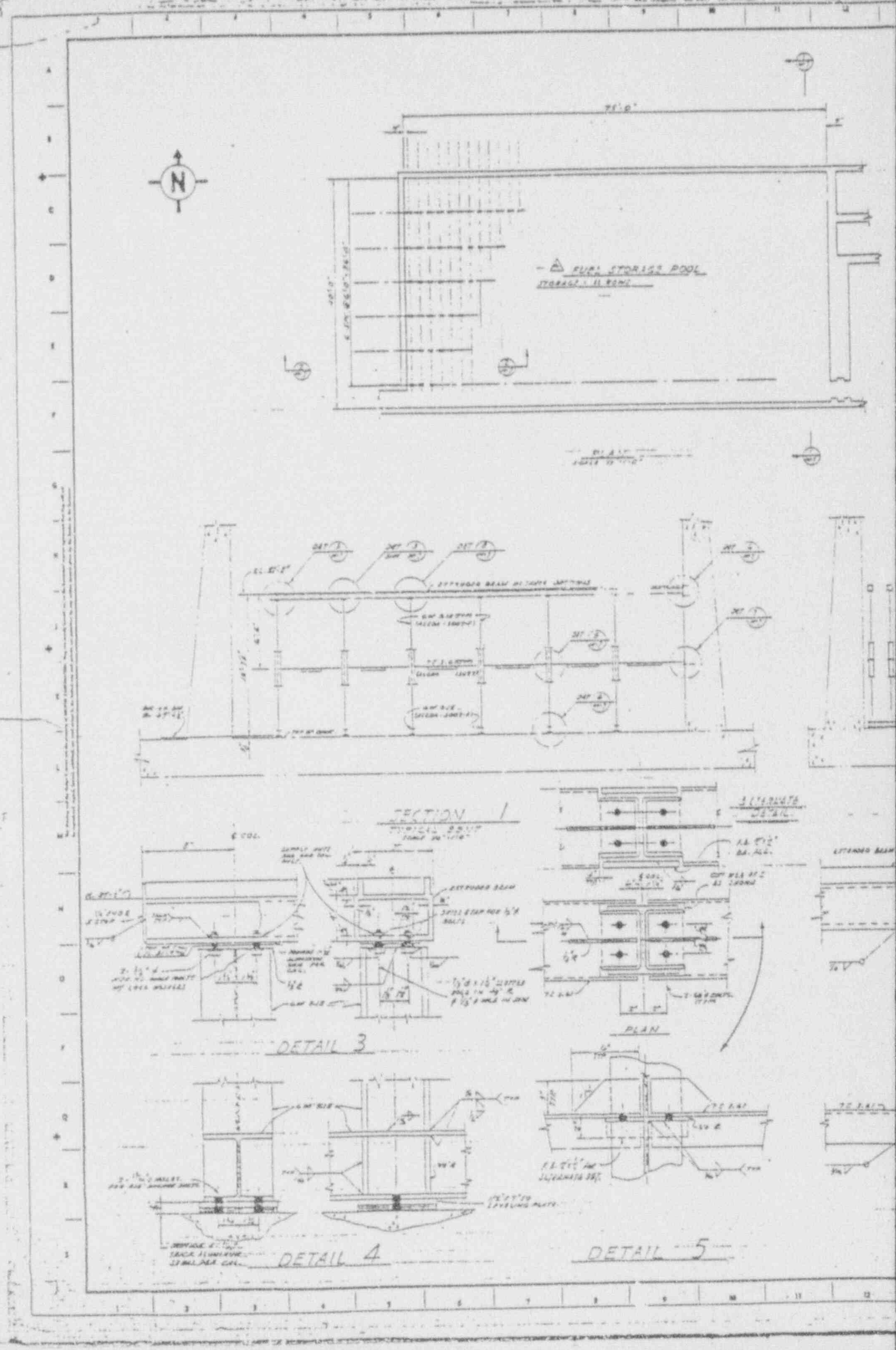


SI
APERTURE
CARD

Also Available On
Aperture Card

Figure B.6.2-2

9301070206-09


N
75'-0"
