B.7.0 WASTE CONFINEMENT AND MANAGEMENT

B.7.1 Waste Management Criteria

Radioactive wastes resulting from WVDP operations include gaseous, liquid and solid low-level radioactive wastes (LLRWs), liquid high-level waste, low-level radioactive mixed waste, and solid TRU/suspect TRU waste. In addition, both hazardous and nonhazardous nonradioactive (i.e., industrial and sanitary) wastes are generated as a result of WVDP activities. Waste handling and processing facilities have been designed to ensure environmental effluent releases are maintained well within discharge guidelines given in DOE Order 5400.5, *Radiation Protection of the Public and the Environment*, and 40 CFR 261, *Identification and Listing of Hazardous Waste*.

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

Note: *DOE Order 5480.23 requires documentation of safety assessments of nuclear facility operations and facilities as well as waste management activities at these facilities. IRTS and Vitrification systems have been designed and constructed for the processing and solidification of high-level wastes. Consequently, a necessary distinction has been made, for the purposes of this SAR, between the waste currently contained in Tank 8D-2 that serves as feed solution to the IRTS and those byproduct streams generated during site operations which are ultimately treated at, or stored in, the LLW2 or Lag Storage facilities. Discussion of high-level waste characteristics and IRTS facilities and processes are contained in Chapters 4 through 6 of this SAR while discussion of IRTS byproduct waste stream handling and treatment are the subject of the sections of this chapter.*

B.7.2 Low-Level Radioactive and Low-Level Radioactive Mixed Wastes

Low-level radioactive wastes (LLRWs) at the WVDP result from IRTS, Main Plant and LWTS operations, as well as from decontamination, maintenance, and construction activities. These wastes include gaseous, liquid and solid LLRW and solid TRU/suspect TRU waste. Contaminated lead metal is the primary source of low-level radioactive mixed waste at the WVDP. This material is currently stored in the Interim Waste Storage Facility (IWSF) and Lag Storage Building as discussed in Section B.7.8. Small quantities of liquid low-level radioactive mixed wastes are generated during analyses in the WVDP Analytical and Process Chemistry laboratories.

Liquid low-level radioactive mixed waste may be stored at the IWSF, Lag Storage Building, or LSA 3 or 4 prior to final disposition.

Airborne radioactive contamination within Project facilities is filtered and exhausted to the environment through the use of building ventilation systems. Gaseous effluents from these systems are monitored per the requirements of the site NESHAPS permit.

Liquid LLRW waste generated in the IRTS/Main Plant is comprised of contaminated water resulting from area or equipment decontamination, system flush water, filter backwash, and laundry operations. Wastewater with a gross beta concentration less than 5E-3 µCi/mL (1.85E2 Bq/mL) is directed to the Low-Level Waste Treatment Replacement Facility (LLWTF) which uses an ion exchange process for liquid radioactive waste decontamination.

The WVDP is currently utilizing the NRC waste classification system prescribed in 10 CFR 61 for Class A, B, and C wastes. Based on this classification system, the primary form of solid LLRWs generated during IRTS/Main Plant operations is Class A waste. This waste is generally compactable material consisting of anticontamination clothing, bags, paper products, rags, analytical sample bottles and other miscellaneous items. Other solid LLRWs include spent ventilation filters, dewatered sludge and resin from LLWTF operations, contaminated wood products, small diameter piping and sheet metal, and failed processing components which have been removed from radioactive service and overpacked before disposal. A representative summary of typical waste, including LLRW, stored at the WVDP is presented in Table B.7.7-1. This table is an example and is provided for information only.

TRU/suspect TRU waste at the WVDP was primarily generated from plant decontamination efforts. At present, there are no major on-going decontamination projects and therefore no significant quantities of TRU/suspect TRU waste being generated. Per the requirements of DOE Order 435.1, *Radioactive Waste Management,* only TRU waste categorized as defense waste is acceptable for disposal at the Waste Isolation Pilot Project (WIPP). Since WVDP TRU waste resulted from reprocessing of spent fuels from both commercial and defense sources, TRU waste is not presently scheduled to be shipped to WIPP. The WVDP has developed a TRU Waste Acceptance Criteria (WAC) which invokes some of the elements of the WIPP WAC program. Discussion of this program is given in WVDP-030, *TRU Waste Certification Program Plan*.

B.7.3 Nonradiological Wastes

Nonradiological wastes generated at the WVDP are of the following basic types:

[1] Hazardous liquid and solid wastes (oils and wipes from maintenance activities, etc);

[2] Nonhazardous, solid wastes (construction and demolition debris, non-construction debris, scrap equipment, maintenance wastes, office trash, packing material, etc.);

[3] Nonhazardous, nonradioactive effluent (sewage, utility room effluent, etc.).

Nonhazardous, nonradioactive solid wastes are disposed of off-site at a licensed landfill facility.

Project effluents are regulated by the New York State Department of Environmental Conservation (NYSDEC) for nonradiological parameters. The combined liquid effluents from the LLWTF (Section B.7.5) and waste water (sanitary sewage) treatment facility (Section B.5.4.7) are monitored to assure compliance with discharge limits identified in the State Pollutant Discharge Elimination System (SPDES) Permit. These waste streams are also monitored for radioactivity.

A small amount of hazardous waste is generated at the WVDP primarily as a result of maintenance, analytical and printing activities. There is no on-site disposal of hazardous waste at the WVDP. Hazardous waste is shipped off-site for treatment and disposal by licensed and approved transporters to permitted commercial treatment, storage, and disposal facilities.

B.7.4 Off-Gas Treatment and Ventilation

Airborne effluents from IRTS and Main Plant equipment are decontaminated by off-gas treatment systems prior to discharge to the environment. Off-gas treatment for vessels in the Waste Tank Farm is provided by the Waste Tank Farm Ventilation System (WTFVS) while the Vessel Off-Gas System (VOG) provides ventilation for vessels in the Main Plant, including vessels associated with the LWTS. Ventilation air from both the Waste Tank Farm Ventilation System and Vessel Off-Gas system is exhausted to the Main Plant stack.

WASTE TANK FARM VENTILATION SYSTEM

The original Waste Tank Farm Ventilation System (WTFVS) provides ventilation to Tanks 8D-1, 8D-2, 8D-3, and 8D-4. The WTFVS is an existing facility constructed in the early 1960s as part of the original reprocessing plant. This ventilation system was not seismically designed or qualified to a design-basis earthquake. If this equipment does fail and/or on-site backup power is lost, negative pressure on the HLW tanks can be maintained by manually placing the tanks onto the Permanent Ventilation System (PVS).

VESSEL OFF-GAS SYSTEM

The Vessel Off-Gas system provides ventilation of exhaust gases from the LWTS evaporator and condenser, as well as a number of other vessels in the LWTS and Main Plant. A summary of vessels ventilated by the VOG is given in Table B.7.4-2.

B.7.4.1 Operating Characteristics

Off-gas treatment and ventilation systems in the Main Plant utilize HEPA filters to ensure adequate removal of radioactive particulates in effluent air streams. HEPA filters are inspected prior to installation and are in-place leak tested at installation and annually thereafter to ensure acceptable operation. HEPA filters used at the WVDP must meet requirements prescribed by the Department of Energy (U.S. Department of Energy, 1988).

WASTE TANK FARM VENTILATION SYSTEM

During normal operation of the Waste Tank Farm Ventilation System air is removed from the tanks at approximately 0.35 m^3/s (750 cfm). Supply air to the tanks is provided from inleakage. Off-gas from Tanks 8D-1 and 8D-2 passes through a condenser to remove water vapor. Off-gas from Tanks 8D-3 and 8D-4 is processed through a caustic scrubber and combines with flow from Tanks 8D-1 and 8D-2. The combined air stream passes through a knock-out drum and demister to remove entrained liquid. Condensate is returned to Tank 8D-2. The air stream is then passed through a heater prior to HEPA filtration and exhaust. Filters and blowers in the Waste Tank Farm Ventilation System are provided with redundant spares connected in parallel to provide exhaust in the event of off-gas equipment failure. Following off-gas treatment, exhaust air is combined in the Main Plant stack with effluent from other activities. The configuration of equipment in the Waste Tank Farm Ventilation System is depicted in Figure B.5.4-1.

During Sludge Mobilization System sludge mixing activities, each mobilization pump adds approximately 130 kW of heat to the Tank 8D-2 contents. The total heat input rate to the tank, with six pumps running, is approximately 790 kW. This is in addition to the 50 kW (70 hp) generated by radioactive decay of the remaining unwashed sludge. Assuming that all the heat generated by the pumps produces water vapor, approximately 1,700 L/h (6.6 gpm) of vapor would pass through the WTFVS. Since the condensers have a design capacity of 4,700 L/h (21 gpm), the water vapor generated from the sludge mixing operation should be condensed efficiently.

VESSEL OFF-GAS SYSTEM

Off-gas ventilated from LWTS and Main Plant vessels passes through the VOG Condenser 6E-3 where it is cooled. Airflow is then through Tank 6D-6 where the condensate generated in 6E-3 is collected and gravity-fed to Tank 6D-3. The stream then passes to the VOG Scrubber 6C-3. During normal operation, the VOG Scrubber contains a minimum level of water and the recirculation pump is off. If off-gas scrubbing is required however, 6C-3 may be charged with a scrubber solution with the off-gas flowing through a cascade of liquid to remove and/or neutralize any chemical vapors. Off-gas leaving Scrubber 6C-3 passes into the VOG cyclone 6V-1 to remove any entrained water and then to heater 6E-4 to raise the dew point temperature prior to filtration.

There are two upstream-filter/fan trains ducted in parallel which provide filter/fan train redundancy. The two final filter trains down stream of the blowers are ducted such that either train can be used with either fan (contrary to the upstream filter ducting configuration). Equipment arrangement in the VOG system is shown in Figure B.5.4-1. Both upstream and downstream filter trains are arranged such that filter change-outs can be accomplished without interrupting system air flow. Upstream (primary) HEPA filters and blowers are located in the Off-Gas Blower Room (OGBR), adjacent to the Off-Gas Cell. Downstream (secondary) HEPA filters are located on the Off-Gas Aisle (OGA) roof. After leaving the final filters the gases are exhausted at a rate of 0.28 m^3/s (600 cfm) to the Main Plant stack where they are discharged to the atmosphere.

The primary Vessel Off-Gas filters located in the Off-Gas Blower Room consist of a roughing filter and HEPA filter in a common frame. The frame is stainless steel and has a 1.3 cm x 1.3 cm (0.5 in x 0.5 in) stainless steel screen on the downstream face. This design precludes bulk filter media from falling into the discharge plenum in the event the filter media fails.

B.7.4.2 Safety Criteria and Assurance

IRTS/Main Plant off-gas treatment and ventilation systems maintain redundant HEPA filters and blowers to ensure ventilation system operability during abnormal operating conditions. Instrumentation has been provided to monitor the integrity of ventilation system filters. A summary of off-gas filter monitoring instrumentation is given in Table B.7.4-1.

Treatment system efficiency is continuously monitored through sampling and monitoring of exhaust air in the Main Plant stack. Effluent air samples are collected weekly and analyzed in the WVDP Environmental Laboratory for gross alpha/beta and tritium activity. In addition, weekly gamma isotopic analyses are performed if gross activity rises significantly. Weekly filter samples are composited quarterly and analyzed for specific radionuclides of interest. The airborne effluent monitoring program is described completely in Section B.8.3.

B.7.5 Liquid Waste Treatment and Retention

During operations of the Main Plant, quantities of wastewater are generated which contain traces of various activation and fission product radionuclides. The purpose of the LLWTS is to intercept radioactive wastewater (from the plant and from the North Plateau); remove and confine radioactivity to the greatest extent practicable; and discharge the treated water at controlled rates to the environs. Figure B.7.5-1 shows a simplified schematic of the LLWTS. Radioactive wastewater from the plant includes, but is not limited to, wastewater from the laundry; miscellaneous low level process wastes including wastes from Main Plant operations; LWTS evaporator overheads; groundwater intrusion into the HLW pans and vaults; NDA Treatment System, Contact Size Reduction, monitoring well development and/or purging and stormwater. Separate and distinct wastewater extracted from the ground water plume northeast of the Main Plant is referred to as the North Plateau wastewater.

Low-level liquid wastes from the plant are collected batch-wise in one of the three interceptors (one old interceptor, two new interceptors [North and South]). Following radiological analysis, batches with gross alpha plus gross beta concentrations below 5E-3 µCi/mL (1.85E2 Bq/mL) are transferred to Lagoon 2. Lagoon 2 water is pumped via transfer pumps to the sump in the Low-Level Waste Treatment Replacement Facility (LLW2), which is located just west of Lagoon 4 (see Figure B.7.7-1). LLW2 houses two (2) skid-mounted equipment trains: Skid A and Skid B. Both are designed to treat 218,000 L/day. Under the current configuration, Skid B treats water from the North Plateau, while Skid A treats water from Lagoon 2. Both skids are capable of preferential removal of strontium-90 and cesium-137 depending on choice of ion exchange resin used. During CY-99, Skid A averaged about 65,000 L/day (17,000 gpd) while Skid B averaged about 25,000 L/day (6,000 gpd). Spent resin from both skids are sluiced directly to shipping containers, dewatered, packaged and transferred to temporary storage in readiness for eventual disposal. The treated liquids are collected batchwise in Lagoon 4 or 5, where further sampling and analyses are conducted. If the treated liquid meets discharge specifications, it is transferred to Lagoon 3. If the treated liquid does not meet discharge specifications, it is either transferred back to Lagoon 2 for recycle through the LLW2 or diluted with soft water, reanalyzed and, if specifications are met, transferred to Lagoon 3. Confirmatory measurements are performed on liquid transferred to Lagoon 3 prior to controlled discharge to the environment via Erdman Brook.

B.7.5.1 Design Objectives

The Low-Level Waste Treatment Replacement Facility is currently configured to process 189,000 L/day (50,000 gpd) of low-level liquid wastes. A comparison of the 1999 LLWTF discharge effluent isotopic concentration to the eight year average discharge isotopic concentration is provided in Table B.7.5-1. The estimated annual water balance for the Low-Level Waste Treatment System for CY-2000 is shown in Figure B.7.5-2.

The process systems in the Low-Level Waste Treatment Replacement Facility have been authorized by the New York State Department of Environmental Conservation for treatment of influent and discharge of the effluent to the environment via permitted outfall 001. The LLWTS is operated in a manner which ensures that effluent concentrations of radionuclides from Lagoon 3 do not exceed the derived concentration guides (DCG) for those nuclides as specified in DOE Order 5400.5. Effluent from Lagoon 3 is also monitored at outfall 001 for nonradiological parameters to demonstrate compliance with limits set forth in the State Pollutant Discharge Elimination System (SPDES) permit for the LLWTS.

Liquid wastes in the LLWTS are contained in the interceptors, lagoons, LLW2 building sump and process vessels. Protection from overflow or spills in LLW2 is ensured through the use of high-level cutoffs, a sloped building floor, moisture/leak sensors, skid catch basins, and drains to the LLW2 sump. These system components are discussed in the following section.

B.7.5.2 <u>Equipment and Systems Description</u>

B.7.5.2.1 Neutralization Pit and Interceptors

Liquid LLRW flowing from WVDP process areas first enters the Low-Level Waste Treatment System at the neutralization pit (see Figure B.7.5-1). Currently, sodium hydroxide (NaOH) or potassium hydroxide (KOH) is added to the wastewater (seasonally) through floor drains in the utility room to maintain a pH of greater than 10 in Lagoon 2 for insect larvae control. Alternatively, effluent streams from the reprocessing plant with the potential for having elevated contamination levels may be directed to the original NFS (old) interceptor.

Interceptors routinely receiving liquid LLRW from the reprocessing plant are dual 87,000 L (23,000 gal) stainless-steel-lined concrete pits (referred to as the North {N} interceptor or the South {S} interceptor) situated in the silty till clay. Each interceptor collects wastewater in batches for sampling and release to Lagoon 2. High-level alarms locally annunciate the potential for over-filling of an interceptor. The interceptors are constructed so that one interceptor will overflow to the other if overfilling occurs. Hydrogen peroxide (H_2O_2) is added, as needed, in summer months to the interceptors to control the growth of algae in Lagoon 2. Wastewater which has been sampled and approved for release is drained by gravity to Lagoon 2. Air sparges in the interceptors are used to keep waste water thoroughly mixed for accurate sampling and to minimize the settling of any material.

The original NFS interceptor located northwest of the stainless-steel-lined interceptor is an unlined concrete pit used to receive plant liquid effluents suspected of having elevated radioactivity contamination levels and other suspect liquids. Upon verification of acceptable radioactivity levels and other parameters of concern, the contents of this interceptor are transferred to the new interceptors via a submersible pump.

B.7.5.2.2 Lagoon System

There are four storage lagoons associated with the LLWTF identified as Lagoon numbers 2 through 5. Lagoons 2 and 3 are large holding basins constructed in the silty till with capacities of 9,100,000 L (2,400,000 gal) and 12,000,000 L (3,300,000 gal), respectively. This silty till has a low hydraulic conductivity and thus provides a level of confinement for the liquid wastes. Lagoons 4 and 5 are synthetic-lined holding basins with capacities of approximately 908,000 L (240,000 gal) and 700,000 L (185,000 gal), respectively.

Lagoon 2 is fed directly from the new interceptors and serves as the feed point for the process skid equipment in LLW2. In addition, effluents from the NDA Liquid Pretreatment System are discharged to Lagoon 2 for treatment in LLW2. Lagoon 2 also serves as an equalizing basin for recycle streams from the LLW2. Lagoon 3 is a surge basin for treated waste and is the point of discharge to the surface waters via Erdman Brook. Lagoons 4 and 5 alternate receiving and discharging LLW2 effluent water to Lagoon 3. Batch collection and sampling practices are used to monitor water quality. These lagoons discharge to Lagoon 3 normally, but can be routed to Lagoon 2 to recycle the water if required.

B.7.5.2.3 Low-Level Waste Treatment Replacement Facility (LLW2)

The LLW2 is a pre-engineered, single-story, metal-sided building (40 feet by 60 feet), located west of Lagoon 4, which houses two skid mounted (7 feet by 18 feet) process equipment modules. The floor is pitched to a longitudinal drain (running along the east wall of the building) which slopes to the sump located in the SE corner of the packaging room. The sump overflows via gravity to Lagoon 2. The LLW2 has HEPA filtration for the Packaging Room which is typically used for resin handling. Air leaving the resin handling area passes through a HEPA filter and the associated portable ventilation unit (PVU) prior to being exhausted through a short stack on the roof of the building. This stack flow is sampled to obtain periodic confirmatory information on radionuclide emissions for the annual National Emission Standards for Hazardous Air Pollutants (NESHAP) Report required by 40 CFR 61 Subpart H.

The low-level radioactive waste water stored in Lagoon 2 is transferred to the 900-gallon sump in the SE corner of the Packaging Room in the LLW2 building via two pumps housed in a shelter located on the berm between Lagoon 2 and 3. From the sump, feed water is pumped via the transfer sump pump to the 800-gallon surge tank.

The North Plateau well pumps remain unchanged. These pumps discharge to the surge tank (1100 gallons) which remains in the trailer associated with the North Plateau ground water. Transfer pumps in the trailer are used to transfer the North Plateau well water from the surge tank in the trailer to the surge tank (800 gallons) in the LLW2 associated with the Skid B.

B.7.5.2.3.1 Liquid Waste Handling

Low-level radioactive waste water treatment activities are performed in LLW2. The initial design and installation configuration requires the operation to process Lagoon 2 wastewater through Skid A and North Plateau waste water through the Skid B.

Due to the design of the IX columns, an annual reduction of spent resins waste will be generated via this process compared to the previous waste volume generated. However, based upon operational data, it is planned to integrate the operations to further optimize the process via treating both waste streams through only one skid at a time. It is permissible for both waste streams to be processed though one skid (assuming proper ion-exchange media is present) since subsequent sampling and analysis determines whether the treated liquids are transferred to Lagoon 3 for discharge to the environment or transferred to Lagoon 2 for recycle. The following sections discuss the independent operation of the Skids A and B.

B.7.5.2.3.1.1 Skid A

The water in the surge tank is pumped via an IX feed pump through the mechanical filter to the process skid for the Lagoon 2 water. The design flow rate is 20 to 50 gpm based on the current total flow rate of approximately 7 million gallons per year. The maximum flow rate is expected to be approximately 40 gpm for the Skid A equipment. The skid has three identical carbon steel ion exchange vessels with non-metallic polymer liner; each ion exchange column is capable of being charged with 50 cubic feet of dry mixed ion exchange resins. The skid mounted equipment has been designed to allow maximum flexibility in the operation of the process, such that the ion exchange columns can be used in series or parallel or by-pass in any order relative to one of the skids. The columns are designed for downflow loading and upflow elution. The pH of the influent to the ion exchange columns is adjusted prior to entering the columns. A chemical feed pump drips concentrated sulfuric acid into the influent stream to adjust the pH to between 6.0 and 8.0. Sulfuric acid used in neutralization is supplied from a 100-gallon tank (located on the sloping floor of the LLW2), which is filled via a drum pump from 55 gallon drums. Receipt and distribution of the concentrated acid is controlled by standard operating procedures. The treated effluent flows to Lagoons 4 and 5 for subsequent radiological and biochemical analyses prior to further transfer. If the effluent meets the requirements for discharge to the environment, the liquid is transferred to Lagoon 3 for discharge; otherwise, the effluent is returned back to Lagoon 2 for further processing.

B.7.5.2.3.1.2 Skid B

The NP wastewater in its associated surge tank is pumped via an IX feed pump through the process equipment on the skid. The design specifications for Skid B equipment are similar to those for Skid A. However, based upon the SPDES permit performance, the maximum flow rate was set at 20 gpm. Operation of Skid B is the same as operation of the Skid A, with the same parameters being monitored and controlled. As shown in Figure B.7.5-1, the effluent from the Skid B is combined with the effluent from the Skid A.

An improvement in the system configuration includes the ability to add a measured amount of sulfuric acid to Lagoon 3 from LLW2. This feature recognizes that the pH in the unbuffered open water of Lagoon 3 rises due to algae growth. The pH range for discharge from Lagoon 3 is between 6.5 and 8.5.

B.7.5.2.3.2 Resin Removal and Handling

The process skids do not have to be shutdown for an ion exchange column to be sluiced out. Only the column with the spent resin needs to be isolated. Soft water will be used to push the resin out of the column and into a shipping container (HIC or B-25 box) located in the packaging room. Here the container will be dewatered and prepared for interim storage. The process skids have to be shutdown to conduct the resin loading operation. In addition, resins in storage from the old Low-Level Waste Treatment System may be repackaged in the packaging room.

B.7.5.2.3.3 Low-Level Waste Treatment Replacement Facility Ventilation

There are three HVAC systems in LLW2. The main heating system includes 4 gas fired radiant heaters located near the ceiling throughout the building. The heater dedicated to the packaging room uses outside air for combustion and discharges to the outside, i.e., the combustion air and combustion products are isolated from the packaging room air. These units are designed to provide enough heat for all normal operations.

The second system consists of the portable ventilation unit (PVU) stationed outside the south wall of the packaging area. This unit will be used whenever personnel are in the packaging room during resin handling activities. The PVU draws air from the packaging room and in addition provides controlled vessel ventilation as appropriate during resin handling activities. Air leaving the resin handling area passes through a HEPA filter and the associated PVU prior to being exhausted through a short stack on the roof of the building (see Section B.7.5.2.3). Contamination levels in the water treatment portion of LLW2 are sufficiently low such that HEPA filtration is not necessary.

The third system is a forced ventilation unit for the office area that provides year-round climate control. Air from the outside is heated or cooled and discharged to the office area such that positive pressure results in the office area.

B.7.5.2.3.4 Instrumentation and Control

Controls on the LLW2 process consist of process sampling, standard operating procedures, and process instrumentation. Samples of the ion exchange effluent provide information on process operation and form the basis for changes in the types and quantities of chemical additives. Primary control parameters for treatment consist of radionuclide concentration and calcium hardness in the feed stream.

Instrumentation is used in the treatment system to monitor the system feed rate, surge tank level, and flow rate through the ion exchange beds. The Skid A influent stream is controlled to a pH range of 6.0 to 8.0. Alarms are indicated on the computer in the LLW2 office.

B.7.5.2.3.5 Fire Protection

The LLW2 was evaluated against DOE Order 420.1, *Facility Safety*, and National Fire Protection Association Codes and Standards. Fire suppression and detection are not part of the facility installation; however, other fire and life safety features provide a satisfactory measure of protection.

The building is of non-combustible or limited-combustible materials. Fire hazards include the gas-fired radiant heaters, recognized industrial electrical malfunctions, and hazards associated with the use of propane and diesel-fueled forklift trucks. The potential for a toxic, biological, and/or radiation incident due to a fire is low. A credible fire is expected to be localized to individual electrical components, with little possibility of fire propagation. Hazardous materials, such as low-level radioactive resins or sulfuric acid (which are inherent to the facility), pose only a minor risk of contamination to the immediate area.

Based upon existing and projected use of the facility, defense-in-depth during system operation is provided by the following fire protection features:

- Minimum fire loading
- Minimum ignition sources
- The 812 All-page system
- Portable fire extinguishers
- Fire Brigade and Local Fire Department response
- On-site fire hydrant coverage

Life safety considerations include emergency lighting, exit signs, and accessible exits along with the passive fire protection inherent in the materials of construction, as well as the relatively benign water treatment process.

The LLW2 is adequately separated from nearby facilities such that the potential for fire spread between facilities is unlikely.

B.7.5.3 <u>02 Building</u>

The 02 Building is a radiological facility that was part of the old LLWTS. It is a two-story steel-framed concrete block building 8.2 m (27 ft) by 11.9 m (39 ft) with an adjacent 7.6 m (25 ft) by 15.2 m (50 ft) bermed concrete slab and an adjacent office area. It is currently undergoing D&D. The chemical tanks which were located on a concrete pad outside of the 02 Building have been emptied and removed.

B.7.6 <u>Liquid Waste Solidification</u>

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

B.7.7 <u>Solid Low-Level Radioactive Wastes</u>

Solid LLRWs generated at the WVDP include Class A, B, and C wastes and TRU and suspect TRU waste. Temporary storage for these wastes is provided by Lag Storage Facility buildings and hardstand areas while waste volume reduction is performed at the Waste Reduction and Packaging Area compactor, Contact Size Reduction Facility, and Container Sorting and Packaging Facility. WVDP solid LLRW storage and volume reduction facilities are fully described in the following sections. Locations of these facilities are depicted in Figure B.7.7-1.

Packaged Class A compactable materials such as small diameter piping, sheet metal, and wood products were volume-reduced in a Supercompactor. The Supercompactor was a trailer-mounted horizontal hydraulic press located adjacent to the Lag Storage building. In 1998 the supercompactor was transferred to the U.S. DOE Savannah River Site.

B.7.7.1 <u>Design Objectives</u>

Waste storage facilities at the WVDP have been designed for the safe storage of wastes packaged to meet the requirements of 10 CFR 61. These facilities provide interim storage for wastes generated at the WVDP prior to final disposal off-site; no radioactive wastes produced at the WVDP are disposed of on-site.

B.7.7.2 Equipment and Systems Description

Facilities providing interim storage for wastes generated on-site are discussed in Section B.7.7.6. Equipment and facilities utilized for volume reduction of solid LLRWs are discussed in the following sections.

B.7.7.2.1 Waste Reduction and Packaging Area Compactor

A small compactor located on the Waste Reduction and Packaging Area (WRPA) dock is used for compacting low activity LLRW. This waste, consisting primarily of anti-C's and paper products, is collected in polyethylene bag-lined 208 L (55 gal) drums throughout the site. Full drums and boxes are transported to the WRPA dock where the bagged waste is transferred from the drums to a 2.5 m^3 (90 ft^3) rectangular steel box and compacted by a 445 kN (50 ton) box compactor. The compactor is vented by a HEPA-filtered ventilation system to provide contamination control.

B.7.7.2.2 Contact Size Reduction Facility

The Contact Size Reduction Facility has been designed for the volume reduction of large low-dose rate (<100 mR/hr) equipment resulting primarily from WVDP decontamination activities in the Main Plant. This equipment consists of process piping, vessels and other equipment formerly housed within shielded cells which were adapted for use in the IRTS. As a result of the nature of this waste, a considerable volume reduction can be realized if this material is cut into pieces which can be packed more efficiently.

The CSRF utilizes plasma arc cutting torches for size reduction, and a high pressure water spray system for decontaminating large items. This equipment is installed in the north room of the Master-Slave Manipulator (MSM) repair shop.

Low dose rate, LLRW packages to be processed in the CSRF are staged in the north airlock pending a preliminary radiation survey to verify that the exposure rate is acceptably low. Wastes that are determined to be acceptable for processing are then transferred to the cutting room. Following safe storage or removal of flammable material from the cutting room, equipment is size-reduced through the use of a plasma arc torch.

Size-reduced materials may be decontaminated prior to packaging for assay and storage. Decontamination capabilities in the CSRF include foam application, high pressure water spray and a liquid abrasive decontamination system.

The liquid abrasive decontamination system is designed to decontaminate material sectioned in the cutting room. This system is designed to use a mixture of abrasive particles, water and air to clean the surfaces of contaminated material. Material to be decontaminated by this system is loaded onto a turntable and transferred into the system decontamination booth for application of the abrasive spray. Currently Tank 15D-6, which receives liquid effluents from the LADS system, is incapable of being jetted. Consequently, the liquid abrasive decontamination system is out-of-service.

Following decontamination, material is air-dried and transferred to an airlock for final survey prior to packaging for return to the Lag Storage Facility for waste classification assay and interim storage pending final disposal off-site.

Ventilation for the CSRF is provided by a room ventilation system and backed up by the Head End Ventilation system of the Main Plant. Room ventilation is provided by a system mounted on the roof of the cutting room. Room ventilation system air flows at a nominal rate of 2.8 m^3/s (6,000 cfm) from the south MSM, vestibules and decontamination room into the cutting room, where it is exhausted through an in-cell spark arrestor and roughing filter and a roof mounted filter train consisting of a roughing filter and two HEPA filters in series prior to discharge to a locally-mounted stack. Ventilation for the MSM decontamination shower booth and liquid abrasive decontamination system decon booth/survey glove box is provided by the Head End Ventilation system. The HEV also provides backup ventilation to the various rooms when the room ventilation system is shut down. A "source capture" system that provides localized ventilation in the cutting room discharges to the room ventilation system.

CSRF ventilation system atmospheric discharges are isokinetically sampled and continuously monitored for alpha and beta activity.

B.7.7.2.3 Container Sorting and Packaging Facility

The Container Sorting and Packaging Facility (CSPF) has been designed to sort, segregate, and repackage LLRW, low-level radioactive mixed waste. This facility is also used to sort between: mixed and non-mixed wastes and to inspect container contents. Future use of the facility may involve segregating higher contamination items from lower contamination ones.

The 40-ft by 28-ft CSPF is a stand-alone facility located within Lag Storage Annex #4 (LSA-4). It is constructed of prefabricated, interlocking modular 22-gage stainless steel panels which form the outside walls, ceiling, and inner partition walls. Some wall and ceiling panels contain plexiglass windows for viewing and external lighting purposes. The concrete floor of LSA-4 serves as the floor of the CSPF. The CSPF consists of a sorting room, drum/box load-in room, drum load-out room, and two airlocks. The box/drum load-in and load-out rooms provide for safe and efficient movement of waste containers in and out of the facility, while the airlocks allow for personnel access and egress to the sorting room.

Unsorted and unsegregated waste packages entering the CSPF are moved into the drum/box load-in room prior to entering the sorting area. Packages are subsequently moved into the sorting area, placed on a lift-and-tilt-table, and opened. The lift-and-tilt-table elevates and tips the container, making the contents easily accessible. The waste is then sorted and segregated. Full drums or boxes containing sorted waste are later moved from the sorting room to the load-out room, covered and decontaminated as needed, and placed back into storage in the Lag Storage Facilities. Other equipment in the sorting room consists of a sorting table with liquid catch basin, drum roller, and an overhead bridge crane.

Adjacent to the CSPF is a stand-alone blower room which houses the ventilation system and other components essential to sorting operations. The CSPF ventilation system consists of a double stack 2000 cfm system with two nominal 1000 cfm blowers. This configuration permits one blower to be taken off-line and have its filter changed while still maintaining ventilation flow at an adequate level. The filter housings are manufactured from 14-gage T-304 stainless steel, adequately reinforced to withstand a negative or positive pressure of 10-in water gage.

The discharge side of the ventilation system from the filter to the discharge point at the exterior of LSA #4 has approximately 12-ft of 14-in diameter stainless steel duct. Two sections of 6-in diameter stainless steel duct direct ventilation air from the blowers to the locally-mounted stack. The stack penetrates the LSA-4 weather structure prior to discharging ventilation air to the atmosphere. Air ventilated by the system is monitored through the use of continuous air monitors.

If electrical power to the two ventilation blowers is lost, an auxiliary blower powered by a dedicated natural gas generator will provide adequate ventilation to the facility. Therefore, failure of both the exhaust blowers or loss of off-site power will not prevent the system from maintaining sub-atmospheric pressure in the CSPF.

Fire detection systems have been installed in the CSPF to provide personnel early warning of a potential fire. A Very Early Warning Smoke Detector System (VESDA) is used as the primary detection system for the CSPF. The VESDA, which can detect particles generated during the pre-combustion stages of a fire, is comprised of an air sampling system, filter assembly, aspiration system, detector and control system.

In addition to the VESDA system, air duct smoke detectors have been installed in the ventilation system. The alarms are monitored through the Central Site Monitoring System by means of the Data Gathering Panel and data transmission lines.

Two manual fire pull stations have been installed in conjunction with the sorting room in the CSPF. One pull station is installed in the sorting room to be activated by the sorting personnel; the second pull station is mounted immediately outside of the CSPF to be activated by anyone observing a fire through one of the panel windows. A Clean Agent Fire Suppression System (CAFSS) has been installed as the fire suppression system in the CSPF.

B.7.7.3 Operating Procedures

Operating procedures for the handling and storage of radioactive waste at the WVDP have been developed per the requirements of DOE Order 435.1 and 10 CFR 61. Update of the waste management program is given WVDP-019, Annual Waste Management Plan. Development of facility procedures is consistent with the development of other procedures at the WVDP, as discussed in Section B.10.3.

B.7.7.4 Characteristics, Concentrations, and Volumes of Solid Waste

Radiological wastes stored in the Lag Storage Facility are comprised of Class A, B, and C low-level waste and TRU and suspect TRU waste packages. A representative summary of waste and volumes generated at the WVDP are presented in Table B.7.7-1. This table is an example and is provided for information only. Estimates of the typical radiological inventory of Lag Storage waste containers are provided in Table B.7.7-2.

Approximately 275 m^3 (9,700 ft^3) of TRU/suspect TRU waste was generated at the WVDP during the period 1984-1993. This volume is being stored pending processing. It is expected that the volume of TRU waste requiring storage will be reduced by decontaminating much of this waste to below the TRU waste classification threshold. TRU waste remaining after final decontamination will be stored and eventually shipped to a federal repository once it becomes available.

Solid LLRWs which contain >0.5% liquid by volume may be stored on the hardstand provided they are double-contained. These wastes are stored pending development of draining, remediation, repackaging, or overpack operations. Upon completion of operations, the waste packages are then transferred to the appropriate storage structure.

B.7.7.5 Packaging

Solid and liquid LLRWs stored in the Lag Storage Facility are packaged at a minimum to meet the requirements of 49 CFR Parts 100 to 178. As LLRW is prepared for off-site shipment/disposal, the criteria of 10 CFR 61 is evaluated and some repackaging may be required. For purposes of criticality control, TRU waste boxes are limited to a maximum of 350 g (0.77 lb) of fissile material per box and TRU waste drums are limited to a maximum of 200 g (0.44 lb) of fissile material per drum. Administrative limits for TRU waste containers are set at 200 g (0.44 lb) fissile material per box and, 125 g (0.28 lb) fissile material per drum. Additionally, TRU waste boxes or drums cannot be stacked more than four high. The detailed analysis to support these limits is included in O-Ahoofe, 1986.

B.7.7.6 Storage and Disposal Facilities

The Lag Storage Facility, which provides interim storage of WVDP wastes prior to final off-site disposal, consists of the Lag Storage Building, Lag Storage Annexes 1, 3, and 4, four hardstands, the pump storage vault, the LSA-4 Shipping Depot, and the Rail Packaging and Staging Area as discussed in following sections. Types of wastes and available storage locations are summarized in Table B.7.7-3. Other storage and disposal facilities include the Interim Waste Storage Facility and the NDA.

Lag Storage Building

The Lag Storage Building is a pre-engineered metal structure supported by a clear span frame and anchored to a 42.7 m long by 18.3 m wide (140 ft x 60 ft) concrete slab foundation. A concrete curb encloses the inner perimeter of the building. The thickness of the concrete slab is 15 cm (10 in) at its high point and slopes downward on all sides to a thickness of 20 cm (8 in). The slab surface was originally coated with an acid-resistant two-coat application of epoxy sealer. The building is designed to withstand a snow loading of 40 pounds per square foot and a design wind loading of 100 miles per hour.

Lag Storage Annex-1

This clear span structure is a pre-engineered frame and fabric enclosure which covers an area of 58 m by 17 m (191 ft x 55 ft) with a height of 7 m (23 ft). The usable area is 51.8 m by 11.3 m by 4.3 m tall (170 ft x 37 ft x 14 ft). The weather structure is constructed using a hot dipped galvanized steel frame which meets ASTM 123. The fabric is a vinyl-coated polyester which is flame resistant and self extinguishing. The structure will support a snow load of 30 pounds per square foot and withstand a design wind velocity of 100 mph. The floor surface of LSA-1 consists of leveled, compacted fine river gravel.

Lag Storage Annex-2

The original LSA-2 facility has been dismantled and the inventory removed to Lag Storage Annexes 3 and 4. If required, a new storage facility similar in design to the current storage structures may be constructed to support future activities.

Lag Storage Annex-3

Lag Storage Annex 3 (LSA-3) is a clear span structure with a pre-engineered frame and steel sheathing and covers an area of 26.8 m by 88.7 m (88 ft x 291 ft). The usable area is 24.4 m by 86.3 m by 6.7 m tall (80 ft x 283 ft x 22 ft). The structure will support a snow load of 40 pounds per square foot and withstand a design wind velocity of 80 mph.

A 6 in high concrete curb encloses the inner perimeter. The thickness of the slab is 7 inches. LSA-3 may be heated by indirect fired, natural gas furnaces as necessary to reduce the impact of the natural freeze-thaw cycle on waste, thus minimizing the deterioration of containers stored at these locations.

Lag Storage Annex-4

Lag Storage Annex 4 (LSA-4) is a clear span structure with a pre-engineered frame and steel sheathing and covers an area of 26.8 m by 88.7 m (88 ft x 291 ft). The usable area is 24.4 m by 86.3 m by 6.7 m tall (80 ft x 283 ft x 22 ft). The structure will support a snow load of 40 pounds per square foot and withstand a design wind velocity of 80 mph.

A six-inch high concrete curb encloses the inner perimeter. The thickness of the slab is 7 inches. LSA-4 may be heated by indirect fired, natural gas furnaces as necessary to reduce the impact of the natural freeze-thaw cycle on waste, thus minimizing the deterioration of containers stored at these locations. LSA-4 provides housing for the Container Sorting and Packaging Facility described in Section B.7.7.2.4.

There is an enclosed passageway connecting LSA-4 with LSA-3. This connector is approximately 20 ft wide and 30 ft long and will allow passage of fork trucks between the two buildings without going outside. There is a roll-up door between LSA-4 and the connector.

LSA-4 Shipping Depot

The LSA-4 Shipping Depot is attached to the south side of LSA-4 and will be used primarily for loading LLW containers on flat bed trailers for shipment off-site for disposal. The foot print of this facility is approximately 7770 ft^2. There is an office space inside the depot.

Hardstand Facilities

The WVDP maintains four hardstands. The first is an asphalt paved area located south of the Chemical Process Cell-Waste Storage Area (CPC-WSA) structure. The second is a compacted gravel pad located west of LSA-3. These pads are used to store containers which are too big to move into the lag storage structures due to length and weight. Additionally, polyethylene containers are stored on these hardstands. Some vessels and equipment where contamination remains sealed inside the piece or is fixed (nonremovable) onto the surface, may be stored unpackaged provided the requirements of WVDP-010 are met to minimize the potential for contamination of the storage area.

The third hardstand area at the WVDP is located in the North Fuel Receiving and Storage Facility yard. This area houses the high integrity containers used to store contaminated resins and filter media from the fuel storage pool water treatment system. Safety issues associated with the storage of these containers are described in WVNS-SAR-012.

Low-level radioactive wastes including wastes containing >0.5% free standing liquids are stored on these hardstands. As needed, tarps may be used to cover some packages providing protection from wind and precipitation.

The fourth hardstand area at the WVDP is located north of the Drum Cell. This hardstand houses contaminated soil contained in roll-offs. The majority of soil contained in these roll-offs is from the excavation of the NDA-LPS interceptor trench discussed in Section B.5.3.4. The roll-offs are fitted with hoops and covers to prevent precipitation collection. Seasonal snow accumulations are removed and regular inspections are required to prevent recurrence of water infiltration problems.

Pump Storage Vault

The pump storage vault is located behind the Lag Storage facility. It is constructed of prefabricated, interlocking modular concrete slabs. The dimensions of the vault are approximately 19 m (63 ft) long, 2.5 m (8 ft) high, 4.5 m (15 ft) wide with wall thickness of 0.6 m (2 ft). The vault stores contaminated mobilization and/or transfer pumps which have been removed from the HLW tanks. In the future the vault may be used to store other contaminated equipment.

Rail Packaging and Staging Area

The Rail Packaging and Staging Area (RPSA) is located west of the NDA and east of the rail spur. It is a 200' by 120' gravel area on which are placed two concrete pads. One pad (the Loading Area), is 90' by 24'. The other concrete pad (the Staging Area) is 64' by 24'. Electrical power is provided to the RPSA but is not currently used.

The RPSA will be used for preparing gondola cars and bulk containers for LLW packaging. This area will also accommodate heavy capacity fork lifts, cranes, and excavators which will be used to place waste into gondola rail cars, sealands, or other bulk containers, as well as loading containers onto flat rail cars. This area will also be used to stage containers containing LLW as well as empty containers for future use.

Interim Waste Storage Facility

The Interim Waste Storage Facility (IWSF) is a pre-engineered metal structure supported by a clear span frame and anchored to a 10.7 m by 10.7 m (35.25 x 35.25 ft) concrete slab foundation. A concrete curb encloses the inner perimeter of the IWSF. The area inside the curb is 10.4 m by 10.4 m (34 x 34 ft) with the concrete slab having a thickness of 20 cm (8 in). The siding and roof is constructed of 26-gauge steel. The interior walls and ceiling are equipped with 10 cm (4 in) thick fiberglass insulation with reinforced vinyl facing. The IWSF is heated by two 15 kilowatt (51,000 BTU) electric heaters to minimize the impact of the natural freeze-thaw cycle. On the northeast corner of the IWSF is a metal 5 m x 3 m 26-gauge metal lean-to addition that houses the fire suppression equipment which consists of a high expansion foam system with one 3,304 L/s (7,000 cfm) foam generator. Foam is generated at a rate sufficient to produce a 1.4 m (5.6 ft) deep layer of fire-suppressing foam across the floor of the IWSF in one minute.

NRC-Licensed Disposal Area (NDA)

The NDA covers a rectangular area of approximately 20,000 m^2 (5 acres) and is located south of the former reprocessing plant. Reprocessing wastes generated by Nuclear Fuel Services (NFS) were disposed of within a U-shaped area along the eastern, northern and western boundaries of the NDA. There are a total of 239 disposal holes in this area. Two types of holes were used for waste burial: deep holes and special holes. Deep holes are generally 81 x 198 cm (32 x 78 in.) by 15 to 21 m (50 to 70 ft.) deep, while the shallower special holes having an average depth of approximately 6m (20 ft.) were excavated with a variety of surface dimensions.

Disposal of decontamination and decommissioning wastes generated by the WVDP occurred in the unused area within the U-shaped NFS burials. Wastes were placed in trenches, except for disposals in four steel-lined caissons 2 m (7 ft.) in diameter and 18 m (60 ft.) deep outside the NFS disposals. Each of the holes and trenches were backfilled and capped with soil excavated on-site.

Historically, the materials disposed in the NDA were categorized according to the radioactivity of the waste. (Radioactivity information was available in the facility operating logs.) Chemical data, not being required by the operating license, was never generated for the wastes. Consequently, the chemical characterization of the waste streams has been based upon historical knowledge and other records of site operations.

No known RCRA-regulated hazardous wastes were disposed in the NDA. Documentation does indicate, however, that RCRA hazardous constituents (6 NYCRR Part 371, Appendix 23) are associated with some of the materials discarded in the unit. Some of these materials, such as lead shielding, may be considered RCRA-regulated wastes if disposed today.

During reprocessing, NFS received approximately 341 rail and truck shipments of fuel from twelve different generators. The fuel was reprocessed in twenty-seven campaigns between 1966 and 1972. Each of the campaigns varied, depending on the types of fuel to be reprocessed and the quality control (QC) required. Generally, however, the chemicals used for the process remained consistent from campaign to campaign.

Liquid waste materials came from the acid fractionator condensate, floor drains in various cells and chemical makeup areas, the analytical laboratory, and from wash solutions from decontamination operations. Miscible liquid wastes generated during the fuel reprocessing campaigns were either treated in the LLWTF or routed to the HLW storage tanks. Immiscible liquids such as the tri-butyl phosphate (TBP)/n-dodecane used in the extraction process were absorbed in vermiculite and disposed in the NDA.

Wastes generated during this phase included fuel hulls, general waste such as radiologically contaminated clothing and equipment, fuel canisters, ruptured fuel encased in concrete, and spent ion-exchange resins and diatomaceous filter media. As a result of all disposal operations, an estimated 417,000 curies (1.54E16 Bq) is in the NDA. Approximately 397,000 curies (1.4E16 Bq) or 95% of the total curies buried in the NDA are attributed to the fuel hulls and hardware from NFS fuel reprocessing activities.

NFS cleanup of the site in preparation for upgrading the process building and post-1975 maintenance activities included decontamination of contaminated areas and equipment to allow access to the process building, disposal of unwanted equipment, cleanup of process materials, and environmental monitoring as required by the license. The types of trace contaminants that may be from this phase include laboratory chemicals and the chemicals used in the process that could have been left as a residue on the equipment.

During the WVDP operations, waste was generated from aggressive decontamination of the process building so it could be used to house the vitrification process and provide a safe work environment for employees. This phase of operations again generated a large amount of radioactive waste consisting of fuel processing equipment, scrap, and related materials that may exhibit trace levels of decontamination chemicals and chemical residues from past fuel processing operations. Minor volumes of Class A waste from these activities are disposed of in the NDA.

