SECTION	EFFECTIVE PAGES	REVISION NO.
Cover Page	Original	1
List of Effective Pages	A, B and C	3
Table of Contents	i through vi	3
List of Tables	vii through viii	2
List of Figures	ix through xi	3
(blank page)	xii	
Commonly Used Acronyms	xiii through xvi	2
Commonly Referenced Isotopes and Elements	xvii	2
1.1	1.1-1 through 1.1-3	1
1.2	1.2-1 through 1.2-2	1
1.3	1.3-1	2
1.4	1.4-1 through 1.4-2	1
1.5	1.5-1 through 1.5-3	1
1.6	1.6-1	2
2.1	2.1-1	1
2.2	2.2-1 through 2.2-27	3
	Tables 2.2-1 through 2.2-3	1
	Figure 2.2-1	2
	Figures 2.2-2 through 2.2-32	1
2.3	2.3-1 through 2.3-65	3
	Table 2.3-1	1
	Figures 2.3-1 through 2.3-3	1
	Figure 2.3-4	3
	Figures 2.3-5 through 2.3-21	1
	Figure 2.3-22	3
	Figure 2.3-23	2
	Figures 2.3-24 through 2.3-25	1
2.4	2.4-1 through 2.4-5	3
	Figure 2.4-1	3
2.5	2.5-1 through 2.5-11	2
	Tables 2.5-1 through 2.5-4	1

SECTION	EFFECTIVE PAGES	REVISION NO.
10	(Deleted)	
Appendix I	I-1 through I-13	Original (11/5/90)
Appendix II	1 through 40 (EE-DEC-0010, Rev D.)	4/17/92
	Appendix B (34 pages)	4/17/92
	Appendix C (63 pages)	4/17/92
	Attachment 1 (10 pages)	4/17/92
	Attachment 2 (2 pages)	4/17/92
Appendix III	1 through 31	11/15/92

INTENTIONALLY LEFT BLANK

**DECOMMISSIONING PLAN
FOR
FORT ST. VRAIN NUCLEAR GENERATING STATION**

TABLE OF CONTENTS

3. PROTECTION OF OCCUPATIONAL AND PUBLIC HEALTH AND SAFETY

6. DECOMMISSIONING TECHNICAL AND ENVIRONMENTAL SPECIFICATIONS

7. DECOMMISSIONING QUALITY ASSURANCE PLAN

8. DECOMMISSIONING ACCESS CONTROL PLAN

APPENDICES

LIST OF FIGURES

INTENTIONALLY LEFT BLANK

2.2 FACILITY DESCRIPTION

2.2.1 General Description

Fort St. Vrain is a High Temperature Gas-Cooled Reactor (HTGR) owned and operated by PSC. Fort St. Vrain's location is approximately 35 miles north of Denver and three and one-half miles northwest of the town of Platteville in Weld County, Colorado.

The site consists of 2798 acres owned by PSC. During the plant operation, approximately one mile square within the site area was designated as the exclusion area, and the licensee maintained complete control over this area. The completed facility is shown in Figure 2.2-1. The basic installation consists of a Reactor Building, a Turbine Building, cooling towers, and an electrical switchyard.

2.2.1.1 Reactor Building

The Reactor Building (Figures 2.2-2 and 2.2-3) houses the prestressed concrete reactor vessel (PCRV), fuel handling area, fuel storage wells (FSWs), fuel shipment preparation facilities, decontamination and radioactive liquid and gas waste processing equipment, and most reactor plant process and service systems. The building is able to withstand wind loadings developed by a 100 mph wind or a tornado of 202 mph total horizontal wind velocity without exceeding yield stresses.

The Reactor Building ventilation exhaust filter system is designed to filter the Reactor Building atmosphere prior to release to the vent stack during both normal and most accident conditions during decommissioning. The Reactor Building is maintained in a subatmospheric condition to ensure that all air leakage will be inward and to minimize unfiltered fission product release from the building. The ventilation system was designed to maintain a subatmospheric condition approximately 1/4-inch water gauge negative. In actual practice, the Reactor Building pressure is normally 0.15 to 0.20 inches water gauge negative, depending on building activities and ventilation system configuration.

The Reactor Building overpressure protection system consists of 94 louvered panels, 4-feet by 8½-feet, each of which provides 12.02 ft^2 of free flow area for a total of 1130 ft^2 of free flow area when fully opened. The louvers are opened by spring pressure and closed (or held closed) by air pressure acting through a pneumatic cylinder. Subatmospheric conditions can be maintained with several louver banks open. The overpressure protection system louvers may be opened on a controlled basis for various reasons (e.g., to provide extra ventilation cooling during hot weather). The louvers must be closed whenever Reactor Building integrity is required.

The PCRV and nuclear steam supply system (NSSS) are located in the west portion of the Reactor Building. The east portion of the Reactor Building houses auxiliary and support systems and facilities such as the FSWs, the hot service facility (HSF), the equipment storage wells (ESWs), storage and laydown areas for various pieces of equipment, radioactive gas and liquid waste storage facilities, and the loading ports for the spent fuel shipping casks (SFSC). The basement area of the Reactor Building contains the building sump/keyway. The volume of the sump/keyway is approximately 44,600 cubic feet.

The Fort St. Vrain Reactor Building is presently designed to withstand the Design Basis Earthquake (DBE) of 0.10 g horizontal ground acceleration at the site without unsafe damage or failure to function. During decommissioning, the Reactor Building will continue to be required to perform its confinement function following a seismic event.

The decommissioning of Fort St. Vrain will not involve any major modifications to the Fort St. Vrain Reactor Building without verification of the seismic qualification. Other than the Reactor Building, no additional seismic analysis of individual decommissioning tasks and removal activities will be required.

The Reactor Building overhead crane is located inside the Reactor Building, over the refueling floor. The Reactor Building crane is the means by which heavy lifting operations and maintenance are performed on the refueling floor of the Reactor Building. The design of the overhead crane conforms to Class "D" crane type specified in the Electric Overhead Crane Institute (EOCI) Specification No. 61 and the AISC Specification "Designs, Fabrication, and Erection of Structural Steel for Buildings" adopted November 30, 1961. All structural steel is ASTM A-36 or better. The crane capacity has been upgraded from 160 tons to a revised capacity of 170 tons. The crane trolley main hook has a capacity of 50 tons and the auxiliary crane hook has a capacity of 17.5 tons.

In order to meet the requirements of the EOCI and AISC specifications, the building girders and crane rails are designed for 125% of the rated load, and the crane bridge girders are designed for an impact loading of not less than 10% of the lifting forces required for 125% of the rated load. The hoisting cable at the main hoist consists of 12 parts of 1-3/8 inch diameter, 6-strand, 37-wire improved plow steel crane rope with a rating of 14.8 tons per part, for a total capacity of 177 tons. The breaking strength of the hoisting cable arrangement is 775 tons.

2.2.1.2 Turbine Building

The Turbine Building (Figures 2.2-2 and 2.2-3) houses the turbine generator with condensing, feedwater, and other auxiliary systems. Included in the Turbine Building is an auxiliary bay area housing the reactor plant ventilation equipment, the controlled personnel access to the Reactor Building, and an area housing the control room and miscellaneous electrical services. The Turbine Building also houses a service and office area which provides space for miscellaneous shops, auxiliary steam system components, and administrative offices.

2.2.1.3 Fuel Storage Building

The Fort St. Vrain Fuel Storage Building is a single level concrete structure located east of the Reactor Building (see Figure 2.2-4). The building is constructed of prestressed concrete panels and twin tees, and is designed to withstand a 202 mph tornado wind and can withstand the design basis tornado missile. This building will be used for decommissioning support.

2.2.2 Prestressed Concrete Reactor Vessel (PCRV) and Internal Components

The PCRV (Figures 2.2-5 and 2.2-6), which contains the NSSS, is a reinforced concrete structure prestressed with steel tendons. Following defueling, the PCRV will contain the majority of the remaining radioactive materials in the Reactor Building.

The Fort St. Vrain systems associated with the PCRV are as follows:

System 11	PCRV and Internal Components
System 12	Control Rod Drive and Orifice Assembly
System 17& System 18	Reactor Reflector and Defueling Elements
System 21	Helium Circulators
System 22	Steam Generators
System 23	Helium Purification System

These systems make up the primary reactor vessel and internal core components located within the PCRV. These systems and components are discussed further in this section and in Section 2.3.

Portions of the PCRV concrete and rebar are expected to remain activated due to direct irradiation from the reactor core. Highly radioactive components will remain inside the PCRV until removed during PCRV decontamination and dismantlement.

Physically, the 15-1/2 foot thick heads and the 9 foot thick concrete walls are constructed around a 3/4-inch thick low-carbon steel liner which forms the internal cavity. The liner is anchored to the concrete at frequent intervals. A core support floor (CSF) is provided within the PCRV in the form of a reinforced 5 foot thick concrete disk with a 3/4-inch carbon steel outer liner, supported by 12 steel core support floor columns from the bottom of the PCRV cavity.

Longitudinal, circumferential and top and bottom crosshead prestressing tendons (448 total) are located in conduits embedded in the PCRV concrete. Tendons are positioned both circumferentially and vertically along the PCRV side walls. There are also tendons across the top and bottom heads in a criss-cross arrangement.

The reactor core arrangement within the PCRV is shown in Figure 2.2-7. The top layer of the core arrangement consisted of hexagonal shaped metal clad reflector blocks (MCRBs) with openings for 37 control rod pairs. The MCRBs provided an inlet plenum for the reactor coolant to the active core. Region constraint devices (RCDs) were located on top of the MCRBs and mechanically interlocked the top layer (not shown on Figure 2.2-7). Hexagonal top reflector elements with coolant channels are located directly below the MCRBs and above the active core region.

The active core was divided into 37 regions and consisted of 1482 fuel elements. Individual fuel elements were hexagonal in cross section and aligned with the coolant channels from the reflector elements and MCRBs. During reactor defueling, the fuel elements are being replaced with defueling elements of identical shape and size. Hexagonal reflector elements are also located to the sides of and below the active core region. Many of the bottom reflector elements contain boronated graphite in Hastelloy cans.

Radially outside of and immediately adjacent to the top, side and bottom hexagonal reflector elements are the large irregular-shaped side reflector blocks. Between the side reflector blocks and the core barrel are the boronated side reflector spacer blocks that contain boronated steel pins and were used for shielding.

The core barrel is a steel cylinder approximately 27 feet 4 inches inside diameter and 29 feet high. The core barrel has 12 upper outer keys and 12 lower outer keys which center the core barrel to the PCRV liner. The lower three feet of the inside surface of the core barrel is insulated. In addition, there are seven thermocouple penetrations located about four feet above the bottom of the core barrel that are between the PCRV liner and the core barrel.

Immediately outboard of the core barrel is a helium interspace area. Outboard of this interspace area is an outer metal insulation cover plate, Kaowool (thermal) insulation, an inner metal insulation cover, another layer of Kaowool, and then the PCRV carbon steel liner. See Figure 2.2-8 for a general arrangement of the thermal barriers.

Below the core region containing the defueling elements, the CSF will bear the weight of the defueling elements and reflectors through the core support posts and the core support blocks. The CSF also is the bottom termination point of the core barrel and has 12 penetrations for the 12 steam generator modules. The CSF is supported from the bottom head of the PCRV with 12 core support floor columns (See Figure 2.2-9). The CSF is a complex component that includes the following features:

1. The CSF is a 29-foot in diameter, 5-foot thick concrete disk, clad with 3/4-inch plate steel, weighing approximately 270 tons.
2. There are 12 conical penetrations which discharged the hot helium gas from the reactor to the steam generators via 12 inlet ducts.
3. The CSF is supported by 12 steel columns that are located near the CSF periphery that are welded to the cladding plate.
4. Within each of the 12 CSF support columns is an array of cooling tubes and instrumentation tubes.
5. All surfaces of the CSF are insulated.
6. There is a monorail spider consisting of twelve heavy structural steel beams in a radial arrangement on the bottom side of the CSF, that were used to position the steam generators during construction.

The lower plenum is below the CSF and houses the steam generator modules (12), circulator diffusers (4), circulators (4) the CSF support columns (12) and the lower floor. A number of instrument and equipment penetrations and wells exist in the PCRV heads and sidewalls.

2.2.3 Balance of Plant Contaminated Components

The systems identified below are considered to be the potentially contaminated balance of plant systems outside of the PCRV at Fort St. Vrain. Decontamination and dismantlement of these BOP systems are discussed in Section 2.3.4:

System 13	Fuel Handling Equipment
System 14	Fuel Storage Facility
System 16	Auxiliary Equipment
System 21	Helium Circulator Auxiliaries
System 23	Helium Purification Auxiliaries
System 24	Helium Storage System
System 46	Reactor Plant Cooling Water System
System 47	Purification Cooling Water System
System 61	Decontamination System
System 62	Radioactive Liquid Waste System
System 63	Radioactive Gas Waste System
System 72	Reactor Building Drain System
System 73	Reactor Building Ventilation System
System 93	Instrumentation and Controls

System 15, fuel and reflector shipping equipment, consists primarily of the shipping casks, truck-trailers, spent fuel container, and cask lifting apparatus and is not a part of the decommissioning project. These equipment items will be retained under their separate 10 CFR 71 license or will be disposed of at some time in the future.

A brief summary of the major components in each of the above balance of plant contaminated systems is as follows:

2.2.3.1 System 13 - Fuel Handling Equipment

The fuel handling equipment that remains contaminated includes the fuel handling machine (FHM, Figure 2.2-10), five reactor isolation valves (Figure 2.2-11) and two refueling sleeves (Figure 2.2-12).

2.2.3.2 System 14 - Fuel Storage Facility

The fuel storage facility (See Figure 2.2-13) consists of nine fuel storage wells constructed of carbon steel liners suspended in concrete pits.

2.2.3.3 System 16 - Auxiliary Equipment

The auxiliary equipment consists of the Auxiliary Transfer Cask (ATC, Figure 2.2-12), ten ESWs (Figure 2.2-14), the HSF (Figure 2.2-15), and three shielding adapters (Figure 2.2-16). Figure 2.2-16 shows a general layout at the location of the various fuel handling and storage system components, and associated auxiliary equipment on the refueling floor.

The ATC is most commonly used to transfer the control rod drive assemblies, refueling sleeves and the shield plugs. The ten ESWs are carbon steel structures embedded in concrete used to store the control rod drive assemblies and the refueling sleeves.

The HSF, constructed of concrete and steel shielding plates, consists of two work areas used for inspection, repair, maintenance, testing and decontamination work. The HSF was designed to allow the decontamination, service and repair of various items of contaminated and activated equipment. The HSF was designed to allow service operations to be performed either indirectly with manipulator equipment or manually after decontamination. Provisions are made for par[illegible]ielding of portions of an assembly while work is proceeding on an exposed portion. The shielding design of the facility conforms to applicable regulations and is adequate for activated or contaminated equipment. In addition, facility design allows the control and disposal of airborne, waterborne, and solid contamination resulting from operation in the HSF. Adequate monitoring for radiation and airborne contamination is available.

The HSF is the shielded general assembly, decontamination and maintenance area, and provides a cell with walls constructed of 3000 psi concrete with steel embedments to provide the necessary shielding. The walls, floor, and ceiling are coated with an epoxy coating to facilitate decontamination.