FUEL STORAGE POOL
PLAN
SECTION 1
EXTRUDED BEAM
PLAN
DETAIL 3
DETAIL 4
DETAIL 5

WVNS- SAR-002
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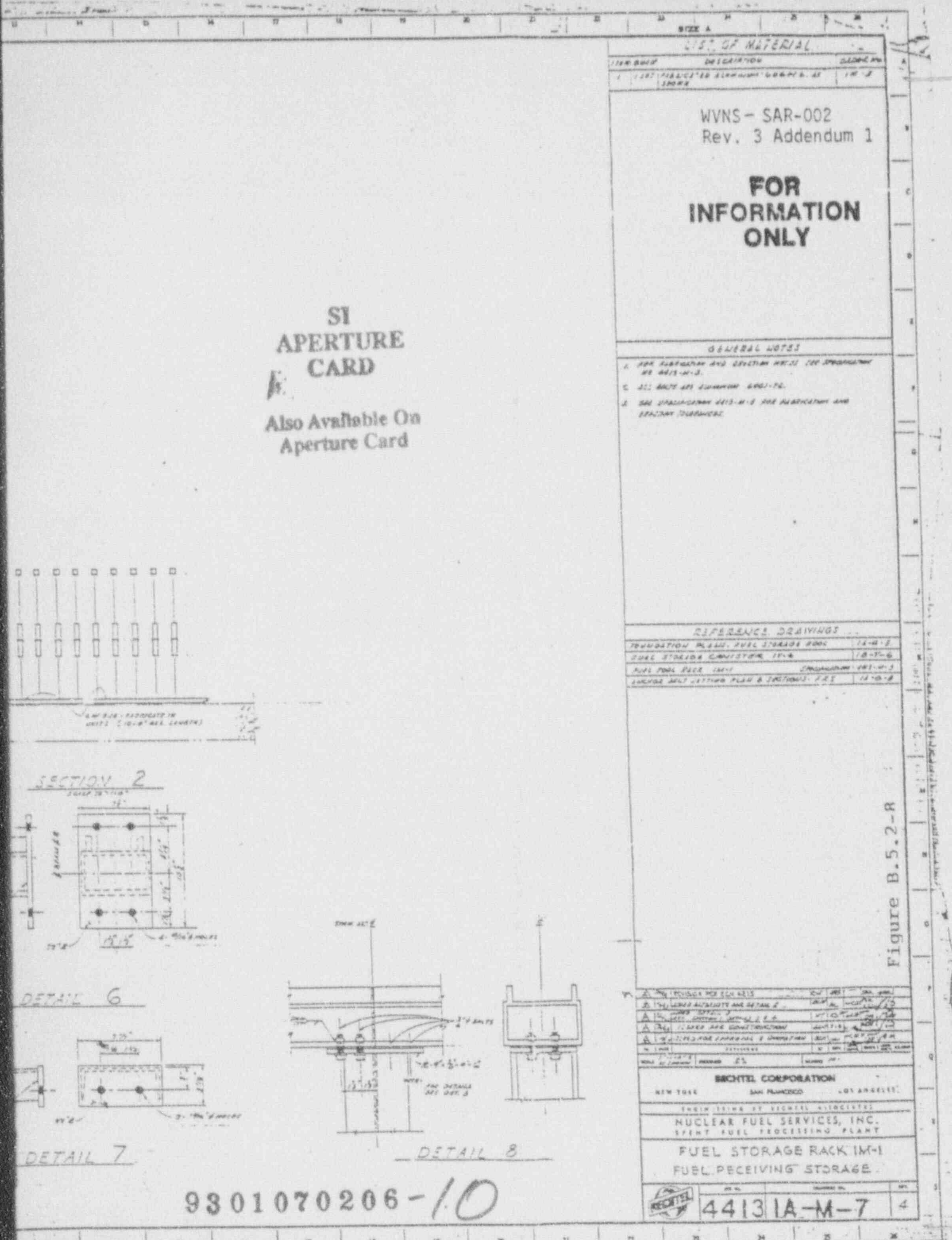
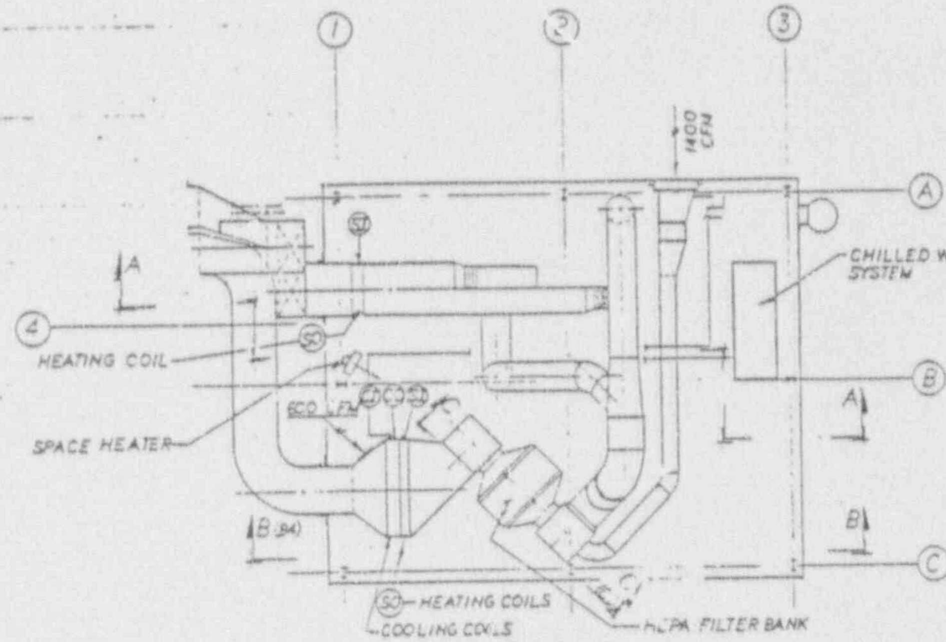
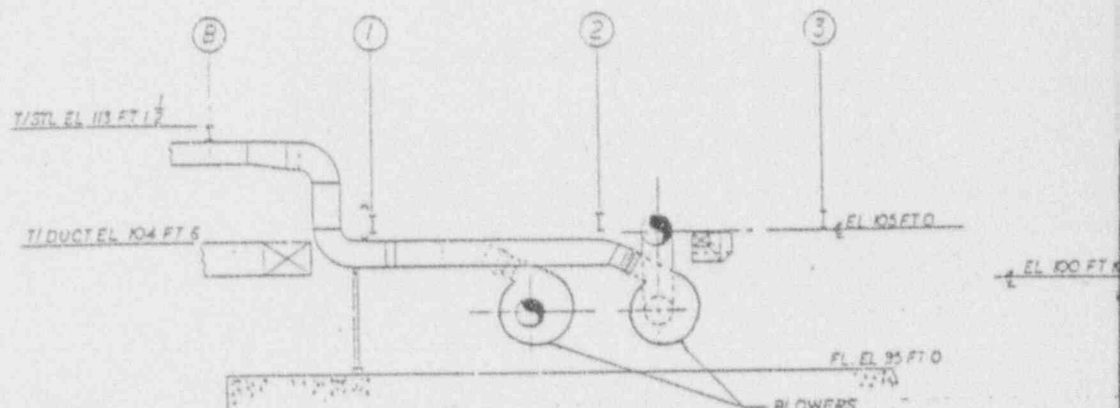