An estimated 10,392 cubic meters (367,000 ft^3) of radioactive waste is buried in the NDA. Approximately 4,587 cubic meters (162,000 ft^3) of the total quantity was disposed during NFS operations, and 5,805 cubic meters (205,000 ft^3) were disposed during WVDP operations. In order to characterize the NDA, a summary profile of the waste streams and the percentage by volume that they represent was prepared for inclusion in the RCRA Facility Investigation Report (WVDP-RFI-018). This profile is included as Table B.7.7-5. The quantity of each waste stream is an approximation that is based on the two NDA waste database systems and the operation logs.

The most predominant waste streams by volume in the NDA according to the profile are contaminated soils at 36%, general process building waste at 20%, and FRS and LLWTF wastes at 14% and 13%, respectively. The combination of these four waste streams makes up almost 83% of the total wastes disposed in the NDA. The remainder of the wastes consists of decontamination-generated debris, scrap material, and equipment; analytical laboratory wastes; TBP/n-dodecane absorbed onto vermiculite; fuel hulls; fuel canisters; ruptured fuel rods; and lead shielding. Depending upon their source, some of these wastes were isolated in particular areas of the NDA. For instance, the hulls are documented as being buried exclusively in the eastern quadrant of the disposal area in the deep holes.

Based on inventory alone, and in accordance with the methodology presented in DOE-STD-1027-92, Hazard Categorization and Accident Analysis Reports, the NDA is a

Category 2 Nuclear Facility. It is postulated that a natural phenomenon-induced accident involving NDA radiological materials could lead to "significant on-site consequences"; therefore, the final hazard categorization for the NDA is Category 2.

B.7.8 Hazardous and Mixed Wastes

Hazardous wastes generated at the WVDP include nonradioactive solid and liquid hazardous wastes and solid and liquid low-level radioactive mixed waste. Programs and facilities at the WVDP provide for the safe interim storage of these wastes prior to shipment for off-site treatment and disposal. Some mixed wastes are neutralized on-site and sent to the interceptors, or to Tank 8D-2.

B.7.8.1 Characteristics and Volumes of Hazardous and Mixed Wastes

B.7.8.1.1 Hazardous Wastes

Hazardous wastes generated on-site from defined waste streams are accumulated in Satellite Accumulation Areas before transfer to the HWSF for storage prior to off-site shipment. These wastes consist primarily of oils from maintenance and analytical laboratory wastes. A summary of the quantity of hazardous waste stored at the WVDP is given in Table B.7.7-1.

B.7.8.1.2 Low-Level Radioactive Mixed Wastes

Low-Level radioactive mixed wastes are radioactive wastes which include hazardous wastes described in 40 CFR 261 and 6 NYCRR 371. These wastes, which may be stored in the Lag Storage Building, LSA-3, LSA-4, CPC-WSA, and the IWSF, are comprised of low-level and TRU radioactive wastes of solid or liquid form that contain heavy metals, combustibles, flammables, PCB-contaminated oils, and PCB-contaminated equipment.

These wastes are packaged for storage according to applicable federal and state environmental regulations and the conditions of the Federal and State Facilities Compliance Act (FSFCA). A summary of the quantity of low-level radioactive mixed waste stored at the WVDP is given in Table B.7.7-1.

B.7.8.2 Storage Facilities

Storage for solid low-level radioactive mixed wastes at the WVDP is provided in the Lag Storage Building and associated annexes discussed above. Storage of liquid low-level radioactive mixed wastes is provided in the Interim Waste Storage Facility (IWSF), LSA-3 and LSA-4. Hazardous wastes generated throughout the site are temporarily stored at the Hazardous Waste Storage Facility prior to shipment off-site for treatment and disposal. Hazardous Waste Storage Facilities are further described below.

Hazardous Waste Storage Facility

Four identical free-standing structures (lockers) located north of the Lag Storage Building are utilized for temporary storage of hazardous wastes generated at the WVDP. The hazardous waste lockers are pre-engineered structures containing segregated 1,000 mL bottles through 85-gallon drums (included bagged waste) of hazardous and nonhazardous wastes. Each locker is 2.4 m x 4.6 m x 2.4 m high (7.9 ft x 15 ft x 7.9 ft) and contains a spill basin beneath a steel grate floor with a capacity of 474 liters (125 gal). The lockers have been designed to contain flammable materials and are equipped with fire suppression devices, remote and local fire alarm systems, explosion proof electrical components, and explosion proof vents.

Chemical Process Cell-Waste Storage Area

The Chemical Process Cell-Waste Storage Area (CPC-WSA) facility consists of a 60.96 m (200 ft) long by 21.34 m (70 ft) wide by 9.14 m (30 ft) high arched, 12-gauge, galvanized steel-panel enclosure. The floor of the CPC-WSA is a gravel pad.

The CPC-WSA primarily contains wastes which were generated during the decontamination of the chemical processing cell (CPC), located in the former reprocessing facility. The area currently contains thirty-five waste storage boxes and forty-five concrete shield module overpacks. The twenty-two waste storage boxes resulting from the CPC decontamination effort consists of twelve jumper boxes, nine vessel boxes, and one general waste storage box. Several of the jumper boxes stored within shielded modules are expected to contain the RCRA wastes, lead and mercury. One hundred fifty jumpers (pipes with special connectors) were loaded into seven inner boxes within the CPC and transferred to the equipment decontamination room (EDR) adjoining the CPC. In the EDR, each inner box was lowered into an outer box (designated as storage boxes J1 through J7, see Table B.7.7-4) with a prepared liner, then sealed and decontaminated before being moved to the CPC-WSA. General waste from the CPC also was loaded into boxes designated J8 through J12 (see Table B.7.7-4) Contact exposure rates were typically 2 R/hr, with one hot spot up to 78 R/hr.

The exterior surfaces of thirteen vessels that had been part of the fuel reprocessing chemical stream in the CPC were steam-cleaned and coated with a clear fixative coating. The vessel internals were inspected with a video camera and all were found to be clean except for: the recycle evaporator, 7C-4 (subsequently loaded into box designated 7C-4) and the low-level waste accountability tank, 7D-10 (subsequently loaded into box designated 7D-10). Both pieces of equipment had a layer of sludge about 0.3 m (1 ft) thick on the bottom. Ten of the vessels were transferred to the equipment decontamination room and loaded into nine boxes (fabricated of carbon steel) with resultant contact exposure rates ranging from 0.1 R/hr to 110 R/hr. The three condensers, 7E-5, 7E-8, and 3E-1, which were originally planned to go into the vessel box with this same designation were actually loaded into jumper box J5 instead.

Final cleanup resulted in six boxes of general waste being loaded into the carbon steel vessel box designated as 7E-5/7E-8/3E-1.

Following transfer of the twenty-two waste boxes with CPC jumpers, vessels, and debris to the CPC-WSA, the waste boxes with the highest dose rates were covered with shielding to reduce general area exposure rates outside the shield modules to below 15mR/hr. The CPC-WSA was planned as a temporary storage area for the twenty-two waste boxes. It is north of the waste tank farm and remote from routine traffic, and is within the site's protected and controlled area.

The twenty-two waste storage boxes are surrounded by forty-five concrete hexagonal shield module overpacks arranged in an oblong circle. These measure 2.06 m (7 feet) across the flats and 3.2 m (10.5 ft) high. To supplement the shielding ability of these overpacks, vertical steel plates were added at select locations, as determined by radiation monitoring, to inhibit streaming. Each shield module contains twenty-one 55-gallon drums. These drums were filled with either contaminated debris, or clean soil, sand, and/or gravel to enhance the shielding capabilities of the overpacks. Of the total 945 drums in the shield modules, 813 drums contain low-level radioactive waste (LLRW), which have been classified as Class A, B, or C LLRW. Of the remaining 132 drums, 128 have been classified as non-radioactive, and 4 are presently unclassified, but assumed to be Class A at this time.

Nine large waste storage boxes are on the west end of the storage pad, and on the east end there are four large waste storage boxes. These boxes have external exposure rates ranging from 1 mR/hr to 30 mR/hr.

The entire storage array is covered with a steel weather structure that shields the storage boxes and shield modules from rain and snow. When a remotely controlled size-reduction facility is available, the CPC equipment in the waste boxes will be volume-reduced and packaged for disposal.

Preliminary estimates by Meigs (1987), updated to include activity estimates for waste box J12, indicated 263 Ci (9.7E12 Bq) of Sr-90, 274 Ci (1.0E13 Bq) of Cs-137, 6 Ci (2.22E11 Bq) of Am-241, and 234 Ci total Pu are present in the twenty-two waste boxes. These activity estimates are based on actual container dose rates as measured during the 1985-1987 period. Isotopic distribution is based on a site-specific, reference spent fuel isotope distribution. Table B.7.7-4 Provides estimates of Cs-137 activity for 1987 as well as estimates of Cs-137 activity decay corrected to the year 1996. Using the following isotopic breakdown for total plutonium, 6.4% Pu-238, 2.1% Pu-239, 1.3% Pu-240, and 90.2% Pu-241, the isotopic activity can be estimated as 15 Ci (5.5E11 Bq) of Pu-238, 5 Ci (1.85E11 Bq) of Pu-239, 3 Ci (1.11E11 Bq) of Pu-240, and 211 Ci (7.8E12 Bq) of Pu-241.

Based on inventory alone, and in accordance with the methodology presented in DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*, the CPC-WSA is a Category 3 nuclear facility. It is postulated that a natural phenomena-induced accident involving CPC-WSA radiological materials could lead to "significant localized consequences"; therefore, the final hazard categorization for the CPC-WSA is Category 3.

As indicated previously, a large volume of Class A, Class B, and Class C low-level radioactive waste is in 208 L (55 gal) drums within the shield modules as well as in the thirteen steel boxes located at the east and west end of the oblong circle of shield modules. Only a small percentage of these have readings above 25 mR/hr, indicating that the total nuclide activity would not increase the overall facility hazard category (since the 22 waste storage boxes are estimated to contain well below Category 2 inventory thresholds).

Prior to final size-reduction and packaging, some of the containers may need to be moved, in which case the following problems could arise:

- Moving the shield modules may require the removal of the 55-gallon drums from the module. Some drums may have corroded during the storage period to date.

- Radiological hazards must be considered when moving the jumper and vessel storage boxes.

- The twelve size-reduction boxes and one special storage box (designated SP-022) outside the shield modules contain radioactive waste that has not been well characterized at this time.

Before any waste container is moved, a radiation work permit (RWP) and an industrial work permit (IWP) will be prepared to ensure that work will be conducted safely and in accordance with WVDP-010, *WVDP Radiological Manual*, and WVDP-087, *WVDP Hoisting and Rigging Manual*.

B.7.8.3 Operating Procedures

Operating procedures for the handling and storage of hazardous and low-level radioactive mixed waste at the WVDP have been developed per the guidance given in WV-996, *Hazardous Waste Management Program*, WVDP-080, *PCB and PCB Contaminated Materials Management Plan*, and WVDP-019, *Annual Waste Management Plan*. These waste management plans have been developed to ensure compliance with the local and federal codes and regulations outlined in Table B.7.1-1. Development of facility operating procedures is consistent with the development of other procedures at the WVDP, as discussed in Section B.10.3.

REFERENCES FOR CHAPTER B.7.0

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

__________. Title 40, Part 261: *Identification and Listing of Hazardous Waste.*

New York Codes, Rules, and Regulations. Title 6, Part 371: *Identification and Listing of Hazardous Waste.*

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

__________. WVDP-019: *Annual Waste Management Plan.* (Latest Revision.) West Valley Nuclear Services Co.

__________. WVDP-030: *TRU Waste Certification Program Plan.* (Latest Revision.) West Valley Nuclear Services Co.

REFERENCES FOR CHAPTER B.7.0 (concluded)

__________. WVDP-080: *PCB and PCB Contaminated Materials Management Plan.* (Latest Revision.) West Valley Nuclear Services Co.

__________. WVDP-RFI-018: *RCRA Facility Investigation Report Volume 2 - Nuclear Regulatory Commission - Licensed Disposal Area* (Rev. 0). September 1995. West Valley Nuclear Services Co., Inc.

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

__________. Safety Analysis Report WVNS-SAR-012: *Safety Analysis Report for Fuel Receiving and Storage Facility.* (Latest Revision.) West Valley Nuclear Services Co.

TABLE B.7.1-1

WASTE MANAGEMENT PLANS, CODES AND REGULATIONS EMPLOYED AT THE WVDP

WASTE MANAGEMENT PLAN	LOCAL AND FEDERAL CODES AND REGULATIONS
ENVIRONMENTAL MANAGEMENT AND MINIMIZATION OF WASTES	
WV-980 - WVNS Environmental Management System WVDP-087 - Waste Minimization/ Pollution Prevention Awareness Plan	DOE Order 231.1 DOE Order 451.1 DOE Order 5400.1 DOE Order 5400.5 DOE Order 5484.1 DOE-EH-0173T 40 CFR, Various sections 6 NYCRR, Various sections
RADIOACTIVE AND MIXED WASTES	
WVDP-019 - Annual Waste Management Plan	DOE Order 435.1 10 CFR 61 40 CFR 264 40 CFR 265 6 NYCRR 373 Federal and State Facility Compliance Agreement Federal Facility Compliance Act
HAZARDOUS WASTE	
WV-996 - Hazardous Waste Management Program WVDP-080 - PCB and PCB Contaminated Materials Management Plan	40 CFR 261-268 40 CFR 270 6 NYCRR 370-374 6 NYCRR 376
INDUSTRIAL WASTE	
WVDP-072 - WVDP Asbestos Management Plan WVDP-164 - Used Oil Management Plan	

TABLE B.7.4-1

SUMMARY OF OFF-GAS FILTER MONITORING INSTRUMENTATION

Ventilation System	Filter Instrumentation						Plenum or Header PAH
	PDR	PDAH	PDCH	PDAL	PDCL	PR	
Vessel Off-Gas (Upstream)	X					X	
Vessel Off-Gas (Downstream)	X	X					
Waste Tank Farm	X	X					

PDR = Pressure Differential Recorder
PDAH = Pressure Differential Alarm High
PDCH = Pressure Differential Control High
PDAL = Pressure Differential Alarm Low
PDCL = Pressure Differential Control Low
PR = Pressure Recorder
PAH = Pressure Alarm High

TABLE B.7.4-2

VESSELS VENTILATED BY THE VESSEL OFF-GAS SYSTEM

Vessel	Description	Cell	Reference Drawing
3D-2	Condensate Catch Tank	LWC	3R-A-1
7D-2	LL Waste Evap Feed Tank	LWC	7R-A-1
7D-8	Tank 6D-3 Overflow Receiver	LWC	7R-A-1
7D-14	Hot Anal Cell Drain Catch Tank	LWC	7R-A-1
13D-7	Solvent Waste Catch Tank	LWC	15R-A-6
13D-8	Solvent Waste Hold Tank	LWC	15R-A-6
4D-8	Empty/OOS - still negative	LWC	15R-A-6
4D-10	Empty/OOS - still negative	LWC	15R-A-6
4D-13	Empty/OOS - still negative	LWC	15R-A-6
6D-3	Catch Tank	OGC	6R-A-1
7D-13	CSS Sump Receiver	Yard	901D-021S2
5D-12A	Empty/OOS - still negative	PPC	5R-A-1
5D-12B	Empty/OOS - still negative	PPC	5R-A-1
5D-13A	Empty/OOS - still negative	PPC	5R-A-1
5D-13B	Empty/OOS - still negative	PPC	5R-A-1
5D-13C	Empty/OOS - still negative	PPC	5R-A-1
5V-1	Empty/OOS - still negative	ULO	5R-A-1
7D-3	Empty/OOS - still negative	ARC	7R-A-2
7D-6	Empty/OOS - still negative	ARPR	7R-A-2
7E-10	Empty/OOS - still negative	AR-OG roof	7R-A-2
7D-11	Empty/OOS - still negative	HAC	7R-A-2
7D-12	Empty/OOS - still negative	HAC	7R-A-2
7E-13	Empty/OOS - still negative	ARC	7R-A-2
	LWTS System Vessels		
71C-001	Organic Ion Exchanger	XC-3	901D-023S1
71C-002	Zeolite Ion Exchanger	XC-3	901D-023S1
71C-003	Zeolite Ion Exchanger	XC-3	901D-023S2
71C-004	Evaporator/Condenser	XC-3	901D-022
71D-005	Distillate Surge Tank	XC-3	901D-022
71D-006	Spent Resin Tank	XC-3	901D-026
71D-007	Spent Zeolite Tank	XC-3	901D-026
71D-008	Filter Backwash Filter Tank	XC-3	901D-026
71D-009	Feed Sample Tank	XC-3	901D-022
71D-011	Low TDS Feed Tank	XC-3	901D-023S2
71V-010	Filter	XC-3	901D-023S1
5D-15A1	Evap Concentrates	UPC	5R-A-1
5D-15A2	Evap Concentrates	UPC	5R-A-1
5D-15B	LWTS Evap Feed Tank	UPC	5R-A-1
14D-7	Acid Add Tank	LXA	901D-059
14D-18	Caustic Add Tank	LXA	901D-059

NOTE: Tanks in the ARC, HAC, XC-1 and XC-2 are out-of-service (OOS) and have been decontaminated (Riethmiller, 1981); however, the VOG still provides negative pressure on the vent lines to these tanks.

TABLE B.7.5-1

COMPARISON OF 1999 LLWTS EFFLUENT ISOTOPIC CONCENTRATIONS TO EIGHT YEAR AVERAGE ISOTOPIC CONCENTRATIONS

ISOTOPE	8 Year Average Lagoon 3 Conc.[1] (µCi/mL)	1999 Average Lagoon 3 Conc. (µCi/mL)	Ratio of 1999 Average to 8 Year Average
Alpha	1.70E-08	1.48E-08	0.87
Beta	6.72E-07	4.11E-07	0.61
H-3	1.62E-05	3.75E-06	0.23
C-14	1.28E-08	2.79E-08	2.18
Sr-90+[2]	1.32E-07	1.17E-07	0.89
I-129	3.52E-09	4.48E-09	1.27
Cs-137+[2]	7.19E-08	1.44E-07	2.00
U-233/234	6.94E-09	6.52E-09	0.94
U-235/236	1.73E-10	2.22E-10	1.28
U-238	3.67E-09	3.71E-09	1.01
Pu-238	2.93E-10	1.84E-10	0.63
Pu-239/240	1.06E-10	9.02E-11	0.85
Am-241	1.10E-10	1.60E-10	1.46

Notes:

1. Based on eight year average of Lagoon 3 concentrations compiled from WVNS Annual Site Environmental Reports, 1992-1999.

2. '+' indicates tabulated activity is the sum of the parent activity and the daughter activity.

3. WVNS Annual Site Environmental Report. June 2000. Table C-2, p. C-4.

TABLE B.7.7-1

TYPICAL INVENTORY OF WASTE STORED AT THE WVDP

Waste Class	Volume or Mass
Low-Level Waste	
Class A	3,762 m^3
Class B	439 m^3
Class C	105 m^3
TRU Waste	60 m^3
Suspect TRU	755 m^3
CPC Waste	180 m^3
Contaminated Soil	2,906 m^3
Mixed Waste	300 m^3
Hazardous Waste	1 m^3

TABLE B.7.7-2

TYPICAL RADIOLOGICAL INVENTORY OF LAG STORAGE WASTE CONTAINERS

Nuclide	Drum Contents [1] (Ci)	Box Contents [2] (Ci)
Sr-90	1.1E-1	1.1E+0
Cs-137	1.5E-1	1.5E+0
Pu-238	4.9E-3	4.9E-2
Pu-239	7.9E-4	7.9E-3
Pu-240	1.3E-3	1.3E-2
Pu-241	9.4E-2	9.4E-1
Am-241	7.0E-3	7.0E-2
Am-243	3.9E-5	3.6E-4
Cm-244	1.5E-3	1.5E-2

Notes:

[1] - Isotopic distribution calculated using ORIGEN2 with the following basis: PWR fuel, 3.3 w/o U-235, 33000 MWD/MTU burnup, 30 MW/MTU specific power. Scaled to 0.15 Ci Cs-137 activity.

[2] - Isotopic distribution calculated as described in [1], scaled to 1.5 Ci Cs-137 activity.

TABLE B.7.7-3

WASTE TYPE AND AVAILABLE STORAGE LOCATIONS IN WVDP LAG STORAGE FACILITIES

Waste Type	Lag	LSA-1	LSA-3	LSA-4	IWSF	HWSF	SAA	Hardstands	NDA	Pump Storage Vault	CPC-WSA
Low-level	X	X	X	X	X			X	X		X
Transuranic	X	X	X	X					X		
Suspect Transuranic	X	X	X	X							X
Mixed	X		X	X	X			X	X		X
Liquids	X		X	X	X			X	X		
Hazardous						X	X		X		
Poly overpacks, Poly wrapped equipment, Oversized containers								X	X		
Equipment contaminated with HLW residue									X	X	X

Table B.7.7-4

Contents, Activity, and Fissile Mass in the Twenty-two Waste Storage Boxes stored in the CPC-WSA[2]

Storage Box Designation	Contents	Contents of Box WT lbs/(kg)	Average Exposure Rate mR/hr	Activity: Cs-137 Ci (Ci)[6]	Fissile Mass (U-235 equivalent) Grams (Estimated)
Twelve Jumper Storage Boxes (double boxed)					
J1[3]	Jumpers and metallic debris		157	1.96 (1.59)	3.507
J2	Jumpers and metallic debris		1,378	17.23 (14.01)	30.831
J3	Jumpers and metallic debris		1,680	21.00[4] (17.08)	37.577
J4[2]	Jumpers and metallic debris		196	2.45 (1.9)	4.384
J5[2]	Jumpers, metallic debris, three condensers (7E-5, 7E-8, and 3E-1)		326	4.08 (3.32)	7.301
J6[2]	Jumpers and metallic debris		226	2.82 (2.29)	5.046
J7[2]	Jumpers and metallic debris		266	3.33 (2.71)	5.959
J8[2]	Metallic debris		329	4.11 (3.34)	7.354
J9[2]	Metallic debris		1,057	13.20 (10.73)	23.619
J10[2]	Metallic debris		1,550	19.36 (15.74)	34.641
J11	Metallic debris		2,034	25.39 (20.65)	45.431
J12	Debris		2,360	29.50 (23.99)	52.785
Nine Vessel Storage Boxes					
3C-1	Fuel dissolver	26,015	2,634	32.93[5] (26.78)	58.925
3C-2	Fuel dissolver	26,015	2,854	35.67[4] (29.01)	63.828
7C-2[2]	LLW Evaporator	14,110	120	1.50 (1.22)	2.684
3E-2/3E-3	Dissolver Condensers	14,991	42	0.53 (0.43)	0.948
7C-4[2]	LLW Evaporator	9.921 (4,500)	98	1.23 (1.00)	2.201
7D-10[2]	LLW Accountability and Neutralizer Tank	9,921 (4,500)	47	0.59 (0.48)	1.056
7C-1[2]	HLW Evaporator	5,512 (2,500)	1,239	15.49 (12.60)	27.718
3D-1	Fuel Accountability and Feed Adjustment Tank	10,582 (4,800)	250	3.13 (2.55)	5.601
7D-4	HLW Accountability and Neutralizer Tank	5,952 (2,700)	743	9.29 (7.55)	16.624

2 *Based on information in memo HB:86:0161, R. Keel and R. Meigs to P. Valenti, November 5, 1986*

3 *These containers contained no equipment that was used to process fissile materials. Cs-137 inventory is due to HLW, so a spent fuel distribution conservativley overestimates fissile content.*

4 *78 R/hr hot spot was assumed to be a short line source.*

5 *7 Ci of Co-60 is assumed as a contributor to the dose rate. It was not included since it is not from the fission product inventory.*

5 *Estimated activity 1987.*

6 *Activity decay corrected to 1996.*

Table B.7.7-4

Contents, Activity, and Fissile Mass in the Twenty-two Waste Storage Boxes stored in the CPC-WSA (concluded)

Storage Box Designation	Contents	Contents of Box WT lbs/(kg)	Average Exposure Rate mR/hr	Activity: Cs-137 Ci (Ci)[6]	Fissile Mass (U-235 equivalent) Grams (Estimated)
One General Waste Storage Box					
7E-5/7E-8/3E-1	Six 44" x 44" (1.12 m x 1.12 m x 1.12 m) boxes full of general waste	2,800 (1,273)	2,360	29.50 (23.99)	52.785
TOTALS				**274.29 (233.05)**	**490.81**

Table B.7.7-5
NDA Waste Disposal Summary Profile

CATEGORY	NFS-162,000 ft³		WVNS-205,000 ft³	MAXIMUM POTENTIAL TOTAL *	APPROXIMATE % OF TOTAL DISPOSAL BY VOLUME	GENERAL LOCATION OF BURIAL
	1966-1975	1976-1981	1982-1986			
Hulls	7,403	0	0	7,403	2%	Southeast Quadrant
Culligans, FRS, Filters	18,002	35,718	1,220	54,940	14%	Throughout NDA, But more predominant in Southeast Quadrant
Solvent-contaminated Materials	4,302	711	367***	5,830	1%	North Boundary Section
LLWTF Sludge	1,384	39,299	12,852	53,535	13%	
HLW and General Process Building Waste	9,093	11,319	56,696	77,108	20%	
Fuel	39	0	0	39	<.01%	Holes 48 & 102-Northeast Quadrant
Contaminated Soils	40,830	1,172	103,305	145,307	36%	North Boundary Section
Scrap, Junk, Debris	4,080	11,588	11,597	27,265	6%	
Analytical	722	321	390	1,433	<1%	
Fuel Canisters	28	7,506	3,780	11,314	3%	Holes 88, 89, 90, and WVDP-11
Lead	4	7,601**	9	7,614	2%	SH-105 and WVDP-8
Miscellaneous	N/A	N/A	10,523	10,523	3%	

(WVDP-RFI-018, Rev. 0)

* Maximum potential disposal of targeted waste streams (will not equal the total amount disposed).
** This quantity is based on the total amount disposed of in SH-105 as the specific quantity for lead was not given. Thought to be very high
*** There was a discrepancy between two different tracking databases, therefore this quantity represents the worse case scenario.

FRS - Fuel receiving and storage
HLW - High-level waste

SR2B751A.DWG

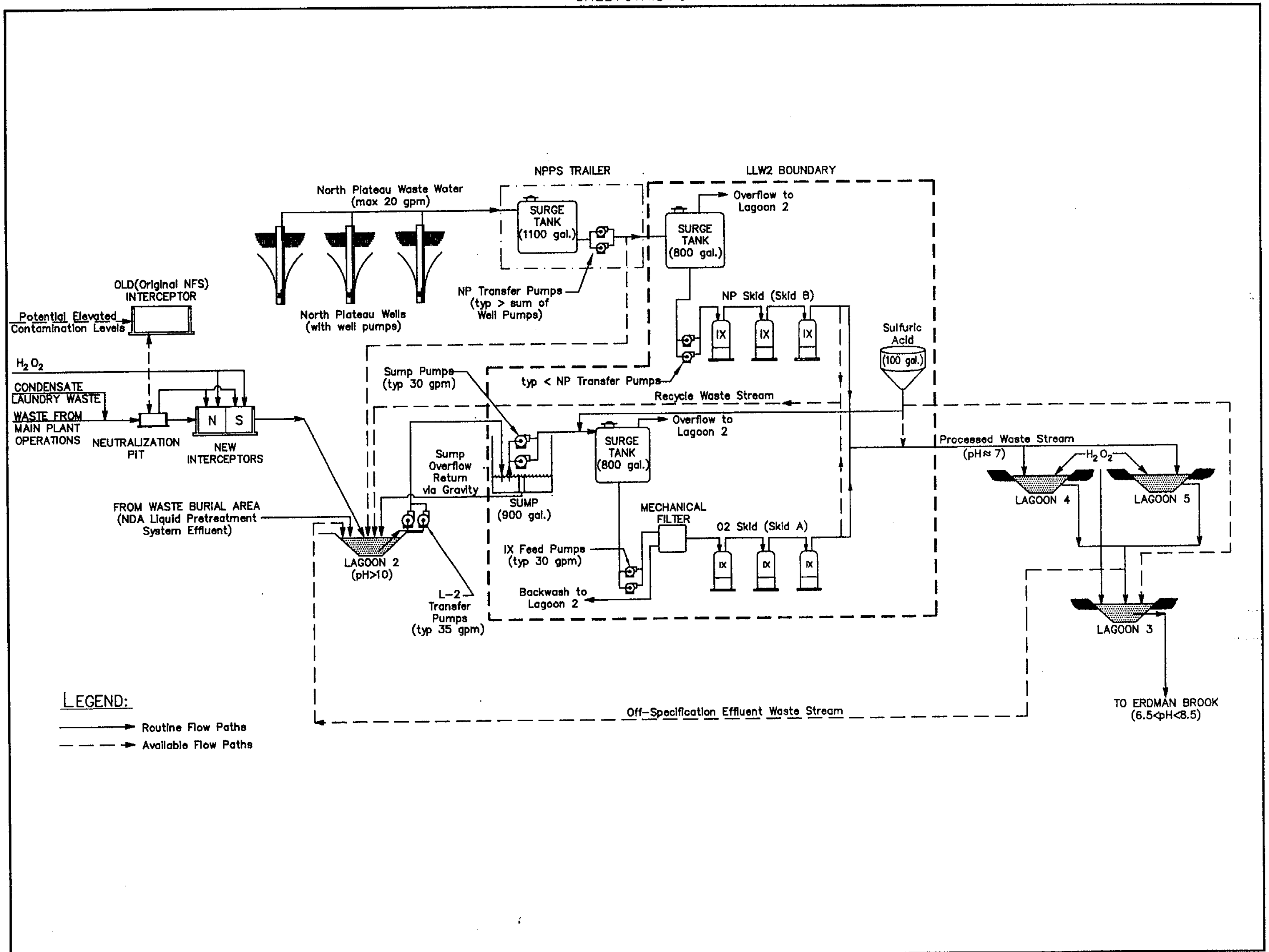

Figure B.7.5-1. Flow Diagram of Low-Level Waste Treatment System

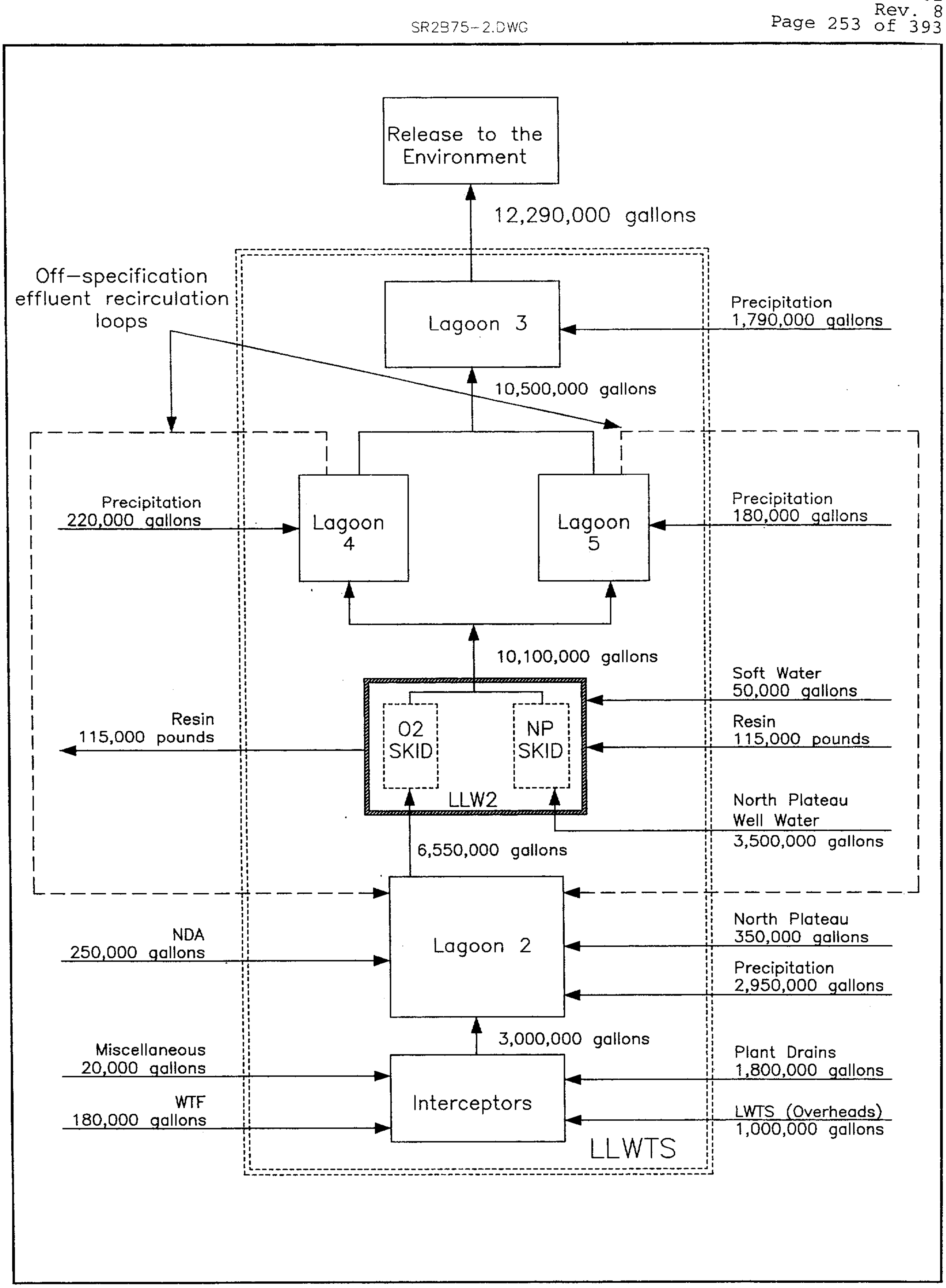

Figure B.7.5-2. Estimated Annual Water Balance for LLWTS for CY–2000

SR2B77-1.DWG

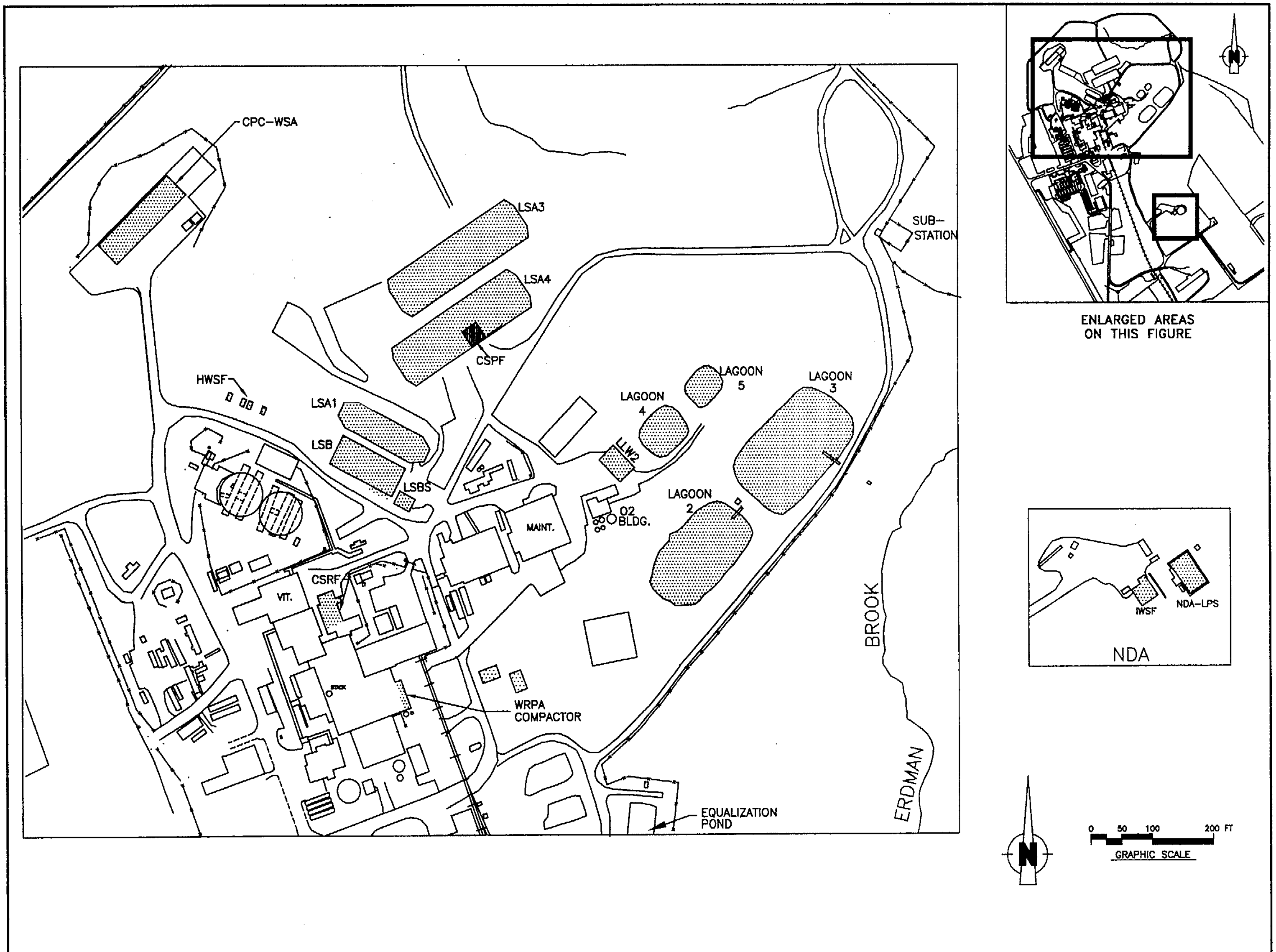

Figure B.7.7-1. Location of Waste Storage and Processing Facilities

B.8.0 HAZARDS PROTECTION

B.8.1 Assuring that Occupational Hazards Exposures Are ALARA

B.8.1.1 Policy Considerations

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

In addition to radiation protection programs, the WVDP has established a comprehensive industrial hygiene and safety program for the identification, assessment and monitoring of nonradiological hazards. Administration of the industrial hygiene and safety program is through WVDP-011, *WVDP Industrial Hygiene and Safety Manual*, which incorporates the guidance of DOE adopted OSHA standards 29 CFR 1910 and 29 CFR 1926.

B.8.1.2 Design Considerations

The prime consideration in maintaining radiation and hazardous material exposures ALARA is ensuring that positive control of these materials is maintained. Design features that ensure the confinement of radioactivity include:

- Ventilation systems which maintain areas containing contamination under negative pressure relative to surrounding occupied areas.

- Remote valving and instrumentation for vessels and components containing radioactive sources.

- Ventilation filtration systems which ensure that effluent air streams are decontaminated before being discharged through a stack to the atmosphere.

- Cell sumps which collect any liquid spills released to the cell.

Additional mitigative design measures include:

- Use of redundancy in primary ventilation components.

- Ability to monitor and control liquid transfers remotely from the control panel within the control room.

- Airlocks which assist ventilation systems in maintaining contamination under negative pressure.

- Equipment design to enable remote replacement of failed components, if necessary.

Design features that ensure that exposures to nonradiological hazardous materials are maintained ALARA include:

- Specially-designed facilities in the New Warehouse which provide storage and isolation of bulk quantities of reactive chemicals.

- Laboratory fume hoods which prevent occupational exposure to analytical reagents.

- Paint spray booths to prevent maintenance personnel exposure to paint fumes.

- Berms, sumps and other spill containment mechanisms.

B.8.1.3 Operational Considerations

In addition to considerations incorporated in facility design, administrative controls are necessary to ensure that personnel hazards exposures are maintained ALARA. Administrative and procedural control is maintained in accordance with the WVNS Industrial Hygiene and Safety Manual (West Valley Nuclear Services Co., Inc., WVDP-011), the Radiological Controls Manual (WVDP-010) and specific standard operating procedures and work instructions. Site operations personnel are fully trained in elements of these programs, as discussed in Section B.10.3.

B.8.2 <u>Sources of Hazards</u>

B.8.2.1 <u>Contained Sources</u>

B.8.2.1.1 Contained Radioactive Material Sources

Radiation sources in the IRTS derive from radioactivity present in high level waste contained in Tank 8D-2 and from the radioactively contaminated zeolite in Tank 8D-1 which results from supernatant and sludge wash solution processing. High level waste in Tank 8D-2 originated during spent nuclear fuel reprocessing in the former Nuclear Fuel Services reprocessing plant. Due to the extended post-reactor storage period, all gamma-emitting isotopes have decayed to levels insignificant with respect to Cs-137.

Radioactively-contaminated solutions in the STS are contained within process vessels located in Tank 8D-1 and in transfer piping between Tanks 8D-1 and 8D-2. Product solution from STS is stored in Tank 8D-3 prior to transfer to the LWTS.

Radiation sources in the LWTS include radioactive solutions in process vessels and piping located in the GCR extension, XC-3, UPC and LWC. Pumps and piping associated with the LWTS are located in niches in the LWA, UWA, and ULO. These facilities are all located in the Main Plant building in areas originally designed to accommodate process solutions with activity levels much greater than those currently processed in the LWTS. Shielding is therefore much greater than that necessary to attenuate exposure rates to acceptable levels.

Feed to the CSS from LWTS Tank 5D-15A1 or Tank 5D-15A2 is received by the Waste Dispensing Vessel in the WDC of the 01-14 Building. This waste is transferred to Mixer 70K-002 or 70K-004 in the CSS Process Room prior to transfer to a 269 L carbon steel drum. Full waste drums are staged in the loadout area prior to transfer to the IRTS Drum Cell for storage.

The Main Plant currently provides support to the IRTS and vitrification and also provides confinement for contamination remaining from former fuel reprocessing operations. Relative to vitrification, the High-Level Waste Interim Storage (HLW canister storage) is located in the former Chemical Process Cell (CPC) and is described in SAR-003, *Safety Analysis Report for Vitrification Operations and High-Level Waste Interim Storage,* Section C.6.5.3, *High-Level Waste Interim Storage.* Radiation sources in the plant include radioactive process solutions in vessels and piping associated with LWTS operations and radioactive particulate contamination associated with ventilation system filters. In addition, contamination in plant cells presents a significant source of radiation; however, no activities are routinely conducted in areas containing high levels of contamination. Sources of radiation associated with ventilation and off-gas operations are located in the Head End Ventilation building, Ventilation Exhaust Cell (VEC), Ventilation Wash Room (VWR) and Off-Gas Cell (OGC). These areas were designed to support operations with concentrations of radioactivity much greater than those associated with current operations. Shielding is therefore significantly greater than that necessary to attenuate current exposure rates to acceptable levels.

Design basis Cs-137 concentrations for IRTS facilities are given in Table B.8.2-1. Design basis gamma curies for Main Plant cells are given in Table B.8.2-2.

B.8.2.1.2 Contained Hazardous Material Sources

Several types of hazardous materials are present in varying quantities throughout the site. The analytical, vitrification and environmental laboratories maintain inventories of a great number of reagents; however, only very small quantities of these reagents are stored at the lab site and storage is provided in a manner that precludes reaction. Due to the physical nature of the IRTS process (i.e., ion exchange [filtration], evaporation [concentration], and solidification) few bulk chemicals are required to support low-level waste processing. Bulk chemicals that are required are primarily those utilized for pH control. These chemicals, which include caustic for SMWS operations, sulfuric acid for neutralization of LLWTS effluent, and nitric acid for utility room operations, are stored at the location of use.

Temporary storage of chemicals to be used throughout the site is provided in a specially-constructed area of the New Warehouse facility. Separate areas of this facility provide storage for various quantities of caustics, acids, and oxidizers, as indicated in Table B.9.1-1. Utilization of process chemicals in the Vitrification Facility is discussed in WVNS-SAR-003.

Maintenance activities require the use of solvents, oils and other lubricants. These materials are stored in secure storage lockers at the maintenance building. Storage for gasoline and diesel fuel is provided in above ground tanks located east of the New Warehouse. Fuel oil for steam boilers and backup equipment is located in storage tanks in the Utility Room, fire pump house, yard east of the Utility Room, STS generator room, and 01-14 Building.

In addition to these hazardous materials, hazardous wastes are stored in several facilities throughout the site. These facilities include the Lag Storage Facility, Interim Waste Storage Facility, Hazardous Waste Storage Lockers and Satellite Accumulation Areas which are described in Section B.7.7.

B.8.2.2 <u>Airborne Hazards Sources</u>

B.8.2.2.1 Airborne Radioactive Material Sources

<u>Supernatant Treatment System</u>

The STS support building is vented to ensure airflow is from regions of low airborne radioactivity to areas of potentially elevated activity. Under normal conditions, clean air within the STS control room and utility areas is exhausted into the valve aisle. Airborne radioactivity in occupied areas is low; however, continuous airborne radioactivity monitors located within the STS are provided to alert personnel to elevated levels of airborne contamination. The primary source of airborne activity in the STS is filtered, dilute, wash water vapor. Airborne radioactivity levels are maintained at less than 0.02 times the derived air concentration (DAC) for all radionuclides under normal and expected abnormal conditions within normally occupied areas of the STS and SMWS control rooms.

<u>Liquid Waste Treatment System</u>

The largest potential source of airborne radioactivity in the LWTS is process solution located in process vessels in XC-3. To minimize the release of airborne contamination into the cell, LWTS vessels are ventilated by the Vessel Off-Gas system. Secondary confinement of airborne radioactivity is provided by the Main Plant ventilation system which ventilates cells in which LWTS equipment is housed. Due to these multiple confinement systems, it is not expected that airborne contamination will be released from these areas; however, a fixed-position air sampler has been placed in the Cell Access Aisle to alert if an increase in airborne radioactivity occurs. A release of airborne radioactivity would not affect LWTS operations personnel as control of the LWTS is conducted from an adjacent, independently ventilated facility.

<u>Cement Solidification System</u>

The sources of airborne radioactivity in the CSS include the radioactive process solutions in the Waste Dispensing Vessel and the mixers. Ventilation in the CSS is provided for both the confinement vessels and the cells. Due to these confinement features and the low concentrations of activity handled by this system, negligible airborne radionuclide concentrations are expected in areas occupied by operating personnel.

Drum Cell

No significant sources of airborne radioactive contamination exist in the Drum Cell.

Main Plant

Sources of airborne radioactive material in the Main Plant have been greatly reduced due to cessation of reprocessing. A small increase in the amount of airborne radioactive particulate activity results from IRTS operations, deterioration of cell penetrations and access ports from former plant operations, operations within the analytical cells, and maintenance operations. Airborne contamination is removed by high efficiency filters in the various ventilation and off-gas systems. Airborne radioactivity levels in routinely accessed areas (e.g., operating aisles and laboratories) are maintained at less than 0.02 times the Derived Air Concentrations values set forth by the Department of Energy (10 CFR 835).