The HSF consists of two areas, one capable of staging relatively short items and the other for longer items such as the FHM internal mast. Both areas are serviced by two remotely operated manipulators for disassembly and handling of components.

The "longer" area is serviced by a movable platform that straddles the longer component supported at preselected vertical locations by means of retractable support pins. At the bottom of the service platform well, provisions have been made for a water pool to act as shielding for possible storage of highly radioactive items if this should ever be needed.

Ventilation is provided by a fresh air inlet located below the service floor elevation and a portable HEPA filter and blower located near the top of the access with ducting to the Reactor Building ventilation exhaust. The filter assembly is removable by either manipulator. Packaging of highly radioactive solid waste is normally performed in the HSF.

The structural design of the HSF is such that no significant release of radioactive material will occur. All radioactive materials will be adequately contained under all normal and abnormal conditions to protect the health and safety of the public. Adequate shielding and ventilation controls are also provided to protect operating personnel.

2.2.3.4 System 21 - Helium Circulator Auxiliaries

The auxiliary equipment for System 21 was used to provide a supply of high pressure water for the helium circulator bearing lubrication and a supply of purified buffer helium to prevent in-leakage of bearing water into the primary coolant. The major equipment items include buffer helium recirculators, heat exchangers, filters, pumps, helium dryers, chemical injection components, containment tanks, and compressors (See Figure 2.2-17).

2.2.3.5 System 23 - Helium Purification Auxiliaries

The System 23 auxiliary equipment was used to assist in purification of the helium used as the primary reactor coolant. The major equipment items include filters, heat exchangers, compressors, and dryers (See Figure 2.2-18).

2.2.3.6 System 24 - Helium Storage System

The primary purpose of the helium storage system was to provide for both storage and transfer of helium from the reactor vessel and the storage tanks. In addition, the helium storage system was used in testing the control rod reserve shutdown system and for various FHM purging operations. The primary equipment items include a helium transfer compressor, storage tanks, surge tank, oil adsorber, and high pressure helium supply tanks (See Figure 2.2-19).

2.2.3.7 System 46 - Reactor Plant Cooling Water System

The reactor plant cooling water system (Figure 2.2-20) provided cooling water for process heat removal from all auxiliary equipment in the reactor plant. Three loops were provided that formed the PCRV circuit (liner cooling tubes), the PCRV auxiliary circuit (closed loop for various systems/components) and the service water circuit (open loop for various systems/components). The major equipment items include surge tanks, pumps, demineralizers, filters, heat exchangers, chemical injection (tank and pump) and recondenser chiller.

2.2.3.8 System 47 - Purification Cooling Water System

The purification cooling water system (two loops) provided cooling water to the helium purification system heat exchangers. The major components are pumps, expansion tanks, exchangers and associated piping (See Figure 2.2-21).

2.2.3.9 System 61 - Decontamination System

The major equipment items include a water heater, a drying air heater, a filter, pumps,a solution tank and a chemical injection system (See Figure 2.2-22). Certain System 61 piping and manual valves may be used during decommissioning.

2.2.3.10 System 62 - Radioactive Liquid Waste System

The Radioactive Liquid Waste System is designed to collect and permit sampling, analysis and monitoring of all aqueous wastes discharged from the reactor plant. Radioactive liquid effluent releases are made from either the Radioactive Liquid Waste System or from the Reactor Building Sump (RBS). Radioactive liquid effluents released from either System 62 or the RBS will be diluted by the cooling tower blowdown flow prior to release to the surrounding surface waters. The rate of liquid waste release, along with the blowdown flow, will be controlled to assure that the concentration following dilution does not exceed 10 CFR 20 limits, that resultant doses do not exceed 10 CFR 50 Appendix I limits, and that the concentration of tritium in downstream surface waters does not exceed EPA Safe Drinking Water Standards in 40 CFR 141.

The maximum liquid waste transfer pump capacity is 10 gpm and the minimum allowed blowdown flow setting is 1100 gpm. Assuming a maximum liquid waste release rate from a liquid waste transfer pump and a minimum blowdown flow setting, a dilution factor of 100 is assured for releases from the Radioactive Liquid Waste System. The maximum capacity of a RBS pump is approximately 60 gpm. Assuming one RBS pump discharges liquid effluent from the RBS and there is a minimum blowdown flow of 1100 gpm, a dilution factor of approximately 18 would be achieved.

All radioactive liquids released, whether from System 62 or the RBS, are discharged via the cooling tower blowdown line to dilute the waste prior to release from the site. Two redundant activity monitors are provided in the radioactive liquid waste discharge line, arranged in one-out-of-two logic. Upon detection of high concentrations of gross gamma activity, these activity monitors will automatically alarm, shutdown the Liquid Waste Transfer Pumps, shutdown the Reactor Building sump pumps, and shut the block valves in the liquid waste discharge line. A flow switch in the cooling tower blowdown line is also provided to automatically alarm, shutdown the Liquid Waste Transfer pumps, shutdown the Reactor Building sump pumps, and shut the block valves in the liquid waste discharge line on low cooling tower blowdown flow. The liquid waste release rate, low blowdown flow setting, and high activity monitors are adjusted in accordance with the ODCM prior to the release. Failure of the selected flow switch will be annunciated in the appropriate control board and will allow normal closure of the liquid waste discharge block valves by the plant operators. If the blowdown flow measuring devices or radiation monitors become inoperable, liquid effluent releases may continue as described in the ODCM. Flow rates can be estimated using Parshall Flume numbers in the liquid discharge pathways.

As shown in Figure 2.2-23, the radioactive liquid waste system includes a 1000 gallon stainless steel lined liquid waste sump which collects aqueous wastes. The sump is equipped with two vertical pumps. Two liquid waste filters are provided in the sump pump discharge line. Two 3000 gallon liquid waste receiver tanks, two demineralizers and a 3000 gallon liquid waste monitor tank are provided. Two liquid waste transfer pumps are provided for disposition of the liquids collected in the waste receiver or monitor tanks.

The System 62 liquid waste filters and liquid waste demineralizers are the only equipment items in the radioactive liquid waste system that are expected to accumulate significant quantities of activity. The radioactive liquid waste demineralizers consist of steel pressure tanks that contain the demineralizer resin in cartridges. Each tank contains one cartridge loaded with resin. Replacement is accomplished by releasing the latch and opening the tank head, attaching a hoist to the cartridge lifting lug, and lifting the cartridge out of the tank. The cartridge can then be lowered into a suitable shipping container. Since replacement will take place before excessive activity levels are built up in a demineralizer, there are no particular radiological problems involved with replacing the cartridge.

Liquids whose activity is expected to be low will be collected in the liquid waste (System 62) sump. From the System 62 sump, the liquids can be pumped by a sump pump to the RBS/keyway via demineralizers installed in the shield water polishing demineralizers' drain line (this pathway was installed so that the System 62 liquid waste receivers, monitor tank, and associated equipment could be decommissioned), or through one of the two filters to one of the two System 62 liquid waste receivers, or sampled and released directly to the common radioactive liquid effluent discharge line. Flow is switched from one filter to the other whenever pressure drop or radiation level indicates the filter is loaded. When a convenient amount of liquid is collected in a receiver, the incoming fluid will be diverted to the second tank, and the first tank will be isolated. The receivers are connected to the transfer pump manifold to allow the tank contents to be recirculated and thoroughly mixed to ensure representative samples. Prior to release, the contents of the receiver will be sampled and analyzed to determine the dilution factor required to assure that the concentration in the cooling tower blowdown flow will not exceed the values specified in the ODCM. Once the liquid waste release rate, blowdown flow and activity monitors have been set, the liquid waste effluent will be pumped by one of the liquid waste transfer pumps to the cooling tower blowdown line for release to the environment. If the gross concentration of radioactivity in the undiluted effluent exceeds ODCM limits, the liquids will be further treated prior to release to the environment.

When one of the two liquid waste receivers reaches a level at which it is desired to discharge the contents, the operator recirculates the contents of the tank for a period of not less than one hour to mix the liquid. Samples are taken from a sample point located at the discharge of the liquid waste transfer pumps. If required, the contents of the liquid waste receiver are pumped from the receiver to the monitor tank through the decontamination system filters and a liquid waste demineralizer to reduce the activity concentration. At a preset level in the liquid waste receiver, the transfer pump will be automatically tripped. The operator then sets up the system to recirculate the liquid waste monitor tank, and a sample is taken and analyzed. Allowable release rates are then determined to ensure that the radioactive liquid waste, when diluted with cooling tower blowdown flow, can be released in accordance with the limits specified in the ODCM.

In addition to System 62, radioactive liquid effluent releases may also be made from the RBS, as was done during plant operation. The automatic protective features described above that govern System 62 releases also function to isolate releases from the RBS in the event of either high effluent gamma activity levels or low cooling tower blowdown flow. The RBS and keyway have a capacity of over 300,000 gallons before they would overflow onto level 1 of the Reactor Building, though it is not planned to fill the RBS and keyway with over 50,000 gallons of water during decommissioning. Two RBS pumps take suction on the RBS, each with a capacity of approximately 60 gpm, and pump water from the RBS to the same liquid waste effluent discharge path used for releases from System 62.

When it is desired to transfer shield water from the PCRV into the RBS, shield water is drained from the PCRV through the shield water system demineralizers into a liner ("bladder") installed in the rectangular keyway section of the RBS. It is also possible to drain water from a liquid waste system (System 62) tank, or pump water from the System 62 sump, into this liner in the RBS keyway via the shield water system polishing demineralizers, and/or additional demineralizers in the polishing demineralizers' drain line. After water in the RBS liner is sampled, analyzed and an allowable release rate determined, water is directed from the liner to the suction of the RBS pumps. The liner reduces the potential for contamination of the RBS concrete. The piping is configured with alternate flow paths to maximize flexibility of processing water in the RBS or liner in the RBS keyway, should further processing of water be desired prior to release. Water can be routed from the RBS by a RBS pump through the shield water system polishing demineralizers, and/or additional demineralizers in the polishing demineralizers' drain line, back to the RBS or to the liner in the RBS keyway. Additionally, water can be pumped from the liner in the RBS keyway, by means of a temporary pump, through the shield water system polishing demineralizers, and/or additional demineralizers in the polishing demineralizers' drain line, back to the liner in the RBS keyway or to the RBS. Valves in the RBS pumps' discharge path are throttled to achieve the desired flow rates. As with System 62 releases, RBS release rates are determined to ensure that the radioactive liquid waste, when diluted with cooling tower blowdown flow, is released in accordance with the limits specified in the ODCM.

As shown on Figure 2.2-23, a three-way ball valve is installed at the connection between the Reactor Building sump discharge line and the radioactive liquid waste discharge line. This ensures that a simultaneous liquid release from more than one system is prevented. The Reactor Building sump effluent is continuously monitored and the release automatically terminated on high activity or low circulating water blowdown flow. A bypass line is installed around the oil separator in the liquid waste system discharge line. This allows the oil separator to be bypassed during a radioactive liquid waste discharge.

The liquid level in the System 62 liquid waste sump is regulated by a level control which starts a sump pump on high level and stops it on low level. Each liquid waste receiver has a level indicator and a high level alarm. Level switches stop the sump pump when a predetermined level is reached in the tank being filled. Switching from a full receiver to the other receiver (less than full) automatically clears the alarm and allows the sump pump to restart. Two activity monitors are provided in the liquid waste discharge line to the cooling tower blowdown line. A low flow switch is provided in the cooling tower blowdown line to detect flow inadequate to effectively dilute the radioactive liquid waste. Upon detection of inadequate blowdown flow, the liquid waste release will automatically be terminated.

It was not possible to fully drain water from the 17 PCRV bottom head penetrations by means of the PCRV shield water system, since the shield water system took suction on the center access penetration, with an alternate suction on a helium circulator penetration. Therefore, the PCRV lower plenum and penetration drain system was installed after the water level was below the elevation of the PCRV lower plenum floor (less than 50,000 gallons remaining in the vessel). This system permitted the shield water system to be isolated and decommissioned, while providing a path for draining the remaining water from the PCRV and its bottom head penetrations. Following isolation of the shield water system, water was drained from each of the PCRV bottom head penetrations, through a sock filter into the System 62 radioactive liquid waste sump. Water was pumped from the System 62 sump through the additional demineralizers that were installed in the shield water system polishing demineralizers' drain line to the liner in the keyway section of the RBS. The capability existed to further process water from either the keyway liner or the RBS with the additional demineralizers. Following necessary processing, water was released from the RBS as described above.

2.2.3.11 System 63 - Radioactive Gas Waste System

The radioactive gas waste system (System 63) was used during the early stages of decommissioning, to collect and monitor potential radioactive gases prior to discharge from the plant via the Reactor Building ventilation exhaust system. Analyses determined that it was not necessary to direct gases displaced from the PCRV to System 63 for holdup and monitoring, since calculations showed worst case potential gas activity concentrations were low. Therefore, System 63 was removed from service and the Reactor Building ventilation exhaust system was modified so that Reactor Building exhaust fans could take a suction on the top of the PCRV, both during initial fill (Section 2.3.3.6.5) and during subsequent operations (Section 2.3.3.8.2). Since gaseous effluent can be released directly from the PCRV, via the Reactor Building ventilation exhaust system, and it is not necessary to holdup gas in System 63 for monitoring, System 63 will be dismantled/decontaminated relatively early in the decommissioning process. Following is a description of System 63 operations, which applied during the early stages of decommissioning, prior to its removal from service.

The radioactive gas waste system collects and monitors potentially radioactive gases discharged from the plant. Gases with activity below 10 CFR 20, 10 CFR 50 and ODCM limits (taking into account atmospheric dilution) are released directly to the reactor plant exhaust filters in the ventilation system for disposal via the reactor plant vent. Gases that are too radioactive for direct release are stored temporarily in surge tanks. Gases that are probably radioactive are transferred directly to the surge tanks. Gas collected in the surge tanks is analyzed to determine isotopic concentrations. With this information, a release rate of the stored gas is established in conformance with the ODCM, and the gas is vented from the tanks at a controlled rate through the reactor plant exhaust filters.

A flow diagram of the gas waste system is shown in Figure 2.2-24. Gas enters the system by means of two inlet headers, a low activity inlet header for gases that are usually of sufficiently low activity levels and flow rates that they can be released directly to the reactor plant exhaust system for disposal, and a high activity inlet header for gases that may be too radioactive for direct release.

Gas that enter the system via the low activity inlet header flows through one of the gas waste filters (one standby). Each filter consists of a prefilter and an "absolute" filter to remove particulate matter. The gas waste filters are designed for 5 psig and full vacuum. These filters are provided with differential pressure indicators and alarms. When either the filter pressure drop or activity becomes excessive, the standby filter is placed in service and the spent filter is replaced.

After leaving the filters, the gas is monitored for activity. If the activity concentration is within the preset limits, the gas is routed directly to the reactor plant exhaust filters for disposal via the gas waste exhaust blowers. If the activity concentration exceeds the preset limits, two interlocked valves are automatically repositioned to divert the gas to the gas waste vacuum tank. Diversion of the gas in this manner activates an alarm in the control room.