Figure B.5.2-R

9301070206-10



FLOOR PLAN



SECTION A-A

REVISIONS

FOR INFORMATION ONLY

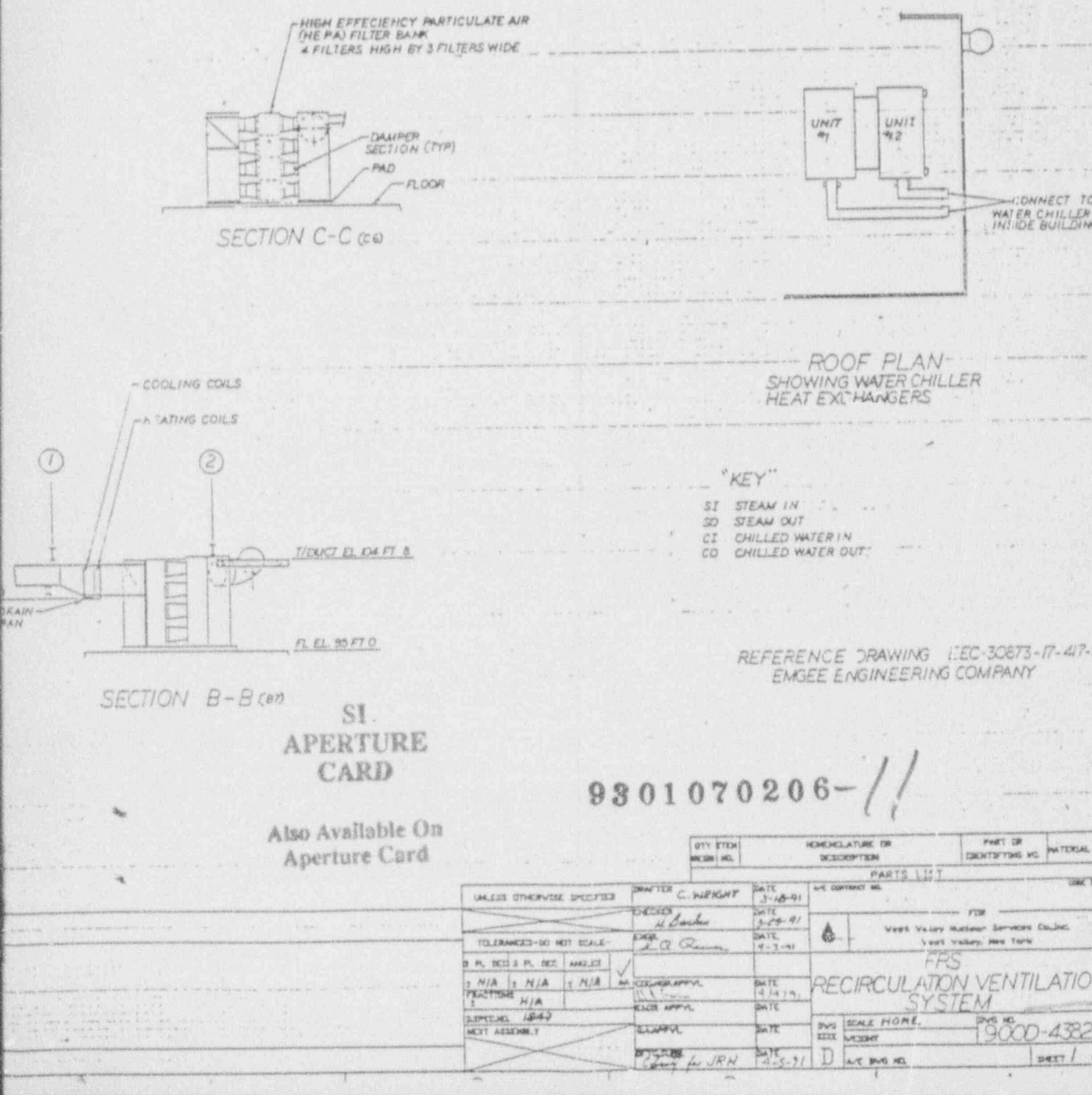
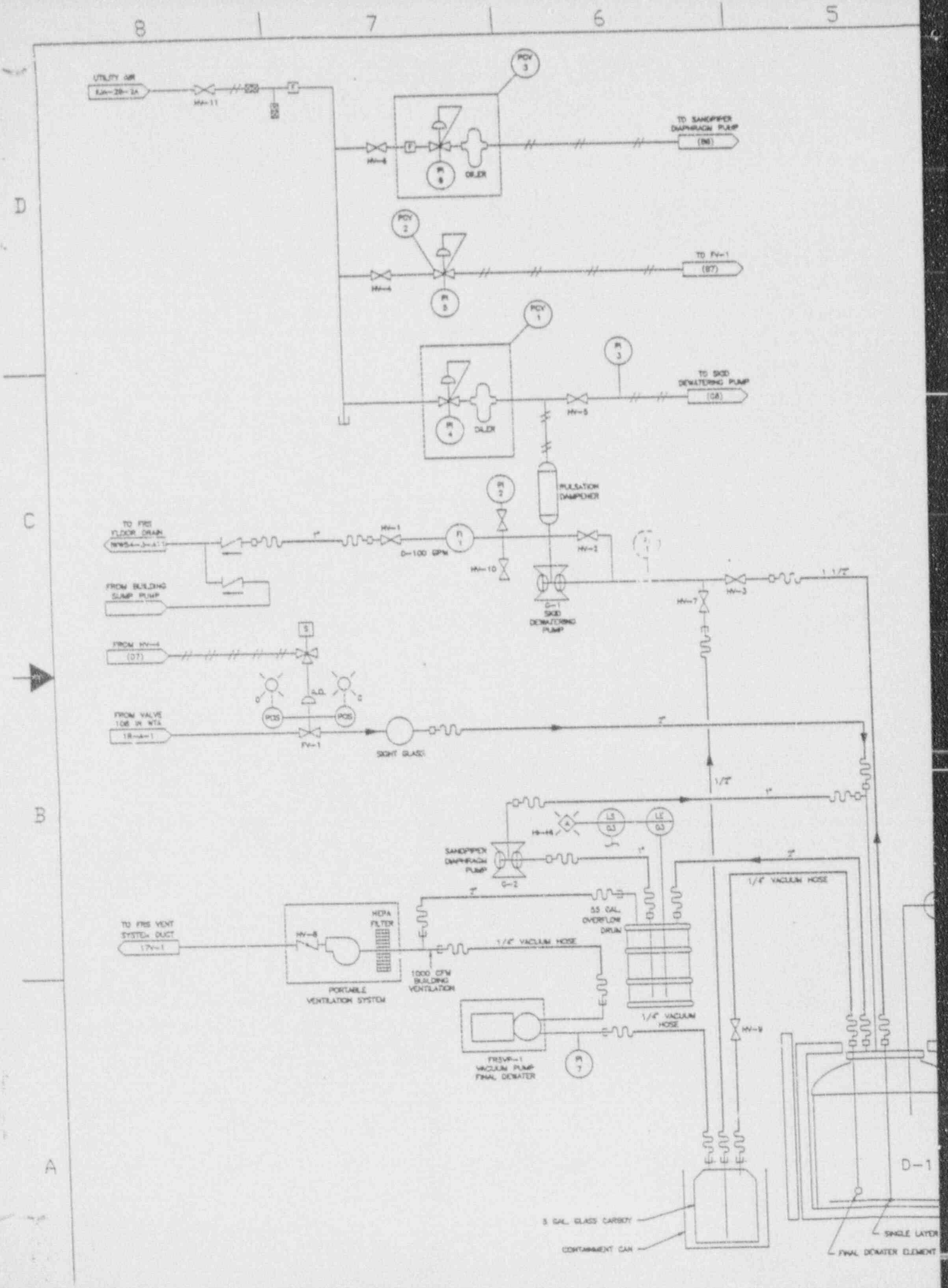