Appropriate levels of respiratory protection are provided (e.g., air-purifying respirators, air-line respirators, and self-contained breathing apparatuses) whenever manned entries are made into airborne radioactivity areas. When dust generating equipment is used, local HEPA filtered airborne contamination control devices are used.

B.8.2.2.2 Airborne Hazardous Material Sources

Asbestos is present in the Waste Tank Farm, Cold Chemical Building, 01-14 Building, and some gasket material throughout the Vitrification Facility. Asbestos fibers may be released during removal or maintenance activities. Silicon dioxide is used as a glass former in the Cold Chemical Building and is released into the air during the makeup of glass formers. Lead can be released into the air during the handling of lead shielding or from the disturbance of lead-based paint. Anhydrous ammonia is stored outside the 01-14 Building and can be released to the atmosphere when the tank is venting, when bleeding off lines, and when filling the tank. Oxides of nitrogen are present in the melter off-gas and may be present in the off-gas trench and in some area of the 01-14 Building. Fumes may also be generated throughout the facility during painting and welding activities.

B.8.3 Hazard Protection Design Features

B.8.3.1 Radiation Protection Design Features

Radiation protection features basic to the design of the IRTS are dedicated to maintaining ALARA radiation exposures to members of the general public and work force. Effective control of radiation exposures depends primarily on design features that provide adequate shielding from all sources of radiation, provide for remote operations and maintenance, confinement of radioactivity within the process, proper ventilation, effluent control, and overall monitoring and surveillance to verify design controls. These physical design features, plus strict adherence to the operational requirements given in WVDP-010, *WVDP Radiological Controls Manual*, provide effective radiation control.

B.8.3.1.1 IRTS and Main Plant Design Features

All radioactive material handling and processing operations in IRTS components occurs within shielded confinement structures. Valves are designed for remote operation. Highly instrumented control rooms include visual display and visual/audible alarm systems, enabling IRTS operators to control processes from a remote location. Equipment and components in radioactive service have been designed for remote removal and replacement should failure occur. High maintenance equipment such as pumps are located in lower exposure rate areas. Ventilation systems provide assurance that materials contained within the cells are not released into operating aisles and areas. Airborne radioactive particulates are removed from the exhausted air and a lower air pressure is maintained in the more highly contaminated areas from adjoining areas. All IRTS and Main Plant areas are maintained with an air pressure differential which directs air into the more highly contaminated areas.

The high-level waste storage tanks provide multiple confinement barriers including: the tank, pan and vault, and the natural silty till surrounding the vault.

B.8.3.1.2 Shielding

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

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

Details of shielding design criteria for IRTS and Main Plant facilities are given below.

IRTS Shielding

Shielding for IRTS component facilities has been designed such that the radiation dose rate in full-time occupancy areas does not exceed 0.25 mrem/hour. This criterion applies to IRTS control rooms. Furthermore, shielding for full-time access areas in IRTS facilities such as operating aisles has been designed such that the dose rate does not exceed 2.5/t mrem/hr in which t is the maximum average time in hours per day that the area is expected to be occupied by one individual. Sufficient shielding has been provided such that the dose to maintenance personnel is less than 0.5 rem/yr for each planned maintenance task. Preoperational shielding calculations for each of the component systems confirmed the adequacy of shielding designs and routine operational area radiation measurements ensure that the design exposure rates are not exceeded. A summary of IRTS shielding evaluations is given in Tables B.8.3-1 and B.8.3-2.

Main Plant Shielding

The Main Plant building was designed to protect operating personnel from the intense radiation fields associated with handling spent nuclear fuel. Shielding for the plant was designed based on a fuel with the following irradiation history:

Burnup	30,000 MWD/MTU
Specific Power	35 MW/MTU
Cooling Time	150 Days

The gamma curies of design fuel are given in Table B.8.2-2. Due to shutdown of reprocessing activities in the plant and decontamination of plant areas, a great reduction in facility radiation levels has been achieved. Summary shielding descriptions for areas of the plant in which routine operations occur or in which high contamination levels exist are given in Tables B.8.3-3 through B.8.3-7 (National Federal Standards, 1970). Certain specific sources listed in these tables have been removed from the plant as a result of decontamination and dismantlement activities by WVNS. However, these sources have been included in the respective table as representing the basis for the original shielding calculations.

In areas of the plant where it was necessary to penetrate shielding walls with pipes, ducts, cables, etc, provisions in the original design features were made to assure the specified shielding requirements of the area. This was done by keeping the penetrations to a minimum in size and quantity, and by avoiding straight streaming paths by using multiple offsets, slopes, shadow shields, etc, as the individual situation required. Future shield penetrations will be evaluated to ensure continued radiological protection.

B.8.3.1.3 Ventilation

Facility ventilation systems are described in Section B.5.4.1. These systems have been designed to ensure contamination confinement during normal operations and to minimize the spread of contamination during abnormal operations. Several features have been included in the system designs to ensure that personnel safety is maintained. Continuous air samples are collected in routinely accessed areas such as stairwells and operating aisles and analyzed to ensure that internal exposures are maintained as low as practicable. Access to areas having elevated airborne contamination levels are restricted through administrative controls set forth in WVDP-010.

Normal airflow is from stairwells and operating aisles to cell service areas and airlocks to process cells. Manned entry to contaminated cells and areas is via airlocks. Airlocks ensure that sufficient negative pressure is maintained between contaminated and uncontaminated areas of the plant. Backup capability has been provided for facility ventilation systems in the event of a power outage. Redundancy is provided for both blowers and filter trains. Discussions of these systems have been provided in Section B.5.4.1.

Facility ventilation systems process air through a series of filters prior to discharge. The final filter in each system is a high efficiency particulate air (HEPA) filter, capable of removing 99.95 percent of aerosol particles greater than 0.3 microns diameter. To assure that facility ventilation systems are performing adequately, ventilation effluent is monitored in system stacks. Pressure differential instruments are calibrated annually, alarms and switchover capabilities are operationally tested quarterly, and HEPA filters are in-place leak tested annually.

The Waste Tank Farm ventilation system and the Vessel Off-Gas system provide ventilation for process vessels located in the Waste Tank Farm, Main Plant and LWTS. Configuration of these systems are discussed in Section B.7.4.

Additional protection from releases of activity into uncontaminated areas during cell or waste tank riser access is assured through the use of temporary confinement tents and portable or temporary ventilation equipment. These temporary confinement systems are erected and operated per approved work procedures which ensure that radiological and industrial safety controls specified in WVDP-010 and WVDP-011 are adequately implemented.

B.8.3.1.4 Radiation and Airborne Radioactivity Monitoring Instrumentation

Radiological monitoring instrumentation used at the WVDP are calibrated in accordance with ANSI N323A-1997. Most radiation detection equipment is calibrated on a six month cycle; however, some instruments are calibrated annually depending on the frequency and type of use or if calibration is performed off-site by a service vendor. Stack exhaust monitors and their calibration are discussed in Section B.8.6.

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

The continuous airborne radioactivity monitoring instrumentation used at the WVDP are the fixed filter type. Separate continuous air monitors (CAMs) are used for the detection of beta-gamma and alpha emitting radioisotopes. The type of CAM which is placed in the work area is determined by, but not limited to, the ratio between alpha and beta radioisotopes, the history of the work area, and the work to be performed.

Table B.8.3-8 provides a summary of information on the backup and/or standby power supply, range, sensitivity, accuracy, calibration frequency, alarm set points, recording devices, location of detectors, readouts and alarms for CAMs. CAMs provide local readout and alarm.

Requirements for air monitoring programs are specified in 10 CFR 835. Additional guidance is contained in DOE Guide 441.1-8, *Air Monitoring Guide*, and American

National Standards and NUREG documents referenced therein. At the WVDP, air monitoring samples are taken in locations throughout the Main Plant building to detect and evaluate airborne radioactive material at work locations and routinely accessed operating aisles. Data obtained by air monitoring is used for assessing the control of airborne radioactive material in the workplace. The WVDP has incorporated the general guidance for placement of air monitors, provided in DOE Guide 441.1-8 and NUREG-1400, into the air monitoring program.

Continuous radiation monitoring capabilities are provided to warn of undesirable trends and/or abnormal conditions. An ARM and a CAM are located in the fresh zeolite dispensing area and in the manipulator operating aisle (minimum of two ARMs and two CAMs for each area). Radiation monitoring is connected to emergency backup power.

Area radiation monitors provide an audible alarm when a preset exposure rate is reached. These instruments operate in the range of 0.1 mR/h to 1.0 R/h. Continuous airborne monitors sample air through a fixed particulate filter at flow rates of 28.3 lpm and will alarm when a preset count rate is reached. The beta CAMs instruments use open window GM detectors, which are sensitive to both beta and gamma activity. The alpha CAMs are solid-state detectors. A summary of radiation monitors in the IRTS and Main Plant is provided in Table B.8.3-9.

There are approximately 11 alpha and 22 beta-gamma continuous air monitors (CAMs) employed at fixed locations in the Main Plant, as indicated in Table B.8.3-10. Additional units (usually between 5 and 10) of both monitoring types are frequently in service in support of work activities performed in the facility. CAM filters are routinely removed from the CAMs twice per week and counted for both gross alpha and gross beta-gamma activity after short-lived radionuclides have decayed to insignificant levels (i.e., approximately one week).

B.8.3.2 Hazardous Material Protection Design Features

Facility design features have been provided to protect against exposure to hazardous materials. The New Main Warehouse contains five engineered segregated storage areas for corrosives, acids, oxidizers, flammables, and health hazards (poisons). Each storage area is equipped with a 15 cm (6-in) deep basin below a steel grate floor to contain spills. Each area also has a separate ventilation system and an automatic fire suppression system and alarm. Access to these rooms is controlled by the Warehouse Manager.

The Hazardous Waste Storage Lockers are pre-engineered lockers containing segregated 208 L (55-gal) drums with a spill basin with a capacity sufficient to contain a spill equivalent to 10% of the volume of material stored in the locker. The lockers are designed to contain flammable materials and are equipped with fire suppression devices and alarms.

The Interim Waste Storage Facility is a heated, metal, Butler-type building used for the collection, sorting, handling, sampling, and interim storage of uncharacterized wastes and suspect radiological mixed wastes. Waste oils and process chemicals are held until classified. The IWSF is equipped with a high-expansion foam fire suppression system and a sump for spill handling.

Exposure to hazardous chemical fumes produced during analytical or painting activities is minimized by conducting analytical procedures in ventilated laboratory hoods and painting activities in ventilated paint booths.

B.8.4 Estimated Collective On-site Dose Assessment

Activities associated with the Main Plant and Waste Processing Facilities include operation of the STS/SMWS/HLWTS, LWTS, CSS, and support activities including analytical and process chemistry, radiological control monitoring, routine maintenance activities and facility surveillance by security and safety personnel. Additional activities include operation of the Fuel Receiving and Storage (FRS) facility systems. Operations conducted in the FRS are discussed in WVNS-SAR-012.

Whole body exposure estimates for personnel providing support for plant activities or performing operations in the Main Plant and Waste Processing Facilities are calculated as part of the WVDP ALARA program. Annual occupational exposures from the WVDP ALARA program for these work groups are given in Table B.8.4-1. Calculations for the ALARA budget provide total exposure estimates for each work group, which are then compared to the actual exposures received. In 1999, the estimated ALARA budget was 107% of the actual dose received from 1999. Due to the nature of the work associated with plant operations (i.e., facilities support), it is not practical to distinguish between dose incurred as a result of Main Plant operations and dose incurred as a result of IRTS support operations conducted in the Main Plant. Therefore, the combined Main Plant/IRTS support dose for each work group is provided in Table B.8.4-1. Based on data from this table and the number of workers in each ALARA group, doses for workers are well within the limits set forth in 10 CFR 835 with an average annual worker exposure of 12 mrem (1.24E-4Sv).

A program of air particulate monitoring is in place for the Main Plant and Waste Processing Facilities to ensure airborne radioactivity levels in routinely occupied areas are well within acceptable limits. This is accomplished by drawing plant air at a constant rate through glass fiber filters placed in holders. These filter assemblies are placed at breathing levels in various locations inside of the Main Plant and Waste Processing Facilities. Radiological analyses of these filters indicate typical airborne radioactivity concentrations of 1E-15 µCi/mL (3.7E-11 Bq/mL) gross alpha and 1E-15 to 1E-14 µCi/mL (3.7E-11 to 3.7E-10 Bq/mL) gross beta with occasional gross beta concentrations of 1E-13 µCi/mL measured in certain areas.

Dose estimates for workers in IRTS and Main Plant facilities due to inhalation of air can be made by using conservative values for gross alpha and gross beta concentrations and assuming the most restrictive nuclides as present on the site. Assuming:

- All gross alpha activity is Am-241
- All gross beta activity is Sr/Y-90
- 2.40E+09 mL of air is inhaled per year per worker (9,600 L/day x 250 days/year)
- A gross alpha concentration of 1 E-15 µCi/mL (3.7E-11 Bq/mL)
- A gross beta concentration of 1 E-13 µCi/mL (3.7E-9 Bq/mL)

The annual estimated inhalation dose per worker is 1.6 mrem (1.6E-2 mSv). Combining this with the annual whole body exposure of 12 mrem (1.2E-1 mSv) gives an estimated annual average occupational dose of 14 mrem (1.4E-1 mSv).

B.8.5 <u>WVDP Hazards Protection Programs</u>

B.8.5.1 <u>WVDP Health Physics Program</u>

A formally documented health physics program for the WVDP has been established in WVNS Policy and Procedure WV-905, *Radiological Protection*. The health physics program is based on requirements set forth in DOE Order 5400.5 and 10 CFR 835. At the WVDP, the health physics Program's site-specific requirements are promulgated in WVDP-010.

IRTS and Main Plant facilities are operated in compliance with the requirements given in WVDP-010. The health physics program for the Project is discussed and summarized in Section A.8.5 of WVNS-SAR-001.

B.8.5.2 <u>WVDP Industrial Hygiene and Safety Program</u>

The WVDP Industrial Hygiene and Safety Program is presented in Section A.8.7 of WVNS-SAR-001.

B.8.6 Estimated Collective Off-site Dose Assessment

B.8.6.1 Effluent and Environmental Monitoring Program

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

B.8.6.1.1 Gas Effluent Monitoring

Currently there are six fixed ventilation stacks that are permitted through the Environmental Protection Agency at the WVDP. The Main Plant ventilation stack monitor and sampler equipment is housed in an insulated building located south of the Main Plant stack base on the Ventilation Exhaust Cell roof. Ventilation air exhausted from the STS is released from the STS PVS stack located at the WTF. Monitoring equipment for the PVS is located in a dedicated structure that is adjacent to the PVS building. The CSS ventilation stack is located on top of the 01/14 Building with sampling and monitoring equipment located inside of the 01/14 Building. The CSRF ventilation stack is located on top of the Main Plant about 25 m (82 ft) north of the Main Plant stack. Ventilation air from the Container Sorting and Packaging Facility (CSPF) is exhausted through the ventilation stack located along the south wall of Lag Storage Area #4. The vitrification facility heating, ventilation and air conditioning system (Vit HVAC) directs the flow of air through the vitrification facility. The Vit stack is located on the west side of the Vitrification Building. Portable ventilation units (PVUs), also known as outdoor ventilation enclosures (OVEs), are also permitted through the Environmental Protection Agency. These units are operated at various locations around the WVDP and provide ventilation to support temporary activities in areas which are not routinely ventilated and where the potential for airborne contamination exists.

Isokinetic air samples from these stacks are continuously drawn and transported to the sampling and monitoring instruments. The sample streams pass through glass fiber particulate filters and charcoal cartridges before returning back to the stack for discharge to the environment. The filters are changed and screened weekly for gross radioactivity. Quarterly composites from both the glass fiber and charcoal filters are analyzed for gamma isotopes, Sr-90, I-129 and actinide isotopes. Currently there are no requirements for monitoring for nonradiological parameters.

Continuous air particulate monitors provide alarm indications should radioactive particulate levels in the exhaust air exceed preset levels. Flow and count-rate sensors will activate a backup vacuum pump and various alarms if equipment failures occur. The systems are provided with auxiliary backup power. Monitoring capabilities are given in Table B.8.3-9.

B.8.6.1.2 Liquid Effluent Monitoring

Liquids that are generated by plant activities are processed through the Low Level Waste Treatment System (LLWTS) before discharge to the environment. Waste liquids are processed in batches to allow greater control and the ability to re-process any liquid that is found to be out of specification. Lagoons 4 and 5 receive effluent from the LLWTS. When the on-line lagoon becomes full, the alternate lagoon is placed on-line and the full lagoon is placed off-line and is sampled for radiological analysis. Upon determination of acceptable radiological levels, the lagoon is drained by gravity to a larger holding basin (Lagoon 3). This procedure continues until Lagoon 3 becomes full. Batch transfers from Lagoons 4 and 5 are then curtailed and Lagoon 3 is sampled for radiological analysis. Upon determination of acceptable radiological levels, the lagoon is discharged to Erdman Brook in a controlled manner with a constant flow. The duration of each discharge event is approximately one week. A lagoon batch is discharged to the environment at intervals of approximately two to three months. Composite and grab samples are collected during each discharge event and analyzed for radiological and non-radiological constituents.

B.8.6.2 <u>Analysis of Multiple Contribution</u>

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

B.8.6.3 <u>Estimated Exposures from Airborne Releases</u>

Airborne emissions result from the ventilation of waste processing vessels, waste storage tanks, contaminated cells in the Main Plant and areas within the Waste Processing Facilities. Ventilation air is filtered prior to discharge to the stacks. The Main Plant stack serves as the discharge point for the main ventilation system, the Head End Ventilation system, the Waste Tank Farm Ventilation System, the Vessel Off-Gas System, and the Fuel Receiving and Storage facility ventilation exhaust. Other smaller stacks (STS, CSS, CSRF, and CSPF) are in close proximity to the Main Plant stack and were used to calculate the exposure from airborne releases.

Dose assessments for discharges from the stacks identified above in addition to the contribution from Portable Ventilation Units (PVUs) were performed. The total airborne activity released per year from the Main Plant, STS, 01-14 Building, CSPF, and CSRF stacks is listed in Table B.8.6-1.

The total effective dose equivalent (TEDE) to the maximally exposed off-site individual (MEOSI) in 1999 was calculated to be 1.1E-02 mrem/yr (1.1E-4 Sv/yr) for airborne discharges from all stacks. The MEOSI is located at approximately 2400 m (7874 ft) east of the Main Plant stack.

B.8.6.4 Estimated Exposures from Liquid Releases

An estimate of the dose to the maximally exposed off-site individual has been calculated using analytical data obtained from lagoon discharge sampling. Table B.8.6-2 lists the dose contributions from isotopes contributing greater than 0.1% to the total dose. The dose to the maximally exposed off-site individual for Main Plant liquid discharges in 1999 was 2.8E-05 rem (2.8-07 Sv).

The WVDP also operates a Waste Water Treatment Facility that processes nonradiological liquid effluents generated primarily from site lavatory facilities. Sewage is processed and collected in a lined holding basin for subsequent discharge to the environment. This effluent is sampled and analyzed for gross alpha and beta activity. This discharge is not considered a radioactive effluent and radiological analyses are used for confirmatory purposes only.

B.8.7 Prevention of Inadvertent Criticality

B.8.7.1 Introduction

Operations involving the handling, processing or storage of fissile materials are evaluated for criticality safety. Criticality safety at the WVDP is achieved primarily through the application of strict administrative controls. Evaluations have shown that there currently is no credible potential for an inadvertent criticality associated with IRTS operations. The potential for a criticality in the General Purpose Cell of the Main Plant does exist.

B.8.7.2 Requirements

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

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

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

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

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

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

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

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

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

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

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

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

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

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

B.8.7.3 Criticality Concerns

Processing activities in the IRTS require the handling and treatment of solutions containing fissionable materials. These processes have the potential to concentrate fissionable materials in the feed solution and consequently present a criticality concern. Specific IRTS processes that have the potential to concentrate fissionable materials include ion exchange in the STS and evaporation in the LWTS evaporator.

Solidified waste cement drums that were produced as a result of original IRTS operations and that are currently stored in the IRTS Drum Cell contain less than 100 nCi of transuranic nuclides per gram of cement. Even if it is assumed that Pu-239 is the only transuranic nuclide, these cement drums are subcritical by a wide margin.

In addition to IRTS facilities and processes, areas of the Main Plant and Lag Storage Facility may contain significant quantities of fissionable material that could present a criticality concern. (Although characterization sufficient to confirm the presence of significant quantities of fissionable materials in these areas has not been performed, it has been assumed that significant quantities do exist based on the best available information.) Areas of the Main Plant that currently contain significant quantities of fissionable material contamination remaining from NFS reprocessing activities principally include the head end cells (i.e., the PMC and GPC) although other areas of the plant, including XC-1 and XC-2, may contain lesser accumulations.

The Lag Storage Facility serves as a temporary storage location for low level and transuranic wastes produced on-site. By definition transuranic (and suspect transuranic) wastes have the potential for containing fissionable materials. Consequently, storage activities in the Lag Storage facility have to be evaluated for criticality concerns.

A more detailed discussion of the criticality concerns for the IRTS, Main Plant, and Lag Storage Facilities is given in the following subsections.

B.8.7.3.1 STS/SMWS Criticality Concerns

Fissionable material in IRTS component systems derives from high-level waste in Tank 8D-2. Table B.8.7-1 presents the total fissionable material inventory of the sludge in Tank 8D-2, the soluble mass present in the first wash solution, and the maximum ion exchange column inventory at cesium breakthrough for the first wash.

Prior to SMWS startup, a comprehensive evaluation of the criticality safety of major vessels and components in the STS was performed. For each major vessel and component the safe concentration or total mass of plutonium was determined assuming the vessel or component to be filled to capacity. The evaluations conservatively assumed the sludge wash solution to be Pu-239 in water with no credit taken for neutron absorption by other nonfissile isotopes or neutron poison materials. Based upon the results of these criticality safety evaluations (Caldwell, 1990; Yuan, 1991), the allowable safe fissile material concentration for the SMWS process was established considering the vessel or component having the minimum critical concentration (i.e., the most restrictive vessel). It was determined that approximately 1.0 kg of Pu-239 loaded onto Ti-treated zeolite within an ion exchange column in a sphere of radius 22.5 cm would result in a $k_{eff} + 2\sigma$ of 0.95 and thus identified the ion exchange columns as the most restrictive vessels. Assessment of the criticality safety evaluations has shown that criticality during STS operations is not credible under normal and abnormal operating conditions (Prowse, 1992).

A summary of the evaluations for individual vessels and components in the STS/SMWS is given below.

Sludge Mobilization in Tank 8D-2

Suspension of the sludge in Tank 8D-2 by the sludge mobilization pumps was specifically analyzed (Caldwell, 1990). The analysis indicates that under conditions of homogenous mixing, as is observed during SMWS operations, an inventory of plutonium and uranium ten times that found in Tank 8D-2 is critically safe. Therefore, Tank 8D-2 can be considered critically safe under normal and expected abnormal conditions based on fissile material concentration.

Zeolite Mobilization in Tank 8D-1

A large margin of criticality safety is maintained in Tank 8D-1 by using the zeolite mobilization pumps to distribute the spent ion exchange zeolite across the geometrically favorable tank bottom. Tests on zeolite distribution within Tank 8D-1 have been performed using a one-sixth scale model (Schiffhauer, 1987) and the results indicate that the zeolite pile is effectively distributed by operating the mobilization pumps. These pumps are operated during and/or following each discharge of zeolite from STS ion exchange columns. This provides adequate mixing and distribution of zeolite.

STS Prefilters

The prefilter is intrinsically safe. Each prefilter is 305 cm (120-in) long with an outside diameter of 10 cm (4 in). This large surface to volume ratio results in neutron leakage sufficient to prevent the $k_{eff} + 2\sigma$ from exceeding 0.95. The rigid 56 cm (22 in) spacing between prefilters ensures that the prefilters are neutronically decoupled from each other.

STS Ion Exchange Columns

The STS ion exchange columns, shown in Figure B.8.7-1, have been determined to be the most restrictive components of the STS due to geometry and process function considerations. The calculation of k_{eff} for several configurations of plutonium loaded zeolite inside the STS ion exchange column was made using the KENO-V Monte Carlo code. Material parameters and cross section sets used for these calculations are given in Table B.8.7-2. The results of parametric evaluations of k_{eff} versus geometry and composition is provided in Table B.8.7-3. These evaluations were made using the KENO-V code and various cross section data sets compiled at the Argonne National Laboratory's IBM mainframe computer systems (Yuan, 1991) and independently verified using TWODANT compiled at the Los Alamos National Laboratory's Cray mainframe computer system (Prowse, 1991). All differences between the calculational model and the actual configuration result in a conservative overestimate of k_{eff}.

Values of k_{eff} were calculated for various size spheres centered in the ion exchange column and having a total mass of 1.0 kg Pu-239 homogeneously distributed within the sphere. (This configuration represents the optimum geometry for a fixed mass confined within a cylinder.) Tabulation of the results of these calculations are given in Table B.8.7-4 while a plot of k_{eff} versus the sphere radius is shown in Figure B.8.7-2. Based on these calculations it was determined that the greatest k_{eff} occurs for a sphere having a radius of approximately 22.5 cm.

To evaluate the limiting Pu-239 mass inside the zeolite column, additional calculations were performed with the same geometric configuration but varying the mass of Pu-239 in the sphere. Results of these calculations are tabulated in Table B.8.7-5. A least squares fit of this data has been plotted in Figure B.8.7-3. It is concluded from this figure that a limit of 1.0 kg of Pu-239 inside the zeolite column will ensure that a $k_{eff} + 2\sigma$ of 0.95 is not exceeded.

The heel remaining from ion exchange column sparging will remain critically safe during normal and accident conditions. The current heel does not contain Ti-treated zeolite and the mass of Pu which could accumulate in the heel is very small.

Comparison of the reactivity of 1.0 kg of Pu-239 within various cylindrical and spherical geometries constrained to the geometry of the ion exchange vessel heel reveals that a cylinder represents the most reactive configuration. As can be seen from Table B.8.7-3, a cylinder 20 cm (8 in) high having a 23 cm (9 in) radius centered in an ion exchange column uniformly loaded with 1.0 kg of Pu-239 is subcritical (Yuan, 1991). As has been shown in Table B.8.7-4 and Figure B.8.7-3, this mass of Pu-239 will remain subcritical even when placed in an optimum geometry (22.5 cm radius sphere).

B.8.7.3.2 LWTS Criticality Concerns

Decontaminated sludge wash solutions are concentrated in the LWTS. Previous experience in the LWTS revealed the accumulation of U and Pu in scale deposits in the evaporator following extended periods of evaporator operation. These deposits are removed through the acid wash procedure described in Section B.6.4.1.3. Calculations by Yuan (1991a) and Caldwell (1991) considered 460 g total of Pu-239 plus Pu-241, 1.88 kg of U-235 and 105.9 kg of U-238 dissolved in water in the bottom of the evaporator (reduced right circular cylinder of radius 60 cm (24 in) and associated height to optimize density and neutron leakage). The results confirmed that the k_{eff} + 2σ for the fissile material configuration could not exceed 0.95. Parks and Dyer (1991) provided an independent validation of the analytical approach. Material mass balance procedures ensure that quantities of fissile plutonium and uranium that accumulate in the evaporator do not exceed bounding masses established by criticality analyses.

B.8.7.3.3 Main Plant Criticality Concerns

The head end cells of the Main Plant (i.e., the PMC and GPC) are currently believed to contain significant quantities of fissile material contamination remaining from NFS reprocessing activities. Elevated radiation levels in the cells and video inspections are the bases for this belief. Although significant accumulations are primarily restricted to the head end cells, other areas of the plant may contain lesser accumulations (e.g. XC-1). This material is critically safe in its current configuration; however, reconfiguration of the material into a different geometry may result in a more reactive condition. Future activities which could modify the configuration of the contaminated debris will be evaluated to ensure activities are performed in a critically safe manner.

Although stabilization and decontamination activities that could increase the reactivity of the accumulated material will not be permitted, cells that contain the greatest fissionable material accumulations as contamination have been provided with sufficient shielding such that the whole body exposure to workers in adjacent operating aisles due to a criticality in the cell would not exceed the whole body exposure limit given in Section B.9.1.3.

The General Purpose Cell currently contains a significant accumulation of contaminated hardware and other material, including fuel hulls, which remain from NFS reprocessing activities. Insufficient cell characterization data has resulted in varying estimates of the fissile material content of this cell (Wolniewicz, 1993 and WHC, 1993). Estimates of the quantity of fissile material in the PMC and GPC based on calculations given in Wolniewicz, 1993 are provided in Table B.8.7-6. The WHC estimate was performed as part of a larger study to evaluate the potential for criticality in the cell.

Through the use of highly conservative assumptions, an analysis by Westinghouse Hanford Corporation (WHC) (WHC, 1993) determined that under appropriate conditions of fuel inventory, enrichment, configuration and moderation a criticality in the cell sump area is possible. A subsequent analysis performed by WHC (WHC, 1996) using more realistic assumptions found that a credible potential for criticality does not exist under the current material configuration, even under the condition of full water moderation. Nevertheless, accepting that under the conditions assumed in the initial WHC analysis a criticality in the cell could be possible, subsequent analyses were performed to determine the occupational and environmental impacts of a criticality in the cell (Wolniewicz, October 1993). An analysis, based on the guidelines given in NRC Regulatory Guide 3.33, *Assumptions used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Fuel Reprocessing Plant*, indicated that an individual in the adjacent operating aisle would receive an approximate whole body exposure of 98 mrem due to a 2E+19 fission event. An additional analysis found that there would be little environmental impact due to the fission gas release (approximately 1 mrem to the maximally exposed off-site individual).

B.8.7.3.4 Lag Storage Facility Criticality Concerns

The Lag Storage Facility, comprised of the Lag Storage Building, Lag Storage Annexes (LSA)-1, -3, -4, and Chemical Process Cell Waste Storage Area (CPC-WSA), provides temporary storage of wastes generated during Project activities. Storage of transuranic (TRU) and suspect TRU wastes, which, by definition, contain fissionable materials, is only permitted in the Lag Storage Building and CPC-WSA.

The Lag Storage Building (see Section B.7.7) receives wastes from current Project activities and provides storage for wastes generated during original facility decontamination and other activities. Matrix materials of the TRU wastes received in the Lag Storage Facility vary widely, from paper anti-contamination clothing materials to sludges and resins. The ratio of fissionable nuclides in these wastes also varies due to a process conducted by NFS to affect the chemical separation of uranium from plutonium. Analyses by O-Ahoofe (Keel, 1984) considered these variables in determining fissionable material mass limits for waste packages received for storage in the Lag Storage Facility. These limits ensure that under no circumstances will reconfiguration of the storage containers (up to a height of four tiers) result in an array having a $k_{eff} + 2\sigma$ greater than 0.95. (The container stacking limit of 4 tiers is not meant to infer that an array height of 5 tiers would represent an unsafe condition. It was selected in the analysis due to the fact that the physical dimensions of the storage facility would not permit an array higher than 4 tiers. Although a fifth tier may be safe, this has not been determined analytically and administrative controls are in place to ensure that the array height does not exceed 4 tiers.) Based on the administrative controls for fissionable materials in containers accepted for storage in the Lag Storage Building and the fact that re-packaging of TRU wastes in the Lag Storage Building is not permitted, it is concluded that a criticality in the Lag Storage Building is incredible.

The Lag Storage Facility complies with criteria associated with the Double Contingency Principle by maintaining administrative control on two independent process parameters: the concentration of fissile material in the waste containers, and the height of the array of waste containers. (Memo FD:99:0049, WVNS, June 29, 1999, and Memo FD:99:0055, WVNS, August 4, 1999.)

The CPC-WSA is utilized for the storage of high activity wastes removed from the former Chemical Process Cell in the Main Plant building and no further additions to the inventory of material in the CPC-WSA are anticipated. Based on the best available information, it has been determined that less than a significant quantity of fissionable material exists in the entire CPC-WSA facility. These estimates have been made based on indirect measurements and historical information (i.e., dose to curie conversions). This historical evidence also indicates that significant quantities of uncharacterized sludges exist in some of the removed equipment. In the current storage configuration this material is critically safe; however, prior to any subsequent decontamination activities that could concentrate the sludges in these vessels, it will be necessary to confirm the concentration of fissionable materials to ensure that criticality safety will be maintained.

B.8.7.4 Criticality Controls

Criticality controls at the WVDP are developed through the guidelines given in WVDP-162 and the references contained therein. Administrative controls are the primary means for criticality control in IRTS systems and Main Plant facilities.

B.8.7.4.1 Engineering Controls

Engineered features designed to prevent an inadvertent criticality are not provided in IRTS systems or Main Plant facilities. The geometry of containers utilized for storage of TRU waste in the Lag Storage Building ensures that a safe concentration of fissionable materials (i.e., mass per unit volume) is maintained; however, the containers themselves are not critically safe. Prevention of inadvertent criticality in the IRTS, Main Plant, and Lag Storage Facility is achieved through the use of administrative controls.

B.8.7.4.2 Administrative Controls

Administrative controls developed through the guidelines and requirements given in WVDP-162 are the primary means for criticality control in the IRTS, Main Plant, and Lag Storage Facilities. These controls ensure that activities that require the storage, processing or handling of fissile or fissionable materials are performed in a manner that provides an acceptable margin to safety for the prevention of an inadvertent criticality.

Accessible areas of WVDP facilities for which administrative controls must be maintained to preclude an inadvertent criticality as a result of the form, quantity or concentration of stored fissile or fissionable material are designated as a criticality control zone. Criticality control zones are posted to indicate a definite boundary and provide a means of accounting for and controlling fissionable material inventory in the designated location. Administrative controls placed on activities conducted in these areas ensure that amounts of moderating material are minimized, that procedures for work involving fissionable material are reviewed by a criticality safety engineer and that fissionable material in an unmoderated criticality zone is maintained as such.

Administrative control limits have been established to ensure that activities conducted within WVDP facilities maintain a wide margin to criticality. Specific process control limits have been developed for the concentration of fissionable materials in liquids transferred between tanks in the Main Plant/IRTS; for the mass of fissionable materials in TRU waste containers stored in the Lag Storage Building;

and, for the mass of fissionable materials (as CPC floor debris) permitted for storage in the GPC. These controls establish limits, dictate surveillance requirements to ensure compliance with the limits, and provide corrective actions for circumstances when it is discovered that the limits are not met.

Ion exchange and evaporation activities conducted within the IRTS have the potential for concentrating fissionable materials. The WVDP has developed a surveillance program to ensure that accumulations of fissionable materials in these systems are monitored. Procedures specify parameters that are measured to determine the loss of fissionable material mass across the STS ion exchange columns and the LWTS evaporator. These procedures ensure that predetermined fissionable material mass limits for these components are not exceeded.

In addition to these controls, criticality safety has been incorporated into the WVDP Integrated Safety Management (ISM) program. ISM hazard screening requirements ensure that activities that involve the handling, storage, transfer, disposal, or processing of fissionable materials, or activities conducted in or potentially impacting areas of the WVDP known to contain fissionable materials are reviewed by a criticality safety engineer.

B.8.7.4.3 Application of Double Contingency

WVDP-162 has been written to ensure that the double contingency principle requirements set forth in DOE O 420.1, Attachment 2, are incorporated into all criticality control elements for activities conducted at the WVDP.

B.8.7.5 <u>Criticality Protection Program</u>

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

B.8.7.5.1 Criticality Safety Organization

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

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

B.8.7.5.2 Criticality Safety Plans and Procedures

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

B.8.7.5.3 Criticality Safety Training

A criticality safety training program has been developed at the WVDP in accordance with the requirement of DOE Order 5480.20A. As indicated in Section A.10.3 of WVNS-SAR-001, criticality safety training is given to individuals who operate, maintain, and/or supervise activities in areas where significant quantities of fissionable materials are stored or handled. Elements of the training program require that each individual receive instruction in nuclear criticality safety including a summary of criticality accident history and nuclear criticality theory, normal procedures, radiation control practices, configuration control, criticality control zones, procedural compliance, and individual responsibility.

B.8.7.5.4 Determination of Operational Nuclear Criticality Limits

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

Safety margins for all calculations performed for WVDP activities and systems are established such that the calculated effective neutron multiplication factors, including all computational uncertainties for a unit, array of units, or systems containing fissionable material is no greater than 0.95, within a 95 percent probability and 95 percent confidence level (i.e., $k_{eff} + 2\sigma \leq 0.95$, where σ is the uncertainty associated with the method of calculation).

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

B.8.7.5.5 Criticality Safety Inspection/Audits

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

B.8.7.5.6 Criticality Infraction Reporting and Follow-Up

Occurrence reporting requirements dictated by DOE O 232.1A, *Occurrence Reporting and Processing of Operations Information*, are implemented at the WVDP through WVNS Policy and Procedure WV-987, *Occurrence Investigation and Reporting*, and WVDP-242. This procedure establishes a system for determining, evaluating, reporting, and correcting occurrences.

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

B.8.7.6 Criticality Instrumentation

DOE O 420.1, Facility Safety, requires that facilities in which the mass of fissionable material exceeds the limits established in paragraph 4.2.1 of ANSI/ANS-8.3-1986, and the probability of a criticality accident is greater than 10^{-6} per year, a criticality alarm system (CAS) shall be provided to cover occupied areas in which the expected dose exceeds 12 rads in free air. For those occupied areas in which the expected dose rate is not anticipated to exceed 12 rads in free air, a criticality detection system (CDS) shall be provided. For DOE purposes, a CAS is defined to be a criticality accident detection device and a personnel evacuation alarm, while a CDS is defined to be an appropriate criticality accident detection device but without an immediate evacuation alarm.

Analyses referenced in Section B.8.7.3 have demonstrated that, although the mass of fissionable material exceeding the limits established in paragraph 4.2.1 of ANSI/ANS-8.3-1986 does exist in certain facility areas, the credible potential for an inadvertent criticality in the IRTS or Main Plant does not currently exist. DOE O 420.1 states that under those circumstances in which an inadvertent criticality accident is determined to be incredible due to the physical form of the fissionable material, or the probability of occurrence is determined to be less than 10^{-6} per year, neither a CAS nor a CDS is required. Nevertheless, analyses of Main Plant stack effluents for Sr-89 and Cs-137 are performed to detect the occurence of a criticality, should one occur. (Calculations have shown that elevated levels of Cs-137 *and* the presence of Sr-89 in stack effluent samples would serve as indicators of an inadvertent criticality [Crotzer, 1994].) Although these analyses are not required, they are performed as a good management practice.

B.8.8 Fire Protection

Fire Hazard Analyses (FHAs) have been and are currently being conducted to comprehensively and qualitatively assess the fire risk within individual fire areas comprising the facilities on-site. A complete discussion of the FHA process and its requirements are given in WVDP-177. DOE Order 420.1 states that FHAs shall be developed for "all nuclear facilities, significant new facilities, and facilities that represent unique or significant fire safety risks." The subject Order also states that FHAs shall be developed using a graded approach. WVNS's proposed approach to performing FHAs for WVDP facilities in accordance with DOE Order 420.1 requirements (Jablonski, 1998) was accepted by OH/WVDP (Provencher, 1998). FHAs that have been developed for nuclear facilities within the scope of this SAR are as follows:

- WVNS-FHA-011, *Fire Hazard Analysis Main Process Plant*, which also covers the Fuel Receiving and Storage Facility, Waste Reduction and Packaging Area, Contact Size Reduction Facility, and Liquid Waste Treatment System. Two addenda to WVNS-FHA-011 have been issued to cover the FRS and the Head End Cells.

- WVNS-FHA-013, *Fire Hazard Analysis Cross-Reference STS/PVS Facilities*

The two FHAs listed above document that there are no "open findings/audit items" or "requirements" (i.e., actions required to correct fire protection deficiencies as regards compliance with mandatory fire protection requirements). The FHAs conclude that the facilities evaluated in them either meet or exceed both the DOE property loss requirements and the Life Safety requirements for special-purpose industrial occupancies or industrial occupancies, as applicable.

An FHA for the Chemical Process Cell Waste Storage Area, a Category 3 nuclear facility, is under development. The "fire service main," which includes site-wide fire water supply, storage, and distribution, is discussed in WVNS-FHA-004, *Fire Hazard Analysis Fire Service Main/Hydrant System*. WVDP-319, *Facility Fire Assessments*, provides fire assessments for various WVDP facilities.

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

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TABLE B.8.2-1

DESIGN BASIS Cs-137 CONCENTRATIONS FOR IRTS FACILITIES

Facility	Design Basis Cs-137 Concentration (µCi/mL)
Supernatant Treatment System	
High-level waste	1,530 [1]
Liquid Waste Treatment System	
Decontaminated HLW	50.0
Cement Solidification System	
Waste Stream	6.0
Cement Drum	3.7 [2]
Drum Cell	
Cement Drum	3.7 [2]

Notes:

[1] Evaluation basis concentration based on first sludge wash Cs-137 concentration

[2] Concentration is 1 Ci Cs-137/drum

TABLE B.8.2-2

**GAMMA CURIES OF DESIGN FUEL FOR THE MAIN PLANT BUILDING*
Basis 1000 kg Uranium 150 Day Cooling**

Isotope	Total Curies	Energy of Gamma Photons MeV	Gamma Photons per Second
Cs-137	9.2E4	0.66	3.1E15
Pm-147	3.2E5	0.12	1.2E15
Eu-155	4.7E3	0.15	1.4E14
Ru-106	6.8E4	2.42	5.0E12
		1.54	1.2E13
		1.05	5.0E13
		0.88	7.5E12
		0.62	2.5E14
		0.50	5.0E14
Ce-144	1.1E6	2.20	4.0E14
		0.70	4.0E14
		0.13	1.0E16
Zr-95	3.8E5	0.75	2.8E16
		0.23	2.8E14
Y-91	2.9E5	1.20	3.2E13
Sr-89	2.1E5	0.92	7.7E11
Ru-103	6.7E4	0.61	1.7E14
		0.50	2.2E15
Ce-141	6.9E4	0.14	1.8E15
La-140	5.2E2	3.00	1.9E11
		2.50	1.0E12
		1.50	1.8E13
		0.80	5.6E12
Gases			
Kr-85	1.3E4	0.52	4.7E12
I-131	2.2	0.72	2.2E9
		0.64	7.4E9
		0.36	6.9E10
		0.16	5.6E8

* Ref: Table 6.60a NFS, 1970. Gamma photons of Ru-106 are actually those of the daughter isotope, Rh-106.

TABLE B.8.3-1

SUMMARY OF STS SHIELDING CALCULATIONS

STS PIPEWAY WALLS - SHIELDING ANALYSIS

(91 cm Thickness of Concrete)

SUPERNATANT STREAM - 6210 µCi/mL Cs-137 Source

Concrete Density g/cc	Exposure Rate mR/hr
2.24	0.477
2.35	0.243
2.50	0.097
2.85	0.011
3.00	0.004
3.20	0.001

STS PIPEWAY ROOF - SHIELDING ANALYSIS

(91 cm Thicknesses of Concrete)

SUPERNATANT STREAM - 6210 µCi/mL Cs-137 SOURCE

Concrete Density g/cc	Exposure Rate mR/hr
2.24	0.450
2.35	0.227
2.50	0.090
2.85	0.010
3.00	0.004
3.20	0.001

TABLE B.8.3-1 (concluded)

SUMMARY OF STS SHIELDING CALCULATIONS

STS VALVE AISLE - FRONT WALL - SHIELDING ANALYSIS

(30 cm [12 in] Thicknesses of Steel)

76, 189, 379 L (20, 50, 100 gals) @ 230 MBq/mL (6210 μCi/mL) ^{137}Cs
SUPERNATANT SOURCE

Front Wall cm (in)	μSv/h 76 L	μSv/h 189 L	μSv/h* 379 L
30 (12)	5.900E-01	1.480E-00	2.950E-00

SOURCE GEOMETRY: (1.5 m [5 ft] X 1.8 m [6 ft] X 1.5 m [5 ft])

STS VALVE AISLE - SIDE WALL - SHIELDING ANALYSIS

(30 cm [12 in] Thicknesses of steel)

76, 189, 379 L (20, 50, 100 gals) @ 230 MBq/mL (6210 μCi/mL) ^{137}Cs
SUPERNATANT SOURCE

Side Wall cm (in)	μSv/h 76 L	μSv/h 189 L	μSv/h* 379 L
30 (12)	5.600E-01	1.420E-00	2.830E-00

SOURCE GEOMETRY: (1.5 m [5 ft] X 1.8 m [6 ft] X 1.5 m [5 ft])

STS VALVE AISLE ROOF - SHIELDING ANALYSIS

(30 cm [12 in] Thicknesses of Steel)
76, 189, 379 L (20, 50, 100 gals) @ 230 MBq/mL (6210 μCi/mL) ^{137}Cs
SUPERNATANT SOURCE

Roof cm (in)	μSv/h 76 L	μSv/h 189 L	μSv/h* 379 L
30 (12)	2.400E-02	52.980E-02	1.180E-01

SOURCE GEOMETRY: (1.5 m [5 ft] X 1.8 m [6 ft] X 1.5 m [5 ft])

* 1 μSv/h = 0.1 mrem/h

TABLE B.8.3-2

RESULTS OF LWTS SHIELDING ANALYSES [1]

Location	Elevation (ft)	Calculated [2] Exposure Rate (mR/hr)	Maximum [3] Exposure Rate (mR/hr)
ULO Pump Room - North Wall	100	2.48	2.5
ULO Pump Room - South Wall	100	2.24	2.5
UPC - N & S Walls	100	2.30	2.5
UPC - West Wall	100	1.00	2.5
UPC - East Wall	100	N/A	2.5
Pipe Chase	131	2.00	2.5

[1] - Performed by Ebasco (1985)

[2] - Calculated exposure rate with additional shielding in place

[3] - Per LWTS Design Criteria

N/A - Not Available

TABLE B.8.3-3

SHIELDING SUMMARY
FOR SOURCE AREA - PROCESS MECHANICAL CELL*

Specific Source - Three fuel assemblies in the following locations in cell:
1 at Disassembly Table 2V-11
1 at Disassembly Saw 2V-2
1 in Feed Mechanism of Shear 2V-4

Shielded Area	Elev. (feet)	Design Exposure Rate mR/hr	Shielding			Remarks
			Thick (m)	Mat'l[1]	C.F.[2]	
Lab Area	131	0.2	1.8	OC	A	
Vent Supply Room	114	1	1.7	OC	A	
Mech. Oper. Aisle	100	1	1.7	OC	A	
Chem. Oper. Aisle	114	1	1.7	OC	A	
Miniature Cell	76	10	1.7	OC	C	
Mech. Crane Room	114	100	0.9	OC	A	
General Purpose Cell	76	100	1.7	OC	C	
Vent Wash Room	114	1	1.5	OC	A	
Ram Equip Room	100	1	1.7	OC	A	

1) OC = Ordinary Concrete
HDC= High Density Concrete

2) C.F. = Controlling Factor

A. Thickness required to attenuate normal operating activity levels in source area to design dose specified.

B. Thickness required to attenuate 1/10 normal operating activity levels in source area to design dose specified, for maintenance operations.