The effluent from the gas waste system is released to the plant ventilation system and undergoes monitoring by two redundant particulate monitors arranged in a one-out-of-two logic.

Two gas waste exhaust blowers (one standby) are provided to maintain the low activity inlet header at a slight vacuum and to discharge low activity gas to the reactor plant exhaust filters for disposal. If the operating blower should fail, the resulting rise in header pressure will automatically start the standby blower, while an alarm alerts the plant operator to the trouble.

Gases that enter the system via the high activity inlet header are normally too radioactive for unidentified or undecayed release and are piped directly to the vacuum tank. These gas wastes will potentially be of such high activity as to make release on an unidentified basis impractical. The gas waste system is therefore arranged to permit identification of the radioisotopes present, and thus allow disposal at the maximum permissible rate on an identified basis.

The gas waste vacuum tank collects gas known to be radioactive. This gas is then compressed by one of the gas waste compressors and is stored in the gas waste surge tanks. The gas waste vacuum tank is a 500 ft^3 tank designed for 15 psig and full vacuum, and is provided with an ASME Code safety valve set at 10 psig. A liquid drain tank is provided to accumulate any liquids collected in the vacuum tank for batch transfer to the liquid waste sump in the radioactive liquid waste system. The liquid drain tank is a 10 ft^3 tank, designed for 15 psig and full vacuum. The gas waste surge tanks are 700 ft^3 tanks designed for 500 psig.

The gas waste compressors normally maintain the operating pressures in the gas waste vacuum tank and the liquid drain tank between 8 and 10 psia. The standby compressor is automatically started if the tank pressure should rise to 12 psia, due either to a failure of the operating compressor or to a gas flow in excess of the compressor capacity (50 acfm). The compressors continue to operate until the gas waste vacuum tank pressure is reduced to 8 psia, at which time the compressors are automatically stopped.

Two gas waste compressors (one standby) are provided. These are three stage reciprocating compressors, enclosed in tanks to collect leakage. The compressor tank vapor spaces are vented to the gas waste vacuum tank. For control of leakage through the piston rod seals, the compressors are equipped with two sealed housings. The inner housing is vented to the compressor suction. The outer housing is maintained at a positive pressure with service air backed up by nitrogen from portable nitrogen cylinders. This arrangement is designed so that any leakage will be air or nitrogen into the compressor, rather than process gas leaking out. Water is used to cool the compressors and the inter and after-coolers.

Compressed and cooled gas from the gas waste compressors is collected in one of the gas waste surge tanks. When the tank pressure reaches a predetermined value, the other tank is manually placed in service and the full tank is isolated by a block valve. A gas sample from the full tank is then obtained, and an analysis performed to determine the radioisotopes present and concentrations, and an allowable release rate. The flow controller in the tank vent line is set at this rate, the tank outlet block valve is opened, and the tank is vented directly to the Reactor Building ventilation exhaust. The Reactor Building ventilation exhaust monitors will alarm for any excessive release of activity and terminate gas waste surge tank venting automatically. In addition, the interlocked valves mentioned previously then automatically divert the discharge from the outlet of the gas waste filters to the gas waste vacuum tank.

The gas waste compressors and the various tanks in the system are protected against overpressure by pressure relief devices, which relieve to the reactor plant exhaust filters. Filter overpressure is prevented in the gas waster filters by pressure controllers located upstream of the filters. The gas waste compressors are equipped with conventional safety valves at the discharge of each stage. These valves relieve to the compressor containment tanks which are, in turn, equalized in pressure with the gas waste vacuum tank. The compressor containment tanks are also equipped with conventional safety valves which discharge to the reactor plant exhaust filters.

These valves protect against inadvertent closure of the pressure equalization lines to the gas waste vacuum tanks. The gas waste surge tanks are each provided with a rupture disk upstream of and in series with a relief valve which discharges to the exhaust blower discharge line. The space between the rupture disk and the relief valve is vented to the gas waste vacuum tank to prevent pressure buildup in this space due to minute leaks. Tank overpressure will blow out the rupture disk, open the relief valve, and safely discharge a portion of the tank contents to the ventilation system exhaust filters. When the pressure surge has been dissipated, the relief valve will close, preventing release of the entire tank contents.

Activity of gases normally leaving the system is monitored at the outlet of the gas waste filters. Alarms are vided to alert the operator if concentrations approach discharge limits. Trip sett.. s are based on the rated flow capacity (80 cfm) of one of the gas waste exhaust bl..wers. If the gas flow rate through the filters should exceed this rate, the permissible activity release rate could be exceeded. For this reason, a high-flow alarm (set at a maximum of 110 scfm) is provided to alert the operator should blower discharge flow exceed normal. A flow recorder is provided in the blower discharge line.

2.2.3.12 System 72 - Reactor Building Drain System

The Reactor Building drain system collects the liquid effluent from various equipment and piping drains for appropriate disposal. The major equipment items include drain tanks, sump, pumps, piping and filters (See Figure 2.2-25). Processing and release of liquid from the System 72 Reactor Building Sump (RBS) is discussed in Section 2.2.3.10.

2.2.3.13 System 73 - Reactor Building Ventilation System

The reactor plant ventilation system provides filtered and heated or cooled air to the Reactor Building for the removal of airborne radioactivity, for personnel comfort, for removal of heat from equipment, piping, and motors, and for control of ambient temperature. The ventilation exhaust filter system is designed to filter the Reactor Building atmosphere prior to release to the ventilation stack during both normal and most accident conditions during decommissioning. The system filters the exhaust air to remove radioactive particulates and discharges the air to the atmosphere through the reactor plant vent extending 10 feet above roof level. The process flow diagram is shown in Figure 2.2-26.

Air handling unit 1A or 1B provides recirculated and outside ventilation air to the Reactor Building, the quantity of outside air being adjusted to allow for building inleakage so as to maintain a negative building pressure and exhaust flow from the Reactor Building. The system is sized to provide a sufficient rate of air change for personnel comfort considering void space and expected occupancy of each area. In most cases, the ventilation air rate is predicated on the heat removal capability, but in all cases there is a sufficient quantity of outdoor air passing through each area to permit the required occupancy in potentially contaminated areas.

Ventilation air to the refueling floor consists of a mixture of outside air. This air is cooled or heated to maintain the desired temperature. The ventilation air in this zone is sufficient to dissipate the heat from external effects, lighting loads and equipment loads. This air, plus leakage from the Reactor Building overpressure relief louvers, is mixed in the refueling floor area, and a portion is routed through the grating in the refueling floor surrounding the PCRV. The balance of the total supply plus leakage is routed back to the supply air handling unit inlet. Perimeter finned tube heat exchangers supplement the heating capacity of ventilation air to offset heat losses from the overpressure relief louvers.

Ventilation air supplied to the area beneath the PCRV is cooled by air handling units located therein. Air through the air handling unit is a mixture of recirculated air and air drawn from outside of the PCRV support ring. A portion of the ventilation air supplied to this zone is exhausted directly to the reactor plant exhaust filters. Pressures in all areas of the Reactor Building are maintained subatmospheric to ensure that any air leakage will be inward, when relied upon to mitigate the consequences of potential accidents, as described in Section 3.4.

Reactor Plant Ventilation Air Supply

All reactor plant ventilation air (except for a fraction of the building leakage) is supplied by one of two air handlers which consist of dampers, 85% efficiency filters, fan, and heating and cooling coils. Temperature of the ventilation air is regulated by controlling the hot or chilled water supply to the heating coils. In the event of a shutdown for repair of the operating air handler, the standby unit is started up, thus assuring 100% ventilation.

Flow of air to each ventilated area is set by manual adjustment of dampers in each main supply duct. Branch supply ducts also equipped with dampers distribute the air throughout each ventilated area as required. Pressure levels for each area are maintained by throttling the supply air to areas maintained at a negative pressure and by throttling the exhaust air from the access control area which is maintained at a positive pressure.

Reactor Plant Exhaust

The ventilation exhaust filter system consists of three trains, one of which is normally in continuous operation. The design flow rate for each train is 19,000 cfm. One train is sufficient to maintain the Reactor Building subatmospheric and thereby minimize unfiltered fission product release from the building. The reactor plant exhaust provides for filtration and high velocity exhaust of any radioactivity inadvertently or accidently released into the Reactor Building. Exhaust air from all locations in the Reactor Building is collected and filtered prior to discharge to the atmosphere via the reactor plant vent extending 10 feet above the roof level (about 180 feet above grade). This ensures sufficient atmospheric dilution to limit activity concentrations at the site boundary to acceptable levels. Air is supplied to the exhaust filters as described previously. Main exhaust ducts from certain areas contain backflow preventers to prevent reverse flow of contaminated air between the various locations in the event of accidental pressure buildup in the refueling floor and the PCRV area.

The exhaust air is subjected to high-efficiency filtration at all times for removal of particulate matter. The particulate filters have a specified and tested capability for removal of 99.9% of particulates that are 0.3 microns and greater in size. The exhaust air is monitored for particulate and gaseous radioactivity in the reactor plant vent (i.e., the common exhaust duct) as it is discharged up the vent to the atmosphere. Radiation alarms from the reactor plant vent monitors are installed in the control room. Gases released from the radioactive gas waste system are routed through the exhaust filters before being discharged to the atmosphere.

In the event that pressure in the PCRV area exceeds atmospheric pressure, the standby exhaust train (exhaust filter and exhaust fan) can be manually started, unless the train is unavailable due to maintenance or repair. Normal operation of the ventilation system controls will throttle fresh air supply to the air handler in order to maintain a negative pressure.

Ventilation System Components

1. Air Handling Units. The two reactor plant air handling units 1A and 1B, which supply a combination of recirculated and outside air to all locations, are each designed to cool 27,200 cfm of air from 99°F (dry bulb), 65°F (wet bulb) to 57°F (dry bulb), 51.5°F (wet bulb).

2. PCRV Ventilation Air Handling Units. The two PCRV ventilation air handling units 1C and 1D are each designed to cool 21,200 cfm of recirculated air from 105°F (db) and 70.5°F (wb) to 57°F (db) and 55.5°F (wb).

3. PCRV Pipe Cavity Ventilation Air Handling Units. The four PCRV pipe cavity ventilation air handling units 1A, 1B, 1C, and 1D are each designed to cool 18,765 cfm of recirculated air from 105°F (db) and 70.5°F (wb) to 60°F (db) and 54.5°F (wb).

4. PCRV Pipe Cavity Chilled Water Heat Exchangers. The two pipe cavity chilled water heat exchangers 1A and 1B are each designed to cool 50,000 cfm of recirculated air from 105°F (db) and 70.5°F (wb) to 57°F (db) and 55.5°F (wb).

5. Exhaust Filters. The reactor plant exhaust filters consist of three units. Exhaust air entering the reactor plant exhaust filters will pass in sequence through a moisture separator and a high efficiency particulate air (HEPA) filter. The HEPA filters conform to the requirements of the following documents:

   (1) MilSpec MIL-F-51068 (D)
   (2) Underwriters Laboratories Standard UL-586-1977
   (3) USNRC Regulatory Guide 1.140
   (4) USNRC Regulatory Guide 1.52, Rev. 2
   (5) ANSI N509 (1976)

   Each of the HEPA filter elements in each filter unit, manufactured from fiberglass conforming to MilSpec MIL-F-51079, is designed for a flow rate of 1200 cfm, with a maximum initial (clean) pressure drop of 1 inch (water gage). Each filter unit is rated at 19,000 cfm to handle the output of its associated reactor plant exhaust fan. The design specification requires that the maximum penetration is 0.03% when tested with thermally-generated monodispersed dioctylphthalate (DOP) smoke having a light-scattering geometric mean droplet diameter of 0.3 microns.

6. Exhaust Fans. The three reactor plant exhaust fans are of the circuit-driven, two-stage, vaneaxial type and are located on the outlet of the reactor plant exhaust filters. Each fan is rated at 19,000 cfm at a total pressure of 12 inches H_2O and is driven by a 60 hp motor.

7. Reactor Building Radioactivity Monitors. The Reactor Building exhaust air is monitored for particulate radioactivity in the reactor plant vent (i.e., the common exhaust duct) as it is discharged to the atmosphere.

2.2.3.14 System 93 - Instrumentation and Control

The portions of the instrumentation and control system that are of interest originate at the PCRV penetrations. These are thermocouple penetrations, process and moisture instrumentation, helium circulator instrumentation, and helium vent piping.

Ventilation Exhaust Monitors

The general design bases for the Reactor Building ventilation exhaust monitoring system are the activity release and radiation exposure limits set by Federal and State regulatory agencies for the protection of personnel and property inside and outside the plant boundaries. The system consists of two particulate monitors, arranged in a one-out-of-two logic.

Ventilation System Air Flow

Ventilation system flow is measured at the outlet of the system. Above-normal flow rates are alarmed to ensure that allowable activity release rates are not exceeded.

Area Radiation Monitors

The area radiation monitors include two monitors, one on the refueling floor and one in the Reactor Building truck bay. These monitors serve as accident monitors to detect unplanned radiation levels in the Reactor Building. The monitors are not relied upon in any accident analysis, but are provided to detect abnormal conditions that could indicate unplanned or accidental radiation levels. Alarm setpoints are specified in the Decommissioning Technical Specifications.

2.2.4 Site Characteristics

2.2.4.1 Demography

The population density in the rural areas surrounding the site is relatively low. The nearest resident is located approximately ½-mile north of the Reactor Building, with the nearest town of Platteville located approximately 3½ miles southeast. This is well outside the proposed EPZ of 100 meters from the Reactor Building. Platteville's population is 1515 based on preliminary 1990 census figures. The nearest population centers with a population over 25,000 are Greeley (60,399), Longmont (51,288), and Loveland (37,173), all based on preliminary 1990 census figures.

2.2.4.2 Geography and Land Use

The site is located in Weld County, Colorado. The area surrounding the site is shown in Figure 2.2-27 with reference circles of 10, 20 and 30 miles radii. The site is located in the South Platte River Valley, approximately thirty-five miles north of Denver. It is located in an agricultural area with gently rolling hills. Grade elevation at the plant is 4,790 feet. The foothills of the Rocky Mountains start to rise about twenty miles west of the site, and the Continental Divide is prominently identified by Longs Peak, located forty miles directly west of the site.

The South Platte River and St. Vrain Creek both pass through portions of the site. These two streams, which join near the northern tip of the site, are not large enough to be used for water transportation.

The general area and land use surrounding the site is predominantly agricultural. The major farm products include grain, feed corn, sugar beets, vegetables, beef cattle, sheep and turkeys. There is also a limited amount of dairy farming in the area.

The industrial facilities in the immediate area are primarily located in the town of Platteville. There are 14 oil/gas wells within a one mile radius of the Reactor Building on Company property.

2.2.4.3 Geology and Seismology

The geologic structure of the general area in which the site is located is shown in Figure 2.2-28. The area lies on the east flank of the Colorado Front Range which is a complexly faulted anticlinal arch on which are superimposed numerous smaller folds and faults. The rocks of the core of the anticlinal arch are Precambrian crystallines, including gneiss, schists, and quartzites which have been intruded by granitic rocks that range in age from Precambrian to Tertiary. On the east flank of the arch are Paleozoic and Mesozoic sedimentary rocks.