Figure B.5.2-7

SI
APERTURE
CARD

Also Available On
Aperture Card

9301070206-11



8
7
6
5
D
C
B
A
UTILITY AIR
HV-11
PCV
3
TO SANDPIPER
DIAPHRAGM PUMP
(B6)
HV-6
DRIER
PCV
2
TO FV-1
(B7)
HV-4
PCV
1
PI
3
TO SKID
DEWATERING PUMP
(C8)
HV-5
DRIER
PULSATION
DAMPENER
PI
2
TO FRS
FLOOR DRAIN
HV-1
PI
1
HV-2
1"
0-100 GPM
HV-10
FROM BUILDING
SUMP PUMP
SKID
DEWATERING
PUMP
HV-7
HV-3
1 1/2"
FROM HV-4
(D7)
S
A.O.
POS
POS
FROM VALVE
108 IN WTA
FV-1
SIGHT GLASS
2"
1/2"
1"
LS
LE
SANDPIPER
DIAPHRAGM
PUMP
2"
1/4" VACUUM HOSE
HEPA
FILTER
55 GAL
OVERFLOW
DRUM
TO FRS VENT
SYSTEM DUCT
HV-8
1/4" VACUUM HOSE
1000 CFM
BUILDING
VENTILATION
PORTABLE
VENTILATION SYSTEM
1/4" VACUUM
HOSE
HV-9
FRSVP-1
VACUUM PUMP
FINAL DEWATER
PI
7
D-1
5 GAL GLASS CARBOY
CONTAINMENT CAN
SINGLE LAYER
FINAL DEWATER ELEMENT

900D-2289

REVISIONS

ZONE	REV	DESCRIPTION	DATE	APPROVED
	0	ER1503		

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FOR INFORMATION ONLY

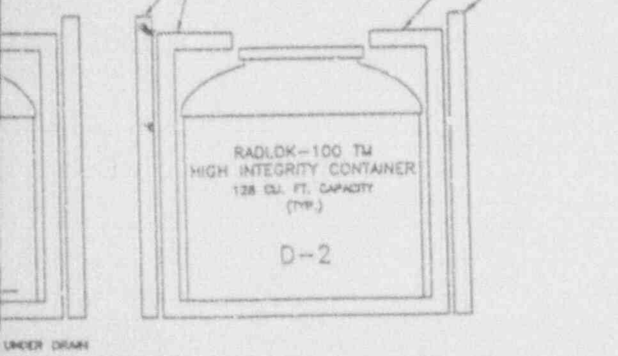
SI APERTURE CARD

Also Available On Aperture Card

Figure B.5.2-6

LEGEND:
UTILITY AIR LINE
HOSE
KAMLOK FITTING

9301070206-12



NOTES:

1. WHEN HIC D-1 IS IN OPERATION HIC D-2 IS THE OFF-LINE SPARE. CONVERSELY, HIC D-1 IS THE OFF-LINE SPARE WHEN HIC D-2 IS IN OPERATION.