C. Thickness provided is dictated by structural reasons and is greater than thickness required for shielding purposes.

D. Thickness arbitrarily set for shielding of a criticality incident.

* Ref: NFS, 1970

TABLE B.8.3-4

SHIELDING SUMMARY
FOR SOURCE AREA - GENERAL PURPOSE CELL*

Specific Source - 6 full scrap drums
6 full fuel baskets in rack 2V-35
1 full fuel basket in transport

Shielded Area	Elev. (feet)	Design Exposure Rate mR/hr	Shielding			Remarks
			Thick (m)	Mat'l[1]	C.F.[2]	
Miniature Cell	76	5	1.3	HDC	C	
Miniature Cell	76	5	1.1	HDC	A	
GPC Oper. Aisle	76	1	1.2	HDC	A	3
GPC Oper. Aisle	76	10	0.9	HDC	A	4
GPC Oper. Aisle	76	10	0.9	OC	A	5
GPC Crane Room	73	100	1.1	OC	A	6
GPC Crane Room	73	100	0.8	HDC	A	6 & 8
Mech Oper. Aisle	100	1	1.7	OC	A	
Scrap Removal	100	10	1.7	OC	C	
Scrap Removal	100	200	0.3	Steel	B	7

1), 2) See Table B.8.3-3
3) Below El 87 feet
4) Near transfer hatches and above El 87 feet
5) Above El 87 feet
6) 10 mR/hr in shutdown status (ie no fuel in transport)
7) Hatch 2V-28
8) Door 2M-7

* Ref: NFS, 1970

TABLE B.8.3-5

SHIELDING SUMMARY
FOR SOURCE AREA - CHEMICAL PROCESS CELL*

Sp.Source - Vessels 3C-1, 3C-2, 3C-3, 3D-1, 7C-4, 7D-10, 7C-4, 7C-1, 7C-2 Plus one fuel basket being loaded, or three dissolvers fully loaded and 1000 R/hr of background.

Shielded Area	Elev. (feet)	Design Exposure Rate mR/hr	Shielding			Remarks
			Thick (m)	Mat'l[1]	C.F.[2]	
Scrap Removal	100	10	0.5	Steel	A	
Mech Oper. Aisle	100	10	1.8	OC	A	
Liquid Waste Cell	100	100	1.8	OC	C	
Off-Gas Cell	100	100	1.5	OC	C	
Office Building	100	1	1.8	OC	A	
Equip. Decon. Room	100	10	1.2	OC	A	
Chem. Oper. Aisle	114.5	1	1.8	OC	A	
Off-Gas Cell	114.5	100	1.5	OC	A	
Chem Viewing Aisle	114.5	1	1.5	OC	C	
Equip. Decon. Room	114.5	10	1.2	OC	A	
Chem Viewing Aisle	117	1	0.6	HDC	A	3
Chem. Crane Room	131	300	0.9	OC	A	
Chem. Crane Room	131	100	0.9	OC	C	
Analytical Decon.	131	1	1.0	OC	A	
Laboratories	131	1	1.0	HDC	A	
ARC-OGC Aisle	131	1	0.9	HDC	A	
Office	131	1	1.0	HDC	A	4
Office	131	1	1.0	OC	A	5
Process Chem. Room	148	1	1.5	OC	A	
Hot Acid Cell	148	100	1.5	OC	C	
Vent Exhaust Cell	148	10	1.5	OC	C	
Roof	148	10	1.5	OC	C	
General Purpose Cell	100	100	1.7	OC	C	
GPC Crane Room	100	10	1.1	OC	A	6

1), 2) See Table B.8.3-3
3) Reflected radiation only
4) At north end of CPC
5) At south end of CPC
6) Liquid sources only
* Ref: NFS, 1970

TABLE B.8.3-6

SHIELDING SUMMARY
FOR SOURCE AREA - EXTRACTION CELL NO. 3*

Specific Source - Vessels 4C-10, 4C-11, 4C-12, 4D-12, 13C-3, 13C-6, 5D-1, 5D-2, 13D-3, 13D-6

Shielded Area	Elev. (feet)	Design Exposure[1] Rate mR/hr	Shielding			Remarks
			Thick (m)	Mat'l	C.F.[1]	
Cell Access Aisle	100'-0"	1	3'-0"	OC	D	
Cell Access Aisle	110'-0"	1	4"	Steel	A	(2)
Prod. Pur Cell	All	100(M)	1'-6"	OC	C	
X Cell No. 2	All	100(M)	1'-6"	OC	C	
Lower Warm Aisle	100'-0"	1	3'-0"	OC	D	
Lower Ext Aisle	114'-6"	1	3'-0"	OC	D	
Upper Warm Aisle	114'-6"	1	3'-0"	OC	D	
Ext Samp Aisle	131'-0"	1	3'-0"	OC	D	
Solv Storage Tanks	131'-0"	1	3'-0"	OC	D	
Upper Ext Aisle	144'-0"	1	3'-0"	OC	D	
Ext Chem Room	160'-0"	1	3'-0"	OC	D	

1) See Table B.8.3-3

2) Door 15 M-11

* Ref: NFS, 1970

TABLE B.8.3-7

SHIELDING SUMMARY
FOR SOURCE AREA - ANALYTICAL CELLS*

ANALYTICAL HOT CELL

Specific Source - Six 5 ml samples at 8,000 Ci/L

Shielded Area	Elev. (feet)	Design Exposure Rate mR/hr	Shielding			Remarks
			Thick (m)	Mat'l[1]	C.F.[2]	
Lower Extrac. Aisle	114	1	0.6	HDC	A	
Chemical Oper. Aisle	114	1	0.6	HDC	A	
Anal. Oper. Aisle	131	1	0.9	OC	A	3
Anal. Oper. Aisle	131	1	0.6	HDC	A	4
ANC Decon. Aisle	131	1	0.3	Steel	A	5
Sample Storage Cell	134	-	0.9	HDC	-	6
Adjacent ANC	131	-	0.5	HDC	-	6
Control Room	144	1	0.9	OC	A	

ANALYTICAL SAMPLE CELL*

Specific Source - 5 L of HAF 660 Ci/L

Shielded Area	Elev. (feet)	Design Exposure Rate mR/hr	Shielding			Remarks
			Thick (m)	Mat'l[1]	C.F.[2]	
Chem. Operating Aisle	114	1	1.4	OC	A	
Anal. Operating Aisle	131	1	0.9	HDC	A	
ANC, Decon Area	132	1	0.9	HDC	A	

1), 2) See Table B.8.3-3
3) Above El 138 feet
4) Below El 138 feet
5) Door 15M-6
6) Arbitrary thickness by NFS

* Ref: NFS, 1970

TABLE B.8.3-8

SPECIFICATIONS OF MONITORING INSTRUMENTS

Specifications	Beta/Gamma Continuous Air Monitor	Alpha Continuous Air Monitor
Standby and/or Backup Power Supply	Portable - None Stack - Plant standby power	Portable - None Stack - Plant standby power
Range	10^1 to 10^5 cpm	10^0 to 10^4 cpm
Sensitivity	Counting efficiency - 30 percent (2π) (Sr-90/Y-90 Disc Source)	Counting efficiency - 10 percent (2π) (Pu-239 Disc Source)
Accuracy	±10 percent	±10 percent
Calibration Method and Frequency	Electronic/Source - Semiannual	Electronic/Source - Semiannual
Alarm Set Point	Portable - 20 cpm above bkg Stack - 4,500 cpm above bkg	Portable - 3,000 cpm above bkg Stack - 200 cpm above bkg
Recording Device	Chart Recorder	Chart Recorder
Readout	Portable - Analog-log scale on unit Stack - Readout in main control room	Portable - Analog-log scale on unit Stack - Readout in main control room
Alarm	Portable - Red beacon visual with bell audible on unit Stack - Alarm in control room	Portable - Red beacon visual with bell audible on unit Stack - Alarm occurs in control room

TABLE B.8.3-9

PROCESS AND EFFLUENT RADIATION MONITORS

Monitor	Type	Range of Measurements
Supernatant Treatment System		
Permanent Ventilation System	alpha beta/gamma	1-10k cpm 10-100k cpm
IX Column A Discharge	gamma	$0-10^7$ cpm
IX Column B Discharge	gamma	$0-10^7$ cpm
IX Column C Discharge	gamma	$0-10^7$ cpm
IX Column D Discharge	gamma	$0-10^7$ cpm
Decontaminated Supernatant Filter Discharge (STS-RE034)	gamma	$0-10^7$ cpm
Pump 50G-007 (STS-RE037)	gamma	$0-10^7$ cpm
Main Plant/Liquid Waste Treatment System		
Main Plant Ventilation Exhaust Monitors	alpha beta/gamma	1-10k cpm 10-100k cpm
Utility Room Plant Cooling Water Return	gamma	$0-10^6$ cpm
Steam Condensate Return	gamma	$0-10^6$ cpm
LWTS 037/070	gamma	$0-10^6$ cpm
LWTS 082/089	gamma	$0-10^6$ cpm
Cement Solidification System		
01-14 Building Ventilation System	alpha beta/gamma	1-10k cpm 10-100k cpm

TABLE B.8.3-10

CONTINUOUS AIRBORNE RADIOACTIVITY MONITORS*

Location	Alpha	Beta/ Gamma	Pump Location**	Location Monitored	Sampling Line Length (m)	Sample Type
General Operating Aisle	---	X	Separate	General Area	---	---
Upper Extraction Aisle	---	X	Separate	General Area	---	---
Analytical Aisle	X	---	Separate	General Area	---	---
Vitrification Lab - north	X	X	Separate	General Area	---	---
Vitrification Lab - south	X	X	Separate	General Area	---	---
Radiochemical Lab - east - west	X(2)	X(2)	Separate	General Area	---	---
Mass Spectroscopy Lab	X	X	Separate	General Area	---	---
Laundry	X	X	Separate	General Area	---	---
WTF - Con Ed Building	X	---	Separate	General Area	---	---
WTF - Main Shelter	---	X	Separate	General Area	---	---
WRPA	---	X	Plant Vacuum	Compactor	3	Remote
CSRF	---	X	Separate	General Area	2	---
Stack Monitor	X(2)	X(2)	Separate	Stack Effluent	21.3	Remote
Uranium Load Out	---	X	Separate	General Area	5	---
Lower Warm Aisle	X	---	Separate	General Area	---	---
Lower Extraction Aisle	X	---	Separate	General Area	---	---
STS Upper Level	---	X	Separate	General Area	---	---
STS Operating Aisle	---	X	Separate	General Area	---	---
PVS	---	X	Separate	General Area	---	---
WTF 8D-1, M-7 Riser	---	X	Separate	General Area	---	---
CSS	---	X	Separate	General Area	---	---

Location	Alpha	Beta/ Gamma	Pump Location**	Location Monitored	Sampling Line Length (m)	Sample Type
EDR	---	X	Separate	General Area	---	---
CSPF Sorting Room	X	X	Separate	General Area	---	---
LLW2 Resin Loadout	---	X	Separate	General Area	---	---

* Quantities and locations of CAMS will vary based on specific Project activities.
** Separate = vacuum pump at CAM location; Plant Vacuum = connected to plant vacuum system.

TABLE B.8.4-1

ANNUAL OCCUPATIONAL EXPOSURES
DUE TO MAIN PLANT AND IRTS OPERATIONS, AND SUPPORT ACTIVITIES *

ANNUAL OCCUPATIONAL EXPOSURES (1999)

ALARA Group	Average Exposure per worker (rem)
Analytical & Process Chemistry	0.035
Construction Projects	0.004
IRTS Operations	0.018
Maintenance	0.023
Others	0.001
Plant Operations	0.029
Radiation Control Operations	0.069
High Level Waste Projects	0.007
Waste Storage Operations	0.040
Decontamination & Decommissioning	0.052

- Exposures as reported for 1999 based on personnel assignments to ALARA groups reported in December, 1999.

TABLE B.8.6-1

AIRBORNE RADIONUCLIDE EMISSIONS FOR THE YEAR 1999 AS REPORTED IN THE 1999 ANNUAL SITE ENVIRONMENTAL REPORT (WVNS, 2000)

Nuclide	Main Plant (Ci)	Contact Size Reduction Facility (Ci)	Supernatent Treatment System (Ci)	Container Sorting and Packaging Facility (Ci)	01-14 Building (Ci)
H-3	6.77E-3	---	3.45E-4	---	---
Co-60	3.90E-8	0.84E-9	6.11E-9	1.50E-9	0.20E-8
Sr-90	1.77E-5	0.53E-9	2.00E-9	<0.29E-9	1.76E-8
I-129	1.90E-3	<0.82E-9	2.44E-6	7.34E-9	3.38E-7
Cs-137	1.03E-4	0.73E-9	2.71E-8	<0.48E-9	<0.69E-8
Eu-154	2.15E-7	<2.84E-9	0.82E-8	4.74E-9	0.72E-8
U-232	1.00E-8	1.00E-10	0.84E-10	2.37E-10	5.98E-10
U-233/234	6.99E-8	2.68E-10	4.22E-9	1.14E-9	1.06E-8
U-235/236	3.31E-9	<3.17E-11	5.61E-10	9.78E-11	7.32E-10
U-238	6.80E-8	1.10E-10	4.55E-9	1.16E-9	1.12E-8
Pu-238	1.17E-7	0.87E-10	4.92E-10	<5.38E-11	3.05E-10
Pu-239/240	1.44E-7	5.19E-11	2.40E-10	<0.58E-11	<0.20E-10
Am-241	5.03E-7	1.34E-10	3.64E-10	1.01E-10	6.16E-10

TABLE B.8.6-2

SITE ANNUAL (1999) MAIN PLANT LIQUID EFFLUENT DISCHARGES

Nuclide[1]	Ci Released[2]	Unit Dose Factor rem/Ci	Annual Estimated Dose rem	Percentage of the Total Dose
Cs-137[3]	4.18E-03	6.10E-03	2.54E-05	94.42
U-232	3.05E-04	1.68E-4	5.10E-08	0.19
C-14	8.10E-04	4.30E-04	3.48E-07	1.29
Am-241	4.66E-06	7.57E-03	3.50E-08	0.13
Sr-90[3]	3.41E-03	2.80E-04	9.55E-07	3.55
Co-60	2.10E-05	1.24E-3	2.60E-07	0.10
I-129	1.30E-04	2.80E-04	3.60E-08	0.13
		Totals	2.69E-05	99.81

Total Dose to the Maximally Exposed Off-Site Individual from Main Plant effluent discharges is = 2.69E-05 rem

[1] Includes only isotopes yielding a dose of ≥ 0.1 percent of the total dose.
[2] Data from WVDP Annual Site Environmental Report for CY 1999 (WVNS, 1999).
[3] Include contributions from daughter isotopes.
[4] Unit Dose Factors from WVDP-065, Rev. 2.

TABLE B.8.7-1

FISSIONABLE MATERIAL INVENTORY FOR TANK 8D-2 AND MAXIMUM ENVELOPE FOR ION EXCHANGE COLUMNS

Nuclide	Total Fissionable Mass in Tank 8D-2 g	Max. Soluble Fissionable Mass for First Wash g	Maximum Fissionable Mass per STS IX Column g
U-233	720	270	0.4
U-234	660	250	0.3
U-235	41,000	16,000	21
U-238	2,300,000	1,000,000	1,300
Pu-238	370	14	3.9
Pu-239	27,000	924	260
Pu-240	5,700	187	52
Pu-241	640	28	7.8
Am-241	20,000	4.1	0.0
Am-243	25,000	5.0	0.0
Cm-244	200	0.0	0.0

- Assumes Cs breakthrough after processing 375,000 L (100,000 gals.) of undiluted sludge wash solution at an alpha plutonium concentration of 0.25 μCi/mL. The maximum column is analyzed as an ion exchange column containing Ti-treated zeolite and envelopes non-treated zeolite usage.

TABLE B.8.7-2

MATERIAL COMPOSITIONS AND ATOM DENSITIES FOR KENO-V CALCULATIONS

Material	Density g/cc	Volume Fraction	Nuclide	Atomic Density atom/barn-cm	Cross-Section Source
Zeolite (SiO2: 62%, Al2O3: 18.9%) (Na2O: 6.31%, Fe2O3: 4.85%) (TiO2: 4.0%, CaO: 1.46%) (K2O: 1.46%, MgO: 0.97%)	1.75	0.4	Si	0.00440	XSDRN
			Al	0.00156	Hansen Roach
			Na	0.00086	Hansen Roach
			K	0.00013	Hansen Roach
			Fe	0.00040	Hansen Roach
			Ca	0.00011	GAM-2
			Ti	0.00031	GAM-2
			Mg	0.00010	XSDRN
			O	0.01300	Hansen Roach
Water	0.9982	0.6	H2O (X(E))	(KENO-V)	Hansen Roach
304 S.S.	7.9	1.0	SS	(KENO-V)	Hansen Roach
Plutonium	17.7	----	Pu-239	Varies	Hansen Roach

TABLE B.8.7-3

SUMMARY OF CRITICALITY EVALUATION FOR THE ZEOLITE COLUMN OF SMWS

Geometry & Composition	Radius & Length Z (cylinder only) (cm) of Pu-239 Retained Vol.	K-effective $\pm\sigma$
0.44 kg of Pu-239, Pu uniformly distributed in the hemisphere	R=22.5 (hemisphere)	0.38137 ± 0.00161
1.0 kg of Pu-239, Pu uniformly distributed in the cylinder	R=45.5 (cylinder) Z=20	0.66491 ± 0.00305
1.0 kg of Pu-239, Pu uniformly distributed in the cylinder	R=23 (cylinder) Z=20	0.89920 ± 0.00464
1.0 kg of Pu-239, optimum geometry, Pu uniformly distributed in the sphere	R=22.5 (sphere)	0.94748 ± 0.00505

TABLE B.8.7-4

k_{eff} FOR A 1.0 kg Pu-239 SPHERE IN THE CENTER OF THE ZEOLITE COLUMN

Geometry	Radius cm	K-effective ± σ
Sphere	15	0.84061 ± 0.00884
Sphere	18	0.91868 ± 0.00542
Sphere	20	0.94102 ± 0.00525
Sphere	21	0.94186 ± 0.00547
Sphere	22	0.94615 ± 0.00479
Sphere	22.5	0.94748 ± 0.00505
Sphere	23	0.94612 ± 0.00445
Sphere	24	0.93074 ± 0.00470
Sphere	26	0.91059 ± 0.00419
Sphere	30	0.83864 ± 0.00367

TABLE B.8.7-5

k_{eff} FOR A 22.5 cm Pu-239 SPHERE IN THE CENTER OF THE ZEOLITE COLUMN

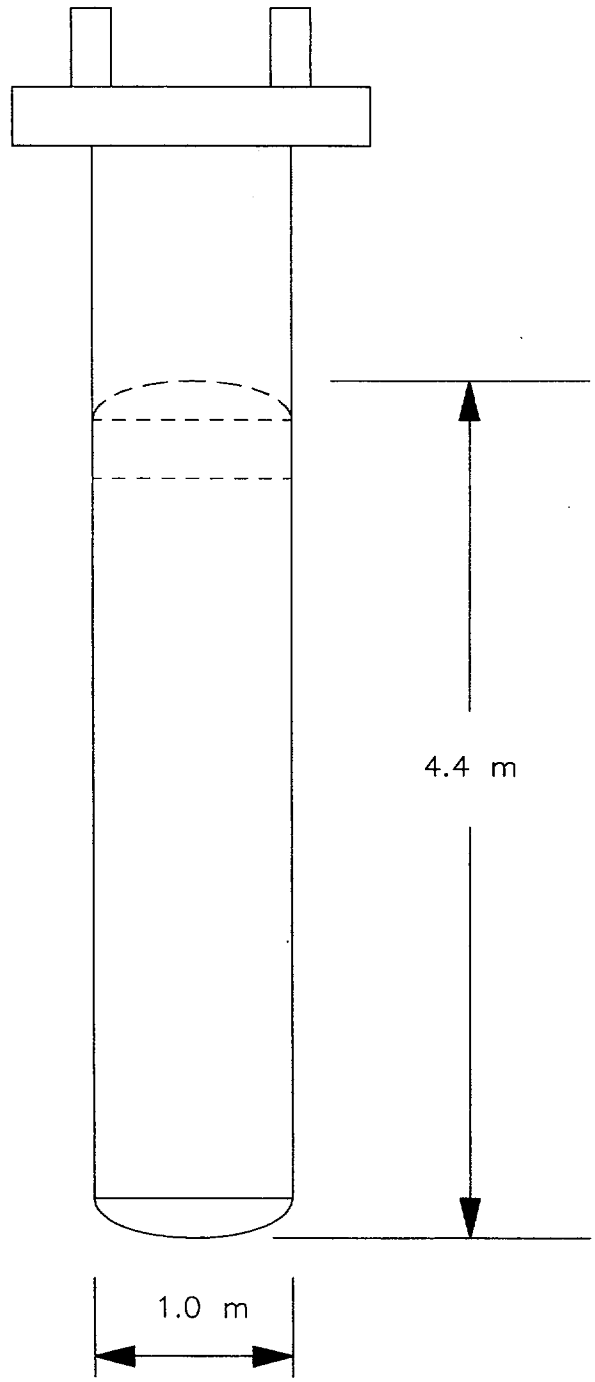
Mass Pu-239 kg	K-effective $\pm\sigma$
0.8	0.88818 ± 0.00436
0.9	0.91380 ± 0.00503
1.0	0.94748 ± 0.00505
1.1	0.95440 ± 0.00517
1.2	0.98249 ± 0.00482

TABLE B.8.7-6

**ESTIMATE OF FISSILE MATERIAL IN MAIN PLANT CELLS
FOR REFERENCE YEAR 1994**

FISSILE ISOTOPE	Estimated Fissile Material Mass		
	PMC (g)	GPC (g)	XC-1 (g)
U-233	4.33E+01	1.47E+02	2.01E+01
U-235	2.05E+03	6.94E+03	9.51E+02
Pu-239	2.81E+02	9.47E+02	1.26E+02
Pu-241	6.32E+00	2.14E+01	2.84E+00

SR2B87-1.DWG



FOR REFERENCE ONLY – NOT TO SCALE

Figure B.8.7-1. STS Ion Exchange Column

B.9.0 HAZARD AND ACCIDENT ANALYSES

B.9.1 Hazard Analysis

The systematic analysis of hazards associated with IRTS operations and facilities and support activities has been accoml`plished in this SAR through the completion of a Process Hazards Analysis (PHA). The PHA is intended to provide a qualitative analysis of the potential sources of hazards and mitigative features associated with facilities and activities discussed in this SAR. Information gained through this analysis is then used in selecting accidents to be further analyzed in a more rigorous quantitative fashion in Section B.9.2 and in grading facility and process descriptions provided throughout the SAR.

B.9.1.1 Methodology

B.9.1.1.1 Hazard Identification

The process of accomplishing the PHA identifies the hazards in terms of quantity, form, location, potential initiating events, and other events which could result in an undesirable consequence. In order to ensure that a comprehensive, systematic analysis was performed, information was obtained from several sources. Primary among these sources were current facility safety documents which identify and evaluate the risks of significant hazards in Project facilities. Site service documents also provided significant information for the PHA. In demonstrating compliance with RCRA spill contingencies the WVDP has developed the Spill Prevention, Controls and Countermeasures Plan (West Valley Nuclear Services Co., Inc., WVDP-043). In support of the site emergency planning program, the *WVDP Hazards Assessment* (West Valley Nuclear Services Co., Inc., WVDP-193) has been developed. These documents supported the identification and evaluation of facility hazards. Additional information for the PHA has been obtained from process flow diagrams, facility operating procedures and miscellaneous site documents referenced therein.

Because many accidents contained in the PHA are of a similar nature (e.g. spills, leaks, fires, etc), bounding accidents may easily be identified through examination of relative inventories. Certain events, however, are more unusual and require quantitative analysis to determine the event probability or consequence. These quantitative analyses are provided in references provided at the end of Table B.9.1-1.

B.9.1.1.2 Hazard Evaluation

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

Qualitative Consequence Classification:

Negligible Negligible on-site and off-site impact on people or the environs.

Low Minor On-site and negligible off-site impact on people or the environs.

Moderate Considerable on-site impact on people or the environs; only minor off-site impact.

High Considerable on-site and off-site impacts on people or the environs.

Qualitative Frequency Classification:

Anticipated ($10^{-1} \geq p > 10^{-2}$) Incidents that may occur several times during the lifetime of the facility.

Unlikely ($10^{-2} \geq p > 10^{-4}$) Accidents that are not anticipated to occur during the lifetime of the facility.

Extremely Unlikely ($10^{-4} \geq p > 10^{-6}$) Accidents that will probably not occur during the life cycle of the facility.

Incredible ($10^{-6} \geq p$) Accidents that are not credible.

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

For each event in Table B.9.1-1, a Risk Factor has been developed that is based on the consequence and frequency for the event. The value of the risk factor is determined from a three-by-three frequency and consequence-ranking matrix, shown in Figure B.9.1-1. Events with negligible consequences were assigned a risk factor of zero (0). Events having either an on-site or an off-site consequence but with probabilities of occurrence less than 1E-6 per year (i.e., incredible events) were assigned a risk factor of "I". Incredible events that have been further evaluated as an accident in Section B.9.2 are assigned a risk factor of "IE" (incredible but evaluated).

B.9.1.2 Hazard Analysis Results

B.9.1.2.1 Hazard Identification

Hazards at the WVDP are of two broad types: radiological and nonradiological (toxicological). These hazards exist as confined and unconfined sources. Table B.9.1-1 presents the PHA on a facility by facility (i.e. location) basis.

In developing potential initiating events, energy sources were identified. The IRTS treatment process and site waste management activities are primarily physical in nature (filtration, concentration, and encapsulation), presenting low inherent operational energy sources as could be present in chemical process facilities. Therefore, severe natural phenomena predominate as the energy source for initiating events considered.

B.9.1.2.2 Hazard Classification

The hazard classification for IRTS, Main Plant and support facilities has been presented in Section B.1.5.

B.9.1.2.3 Hazard Evaluation

B.9.1.2.3.1 Summary of Significant Worker-Safety Features

Though worker hazards protection is provided by engineered facility features, the most significant facility worker-safety feature, namely cell shield walls, are passive in nature. Therefore, the primary operational worker-safety features identified in the hazards analysis are administrative controls. Specifically, worker protection from radiological hazards is controlled through the requirements of the WVDP Radiological Controls Manual (West Valley Nuclear Services Co., Inc., WVDP-010) while worker protection from nonradiological hazards is controlled through the requirements of the WVNS Industrial Hygiene and Safety Manual (West Valley Nuclear Services Co., Inc., WVDP-011).

B.9.1.2.3.2 Accident Selection

The identification of accidents presenting the greatest risk to on-site individuals and the off-site public is one of the primary goals of the PHA. Accidents selected for more rigorous quantitative evaluation are presented in Section B.9.2. These accidents result from IRTS process operations and support activities that were determined to present the greatest risks based on accident consequence and probability. Accidents selected were those identified in the PHA as having a risk factor greater than or equal to 3. In addition, events with incredible probabilities of occurrence, but representing the bounding accident for certain classes of events, were also selected for evaluation (identified with a risk factor of "IE" in Table B.9.1-1).

Accidents selected for further evaluation are:

1) Main ventilation system HEPA bank failure
2) Hydrogen peroxide spill
3) Transformer leak
4) Fire in Lag Storage Facility
5) 8D-2 Tank and vault failure
6) LLWTS Lagoon 2 failure

While other accidents of a similar nature were identified in the hazards analysis, these accidents were selected due to their bounding risk for exposure, both radiological and nonradiological. As a result, operational accidents 1 through 4 above also represent the evaluation basis accidents for the activities analyzed in this SAR. Accidents 5 and 6 represent the beyond evaluation basis accidents analyzed in this SAR.

No major accidents or hazardous situations have occurred throughout the operational history of the site.

B.9.1.3 <u>WVDP Evaluation Guidelines (EGs)</u>

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

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

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

For manmade accidents with either internal or external initiators, radiological EBAs are compared to EGs over the frequency spectrum of 0.1 to 1E-06 events per year. Toxicological EBAs are compared to EGs based on Secretary of Energy Notice (SEN)-35-91, *Nuclear Safety Policy,* over the frequency spectrum 0.1 to 1E-04 per year.

Public Radiological EG: Manmade EBAs shall not cause doses to the maximally exposed off-site individual (MOI) greater than: (1) 0.5 rem (0.005 Sv) for accidents with estimated frequencies <1E-01 per year but ≥1E-02 per year; (2) 5 rem (0.05 Sv) for accidents with estimated frequencies <1E-02 per year but ≥1E-04 per year; and (3) 25 rem (.25 Sv) for accidents with estimated frequencies <1E-04 per year but >1E-06 per year. Manmade EBAs with estimated frequencies ≤1E-06 per year are not considered credible. These EGs are depicted graphically in Figure B.9.1-2.

Public Toxicological EG: For manmade EBAs with an estimated frequency of <0.1 per year but ≥1E-04 per year, the risk of prompt fatality to an average individual in the vicinity of the WVDP from accidents shall not exceed one-tenth of one percent (0.1%) of the sum of prompt fatalities resulting from other accidents to which members of the population are generally exposed. For the purposes of this SAR, this requirement shall be met by comparison of off-site hazardous material concentrations to the ERPG-2 or TEEL-2 value for that material.

On-Site Radiological EG: Manmade EBAs shall not result in calculated doses at the on-site evaluation point (OEP) (640 meters) greater than: (1) 5 rem (0.05 Sv) for accidents with estimated frequencies <0.1 per year but ≥1E-02 per year; (2) 25 rem (0.25 Sv) for accidents with estimated frequencies <1E-02 per year but ≥1E-04 per year; and (3) 100 rem (1.0 Sv) for accidents with estimated frequencies of <1E-04 per year but >1E-06 per year. Manmade EBAs with estimated frequencies ≤1E-06 per year are not considered credible. These EGs are depicted graphically in Figure B.9.1-3.

On-Site Toxicological EG: On-site numerical EGs shall not be required for safety assurance in the analysis of manmade accidents. For the purpose of providing a perspective for accident consequences, on-site hazardous material concentrations shall be compared against the ERPG-3 or TEEL-3 concentration for that material.

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

Public Radiological EG: Natural phenomena induced EBAs shall not cause doses to the MOI greater than 25 rem (0.25 Sv).

Public Toxicological EG: The risk to an average individual in the vicinity of the WVDP for prompt fatalities that might result from natural phenomena induced EBAs shall not exceed one-tenth of one percent (0.1%) of the sum of prompt fatalities resulting from other accidents to which members of the population are generally exposed. For the purposes of this SAR, this requirement shall be met by comparison of off-site hazardous material concentrations to the ERPG-2 or TEEL-2 value for that material.

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

On-Site Toxicological EG: On-site numerical EGs shall not be required for safety assurance in the analysis of accidents induced by natural phenomena.

B.9.2 Accident Analyses

B.9.2.1 Methodology

Accident analyses are performed through the use of established and accepted references and computer codes. Computer codes used in accident analyses are verified per approved procedures prior to use. Accidents analyzed in this SAR represent the bounding accident for a particular event type (i.e., spills, filter failure, liquid release, etc). Events presenting the greatest risk have been identified through process hazards analysis.

Analyses to evaluate the consequences of airborne radiological releases utilize source terms developed from guidance given in DOE-HDBK-3010-94, *Airborne Release Fractions/Rates and Respirable Fractions for Non-Reactor Nuclear Facilities;* site specific dispersion factors calculated using the PAVAN computer codes; and radiological dose conversion factors given in DOE/EH-0070, *External Dose-Rate Conversion Factors for Calculation of Dose to the Public*, and DOE/EH-0071, *Internal Dose Conversion Factors for Calculation of Dose to the Public*.

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

The PAVAN program uses meteorological data in the form of joint frequency distributions of hourly averages of wind direction and wind speed by atmospheric stability class. Wind direction is distributed into 16 sectors (N, NNE, NE,...) and atmospheric stability is distributed into 7 classes (A-G). For each of 16 downwind sectors, the program calculates χ/Q values for each combination of wind speed and atmospheric stability at the site boundary for the respective sector. The χ/Q values calculated for each sector are then ordered from greatest to smallest and an associated cumulative frequency distribution is derived based on the frequency distribution of wind speed and stabilities for that sector. The program then determines for each sector an upper envelope curve based on these data such that no plotted point is above the curve. From this upper envelope the χ/Q value which is equalled or exceeded 0.5% of the total time is obtained. The maximum 0.5% χ/Q value from the 16 sectors becomes the maximum sector χ/Q value, which has been used in consequence analyses in this SAR. See WVDP-065, *Radiological Parameters for Assessment of West Valley Demonstration Project Activities*, for additional information.

Another technique that can be employed to develop conservative χ/Q values is directionally independent (i.e., determined on an "overall site basis"), as opposed to the 0.5% directionally dependent technique discussed in the previous paragraph. Regarding the directionally independent technique, Regulatory Guide 1.145 states the following: "An overall cumulative probability distribution for all directions combined should be constructed. A plot of χ/Q versus probability of being exceeded should be made, and an upper bound curve should be drawn. The 2-hour χ/Q value that is exceeded 5 percent of the time should be selected from this curve as representing the dispersion condition indicative of the type of release being considered." Based on guidance presented in Regulatory Guide 1.145, Section 4, the higher value of the maximum sector χ/Q, (0.5% value) or the 5 percent overall site χ/Q, should be used in evaluations. WVDP-065 used the guidance presented above to determine that the maximum sector 0.5% χ/Q value would be used for the maximally exposed individual dose calculations. A maximum sector 0.5% χ/Q value was calculated for elevated (stack) and ground level releases. This SAR applies these χ/Q values for dose calculations performed at a distance producing maximum exposure for site-specific 95% meteorology. The expression "site-specific 95% meteorology" is often used to communicate the 5% directionally independent technique although maximum sector 0.5% χ/Q values are used for dose calculations. The rationale for this is established in Regulatory Guide 1.1.45, which states that "Selection of the 0.5 percent level is based on an equality, without consideration of plume meander, between the 5 percent directionally independent evaluation of χ/Q and the 0.5 percent directionally dependent evaluation of χ/Q averaged over a reasonably representative number of existing nuclear power plant sites." Given the established equality between the two techniques, and given the fact that the expression "site-specific 95% meteorology" is often used to communicate the 5% directionally independent technique, the terminology "site-specific 95% meteorology" is used in this SAR to communicate conservatively developed site-specific χ/Q values.

Analyses to evaluate the consequences of liquid radiological releases utilize source terms developed based on the guidance given in DOE-HDBK-3010-94 and radiological dose conversion factors given in DOE/EH-0070 and DOE/EH-0071.

The consequences of nonradiological releases are calculated using a standard computer code. The Emergency Prediction Information Code (EPIcode), version 6.0, was used to model the atmospheric dispersion of nonradiological source terms from the postulated accident scenarios described in Section B.9.2.3. EPIcode is endorsed by the DOE as a useful tool for helping emergency planners estimate potential impacts from atmospheric releases of toxic substances. EPIcode used a straight-line Gaussian Plume Model to calculate peak ground-level concentrations downwind of a release. It allows the user to choose the meteorological and environmental conditions of the release, including the Pasquill-Gifford Stability Class (A-F), ground wind speed, type of terrain, effective release height, ambient temperature, and sampling time. EPIcode does not account for terrain effects, plume buoyancy, or wake effects due to nearby structures.

B.9.2.1.1 Initiating Event Summary

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

B.9.2.1.2 Scenario Development

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

B.9.2.1.3 Source Term Analysis

For radiological accident scenarios, source terms are calculated based on the method described in DOE-HDBK-3010-94. This calculation requires quantification of Material-at-Risk (MAR), Damage Ratio (DR), Airborne Release Fraction (ARF) or Accident Release Rate (ARR), Respirable Fraction (RF), and Leakpath Factor (LF) and is given as: Source Term = MAR x DR x ARF x RF x LPF.

In order to bound the consequences of accidents analyzed, source terms used in this SAR are often based on the entire inventory of material at risk; that is, damage ratios and leakpath factors, as described in DOE-HDBK-3010-94, are assumed to be equal to 1. Source terms for nonradiological releases are calculated by EPIcode based on the quantity of material at risk.

B.9.2.1.4 Consequence Analysis

Consequences of radiological accidents in this SAR are calculated for both on-site and off-site individuals. Consequences are calculated for several meteorological conditions: Stability class "D", wind speed 4.5 m/s; Stability class "F", wind speed 1 m/s; and site-specific 95% meteorology. On-site doses are calculated at the On-site Evaluation Point (OEP), located 640 m (2100 ft) from the center of the accident release. Dose to off-site individuals is calculated at the nearest site boundary (1050 m (3400 ft), as shown in WVDP-065), and at the distance producing maximum exposure for site-specific 95% meteorology, namely 1700 m (5577 ft) for an elevated (stack) release, and 2350 m (7700 ft) for a ground level release.

Consequences due to the release of radioactive liquids are calculated through multiplication of an ingested source quantity by dose factors given in DOE/EH-0071. The ingested source quantity is determined from the original accident source term, subsequent source dilution and ingestion rates taken from NRC Regulatory Guide 1.109, *Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Appendix I.*

Consequences of nonradiological airborne releases are calculated for individuals at the OEP 640 m (2100 ft) and site boundary 1050 m (3400 ft). The magnitude of these consequences are calculated through use of the EPIcode which uses a straight-line Gaussian Plume Model to calculate peak ground-level concentrations downwind of a release.

B.9.2.1.5 Comparison to Guidelines

Guidelines utilized for the comparison to accident analysis consequences are given in Section B.9.1.3. Guidelines for both radiological and toxicological consequences due to operating and natural phenomena accidents are provided. Maximum acceptable consequences for radiological accidents are given in Figures B.9.1-2 and B.9.1-3. Maximum acceptable concentrations for nonradiological accidents are defined as ERPG-2 or TEEL-2 for off-site evaluations and ERPG-3 or TEEL-3 for on-site evaluations, regardless of the probability of occurrence. For the purposes of evaluation of Unreviewed Safety Questions, these consequences present the authorization basis risk for activities conducted in facilities within the scope of this SAR.

B.9.2.2 <u>Operational Accidents</u>

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

B.9.2.2.1 Ventilation System Filter Failure

B.9.2.2.1.1 Scenario Development

The Main Plant ventilation system provides contamination control for several areas in the Main Plant building including those areas associated with LWTS and Analytical Laboratory operations. Currently, site ventilation systems do not provide instruments for real-time filter activity measurements. An indication of filter activity is obtained from routine radiation surveys of adjacent operating areas. Highly conservative assumptions have been made regarding the accident source term.

The ventilation system is comprised of a bank of 30 roughing filters in series with a bank of 30 HEPA filters. The deterministic assumption has been made that a pressure excursion in the ventilation system results in the rupture and subsequent release of the entire bank of 30 HEPA filters.

B.9.2.2.1.2 Source Term Analysis

Contamination remaining from original NFS reprocessing activities is the primary source of contamination deposited on Main Plant ventilation filters. Several years of operational experience has shown that ventilation system filter changeout is initiated as a result of high filter differential pressure rather than exposure rate. Nevertheless, the assumption has been made that activity loading on Main Plant HEPA filters results in a surface exposure rate of 10 R/hr. Calculations have shown that an exposure rate of 1 R/hr at the face of a HEPA filter corresponds to a Cs-137 loading of 0.75 Ci (2.8E10 Bq) (Peterson, 1985). Assuming an exposure rate of 10 R/hr per filter, 0.75 Ci (2.8E10 Bq) of Cs-137 per R/hr measured exposure rate and 30 filters results in a filter bank Cs-137 activity loading of 225 Ci (8.3E12 Bq).

The total filter activity is found by scaling the nuclides given in Table B.9.2-1 to 225 Ci (8.3E12 Bq) of Cs-137. Based on this source inventory, the source term given in Table B.9.2-1 was computed from ARF and RF values given in DOE-HDBK-3010-94. These values are 1.0E-2 and 1.0, respectively.

At the Cs-137 levels assumed, calculations have determined that the exposure rate in the adjacent operating aisle would approach 1 R/hr, inconsistent with the ALARA principle given in WVDP-010. In addition, activity concentrations measured from dust samples collected from HEV roughing filters indicate that in order to achieve the assumed exposure rates, approximately 150 kg (330 lb) of dust would need to be deposited per filter (4.5 MT/filter bank). Based on these considerations it is therefore believed that the assumed inventory bounds any conceivable filter failure scenario.

B.9.2.2.1.3 Analysis of Results

Table B.9.2-1 presents the dose at the on-site evaluation point and to off-site individuals from the complete failure of the Main Plant ventilation system filter bank for various meteorological conditions. The maximum total effective dose equivalent at the on-site evaluation point has been calculated to be 6.4 rem, as shown in Table B.9.2-1. The maximum total effective dose equivalent received by an off-site individual has been calculated to be 2.7 rem (0.027 Sv). It should be noted that these doses have been calculated based on the highly conservative assumptions of very high filter activity loading and total filter bank failure. It is not expected that either of these analysis basis parameters would be approached.

B.9.2.2.1.4 Comparison to Guidelines

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

Radiological evaluation guidelines given in Section B.9.1.3 state that total effective dose equivalent to the maximally exposed off-site individual due to an operational accident shall not be greater than 0.5 rem (0.005 Sv) for accidents with estimated frequencies < 0.1 event per year but ≥ 0.01 event per year; 5 rem (0.05 Sv) for accidents with estimated frequencies < 1E-2 event per year but ≥ 1E-4 event per year; and 25 rem (0.25 Sv) for accidents with estimated frequencies < 1E-4 event per year but ≥ 1E-6 event per year.

For the on-site evaluation point, the dose limit is 5 rem (0.05 Sv) TEDE for accidents with estimated frequencies <0.1 event per year but ≥0.01 event per year; 25 rem (0.25 Sv) TEDE for accidents with estimated frequencies <1E-2 event per year but ≥1E-4 event per year; and 100 rem (1.0 Sv) TEDE for those accidents with estimated frequencies <1E-4 event per year but ≥1E-6 event per year.

The doses to the maximally exposed off-site individual (2.7 rem TEDE) and the maximum exposure at the on-site evaluation point (6.4 rem TEDE) due to the failure of the bank of ventilation system filters are below the radiological dose acceptance criteria specified in Section B.9.1.3.

B.9.2.2.2 Hydrogen Peroxide Spill

B.9.2.2.2.1 Scenario Development

Technical grade 35% hydrogen peroxide is stored in 1,249 L (330 gal) polyethylene totes in the Oxidizer Room of the New Warehouse. Due to installed safety features in the Oxidizer Room (collective berm, automatic sprinkler system, controlled ventilation) and administrative controls which limit access to the room, a spill and/or a fire within the oxidizer room would be of limited severity, with no consequence to the off-site public.

A maximum of one tote can be placed on a forklift for transport at any time. With only the tote as a barrier to release to the environment, a significant spill outside of the Oxidizer Room presents a risk requiring further evaluation.

B.9.2.2.2.2 Source Term Analysis

The release rate of the hydrogen peroxide from the tote is not significant since evaporation is the dominant release mechanism. It was assumed that one tote spills its entire contents outside the warehouse 1240 L (330 gal). For conservatism, it was assumed that the temperature of the hydrogen peroxide was 30°C (86°F) and that the total quantity spilled from the tote formed a uniform 1 mm (0.039 in) deep pool.

B.9.2.2.2.3 Analysis of Results

For Pasquill-Gifford stability class "F" and 1 m/s wind speed conditions, EPIcode calculated ground-level concentrations of 17.5 ppm at the 640 m (2100 ft) on-site evaluation point (OEP) and 8.75 ppm at the 1050 m (3400 ft) site boundary. Conditions of stability class "D" and 4.5 m/s wind speed resulted in ground-level concentrations of 3.08 ppm at the OEP and 1.50 ppm at the 1050 m (3400 ft) site boundary. The 17.5 ppm OEP concentration is less than the 100 ppm ERPG-3 level, and the 8.75 ppm concentration at the 1050 m (3400 ft) site boundary is well below the ERPG-2 level of 50 ppm.

B.9.2.2.2.4 Comparison to Guidelines

Toxicological evaluation guidelines specified in Section B.9.1.3 state that dosages to the maximally exposed off-site individual due to an operational accident shall not exceed the ERPG-2 or TEEL-2 dosage. Though on-site numerical EGs are not required for the analysis of manmade accidents, on-site hazardous material concentration are compared against the ERPG-3 or TEEL-3 concentrations for that material.

The dosage to the maximally exposed off-site individual (8.75 ppm) and the maximum dosage at the on-site evaluation point (17.5 ppm) due to a hydrogen peroxide spill are below the ERPG comparison values of 50 ppm and 100 ppm, respectively.

B.9.2.2.3 Main Plant Transformer Rupture

B.9.2.2.3.1 Scenario Development

The Main Plant transformer contains approximately 2200 L (586 gal) of PCB-contaminated Wemco "C" oil, a hydrotreated light napthenic that is a confirmed carcinogen, which serves as a heat-transfer media. The concentration of PCBs in the oil has been measured at 292 ppm, or about 0.64 L (0.17 gal). This contamination came from residual amounts of PCBs in the equipment used to fill the transformer.

The transformer containment could be breached by external impact or structural failure due to a seismic event. Once a puncture/failure has occurred, the coolant (PCB-laden oil) above the failure location would empty to the gravel below at a rate dependent upon the size of the failure. Evaporation of the pool formed by the spill and airborne transport of the PCBs would follow.

B.9.2.2.3.2 Source Term Analysis

The release rate of contaminated oil from the transformer is not important since evaporation of the PCBs is the dominant release mechanism. The rate of evaporation from the oil depends on the ambient temperature, wind speed, and surface area formed as a result of the spill. To achieve the maximum evaporation rate, it was assumed that the temperature of the oil was 30°C (86°F) and that the total quantity of oil in the transformer formed a uniform 1 mm (0.039 in) deep pool.

B.9.2.2.3.3 Analysis of Results

For Pasquill-Gifford stability class "F" and 1 m/s wind speed conditions, EPIcode calculated ground-level concentrations of 6.4E-03 mg/m^3 and 3.1E-03 mg/m^3 on-site and off-site, respectively. Conditions of stability class "D" and 4.5 m/s wind speed resulted in slightly lower concentrations downwind, 1.1E-03 mg/m^3 and 5.5E-04 mg/m^3, respectively. These concentrations are well below the 5 mg/m^3 ERPG-2 level for PCBs. (The TEEL-3 level is also 5 mg/m^2.)