The regional structure of this part of Colorado is characterized by sedimentary rocks dipping eastward into the Denver Basin. Along the mountain front the regional structural pattern is interrupted by relatively small, en echelon anticlines that plunge to the southeast. In addition to the fold axes, two groups of faults have been recognized. The most notable occurs along the mountain front and includes a series of faults extending in a generally northwest-southeast direction from the Precambrian into the Paleozoic-Mesozoic sediments. The second group of faults has been recognized primarily in coal mines, located generally east of Boulder. These faults have a northeast-southwest orientation. Both groups of faults are relatively high angle faults.

The faults and the minor folds are related to the uplift of the Front Range which began in Late Cretaceous and continued into the Tertiary. The original field examination and photo interpretation of the area surrounding the site location failed to indicate any evidence of recent movement along any of the known faults. There is no known evidence of any recent seismic activity in the immediate area to have caused any subsequent movement.

The subsoils at the site are St. Vrain-Platte River alluvial sands and gravel overlying Pierre shale bedrock. Generally, 3 to 8 feet of loose to very loose clean sands (with occasional silty and clay lenses) are underlain by 30 to 35 feet of medium dense, fine alluvial sands. These sands are underlain with 4 to 11 feet of medium dense to dense, slightly clay, sandy gravel. Continuing under the gravel, hard to very hard interlayered sandstone and claystone bedrock is found at depth 46 to 51 feet. Free water was found at a depth of about 23 feet. Estimated contours of the surface of the bedrock and the free water level are shown in Figures 2.2-29 and 2.2-30. The shallow loose sands are capable of supporting only low foundation pressures, the medium dense sand will support moderate foundation pressures, and the bedrock will support high foundation pressures.

2.2.4.4 Hydrology

The site location is between the South Platte River and St. Vrain Creek about two miles south of the confluence of these two streams. Surface water rights are owned in four ditches which traverse portions of the site area. In addition, nineteen shallow wells are located on the site area.

Flow of ground water on the site is toward the alluvial deposits of both the South Platte River and St. Vrain Creek. The contours of the water table indicate that the flow of ground water is predominately toward the South Platte River Valley (Figure 2.2-30). Much of the ground water comes from the South Platte River and St. Vrain Creek, such that the water table changes with the flow rate (elevation) in the two streams. Total precipitation, mostly in the form of rain, in the South Platte Valley is small and contributes relatively little to the ground water.

2.2.4.4.1 Plant Water Supply

When the plant was operating, cooling water for the plant was supplied by the main cooling tower and the service water tower. Make-up water for the main cooling tower was obtained from water diverted from the South Platte River and St. Vrain Creek, and supplemented by water from a system of six shallow wells. Make-up water for the service water tower is supplied by the domestic water system, with back-up from the shallow well system. Potable water and water for closed systems in the plant, such as the secondary coolant system, is supplied by the domestic water line, which is connected to a main of the local water district. The local water district is the Central Weld County Water District, whose source of supply is Colorado Big Thompson Project water from Carter Lake, which is located about twenty miles west of the site. The arrangement of the various water supply systems is shown in Figure 2.2-31.

2.2.4.4.2 Plant Effluent

The same liquid effluent release path will be used during decommissioning as was used during normal plant operations. As shown in Figure 2.2-31, diluted liquid effluent will normally be released from the Fort St. Vrain restricted area to the Goosequill Ditch. From the concrete lined Goosequill Ditch, liquid effluent flows into the Jay Thomas Ditch, where additional dilution may occur, and then on to a 25 acre farm pond that contains about 32 million gallons of water. Water flows approximately 8700 feet from the plant to the farm pond.

Dilution water will be taken from the surrounding rivers (South Platte River or St. Vrain Creek) and released via the cooling tower blowdown line, where it is mixed with radioactive liquid effluent. Availability of 2000 gpm for the dilution water flow is assured throughout decommissioning, as this is less than the 4100 gpm circulating water makeup flow (which is 35% of the surface water rights owned by PSC) that was available during normal plant operations.

Miscellaneous turbine plant drains such as floor drains, the Turbine Building sump, and yard drains, are normally directed to the South Platte River via the continuation of the Goosequill ditch to the farm pond. A diversion box is provided in the Turbine Building drain line so that effluent can normally be directed into the Goosequill ditch. Under abnormal conditions which prevent discharge via the Goosequill ditch, effluent is alternatively directed to the St. Vrain Creek via a slough. The reactor plant drains flow to a diversion box from which the flow can be directed to the South Platte River via the continuation of the Goosequill ditch or to the St. Vrain Creek via a slough.

Three lined evaporation ponds (total surface area of 3.6 acres) are present and were utilized to receive chemically treated effluent (primarily produced by periodic regeneration of plant demineralizers) while the plant was operating. Two ponds are located a few hundred feet northeast of the plant building. The other pond is located south of the switchyard.

Use of surface water downstream from the site is limited almost entirely to irrigation. A diagram of the major tributaries and irrigation ditches on the South Platte River between the gaging stations at Henderson and Kersey is shown on Figure 2.2-32. The plant site is located just upstream of the junction with the St. Vrain Creek, adjacent to the Jay Thomas Ditch.

Analyses for the reactor site were conducted on the amount of diversion and stream flows of the nearby water ways. From these original analyses, it was concluded that effluent from the plant would be carried primarily by the South Platte River except during the irrigation season with allowance for reservoir storage. Effluent in irrigation water would enter ground water in the alluvium and would eventually be transported back into the strata bed of the South Platte River. There have been no significant changes in the waterway flows or diversions to require new analyses.

The sources of public water supplies within thirty miles of the site are given in Table 2.2-1. There are two towns downstream within this radius that presently obtain part or all of their water from wells in the alluvium of the South Platte River: Gilcrest and LaSalle. It has been common practice for farmers to obtain domestic water from shallow wells in the alluvium. Many of those who formerly used shallow wells as their source of domestic water now obtain water from the Central Weld County Water District. This same district is the source of domestic water for the plant.

2.2.4.5 Meteorology

2.2.4.5.1 General Climate

The general climate around the Fort St. Vrain reactor site is typical of the Colorado eastern-slope plains region. In this semi-arid region the precipitation averages 10 to 15 inches a year, mostly from thunderstorms in late spring and summer. The annual free water surface evaporation rate is about 45 inches per year (Ref. 4).

The wind records show no dominant direction, although winds out of the north by northeast segment do occur with the greatest frequency. The winds are generally light (10 mph), with higher velocities occurring during various atmospheric disturbances.

The weather is generally mild. Most seasons are characterized by low humidity and sunny days, with occasional, short-lived storms bringing precipitation into the area. Relative humidity averages about 40 percent during the day and 65 percent at night. Thermal radiation losses resulting from lack of cloud cover provide considerable variation in temperature from night to day. Although snowfall may be significant, the snow cover is usually melted in a few days.

2.2.4.5.2 Severe Weather

Tabulated below are temperature and precipitation records for three cities within 20 miles of Fort St. Vrain (see Figure 2.2-27). The recording periods were 1973-1988 (Brighton), 1931-1988 (Longmont), and 1967-1988 (Greeley).

	Brighton	Longmont	Greeley
Max. Temp. (degrees F)	101	106	103
Min. Temp. (degrees F)	-23	-36	-25
Max. Precip. - Day (in.)	2.73	4.04	3.20
Max Snowfall - Month (in.)	22.1	32.1	37.3

Based on information extracted from archived weather data collected from Fort St. Vrain's 60 meter meteorological tower for the period 1986 through 1989, the following weather extremes were observed:

Maximum Temperature	104 degrees F
Minimum Temperature	-26 degrees F
Maximum Wind Velocity	48 mph at wind direction 6.5 degrees (NNE)

Seasonally, winds tend to be strongest in the late winter and spring, the season with high chinook frequency, and again in the summer, when thunderstorms occur frequently. Strong winds, especially under chinook conditions, have been observed on various occasions in eastern Colorado. The chinook winds are strongest immediately to the east of the mountain ridge and diminish rapidly over the plains with increasing distance from the mountains.

The measurement records at the site from July 1986 to December 1989 reveal a prevalence of northerly and southerly winds caused by the shallow depression of the St. Vrain Creek and the South Platte River and by the proximity of the Rocky Mountains. The meteorological data for this period for the wind speed and duration and frequency of distribution is contained in Tables 2.2-2 and 2.2-3, respectively.

Northeastern Colorado has moderate thunderstorm activity. The region near Fort St. Vrain averages 50 days/year in which thunder and lightning occur. The majority of these thunderstorms are present from late spring through the summer.

The Fort St. Vrain site is located in a region that typically experiences five tornadoes per year per 10,000 square miles. The peak tornado activity occurs in the month of June. According to the National Weather Service, 117 tornadoes occurred in Weld County during the period 1950-1987.

INTENTIONALLY LEFT BLANK

2.3 DECOMMISSIONING ACTIVITIES AND PLANNING

2.3.1 Introduction

Decommissioning of Fort St. Vrain includes the dismantlement, decontamination and disposal of radioactively contaminated or potentially contaminated material and components within the PCRV, and in contaminated or potentially contaminated balance of plant systems, and on the remaining site, followed by the final radiation survey. Some of the activities described in this section will be performed prior to approval of the Proposed Decommissioning Plan, and are considered plant closure activities in preparation for decommissioning. Section 2.2 provided a description of the facility and site characteristics. The activated and contaminated portions of Fort St. Vrain which will be decontaminated, dismantled and removed during the decommissioning process are identified in Sections 2.2, 2.3 and 3.1. The specific tasks to be performed to accomplish this goal are discussed in this section. Although personnel conducting the dismantling activities will be exposed to radiation above background levels, the dismantling and decontamination activities have been developed to limit exposure to and control radioactive material in order to maintain occupational doses As Low As Reasonably Achievable (ALARA). Exposure estimates to accomplish the individual tasks and overall project are also provided.

To accomplish the decommissioning of Fort St. Vrain, substantial portions of the existing plant will be dismantled and removed. However, Reactor and Turbine Building components and structures which are not radioactive above releasable limits will remain.

The decommissioning project is divided into three major work areas:

1. Decontamination and dismantlement of the PCRV.
2. Decontamination and dismantlement of the contaminated or potentially contaminated balance of plant systems.
3. Site cleanup and final site radiation survey.

Site cleanup involves pre- and post-decommissioning surveys of the site, and the radiological decontamination necessary to meet the regulatory guidelines to allow release for unrestricted use. These activities are discussed in detail in Section 4 and are not addressed in this section.

PCRV Decontamination and Dismantlement Activities

The following are the major activities involved in dismantling and removing the radioactive portions of the PCRV. These activities will be discussed in further detail in the following Section 2.3.3:

1. Initial PCRV Preparation
2. Removal of the Helium Circulators
3. Steam Generator Disassembly
   a. Initial Preparation
   b. Removal of Steam Generator Secondary Assembly
4. Removal of Activated Components using the ATC and FHM
5. Detensioning and Removal of Pretensioned Tendons
6. Flooding of the PCRV
7. PCRV Top Head Concrete and Liner Removal
8. Dismantling PCRV Core Components
9. Removing the Core Barrel
10. Removal of the Core Support Floor
11. Disassembling the PCRV Lower Plenum
12. Final Dismantling, Decontamination, and Cleanup Activities

A technical evaluation is provided in Section 2.3.2 which provides the basis for the technical approach selected to decontaminate and dismantle the PCRV. A brief description is also provided to identify various techniques which were considered for removal of the PCRV activated concrete.

Balance of Plant System Decontamination and Dismantlement Activities

The balance of plant systems that are contaminated or potentially contaminated above releasable limits and will require decontamination or dismantlement are identified in Section 2.2.3. Work activities associated with these systems are discussed in paragraph 2.3.4 of this section.

2.3.2 Technical Approach Selection

2.3.2.1 Options Considered for Removal of the PCRV

Key elements of the decommissioning plan include the techniques to be used to remove the internal components from the PCRV and to remove the activated concrete from the PCRV structure. This technical approach is based on filling the PCRV with water for shielding while internal components are being removed and using diamond-wire cutting to remove the activated concrete from the PCRV structure. These methods provide the decommissioning project with the optimum schedule, cost, ALARA, risk, and safety considerations for decommissioning the PCRV. A detailed description of the PCRV disassembly techniques and the basis for selecting them are described below.

Two basic methods to disassemble the PCRV were considered: (1) in-air (dry) disassembly, and (2) filling the PCRV with water to provide shielding. Two possible methods of in-air dismantlement were also evaluated, considering factors of ALARA, safety, risks, schedule and cost. The two in-air methods evaluated were fully remote disassembly through the refueling penetrations in the top head, and partially remote disassembly from a massive shielded work platform with the top head removed. The following paragraphs provide an evaluation of each method, discussion of advantages and disadvantages, and a determination of its acceptability.

2.3.2.1.1 <u>Fully Remote, In-Air Disassembly:</u>

The fully remote, in-air approach to the PCRV disassembly relied upon the extensive use of complex remote tooling and resultant limited view of dismantlement operations, which would produce less than predictable results. Although use of remote operations would potentially result in the best ALARA and safety records, all activities would be performed with highly specialized robots. Therefore, the risk of failure or project delays would be greater due to potential breakdowns or delays, lack of reliable backup techniques, and lack of adequate contingency plans. Design, fabrication and testing of specialized robotics would also have to occur in a relatively short period of time, which could cause unnecessary delays in the project schedule. Additionally, removal of the CSF would be extremely difficult, since it is too massive (270 tons) for practical remote removal.

2.3.2.1.2 <u>Partially Remote, In-Air Disassembly:</u>

Partially remote in-air disassembly of the PCRV relied upon a massive shielded work platform that would be required to protect workers from radiation exposure during disassembly. Access ports would be required in this platform through which hand-held, pole-type tools could be inserted to perform the disassembly when the platform is properly indexed over the work location. Using this approach, radiation exposure would be increased because of the extended stay times resulting from restricted tool access. Removal of the top head and the top of the PCRV liner for installation of the work platform would be difficult because of high radiation levels and would probably require remote operations.

2.3.2.1.3 Flooding the PCRV:

The final approach evaluated was to flood the PCRV cavity with water. This selected approach will provide optimum shielding and contamination control and will allow the PCRV disassembly to be completed with optimum balance of schedule, cost, ALARA exposure and minimum risks. Additionally, there is an inherent added measure of safety due to the passive nature of the water for shielding and contamination control. Dismantling operations are greatly simplified by "line of sight" manipulations as a result of direct viewing of the entire cavity.

2.3.2.1.4 Conclusion

The evaluation of the two "dry" (in-air) approaches against the "wet" approach for PCRV disassembly favors filling the PCRV with water for shielding during disassembly. Additionally, it is noted that the "dry" techniques are not completely dry, since large volumes of water are required for any abrasive process used to cut the activated concrete inside the PCRV. Therefore, water would be introduced into the PCRV in each of the "dry" dismantlement options considered.

2.3.2.2 Techniques Considered for Removal of PCRV Activated Concrete

Diamond wire cutting and abrasive water-jet cutting were evaluated for removing activated concrete from the PCRV walls. Diamond wire cutting was chosen as the method for cutting most of the concrete into sections because this proven technology lends itself well to the PCRV concrete removal activities.