QTY REQD	ITEM NO.	NOMENCLATURE OR DESCRIPTION	PART OR IDENTIFYING NO.	MATERIAL OR SPEC

PARTS LIST

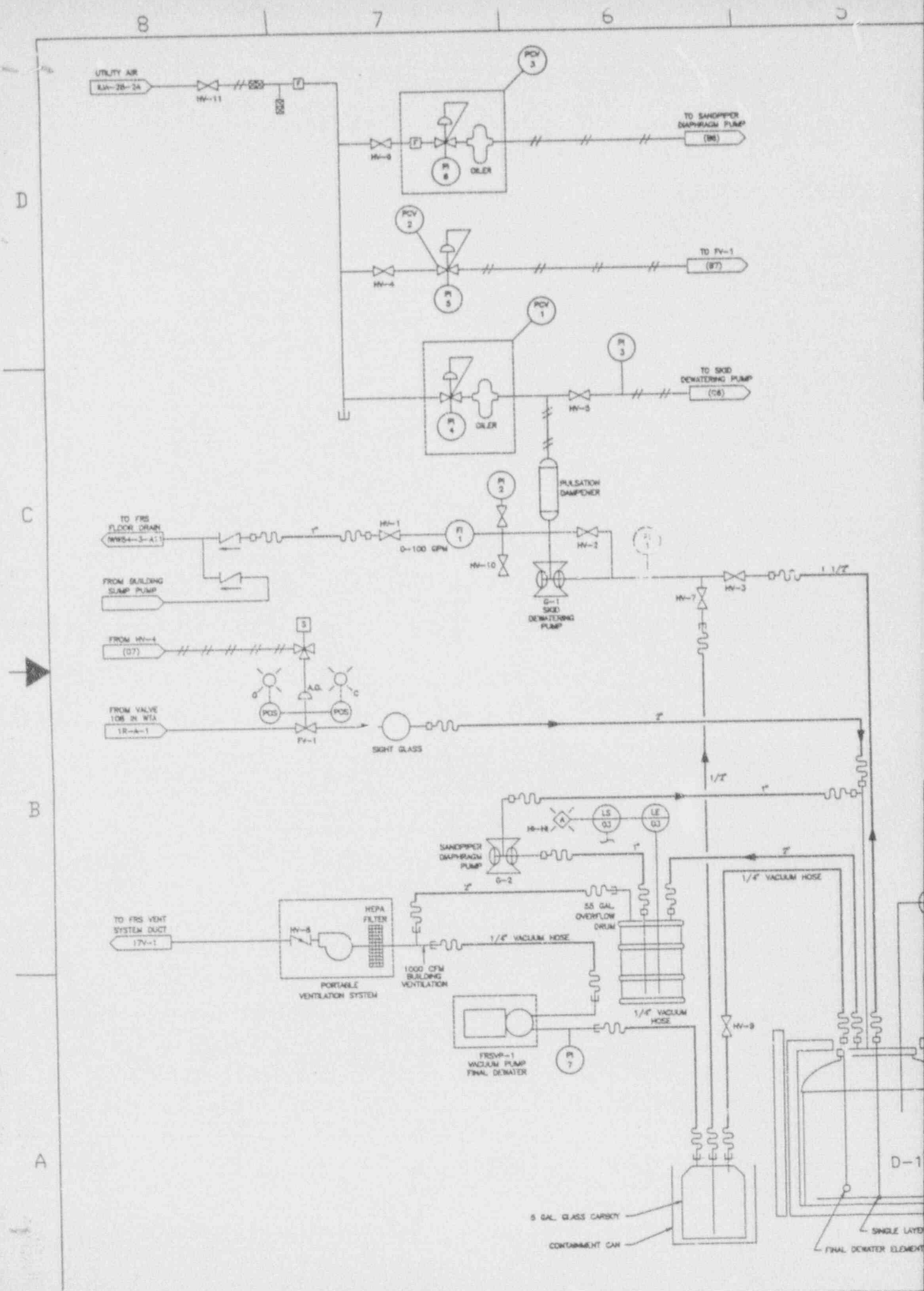
UNLESS OTHERWISE SPECIFIED
TOLERANCES-DO NOT SCALE
2 PL DEC | 3 PL DEC | ANGLES
FRACTIONS
D.SPEC.NO. 1274
NEXT ASSEMBLY