B.9.2.2.3.4 Comparison to Guidelines

Guidelines for toxicological dosages are those specified in Section B.9.2.2.2.4. The dosage to the maximally exposed off-site individual (3.1E-3 mg/m^3) and the maximum exposure at the on-site evaluation point (6.4E-3 mg/m^3) due to a transformer leak are below the toxicological dosage acceptance criteria (5 mg/m^3 and 5 mg/m^3, respectively) specified in Section B.9.1.3.

B.9.2.2.4 Fire In Lag Storage Facility

B.9.2.2.4.1 Scenario Development

The Lag Storage Building and Lag Storage Annexes 1, 3, and 4 contain approximately 7000 m^3 (2.5E5 ft^3) of mixed, transuranic, and low-level waste stored in carbon steel drums and boxes. Of this inventory, 1650 m^3 (5.8E4 ft^3) is estimated to be combustible material such as anti-contamination clothing and flammable liquids. Lag Storage Annex #4 contains the most combustible waste by volume at approximately 620 m^3 (2.2E4 ft^3)

A fire could occur in the Lag Storage Facility, causing some of the combustible material to ignite and burn. This analysis assumes that a fire occurs in Lag Storage Annex #4 and that the combustion of waste results in airborne transport of radionuclides from the facility.

B.9.2.2.4.2 Source Term Analysis

The entire volume of combustible waste in Lag Storage Annex #4 is assumed to be consumed by the fire. Release of radionuclides to the environment occurs via airborne transport assuming ARF and RF values given in DOE-HDBK-3010-94 for thermal stress of packaged mixed waste. Bounding values of 5E-4 for ARF and 1.0 for RF are used. All major radionuclides (those contributing >0.1% of the total TEDE) are considered in the dose calculations. Typical radionuclide inventories for Lag Storage waste containers are given in Table B.7.7-2.

B.9.2.2.4.3 Analysis of Results

Radiological consequences were analyzed for airborne pathways only. Table B.9.2-2 presents the dose at the on-site evaluation point and to off-site individuals from a fire in Lag Storage Annex #4 for various meteorological conditions. The maximum Total Effective Dose Equivalent (TEDE) at the on-site evaluation point is 6.2 rem (0.062 Sv), while the maximum TEDE received by an off-site individual is 2.9 rem (0.029 Sv).

Although all combustible material in Lag Storage Annex #4 is assumed to be consumed by the fire, it is unlikely that a fire of sufficient intensity to breach and completely burn all combustible waste containers would develop before mitigative actions were taken. The envisioned fire for this accident scenario is comparable to the fire that one might postulate for the "maximum possible fire loss" (MPFL). The MPFL would likely include the complete loss of the building structure, including all exterior walls, the roof, and possibly the steel truss structure. The most likely source of a MPFL scenario would be a natural gas leak in the Sterling/Alton gas-fired appliance. The escaping gas could find an ignition source resulting in a fire, and a possible explosion, engulfing Lag Storage Annex #4. Since the likelihood of such a catastrophic scenario is very small, actual doses received by on-site and off-site individuals from a fire-related incident are likely to be considerably lower than doses postulated by this accident scenario.

B.9.2.2.4.4 Comparison to Guidelines

Guidelines for operational accidents are those discussed in Section B.9.2.2.1.4. The doses to the maximally exposed off-site individual (2.9 rem TEDE) and the maximum exposure at the on-site evaluation point (6.2 rem TEDE) due to a fire in Lag Storage Annex #4 are below the radiological dose acceptance criteria specified in Section B.9.1.3.

B.9.2.3 <u>Natural Phenomena Events</u>

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

B.9.2.3.1 Earthquake Induced Failure of Tank 8D-2 Roof and Vault

B.9.2.3.1.1 Scenario Development

The HLW remaining from the PUREX process, the THOREX campaign, and Phase I pretreatment operations at the WVDP is contained in underground storage tank 8D-2. As shown in Figure B.5.1-1 the HLW tanks are located in the Waste Tank Farm (WTF) north of the Vitrification Facility. The primary barriers (storage tanks, risers, vaults, and pump pits) have been designed to withstand design basis earthquake described in Section B.5.2. It is assumed that the occurrence of a severe earthquake greater than six times the design basis (0.1g) causes the roof of Tank 8D-2 and its vault to collapse, exposing the tank contents to the atmosphere. The evaporated Vit feed solution is released without passing through any filtration. Tank 8D-2 contains the entire feed inventory for the WVDP high-level waste processing system (Vitrification). Failure of this vessel was therefore selected for analysis due to the large radionuclide inventory in the tank relative to other vessels.

B.9.2.3.1.2 Source Term Analysis

For this analysis it is assumed that the occurrence of a severe earthquake causes the roof of the vault and tank to collapse, allowing the evaporation of liquid from the tank. The evaporated solution is assumed to be released directly to the environment. The source term quantity is based on bounding airborne release rates given in DOE-HDBK-3010-94 for evaporative loss from pools. Bounding release rates of 4E-8 per hour for non-volatiles and 1.0 for volatiles were applied to the source inventory given in Table B.4.1-1. The respirable fraction for all releases is assumed to be 1.0.

B.9.2.3.1.3 Analysis of Results

Radiological consequences were analyzed for airborne pathways only. All major radionuclides (those contributing >0.1% of the total TEDE) were used in the dose calculations. Table B.9.2-3 presents the dose at the on-site evaluation point and to off-site individuals from the catastrophic failure of the 8D-2 tank and vault for several meteorological conditions. The maximum total effective dose equivalent (TEDE) at the on-site evaluation point is 21 rem (0.21 Sv). The maximum TEDE to the off-site individual has been calculated to be 9.9 rem (0.099 Sv).

B.9.2.3.1.4 Comparison to the Guidelines

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

The dose to the maximally exposed off-site individual due to Tank 8D-2 vault failure (9.9 rem TEDE) is well below the radiological dose acceptance criteria specified in Section B.9.1.3 for natural phenomena events.

B.9.2.3.2 Earthquake Induced Failure of LLWTS Storage Lagoon 2

B.9.2.3.2.1 Scenario Development

Lagoon 2 provides temporary storage for low-level liquid wastes generated at the WVDP prior to decontamination and eventual release to the environment. As indicated in Figure B.7.7-1, Lagoon 2 is adjacent to the embankment leading to the Erdman Brook drainage basin. This embankment has not been designed to withstand seismic accelerations and the deterministic assumption is made that the lagoon basin wall fails as a result of a seismic event. Furthermore it has been assumed that Lagoon 3 has just been emptied, thereby reducing the strength and ability of the wall between Lagoons 2 and 3 to withstand a seismic event. This assumption provides additional conservatism as the contents of Lagoon 3 would act to dilute the source strength of Lagoon 2 due to decontamination by the LLWTS.

B.9.2.3.2.2 Source Term Analysis

The source term for this analysis is based on a seven year average of radionuclide concentrations measured in Lagoon 3 discharge corrected for decontamination at the LLWTS. (Derivation of Lagoon 2 concentrations from Lagoon 3 concentrations is necessitated by the fact that comprehensive analyses of radionuclides in Lagoon 2 is not routinely performed.) The calculated Lagoon 2 composition is then scaled to a maximum gross beta concentration of 5E-3 uCi/mL to achieve the analysis basis concentration which is given in Table B.7.5-1. For this analysis Lagoon 2 is assumed to be full to its operating capacity (9,500,000 L) with water having this radionuclide composition.

B.9.2.3.2.3 Analysis of Results

Table B.9.2-4 presents the dose to the maximally exposed off-site individual from ingestion of 2 L (0.53 gal) of contaminated water taken from Cattaraugus Creek. Cattaraugus Creek has an average flow rate of 3,620,000 L/h (West Valley Nuclear Services Co., Inc., WVDP-065). It has been assumed that the entire volume of Lagoon 2 reaches Cattaraugus Creek in 1 hour. The committed effective dose equivalent to the maximally exposed off-site individual is calculated to be approximately 0.41 rem (0.0041 Sv).

B.9.2.3.2.4 Comparison to Guidelines

Guidelines for natural phenomena are those discussed in Section B.9.2.3.1.4. The dose to the maximally exposed off-site individual due to a lagoon embankment failure (0.41 rem CEDE) is well below the radiological dose acceptance criteria specified in Section B.9.1.3.

B.9.2.4 Accident Analysis Summary

A summary of the consequences of accidents analyzed in this SAR is provided in Table B.9.2-5. All accidents analyzed are within the evaluation guidelines given in Section B.9.1.3. The failure and release of the entire liquid fraction of Tank 8D-2 results in a total effective dose equivalent to the maximally exposed off-site individual of 9.9 rem (0.099 Sv). This represents the bounding accident for radiological releases. The spill of hydrogen peroxide results in an exposure to an off-site individual of 8.75 ppm and an exposure to an on-site individual of 17.5 ppm. These exposures are below the toxicological dosage EGs. This represents the bounding accident for nonradiological releases.

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TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
SUPERNATANT TREATMENT SYSTEM AND WASTE TANK FARM					
High Level Waste Transfer System					
Zeolite transfer from tank 8D-1 to 8D-2: Maximum pump rate - 110 gpm.					
HLW Zeolite (See Table B.4.1-1)	1) Leak at jumper connection 2) Transfer line rupture 3) Transfer line rupture and trench failure due to seismic event	- Double-walled stainless steel piping - Concrete trench with sealed covers - Stainless steel lined concrete pump pits with sealed covers - Leak detectors in transfer lines and pump pits - Shielding - 2 feet of concrete - Clayey till	1) Negligible 2) Negligible 3) Negligible	1) Unlikely 2) Extremely unlikely 3) Incredible [7]	1) 0 2) 0 3) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
Supernatant Treatment System					
Airborne Contamination	1) Loss of line power 2) Backflow due to direct tornado strike 3) Loss of confinement barrier integrity due to tornado	- Redundant vent system blowers - Seals on valve aisle penetrations - Multiple barriers to direct cell access - Redundant vent system filter trains - Monolithic cell structure	1) Negligible 2) Negligible [1] 3) Negligible [1]	1) Anticipated 2) Extremely Unlikely 3) Extremely Unlikely	1) 0 2) 0 3) 0
Process Lines Between Vessels					
High level waste (See Table B.4.1-1)	1) Process line leak 2) Process line failure 3) Process line and confinement loss due to seismic event	- Clayey till	1) Negligible 2) Negligible 3) Low	1) Unlikely 2) Extremely unlikely 3) Incredible	1) 0 2) 0 3) IE

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES
(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
No ID # **Volume: 19,000 L** Name: Caustic Addition System Location: Yard north of HLW tanks Construction: N/A					
NaOH	1) Tanker truck leak 2) Tanker truck failure	- Portable berm - Inflatable berm	1) Negligible 2) Negligible	1) Unlikely 2) Extremely unlikely	1) 0 2) 0
8D-1 **Volume: 2,800,000 L** Name: PUREX HLW Tank Location: Waste Tank Farm Construction: Carbon steel					
Cs loaded zeolite (See Table B.4.1-1)	1) Loss of ventilation 2) Tank leak due to corrosion 3) Tank failure due to seismic event 4) Tank and vault failure due to seismic event	- 1/2 inch carbon steel tank - Liquid level indicators - Pressure indicators - Carbon steel pan - Reinforced concrete vault - Clayey till	1) Negligible 2) Negligible 3) Negligible 4) Low	1) Unlikely 2) Anticipated 3) Extremely unlikely 4) Incredible	1) 0 2) 0 3) 0 4) I
8D-2 **Volume: 2,800,000 L** Name: PUREX HLW Tank Location: Waste Tank Farm Construction: Carbon steel					
High level waste (See Table B.4.1-1)	1) Loss of ventilation 2) Tank leak due to corrosion [1] 3) Tank failure due to seismic event 4) Tank and vault failure due to seismic event 5) Tank over-pressurization due to H_2 explosion	- 1/2 inch carbon steel tank - Liquid level indicators - Pressure indicators - Carbon steel pan - Reinforced concrete vault - Clayey till - Tank ventilation and natural convection to remove H_2 buildup [2]	1) Negligible 2) Negligible 3) Low 4) Moderate 5) Moderate	1) Unlikely 2) Anticipated 3) Extremely unlikely 4) Incredible 5) Incredible [2]	1) 0 2) 0 3) 1 4) IE 5) I

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES
(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
8D-3 **Volume: 57,000 L** Name: STS Product Tank Location: Waste Tank Farm Construction: Stainless steel					
26) Processed HLW 27) VF CFMT overheads [See 1) Table B.4.1-1, and 2) Crocker, 1989]	1) Loss of ventilation 2) Tank leak 3) Tank failure due to seismic event 4) Tank and vault failure due to seismic event	- Stainless steel tank - Liquid level indicators - Pressure indicators - Reinforced concrete vault lined with stainless steel - Clayey till	1) Negligible 2) Negligible 3) Negligible 4) Moderate	1) Unlikely 2) Extremely unlikely 3) Incredible 4) Incredible	1) 0 2) 0 3) 0 4) I
8D-4 **Volume: 57,000 L** Name: Waste Header Effluent Tank Location: Waste Tank Farm Construction: Stainless steel					
VF CFMT overheads (See Crocker, 1989)	1) Loss of ventilation 2) Tank leak 3) Tank failure due to seismic event 4) Tank and vault failure due to seismic event	- Stainless steel tank - Liquid level indicators - Pressure indicators - Reinforced concrete vault lined with stainless steel - Clayey till	1) Negligible 2) Negligible 3) Low 4) Moderate	1) Unlikely 2) Extremely unlikely 3) Incredible 4) Incredible	1) 0 2) 0 3) I 4) I

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES
(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
50C-001 Volume: 7,200 L Name: IX Column Location: Tank 8D-1 Construction: Stainless steel **50C-002 Volume: 7,200 L** Name: IX Column Location: Tank 8D-1 Construction: Stainless steel **50C-003 Volume: 7,200 L** Name: IX Column Location: Tank 8D-1 Construction: Stainless steel **50C-004 Volume: 7,200 L** Name: IX Column Location: Tank 8D-1 Construction: Stainless steel					
High level waste (See Table B.4.1-1)	1) Column leak due to corrosion 2) Column failure due to over-pressurization 3) Column failure due to seismic event	- Stainless steel IX column - Pressure indicators - Low, high level alarms - 8D-1 carbon steel tank - Liquid level indicators - Pressure indicators - Carbon steel pan - Reinforced concrete vault - Clayey till	1) Negligible 2) Negligible 3) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible	1) 0 2) 0 3) 0
50D-001 Volume: 6,535 L Name: Supernatant Feed Tank Location: Tank 8D-1 Construction: Stainless steel					
High level waste (See Table B.4.1-1)	1) Tank leak 2) Tank failure due to seismic event 3) Tank failure due to over-pressurization 4) Overflow	- Stainless steel tank - 8D-1 carbon steel tank - Liquid level indicators - Pressure indicators - Carbon steel pan - Reinforced concrete vault - Clayey till	1) Negligible 2) Negligible 3) Negligible 4) Negligible	1) Extremely unlikely 2) Incredible 3) Extremely unlikely 4) Unlikely	1) 0 2) 0 3) 0 4) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
50D-002 Name: Zeolite Batch Tank Location: STS Building					
Fresh Zeolite	1) Tank leak 2) Tank failure due to seismic event	- Confinement provided by STS building	1) Negligible 2) Negligible	1) Extremely unlikely 2) Extremely unlikely	1) 0 2) 0
50D-003 Name: Fresh Water Tank Location: STS Operating Aisle					
Fresh water	1) Tank leak 2) Tank failure due to seismic event		1) Negligible 2) Negligible	1) Extremely unlikely 2) Extremely unlikely	1) 0 2) 0
50D-004 Volume: 8,110 L Name: Sluice Lift Tank Location: Tank 8D-1 Construction: Stainless steel					
Sluice water (See Table B.4.1-1)	1) Tank leak 2) Tank failure due to seismic event	- Stainless steel tank - High and low level alarms - 8D-1 carbon steel tank - Carbon steel pan - Reinforced concrete vault - Clayey till	1) Negligible 2) Negligible	1) Unlikely 2) Extremely Unlikely	1) 0 2) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES
(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
50D-005 Name: Water Break Tank Location: STS Valve Aisle					
STS process upset safeguard tank	1) Tank leak 2) Tank failure due to seismic event 3) Overflow	- Stainless steel tank - STS valve aisle sump - STS building - High and low level alarms	1) Negligible 2) Negligible 3) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Unlikely	1) 0 2) 0 3) 0
50D-006 Name: Air Break Tank Location: STS Valve Aisle					
STS process upset safeguard tank	1) Tank leak 2) Tank failure due to seismic event	- Stainless steel tank - STS valve aisle sump - STS building - High and low level alarms	1) Negligible 2) Negligible	1) Extremely unlikely 2) Extremely unlikely	1) 0 2) 0
50D-007 Name: Fines Collection Tank Location: STS Building					
Fresh Zeolite	1) Tank leak 2) Tank failure due to seismic event	- Confinement provided by STS building	1) Negligible 2) Negligible	1) Extremely unlikely 2) Extremely unlikely	1) 0 2) 0
50D-008 Volume: 57 L Name: Brine Chiller Expansion Tank Location: STS Building Construction: Stainless steel					
Brine Sodium nitrate	1) Tank leak 2) Tank failure due to seismic event	- Concrete berm - STS operating aisle sump - Drains to tank 8D-1	1) Negligible 2) Negligible	1) Extremely unlikely 2) Extremely unlikely	1) 0 2) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
50E-001 Name: Supernatant Cooler Location: Tank 8D-1 Construction: Stainless steel					
High level waste (See Table B.4.1-1)	1) Cooler leak 2) Cooler failure due to seismic event	- Stainless steel cooler - 8D-1 carbon steel tank - Liquid level indicators - Pressure indicators - Carbon steel pan - Reinforced concrete vault - Clayey till	1) Negligible 2) Negligible	1) Extremely unlikely 2) Incredible	1) 0 2) 0
50E-002 Volume: 220 L Name: Brine/Heat Exchanger Location: STS Operating Aisle Construction: Stainless steel					
Brine solution of sodium nitrate	1) Tank leak 2) Tank failure due to seismic event	- Stainless steel tank - Concrete berm - Operating aisle sump - Sight gauge - Drains to tank 8D-1	1) Negligible 2) Negligible	1) Extremely unlikely 2) Extremely unlikely	1) 0 2) 0
50F-001 Volume: N/A Name: Prefilter Location: Tank 8D-1 Construction: Stainless steel					
High level waste (See Table B.4.1-1)	1) Filter leak 2) Filter failure due to seismic event 3) Filter failure due to over-pressurization	- 8D-1 carbon steel tank - Stainless steel filter housing - Temperature indicator - Pressure indicator - Reinforced concrete vault - Clayey till - Low, high level alarms	1) Negligible 2) Negligible 3) Negligible	1) Extremely unlikely 2) Incredible 3) Extremely unlikely	1) 0 2) 0 3) 0
50F-002 Volume: 530 L Name: Sand Postfilter Location: Tank 8D-1 Construction: Stainless steel					
Processed high level waste solution (See Table B.4.1-1)	1) Filter leak 2) Filter failure due to seismic event 3) Filter failure due to over-pressurization	- 8D-1 carbon steel tank - Stainless steel filter housing - Pressure indicator - Reinforced concrete vault - Clayey till - High level alarms	1) Negligible 2) Negligible 3) Negligible	1) Extremely unlikely 2) Incredible 3) Extremely unlikely	1) 0 2) 0 3) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
50V-001 Volume:	Name: Chiller	Location:		Construction: Stainless steel	
High level waste (See Table B.4.1-1)	1) Chiller leak 2) Chiller failure due to seismic event	- Stainless steel chiller - 8D-1 carbon steel tank - Liquid level indicators - Pressure indicators - Carbon steel pan - Reinforced concrete vault - Clayey till	1) Negligible 2) Negligible	1) Extremely unlikely 2) Incredible	1) 0 2) 0
Waste Tank Farm Ventilation System					
Waste Tank Farm Ventilation System					
1) Airborne contamination 2) Contaminated liquid	1) Loss of contaminated airborne confinement due to filter failure 2) Loss of contaminated airborne confinement due to vent duct failure 3) Loss of contaminated liquid confinement due to condensate piping failure/leak 4) Loss of contaminated liquid confinement due to vessel failure	- WTF HEPA filter differential pressure indication/alarm - Redundant HEPA filter trains and blowers - Ventilation system backup via PVS - Containment of WTF ventilation system vessels by WTF shelter; drains to 8D-1 or 8D-2	1) Negligible 2) Negligible 3) Negligible 4) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Extremely unlikely 4) Extremely unlikely	1) 0 2) 0 3) 0 4) 0
8D-6 Volume: 1,900 L	Name: WTF Off-Gas Knock-Out Pot	Location: WTF Shelter		Construction: Carbon steel	
Liquid LLW (See Table B.4.1-1)	1) Tank leak due to corrosion 2) Tank failure due to seismic event 3) Overflow	- Containment by WTF shelter - Overflow to 8D-1 or 8D-2	1) Negligible 2) Negligible 3) Negligible	1) Extremely unlikely 2) Incredible 3) Unlikely	1) 0 2) 0 3) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
8D-7 **Volume: 950 L**	Name: WTF Off-Gas Relief Tank	Location: WTF shelter		Construction: Carbon steel	
Liquid LLW (See Table B.4.1-1)	1) Tank leak due to corrosion 2) Tank failure due to seismic event 3) Overflow	- Containment by WTF shelter - Overflow to 8D-1 or 8D-2	1) Negligible 2) Negligible 3) Negligible	1) Extremely unlikely 2) Incredible 3) Unlikely	1) 0 2) 0 3) 0
8C-1 **Volume: 2,650 L**	Name: Off-Gas Caustic Scrubber	Location: WTF Shelter		Construction: Carbon steel	
Liquid LLW (See Table B.4.1-1)	1) Scrubber leak due to corrosion 2) Scrubber failure due to seismic event 3) Overflow	- Containment by WTF shelter - Overflow to 8D-6 - High/low level alarm	1) Negligible 2) Negligible 3) Negligible	1) Extremely unlikely 2) Incredible 3) Unlikely	1) 0 2) 0 3) 0
MAIN PLANT AND LIQUID WASTE TREATMENT SYSTEM					
Main Plant					
Airborne Contamination	1) Loss of line power 2) Backflow due to direct tornado strike 3) In-cell explosion or fire 4) HEPA filter failure 5) HEPA filter fire	- Redundant vent system blowers - Backflow filters on highly contaminated cells - Seals on penetrations - Multiple barriers to direct cell access (airlock, crane room, shield door) - No explosives contained in cells - Minimal or no combustibles in cells - Redundant vent system filter trains - Cell pressure indicators	1) Negligible 2) Negligible 3) Low 4) Moderate 5) Low	1) Anticipated 2) Extremely unlikely 3) Extremely unlikely 4) Extremely unlikely 5) Extremely unlikely	1) 0 2) 0 3) 1 4) 3 5) 1

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
Criticality	1) Accumulation of fissile material into critical configuration 2) Moderation of existing accumulation 3) Plateout of fissile material in vent ducting or LWTS evaporator 4) Concentration of fissile material in LWTS evaporator acid wash solution	- Strict administrative controls on storage and handling of fissile material - Substantial shielding by cells containing significant quantities of fissile material (as contamination) - Sumps and sump alarms to indicate inleakage of water - Routine plant radiation surveys to determine contamination accumulations - Critically safe design of the evaporator - Evaporator, waste vessels in heavily shielded cells	1) Low 2) Low [3] 3) Moderate [4 & 5] 4) Low [4 & 5]	1) Extremely unlikely 2) Extremely unlikely [3] 3) Incredible 4) Incredible	1) 1 2) 1 3) I 4) I
Contaminated Liquid	1) Overflow of cell sump 2) Transfer pipe failure 3) LWTS pump leak	- Cell sump alarm - Berms in cells containing active vessels - LWTS pumps contained in sealed niches with drains back to extraction cell	1) Negligible 2) Negligible 3) Negligible	1) Unlikely 2) Extremely unlikely 3) Anticipated	1) 0 2) 0 3) 0
Natural Gas	1) Explosion due to accumulation	- Only minor amount of gas piping in confined areas - No piping in areas containing hazardous material	1) Negligible [1]	1) Extremely unlikely	1) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
		Head End Cells			
Fuel/Hull Material	1) Criticality due to accumulation of water (moderator) in the cell. 2) Metal fire involving Zr cladding or UO_2 3) Fire/explosion due to ignition of high concentrations of airborne, finely-divided particulate material 4) Fire in HEV filter bank due to pyrophoric reaction in HEC 5) Unrestricted release to environment due to HEV filter failure following fire	- Accumulation of fissile material in GPC critically safe in current configuration, even with full water moderation - Floor drain and floor hatches in PMC prevent accumulation of water in PMC - Fuel and clad material primarily in bulk form which inhibits pyrophoric reactions - Sufficient shielding in the GPC to protect worker from inadvertent criticality - No significant off-site doses due to atmospheric dispersion of fission gases - Fuel and clad material in oxidized state which inhibits pyrophoric reactions - Contaminated area of cell lined with stainless steel which prevents accumulation of static electricity - Minimal amount of smoke/embers associated with pyrophoric reactions that could become entrained in cell ventilation airstreams - Ventilation for HECs possible through Main Plant Ventilation System (Embers or burning material from HEC less likely to reach Main Plant Ventilation filters)	1) Low 2) Low 3) Low 4) Low 5) Low	1) Incredible 2) Unlikely 3) Unlikely 4) Incredible 5) Extremely Unlikely	1) I 2) 2 3) 2 4) I 5) 1

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
XC-1 and XC-2					
Residual Radioactive Material	1) Criticality due to accumulation of water (moderator) in XC-1 or XC-2 2) Fire or explosion in XC-1 or XC-2 3) Uncontrolled release to environment due to Main Plant ventilation exhaust system filter failure following fire	- Though estimated fissile material quantities in each cell slightly exceed mass single parameter limits for a uniform aqueous solution, the distribution and immobility of the fissile material make the cells critically safe, even with full water moderation - Sufficient shielding to protect worker from inadvertent criticality - Fissile material in oxidized state which inhibits pyrophoric reactions - Cells lined with stainless steel which prevents accumulation of static electricity - Very limited amount of smoke/embers that a fire would generate	1) Low 2) Low 3) Low	1) Incredible 2) Extremely Unlikely 3) Extremely Unlikely	1) I 2) 1 3) 1
Main Plant Tanks					
7D-2 **Volume: 32,220 L**	Name: LLW Collection Tank		Location: LWC	Construction: Stainless Steel	
Liquid LLW (See Table B.4.1-1)	1) Tank leak 2) Tank failure due to seismic event 3) Tank and ventilation failure due to seismic event 4) Tank, ventilation, and LWC failure due to seismic event 5) Overflow	- Stainless steel tank - Level recorder - High and low level alarms - Spills handled by LWC sump - Overflow contained by Tank 6D-3 - LWC, reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES
(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
7D-8 Volume: 11,360 L	Name: Rework Evaporator Feed Tank	Location: LWC		Construction: Stainless Steel	
Liquid LLW (Activity indeterminate)	1) Tank leak 2) Tank failure due to seismic event 3) Tank and ventilation failure due to seismic event 4) Tank, ventilation, and LWC failure due to seismic event 5) Overflow	- Stainless steel tank - Level recorder - Spills handled by LWC sump - Overflow contained by Tank 6D-3 - LWC, reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0
7D-14 Volume: 1,900 L	Name: Hot Analytical Cell Drain Catch Tank	Location: LWC		Construction: B334 Hastelloy C	
Liquid LLW (Activity indeterminate)	1) Tank leak 2) Tank failure due to seismic event 3) Tank and ventilation failure due to seismic event 4) Tank, ventilation, and LWC failure due to seismic event 5) Overflow	- Corrosion resistant tank - Level recorder - Level alarm - Spills handled by LWC sump - Overflow contained by Tank 6D-3 - LWC, reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0
13D-8 Volume: 2,570 L	Name: Cell Sump Receiver	Location: LWC		Construction: Stainless Steel	
LWC sump receiver (Activity indeterminate)	1) Tank leak 2) Tank failure due to seismic event 3) Tank and ventilation failure due to seismic event 4) Tank, ventilation, and LWC failure due to seismic event 5) Overflow	- Stainless steel tank - Level recorder - High and low level alarms - Spills handled by LWC sump - Overflow contained by Tank 6D-3 - LWC, reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
4D-2 Volume: 4,160 L Name: Partition Cycle Waste C/H Tank (XC-1 sump receiver) Location: XC-1 Construction: Stainless Steel					
Liquid LLW (Activity indeterminate)	1) Tank leak 2) Tank failure due to seismic event 3) Tank and ventilation failure due to seismic event 4) Tank, ventilation, and XC-1 failure due to seismic event 5) Overflow	- Stainless steel tank - Level recorder - Spills handled by XC-1 sump - Overflow contained by Tank 6D-3 - XC-1, reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0
Analytical and Process Chemistry Laboratories					
Laboratory Reagents (Acids, Oxidizers, Corrosives, Poisons, Flammables)	1) Significant spill	- Small quantities used - Different categories of chemicals stored in separate locations - Isolation from environment - Restricted access/use - Chemical handling/storage per WVDP-011	1) Negligible	1) Unlikely	1) 0
Liquid Waste Treatment System					
35104 Volume: 22,000 L Name: LLW Collection Tank Location: GCR Extension Construction: Stainless Steel					
Processed high level waste solution (See Table B.4.1-1)	1) Tank Leak 2) Tank failure due to seismic event 3) Tank and ventilation failure due to seismic event 4) Tank, ventilation, and GCR Extension failure due to seismic event 5) Overflow	- Stainless steel tank - Level recorder - High and low level alarms - Sump in GCR - High level alarms in sump - GCR extension, reinforced concrete walls	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
5D-15B Volume: 56,950 L Name: Evaporator Feed Tank Location: UPC Construction: Stainless Steel					
Processed high level waste solution (See Table B.4.1-1)	1) Tank leak 2) Tank failure due to seismic event 3) Tank and ventilation failure due to seismic event 4) Tank, ventilation, and UPC failure due to seismic event 5) Overflow	- Stainless steel tank - Level recorder and indicator - Low and high alarms - Spills handled by UPC sump - Sump high level alarm - Stainless steel liner in UPC - Overflow contained by Tank 6D-3 - UPC, reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely Unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0
5D-15A1 Volume: 38,150 L Name: Evaporator Concentrates Tank Location: UPC Construction: Stainless Steel					
Evaporator concentrates (See Table B.4.1-1)	1) Tank leak 2) Tank failure due to seismic event 3) Tank and ventilation failure due to seismic event 4) Tank, ventilation, and UPC failure due to seismic event 5) Overflow	- Stainless steel tank - Level recorder - Level indicator - High and low alarms - Spills handled by UPC sump - Sump high level alarm - Stainless steel liner in UPC - Overflow contained by Tank 6D-3 - UPC reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0
5D-15A2 Volume: 18,990 L Name: Evaporator Concentrates Tank Location: UPC Construction: Stainless Steel					
Evaporator concentrates (See Table B.4.1-1)	1) Tank leak 2) Tank failure due to seismic event 3) Tank and ventilation failure due to seismic event 4) Tank, ventilation, and UPC failure due to seismic event 5) Overflow	- Stainless steel tank - Level recorder - Level indicator - High and low alarms - Spills handled by UPC sump - Sump high level alarm - Stainless steel liner in UPC - Overflow contained by Tank 6D-3 - UPC reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
71D-001 Volume: 535 L Name: Organic IX Column Location: XC-3 Construction: Stainless Steel					
Organic IX media and low TDS process solution (See Table B.4.1-1)	1) Column leak 2) Column failure due to seismic event 3) Column and ventilation failure due to seismic event 4) Column, ventilation, and XC3 failure due to seismic event 5) Column failure due to over-pressurization	- Stainless steel column - Spills handled by XC-3 sump - Sump high level alarm - Stainless steel liner in XC3 - XC3, reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0
71D-002 Volume: 1,820 L Name: Zeolite IX Column Location: XC-3 Construction: Stainless Steel					
Zeolite and low TDS processed high level waste solution (See Table B.4.1-1)	1) Column leak 2) Column failure due to seismic event 3) Column and ventilation failure due to seismic event 4) Column, ventilation, and XC3 failure due to seismic event 5) Column failure due to over-pressurization	- Stainless steel column - Spills handled by XC-3 sump - Sump high level alarm - Stainless steel liner in XC3 - XC3, reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
71D-003 Volume: 1,820 L Name: Zeolite IX Column Location: XC-3 Construction: Stainless Steel					
Zeolite and high level waste solution and distillate (See Table B.4.1-1)	1) Column leak 2) Column failure due to seismic event 3) Column and ventilation failure due to seismic event 4) Column, ventilation, and XC3 failure due to seismic event 5) Column failure due to over-pressurization	- Stainless steel column - High level alarm - Spills handled by XC-3 sump - High level alarm in the sump - Stainless steel liner in XC3 - XC3, reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0
71D-004 (31017) Volume: 5,680 L Name: LWTS Evaporator Location: XC-3 Construction: Stainless Steel					
Processed high level waste solution (See Table B.4.1-1)	1) Evaporator leak 2) Evaporator failure due to seismic event 3) Evaporator and ventilation failure due to seismic event 4) Evaporator, ventilation, and XC3 failure due to seismic event 5) Evaporator failure due to over-pressurization	- Stainless steel Evaporator - Level recorder - High, low alarms - Spills handled by XC-3 sump - Sump high level alarm - Stainless steel liner in XC3 - XC3, reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES
(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
71D-005 Volume: 3,785 L Name: Distillate Surge Tank Location: XC-3 Construction: Stainless Steel					
Processed high level waste distillate (See Table B.4.1-1)	1) Tank leak 2) Tank failure due to seismic event 3) Tank and ventilation failure due to seismic event 4) Tank, ventilation, and XC3 failure due to seismic event 5) Overflow	- Stainless steel tank - Level indicator - Spills handled by XC-3 sump - High level alarm in the sump - Stainless steel liner in XC3 - Overflow contained by Tank 6D-3 - XC3, reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0
71D-006 Volume: 4,650 L Name: Spent Resin Tank Location: XC-3 Construction: Stainless Steel					
Spent resin (See Table B.4.1-1)	1) Tank leak 2) Tank failure due to seismic event 3) Tank and ventilation failure due to seismic event 4) Tank, ventilation, and XC3 failure due to seismic event 5) Overflow	- Stainless steel tank - Level indicator - Spills handled by XC-3 sump - High level alarm in the sump - Stainless steel liner in XC3 - Overflow contained by Tank 6D-3 - XC3, reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0
71D-007 Volume: 4,650 L Name: Spent Zeolite Tank Location: XC-3 Construction: Stainless Steel					
Spent zeolite (See Table B.4.1-1)	1) Tank leak 2) Tank failure due to seismic event 3) Tank and ventilation failure due to seismic event 4) Tank, ventilation, and XC3 failure due to seismic event 5) Overflow	- Stainless steel tank - Level indicator - Spills handled by XC-3 sump - High level alarm in the sump - Stainless steel liner in XC3 - Overflow contained by Tank 6D-3 - XC3, reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
71D-008 Volume: 2,950 L Name: Filter Backwash Receiver Tank Location: XC-3 Construction: Stainless Steel					
Filter backwash solution (See Table B.4.1-1)	1) Tank leak 2) Tank failure due to seismic event 3) Tank and ventilation failure due to seismic event 4) Tank, ventilation, and XC3 failure due to seismic event 5) Overflow	- Stainless steel tank - Level indicator - Spills handled by XC-3 sump - High level alarm in the sump - Stainless steel liner in XC3 - Overflow contained by Tank 6D-3 - XC3, reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0
71D-009 Volume: 380 L Name: Feed Sample Tank Location: XC-3 Construction: Stainless Steel					
Processed high level waste solution (See Table B.4.1-1)	1) Tank leak 2) Tank failure due to seismic event 3) Tank and ventilation failure due to seismic event 4) Tank, ventilation, and XC3 failure due to seismic event 5) Overflow	- Stainless steel tank - Level indicator - Spills handled by XC-3 sump - High level alarm in the sump - Stainless steel liner in XC3 - Overflow contained by Tank 6D-3 - XC3, reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0
71D-011 Volume: 380 L Name: Low TDS Feed Tank Location: XC-3 Construction: Stainless Steel					
Processed high level waste (See Table B.4.1-1)	1) Tank leak 2) Tank failure due to seismic event 3) Tank and ventilation failure due to seismic event 4) Tank, ventilation, and XC3 failure due to seismic event 5) Overflow	- Stainless steel tank - Level indicator - High and low level alarms - Spills handled by XC-3 sump - High level alarm in the sump - Stainless steel liner in XC3 - Overflow contained by Tank 6D-3 - XC3, reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
71E-001 Volume: N/A	Name: Evaporator Reboiler	Location: XC-3	Construction: Stainless Steel		
Evaporator concentrates (See Table B.4.1-1)	1) Reboiler leak 2) Reboiler failure due to seismic event 3) Reboiler and ventilation failure due to seismic event 4) Reboiler, ventilation, and XC3 failure due to seismic event	- Stainless steel Reboiler - High level alarm - Spills handled by XC-3 sump - High level alarm in the sump - Stainless steel liner in XC3 - XC3, reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible	1) 0 2) 0 3) 0 4) I
71E-005 Volume: N/A	Name: Concentrates Cooler	Location: XC-3	Construction: Stainless Steel		
Evaporator concentrates (See Table B.4.1-1)	1) Cooler leak 2) Cooler failure due to seismic event 3) Cooler and ventilation failure due to seismic event 4) Cooler, ventilation, and XC3 failure due to seismic event	- Stainless steel cooler - Spills handled by XC-3 sump - High level alarm in the sump - Stainless steel liner in XC3 - XC3, reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible	1) 0 2) 0 3) 0 4) I
14D-7 Volume: 375 L	Name: HNO_3 Tank	Location: LXA	Construction: Stainless Steel		
2 M nitric acid	1) Tank leak 2) Tank failure due to seismic event 3) Tank and ventilation failure due to seismic event 4) Tank, ventilation, and LXA failure due to seismic event 5) Overflow	- Stainless steel tank - Level indicator - High level alarm - Drains to interceptor - Overflow contained by Tank 6D-3 - Berm (full capacity)	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES
(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
14D-18 Volume: 375 L Name: NaOH Tank Location: LXA Construction: Stainless Steel					
Sodium hydroxide	1) Tank leak 2) Tank failure due to seismic event 3) Tank and ventilation failure due to seismic event 4) Tank, ventilation, and LXA failure due to seismic event 5) Overflow	- Stainless steel tank - Level indicator - High level alarm - Drains to interceptor - Overflow contained by Tank 6D-3 - Berm (full capacity)	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0
Vessel Off-Gas System					
Airborne contamination	1) Filter failure	- Differential pressure monitoring instrumentation - Backup HEPA filters	1) Negligible	1) Unlikely	1) 0
Contaminated liquid	1) Seismic failure of OGC vessel 2) OGC vessel leak 3) Failure of pump 6G-2 seal 4) Overfill of 6D-3 or 6C-3	- Sumps and sump alarms in OGC - OGC bermed - Pumps contained in sealed niche with drains to OGC sump - Level indicator on vessels - Level alarm on 6D-3	1) Negligible 2) Negligible 3) Negligible 4) Negligible	1) Extremely unlikely 2) Unlikely 3) Unlikely 4) Unlikely	1) 0 2) 0 3) 0 4) 0
6D-3 Volume: 910 L Name: VOG Condensate Catch Tank Location: OGC Construction: Stainless Steel					
Liquid LLW (See Table B.4.1-1)	1) Tank leak 2) Tank failure due to seismic event 3) Tank and ventilation failure due to seismic event 4) Tank, ventilation, and OGC failure due to seismic event 5) Overflow	- Stainless steel tank - Level indicator - High level alarm - Spills handled by OGC sump - Overflow contained by Tank 7D-8 - OGC, reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
6D-6 Volume: 240 L	Name: VOG Condensate Knockout Pot	Location: OGC	Construction: Stainless Steel		
Liquid LLW (See Table B.4.1-1)	1) Tank leak 2) Tank failure due to seismic event 3) Tank and ventilation failure due to seismic event 4) Tank, ventilation, and OGC failure due to seismic event 5) Overflow	- Stainless steel tank - Spills handled by OGC sump - Overflow contained by Tank 6D-3 - OGC, reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low 5) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible 5) Unlikely	1) 0 2) 0 3) 0 4) I 5) 0
6C-3 Volume: 1,500 L	Name: VOG Scrubber	Location: OGC	Construction: Stainless Steel		
Liquid LLW (potentially caustic) (See Table B.4.1-1)	1) Scrubber leak 2) Scrubber failure due to seismic event 3) Scrubber and ventilation failure due to seismic event 4) Scrubber, ventilation, and OGC failure due to seismic event	- Stainless steel tank - Spills handled by OGC sump - Overflow contained by Tank 6D-3 - OGC, reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible	1) 0 2) 0 3) 0 4) I
6V-1 Volume: 200 L	Name: VOG Cyclone	Location: OGC	Construction: Stainless Steel		
Liquid LLW (See Table B.4.1-1)	1) Cyclone leak 2) Cyclone failure due to seismic event 3) Cyclone and ventilation failure due to seismic event 4) Cyclone, ventilation, and OGC failure due to seismic event	- Stainless steel tank - Spills handled by OGC sump - OGC, reinforced concrete	1) Negligible 2) Negligible 3) Negligible 4) Low	1) Extremely unlikely 2) Extremely unlikely 3) Incredible 4) Incredible	1) 0 2) 0 3) 0 4) I