Abrasive water-jet cutting was determined to be feasible for much of the concrete cutting but has been minimized to limit the production of contaminated abrasive waste and because of related ALARA considerations. The abrasive water-jet is presently being considered for one application, cutting of the CSF.

The following techniques were also evaluated and were determined to be less desirable for the following reasons: (1) Expanding grout and explosives could be used to break apart the PCRV concrete, but were less desirable because of the heavy reinforcement of the concrete and the presence of the PCRV liner on the face of the concrete; (2) Thermal techniques were evaluated but were less desirable due to tool positioning difficulties, which could cause cost and schedule concerns; (3) Mechanical impact was evaluated but was less desirable due to structural considerations (with the exception of removal of portions of the lowest concrete in the top head).

2.3.3 PCRV Dismantlement and Decontamination

2.3.3.1 Overview of PCRV Dismantlement Activities

The major decommissioning task is the dismantlement and decontamination of the radioactive portions of the PCRV. A description of the PCRV is provided in Section 2.2 and illustrated on Figure 2.2-6. It should be noted that the steps identified in the following paragraphs represent preliminary planning and may change during the detailed engineering and work development that will occur during the planning phase.

This section provides a description of the expected steps necessary to dismantle and decontaminate the PCRV. Initial dismantlement of the PCRV will include removal of selected PCRV internal components and removal of portions of the steam generators. The selected internal PCRV components will be removed from the upper portion of the PCRV using the Fuel Handling Machine (FHM) and Auxiliary Transfer Cask (ATC). These components may include the 37 control rod metal clad reflector blocks (MCRBs), all but six of the 276 non-control rod hexagonal MCRBs, and certain helium purification components. Simultaneously, the non-contaminated portion of the steam generators (also called the steam generator secondary assemblies) will be removed from the lower portion of the PCRV to provide access for detachment of the contaminated steam generator primary assemblies (See Figure 2.2-6).

To facilitate the removal of the remaining reactor core components, the reactor cavity will be flooded with water. As discussed in Section 2.3.2, flooding the PCRV will provide shielding for the workers associated with PCRV dismantlement activities. After the steam generator secondary assemblies are removed from the bottom of the PCRV, the PCRV bottom head and side wall penetrations will be sealed, the PCRV Shield Water System will be connected, and the PCRV will be flooded.

To gain entry to the PCRV cavity, a plug of concrete will be removed from the top of the PCRV. Selected PCRV prestressing tendons (See Figure 2.2-6) will be either (1) detensioned and removed, or (2) detensioned and left in-place. The top head plug will be cut into sections of appropriate size such that the weight and dimensions will allow them to be handled with the Reactor Building crane and permit them to be moved out of the building. After the majority of the concrete has been removed from the PCRV top head, the 3/4-inch steel PCRV liner plate will be cut and removed with the remaining concrete, together with the top head liner insulation. A detailed discussion of this activity is provided in Section 2.3.3.7.

Once access is gained to the PCRV cavity, a work platform will be installed at the approximate elevation of the top of the PCRV where the liner and concrete have been removed. Working from this platform, workers will remove core components, including the remaining MCRBs, defueling elements, hexagonal reflector blocks, large side reflector blocks, side spacer blocks, core support blocks and core support posts. This activity is described in Section 2.3.3.8.

Once the core internals have been removed, the core barrel (a large carbon steel cylinder) will be removed by cutting it into pieces sized to fit in radwaste containers. (See Section 2.3.3.9)

Following removal of the core barrel, the PCRV water level will be lowered and the CSF insulation removed, in preparation for removal of the CSF. The CSF is a 29-foot diameter, 5-foot thick disk of reinforced concrete within a 3/4-inch steel casing weighing approximately 270 tons. The CSF will be detached from the twelve CSF columns and the twelve steam generator inlet ducts and lifted with a hydraulic jacking system to the PCRV top head region. The jacking system will then lower the CSF onto supports on the ledge in the cavity where the PCRV top head was removed. Once supported, the CSF will be sectioned into segments small enough for handling by the Reactor Building crane. This activity is discussed in Section 2.3.3.10.

Once the CSF is removed, the PCRV lower plenum is exposed and the helium circulator diffusers and steam generator primary modules can be removed. These activities are discussed in Section 2.3.3.11.

The removal of the steam generator primary assemblies completes the removal of the major PCRV radioactive components. Remaining radioactive components include the activated "beltline concrete" around the reactor core region, the PCRV liner, liner insulation and insulation cover plates, and the PCRV lower floor with its supports. The activated beltline concrete is the PCRV region that was adjacent to the reactor core. It is estimated that this activated region is defined by a cylinder with an 18 to 24 inch wall thickness and a height of 40 feet. This section of PCRV sidewall will be removed by cutting and removing vertical segments. The activated liner plate, insulation and cover plates will be removed with the concrete. These activities are discussed in Section 2.3.3.12.

In the lower portion of the PCRV cavity (below the CSF), the insulation and insulation cover plates will be removed from the PCRV liner. The lower floor and all support members, insulation and other components will be removed, and the exposed PCRV liner will be surveyed and decontaminated as appropriate. These activities are also discussed in Section 2.3.3.12.

2.3.3.2 Initial PCRV Preparation

Initial tasks to be completed in preparation for dismantling the PCRV will include acquiring tooling, setting up training mockups, installation of the PCRV Shield Water System, and craft personnel training in accordance with Section 2.6.

Preparation activities include any modifications or revisions to existing facilities and equipment and installation of new facilities and equipment that would be necessary for their use in supporting the decommissioning operations. No major facility modifications are required that will affect the safety of the facility.

The refueling deck equipment hatch and truck bay door may be enlarged to allow passage of larger items. The Reactor Building crane may also be re-reeved to provide additional vertical travel which will allow the 170-ton main hook to travel from the refueling floor to ground level. This re-reeved configuration is consistent with the crane configuration used during original plant construction and may be necessary to provide the lifting capacity to lift heavy loads, such as large concrete sections, when components are removed from within the PCRV.

The need for extensive waste handling facilities in addition to those already present has been minimized by proper sequencing of the dismantlement activities and by proper management of the radioactive waste program, as described in Section 3.3. Off-site facilities will be utilized when necessary and practical for waste processing and final packaging. Proper task planning and sequencing will aid in minimizing accumulation of radioactive waste on site.

A self-contained mobile laundry facility to clean all contaminated protective clothing, as well as a self-contained mobile respirator cleaning facility to clean contaminated respirators, will be utilized. The PCRV Shield Water System, installed to maintain water purity in the flooded PCRV, will be discussed in Section 2.3.3.6.

Following helium circulator machine assembly removal, as identified in Section 1.5.2 (See Figure 2.3-1), the diffuser shutoff valve assemblies will be disconnected from the PCRV penetrations and the penetrations will then be sealed by installing a closure fixture designed to withstand pressure when the PCRV is flooded with water.

2.3.3.3 Steam Generator Disassembly

2.3.3.3.1 Initial Steam Generator Disassembly

Each of the twelve steam generators consists of a primary assembly and a secondary assembly (Figure 2.3-2). The primary assembly is located within the PCRV lower plenum and the secondary assembly is located beneath the primary assembly inside a PCRV bottom head steam generator penetration. The primary assembly is contaminated and the secondary assembly is not expected to be contaminated. However, in order to remove the primary assembly, the secondary assembly must first be removed from beneath the PCRV.

The removal of the insulation from the steam generator secondary side piping will be limited to the sections of feedwater, main steam, hot reheat, and cold reheat piping that need to be severed for the steam generator secondary side removal. Prior to removal, the insulation will be tested for asbestos content. If asbestos is present, appropriate controls will be implemented for removal of the insulation. Following the removal of the insulation, the main steam, feedwater, hot reheat and cold reheat piping will be cut which will allow the secondary side of the twelve (12) steam generators to be removed.

2.3.3.3.2 Removal of Steam Generator Secondary Assembly

Removal of the steam generator secondary assemblies (See Figure 2.3-2) will be accomplished in the reverse of the original construction installation sequence. The steam generator secondary assemblies are expected to be free of contamination.

The Marmon clamp (See Figure 2.3-2) will be removed from the lower end of the steam generator secondary assembly. This will allow withdrawal of the hot reheat piping from the steam generators. Because of the length of the hot reheat pipe, it will be severed into several sections as it is being withdrawn from the steam generators.

The cold reheat pipe will then be severed by remote operations at the threaded connection below the primary closure dome. Severing this connection remotely will make it unnecessary to send an individual inside the cold reheat pipe, as was done during installation. After the top of the cold reheat pipe has been cut, the lower reheat nozzle assembly will be cut free of the steam generator secondary assembly at an elevation below the feedwater ring header. This will allow the withdrawal of the cold reheat pipe from the steam generator for disposal.

After the cold reheat piping has been removed, the 40 feedwater, instrument, and steam tubes will be cut remotely below the primary closure dome. The steam generator secondary assembly will then be rigged for lowering. The secondary closure weld will be cut and the steam generator secondary assembly will be lowered out of the PCRV penetration liner. The Rucker machine, which is a large turntable designed to handle heavy loads under the PCRV, will be used to handle the steam generator secondary assemblies in the reverse order of the installation operations.

In order to detach the primary assembly from the penetration liner, the final step will be to remove the 36 nuts that attach the primary module to the penetration liner flange. The steam generator primary assembly is stabilized by the steam generator shroud connection to the lower floor and the helium duct connection to the CSF.

After each of the twelve steam generator primary assemblies is detached from its respective penetration, it will then be removed through the top of the PCRV after the CSF is removed. This is discussed further in Section 2.3.3.11.

When cutting operations have been completed, the interior of the penetration liner may be sprayed with a strippable coating to ease future decontamination operations. A new secondary closure plate will be welded in place to seal the penetration liner in preparation for flooding the PCRV.

In parallel with the removal of the steam generator secondary assemblies, the PCRV lower plenum (See Figure 2.3-3) will be entered through the PCRV bottom head access penetration after removal of the shield plug. A radiological survey of this area will be performed to determine radiation levels and major contributors in this area. Still photographs and video recordings will also be made to assist in mockup design and training for eventual dismantlement of the PCRV lower plenum.

2.3.3.4 Removal of Activated Components Using the ATC and FHM

Selected activated components will be removed from the PCRV using the ATC and the FHM. Use of this equipment will provide shielding while transferring highly radioactive components from the PCRV to shipping casks with minimal personnel exposure. The 37 control rod MCRBs and all but six of the 276 non-control rod hexagonal MCRBs will be removed from the PCRV by the FHM. Removal of the highly radioactive components with the FHM is the preferred method to maintain personnel exposure ALARA. However, use of the FHM for this purpose is dependent on its operability, and its availability has not been relied upon as the basis for removal of these components. Removal of certain components in the helium purification wells and penetrations, and placement of the refueling sleeves, will be performed by the ATC.

As identified in Section 1.5.2, the RCDs, CRDOAs, and high temperature helium purification equipment may have been previously removed.

2.3.3.5 Detensioning and Removal of Pretensioned Tendons

2.3.3.5.1 Tendon Removal

Concurrent with operations discussed in Sections 2.3.3.2 through 2.3.3.4 is the detensioning of selected tendons in the PCRV. The PCRV has a total of 448 prestressing tendons made up of vertical, circumferential, and top and bottom cross head tendons (See Figure 2.2-6). The following identifies the number of tendons planned to be detensioned and the number of tendons planned to be both detensioned and removed for each of the various tendon types. The exact tendons affected could change somewhat based on conditions encountered during decommissioning.

Type of Tendon	Number of Tendons	Number to Be Detensioned	No. of To Be
Vertical	90	90	90
Circumferential	310	70	24
Top Cross Head	24	24	24
Bottom Cross Head	24	0	0
TOTALS:	448	184	138

Temporary scaffolding will be erected to facilitate tendon removal. Tendons will be detensioned by cutting or grinding individual tendon wire buttonheads.

2.3.3.5.2 Vessel Integrity

The modified PCRV structure was evaluated for the loadings produced by the dead weight of the PCRV structure and components, the lifting operations of the CSF, and a design basis seismic event. The structural evaluation considered the detensioning and removal of all 24 top cross head tendons, all 90 vertical tendons, and all circumferential tendons from group 9 through group 19. In addition, it was conservatively assumed that all circumferential tendons (inner, middle and outer) were detensioned even though it is planned to only detension the inner and middle tendons in the top head and the inner tendons in the belt line region.

To evaluate this modified structure, a simplified free-body lumped mass model fixed at the basement floor of the PCRV structure was developed for analysis with the STAAD-III/ISDS (Ref. 5) computer code. Inputs to the analysis included NRC Regulatory Guide 1.60 "Design Response Spectra for Seismic Design of Nuclear Power Plants" (Ref. 6) design response spectra normalized to the Fort St. Vrain specific "double design earthquake" (referred to as the Design Basis Earthquake (DBE)) ground motions with NRC Regulatory Guide 1.61 "Damping Values for Seismic Design of Nuclear Power Plants" (Ref. 7) damping values. Specifically, the FSV Operating Basis Earthquake (OBE) ground motions of 0.05g horizontal and 0.033g vertical accelerations and DBE ground motions of 0.10g horizontal and 0.067g vertical accelerations were used in the analysis. In addition, the concrete has a compressive strength of f'c = 6000 psi and the reinforcing steel was conservatively assumed to be Grade 40 steel with a F_y = 40,000 psi, although it is Grade 60 or better. The damping values are 2% horizontal and vertical for the OBE, and 2% vertical and 5% horizontal for the DBE.

The modified structure was conservatively evaluated for the loadings produced by the dead weight of the PCRV structure (assuming the PCRV is flooded with water), PCRV internal components, the lifting operations of the CSF, and OBE and DBE events. The resulting forces and moments in each of the individual cross sections of the PCRV were used to develop concrete compressive and reinforcing steel tensile stresses. In the development of the concrete and reinforcing steel stresses, all vertical, cross-head and circumferential pre-stressing tendons were considered detensioned.

The resulting concrete compressive and reinforcing steel tensile stresses are provided below:

CROSS SECTION	LOADING	CONCRETE COMPRESSIVE STRESS (psi)	REINFORCING STEEL TENSILE STRESS (psi)
	Lift	7.7	0.0
	OBE Seismic[1]	22.0	10.2
	DBE Seismic[1]	25.1	23.7
Belt Line Region	DW	134.6	0.0
	Lift	6.1	0.0
	OBE Seismic[1]	298.9	814.8
	DBE Seismic[1]	381.5	1,268.0

[1] - Includes the effects of Dead Weight (DW).

To determine the margins of safety, the worst case load combinations of dead weight, lift loads, and OBE and DBE seismic events were considered using the following equations, consistent with the "Building Code Requirements for Reinforced Concrete" (Ref. 8):

Equation-1: $U_1 = 0.75\ (1.4\ DW + 1.7\ Lift + 1.87\ OBE\ Seismic)$

Equation-2: $U_2 = 0.75 (1.4\ DW + 1.7\ Lift + 1.4\ DBE\ Seismic)$

The total combined concrete compressive stress was compared to the "Limit Condition 2" compressive stress allowable of 0.85 f'c or 5100 psi as outlined by Section E.1.2.6.2 of the FSV FSAR. The reinforcing steel allowable tensile stress was considered to be 90% of the yield stress of Grade 40 steel, or 36,000 psi. The margins of safety, where margin equals (allowable stress)/(combined actual stress), are summarized below:

SECTION				
	EQ-1	EQ-2	EQ-1	EQ-2
Belt Line	8.97	9.28	31.5	27.0

In summary, the structural evaluation has determined that the resulting concrete compressive and reinforcing steel tensile stresses are well within allowable limits. Furthermore, adequate margin of safety exists for all loading conditions specified. The potential for cracking of concrete in the modified top head and beltline regions has been reviewed and, considering the relatively low tensile stress in a conservative number of reinforcing bars, cracking due to tension in the concrete is not expected.