DRAFTER LINDA KOMENDA DATE 5/11/88
CHECKER DATE
ENGR. DATE
COG.ENGR.APPVL. DATE
ENGR. APPVL. DATE
QA APPVL. DATE
DATE

A/E CONTRACT NO.
FOR West Valley Nuclear Services Co.,Inc. West Valley, New York

FRS
RADWASTE TREATMENT SYSTEM
PIPING AND INSTRUMENT DIAGRAM

DWG SIZE D | SCALE NONE | WEIGHT | DWG NO. 900D-2289 | REV 0



8
7
6
D
C
B
A
UTILITY AIR
HV-11
PCV 3
TO SANDPIPER DIAPHRAGM PUMP
HV-6
PI 6
OILER
PCV 2
TO FV-1
(07)
HV-4
PI 5
PCV 1
PI 3
TO SKID DEWATERING PUMP
(08)
HV-5
PI 4
OILER
PULSATION DAMPENER
PI 2
TO FRS FLOOR DRAIN
HV-1
FI 1
0-100 GPM
HV-2
HV-10
G-1 SKID DEWATERING PUMP
HV-7
HV-3
1 1/2"
FROM BUILDING SUMP PUMP
FROM HV-4
(07)
S
A.O.
POS
POS
FROM VALVE 10B IN WTA
1R-A-1
FV-1
SIGHT GLASS
2"
1/2"
1"
LS
LE
HI-HI
SANDPIPER DIAPHRAGM PUMP
G-2
1"
2"
1/4" VACUUM HOSE
2"
55 GAL. OVERFLOW DRUM
TO FRS VENT SYSTEM DUCT
17V-1
HV-8
HEPA FILTER
PORTABLE VENTILATION SYSTEM
1000 CFM BUILDING VENTILATION
1/4" VACUUM HOSE
1/4" VACUUM HOSE
FRSVP-1 VACUUM PUMP FINAL DEWATER
PI 7
HV-9
5 GAL. GLASS CARBOY
CONTAINMENT CAN
D-1
SINGLE LAYE
FINAL DEWATER ELEMENT

ZONE	REV	REVISIONS DESCRIPTION	DATE	APPROVED
	0	ER1503	[illegible]	[illegible]

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SI
APERTURE
CARD

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Aperture Card

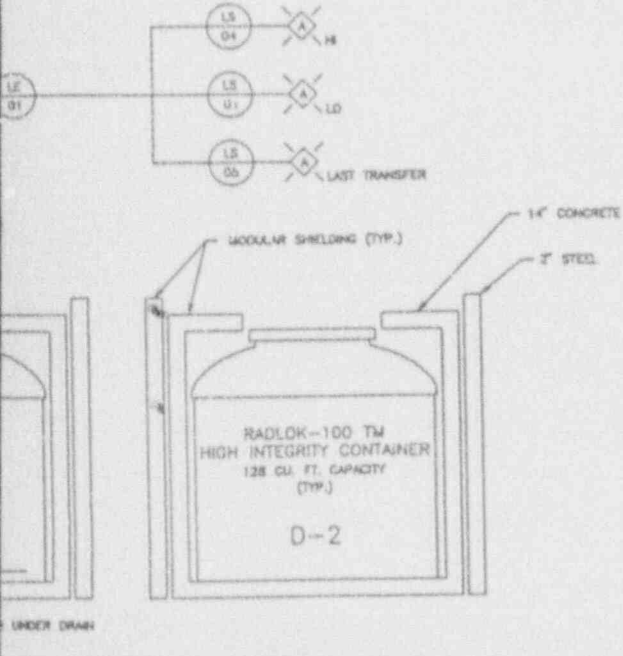
Figure B.5.2-5

LEGEND–

UTILITY AIR LINE

HOSE

KAMLOK FITTING



9301070206–13

NOTES–

1. WHEN HIC D–1 IS IN OPERATION HIC D–2 IS THE OFF–LINE SPARE. CONVERSELY, HIC D–1 IS THE OFF–LINE SPARE WHEN HIC D–2 IS IN OPERATION.

QTY REQD	ITEM NO.	NOMENCLATURE OR DESCRIPTION	PART OR IDENTIFYING NO.	MATERIAL OR SPEC

PARTS LIST

UNLESS OTHERWISE SPECIFIED
TOLERANCES–DO NOT SCALE
2 PL. DEC ± | 3 PL. DEC ± | ANGLES ±
FRACTIONS ±
D.SPEC.NO. 1274
NEXT ASSEMBLY

DRAFTER LINDA KOMENDA DATE 5/11/88
CHECKER [signature] DATE [illegible]
ENGR. [signature] DATE [illegible]
COG.MGR.APPVL. T.HUGHES DATE 11-4-88
S.&.S APPVL. [signature] DATE 11/15/88
Q.A.APPVL. [signature] DATE 11-15-88
DEPT.SUPV. [signature] DATE 11/2/88

A/E CONTRACT NO.

FOR
West Valley Nuclear Services Co., Inc.
West Valley, New York

FRS
RADWASTE TREATMENT SYSTEM
PIPING AND INSTRUMENT DIAGRAM

DWG SIZE D | SCALE NONE | WEIGHT | A/E DWG NO.
DWG NO. 900D-2289 | REV 0
SHEET 1 OF 1