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
CEMENT SOLIDIFICATION SYSTEM					
Airborne contamination	1) Loss of line power to blower 2) Failure of HEPA filter 3) Explosion in WDC or Process Cell 4) Backflow due to direct tornado strike	- Redundant vent blowers - Isolation dampers on final HEPA filters - No explosives stored in cell - Barriers to direct cell access (airlocks)	1) Negligible 2) Negligible 3) Negligible 4) Negligible	1) Anticipated 2) Unlikely 3) Incredible 4) Extremely unlikely	1) 0 2) 0 3) 0 4) 0
Contaminated liquid/cement release (See Table B.4.1-1)	1) Seismic failure of transfer line from LWTS to CSS 2) Seismic failure of WDV 3) Failure of WDV seal 4) Failure of transfer line in process cell 5) Faulty discharge valve on cement mixer 6) Drum not in position under fill head 7) Lid not removed from drum to be filled	- Sumps in WDC and Process Cell - Berm in WDC - Valve position indicator/load cell on mixers - Position indicator/load cell on drum conveyor station - Lid vacuum indicator - CCTV to allow viewing of cell operations	1) Low 2) Negligible 3) Negligible 4) Negligible 5) Negligible 6) Negligible 7) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Unlikely 4) Extremely unlikely 5) Unlikely 6) Unlikely 7) Unlikely	1) 1 2) 0 3) 0 4) 0 5) 0 6) 0 7) 0
70D-001 Volume: 1,900 L Name: Waste Dispensing Vessel Location: CSS WDC Construction: Stainless Steel					
LWTS concentrates (See Table B.4.1-1)	1) Vessel leak 2) Vessel failure due to seismic event 3) Overflow	- Stainless steel vessel - Level recorder - Level indicator - Low, high level alarms - Spills contained by WDC sump - High level alarm in sump - Stainless steel liner in WDC - WDC, reinforced concrete	1) Negligible 2) Negligible 3) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Unlikely	1) 0 2) 0 3) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES
(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
70V-001 Volume: 1,160 L Name: Additive Day Tank Location: CSS Change Room Construction: Carbon Steel **70V-001 Volume: 5,700 L** Name: Additive Bulk Storage Tank Location: CSS Change Room Construction: Polyethylene					
Sodium silicate	1) Tank leak 2) Tank failure	- Carbon steel tank or polyethylene tank - Level indicator - Berm	1) Negligible 2) Negligible	1) Extremely unlikely 2) Extremely unlikely	1) 0 2) 0
70K-002 Volume: 114 L Name: High Sheer Mixer Location: CSS Process Room Construction: Carbon Steel **70K-004 Volume: 114 L** Name: High Sheer Mixer Location: CSS Process Room Construction: Carbon Steel					
LWTS concentrates (See Table B.4.1-1)	1) Mixer leak 2) Mixer failure due to seismic event 3) Overflow	- Mixer housing - Level recorder - Level indicator - Low, high alarms - Spills handled by Process Room sump - High level alarm in sump	1) Negligible 2) Negligible 3) Negligible	1) Extremely unlikely 2) Extremely unlikely 3) Unlikely	1) 0 2) 0 3) 0
70D-004 Volume: 70 m^3 Name: Cement Silo Location: South of 01-14 Building Construction: Carbon Steel					
Dry portland cement	1) Silo failure	- None	1) Negligible	1) Extremely unlikely	1) 0
70V-010 Volume: 0.42 m^3 Name: Cement Day Bin Location: Second floor of 01-14 Building Construction: Carbon Steel					
Dry portland cement	1) Bin failure	- None	1) Negligible	1) Incredible	1) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
7D-13 Volume: 7,700 L Name: Lab Drains/Decon Catch Tank Location: Yard, west of Plant Construction: Stainless Steel					
Liquid LLW (See Table B.4.1-1)	1) Tank leak 2) Tank failure due to seismic event 3) Overflow	- Level indicator - Overflow contained by Tank 6D-3	1) Negligible 2) Negligible 3) Negligible	1) Unlikely 2) Extremely unlikely 3) Unlikely	1) 0 2) 0 3) 0
DRUM CELL					
Solidified cement drums located in the Drum Cell, 71 gallons					
Solid LLW (See Table B.4.1-2)	1) Drum failure	- Carbon steel drum - Concrete shield walls - Administrative controls on waste form integrity requirements	1) Negligible	1) Extremely unlikely	1) 0
LOW LEVEL WASTE TREATMENT SYSTEM					
Low Level Liquid Radioactive Waste; Spent Resin;	1) Introduction of high activity waste into system 2) Failure of transfer line (interceptors to Lagoon 2 or Lagoon 2 to LLW2) 3) Loss of spent IX resin containment in LLW2	- Analysis of liquids in interceptors prior to transfer to Lagoon 2 - Sloped floor to floor drains in LLW2 where IX resin is handled	1) Negligible 2) Negligible 3) Negligible	1) Unlikely 2) Extremely unlikely 3) Unlikely	1) 0 2) 0 3) 0
No ID # Volume: 1,900 L Name: Neutralization Pit Location: Yard, east of Plant Construction: S.S. lined concrete					
Liquid Low-Level Waste (See Table B.7.5-1 Max. Lagoon 2 Conc.)	1) Failure of pit containment 2) Overflow	- Pit located in silty till - Routine surveillance by qualified operators	1) Negligible 2) Negligible	1) Extremely unlikely 2) Unlikely	1) 0 2) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
No ID # Volume: 87,000 L Name: N/S Interceptors Location: Yard, east of Plant Construction: S.S. lined concrete					
Liquid LLW (See Table B.7.5-1 Max. Lagoon 2 Conc.)	1) Failure of pit containment 2) Overflow	- Interceptors located in silty till - Overflow to off-line interceptor - High level alarm - Routine area surveillance by qualified operators	1) Negligible 2) Negligible	1) Extremely unlikely 2) Unlikely	1) 0 2) 0
No ID # Volume: 57,000 L Name: Old Interceptor Location: Yard, east of Plant Construction: Concrete					
Liquid LLW (See Table B.7.5-1 Max. Lagoon 2 Conc.)	1) Failure of pit containment 2) Overflow	- Interceptor located in silty till - Routine surveillance by qualified operators	1) Negligible 2) Negligible	1) Extremely unlikely 2) Unlikely	1) 0 2) 0
No ID # Volume: 9,500,000 L Name: Lagoon 2 Location: Yard, east of Plant Construction: Clay-lined basin **No ID # Volume: 12,000,000 L** Name: Lagoon 3 Location: Yard, east of Plant Construction: Clay-lined basin					
Liquid LLW (See Table B.7.5-1 Max. Lagoon 2 Conc.)	1) Seismic failure of basin containment 2) Overflow of Lagoon 2 3) Overflow of Lagoon 3 4) High activity, Lagoon 3 5) Loss of lagoon confinement integrity	- Basins constructed in silty till - Level in basins maintained with sufficient margin to accommodate precipitation - Lagoon 2 overflow to Lagoon 3 - Lagoon 3 sampled prior to discharge; off-spec solutions to Lagoon 2 - Monitoring well downgradient	1) Moderate 2) Negligible 3) Negligible 4) Negligible 5) Negligible	1) Extremely unlikely 2) Unlikely 3) Unlikely 4) Unlikely 5) Extremely unlikely	1) 3 2) 0 3) 0 4) 0 5) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
No ID # Volume: 770,000 L Name: Lagoon 4 Location: Yard, east of Plant Construction: Synthetic-lined basin **No ID # Volume: 630,000 L** Name: Lagoon 5 Location: Yard, east of Plant Construction: Synthetic-lined basin					
Decontaminated LLW (See Table B.7.5-1 Max. Lagoon 2 Conc.)	1) High activity in lagoon 2) Loss of lagoon confinement integrity 3) Overflow	- Analysis of contents prior to transfer to Lagoon 3 - Monitoring wells downgradient - Impermeable synthetic liners - Routine area surveillance by qualified operators during operation	1) Negligible 2) Negligible 3) Negligible	1) Unlikely 2) Extremely unlikely 3) Unlikely	1) 0 2) 0 3) 0
02-D-0301 Volume: 100 Gallons Name: Sulfuric Acid Tank Location: LLW2 Construction: Polyethylene					
93% H_2SO_4	1) Vessel leak 2) Vessel failure 3) Fill line rupture 4) Overflow	- Vessel on sloped floor to drain emptying into sump - Level indicator - Handling and storage activities conducted per WVDP-011	1) Negligible 2) Low [WVDP-193] 3) Negligible 4) Negligible	1) Unlikely 2) Extremely unlikely 3) Extremely unlikely 4) Extremely unlikely	1) 0 2) 1 3) 0 4) 0
02-D-0102 Volume: 800 Gallons Name: Surge Tank A Location: LLW2 Construction: Carbon Steel					
Liquid LLW	1) Vessel leak 2) Vessel failure 3) Overflow	- Vessel on sloped floor to drain emptying into sump - Level indicator - Overflow to Lagoon 2	1) Negligible 2) Negligible 3) Negligible	1) Unlikely 2) Extremely unlikely 3) Anticipated	1) 0 2) 0 3) 0
02-D-0202 Volume: 800 Gallons Name: Surge Tank A Location: LLW2 Construction: Carbon Steel					
Liquid LLW	1) Vessel leak 2) Vessel failure 3) Overflow	- Vessel on sloped floor to drain emptying into sump - Level indicator - Overflow to Lagoon 2	1) Negligible 2) Negligible 3) Negligible	1) Unlikely 2) Extremely unlikely 3) Anticipated	1) 0 2) 0 3) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
02-C-0104 02-C-0105 02-C-0106 02-C-0204 02-C-0205 02-C-0206 **Volume: 50 ft^3** Name: Ion Exchange Columns Location: LLW2 Construction: Carbon Steel					
Liquid LLW Contaminated IX Resin (See Table 7.5-1 Max. Lagoon 2 Conc.)	1) Column leak 2) Column failure 3) Column over pressurization 4) Natural phenomena induced release of all resin in ion exchange columns (no workers assumed to be in vicinity) 5) Operational mishap releases spent resin during sluicing of resin from ion exchange columns to storage containers 6) Storage containers breached during transfer to temporary storage	- Drains to skid pan which drains to floor drain which empties into sump - Pressure indictor - Established procedures and training for operations and sluicing activities - Established procedures and training for forklift and other vehicle operations - Spent resin contained in formidable, sealed, noncombustible storage containers	1) Negligible 2) Negligible 3) Negligible 4) Negligible 5) Low 6) Negligible	1) Unlikely 2) Extremely unlikely 3) Unlikely 4) Extremely Unlikely 5) Unlikely 6) Anticipated	1) 0 2) 0 3) 0 4) 0 5) 2 6) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
		NORTH PLATEAU PUMP SYSTEM			
Contaminated Groundwater	1) System breach for any reason (e.g., natural phenomena or operational mishap) leads to environmental release and/or momentary worker inundation in contaminated groundwater 2) Failure of transfer line (NPPS Surge Tank to NP Surge Tank in LLW2)	- Maximum combined flow from all three wells limited to 20 gpm - Equipment enclosed in shelter - Alarms and interlocks on various parameters, including well enclosure leak detection alarm, and high-high level in surge tank alarm (which auto-stops well pumps also). All alarms transmitted to Keltron Panel in Main Security Gate House.	1) Negligible 2) Negligible	1) Anticipated 2) Unlikely	1) 0 2) 0
02-T-0101	**Volume: 900 Gallons**	Name: Sand Filter (Tank)	Location: LLW2 Construction: Carbon Steel		
Liquid Low-Level Waste	1) Filter tank leak 2) Filter tank overflow 3) Filter tank failure	- Floor slopes to floor drain which empties into sump - Level instrumentation - Overflow to LLW2 sump	1) Negligible 2) Negligible 3) Negligible	1) Unlikely 2) Anticipated 3) Extremely unlikely	1) 0 2) 0 3) 0
		NDA Interceptor Trench Liquid Pretreatment System (LPS)			
Solvent	1) Major spill/release for any reason 2) Fire	- Alarms and interlocks on various parameters, including high level alarms on tanks - Berms around tanks and piping - Equipment enclosed in shelter - Established procedures and training for system operations and handling and transfer of drums containing hazardous material - Continuous Air Monitors in key locations	1) Negligible 2) Low	1) Unlikely 2) Unlikely	1) 0 2) 2

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
Contaminated Water in the LPS System/Tanks	1) Release to the environment of all or major portion of water in the LPS system/tanks for any reason	- Same as above for solvent hazard	1) Negligible	1) Unlikely	1) 0
Spent (or Partially Used) Granular Activated Carbon	1) Major spill/release for any reason 2) Fire 3) Fire with solvent fire as the ignition source	- Same as above for solvent hazard	1) Negligible 2) Low 3 Low	1) Unlikely 2) Extremely unlikely 3) Unlikely	1) 0 2) 1 3) 2
UTILITY ROOM AND YARD					
PCBs in Main Plant Transformer	1) Puncture resulting in spill 2) Transformer fire	- 8 ft. chain link fence surrounding transformer	1) Moderate 2) Low	1) Extremely unlikely 2) Unlikely	1) 3 2) 2
Sodium Hypochlorite 208 L	1) Rupture of drum 2) Fire involving drum		1) Negligible 2) Negligible	1) Unlikely 2) Unlikely	1) 0 2) 0
50% NaOH 1670 L	1) Puncture resulting in spill	- Administrative control prohibiting forklift use in Utility Room	1) Negligible	1) Unlikely	1) 0
93% H_2SO_4 210 L	1) Puncture resulting in spill	- Administrative control prohibiting forklift use in Utility Room	1) Negligible	1) Unlikely	1) 0
31D-2 **Volume: 38,000 L** Name: Fuel Oil Storage Tank Location: Yard, east of Plant Construction: Carbon steel					
No. 2 Fuel Oil	1) Tank leak 2) Tank failure	- Spill basin (87,000 L) under tank - Sight glass and level indicator	1) Negligible 2) Negligible	1) Unlikely 2) Unlikely	1) 0 2) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
G-01Volume: 3,750 L Name: Diesel Fuel Storage Tank Location: Yard, east of New Warehouse Construction: Concrete with Double Steel Liner **G-02Volume: 7,500 L** Name: Gasoline Storage Tank Location: Yard, east of New Warehouse Construction: Concrete with Double Steel Liner					
Gasoline, Diesel Fuel	1) Tank leak with no fire 2) Tank failure with no fire 3) Gas Tank leak with ensuing fire 4) Diesel Tank failure with ensuing fire	- Multiple confinement barriers - Leak detection equipment installed - Anti-siphon device - Pump integral to tank	1) Negligible 2) Negligible 3) Low 4) Moderate	1) Unlikely 2) Extremely unlikely 3) Extremely unlikely 4) Incredible	1) 0 2) 0 3) 1 4) I
LAG STORAGE AND WASTE COMPACTION					
Lag Storage Facilities					
Airborne Contamination	1) Container failure due to tornado strike 2) Container failure due to seismic event 3) Container failure due to lightning strike 4) Fire originating external to containers leads to breaching of containers 5) Container explodes or overpressurizes due to gas buildup, spontaneous heating/combustion, or exothermic reaction due to mixing of incompatible chemicals 6) Forklift induced breaching of a container (e.g., tine puncture, forklift dropped or crushed) 7) Failure of pressurized bottle (possibly being used for maintenance work) leads to breaching of a container 8) Propane tank on propane powered forklift leaks and leads to detonation that breaches containers, but no ensuing fire 9) Sufficiently energetic event in nearby facility,	- Wastes contained in formidable, sealed, DOT-approved noncombustible containers - No significant combustion sources stored in facility - Fire detection provided in some key areas, such as the Lag Storage Building, and manual pull stations - WVDP Fire Brigade and West Valley Volunteer Hose Company - Established procedures and training for forklift operations - Design and installation of electrical equipment to accepted electrical industry standards - Security roving watches - Procedures that govern the types (mixtures) of waste materials that can be placed in a given container, and the amounts of radioactive materials per container - Emergency Team's response after event - Prompt evacuation of personnel in the area of the event and downwind - Periodic inspections of container integrity	1) Negligible [1] 2) Low 3) Low 4) Moderate [1] 5) Low 6) Negligible 7) Low 8) Low 9) Moderate 10) Negligible	1) Extremely Unlikely 2) Unlikely 3) Extremely Unlikely 4) Extremely Unlikely 5) Unlikely 6) Antici-pated 7) Extremely Unlikely 8) Extremely Unlikely 9) Incredible 10) Anticipated	1) 0 2) 2 3) 1 4) 3 5) 2 6) 0 7) 1 8) 1 9) I 10 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
Gamma Radiation External Exposure from Radioactive Material Displaced from Its Container	Same as events 1) through 10) above for airborne contamination hazard	- Same as above for airborne contamiantion hazard - Extremely low likelihood of any gamma source in a waste container that could give even a one rem dose over many minutes and in close proximity to a receptor	1) Negligible 2) Negligible 3) Negligible 4) Negligible 5) Negligible 6) Negligible 7) Negligible 8) Negligible 9) Negligible 10) Negligible	1) Extremely Unlikely 2) Unlikely 3) Extremely Unlikely 4) Extremely Unlikely 5) Unlikely 6) Anticipated 7) Extremely Unlikely 8) Extremely Unlikely 9) Incredible 11) Anticipated	1) 0 2) 0 3) 0 4) 0 5) 0 6) 0 7) 0 8) 0 9) I 10) 0
Liquid Release Leading to Skin Contamination and/or Ingestion Dose	Same as events 1) through 10) above for airborne contamination hazard	- Same as above for airborne contamination hazard - Extensive washing with special cleansers as necessary would be performed immediately to remove skin contamination - Training to keep hands away from eyes, nose, mouth when contaminated, thereby minimizing likelihood of ingestion dose - Generally very low concentrations of alpha particle emitting radionuclides in waste - Very little free liquids as of function of volume of all wastes	1) Negligible 2) Low 3) Low 4) Low 5) Low 6) Negligible 7) Negligible 8) Low 9) Low 11) Negligible	1) Extremely Unlikely 2) Unlikely 3) Extremely Unlikely 4) Extremely Unlikely 5) Unlikely 6) Antici-pated 7) Extremely Unlikely 8) Extremely Unlikely 9) Incredible 12) Anticipated	1) 0 2) 2 3) 1 4) 1 5) 2 6) 0 7) 0 8) 1 9) I 10) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
Exposure to Hazardous (Non-Radioactive) Material Displaced from Its Container	Same as events 1) through 10) above for airborne contamination hazard	- Same as above for airborne contamination hazard - Extensive washing with special cleansers as necessary would be performed immediately for skin exposure - Training to keep hands away from eyes, nose, mouth when exposed to hazardous materials	1) Negligible 2) Low 3) Low 4) Low 5) Low 6) Negligible 7) Negligible 8) Low 9) Low 10) Negligible	1) Extremely Unlikely 2) Unlikely 3) Extremely Unlikely 4) Extremely Unlikely 5) Unlikely 6) Anticipated 7) Extremely Unlikely 8) Extremely Unlikely 9) Incredible 10) Anticipated	1) 0 2) 2 3) 1 4) 1 5) 2 6) 0 7) 0 8) 1 9) I 10) 0
Chemical Process Cell - Waste Storage Area (CPC-WSA)					
Airborne Contamination	1) Container failure due to tornado strike 2) Container failure due to seismic event 3) Container failure due to lightning strike 4) Fire originating external to containers leads to breaching of containers 5) Improper hoisting and rigging activities and/or equipment failure leads to impact (e.g., dropping) induced breaching of a container 6) Sufficiently energetic event in nearby facility, fuel storage tank, or gas powered vehicle breaches containers 7) Container breach due to corrosion or other material defect, or container manufacturing deficiency	- Wastes contained in formidable, sealed, noncombustible containers - No significant combustion sources - WVNS Fire Brigade and West Valley Volunteer Hose Company. - Established procedures and training for rigging and hoisting operations - Security roving watches - Emergency Team's response after event - Prompt evacuation of personnel in the area of the event and downwind - Periodic inspections of container integrity	1) Low 2) Low 3) Low 4) Low 5) Low 6) Moderate 7) Low	1) Extremely Unlikely 2) Unlikely 3) Extremely Unlikely 4) Incredible 5) Unlikely 6) Incredible 7) Unlikely	1) 1 2) 2 3) 1 4) I 5) 2 6) I 7) 2

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
Gamma Radiation External Exposure Beyond That Normally Incurred	1) Tornado strike 2) Seismic event 3) Lightning strike 4) Fire originating external to containers (e.g., Sprung structure catches on fire) leads to breaching of containers or reconfiguration of containers 5) Improper hoisting and rigging activities and/or equipment failure leads to impact (e.g., dropping) induced breaching of a container or reconfiguration of containers 6) Sufficiently energetic event in nearby facility, fuel storage tank, or gas powered vehicle breaches or reconfigures containers 7) Container breach due to corrosion or other material defect, or container manufacturing deficiency	- Wastes contained in formidable, sealed, noncombustible containers - No significant combustion sources - WVNS Fire Brigade and West Valley Volunteer Hose Company. - Established procedures and training for rigging and hoisting operations - Security roving watches - Emergency Team's response after event - Prompt evacuation of personnel in the area of the event - Periodic inspections of containers' integrity and configuration	1) Low 2) Low 3) Low 4) Low 5) Low 6) Moderate 7) Low	1) Extremely Unlikely 2) Unlikely 3) Extremely Unlikely 4) Extremely Unlikely 5) Unlikely 6) Incredible 7) Unlikely	1) 1 2) 2 3) 1 4) 1 5) 2 6) I 7) 2
Waste Reduction and Packaging Area Compactor					
Solid LLW (See Table B.7.7-2)	1) Failure of HEPA filter	- Administrative controls precluding facility operation without ventilation support - HEPA filter differential pressure indicator	1) Negligible	1) Unlikely	1) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES
(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
		Contact Size Reduction Facility			
Solid LLW (See Table B.7.7-2)	1) Failure of HEPA filter 2) Failure of ventilation system blower 3) Fire	- High/low differential pressure instrumentation of HEPA filters - Backup ventilation support provided by Main Plant HEV - Administrative controls prohibiting combustibles in cutting area	1) Negligible 2) Negligible 3) Negligible	1) Unlikely 2) Unlikely 3) Extremely unlikely	1) 0 2) 0 3) 0
15D-6 Volume: 5,700 L	Name: HEV/CSRF Waste Catch Tank	Location: Yard east of MSM	Construction: Stainless Steel		
Liquid LLW [1] 6E-4 μCi/mL Sr-90 0.15 μg/g Total U	1) Tank leak 2) Tank failure	- Tank constructed of stainless steel - High level alarm - Tank located in silty till	1) Negligible 2) Negligible	1) Unlikely 2) Extremely unlikely	1) 0 2) 0
		STORAGE AREAS			
		Hazardous Waste Storage Facility			
Hazardous Wastes	1) Container failure/leak 2) Fire	- Audible and visible spill detection alarm - 255-gal capacity sump - Individual locker vents - Explosion-proof electrical lights, fixtures, and switches - Automatic dry chemical fire extinguishing system - Exterior local visual fire alarm and light on each unit - Restricted access/use - Quantity restrictions imposed by WVDP-073	1) Negligible 2) Low [6]	1) Unlikely 2) Extremely unlikely	1) 0 2) 1

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
Interim Waste Storage Facility					
Miscellaneous hazardous and mixed wastes	1) Container failure 2) Fire	- Collective berm for all containers - Class A (automatic) dry foam fire suppression	1) Negligible 2) Low [6]	1) Unlikely 2) Extremely unlikely	1) 0 2) 1
New Warehouse					
New Warehouse - Acid Room					
$Zr(NO_3)_4$ 1460 L	1) Puncture resulting in spill 2) Fire	- Berms in each individual room (but not general warehouse) - Each room has its own vent system - Restricted access controlled by warehouse manager - Automatic sprinkler system	1) Negligible 2) Low	1) Unlikely 2) Extremely unlikely	1) 0 2) 1
$ZnBr_2$ 1250 L	1) Puncture resulting in spill 2) Fire	- Berms in each individual room (but not general warehouse) - Each room has its own vent system - Restricted access controlled by warehouse manager	1) Negligible 2) Low	1) Unlikely 2) Extremely unlikely	1) 0 2) 1
H_2SO_4 760 kg	1) Puncture resulting in spill 2) Fire	- Berms in each individual room (but not general warehouse) - Each room has its own vent system - Restricted access controlled by warehouse manager	1) Negligible 2) Low	1) Unlikely 2) Extremely unlikely	1) 0 2) 1
New Warehouse - Oxidizer Room					
HNO_3 10 L	1) Spill from puncture/container failure 2) Over-pressurization during a fire	- Berms in each individual room (but not general warehouse) - Each room has its own vent system - Restricted access controlled by warehouse manager	1) Low 2) Low	1) Extremely unlikely 2) Extremely unlikely	1) 1 2) 1

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
$NaNO_3$ 360 kg	1) Puncture resulting in spill 2) Fire	- Berms in each individual room (but not general warehouse) - Each room has its own vent system - Restricted access controlled by warehouse manager	1) Negligible 2) Low	1) Unlikely 2) Extremely unlikely	1) 0 2) 1
$NaNO_2$ 680 kg	1) Puncture resulting in spill 2) Fire	- Berms in each individual room (but not general warehouse) - Each room has its own vent system - Restricted access controlled by warehouse manager	1) Negligible 2) Low	1) Unlikely 2) Extremely unlikely	1) 0 2) 1
H_2O_2 1250 L	1) Spill from puncture/container failure 2) Over-pressurization during a fire	- Berms in each individual room (but not general warehouse) - Each room has its own vent system - Restricted access controlled by warehouse manager	1) Moderate 2) Low	1) Extremely unlikely 2) Extremely unlikely	1) 3 2) 1
New Warehouse - Caustic Room					
KOH 210 L	1) Puncture resulting in spill 2) Fire	- Berms in each individual room (but not general warehouse) - Each room has its own vent system - Restricted access controlled by warehouse manager	1) Negligible 2) Low	1) Unlikely 2) Extremely unlikely	1) 0 2) 1
NaOH 320 kg	1) Puncture resulting in spill 2) Fire	- Berms in each individual room (but not general warehouse) - Each room has its own vent system - Restricted access controlled by warehouse manager	1) Negligible 2) Low	1) Unlikely 2) Extremely unlikely	1) 0 2) 1

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
		NRC-LICENSED DISPOSAL AREA			
Release of buried radioactive waste	1) Seismic event 2) Airplane crash 3) Meteorite strike	- Waste is buried	1) Low 2) High 3) High	1) Unlikely 2) Incredible 3) Incredible	1) 2 2) I 3) I
		VITRIFICATION TEST FACILITY			
Nonradioactive chemicals used for melter feed	1) Tank/vessel (e.g., SMT, FHT, MFT, CHT) leak 2) Tank/vessel failures due to seismic event 3) Tank/vessel overflow 4) Fire leading to tank/vessel breach 5) Tank/vessel off-gas system malfunction/failure	- Formidable and corrosion resistant tank/vessel materials of construction - Liquid level indicators - Pressure indicators - Emergency vents open on high pressure - Sloping floor directs spilled materials to bermed area along North wall of VTF - Auto-termination of feed process on high melter pressure, low scrubber water flow, or select off-gas system parameter abnormalities - Very small amounts of combustible material in the VTF making fire of any intensity or duration very remote - WVDP Fire Brigade and West Valley Volunteer Hose Company. - Off-gas analyzers and associated alarms	1) Negligible [8] 2) Low [8] 3) Negligible 4) Low 5) Negligible	1) Unlikely 2) Unlikely 3) Unlikely 4) Extremely Unlikely 5) Anticipated	1) 0 2) 2 3) 0 4) 1 5) 0

TABLE B.9.1-1

PROCESS HAZARDS ANALYSIS FOR THE MAIN PLANT AND WASTE PROCESSING FACILITIES

(All footnotes are located at the end of the table) (continued)

Hazard *	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor **
Ammonia (used in fluidized bed reactor)	1) Failure of an ammonia storage cylinder 2) Concurrent failure of 3 or 4 ammonia storage cylinders 3) Off-gas system malfunction releases excessive ammonia from VTF stock	- Off-gas analyzers and associated alarms - Stand-alone ammonia storage room (ASR), normally unoccupied - Industry standard ammonia storage cylinders - Ammonia detectors (and alarms) located in the area of the ASR	1) Negligible [8] 2) Low [8] 3) Negligible	1) Unlikely 2) Extremely Unlikely 3) Anticipated	1) 0 2) 1 3) 0

* Materials-at-risk are determined using:
the vessel volume indicated, in conjunction with the referenced hazard concentration; or the volume of hazardous material indicated.

** See Section B.9.1.1.2 for an explanation of Risk Factor.

References:

[1] - Dames & Moore, 1995

[2] - Prowse, 1991

[3] - Wolniewicz, 1993

[4] - Roberts, 1990

[5] - Yuan, 1991

[6] - WVDP-193

[7] - Gates, 1994

[8] - Kupp, 1992

TABLE B.9.2-1

FAILURE OF MAIN PLANT HEPA FILTER BANK

Assumptions:
Airborne Release Fraction (ARF)[1] 1.0E-2 Damage Ratio (DR)[1] 1.0
Respirable Fraction (RF)[1] 1.0 Leakpath Factor (LPF)[1] 1.0

Number of Failed HEPA Filters 30 HEPA Filter Exposure Rate 10 R/hr
Release Height 60 m HEPA Filter Cs-137 Activity[2] 225 Ci

Receptor Location			640 m	640 m	640 m	1050 m	1050 m	1700 m	
Stability Class, Wind Speed			D, 4.5m/s	F, 1m/s	95%	D, 4.5m/s	F, 1m/s	95%	
Dispersion (X/Q)			1.59E-06 s/m^3	2.70E-11 s/m^3	1.63E-04 s/m^3	5.54E-06 s/m^3	1.03E-07 s/m^3	6.72E-05 s/m^3	
Nuclide	Normalized Spent Fuel Activity[3]	Source Term (Ci)	On-Site Dose (rem)	On-Site Dose (rem)	On-Site Dose (rem)	Off-Site Dose (rem)	Off-Site Dose (rem)	Off-Site Dose (rem)	Percent Dose Contribution
Pu-238	5.32E-02	1.20E-01	2.91E-02	4.95E-07	2.99E+00	1.02E-01	1.89E-03	1.23E+00	46.5%
Am-241	1.58E-02	3.56E-02	9.80E-03	1.66E-07	1.01E+00	3.42E-02	6.35E-04	4.14E-01	15.6%
Pu-239	1.36E-02	3.07E-02	8.29E-03	1.41E-07	8.50E-01	2.89E-02	5.37E-04	3.50E-01	13.2%
Pu-240	1.04E-02	2.34E-02	6.32E-03	1.07E-07	6.48E-01	2.20E-02	4.09E-04	2.67E-01	10.1%
Pu-241	5.14E-01	1.16E+00	6.12E-03	1.04E-07	6.27E-01	2.13E-02	3.96E-04	2.59E-01	9.8%
Sr-90	9.26E-01	2.08E+00	1.44E-03	2.45E-08	1.48E-01	5.03E-03	9.34E-05	6.10E-02	2.3%
Cm-244	2.31E-03	5.20E-03	7.43E-04	1.26E-08	7.62E-02	2.59E-03	4.81E-05	3.14E-02	1.2%
Am-243	7.27E-04	1.64E-03	4.50E-04	7.64E-09	4.62E-02	1.57E-03	2.92E-05	1.90E-02	0.7%
U-232	2.41E-04	5.43E-04	1.93E-04	3.27E-09	1.98E-02	6.71E-04	1.25E-05	8.14E-03	0.3%
Am-242m	1.22E-04	2.75E-04	7.44E-05	1.26E-09	7.62E-03	2.59E-04	4.82E-06	3.14E-03	0.1%
U-233	3.40E-04	7.65E-04	5.26E-05	8.94E-10	5.40E-03	1.83E-04	3.41E-06	2.22E-03	0.1%
Cs-137	1.00E+00	2.25E+00	3.84E-05	6.53E-10	3.94E-03	1.34E-04	2.49E-06	1.62E-03	0.1%
		TOTAL TEDE	6.27E-02	1.07E-06	6.43E+00	2.19E-01	4.06E-03	2.65E+00	99.9%

Notes:
[1] - Based on Section 5.4.2.2, DOE-HDBK-3010-94.
[2] - HEPA activity based on 1 R/hr per 0.75 Ci Cs-137; Ref: WVNS Letter HE:85:0016.
[3] - Based on normalized spent fuel activity; Ref: CN:93:0015.

TABLE B.9.2-2

LAG STORAGE FACILITY FIRE

Assumptions:
Airborne Release Fraction (ARF)[1] 5.0E-4 Damage Ratio (DR)[2] 0.375
Respirable Fraction (RF)[1] 1.0 Leakpath Factor (LPF) 1.0

Receptor Location			640 m	640 m	640 m	1050 m	1050 m	2350 m	
Stability Class, Wind Speed			D, 4.5m/s	F, 1m/s	95%	D, 4.5m/s	F, 1m/s	95%	
Dispersion (χ/Q)			6.35E-05 s/m³	1.49E-03 s/m³	7.26E-04 s/m³	2.85E-05 s/m³	6.85E-04 s/m³	7.07E-04 s/m³	
Nuclide	Combustible Activity (Ci)	Source Term (Ci)	On-Site Dose (rem)	On-Site Dose (rem)	On-Site Dose (rem)	Off-Site Dose (rem)	Off-Site Dose (rem)	Off-Site Dose (rem)	Percent Dose Contribution
Am-241	5.48E+01	1.03E-02	1.13E-01	2.65E+00	1.29E+00	5.07E-02	1.22E+00	1.26E+00	42.7%
Pu-238	3.89E+01	7.30E-03	7.10E-02	1.67E+00	8.12E-01	3.19E-02	7.66E-01	7.91E-01	26.9%
Pu-241	7.44E+02	1.40E-01	2.95E-02	6.93E-01	3.37E-01	1.32E-02	3.18E-01	3.29E-01	11.2%
Pu-240	1.02E+01	1.91E-03	2.06E-02	4.83E-01	2.35E-01	9.23E-03	2.22E-01	2.29E-01	7.8%
Cm-244	1.19E+01	2.23E-03	1.27E-02	2.99E-01	1.46E-01	5.72E-03	1.37E-01	1.42E-01	4.8%
Pu-239	6.19E+00	1.16E-03	1.25E-02	2.94E-01	1.43E-01	5.62E-03	1.35E-01	1.39E-01	4.7%
Sr-90	8.35E+02	1.57E-01	4.34E-03	1.02E-01	4.96E-02	1.95E-03	4.68E-02	4.83E-02	1.6%
Am-243	2.82E-01	5.29E-05	5.82E-04	1.37E-02	6.65E-03	2.61E-04	6.28E-03	6.48E-03	0.2%
Cs-137	1.18E+03	2.22E-01	1.51E-04	3.55E-03	1.73E-03	6.79E-05	1.63E-03	1.69E-03	0.1%
		TOTAL TEDE	2.64E-01	6.21E+00	3.02E+00	1.19E-01	2.85E+00	2.94E+00	100.0%

Notes:

[1] - Based on Section 5.2.1.1, DOE-HDBK-3010-94.
[2] - Based on ratio of combustible waste volume in LSA #4 to total combustible waste volume in Lag Storage Facilities; values taken from WVNS Waste Management Operations LLW database.

TABLE B.9.2-3

FAILURE OF TANK 8D-2 VAULT

Assumptions:

Airborne Release Rate (ARR)	4.0E-8 hr^{-1}	- see note [1] below	Damage Ratio (DR)	1.0
Respirable Fraction (RF)	1.0	- see note [1] below	Leakpath Factor (LPF)	1.0

Release Duration	24 hr	
Release Height	0 m	(Ground Level Release)

Receptor Location			640 m	640 m	640 m	1050 m	1050 m	2350 m	
Stability Class, Wind Speed			D, 4.5m/s	F, 1m/s	95%	D, 4.5m/s	F, 1m/s	95%	
Dispersion (χ/Q)			6.35E-05 s/m^3	1.49E-03 s/m^3	7.26E-04 s/m^3	2.85E-05 s/m^3	6.85E-04 s/m^3	7.07E-04 s/m^3	
Nuclide	8D-2 Inventory[2] (Ci)	Source Term (Ci)	On-Site Dose (rem)	On-Site Dose (rem)	On-Site Dose (rem)	Off-Site Dose (rem)	Off-Site Dose (rem)	Off-Site Dose (rem)	Percent Dose Contribution
Am-241	5.37E+04	5.16E-02	5.67E-01	1.33E+01	6.48E+00	2.55E-01	6.12E+00	6.31E+00	64.1%
Sr-90	5.79E+06	5.55E+00	1.54E-01	3.60E+00	1.76E+00	6.90E-02	1.66E+00	1.71E+00	17.4%
Pu-238	7.93E+03	7.62E-03	7.41E-02	1.74E+00	8.47E-01	3.32E-02	7.99E-01	8.25E-01	8.4%
Cm-244	6.08E+03	5.84E-03	3.33E-02	7.82E-01	3.81E-01	1.50E-02	3.59E-01	3.71E-01	3.8%
Pu-239	1.63E+03	1.56E-03	1.69E-02	3.96E-01	1.93E-01	7.57E-03	1.82E-01	1.88E-01	1.9%
Pu-240	1.19E+03	1.15E-03	1.24E-02	2.90E-01	1.41E-01	5.55E-03	1.33E-01	1.38E-01	1.4%
Pu-241	6.05E+04	5.80E-02	1.23E-02	2.88E-01	1.40E-01	5.51E-03	1.32E-01	1.37E-01	1.4%
Cs-137	6.32E+06	6.07E+00	4.14E-03	9.71E-02	4.73E-02	1.86E-03	4.47E-02	4.61E-02	0.5%
Am-243	3.47E+02	3.33E-04	3.66E-03	8.59E-02	4.19E-02	1.64E-03	3.95E-02	4.08E-02	0.4%
Am-242m	2.85E+02	2.73E-04	2.95E-03	6.92E-02	3.37E-02	1.32E-03	3.18E-02	3.28E-02	0.3%
Ac-227	9.46E+00	9.08E-06	1.29E-03	3.02E-02	1.47E-02	5.77E-04	1.39E-02	1.43E-02	0.1%
Cm-243	1.16E+02	1.11E-04	8.24E-04	1.93E-02	9.42E-03	3.70E-04	8.89E-03	9.18E-03	0.1%
I-129	1.84E-01	1.84E-01	6.99E-04	1.64E-02	7.99E-03	3.14E-04	7.54E-03	7.78E-03	0.1%
		TOTAL TEDE	8.84E-01	2.07E+01	1.01E+01	3.97E-01	9.54E+00	9.85E+00	99.9%

Notes:
[1] - Based on Section 3.2.4.5 (liquid covered with debris), DOE-HDBK-3010-94.
[2] - Ref: WVNS memo EK:89:0232 (Vit Mass Balance, Rev. 7).

TABLE B.9.2-4

**LOW LEVEL WASTE TREATMENT FACILITY ACCIDENT
EARTHQUAKE INDUCED LAGOON FAILURE**

ISOTOPE [1]	Max. Lagoon 2 Concentration [2] (uCi/mL)	Activity Released [3] (Ci)	Cattaraugus Creek Concentration [4] (Ci/L)	Activity Ingested [5] (Ci)	Dose Conversion Factor [6] (rem/Ci)	CEDE Off-site (rem)
Cs-137	3.23E-03	3.07E+01	2.34E-06	4.68E-06	5.00E+04	2.34E-01
Sr-90	8.82E-04	8.38E+00	6.39E-07	1.28E-06	1.40E+05	1.79E-01
					TOTAL CEDE	4.13E-01

Notes:

[1] - Based on all nuclides expected to be present in Lagoon 2. Nuclides given here represent those that contribute greater than 0.1% of the CEDE.

[2] - Based on interceptor discharge limit of 5E-3 uCi/mL gross beta activity.

[3] - Based on Lagoon 2 capacity of 9,500,000 L.

[4] - Based on Cattaraugus Creek flow rate of 3,620,000 L/hr - WVDP-065.

[5] - Based on 2L water ingested by Maximally Exposed Off-Site Individual (MEOSI).

[6] - Dose conversion factors from WVDP-065.

Table 9.2-5

SUMMARY OF CONSEQUENCES OF IRTS, MAIN PLANT AND SUPPORT FACILITY ACCIDENTS

Accident Scenario	Maximum Off-Site Dose/Dosage	Maximum On-Site Dose/Dosage	Evaluation Guidelines Level
Main Ventilation HEPA Bank Failure	2.7 rem	6.4 rem	On-site - 100 rem
			Off-site - 25 rem
Hydrogen Peroxide Spill	8.75 ppm	17.5 ppm	On-site - ERPG-3 (100 ppm)
			Off-site - ERPG-2 (50 ppm)
Transformer Leak of PCBs	3.1E-3 mg/m^3	6.4E-3 mg/m^3	On-site - TEEL-3 (5 mg/m^3)
			Off-site - TEEL-2 (5 mg/m^3)
8D-2 Tank and Vault Failure	9.9 rem	21 rem	On-site - Natural Phenomena, N/A
			Off-site - 25 rem
Fire in Lag Storage Facility	2.9 rem	6.2 rem	On-site - 100 rem
			Off-site - 25 rem
LLWTS Lagoon 2 Failure	4.1E-1 rem	N/A	On-site - Natural Phenomena, N/A
			Off-site - 25 ppm

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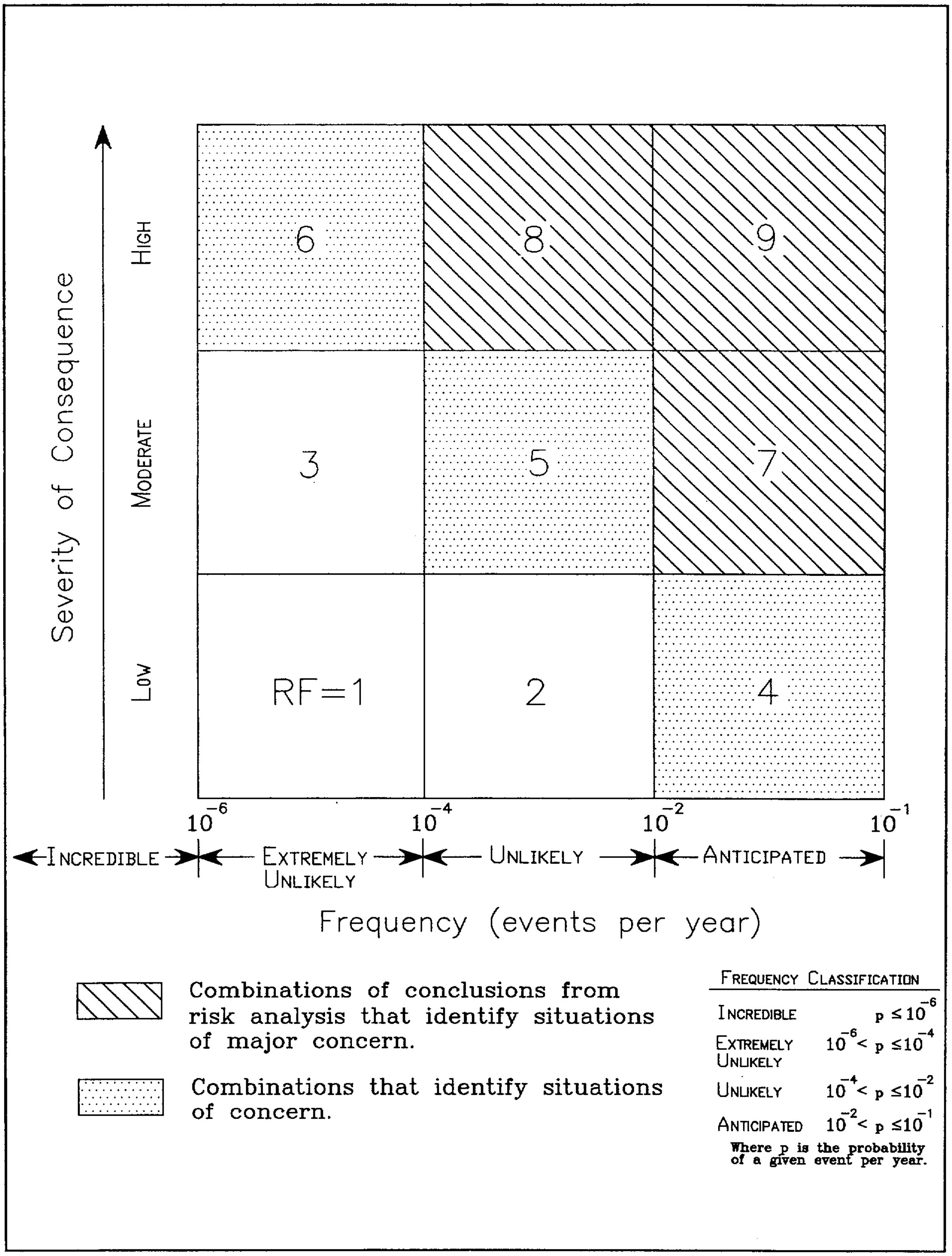

Figure B.9.1-1. Process Hazards Analysis Risk Bins

SR2B91-2.DWG

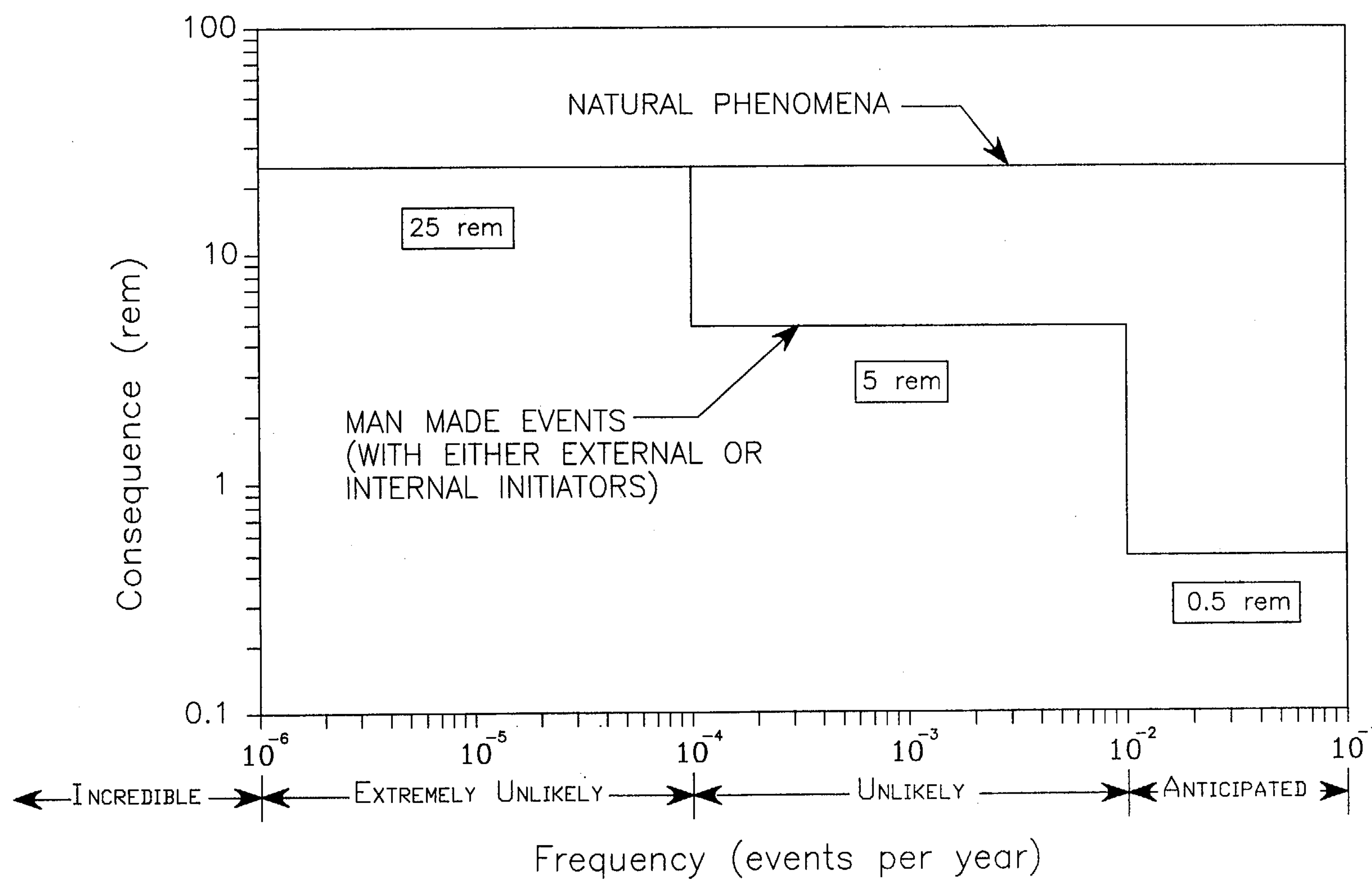

Figure B.9.1-2. Evaluation Guidelines for the Off–site Evaluation Point for Radiological Accidents

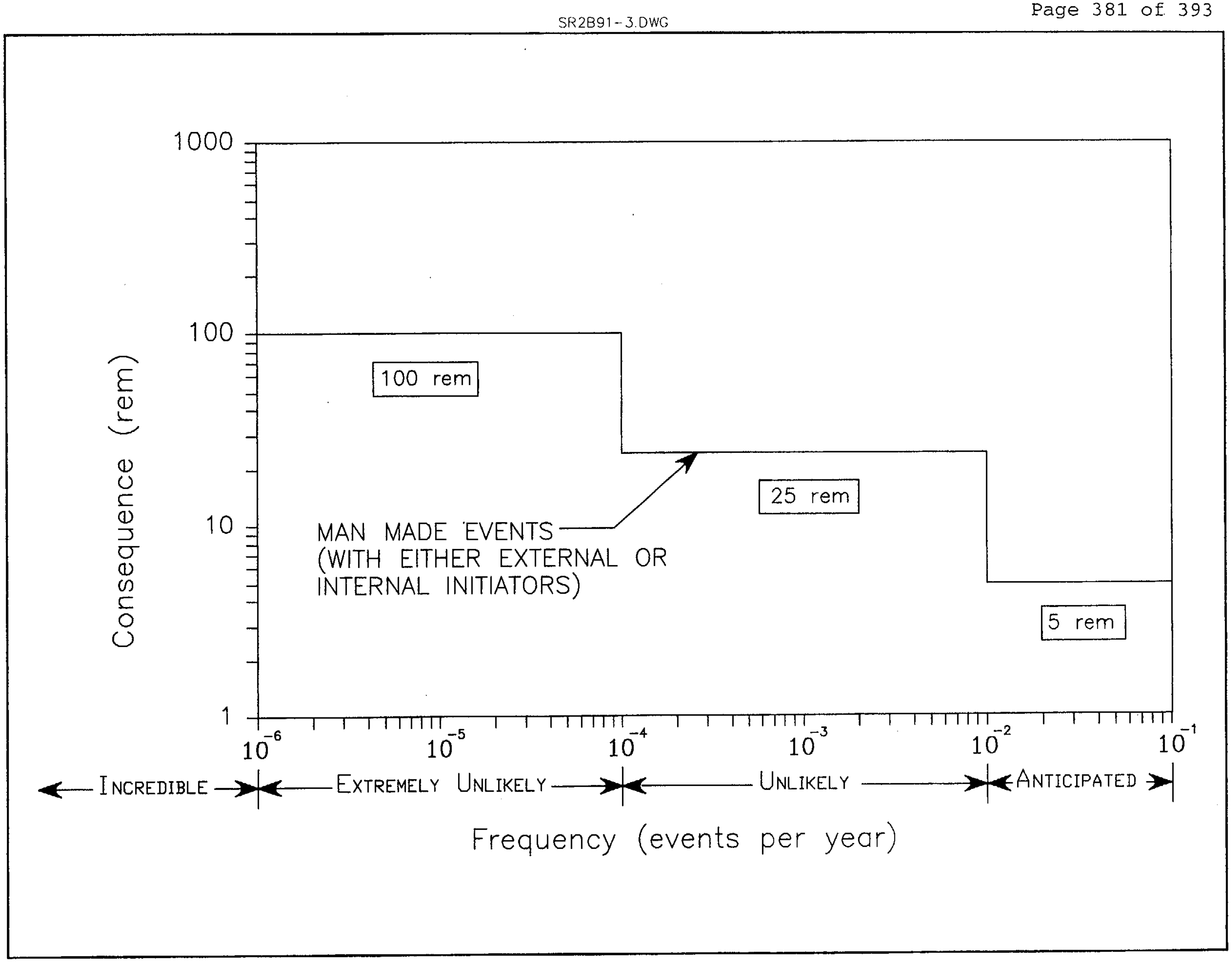

Figure B.9.1-3. Evaluation Guidelines for the On-site Evaluation Point for Radiological Accidents

B.10.0 CONDUCT OF OPERATIONS

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

B.10.1 Management, Organization, and Institutional Safety Provisions

B.10.1.1 Organizational Structure

The major facilities discussed in WVNS-SAR-002 fall under the authority of one of the following major organizational groups at the WVDP: 1) Site Operations & Facility Closure Projects, 2) High-Level Waste (HLW) Projects, or 3) Waste, Fuel & Environmental Projects. Figures B.10.1-1, B.10.1-2, and B.10.1-3 present the structure of these organizations. Site Operations & Facility Closure Projects has responsibility for the operation and engineering of the Main Plant, HLW Projects has responsibility for operation and engineering of most facilities associated with the IRTS, and Waste, Fuel & Environmental Projects has responsibility for waste management services and spent fuel shipping.