2.3.3.6 Flooding of the PCRV

2.3.3.6.1 Preparation for Flooding the PCRV

Once operations described in Sections 2.3.3.2 through 2.3.3.4 have been completed, activities may proceed to install the PCRV Shield Water System and flood the PCRV. A network of PCRV liner cooling tubes (System 46) and the tendon tubes within the PCRV concrete wall creates a potential pathway for water leakage and the spread of contamination during the cutting of the PCRV concrete. To block these potential leak paths and prevent the spread of contamination, the liner cooling tubes and selected tendon tubes will be sealed with grout or other suitable sealing methods.

Before flooding the vessel, all PCRV penetrations that are below the PCRV waterline and have had their instrumentation removed (including instrument penetration internal components and other items) will be sealed. These penetrations will be sealed by either one or a combination of the following: cutting and capping just outside the PCRV or by installation of bolted and gasketted blind flanges. Where welding is utilized, the welds will be non-destructively tested per applicable codes.

Some of the instrumentation was removed from PCRV penetrations prior to flooding the PCRV. High dose rates encountered on the core outlet thermocouple assemblies precluded their complete removal prior to PCRV flooding, and it was determined that underwater removal was preferable to limit occupational radiation exposures. Therefore, preparations for flooding the PCRV included installation of a push rod assembly and redundant wiper seals in each core outlet thermocouple penetration, which will permit underwater removal of the thermocouple assemblies, as described in Section 2.3.3.8.6 (Ref. 16).

Two PCRV low point penetrations (the bottom head access penetration and one helium circulator penetration) will be sealed with specially designed cover plates before the PCRV is flooded. These closures will provide alternate suction and fill paths for the PCRV Shield Water System described below. The welded connections will be non-destructively tested. After verifying that the steam generator and helium circulator penetrations are sealed, the PCRV Shield Water System will be installed.

2.3.3.6.2 <u>Expected Conditions Within the Flooded PCRV</u>

1. <u>Radionuclides:</u> The radionuclides of concern that will be encountered during dismantlement operations inside the PCRV have been previously identified in the activation analysis provided as Appendix II and are summarized in Table 3.1-2. A fraction of each of these radionuclides is expected to leach into the water from the graphite when the PCRV is flooded.

The principal radionuclides of concern are tritium, Co-60, Fe-55 and Cs-137. Of these, Co-60 is expected to provide the majority of the whole body exposure to occupational workers as a result of dismantlement operations. These radionuclides will appear in particulate and ionic form, and the PCRV Shield Water System will be designed to remove the principal radionuclides.

Although not a major contributor to whole body exposures, the other major radionuclide of concern is tritium. Since the tritium cannot be removed by processing through filters or demineralizers, it will be processed and released using liquid effluent discharge operations in accordance with 10 CFR 20 limits. The maximum tritium concentration shall not exceed the limit specified in the Decommissioning Technical Specifications.

2. Particulates: During PCRV dismantlement evolutions, debris will be generated from handling graphite blocks, concrete cutting operations, insulation, and dross from underwater cutting operations. Various size particles of debris are expected to be generated from the various cutting methods to be employed during PCRV dismantlement operations, including diamond wire cutting (PCRV top head), oxy-acetylene cutting, thermitic rod cutting, and underwater plasma arc cutting. Large particles will settle downward and remain in the PCRV. Smaller particles will be circulated with the water and will be removed by the PCRV Shield Water System. Suitable provisions will be included in the system design to collect this debris and prevent it from damaging system components.

Some graphite dust is expected to become waterborne after the PCRV is flooded. The need to filter this graphite has been incorporated into the design and filter sizing of the PCRV Shield Water System. The possibility of breakdown of the Kaowool insulation (described in Section 2.2.2 and Figure 2.2-8), attached to the PCRV liner immediately outboard of the core barrel, has also been considered. However, based on information from the manufacturer, this insulation is not expected to break down when immersed in water and therefore will not be a factor in the design of the system filtration trains.

2.3.3.6.3 PCRV Shield Water System - Design Considerations

The primary function of the system is to provide water shielding to minimize personnel exposure during dismantlement operations internal to the PCRV. The system will also be designed to provide a means to meet 10 CFR 20 discharge limits for the radionuclides identified above and ensure compliance with 10 CFR 50 Appendix I guidance for radioactive liquid waste discharges to unrestricted areas. Specifically, the system design will provide:

(1) an acceptable method to reduce tritium inventory by liquid effluent discharge operations.

(2) an acceptable radioactive liquid waste processing path to reduce the concentrations of fission and activation products for discharge to unrestricted areas, as well as control radionuclide concentrations in the PCRV water inventory to maintain occupational exposures in the work area ALARA.

In addition to these regulatory criteria, the system will also be designed to meet the following non-regulatory considerations:

(1) maintain acceptable water clarity to conduct underwater dismantlement operations.
(2) minimize corrosion and biological fouling by suitable chemistry control.
(3) provide a means of initial fill of the PCRV, as well as the ability for makeup with demineralized water to compensate for system losses due to effluent discharges and evaporation.

The recommendations of Regulatory Guide 1.143 "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants" (Ref. 9) will be used in system design.

This system will be designed to maintain occupational radiation exposures within 10 CFR 20 regulatory limits and as low as is reasonably achievable (ALARA). The recommendations of Regulatory Guide 8.8 "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be ALARA" (Ref. 10) will also be incorporated (to the degree applicable) into system design.

2.3.3.6.4 PCRV Shield Water System - General Design Information

The PCRV Shield Water System is shown in Figures 2.3-4. The system will consist of two parallel trains of equipment, each sized for 500 gpm (or 50%) of the total flow. This total design flow rate (1000 gpm) will provide a turnover rate of approximately five PCRV volumes per day. Based on discussions with personnel involved in the TMI cleanup, this rate is considered adequate. The system will be designed to allow a complete train or individual components to be removed from service for preventive or corrective maintenance Provisions will be included for the addition of another complete train if required. The trains are cross-connected to permit the pumps and filters to be used interchangeably between the two trains.

Maximum flexibility will be designed into the system to minimize the impact of individual component failure on system availability. Sufficient valves, bypasses and interconnecting piping will be utilized to allow continued system operation during scheduled maintenance or in the event of a component failure. Remote-indicating radiation detectors will be used to monitor dose rates on components in high radiation areas, such as at strainers, filters and demineralizers.

A. Filtration Trains:

The purpose of the system filtration trains will be to maintain PCRV water clarity by removing suspended solids and particulate matter, and to reduce concentrations of suspended radioactive particulates. In order to maintain optimum water clarity, suction of the PCRV Water Shield System will be taken from the bottom of the PCRV and clarified water will be returned to the top of the vessel. The system will have two filtration trains consisting of the following components:

1. Clarifying Pump Suction Strainer: A strainer will be installed in the suction line of each clarifying pump to prevent equipment damage due to large particulate debris. The strainers will be duplex type, and provisions for radiation monitoring of these filters will be included in the design.

2. Clarifying Pump: Each train will have one clarifying pump. The pumps will be horizontal, centrifugal process pumps. Each pump will have a capacity of 500 gpm through the associated train of equipment and will return the clarified water to the PCRV.

3. Filter Trains: Each filter train will consist of two filters, with the filters mounted in a series arrangement. Bypasses will be provided to allow each filter to be operated individually or in series with other filters. The micron sizing of filter elements will be determined by on-going decommissioning operations. Filter element change out requirements will be based on a maximum differential pressure of 20 psid or a maximum radiation reading of 1 R/hr. It is expected that approximately five micron filters will be effective in removing particulates

after the PCRV is flooded. Provisions will be included for shielding and radiation monitoring.

B. Demineralizer Train:

The system will also be equipped with a sidestream demineralizer train. The purpose of the demineralizer train will be to reduce concentrations of dissolved radionuclides (specifically Co-60, Fe-55 and Cs-137) to levels that will allow discharge to unrestricted areas, as well as reduce concentrations in the PCRV to minimize radiation exposure to occupationally exposed personnel. The demineralizer train will consist of the following components:

1. Demineralizers: The demineralizers enable removal of the principle dissolved radionuclides of concern (Co^{++}, Fe^{++}, Cs^{+}) from the shield water, by means of ion exchange resin. There are 8 roughing demineralizer vessels, arranged in parallel, and 2 polishing demineralizer vessels, arranged in series. The polishing demineralizers are used to further process water being routed to the radioactive liquid waste system (System 62). During normal operation, 4 roughing demineralizer vessels will be in service, with a flow of approximately 50 gpm to each vessel. If the shield water contains a high level of dissolved radioactive nuclides, a design flow rate of up to 400 gpm can be established through all 8 roughing demineralizers. A return line directs effluent from the roughing demineralizers back to the top of the PCRV. A portion of the water discharged from the roughing demineralizers can be routed through the polishing demineralizers to the liquid waste system. Normal flow through the two polishing demineralizers is 10 to 50 gpm. As water is discharged from the shield water system via the polishing demineralizers, an equivalent volume of demineralized water must be added to maintain PCRV level. Shielding is provided to protect personnel working with the demineralizer system, and to keep general area radiation levels within acceptable limits. Remote radiation monitors are installed.

2. Resin Fines Filter: One cage-type filter will be provided to prevent the loss of resin fines from the demineralizer and possible discharge into the PCRV. This filter will be designed for a minimum capacity of 100 gpm and to retain 98% of all particles greater than 5 micron at 15 psid.

C. Chemical Addition Train:

The system will also include a chemical addition train. The purpose of the chemical addition train will be to minimize corrosion by suitable chemistry control within the PCRV system and to minimize biological fouling. The chemical addition train will consist of the following components:

1. Chemical Addition Tanks: Two 100-gallon chemical addition tanks will be included. For the initial PCRV shield water chemistry control program, described in Section 2.3.3.6.5, one tank was used to contain Calgon LCS-20 (corrosion inhibitor) and the other hydrogen peroxide (biocide). The tanks will be used to add chemicals to the system for the maintenance of proper chemistry and to control biological fouling.

2. Chemical Addition Pumps: Two chemical addition pumps will be included.

D. Submersible Skimmer and Vacuum System:

Consistent with experience gained conducting underwater operations at TMI, a submersible skimmer and vacuum subsystem has been included in the system design to maintain adequate surface visibility, to reduce surface contamination and to provide vacuuming capability.

E. System Controls:

The system incorporates remote manual controls on a central control panel located in a non-congested area on level 1 of the Reactor Building. All major operations of the system, including flow adjustments and valve positioning, will be performed from the panel. Pumps will be controlled both locally and remotely from the central control panel. Alarms for either high differential pressure or high radiation will notify the operator of the need to replace filter elements or the demineralizer resins. PCRV water level indication and PCRV high/low water level alarms will be included in the design to facilitate system operation and control.

F. Other General Design Information:

The filters and demineralizers will be shielded to reduce radiation fields in the immediate vicinity of these components during operation. The system will have isolation valves at the outlets from the PCRV to the suction of the PCRV Shield Water System to allow isolation if a problem should develop in the system. There will also be other valves to allow isolation of portions of the system for maintenance or repair. A connection will also be provided for an additional train, if necessary.

The major components of the system will be prefabricated on skids with drip pans to contain potential leakage, and will be installed in low occupancy areas of the Reactor Building to minimize personnel exposure. Skids that include system filters will be shielded. The operating controls and chemical addition skids will be located on level 1 of the Reactor Building. The skids will be interconnected with other skids as well as with the suction piping from the bottom of the PCRV and the return piping to the top of the PCRV. Drains from the PCRV Shield Water System will be directed to the existing Fort St. Vrain Radioactive Liquid Waste System (System 62) to permit the use of the existing effluent discharge paths and radioactivity monitoring and controls.

2.3.3.6.5 PCRV Shield Water System - Operation

A. Initial Fill of the PCRV

After system installation and check out, the first operation of the system will be during the initial fill of the PCRV. As opposed to normal system operation, the initial fill will be from the bottom of the PCRV via the suction piping. The initial fill will be accomplished prior to the final cutting and removal of the PCRV top head concrete, and prior to gaining access into the PCRV internal cavity. Demineralized water for the initial PCRV fill (estimated to require approximately 325,000 gallons) will be from the portable demineralized water makeup system, which is described in a following section.

As the PCRV is being filled, the displaced air and gas will be passed through a portable HEPA filter system attached to a penetration in the PCRV head. Using temporary ventilation ducting, the displaced air from the HEPA filter will then be routed to the existing Reactor Building Ventilation Exhaust System (System

73). The PCRV water will be sampled and analyzed daily during initial filling to verify that the tritium concentration remains below the Technical Specification limits. In addition, the air displaced during the filling process will be sampled for tritium and analyzed daily until the filling operation is completed to ensure that discharges will be in compliance with the Fort St. Vrain Offsite Dose Calculation Manual (ODCM Ref. 11). The PCRV will be inspected for leaks during the initial fill process. The fill operation is expected to take several days. After the PCRV has been filled, the air will be sampled and analyzed for tritium at the same frequency required by Technical Specification LC 3.4 for water until the air in the PCRV has been adequately characterized.

The Decommissioning Technical Specifications require that the PCRV water be sampled and analyzed daily for tritium concentration during the initial fill of the PCRV. Following the initial fill, sample frequency may be reduced to weekly after the tritium concentration has decreased to less than 0.1 μCi/cc. Limits have been established in the Decommissioning Technical Specifications to assure that tritium activity concentrations in the PCRV Shield Water System will not exceed those postulated in the decommissioning accident analyses.

B. Normal System Operation

Once the PCRV has been filled, the PCRV Shield Water System lineup will be restored to take a suction from the bottom of the PCRV, with the return flow to the top of the PCRV. The system will be operated to establish and maintain water clarity, water chemistry and to minimize waterborne concentrations of radionuclides. The demineralizer will be placed into service as required. Filters and demineralizers will be monitored for differential pressure and radiation levels to determine when replacement is required. It is expected that approximately 60 days will be available to operate the system to establish water clarity and reduce radionuclide concentrations before the PCRV top head is removed.

1. System Recirculation to the PCRV:

The normal operational mode of the system will be with both trains processing PCRV water at a total flow rate of approximately 1000 gpm. Both clarifying pumps will take suction from the bottom of the PCRV (total flow rate - 1000 gpm) and will process the water through two parallel trains of filters. During system operation, water flows from the flooded PCRV through duplex strainers to the suction of the clarifying pumps. From the pumps, the water is processed through the filters before returning to the PCRV. A minimum side stream flow of 20% of the water flow rate will be processed through the demineralizers for removal of radionuclides, namely Co-60, Fe-55 and Cs-137. The filter trains will be set up in a series arrangement depending on the turbidity conditions in the PCRV. Initially, the system will be operated without filter elements installed. Samples will be taken to characterize particulates and determine the appropriate filter micron sizes. As water clarity dictates, filter elements and filter alignment will be changed as needed to support ongoing operating conditions. The clarified water will be returned to the top of the PCRV.