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

B.10.1.2 Organizational Responsibilities

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

B.10.1.3 Staffing and Qualifications

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

B.10.1.4 Safety Management Policies and Programs

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

B.10.2 Procedures and Training

B.10.2.1 Procedures

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

B.10.2.2 Training

A description of the WVNS training program is presented in Section A.10.3 of WVNS-SAR-001.

B.10.3 Initial Testing, In-Service Surveillance, and Maintenance

B.10.3.1 Initial Testing Program

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

The first component of the IRTS to become operational was the STS (Supernatent Treatment System), in 1988. This system, as well as each succeeding component of the IRTS, met the requirements of the initial testing (preoperational) program described in Section A.10.2 of WVNS-SAR-001 prior to being declared operational.

B.10.3.2 In-Service Surveillance and Maintenance Program

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

B.10.4 Operational Safety

B.10.4.1 Conduct of Operations

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

B.10.4.2 Fire Protection

The WVDP Fire Protection Program is discussed in Section A.4.3.6 of WVNS-SAR-001.
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B.10.5 Emergency Preparedness Program

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

B.10.6 Decontamination and Decommissioning

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

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

REFERENCES FOR CHAPTER B.10.0

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

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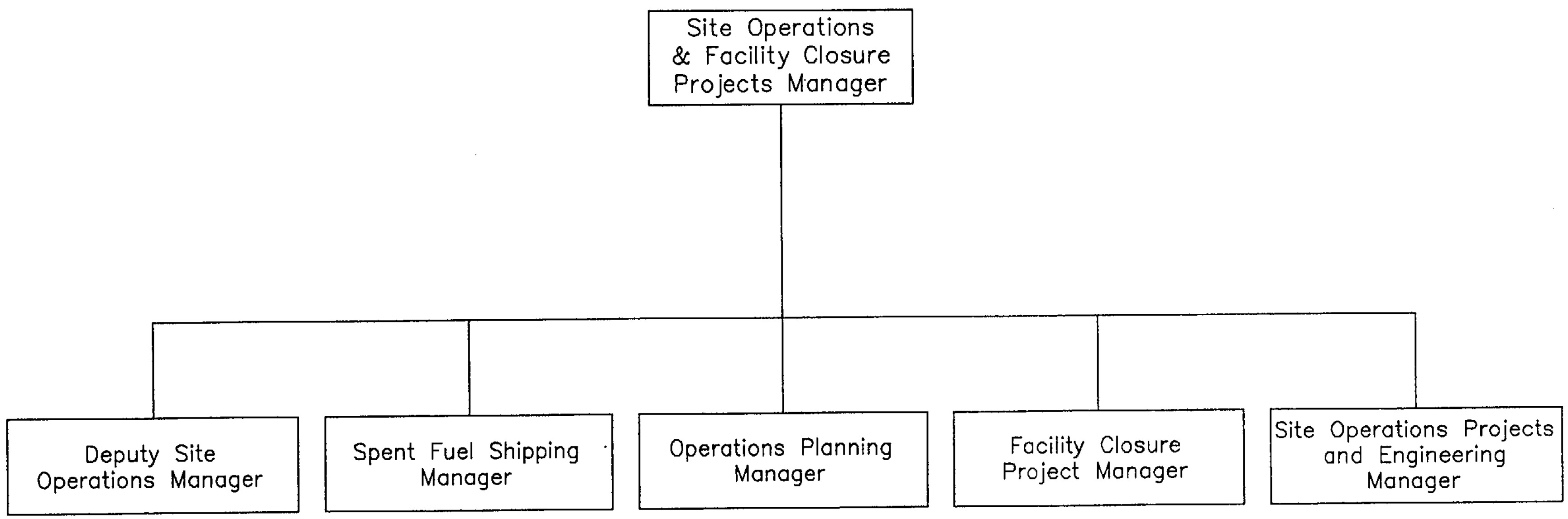

Figure B.10.1-1. Site Operations & Facility Closure Projects Organization

SR2B1012.DWG

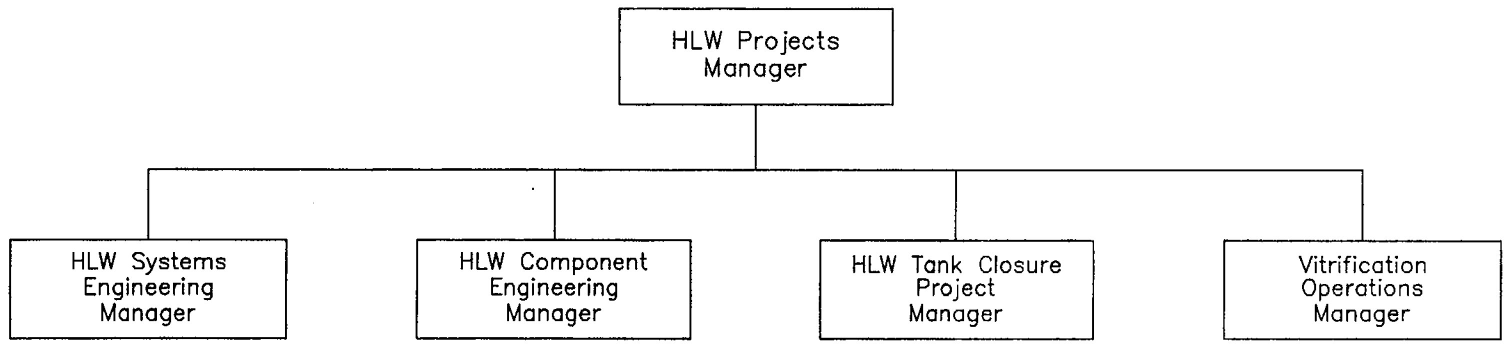

Figure B.10.1-2. HLW Projects Organization

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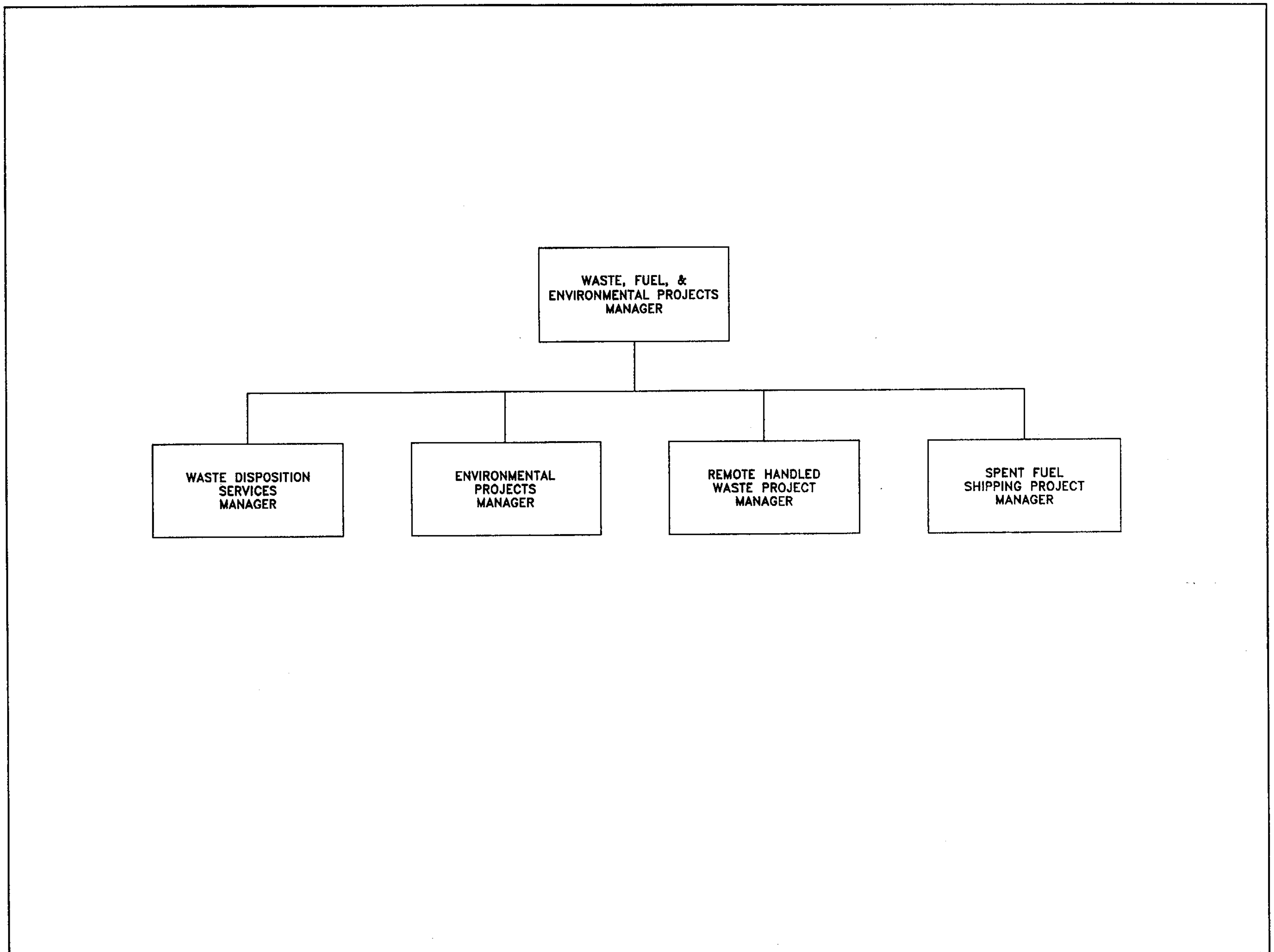

Figure B.10.1-3 Waste, Fuel, & Environmental Projects Organization

B.11.0 TECHNICAL SAFETY REQUIREMENTS

B.11.1 Introduction

The objective of this chapter is to provide information that will satisfy the requirements of DOE Order 5480.23, Topic 16 relating to the derivation of Technical Safety Requirements (TSRs). This chapter is intended to link the accident analyses, through descriptions of the Safety Class structures, systems, and components (SSCs) to TSR documents. The TSR document, as stated in DOE Order 5480.22, is intended to constitute an agreement or contract between DOE and WVNS regarding the safe operation of the WVDP facilities.

Safety Class SSCs are those structures, systems, or components whose preventive and/or mitigative functions are necessary to maintain the consequence of an accident below the off-site evaluation guidelines provided in Section B.9.1.3. Because the accidents analyzed in Chapter B.9.0 do not rely on protective or mitigative features to maintain dose consequences below the evaluation guidelines, no TSRs are required for this SAR.

B.11.2 Requirements

This SAR meets the requirements in DOE Orders 5480.23 and 5480.22, with respect to TSRs.

B.11.3 TSR Input

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

B.11.3.1 Safety Limits and Limiting Conditions for Operation

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

Therefore, no Safety Limits and Limiting Conditions for Operation are required for facilities within the scope of this SAR.

B.11.3.2 Design Features

The primary passive safety features in the IRTS and Main Plant are the high level waste tanks and vaults in the Waste Tank Farm and the massive concrete shield walls of the Main Plant. IRTS, Main Plant and support facility design features are described in Chapters 4, 5, and 6.

B.11.3.3 Administrative Controls

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

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

Occupational exposure to hazardous materials and/or conditions is thus regulated by the Department of Energy through its contractual commitment by WVNS for the safe operation of the WVDP facilities through compliance with these DOE Orders. Additionally, occupational exposure limits are imposed upon WVNS through Federal and State regulations, as well as through provisions in the Occupational Safety and Health Act administered by OSHA. Consequently, no TSR administrative controls for occupational exposure are required for facilities within the scope of this SAR.

B.11.4 Interface With TSRs From Other Facilities

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

REFERENCES FOR CHAPTER B.11.0

U.S. Department of Energy. February 25, 1992. Change 2 (January 23, 1996.) DOE Order 5480.22: *Technical Safety Requirements*. Washington, D.C.: U.S. Department of Energy.

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

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

B.12.0 QUALITY ASSURANCE

The Quality Assurance Program (QAP) at the WVDP is implemented on a site-wide basis and is applied in compliance with the QA Rule, 10 CFR 830.120, *Quality Assurance Requirements*, and DOE O 414.1, *Quality Assurance*. Definition and description of the WVNS QAP is provided by the OH/WVDP-approved WVNS document WVDP-111, *Quality Assurance Program* (WVNS), which includes the WVNS QA Program Implementation Plan which, in turn, includes the 10 CFR 830.120 implementation matrix identifying appropriate and applicable requirements of the WVNS QAP and other WVNS procedures to be used in determining compliance with the QA Rule.

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

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

REFERENCES FOR CHAPTER B.12.0

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

West Valley Nuclear Services Co., Inc. WVNS Implementation Plan for QA Rule 10 CFR 830.120

__________. WVDP-111: Quality Assurance Program Plan. (Latest Revision.)

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

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

WVNS RECORD OF REVISION

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	12/23/95
3	General Revision per ECN #6026 and ECN #9495 Incorporated into WVNS-SAR-002 Rev. 3, are the following documents: WVNS-SAR-004, WVNS-SAR-005, WVNS-SAR-006, WVNS-SAR-007, WVNS-SAR-008, WVNS-SAR-009, WVNS-SAR-010, WVNS-SAR-013 and WVDP-096	All	01/16/96
4	ECN #10541 Represents an annual update of WVNS-SAR-002 as required by DOE Order 5480.23	All	02/05/97
4	Per ECN #10635 - Cancellation of WVNS-SAR-002, Addendum 1, Rev. 3, "Safety Analysis Report for the Fuel Receiving & Storage Facility". Issuance of WVNS-SAR-012, Rev. 0, will provide the authorization basis required for operation of the Fuel Receiving & Storage Facility. (formerly contained in WVNS-SAR-002, Addendum 1)	Addendum 1	05/29/97
5	ECN #11223 represents an annual update of WVNS-SAR-002 as required by DOE Order 5480.23. Incorporates and updates material from WVNS-ASA-001, Rev. 1, and WVNS-SAR-022, Rev. 0	All	02/27/98

WVNS RECORD OF REVISION CONTINUATION FORM

Rev. No.	Description of Changes	Revision On Page(s)	Dated
6	ECN #11801 represents an annual update of WVNS-SAR-002 as required by DOE Order 5480.23. Incorporates and updates material from the LLW2, the NDA, the VTF, and the HEC. Removed reference to the Supercompactor. Added information and conclusions from various FHAs, including information on lightning protection.	All	12/30/98
7	Annual update of WVNS-SAR-002 as required by DOE Order 5480.23. Double contingency analyses documented, and TRU waste storage in LSA 1, 3, and 4 covered. Figures and associated text updated to reflect current operating practice. Scenario of a hydrogen peroxide spill reanalyzed, and organization charts updated. Added description of Permeable Treatment Wall. Discussion of tank corrosion updated and the estimated frequency of a potential tank leak due to corrosion has been modified. System descriptions updated as appropriate, including electrical and compressed air.	All	
8	Annual update of WVNS-SAR-002 as required by 10 CFR 830.202.(c).(2). Figures and associated text updated to reflect current operating practice. System descriptions updated as appropriate, including electrical and compressed air. Added Addendum 4 to this revision. DOE approval contained in Letter DW:2002:0029, dated 01/22/02. Per ECN #13341	All	03/01/02

ADDENDUM 4

West Valley Demonstration Project

Doc. ID Number WVNS-SAR-002, Addendum 4

Revision Number 0

Revision Date 03/01/2002

Engineering Release #4650

Controlled Copy No. 6

SAFETY ANALYSIS REPORT ADDENDUM FOR

HEAD END CELL DECONTAMINATION AND WASTE PACKAGING

DAMES & MOORE FOR
WVNS ENVIRONMENT, SAFETY, QUALITY ASSURANCE AND LABORATORY OPERATIONS

JUNE 30, 2000

APPROVED HAZARD CATEGORIZATION: 2

DATE OF APPROVAL: September 24, 1993 REFERENCED DOCUMENT: WD:93:1167

APPROVALS: Lettie J Chilson 8/24/00
L. J. Chilson, Manager
Safety Analysis and Integration

Ellery D. Savage 8-25-00
E. D. Savage, Manager
Environment, Safety, Quality Assurance and Laboratory Operations

8/28/00
J. L. Little, Chairman
WVNS Radiation and Safety Committee

Joseph C. Wolniewicz 8/28/00
Independent Reviewer, Date
(Not required unless SAR contains new criticality analyses)

Westinghouse
Government Services Group

West Valley Nuclear Services Co.
10282 Rock Springs Road
West Valley, NY 14171-9799

SAR:0007378.01

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LIST OF ACRONYMS AND ABBREVIATIONS

ALARA	As Low As Reasonably Achievable
ARF	Airborne Release Fraction
CCTV	Closed Circuit Television
DOE	Department of Energy
ft	Feet
GCR	General Purpose Cell Crane Room
GCRX	GPC Crane Room Extension
GOA	GPC Operating Aisle
GPC	General Purpose Cell
HEC	Head End Cells
HEPA	High Efficiency Particulate Air
HEV	Head End Ventilation
kg	Kilogram
lb	Pound
LLW3	Low-Level Waste Waste Treatment Replacement Facility
LWTS	Liquid Waste Treatment System
m	Meter
MCNP	Monte Carlo N-Particle
mrem	Millirem
mSv	milli-Sievert
NCSE	Nuclear Criticality Safety Evaluation
NFS	Nuclear Fuel Services, Inc.
NRC	Nuclear Regulatory Commission
PHA	Process Hazards Analysis
PMC	Process Mechanical Cell
PMCR	Process Mechanical Cell Crane Room
RFs	Respirable Fraction
SAR	Safety Analysis Report
SRR	Scrap Removal Room
Sv	Sievert
TEDE	Total Effective Dose Equivalent
TSR	Technical Safety Requirement
WHC	Westinghouse Hanford Corporation
WVDP	West Valley Demonstration Project

B.1.0 INTRODUCTION AND GENERAL DESCRIPTION

This Safety Analysis Report (SAR) Addendum addresses the collection, processing (e.g., drying, decontamination, and size reducing), packaging, and storage of debris located in the Head End Cells (HECs) of the Main Plant at the West Valley Demonstration Project (WVDP). Some of the debris contains fissile material. This SAR Addendum constitutes a supplement to WVNS-SAR-002, *Safety Analysis Report for Low-Level Waste Processing and Support Activities*, as defined in WV-365, *Preparation of WVDP Safety Documents*. Table B.1.6-1 of WVNS-SAR-002 shows the correlation between the 12 sections of this SAR Addendum and the SAR topics presented in DOE Order 5480.23, *Nuclear Safety Analysis Reports*. The graded approach to the development of safety analysis documentation was used to ensure that the extent of descriptive information and accident analysis is commensurate with the risk associated with decontamination activities discussed in this SAR Addendum. The activities discussed in this SAR Addendum have very little potential for adverse effects to human health or the environment.

The HECs, which include the Process Mechanical Cell (PMC) and its crane room, the General Purpose Cell (GPC) and its crane room, and the Scrap Removal Room (SRR), were used by Nuclear Fuel Services Incorporated (NFS) to mechanically process and handle irradiated nuclear fuel assemblies. As a result of routine NFS operations a significant amount of high-activity, fissile-bearing debris accumulated in these cells and rooms, the majority of which is in the PMC and the GPC. This debris includes fine particulate material such as fuel and saw fines that were generated during the mechanical processing of the fuel assemblies and the fuel elements; fuel assembly hardware; small equipment used to support operations; and leached and potentially unleached fuel hulls. Additional debris, principally laboratory equipment and other laboratory waste, was introduced into the PMC by the WVDP during cleanup activities in the Sample Storage Cell in the 1980s. After packaging, fissile-bearing debris from the HECs is to be stored in one or more of the at-grade or above-grade cells or rooms within the Main Plant until a disposal facility becomes available; however, the option to store the fissile-bearing debris at-grade or above-grade in a facility other than the Main Plant is also available. The benefit in collecting, packaging, and storing this debris is that the HECs will be placed in a safer and more stable condition that can be economically monitored and maintained until final closure activities begin in the Main Plant.

As stated in WVDP-227, *WVDP Facility Identification and Classification Matrix*, the Main Plant is a Hazard Category 2 nuclear facility. The HECs are part of the Main Plant, and, therefore, activities described in this SAR Addendum are encompassed by the hazard categorization for the Main Plant.

B.2.0 SUMMARY SAFETY ANALYSIS

The process hazards analysis (PHA) for the activities discussed in this SAR Addendum is presented in Table B.9.1-1 of this document. The PHA identified only one event as being of sufficient risk to warrant additional analysis. The event is a fire in the HECs, in particular the GPC, where a significant inventory of transuranic materials are expected to be present. A fire in the GPC that affects much or all of the radioactive debris in the GPC is extremely unlikely; nevertheless, such a fire was selected for consequence assessment purposes. The maximum total effective dose equivalent (TEDE) at the on-site evaluation point (640 m from the facility) was calculated to be 6.09 rem (0.0609 Sv), while the maximum TEDE received by an off-site individual (1050 m from the facility) was calculated to be 2.51 rem (0.0251 Sv). These doses are below the Evaluation Guidelines presented in Section B.9.1.3 of WVNS-SAR-002 for this extremely unlikely event.

Nuclear Criticality Safety Evaluation for the Handling and Storage of Fissile-bearing Debris in the Head End Cells provides the criticality and contingency analyses that have been performed to support the safe handling, processing, and storage of fissile waste materials located in the HECs. *Nuclear Criticality Safety Evaluation for the Handling and Storage of Fissile-bearing Debris in the Head End Cells* specifies design features and administratively controlled limits and requirements to reduce the likelihood of a criticality event during the handling, processing, and storage of HEC fissile-bearing debris. This evaluation concludes, in Section 8.0, that "storage area(s), and areas where handling and processing activities will occur, will remain subcritical under all normal and credible abnormal and accident conditions."

B.3.0 SITE CHARACTERISTICS

Site characteristics of the WVDP are provided in WVNS-SAR-001, *Project Overview and General Information.* Geological and seismological information pertinent to the Main Plant can be found in Section B.3.6 of WVNS-SAR-002.

B.4.0 PRINCIPAL DESIGN CRITERIA

The Main Plant was constructed in 1964 by Nuclear Fuel Services in accordance with NRC license CFS-1 and various criteria in effect at the time, as documented in a U.S. Atomic Energy Commission-approved Final Safety Analysis Report (FSAR) (Nuclear Fuel Services, 1970). The Head End Cells of the Main Plant were constructed to these criteria. Significant additions or modifications to the facility are required to comply with DOE Order 420.1 and the associated editions of the references therein. Crane room enclosures required for access to the Process Mechanical Cell Crane Room (PMCR) and General Purpose Cell Crane Room Extension (GCRX) have been fabricated in accordance with this criteria. No other structures or major process systems are being added to the Main Plant to accomplish the activities described in this SAR Addendum.

B.4.1 Purpose of the Head End Cells

The Head End Cells were used by NFS to mechanically process and handle irradiated nuclear fuel assemblies. This SAR Addendum addresses the decontamination of the HEC areas; consequently, original process equipment will not be relied upon to serve its original process function. Cell equipment that is required to support decontamination such as cranes, manipulators, and shield windows have been replaced or refurbished. A discussion of the materials to be handled in this area can be found in Section B.8.7-6 of WVNS-SAR-002, which provides estimates of fissile material located in the PMC and GPC.

B.4.2 Structural and Mechanical Safety Criteria

Section B.4.2 of WVNS-SAR-002 discusses the structural and mechanical safety criteria for the Main Plant.

B.4.3 Safety Protection Systems

Discussions of protection through defense-in-depth, protection by equipment and instrumentation selection, nuclear criticality safety, radiological protection, fire and explosion protection, radioactive waste handling and storage, and industrial and chemical safety, as provided in Sections B.4.3.2 through B.4.3.8, respectively, of WVNS-SAR-002, are applicable for activities addressed in this SAR Addendum.

B.4.4 Classification of Systems, Structures, and Components

The classification (i.e., safety class and quality level) of equipment and components used to cleanup the HECs is as described in Section B.4.4 of WVNS-SAR-002. (The "safety class" noted here is not related to "safety class" as used in DOE-STD-3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*. The analyses presented in this SAR Addendum do not result in the need for safety-class structures, systems, or components as defined in DOE-STD-3009-94.)

B.5.0 FACILITY DESIGN

B.5.1 Summary Description

A summary description of the WVDP is provided in Section B.5.1 of WVNS-SAR-002.

B.5.2 Head End Cell Description

The Main Plant is described in Section B.5.2 of WVNS-SAR-002. Section B.5.2.4.9 of WVNS-SAR-002 provides a detailed description of HEC facilities, which are illustrated in Figures 1 and 2 of this SAR Addendum. The following describes modifications to the facility required to accomplish cell decontamination.

A Process Mechanical Crane Room (PMCR) enclosure and GPC Crane Room Extension (GCRX) enclosure are provided to support the replacement of cranes and manipulators located in the PMC and GPC, and for use as a maintenance area for the newly installed bridge-mounted manipulator in the PMC and GPC, respectively. The PMCR enclosure is a steel-framed structure built on the roofs of the PMCR and the adjoining Master Slave Manipulator Repair Shop. The enclosure was installed to accommodate the removal of the two bridge cranes and bridge-mounted power manipulator located in the PMC, and to allow the installation of the replacement bridge-mounted power manipulator in the PMC. The part of the enclosure built on the PMCR roof is located over the rolling hatch cover contained in the roof slab of the PMCR. This hatch cover is removed as necessary to allow access into the PMCR. The structural steel framework of the enclosure is tied into the existing structural steel in the Main Plant. The enclosure is equipped with a bridge crane. The enclosure is ventilated by airflow from the North Operating Aisle through the Change Room exiting near the rolling hatch.

The GCRX enclosure is a steel-framed structure built over the roof hatches of the GCRX. These hatch covers are removed as necessary to allow access into the GCRX. The enclosure was installed to facilitate the removal of the bridge crane and the bridge-mounted power manipulator in the GPC, and to allow the installation of the replacement bridge-mounted power manipulator in the GPC.

The two bridge cranes and the bridge-mounted power manipulator in the PMC have been removed and replaced with a single bridge equipped with an auxiliary two-ton capacity chain hoist and a bridge-mounted, remote-controlled power manipulator. The bridge crane and the bridge-mounted power manipulator in the GPC have also been removed and replaced with a single bridge equipped with an auxiliary two-ton capacity chain hoist and a bridge-mounted, remote-controlled power manipulator. The bridges, hoists, and power manipulators are nearly identical in both the PMC and GPC. However, the width of the bridge in the GPC is different from the width of the bridge in the PMC. The auxiliary chain hoists are mounted on a separate I-beam attached to the bridge assemblies and both have a 1800-kg (4000-lb) lifting capacity. The chain hoist in the PMC also has a 15-m (50-ft) travel range and lift capacity allowing it vertical access from the bridge rails in the PMC, through the PMC/GPC hatch, and down to the floor of the GPC.

Several of the manipulators in the PMC and GPC have been recently removed and replaced to support debris collection and packaging activities. Manipulators used in the PMC and GPC are similar to equipment originally installed in both of these cells.

B.5.3 <u>Support Systems</u>

Main Plant and IRTS support systems are described in Section B.5.3 of WVNS-SAR-002.

B.5.3.1.9 Head End Cell Fire Protection

Addendum 1 to WVNS-FHA-011, *Fire Hazard Analysis Main Process Plant*, provides the fire hazards analyses necessary to support the activities discussed in this SAR Addendum. Addendum 1 to WVNS-FHA-011 states that there "is no fixed fire suppression system coverage of the HECs," and recommends under the "requirements" portion of the document that Class A, B, and D fire suppression agent (e.g., FEM-12) must be placed in the HECs so as to be available for manual application on a fire or around a fire (using remotely controlled equipment such as a bridge-mounted power manipulator). Addendum 1 to WVNS-FHA-011 contains several other recommendations under the "improvements" portion of the document. "Improvements" are those items that "recognize an industrial standard or a best practice, improve protection and risk associated with proposed activities, and further safety comparable to the identified hazards." Recommendations under the "improvements" portion of the document are noted below.

- Ordinary combustible materials should not be introduced into or allowed to accumulate in the HECs.

- Ordinary combustible materials should not be discarded within the same containers used for the collection of zirconium chips or fines.
- Adequate separation should exist between hot (thermal) work operations and other in-cell activities.
- Equipment used to collect, size reduce, decontaminate, package and store the loose fissile-bearing debris in-cell should be noncombustible.
- Equipment with hydraulic systems should use a less hazardous (i.e., noncombustible) hydraulic oil. Nonflammable coolants or lubricants should be used for wet grinding, cutting, or sawing operations.
- Wetted fines should be dried at a temperature not exceeding 230 deg F.
- Existing or new fire detection is recommended. Existing fire detectors have been deactivated.

B.5.4 Description of Service and Utility Systems

Service and utility system descriptions, including descriptions of the Main Plant and Head End Cell Ventilation systems, are provided in Section B.5.4 of WVNS-SAR-002.

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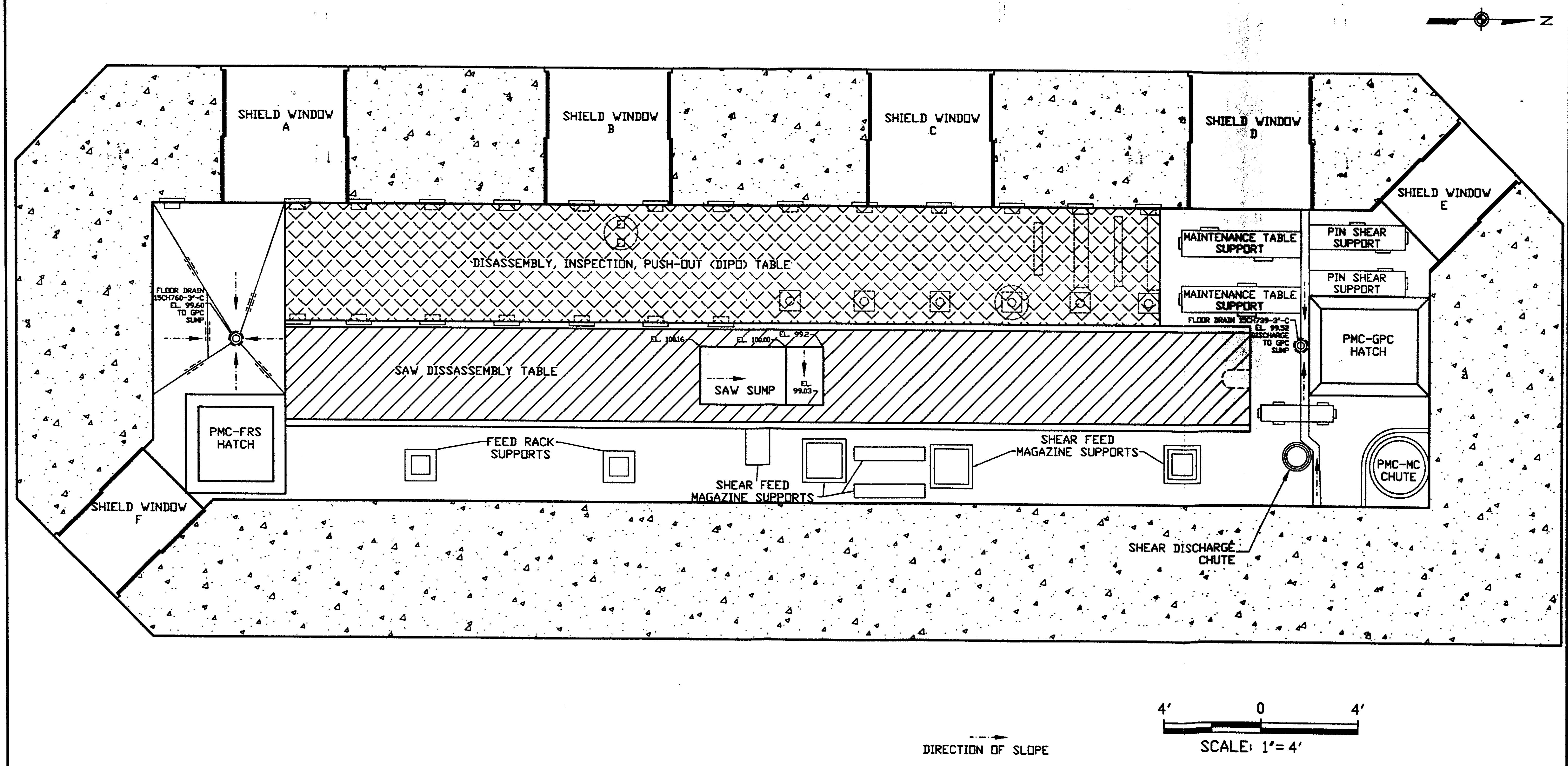

Figure 1 Process Mechanical Cell Plan View

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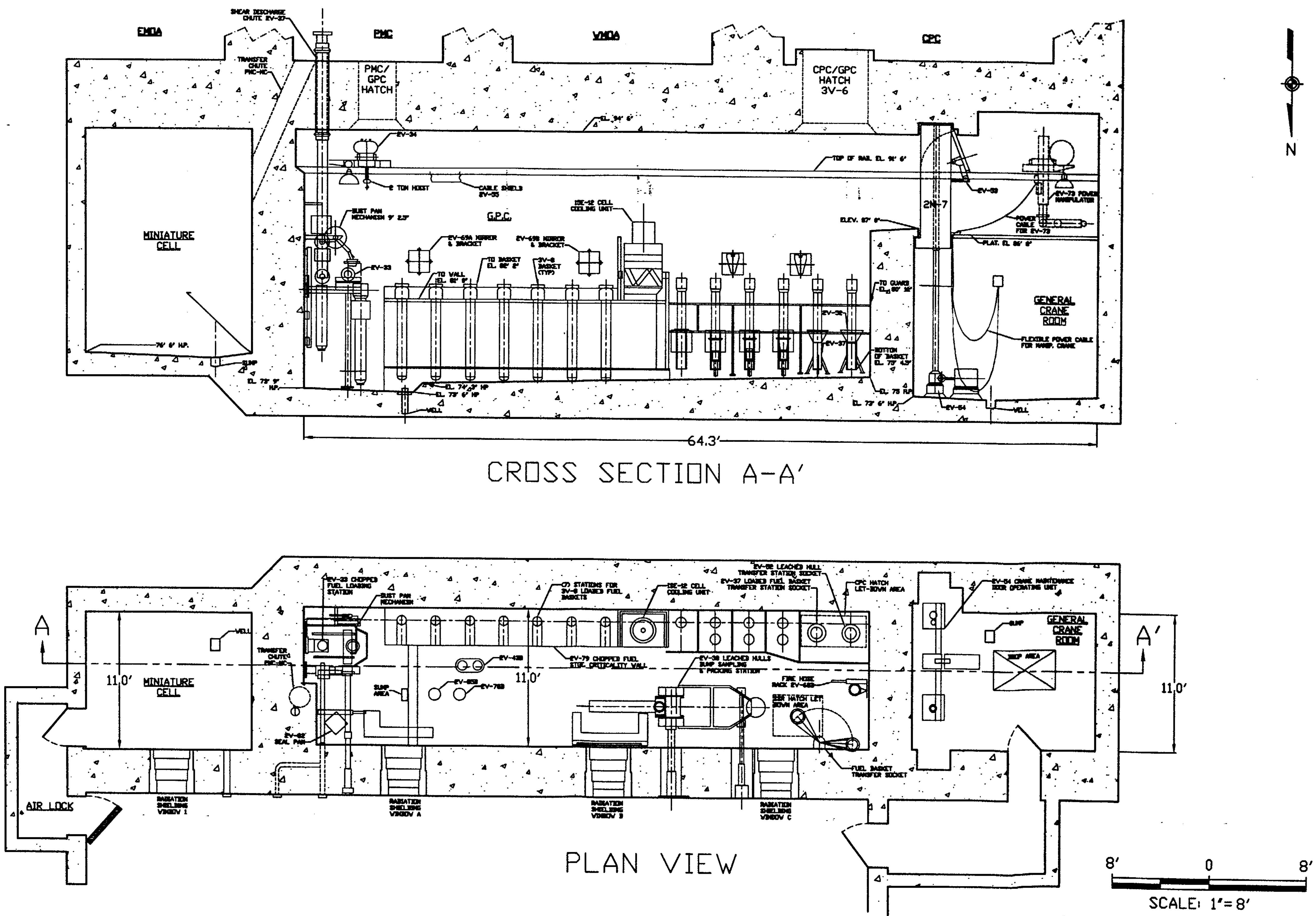

Figure 2 General Purpose Cell Section and Plan View

B.6.0 PROCESS SYSTEMS

The HECs contain a heterogeneous mixture of debris ranging from large objects such as drums and shear gags to fine particulate material such as fuel and saw fines. A variety of methods and tooling may be employed to collect, size reduce, decontaminate, dry, package, and place this debris into safe storage. An overview of the operations required to accomplish these activities is provided in Section 6.1. Equipment required to accomplish these activities are described in Section 6.2.

B.6.1 Overview of Operations

This SAR Addendum addresses the operations required to decontaminate the highly contaminated Head End Cells of the Main Plant. These cells contain a variety of contaminated bulk materials such as drums, drum lids, lifting equipment, and hoses; activated equipment and materials such as fuel assembly hardware and saw fines; and fissile-bearing materials such as fuel hulls and fuel fines. Equipment in the cells that will be necessary to support decontamination activities include manipulators (both power and manual), cranes, manipulator and crane tooling (end effectors), and collection equipment such as scoops, shovels, or vacuums.

Some of the larger debris in the PMC and GPC may need to be size-reduced to allow this material to be packaged. Size reduction of this larger debris may be done in-place at the point of collection or the debris may be transferred to an in-cell size reduction station. Depending on the type of material to be size reduced, mechanical or thermal size reduction equipment may be used.

Cutting operations introduce the potential for an in-cell fire or explosion due to the generation of sparks during certain sawing operations or due to the generation of sparks or hot slag during torch cutting. WVNS-FHA-011, "Fire Hazard Analysis Addendum - Main Process Plant (Head End Cell)," acknowledges that these activities present the potential for a fire incident or related event in-cell and recommends minimizing the potential of this risk by eliminating the combustibles in cell prior to size reduction and by establishing a "hot work area" in the cell in which size reduction activities would be performed. Cutting operations that employ hydraulic equipment (such as nibblers or shears) or saws that use toothed-blades (versus abrasive discs) do not present an increased potential for a fire in the cell.

Mechanical size reduction methods may include the use of electrical-, pneumatic-, or hydraulic-powered shears and nibblers and common industrial saws such as reciprocating saws, chop saws, guillotine saws, power hacksaws, and portable band saws. Thermal methods using plasma-arc or laser technologies may also be used to size reduce material that cannot be reduced by mechanical methods. Since methods such as plasma-arc generate a large amount of airborne particulate material, a source capture system will be installed in the PMC or GPC to prevent premature fouling and change-out of the Head End Ventilation (HEV) System filters if this equipment is used. An enclosure designed to house the size reduction equipment and to capture any particulates released during size reduction activities may also be employed if warranted by the size reduction technology selected. The specific type of size reduction equipment used will depend on the composition of the debris to be size reduced. For example, combustible debris and pyrophoric material such as Zircaloy-clad fuel will not be size-reduced using thermal cutting methods or by mechanical methods that are capable of emitting sparks and starting a fire.

Bulk contaminated debris in the PMC and GPC may require decontamination to meet waste management or ALARA objectives. This equipment will be decontaminated as necessary using technologies that have been previously deployed for equipment or cell decontamination at the WVDP such as high pressure water, steam, detergent, and CO_2.

Mechanical or water-based decontamination methods have been evaluated for their effectiveness in removing surface contamination from larger debris and equipment in-cell. Mechanical methods include CO_2 pellet or compressed air systems that decontaminate debris by mechanically dislodging removable surface contamination. Dislodged contamination is then collected and packaged. Water-based methods including water washes, steam cleaning, and water-based detergent or caustic washes may be used to remove surface contamination from the surfaces of contaminated equipment or debris. Liquid decontamination will be performed in a stall or booth that will isolate the decontamination solution from areas containing fissile-bearing debris to minimize the potential for material moderation. Should a liquid decontamination system be used, a liquid collection, transfer, and treatment system will be installed to manage the spent decontamination liquids.

Particulate debris in the HECs includes fuel hulls and fine particulate material consisting primarily of saw fines and sheared fuel particles. Fine debris may contain fissile material and may be collected using shovel- or clam-shell-type scoops or a vacuum system tethered and deployed in-cell by a manual manipulator or the bridge-mounted power manipulator. Clam-shell or shovel-type scoops similar to those that were previously employed by NFS for this activity in the HECs may be used to collect debris on the HEC floors and to place it into containers. Fine debris may also be collected with a vacuum system that includes a hose, a particulate debris collection container, and a fine particulate filter. The bulk of the vacuumed debris is collected in the particulate debris collection container. Particulate material passing through the vacuum exhaust is collected in the fine particulate filter before the exhaust air is discharged into the HEC.

The fine debris from the GPC collected in the vacuum's particulate debris collection container may contain fissile material and water. This debris will be dried before it is placed into storage to reduce the potential for inadvertent criticality. Fissile-bearing debris collected in the GPC is dried using a specially-designed drying unit that is heated with electrical or steam heating elements. Once dried, the particulate debris collection containers are placed in a close-packed array within a 114-liter (30-gallon) storage drum to maximize the use of the available storage volume. Drums containing dry uranic and transuranic debris are vented with specially-designed vents that release hydrogen and other gases generated through radioactive processes, and also prevent the entry of water that would serve as a moderator.

Bulk contaminated debris collected from the HECs will be packaged into standard waste containers (i.e., drums or boxes); however, fissile-bearing debris must be packaged into 114-liter (30-gallon) carbon steel drums. After packaging, fissile-bearing debris from the HECs is stored in one or more of the at-grade or above-grade cells or rooms within the Main Plant until a disposal facility becomes available; however, the option to store the fissile-bearing debris at-grade or above-grade in a facility other than the Main Plant is also available.

Once the loose debris is removed from a given HEC, the interior surfaces of the cell may be decontaminated using mechanical decontamination methods such as vacuum, high-pressure water wash, and CO_2 methods. Spent wash water will be collected and transferred out of the HEC for management in the Low-Level Waste Treatment Replacement Facility (LLW2), Liquid Waste Treatment System (LWTS), Vitrification facility, or a vendor-supplied wastewater treatment unit. Significant water incursion into the PMC has not been observed and it has been assumed that minimal moderator exists in the PMC due to the presence of floor drains and hatches that would allow water that does enter the cell to drain to the GPC. Activities in the

PMC involving the use of liquid decontamination agents prior to the collection and packaging of small fissile-bearing debris is prohibited.

B.6.2 Equipment Descriptions

The power manipulators in both the PMC and GPC are an electrically powered PaR Systems Model #3000. The manipulators are used to perform a variety of operations. The manipulators lift objects and, when equipped with a variety of tools and end effectors, cut, grind, and shear objects as necessary. The manipulators are capable of rotating and pivoting at the shoulder, elbow, and wrist joints, and are able to grasp objects with the manipulator hands. A telescoping tube assembly connects the manipulators to the bridge cranes allowing the manipulators to be maneuvered in three-dimensional space. Both telescoping tube assemblies have a lifting capacity of 450 kg (1000 lb) plus the weight of the manipulator. The manipulators are remotely controlled from a portable control console located outside of the PMC in the mechanical operating aisles and outside the GPC in the GPC Operating Aisle (GOA). The manipulators are equipped with a 115-volt AC/DC electrical receptacle that supplies power to small power tools up to 0.5 horsepower.

Both bridge cranes are equipped with two high-resolution, radiation-hardened, closed circuit television (CCTV) cameras with an integral pan-and-tilt system. The radiation hardened cameras are able to tolerate a minimum absorbed dose of 10^6 Gray (10^8 rad). The cameras are able to rotate up to 355 degrees and are able to tilt 90 degrees down from the horizontal and 45 degrees up from the horizontal.

The power manipulator and the manual manipulators are equipped with mechanical hands that allow these pieces of equipment to grasp larger objects and place them into storage containers. The manipulators may also be equipped with grapples and clam-shell scoops to pick up large debris. Smaller debris in the HECs is collected using a clam-shell or shovel type scoop or a vacuum system which is tethered and deployed in-cell by a manual manipulator or the bridge-mounted power manipulator.

A typical vacuum system is one with a dual-drum design that includes a pre-separator drum that acts as a knock-out pot, a particulate collection suction drum, and an electric power head containing up to three vacuum motors. A hose connected to the vacuum drums is used to pick up fine debris from the HECs. The fine particulate material entrained in the vacuum exhaust flow passing through the collection container is captured in a HEPA filter before the vacuum exhaust is discharged into a given HEC.

Larger debris in the HECs such as hoses, drums, drum lids, and lifting equipment may need to be size reduced before they are packaged in containers. Size reduction is performed either in-place or after transferring debris to a designated size reduction location. A variety of size reduction equipment utilizing mechanical or thermal methods may be used to size reduce larger debris. The type of equipment to be used depends on the type of material being size reduced. Large debris within a cell will be size reduced through the use of mechanical or thermal equipment that will be remotely operated via a power manipulator or manual manipulators. Mechanical size reduction equipment includes saws, power shears, and nibblers. This equipment may be either electrically, hydraulically, or pneumatically actuated.