2. System Processing Via The Demineralizer:

A minimum sidestream flow of approximately 20% of the total flow (200 gpm) will be taken from the return line downstream of the filtration units and processed through demineralizers. The flow will be adjusted as required to maintain acceptable radiation levels on the work platform to minimize personnel exposure. Effluent from the demineralizer train can also be routed back to the system return lines for recirculation to the PCRV or, after sampling, routed to the effluent discharge connections as described in the following paragraph. Suitable provisions will also be provided for additional demineralizer capacity as required.

3. Discharge Via Radioactive Liquid Waste System (System 62):

All liquid waste from the Fort St. Vrain decommissioning will be routed through the existing radioactive liquid waste system (System 62) discharge line that was utilized during normal plant operations. Further details of this system are provided in Figure 2.2-23 and Section 2.2.3.10. Discharges will also be performed in compliance with the NPDES permit in effect at that time.

4. Sampling Operations:

Initially, any releases will be batch mode releases. Prior to liquid effluent discharge operations, representative samples obtained from the PCRV Shield Water System will be analyzed for principal radionuclides to ensure that the concentrations of radionuclides discharged to the environment do not exceed the values specified in 10 CFR 20. Samples will also be taken to verify maintenance of suitable water chemistry, water clarity and radionuclide concentrations. Sample locations will be included (see Figure 2.3-4) at the suction line from the PCRV, at the outlet of the filter trains, and at the outlet of the demineralizers.

5. Chemistry Control:

Two different chemistry control programs have been used in the PCRV shield water to minimize corrosion and maintain acceptable water clarity. Calgon LCS-20 was initially selected as the water treatment program for the PCRV shield water system. Calgon LCS-20 consists of a solution of sodium nitrite, sodium tetraborate and sodium hydroxide. The Calgon LCS-20 program used sodium hydroxide to maintain a basic pH (approximately 9 to 10), with nitrites to scavenge any oxygen in the shield water, thus minimizing corrosion of the carbon steel PCRV liner and liner insulation cover plates. Laboratory tests demonstrated that this water chemistry would be effective in inhibiting carbon steel corrosion. A polymerization compound was also added to resolve a colloidal suspension problem caused by entry of some concrete cutting slurry into the shield water. This water treatment proved adequate during removal of the graphite core components (including the core support blocks and posts), core barrel removal and removal of core support floor upper insulation.

Water clarity problems were encountered during core support floor upper insulation removal and steam generator inlet duct cutting operations, which were being performed by underwater divers, as described in Section 3.2.12. During diving operations in the PCRV, air used by the divers continuously bubbled to the surface, depleting nitrites which react with oxygen to form nitrates. This necessitated the addition of substantial amounts of chemicals to maintain the desired nitrite concentration, resulting in formation of a complex colloid that caused deterioration in water clarity.

The shield water chemistry control program was revised to end the LCS-20 water treatment, and permit addition of aluminum sulfate ("alum") to address the colloidal suspension problem and restore visibility. The revised shield water chemistry control program is more compatible with oxygenated water produced by underwater diving operations, using a neutral pH, without the addition of oxygen scavenging chemicals. The pH is maintained by periodic additions of sulfuric acid. A polymer(s) is added to flocculate materials, enhancing their removal. Laboratory tests determined that this water treatment program would be effective in re-establishing water clarity, with a pH of approximately 6.5 optimal, and that this program had significant benefits over use of other alternative water treatment programs. It is possible for some corrosion of carbon steel in the PCRV to occur using the second water chemistry control program. However, it is considered that corrosion will be minimal in the relatively short decommissioning time frame, and a small amount of rust in the water will not significantly reduce visibility.

Hydrogen peroxide will be added to the Shield Water system in batches, as necessary, to control biological fouling. A system will be provided for the purpose of adding chemicals to the PCRV shield water, consisting of two 100-gallon tanks and two chemical addition pumps. The system will initially be used to add chemicals during the initial fill of the PCRV. The system will then be used to batch feed chemicals as required until tritium levels are reduced to a level where continuous effluent discharge operations are acceptable. To support a continuous discharge operation, it will be necessary to continuously feed chemicals.

6. <u>System Interfaces:</u>

The PCRV Shield Water System will interface with and require support from the existing site systems:

(1) Portable Demineralized Water Makeup System

The PCRV Shield Water System will require a supply of demineralized water at a flow rate of up to 50 gpm at 100 psig. Demineralized water is required for system makeup, replacement of water removed by effluent discharge operations, chemical additions, and to replace evaporative losses. The demineralized water supplied to the PCRV Shield Water System must meet typical industry standards for oxygenated, deionized water. The demineralized water for the initial PCRV fill and for makeup due to subsequent effluent discharge operations will be from the portable demineralized water makeup system.

(2) Radioactive Liquid Waste System (System 62)

Tritium inventories will be initially controlled and subsequently reduced using effluent discharge operations, as necessary. The discharge from the PCRV Shield Water System will initially be to either the existing plant liquid waste holdup and monitoring tanks for processing and subsequent discharge or to the Reactor Building Sump (RBS), as described in Sections 2.2.3.10, 3.3.2.2 and 3.3.2.3. As tritium levels are reduced below the 10CFR20 MPC, the discharge may be permitted to flow directly from the PCRV to the radioactive liquid effluent discharge line, as discussed in Section 3.3.2.2.

(3) Electrical Power

Tie-ins to the site supplied power of 480 VAC 3-phase will be required. The skids will be pre-wired and transformers provided to facilitate interconnections.

(4) Compressed Air System

A source of dry compressed air at a nominal value of 90 psig is required to support system operations, particularly for dewatering the filters and demineralizers.

(5) Heating, Ventilation and Air Conditioning

The PCRV Shield Water System will be located within the Reactor Building to provide the required environmental conditions. No special or additional environmental conditions are required. The PCRV will require a temporary connection to the Reactor Building ventilation exhaust system to accommodate the displaced air during the initial filling of the PCRV.

C. System Maintenance

Methods for handling the replacement of radioactive strainers, filter elements and the change out of demineralizer resins will be designed for ease of replacement and will incorporate ALARA concepts, consistent with the recommendations of Regulatory Guide 8.8 (Ref. 10). These components will be shielded as necessary to minimize occupational radiation exposures.

The system will have sufficient interconnecting piping and isolation valves to allow repair or maintenance on a portion of the system while the remainder of the system continues in operation. In the unlikely event of a leak within the system, the entire system will be isolable from the PCRV, or that portion of the system with the leak will be isolated. The strainers, filters, and demineralizers will be designed to minimize exposure during maintenance. The strainers will be provided with inserts for ease of handling during replacement. The filters will be provided with vents and drains, and filter cartridges will be removed into shielded containers, if necessary, to minimize exposure. Filters and demineralizers will be shielded and strainers, filters and demineralizers provided with radiation monitors.

D. System Removal

The system will be used to support ongoing decommissioning operations. When the system is no longer required, it will be dismantled and treated as contaminated BOP equipment and piping. The system will be surveyed and decontaminated or disposed of as radioactive waste, as necessary.

2.3.3.7 PCRV Top Head Concrete and Liner Removal

The PCRV top head concrete and top carbon steel liner will be removed in two phases:

- Removal of 12 large sections of PCRV top head concrete.
- Removal of a final approximately 1 inch thick layer of activated concrete and the top of the carbon steel PCRV liner.

It is planned that the PCRV top head will be cut using diamond wire techniques, and removed in sections that can be handled with the Reactor Building crane. These sections will be cut so as to leave a thin horizontal layer of concrete above the PCRV liner. The remaining layer of activated concrete and liner will be removed by thermal methods (oxy-lance). This sequence is performed in this manner to prevent inadvertently breaching the PCRV liner and minimize exposure of equipment and personnel to radioactive material.

Prerequisite activities that are necessary to begin removal of the top head concrete include the following:

1. Detensioning and removal of selected tendons as discussed in Section 2.3.3.5.1.
2. Removal of selected highly radioactive components (control rod elements and metal clad reflector blocks) from the reactor core with the Fuel Handling Machine (FHM) as described in WBS Nos. 2.3.1.8.2 and 2.3.3.4 of the Decommissioning Cost Estimate and discussed in Sections 1.5.2 and 2.3.3.4.
3. Plugging the PCRV cooling tubes, top head penetrations, and tendon conduits as discussed in Section 2.3.3.6.1 and described in WBS No. 2.3.2.3 of the Decommissioning Cost Estimate.
4. Removal of helium purification equipment from PCRV top head wells using the Auxiliary Transfer Cask (ATC) as described in WBS No. 2.3.1.9 of the Decommissioning Cost Estimate and in Section 2.3.3.4.
5. Sealing of PCRV penetrations which are below the PCRV waterline and have had their instrumentation removed as discussed in Section 2.3.3.6.1.
6. Removal of interfering piping, instrumentation, and electrical components.
7. Flooding the PCRV prior to liner removal and acquiring access to the PCRV internal cavity as described in Section 2.3.3.6.

Plugging of the cooling tubes is a necessary requirement to mitigate the spread of contamination from the diamond wire cutting operation. The refueling, high temperature filter adsorber, and access penetrations in the top head in the path of the diamond wire saw will be plugged prior to diamond wire cutting operations to limit the amount of cutting slurry entering the PCRV cavity. Certain penetrations will be designated for use to draw air from the cavity and provide a negative pressure in the cavity. This air will be exhausted to the Reactor Building Ventilation System (System 73) for discharge and will be monitored for concentrations of tritium and other radionuclides.

The first phase, removal of 12 large sections of top head concrete, consists of the following major activities:

1. Seal the top head penetrations to prevent debris from entering the PCRV.
2. Set up the core drilling machines on the external wall of the PCRV to create six horizontal core drilled holes. (Figure 2.3-5).
3. Thread the diamond wire through the intersection points of the cored holes to make a loop to allow cutting of the concrete (Figure 2.3-6).
4. Make 12 vertical core drilled holes to intersect with the horizontal cored holes (Figure 2.3-7). The first two vertical holes will be inclined to facilitate removal of the first concrete segment.
5. Make the horizontal cut, then make the 12 vertical sectioning cuts and the 12 vertical back cuts using the diamond wire method (Figure 2.3-8).
6. Rig the sections for removal.

The concrete sections will be cut using the diamond wire cutting process, consisting of a wire with collars containing a diamond-matrix and made to length for each individual cut, and a hydraulic pulley drive system to circulate the wire. The diamond wire is routed to envelop the cut area and then returned to a drive wheel on the drive system. The wheel rotates and pulls the wire through the cut areas. Hydraulic cylinders control the tension of the wire. Once the cut is started, the tension and drive wheel speed are adjusted to optimize cutting efficiency.

The diamond wire cutting process will utilize appropriate radiological engineering controls to contain the cutting slurry and control airborne radioactivity. The diamond wire saw uses water as a coolant and lubricant for the cutting process. However, the coolant water is independent of the PCRV Shield Water System. A water collection system will collect the cutting slurry, decant the slurry and recycle the water. In addition, airborne and loose surface contamination control will be achieved by containing the diamond wire path and drive units(s) in a containment tent(s) served by HEPA ventilation.

Removal of 12 sections of tophead concrete, of which the lower portions of each section may be activated, will be accomplished utilizing the Reactor Building crane. This will leave a thin laye. .f activated concrete covering the PCRV liner. Specially engineered lifting attachments will be used to safely handle the heavy components, consistent with the requirements of 29 CFR 1926 and applicable ANSI standards. The concrete sections will be moved to a containment tent on the refueling floor where each large wedge-shaped segment will be cut into three approximately equal size smaller wedges using the diamond wire cutting methodology to make two horizontal cuts. Whereas the 12 large segments removed from the PCRV will each weigh approximately 110 tons, and will require use of the Reactor Building crane's 170 ton hoist for removal from the PCRV and transport to a containment tent, the smaller wedges will weigh approximately 37 tons. The smaller wedges will be handled with the 50 ton hoist of the Reactor Building crane, which has the capability to extend down to grade level in the Reactor Building truck bay. Section 3.4.3 evaluates the postulated drop of one of these wedges while it is being lowered down the truck bay, assuming the wedge was located adjacent to the top head liner while it was in the PCRV (most highly activated concrete).

The first phase activity will account for the majority of the effort that will be spent to remove the top head concrete and liner. Occupational exposure is expected to be negligible during the core boring and concrete cutting operations on the top head due to the relatively low radiation fields external to the PCRV. The second phase consists of removal of the final thin (approximately 1 inch thick) concrete layer and the top of the PCRV carbon steel liner. In this activity, the PCRV liner plate and adjoining concrete will be sectioned using a long handled (10 ft. initial length) thermal torch ("oxy-lance"). Concrete near the circumference of the steel liner will be sectioned with a circular concrete saw. The concrete saw is a hydraulic-driven, track-mounted, manually-operated saw. The concrete saw will be used to score the cut lines around the circumference. Jack hammers will then be used to chip away the concrete around the circumference to expose the carbon steel liner.

Prior to sectioning of the top head liner, the Decommissioning Rotary Work Platform (DRWP) will be installed on the ledge formed by the horizontal concrete cut (Figure 2.3-10). This platform will be a rotating platform with openings to provide access to all sections of the PCRV. The rotating platform is approximately 36 feet in diameter and rides on a circular track (rails) set 15 feet below the top of the PCRV. A stationary platform will fill in the voids between the circular rotating platform and the hexagonal opening cut in the PCRV. The Airborne Contamination Control System (ACCS) provides a flow path for air to move from the refueling floor down through the tool slots of the DRWP, then upward through exhaust ducts from the stationary work platform over the PCRV wall to the Reactor Building Exhaust system. The Reactor Building exhaust fans provide the air flow for the ACCS. ACCS ducting and shield water system piping pass through openings in the stationary platform. Openings or tool slots in the rotating platform provide access to in-vessel components. A wiper seal between the rotating and stationary platforms minimizes air flow between the two platforms so that most of the air flow is downward through the tool slots. The tool slot openings are sized to ensure a downward flow of air.

The ACCS provides for control of contamination during cutting of the PCRV top head liner. Smoke and airborne activity generated by the liner cutting operations will be pulled down through the DRWP tool slots, pass through a filter in the ACCS ducting, then through the Reactor Building ventilation exhaust system, including the System 73 HEPA filters and activity monitors, before being released from the Reactor Building. Therefore, workers performing the liner cutting operations will not be exposed to significant levels of airborne activity. Cutting operations will cease in the event of loss of ACCS ventilation flow.

The thermal torch (oxy-lance) will be used to cut through the thin layer of concrete remaining, the liner, insulation and cover plates to free sections for removal (see Figure 2.3-9). The layout and sequencing of cuts will take into consideration the structural stability of the liner during the disassembly process. The concrete/liner/insulation segments will be removed and packaged for disposal.

Of the two phases, the second phase represents the greatest potential for personnel exposure. The PCRV liner plate and the remaining thin layer of activated concrete will be uncovered as the segments of the top head concrete are removed. The PCRV liner plate is estimated to have radiation levels of up to 600 mRem/hr on contact. This estimated exposure rate is a conservative interpretation of information provided in the activation analysis for the bottom side of the cover plate, insulation, liner plate and activated concrete. Shielding for the workers will be utilized as appropriate for the close operations such as installing the saw tracks, operation of the saw and thermal cutting.