Thermal methods such as plasma-arc or laser technologies may also be used to size reduce debris in the HECs. A plasma-arc torch uses a high-velocity stream of high-temperature ionized gas (plasma) to cut through metals by melting and forcefully displacing the molten metal. The plasma is produced by sending a compressed gas past an electrode that generates a high voltage that ionizes the gas stream. The electrical conductivity of the plasma causes the electric arc to transfer to the metal being cut. A secondary stream of non-ionized gas is expelled around the plasma stream to shield the cut. If used, the plasma-arc torch, including the torch electrode and nozzle, will be located in-cell in a mobile enclosure containing a source capture system to capture airborne particulate material generated during cutting operations. The plasma-arc torch will be deployed by the power manipulator or a manual manipulator.

Mechanical or water-based decontamination methods may be used to dislodge removable surface contamination from equipment or large debris. Mechanical methods may use a CO_2 or compressed air system to decontaminate debris. CO_2 decontamination uses CO_2 pellets (dry ice) as the cleaning medium which will be propelled by compressed air to dislodge removable surface contamination from equipment or debris. The radioactive debris removed by this method can be collected and packaged. There are no secondary wastes produced using CO_2 decontamination as the CO_2 sublimates. A compressed air decontamination system may be used to mechanically remove the surface contamination from the external surfaces of contaminated debris by forcefully dislodging the removable contamination. If such decontamination systems are used, a source capture system will be installed to prevent premature fouling of the HEV System exhaust filters. The filtered air stream exiting the source capture system will be exhausted (in an upward direction) to the cell.

Water washes, steam cleaning, and chemical decontamination methods using water-based detergents or caustic may be used to strip removable surface contamination from the external surfaces of contaminated equipment or debris. Such treatment will be located outside of areas containing fissile-bearing debris because of criticality concerns. If a water-based decontamination system is used, a liquid collection, transfer, and treatment system will be installed to manage the spent decontamination liquids. Water-based decontamination will be performed within an enclosure equipped with a liquid collection and transfer system. Steam and water used to decontaminate debris will be supplied by the existing steam and utility water supply lines. Detergents and caustic will be stored outside of the cells and will be transferred into the HECs via utility lines and piped to wands and spray heads in the decontamination enclosure.

Fine debris in the HECs are collected in the vacuum system's debris collection containers. Since debris collected from the GPC may potentially contain fissile material and water, the debris will be dried before it is placed into final storage. Fissile material in debris collection containers will be dried using a pail or drum heater that uses a jacketed heating system with electrical or steam heating elements. It is anticipated that the monitoring of moisture removal from collection containers during drying operations will be accomplished through the use of load cells which will indicate the dryer/container weight throughout the drying process, although a comparable technique could be used provided the technique demonstrates that water has been removed from the containers.

Dry fissile-bearing debris collected from the HECs will be stored in 114-liter (30-gallon) carbon steel drums that are approximately 46.0 cm (18.1 in) in diameter and 72.4 cm (28.5 in) in height (outside). The dimensions of the drums allow them to be placed in a shielded overpack, if necessary, to allow for transport to a storage area outside the Main Plant.

Drums containing fissile-bearing debris are vented to release hydrogen and other gases that originate through radiation-related processes. Each of the 114-liter (30-gallon) drums containing fissile-bearing debris is vented with a HEPA filter that can withstand at least 121.9 cm (48 in) water column without allowing water entry into a container (such as the Nuclear Filter Technology Incorporated NucFil 013 filter with Gore-Tex). A limit on the number and size of fissile-bearing debris containers that may be present in various cell locations has been stipulated in *Nuclear Criticality Safety Evaluation for the Handling and Storage of Fissile-bearing Debris in the Head End Cells.* Criticality design and administrative controls for these components are discussed in Section B.8.7 of this SAR Addendum.

B.7.0 WASTE CONFINEMENT AND MANAGEMENT

The information on waste confinement and management provided in Chapter B.7.0 of WVNS-SAR-002 is applicable to the activities discussed in this SAR Addendum. Waste containers (i.e., drums or boxes) used for storage of HEC debris will be temporarily stored at the WVDP and will ultimately be disposed of at a permanent off-site disposal facility following characterization. Equipment required for collection, drying, and size reduction will become contaminated and may be decontaminated and utilized elsewhere at the WVDP, or may be disposed of as low-level waste. The total volume of such expended equipment is not expected to be large due to the scope of work involved and the nature of the equipment being used to clean up the HECs.

B.8.0 HAZARDS PROTECTION

B.8.1 Assuring that Occupational Hazards Exposures are ALARA

Policies, programs, and design considerations related to maintaining hazards exposures ALARA during HEC operations are addressed by the policies, programs, and design considerations discussed in Section B.8.1 of WVNS-SAR-002 for the Main Plant.

B.8.2 Sources of Hazards

B.8.2.1 Contained Sources

The PMC and GPC contain a significant amount of high activity wastes. Nuclides in these wastes include fission products, activation products, transuranics, and other long-lived alpha emitters. A partial characterization of the wastes in these cells was performed in 1986 (Vance, 1986). Based on this characterization, an estimate of residual contamination in the PMC and GPC was prepared (Wolniewicz, 1993). Residual activity estimates for the PMC and GPC, which are contained in Table B.8.2-3 of this SAR Addendum, were calculated by inferring the activity of Cs-137 in the area based on measured exposure rates. The proportion of other nuclides in the waste relative to Cs-137 was calculated based on the type of fuel processed in the cell and an average fuel irradiation. Based on the activity estimates in Wolniewicz, 1993, an estimate of the mass of residual fissile material in the PMC and GPC has also been calculated. These inventories are documented in Table B.8.7-6 of WVNS-SAR-002.

B.8.2.2 Airborne Hazards Sources

The Head End Cells contain a significant amount of loose particulate contamination generated during original NFS operations in these areas. Decontamination activities in the HECs will inevitably mobilize this contamination; however, the cell ventilation system, which is described in Section B.5.4.1.1.3 of WVNS-SAR-002, is expected to maintain control of any mobilized contamination.

B.8.3 Hazard Protection Design Features

B.8.3.1 Radiation Protection Design Features

General radiation protection design features for the Main Plant are discussed in Section B.8.3.1 of WVNS-SAR-002.

B.8.3.1.2 Shielding

The design basis for shielding in the Main Plant is given in Section B.8.3.1.2 of WVNS-SAR-002. Currently, the exposure rates in the operating areas of the Main Plant range from 0.1 to 5.0 mR/hour, with most areas having an exposure rate of less than 1 mR/hr. Modifications to the Head End Cells required to support cell decontamination (such as shield window replacement) are designed such that the exposure rate in occupied areas of the Main Plant will not be increased.

B.8.4 Estimated Collective On-site Dose Assessment

The collective on-site dose for the combined Main Plant/IRTS work groups is given in Section B.8.4 of WVNS-SAR-002 and summarized in Table B.8.4-1 of the SAR. Decontamination activities in the HEC will necessitate manned entry to HEC support areas such as crane maintenance rooms; however, all work in the cells themselves will be conducted remotely due to the significant in-cell radiation levels. These levels, though, are significantly below the facility's original shielding design basis and all material in these areas has undergone significant (approximately 30 year) post-reactor decay. Based on engineering judgment, the average annual exposure per worker will increase slightly for some work groups because of the performance of activities discussed in this SAR Addendum, but will remain far below the administrative dose limit at the WVDP of 500 mrem (5 mSv) per year for a given worker. In general, the dose rates in the operating aisles around the GPC and PMC are very small (i.e., around 1 mrem/hr [0.01 mSv/hr]).

B.8.5 WVDP Hazards Protection Programs

WVDP hazards protection programs are described in Section B.8.5 of WVNS-SAR-002.

B.8.6 Estimated Collective Off-Site Dose Assessment

Section B.8.6 of WVNS-SAR-002 addresses the estimated collective off-site dose assessment. Based on engineering judgment, activities discussed in this SAR Addendum are not expected to contribute significantly to the existing environmental radiological source term.

B.8.7 Prevention of Inadvertent Criticality

B.8.7.3 Criticality Concerns

Decontamination of the GPC and PMC presents criticality concerns during the collection, processing, and storage of fissile-bearing materials. In the GPC, potentially moderated fissile-bearing wastes must be collected and packaged. This activity alters the configuration of the material from that currently evaluated in the SAR and potentially results in a more reactive configuration than currently exists in the slab configuration in which the material now lies. Packaged waste will be relocated to a dryer station and upon drying will be removed to a storage array.

Collection, packaging, processing, and storage activities necessary to decontaminate the HECs require that containers of fissile-bearing material be brought in proximity to each other. These activities present the potential for inadvertent criticality under certain conditions of moderation and spacing when the containers are filled with fuel-bearing materials having an enrichment of 5 w/o U-235. Little validated characterization data for the debris in the HEC exists; consequently, conservative assumptions were made in the analyses to evaluate the reactivity of decontamination operations. These assumptions are:

- The fuel in the debris is unirradiated;
- The fuel has a U-235 enrichment of 5 w/o (97.5% of the fuel reprocessed by NFS had effective enrichments of less than 3.3 w/o);
- Fuel hulls and fine particulate debris are assumed to be unreprocessed fuel;
- Arrays of fissile-bearing debris storage containers are assumed to be in an infinite two-dimensional hexagonal array;
- Fissile-bearing debris inside collection and storage containers is assumed to be arranged in an optimum configuration (i.e., hexagonal-pitch array).

Analyses to determine the reactivity of various normal and accident conditions are documented in the *Nuclear Criticality Safety Evaluation for the Handling and Storage of Fissile-Bearing Debris in the Head End Cells* (hereafter referred to as the NCSE [nuclear criticality safety evaluation]). Calculations referenced in the analysis were performed using the personal computer (PC) version of MCNP 4A, which was run on a Compaq Pentium III, 550 MHz system, operating in a Windows NT environment. MCNP 4A Software Validation Plan and Report for Criticality Calculations provides documentation of the validation activities that were performed to support the criticality calculations addressed in this NCSE. The bias and bias uncertainty related to the use of MCNP for evaluation of moderated and unmoderated systems containing low-enriched uranium are addressed in MCNP 4A Software Validation Plan and Report for Criticality Calculations. Through an evaluation of the code bias, a subcritical limit of 0.93 was selected for use in this NCSE in assessing the criticality safety of normal operations.

The NCSE explicitly evaluates the criticality safety of each of four conditions: collection/packaging and storage under normal conditions and collection/packaging and storage under abnormal and accident conditions. These conditions are further discussed below.

B.8.7.3.1 <u>Normal Operations</u>

Normal collection, packaging, and storage operations have been evaluated to determine the criticality safety of these operations and have been found to be subcritical by a wide margin. A summary of the calculated k-eff for normal operations conditions is provided in Table B.8.7-1 of this Addendum. The following describes the conditions anticipated during normal operations.

Collection

Under normal collection conditions small particulate debris will be collected with a clam shell scoop, vacuum, or similar device which will be selected to minimize the potential for creating a more geometrically favorable arrangement of fissile material. It is expected that the sump area of the GPC will contain little or no water throughout the decontamination of the cell. Although historical evidence of water infiltration into the area exists, current video of cell conditions indicates that the cell appears to be dry and the cell sump level indicator registers zero water level in the sump. Consequently, it is expected that under normal collection and packaging operations, little water will be transferred to a collection container.

Packaging

Containers used for the collection of fissile material will be located in areas that reduce the potential for interaction of collected material with other fissile material accumulations. The interaction of material in the collection container with accumulations of fissile-bearing debris in the GPC sump, in containers of dried waste in storage arrays, or in waste containers in the drum dryer will be minimized by locating the collection container in an area away from other accumulations. During material collection in the GPC sump the collection container will be located in an area outside the sump region; the presence of a collection container in the GPC sump during material collection is considered an abnormal condition. The interaction of the collection container with the small accumulations of debris scattered on the cell floor or surfaces is not a criticality concern because the configuration of this material (low density slab) is such that the material does not have the potential to significantly affect the reactivity of the system.

Storage

Fissile-bearing debris collected in the HECs will be stored in 114-liter (30-gallon) carbon steel drums. Any moisture or free liquid will be removed from the drum to the greatest extent practicable during drying operations prior to placement of the drum in a storage array. In addition, storage locations for drum arrays are not expected to be susceptible to flooding. Even the GPC, which has experienced periods of water accumulation due to roof leaks, is not susceptible to water intrusion due to surface flooding. Therefore, under normal conditions, there will be minimal internal and external moderation of waste package in the storage array.

B.8.7.3.2 <u>Abnormal Operations and Accidents</u>

Abnormal operations and accidents associated with the collection, handling, and storage of fissile-bearing materials in the HEC have been evaluated to determine the margin of criticality safety during these operations. Evaluated conditions are discussed below.

Collection

Accidents and abnormal operations that could occur during the collection and packaging of fissile-bearing particulate debris include dropping collected material onto debris in the sump area of the GPC, collection of a significant amount of water into a storage container, and collection of material into a waste container located in the sump area of the GPC.

The condition in which fissile-bearing debris in the GPC sump is rearranged into a hemispherical pile represents the bounding case for a reconfiguration of fissile-bearing debris in the HECs. It is assumed that moderator exists in the sump up to an elevation of 74 ft 3 in. This elevation corresponds to the point at which the floor of the GPC intersects the west side of the sump. Water rising above this elevation would extend onto the floor of the GPC, and no mechanism for this amount of water infiltration into the cell has been identified. The calculated keff + 2σ for the condition in which fuel is relatively densely packed is 0.70, which is less than the upper subcritical limit of 0.93. This system remains safe even when up to one third of the volume of the sump is occupied with water, with the remainder occupied by uniformly arranged evaluation basis fuel.

Handling

Accidents and abnormal operations that could occur during the handling of fissile debris containers in the HECs include placing two drums containing moderated fissile-bearing material in proximity to each other and placing multiple drums of moderated fissile material in proximity to a storage array. The NCSE has evaluated the condition in which two drums of moderated fissile material are placed in proximity to each other and concludes that this system will remain subcritical, even when the fuel in the drum is configured to optimize reactivity and over a third of the volume of the drum is water.

The NCSE also evaluates the condition in which two drums of moderated fissile material are brought into proximity with an array of dried fissile debris storage containers. The size of the array, which corresponds to a number of containers that is nearly an order of magnitude greater than has been estimated as being necessary to package all of the small particulate debris in the GPC, was selected to represent the maximum number of waste containers that could reasonably be stored in the GPC. The analysis assumes that fuel is optimally arranged and that the waste containers are bounded on all sides and on the bottom by concrete of sufficient thickness to appear infinite. Based on these assumptions it has been found that the keff + 2σ for the system is 0.92, which is below the upper subcritical limit of 0.93.

Storage

Arrays of HEC fissile debris storage containers will typically be located in above grade storage locations where the potential for flooding is incredible; however, containers of dried fissile debris wastes may also be staged in the GPC prior to transfer to an above grade storage location. Abnormal operations and accidents important to criticality safety that may involve an array of fissile debris storage containers include external moderation of the storage array (with or without subsequent internal moderation of the containers in the array), inadvertent placement of undried (i.e., potentially moderated) waste packages in proximity to the storage array, and breaching of containers in the array due to fire or falling as a result of a seismic event or due to striking.

The external moderation of an array of waste drums containing dried fissile materials has been evaluated in the NCSE. Significant external moderation of a storage array is only a concern in the GPC, where the possibility of flooding of the cell is extremely unlikely. In other locations where HEC fissile debris may potentially be stored, full flooding of the array is incredible. Consequently, arrays of storage containers in the GPC are limited to one tier, which has been shown to be safe under all conditions of external moderation, whereas the height of an array of HEC fissile debris outside of the GPC is limited to three tiers.

The internal moderation of wastes packages contributes significantly to the reactivity of a storage array. Normal operations will minimize the quantity of water in a storage drum through evaporation. These wastes are not expected to be rewetted following processing, although the potential exists due to the fact that drums containing high concentrations of alpha contamination must be vented to prevent the accumulation of explosive gases, which may be generated through processes such as the radiolytic decomposition of water. Drums containing dried HEC fissile-bearing debris will be ventilated with a HEPA filter that can withstand at least 120 cm (48 in) water column without allowing water entry into a container (Nuclear Filter Technology Incorporated NucFil 013 filter with Gore-Tex, or equivalent) to minimize the potential for introducing water into a sealed, dried waste drum.

Water moderator in an unprocessed collection container may also affect the reactivity of a dry, unmoderated storage array if a collection container containing a sufficient amount of water is brought in proximity to the array. Although it is not anticipated that unprocessed wastes will be placed in the vicinity of the storage array because of pragmatic operational constraints, there are no physical barriers or administrative controls that would prevent a drum of moderated waste material from being placed near an array of dried drums. Analyses referenced in the NCSE indicate that even when two drums of moderated fissile debris wastes are placed adjacent to a 7 drum x 7 drum x 3 tier array of dried waste containers, the $k_{eff} + 2\sigma$ is 0.92, which is less than the selected upper subcritical limit of 0.93.

Finally, breaching a container, either through increased internal pressure as a result of a fire or through rupture due to falling, presents the potential for moderation of a previously unmoderated accumulation of fissile material. The potential for a fire in the HEC is minimal due to the low amount of combustible material in the area. Furthermore, activities in the HEC that present the potential for increased fire risk, such as cutting operations, will be conducted in engineered enclosures that contain hot sparks. Fire suppression systems that will be employed in the HECs will not have sufficient volume or pressure to disturb a drum in its storage location. Dislocation of a drum from the storage array due to a seismic event or impact with a handling device such as a PaR or manipulator arm is a credible abnormal operation or accident event due to the fact that there will be no mechanical restraint around the storage area to prevent such an occurrence. However, debris piles formed from breaching a container in such a way is only a criticality concern in those areas where moderator exists. As discussed earlier, this is only the case in the GPC, and as stated, the storage of drums in the GPC is limited to single-tier arrays.

A summary of results for all evaluated abnormal and accident conditions is given in Table B.8.7-1 of this Addendum.

B.8.7.4 Criticality Controls

B.8.7.4.1 Design Features

Decontamination operations in the Head End Cells will rely on engineered features to reduce the likelihood of a criticality event during the handling, processing, and storage of HEC fissile-bearing debris. Equipment requirements specified in the NCSE for HEC decontamination activities are summarized below.

- Only 114-liter (30-gallon) carbon steel drums, 72.4 cm + 1 cm (28.5 in + 0.4 in) high (outside) with an inside diameter of 46.0 cm + 1 cm (18.1 in + 0.4 in), shall be used for the storage of HEC fissile-bearing debris. (The selected tolerance of 1 cm (0.4 in) corresponds to a variation in geometry having an insignificant affect on reactivity for a waste package or array of packages.)

- The volume of an HEC fissile material debris collection or processing container shall not exceed the dimensions of an HEC storage container specified above.

- Each 114-liter (30-gallon) HEC fissile-bearing debris storage drum shall be vented with a HEPA filter that can withstand at least 122 cm (48 in) water column without allowing water entry into a container (Nuclear Filter Technology Incorporated NucFil 013 filter with Gore-Tex, or equivalent).

Criticality analyses documented in *Nuclear Criticality Safety Evaluation for the Handling and Storage of Fissile-Bearing Debris in the Head End Cells* have only evaluated the criticality safety of fissile-bearing debris which is contained in 114-liter (30-gallon) carbon steel drums. No parametric analyses have been performed for container size; therefore, storage is only permitted in containers having the dimensions prescribed above. Storage in containers having these dimensions ensures a maximum fissile content per drum (based on process knowledge of fissile-bearing materials in the cell) and ensures a routine geometry in the storage array. collection and processing of fissile-bearing materials can be performed using containers with smaller dimensions than those indicated; however, these containers must be over-packed in 114-liter (30-gallon) carbon steel drums for storage.

The safety of the fissile material array is assured only when the presence of moderator in the storage containers is minimized. This is accomplished initially through the drying of collection containers to remove liquid to the greatest extent practicable, and is assured subsequently through the use of water resistant HEPA filters which prevent the infiltration of water into a sealed drum even in the event that the drum is subjected to a significant head of water.

B.8.7.4.2 Administrative Controls

Administrative controls will be required for operations involving the collection, packaging, and storage of HEC fissile-debris wastes to ensure that the potential for inadvertent criticality is minimized. The following administrative controls for HEC decontamination operations are identified in the NCSE.

- No more than two containers of moderated (i.e., unprocessed) fissile material may exist in any HEC at a given time.
- Free liquid in HEC fissile-bearing debris storage containers shall be removed to the greatest extent practicable and in no case shall exceed 5 v/o water.
- Drums in the GPC containing HEC fissile-bearing debris shall not be stacked more than one tier high.
- Drums outside the GPC containing HEC fissile-bearing debris shall not be stacked more than three tiers high.

As indicated previously, the safety of a large array of drums containing HEC fissile-bearing debris is assured only when minimal moderator exists in the drums. Water is the only significant moderator that has been identified in the HECs. Therefore minimizing the amount of water in the fissile-bearing debris storage containers will significantly reduce the potential for inadvertent criticality.

The probability of a criticality in the HEC, as indicated in Table B.9.1-1 of this Addendum, has been determined based on the likelihood of concurrent failure of several barriers that are in place to prevent the occurrence of a criticality event. Moderation of dried fissile-bearing materials in a large storage array represents one of the greatest risks for criticality. Although several steps will be taken to limit the potential for moderator in a storage drum, the risk of inadvertent criticality will be further minimized through administrative controls that specify a limit on the size of storage arrays that may be located in below grade areas.

B.8.7.4.3 Application of Double Contingency

A contingency is "a possible but unlikely change in a condition/control important to the nuclear criticality safety of a fissionable material operation that would, if it occurred, reduce the number of barriers (either administrative or physical) that are intended to prevent an accidental nuclear criticality" (DOE-STD-3007-93). The following contingencies exist for the storage of 114-liter (30-gallon) carbon steel drums containing fissile-bearing debris from the HECs.

Fissile Mass - There would have to be much more fissile mass than that estimated to be present in the HECs (i.e., estimates of the fissile mass in the HECs would have to be in extreme error).

Moderation - At any one time, no more than two drums of HEC fissile-bearing debris shall contain a significant quantity (greater than 5 v/o) of water.

Spacing - HEC fissile debris waste shall be stored only in 114-liter (30-gallon) carbon steel drums and shall be collected in containers having dimensions not exceeding those of a storage container.

The double contingency analysis contained in Table 1 of the NCSE evaluates these contingencies and assesses their integrity under abnormal operation and accident conditions. This evaluation concluded that all contingencies are sufficiently protected that failure of any one would be an extremely unlikely event.

DOE O 420.1, Section 4.3, states that: "An exception to the application of double contingency, where single contingency operations are permissible, is presented in paragraph 5.1 of ANSI/ANS-8.10-1983. This exception applies to operations with shielding and confinement (e.g., hot cells or other shielded facilities)." ANSI/ANS-8.10-1983, Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement, states that "The number of contingencies may be reduced to unity where the principles of this standard are met and the probability of failure of any primary control is shown to be low." Although the HECs meet the criteria for single contingency due to the significant degree of personnel protection afforded by the substantial shield walls of the cells, several contingencies beyond the minimum required are in place to protect against the consequences of an inadvertent criticality.

Table B.8.2-3

Estimate of Activity in the Head End Cells (1993)

PMC Activity		GPC Activity	
Isotope	(Ci)	Isotope	(Ci)
C-14	2.43E-02	C-14	8.23E-02
Fe-55	5.83E-02	Fe-55	1.97E-01
Ni-59	1.66E-02	Ni-59	5.61E-02
Co-60	4.34E+00	Co-60	1.47E+01
Ni-63	1.63E+00	Ni-63	5.50E+00
Se-79	7.18E-03	Se-79	2.43E-02
Sr-90	1.13E+03	Sr-90	3.82E+03
Y-90	1.13E+03	Y-90	3.82E+03
Zr-93	4.38E-02	Zr-93	1.48E-01
Nb-93m	3.43E-02	Nb-93m	1.16E-01
Tc-99	3.02E-01	Tc-99	1.02E+00
Ru-106	3.87E-04	Ru-106	1.31E-03
Rh-106	3.87E-04	Rh-106	1.31E-03
Pd-107	2.36E-04	Pd-107	7.99E-04
Sb-125	1.92E-01	Sb-125	6.51E-01
Te-125m	4.69E-02	Te-125m	1.59E-01
Sn-126	7.73E-03	Sn-126	2.62E-02
Sb-126m	7.73E-03	Sb-126m	2.62E-02
Sb-126	3.09E-03	Sb-126	1.05E-02
Cs-134	4.66E-01	Cs-134	1.57E+00
Cs-135	2.87E-02	Cs-135	9.70E-02
Cs-137	1.23E+03	Cs-137	4.16E+03
Ba-137m	1.16E+03	Ba-137m	3.94E+03
Ce-144	1.20E-05	Ce-144	4.06E-05
Pr-144	1.19E-05	Pr-144	4.01E-05
Pm-147	1.16E+01	Pm-147	3.93E+01
Sm-151	3.65E+01	Sm-151	1.23E+02
Eu-152	6.14E-02	Eu-152	2.08E-01

Table B.8.2-3

Estimate of Activity in the Head End Cells (1993)
(Concluded)

PMC Activity		GPC Activity	
Isotope	(Ci)	Isotope	(Ci)
Eu-154	1.44E+01	Eu-154	4.86E+01
Eu-155	1.75E+00	Eu-155	5.92E+00
U-232	2.96E-01	U-232	1.00E+00
U-233	4.18E-01	U-233	1.42E+00
U-234	1.99E-01	U-234	6.73E-01
U-235	4.42E-03	U-235	1.50E-02
U-236	1.31E-02	U-236	4.44E-02
U-238	3.72E-02	U-238	1.26E-01
Np-237	4.71E-03	Np-237	1.59E-02
Np-239	8.90E-01	Np-239	3.01E+00
Pu-238	6.74E+01	Pu-238	2.28E+02
Pu-239	1.74E+01	Pu-239	5.87E+01
Pu-240	1.32E+01	Pu-240	4.45E+01
Pu-241	6.51E+02	Pu-241	2.20E+03
Pu-242	1.68E-02	Pu-242	5.67E-02
Am-241	1.94E+01	Am-241	6.57E+01
Am-242	1.47E-01	Am-242	4.98E-01
Am-242m	1.48E-01	Am-242m	5.00E-01
Am-243	8.90E-01	Am-243	3.01E+00
Cm-242	1.22E-01	Cm-242	4.12E-01
Cm-243	4.80E-03	Cm-243	1.62E-02
Cm-244	2.83E+00	Cm-244	9.56E+00
Cm-245	4.31E-04	Cm-245	1.46E-03
Cm-246	6.82E-05	Cm-246	2.31E-04
TOTAL	5.50E+03 Ci	TOTAL	1.86E+04 Ci

Ref: Wolniewicz, 1993

Table B.8.7-1

Summary of Criticality Analysis Results for Normal and Accident Conditions

ARRAY	MODERATION INTERNAL	MODERATION EXTERNAL	U-235 w/o ENRICHMENT	$K_{eff}(K_{\infty})+2\sigma$	REFERENCE CALCULATION
NORMAL OPERATIONS					
COLLECTION & HANDLING					
2 x 1	10 v/o water	Dry	5	0.5627	BUF-2000-067
STORAGE					
∞ x ∞ x 1 (GPC)	Dry	Dry	5	0.6630	BUF-2000-052
∞ x ∞ x 3	Dry	Dry	5	0.7107	BUF-2000-052
∞ x ∞ x 3	5 v/o water	Dry	5	0.8700	BUF-2000-059
ABNORMAL OPERATIONS & ACCIDENT CONDITIONS					
COLLECTION & HANDLING					
Hemispherical Pile	33 v/o water in lower 9"	Dry	5	0.93	BUF-2001-021
2 x 1 next to 7 x 7 x 3	33 v/o water in 1 x 1	Dry	5	0.9187	BUF-2001-019
2 x 1	~40 v/o water (w/p=0.7)	Dry	5	<0.93	BUF-2000-067
2 x 1	50 water v/o (w/p=1)	Dry	3.327	0.9063	BUF-2000-067
STORAGE					
∞ x ∞ x 1 (GPC)	Dry	Full	5	0.9149	BUF-2000-052
2 x 1 next to 7 x 7 x 3	33 v/o water in 1 x 1	Dry	5	0.9187	BUF-2001-019

B.9.0 HAZARDS AND ACCIDENT ANALYSES

B.9.1 Hazard Analysis

The systematic analysis of hazards associated with the cleanup of the HECs has been accomplished in this SAR Addendum through the completion of a PHA. The PHA is intended to provide a qualitative analysis of the potential hazards and protective and mitigative features associated with cleaning up the HECs. Information gained through this analysis is then used in selecting accidents to be further analyzed in a more rigorous manner.

The purpose of accomplishing a PHA is to identify the hazards in terms of quantity, form, location, potential initiating events, and other events which could result in an undesirable consequence. A graded approach was used in the development of the PHA shown in Table B.9.1-1 of this Addendum. Information used in developing the PHA was obtained from several sources, including the *Head End Cell Material Handling and Processing Plan*, WVNS-SAR-002, and miscellaneous site documents referenced therein.

Debris identified in association with cleanup of the HECs may contain varying amounts of hazardous materials. Hazardous metals such as mercury exist in minor amounts, while uranium, plutonium, and lead exist in larger quantities. Though the toxicological hazard posed by these metals is evaluated in the PHA, the form and inventory of these metals preclude any significant on-site and off-site consequences associated with the postulated accident scenarios. For this reason, accident analyses evaluate radiological consequences only.

Evaluation of hazards for the PHA required the qualitative assessment of event consequences and frequencies. Qualitative consequence and frequency classifications used in Table B.9.1-1 of this Addendum are as follows.

Qualitative Consequence Classification:

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

Qualitative Annual Frequency Classification:

Anticipated ($10^{-1} \geq p > 10^{-2}$) Incidents that may occur several times during the lifetime of the facility.

Unlikely ($10^{-2} \geq p > 10^{-4}$) Accidents that are not anticipated to occur during the lifetime of the facility.

Extremely Unlikely ($10^{-4} \geq p > 10^{-6}$) Accidents that will probably not occur during the life cycle of the facility.

Incredible ($10^{-6} \geq p$) Accidents that are not credible.

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

For each event in Table B.9.1-1 of this Addendum, a risk factor has been developed that is based on the consequence and frequency for the event. The value of the risk factor is determined from a three-by-three frequency and consequence-ranking matrix, shown in Figure B.9.1-1 of WVNS-SAR-002. Events with negligible consequences were assigned a risk factor of zero (0). Events having either an on-site or an off-site consequence but with a frequency of occurrence less than 1E-6 per year (i.e., incredible events) were assigned a risk factor of "I".

While worker hazards protection is provided by engineered facility features, the most significant facility worker-safety feature, namely, cell shield walls, is passive in nature. Another significant worker-safety facility design feature is the HEC ventilation exhaust system which minimizes the discharges of radioactive material to the environment. Therefore, the primary operational worker-safety features identified in the hazards analysis are administrative controls. Specifically, worker protection from radiological hazards is controlled through the requirements of the *WVDP Radiological Controls Manual* (WVDP-010), while worker protection from non-radiological hazards is controlled through the requirements of the *WVNS Industrial Hygiene and Safety Manual* (WVDP-011).

B.9.2 Accident Analysis

The identification of accidents presenting the greatest risk to on-site individuals and the off-site public is one of the primary goals of the PHA. Evaluation of accident risk was based on a qualitative judgment of accident consequence and likelihood. Accidents selected were those identified in the PHA as having a risk factor greater than or equal to three. The only accident with a risk factor of three or greater is a fire that engulfs much or all of a cell or room. A fire in the GPC was evaluated because the GPC is expected to contain more hazardous materials than the other HECs combined.

The Evaluation Guidelines presented in Section B.9.1.3 of WVNS-SAR-002 are applicable for evaluating the consequences from a fire in the GPC. The accident analysis methodology discussion presented in Section B.9.2.1 of WVNS-SAR-002, and the subsections to Section B.9.2.1, are applicable to the analysis of a fire in the GPC.

B.9.2.2.5 Fire in the General Purpose Cell

B.9.2.2.5.1 Scenario Development

A fire in the GPC is postulated to occur while conducting cleanup activities. Consistent with the development of an unmitigated accident scenario, no credit is taken for HEPA filtration of airborne releases or for fire suppression efforts. The fire is assumed to consume combustible materials present in the GPC and thermally stress metallic fissile-bearing debris. This results in the airborne release of radioactive material from the GPC through the Main Plant stack.

B.9.2.2.5.2 Source Term Analysis

For the analysis it was conservatively assumed that the entire estimated GPC radionuclide inventory, presented in Table B.8.2-3 of this SAR Addendum, was subjected to the fire. All radionuclide activities were decay-corrected to the year 2000. The release of radionuclides to the environment is assumed to occur, via airborne transport, using the airborne release fractions (ARFs) and respirable fractions (RFs) deemed appropriate for a given radionuclide as given in Chapters 4.0 and 5.0 of DOE-HDBK-3010-94, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*. An ARF of 5E-03 and RF of 1.0 was used for uranium radionuclides, while an ARF of 5E-04 and RF of 0.7 was used for plutonium radionuclides. An ARF of 1.0 and RF of 1.0 was used for C-14, and an ARF of 0.15 and RF of 1.0 was used for cesium radionuclides. An ARF of 6E-03 and RF of 0.1 was used for all other radionuclides, except Co-60, in which an RF of 1E-02 was used. The damage ratio was set to 1.0, and the leak path factor, as recommended in DOE-STD-3009-94, Appendix A, was set to 1.0.

B.9.2.2.5.3 Consequence Analysis

Table B.9.2-1 of this SAR Addendum presents the dose to a receptor at the on-site evaluation point (640 m) and at the site boundary (1050 m), and the dose to the maximally-exposed off-site individual for various meteorological conditions. Only those radionuclides that contribute greater than 0.1% to the total effective dose equivalent (TEDE) are shown. The maximum TEDE at the on-site evaluation point was calculated to be 6.09 rem (0.0609 Sv), while the maximum TEDE received by an off-site individual was calculated to be 2.51 rem (0.0251 Sv).

B.9.2.2.5.4 Comparison to Guidelines

Section B.9.1.3 of WVNS-SAR-002 defines the means by which safety assurance is shown by providing numerical criteria against which to judge the results of the accident analysis.

Radiological evaluation guidelines given in Section B.9.1.3 state that the total effective dose equivalent to the maximally exposed off-site individual due to an operational accident shall not be greater than 0.5 rem (0.005 Sv) for accidents with estimated frequencies ≤0.1 event per year but >0.01 event per year; 5 rem (0.05 Sv) for accidents with estimated frequencies ≤1E-2 event per year but >1E-4 event per year; and 25 rem (0.25 Sv) for accidents with estimated frequencies ≤1E-4 events per year but >1E-6 events per year.

For the on-site evaluation point, the dose limit is 5 rem (0.05 Sv) TEDE for accidents with estimated frequencies ≤0.1 event per year but >0.01 event per year; 25 rem (0.25 Sv) TEDE for accidents with estimated frequencies ≤1E-2 event per year but >1E-4 event per year; and 100 rem (1.0 Sv) TEDE for accidents with estimated frequencies ≤1E-4 events per year but >1E-6 events per year.

The doses to the maximally exposed off-site individual (2.5 rem TEDE) and the maximum exposure at the on-site evaluation point (6.1 rem TEDE) due to a fire in the GPC are below the Evaluation Guidelines presented in Section B.9.1.3 of WVNS-SAR-002 for this extremely unlikely event.

Table B.9.1-1

Process Hazards Analysis for Cleanup of the Head End Cells

Hazard	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor
HEAD END CELLS					
Fissile-bearing Debris (e.g., Fuel/Hull Material)	1) Criticality during handling or processing activities (which occur in heavily shielded rooms/cells) 2) Criticality after drums are placed in storage (option exists that storage may be in a location that is not heavily shielded)	- Admin control on waste package size - Admin control on drum stacking height - Admin control on handling and processing equipment volumes - Moderator in package minimized through waste drying prior to packaging - Fissile-bearing debris dried prior to placement in storage drums - Potential for dried waste moderation minimized through the use of drum venting devices that can withstand at least 48 in. w.c. without allowing water entry and by providing long-term storage of filled drums in above-grade areas - Low reactivity of unmoderated fissile material in the HECs - Low areal density of fissile materials in HECs - Floor drain and floor hatches in PMC which prevent accumulation of moderator in PMC - Prohibition on the use of liquid decontamination agents and other moderators in areas containing unpackaged fissile materials - Operator training in criticality safety requirements - Massive cell shielding to protect workers from a criticality event - Negligible off-site doses due to atmospheric dispersion of fission gases from a criticality event	1) Low 2) Moderate	1) Extremely Unlikely 2) Incredible	1) 1 2) I

Table B.9.1-1

Process Hazards Analysis for Cleanup of the Head End Cells (Continued)

Hazard	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor
Combustible Radioactive Materials	1) Localized area fire (i.e., a "small" fire) from a pyrophoric reaction (metal fire) or combustible material fire ignited by sparks or flame from handling and processing equipment 2) Large area fire (i.e., a fire engulfing much or all of a cell or room) from a pyrophoric reaction (metal fire), or a metal or combustible material fire ignited by sparks or flame from handling and processing equipment 3) Limited breaching of the HEV system exhaust filter bank due to a fire in a given HEC 4) Major damage (e.g., catastrophic destruction) of the HEV system exhaust filter bank due to a fire in a given HEC with failure to align the Main Plant Ventilation system to service the HECs	- Fuel and clad material primarily in bulk form which inhibits pyrophoric reactions - Fuel and clad material in oxidized state which inhibits pyrophoric reactions - Equipment grounded when necessary to reduce accumulation of static electricity - Contaminated area of cell lined with stainless steel which would prevent the accumulation of static electricity - Buffer area established around thermally hot size reduction activities - Fire suppression capabilities - Minimal amount of smoke and embers associated with pyrophoric reactions that could become entrained in HEV system exhaust airstream - Ventilation for HECs possible through Main Plant Ventilation system - Size reduction activities performed only on non-pyrophoric materials	1) Negligible 2) Moderate 3) Low 4) Moderate	1) Unlikely 2) Extremely Unlikely 3) Extremely Unlikely 4) Incredible	1) 0 2) 3 3) 1 4) I
Chemical Reactions	1) Chemical reaction of fuel/hull material with decontamination agents resulting in dissolution or release of toxic or reactive nuclides	- Decontamination limited to non-acid means (e.g., high pressure water spray, CO_2 decontamination) prior to fuel/hull retrieval - Ventilation system which would prevent release of toxic off-gases to adjacent operating areas	1) Low	1) Extremely Unlikely	1) 1

Table B.9.1-1

Process Hazards Analysis for Cleanup of the Head End Cells

(Concluded)

Hazard	Event	Protective and Mitigative Systems	Consequences	Frequency	Risk Factor
Airborne Contamination in Operating Areas	1) Release of contamination to operating areas due to the extended unavailability of the HEV system or pressure transient in a HEC 2) Rupture of closed drums due to overpressurization caused by accumulating gas generated through radiological processes, in particular the decomposition of water	- Canisters of high pressure air or CO_2 that may be used for decontamination purposes are stored outside HECs - HEPA-filtered backflow dampers provided for HECs - Large combined interior volume of HECs capable of accommodating pressure excursion - Debris is dried prior to being packaged, minimizing chance of radiation induced decomposition of water - Drums containing fissile-bearing debris are vented with filtered venting devices - Redundancy in HEV system - Main Plant Ventilation system can be configured to service HECs	1) Low 2) Negligible	1) Unlikely 2) Extremely Unlikely	1) 2 2) 0
Hazardous (Toxic) Metals (e.g., Pb, Hg, Pu, U)	1) Fire involving toxic metals	- Fuel and clad material primarily in bulk form which inhibits pyrophoric reactions - Fuel and clad material in oxidized state which inhibits pyrophoric reactions - Equipment grounded to reduce accumulation of static electricity - Contaminated area of cells lined with stainless steel which would prevent the accumulation of static electricity - Fire suppression capabilities	1) Negligible	1) Extremely Unlikely	1) 0

Table B.9.2-1

Receptor Doses from a Fire in the General Purpose Cell

Assumptions: Leakpath Factor (LPF) 1.0 Damage Ratio (DR) 1.0
Breathing Rate 3.33E-4 m^3/s Airborne Release Fractions and Respirable Fractions as indicated in Section B.9.2
Release Height 60 m

Receptor Location		640 m	640 m	640 m	1050 m	1050 m	1700 m	
Stability Class, Wind Speed		D, 4.5 m/s	F, 1 m/s	95%	D, 4.5 m/s	F, 1 m/s	95%	
Dispersion (χ/Q)		1.59E-6 s/m^3	2.7E-11 s/m^3	1.63E-4 s/m^3	5.54E-6 s/m^3	1.03E-7 s/m^3	6.72E-5 s/m^3	
Nuclide	Source Term (Ci)	On-Site Dose (rem)	On-Site Dose (rem)	On-Site Dose (rem)	Off-Site Dose (rem)	Off-Site Dose (rem)	Off-Site Dose (rem)	Percent Dose Contribution
Pu-238	7.55E-02	1.84E-02	3.14E-07	1.89E+00	6.41E-02	1.19E-03	7.77E-01	30.96%
Am-241	5.15E-02	1.42E-02	2.42E-07	1.45E+00	4.94E-02	9.18E-04	5.99E-01	23.86%
Cs-137	5.31E+02	8.99E-03	1.53E-07	9.23E-01	3.14E-02	5.38E-04	3.80E-01	15.15%
Pu-239	2.05E-02	5.54E-03	9.46E-08	5.69E-01	1.93E-02	3.59E-04	2.34E-01	9.34%
Pu-240	1.56E-02	4.20E-03	7.17E-08	4.31E-01	1.46E-02	2.72E-04	1.78E-01	7.08%
Pu-241	5.50E-01	2.91E-03	4.96E-08	2.98E-01	1.01E-02	1.89E-04	1.23E-01	4.90%
U-232	4.67E-03	1.66E-03	2.83E-08	1.70E-01	5.78E-03	1.07E-04	7.01E-02	2.79%
Sr-90	1.93E+00	1.33E-03	2.27E-08	1.36E-01	4.63E-03	8.61E-05	5.62E-02	2.24%
Cm-244	4.39E-03	6.26E-04	1.07E-08	6.43E-02	2.19E-03	4.06E-05	2.65E-02	1.06%
Am-243	1.80E-03	4.96E-04	8.47E-09	5.09E-02	1.73E-03	3.22E-05	2.10E-02	0.84%
U-233	7.10E-03	4.88E-04	8.33E-09	5.01E-02	1.70E-03	3.17E-05	2.07E-02	0.82%
U-234	3.39E-03	2.33E-04	3.97E-09	2.39E-02	8.12E-04	1.51E-05	9.85E-03	0.39%
Am-242	2.89E-04	9.33E-05	1.59E-09	9.58E-03	3.25E-04	6.05E-06	3.95E-03	0.16%
Th-228	5.27E-04	8.65E-05	1.48E-09	8.87E-03	3.02E-04	5.61E-06	3.66E-03	0.15%
Am-242m	2.91E-04	7.84E-05	1.34E-09	8.04E-03	2.73E-04	5.08E-06	3.32E-03	0.13%
	Total TEDE	5.93E-02	1.01E-06	6.09E+00	2.07E-01	3.85E-03	2.51E+00	99.86%

B.10.0 CONDUCT OF OPERATIONS

Information relating to the conduct of operations as presented in Chapter B.10.0 of WVNS-SAR-002 is applicable to the activities discussed in this SAR Addendum.

B.11.0 TECHNICAL SAFETY REQUIREMENTS

B.11.1 Introduction

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

Safety Class SSCs are those structures, systems, or components whose preventative and/or mitigative functions are necessary to maintain the consequences of an accident below the off-site evaluation guidelines provided in Section B.9.1.3 of WVNS-SAR-002. Because the accidents analyzed in Chapter B.9 of this Addendum do not rely on protective or mitigative features to maintain does consequences below the evaluation guidelines, no TSRs are required for the activities addressed in this SAR Addendum.

B.11.2 Requirements

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

B.11.3 TSR Input

There are no enveloping Evaluation Basis Accidents that exceed the Evaluation Guidelines established in WVNS-SAR-002. There are no active Safety Class SSCs in facilities within the scope of this SAR, nor are there any Safety Class SSCs which are under the direct control of operators of facilities within the scope of this SAR Addendum.

B.11.3.1 Safety Limits and Limiting Conditions for Operation

There are no evaluation basis accidents which require active Safety Class SSCs nor Safety Class SSCs under the direct control of operators of facilities within the scope of this SAR to mitigate the consequences or prevent the occurrence to meet the

Evaluation Guidelines (EGs) specified in WVNS-SAR-002. Initial accident conditions under the direct control of the operator have been analyzed at the maximum credible worst-case conditions (e.g., maximum credible cell inventory).

Therefore, no TSR Safety Limits or TSR Limiting Conditions for Operation (LCOs) are required for facilities or activities within the scope of this SAR. The WVDP has initiated Process Safety Requirements (PSRs) which contain PSR LCOs as well as associated Surveillance Requirements (see Sections B.4.3.2.6 of WVNS-SAR-002 and Section B.11.3.3 of this Addendum).

B.11.3.2 Design Features

The primary design safety feature of the PMC and GPC relevant to facility decontamination is significant shield walls of each of these areas. Analyses referenced in Section B.8.7.3.3 of WVNS-SAR-002 have concluded that the radiation attenuation provided by this shielding will limit the maximum operator exposure due to a inadvertent criticality to less than 100 mrem.

B.11.3.3 Administrative Controls

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

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

Engineered and administrative controls are provided for HEC decontamination activities to ensure that the occurrence of an inadvertent criticality or other operational mishap is prevented. Administrative controls for the prevention of an inadvertent criticality at the WVDP are developed through the guidelines given in

WVDP-162, *WVDP Nuclear Criticality Safety Program Manual*, and the references contained therein.

Safe conditions are maintained during decontamination operations through the use of administrative controls that limit the size of the collection and storage containers, limit the amount of water contained in a storage container in a storage array, and specify the parameters for array size. These administrative controls, which will be reflected in Process Safety Requirements, are provided to ensure that an inadvertent criticality does not occur in a HEC during area decontamination.

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

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

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

B.11.4 Interface With TSRs from other Facilities

There are no TSRs from other facilities that interface with the HEC.

B.12.0 QUALITY ASSURANCE

Information relating to quality assurance as presented in Chapter B.12.0 of WVNS-SAR-002 is applicable to the activities discussed in this SAR Addendum.

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WVNS RECORD OF REVISION

Rev. No.	Description of Changes	Revision On Page(s)	Dated
0	Original Issue	All	03/01/02