2.3.3.8 Dismantling PCRV Core Components

2.3.3.8.1 General Description - Graphite Blocks

Following the removal of the PCRV top head and installation of the work platform, PCRV core components will be removed. These activities will include the removal of various types of graphite blocks and other reactor internals within the core barrel down to the CSF. A listing of the types of graphite blocks that will be removed during decommissioning activities is identified in Table 2.3-1.

The top layers of blocks (i.e., metal-clad reflectors and some hex reflector blocks without Hastelloy cans) will be removed using the Fuel Handling Machine (FHM) and current plant methods to transfer the components from the PCRV to a shipping container, as discussed in Section 2.3.3.4 and WBS No. 2.3.1.8.2 of the Decommissioning Cost Estimate. Use of the FHM will provide the necessary shielding and containment while transferring components from the PCRV to the shipping containers with minimal personnel exposure. The FHM may be used to remove blocks other than those on the top layer of the core, including defueling elements and hexagonal graphite reflector blocks. However, use of the FHM for this purpose is dependent on its operability, and its availability has not been relied upon as the basis for removal of these components. The following sections provide a discussion for removal of these graphite blocks using manually operated tools.

2.3.3.8.2 General Arrangement of Work Area for Graphite Block Removal

The arrangement of the work area that will be typically used for removal of all types of graphite blocks is provided in Figure 2.3-10. The PCRV will have been filled with shield water to a level approximately 4 feet above the graphite blocks, but below the top of the PCRV liner. Suitable controls will be implemented to prevent water from splashing or the water level from approaching the exposed concrete above the top of the PCRV liner. These controls are necessary to prevent potential contamination of the concrete.

The Work Platform will have been installed on the ledge at the bottom of the hex opening in the PCRV. The Work Platform will be designed with the capability of rotating to provide access to all areas of the core. It will have two access openings to allow insertion and removal of tools and components, which will permit operations to proceed in parallel. A floor will be installed between the platform and the walls of the PCRV at the level of the Work Platform. There will be three jib cranes installed on the refueling floor level to service the access openings in the platform. The Reactor Building crane will also be available to service the platform and the remainder of the refueling floor area.

A ventilation system will be installed to provide control of airborne contamination, including tritium. Air will be drawn from the refueling floor to the Work Platform, down through the access openings in the platform, and then exhausted to the Reactor Building Ventilation (exhaust) System (System 73) for discharge. The discharges from the Reactor Building Ventilation System will be monitored in accordance with the FSV Offsite Dose Calculation Manual (ODCM) (Ref. 11). The airflow from uncontaminated areas to contaminated areas through the Work Platform will minimize personnel exposure to airborne contamination.

The area on the Work Platform will be quite large, approximately 43 feet across the corners of the hexagonal opening. This will provide the capability to move personnel on the Work Platform to a considerable distance away from an operation if a significant radiation field is encountered. During dismantlement operations, workers on the work platform will be protected from direct radiation and airborne contamination during removal of core components from the open PCRV. Radiation protection features include:

- Core dismantlement will be performed underwater, shielding workers and minimizing airborne particulate radioactivity.
- PCRV Shield Water System will strip soluble radionuclides from the shield water. Tritium inventory control is discussed in Section 3.3.2.3 of this plan.
- The ventilation system will ensure a positive downward flow of air over the workers. Exhaust ducts under the work platform will carry air through a filter, then to the existing plant ventilation system.
- Procedures and equipment for core dismantlement and operation of the work platform will be provided to minimize radiation exposure to workers.
- All work will be performed in accordance with approved Radiation Work Permits.

2.3.3.8.3 Special Considerations During Graphite Block Removal

The graphite block removal tasks represent a significant portion (22%) of the project's total person-Rem estimate. Due to the repetitive nature of the tasks, even small successful reduction measures will result in a significant savings of cumulative exposure. Although this process will benefit from additional future reviews and improvements, the following considerations are being taken to reduce personnel exposures for this series of jobs:

- Use of the PCRV Shield Water System.
- Use of the Work Platform will improve worker efficiency and safety.
- Utilization of a ventilation system to move evaporated tritium and other airborne contaminants away from the work area under and around the platform.
- Use of long handled tools and submerged staging areas to perform potential high exposure activities underwater.
- Use of temporary shielding as appropriate to maintain exposures ALARA.
- Installation of additional area radiation monitors (ARMS) with local alarm features to detect unexpected dose rates around the platform work areas.
- Audio-visual communication equipment to support remote surveillance of activities and equipment operations.
- Use of shielding bell(s).

2.3.3.8.4 Prerequisites for Graphite Block Removal

Prerequisite activities that are necessary to begin removal of the graphite blocks include the following:

1. Flooding the PCRV with shield water as described in Section 2.3.3.6.
2. Removal of the top head concrete and liner as discussed in Section 2.3.3.7.
3. Installing the PCRV work platform as discussed in Section 2.3.3.7.
4. Radiological survey of the work area and installation of temporary shielding if necessary for ALARA purposes.

2.3.3.8.5 General Graphite Block Removal Sequence

The following is the general sequence of operations that will be used for removal of all graphite blocks:

1. Removing
2. Staging
3. Loading
4. Transferring
5. Unloading
6. Packaging

Since this general sequence of operations will be used for the removal of all types of graphite blocks, the discussion of the six steps provided below are applicable to the removal of the defueling elements, hex reflectors (with and without hastelloy cans), the large permanent reflector blocks, side spacer blocks with boronated pins, and core support blocks and posts. Relative locations of the graphite blocks within the core area around the circumference of the core, are shown in Figure 2.3-11. The activities that are specific for one type of block are discussed in the subsection following this general discussion.

(1) Removing:

The blocks will be lifted from their position in the PCRV core area, and placed in a transfer basket that is below the surface of the water (see Figure 2.3-12). This will be accomplished using remotely engaged long handled tools (LHT's) attached to an overhead jib crane that is operated by personnel on the Work Platform. The workers will be working from the Work Platform that will be installed over the flooded PCRV. The tool for handling hex reflectors, defueling elements, and large permanent reflector blocks will be an expanding collet type similar to that used in the Fuel Handling Machine (FHM). The end of the tool will be inserted into the reverse counterbored hole in the top of the block with the end of the tool retracted. The end of the tool will then be expanded in the larger diameter in the lower portion of the hole and the block will be lifted. The side spacers will primarily be handled by attaching a lifting bail to the top of the block using the existing threaded holes in the graphite block. Some of the side spacers may be lifted by means of a bucket device. The hex shaped core support blocks will be handled using a three pronged gripper tool. The prongs will be inserted into three of the six holes in the core support block and then contracted to grapple the block. The irregular shaped perimeter floor blocks will be handled using a fork lift type tool since there are no holes in the block which can be used for grappling. The core support block posts will be handled by a tool which slips over the top of the post and is contracted to grip the post.

(2) Staging:

After removal from the PCRV core area and while still submerged, the blocks will be lifted and placed in a transfer basket located on a transfer stand attached to the work platform (see Figures 2.3-12). During this operation, the block will remain submerged underwater. The LHT will be disengaged and removed, leaving the block temporarily stored in the basket.

(3) Loading:

A shield bell will then be lowered into position over the transfer basket and seated on the work platform, as shown in Figure 2.3-12. A grappling tool will be lowered from the inside of the shielding bell, engage the transfer basket, and lift the basket into the shield bell. The shield bell guide pins in the transfer stand will provide the necessary alignment for engagement of the tool. The actual raising of the basket will be accomplished in a few minutes. After the basket has been raised into the shield bell, water contained in the basket and blocks will drain back into the PCRV through holes in the bottom and sides of the basket.

The shield bell will then be lifted to just above the floor of the Work Platform, and a catch pan built into the shield bell bottom (see Figure 2.3-13) will be installed under the shield bell to contain possible drippings of contaminated water. The shield bell bottom will be strong enough to retain the basket in the shield bell in the unlikely event that the grappling mechanism should fail. The bottom will also provide shielding at the bottom of the shield bell. During loading operations, radiation levels in the immediate vicinity of the shield bell will be closely monitored and personnel access to the affected area will be limited by administrative procedural controls.

The expected dose rates on the Work Platform, both with and without the shielding bell, are shown in Figure 2.3-15 for the large permanent side reflector block, in Figure 2.3-16 for the hex reflector blocks without hastelloy cans, in Figure 2.3-17 for the hex reflector blocks with hastelloy cans, and in Figure 2.3-18 for the side spacers without boronated pins.

(4) Transfer:

As the shield bell is moved from the work platform (using the Reactor Building crane) to the packaging area, nonessential personnel will be required to stay clear of the area to create a clear path for movement of the load.

(5) Unloading:

Unloading of the shield bell into a liner contained in a storage/shipping cask will be accomplished by removing the shield bell bottom and lowering the basket from the shield bell into the liner (see Figure 2.3-14). An alignment fixture will be used as necessary, to align the shield bell to lower the basket into the liner. While the basket is being unloaded, any water accumulated in the catch pan will be removed and disposed of or returned to the PCRV.

(6) Packaging:

After the basket has been loaded into the storage/shipping container, water absorbing material will be added in sufficient quantity to absorb any incidental water. The liner top will then be secured to the top of the liner. If the basket has been loaded into a shipping cask, the top will be installed and secured for shipment. If the basket has been placed in a storage cask, the top will be placed on the cask to provide shielding if necessary until ready for shipment. When ready for shipment, the top will be removed from the storage cask and the liner will be transferred to a shipping cask using an overhead crane. Any necessary shielding will be provided during the transfer of the liner from the storage cask to the shipping cask.

2.3.3.8.6 Description of Activities Specific to Block Type

A. Defueling Elements

Removal of the defueling elements not removed with the FHM will be handled as described in the six steps described in the previous subsection. The defueling elements are not activated and were uncontaminated when installed in the core. However, cross-contamination is expected to have occurred during reactor flooding and suitable contamination control procedures will be implemented to handle the defueling elements. Due to the very low radiation levels (< 1 mR/hr), it will not be necessary to stage the defueling elements and they may not be loaded into a shield bell for transfer to the shipping container.

B. Large Side Reflector Blocks

A typical large side reflector block is shown in Figure 2.3-19. There are 312 large side reflectors, ranging from approximately 522 to 2030 lbs. each, around the circumference of the core as shown in Figure 2.3-11. The initial task is to remove the 24 upper reflector keys, which must be accomplished in order to remove the side reflector blocks. The keys will be detached by removing the five nuts per key or by thermally cutting the keys. The large side reflector blocks will then be removed and processed using the general steps described above. The large side reflector blocks will be handled using a dual collet tool inserted into the reverse counterbored holes.

C. Hex Reflector Blocks Without Hastelloy Cans

Removal of the bottom, side and top hex reflector blocks without hastelloy cans not removed with the FHM will be handled as described in the eight steps described in the previous subsection. The position of the hex reflector blocks without hastelloy cans in the core is shown in Figure 2.3-11.

D. Hexagonal Graphite Blocks With Hastelloy Cans

Each Hastelloy can hex reflector block has 0.531-inch diameter holes to accommodate Hastelloy cans. There are 270 hex reflector blocks that contain 72 Hastelloy cans and 2 hex reflectors that contain only 4 Hastelloy cans each. The Hastelloy cans are 0.51 inches in diameter, approximately 8 inches long, and contain boronated graphite. The location of Hastelloy can hex reflector blocks in the PCRV is shown in Figure 2.3-11. Table 2.3-1 indicates that the hex reflector blocks with Hastelloy cans have one of the highest radiation levels (300 R/hr) of those irradiated components to be removed from the PCRV with manually operated tools.

Removal of the hex reflector blocks with Hastelloy cans not removed with the FHM will also be handled as described in the six steps described in the previous subsection. The Hastelloy cans in the hex reflector blocks are not expected to fall out of the block. Therefore, removal of the Hastelloy cans using a dumping or tipping operation will not be attempted. Removing the Hastelloy cans from the graphite blocks would require the use of some mechanical method (broaching, cutting, pressing, and/or crushing). After considering the methods for removing the Hastelloy cans and comparing them to the alternative of leaving the Hastelloy cans in the blocks, it was decided that the Hastelloy cans would be left in the blocks. Leaving the Hastelloy cans in the blocks eliminates the need for special equipment to remove the Hastelloy cans, simplifies the process of block removal, and will minimize personnel exposure.

E. Side Spacer Blocks With Boronated Pins

The side spacer blocks with boronated pins are shown in Figure 2.3-20. There are 1152 boronated side spacer blocks weighing approximately 100 - 150 lbs each. Their location in the core is shown in Figure 2.3-11 and dimensions of the pins are shown in Figure 2.3-21. Removal of the side spacer blocks will be handled as described in the six steps described in the previous section. Based on expected radiation levels, the pins and blocks may be shipped in the same type of shipping container. Therefore the pins will remain in the blocks during removal and shipping. The blocks will be lifted vertically from their position in the PCRV and placed in a vertical position in the transfer basket. The drainage holes in the basket will be sized to retain any pins which might fall out of the blocks during transfer from the PCRV to the shipping cask.

F. Core Support Blocks and Posts

There are 37 hex Core Support Blocks (CSB) and 24 irregular shaped perimeter CSBs. The CSBs are supported by 183 posts. Removal of the CSBs will be handled as described in the six steps described in the previous section.

The defueling blocks and hex reflectors must be removed prior to removal of the hex CSBs. Thermocouple assemblies run from the north west to south east through the CSBs. These assemblies may be removed by pulling them out of penetrations in the north west side of the PCRV or may be removed by personnel on the rotary work platform using long handled tools. The assemblies must be removed from a given CSB prior to its removal. In accordance with the Decommissioning Technical Specifications, blind flanges for the seven core outlet thermocouple penetrations may be removed, one at a time, during underwater removal of the thermocouple assemblies. During this time, PCRV shield water leakage will be prevented by redundant seals on the thermocouple removal tools. The assemblies will be placed in a container or basket under water for movement to the radwaste area for packaging and disposal in a manner similar to that described above.

The large side reflectors and side spacer blocks sit on top of the perimeter CSBs. The respective blocks and the lower reflector key must be removed prior to removal of a perimeter floor block. The 24 lower keys, which are made of Hastelloy X, will have estimated radiation levels of 10 R/hr at 1 meter. The lower reflector keys will be placed in a container or basket under water for movement to the radwaste area for packaging and disposal in a manner similar to that described above.

2.3.3.9 Removing the Core Barrel

The following prerequisites must be completed to begin dismantlement of the core barrel:

1. The PCRV top head concrete has been removed.
2. The PCRV has been flooded above the core barrel with shield water and water clarity has been established as discussed in Section 2.3.3.6.
3. Reactor core graphite blocks have been removed from the PCRV to a level low enough to permit the cutting of a core barrel section.

The core barrel and core barrel keys will be segmented underwater using remotely operated cutting equipment after the graphite core components are removed. However, if radiological surveys in the core barrel indicate that actual radiation and contamination levels are low, the PCRV water level will be progressively lowered and the core barrel and outer keys will be thermally cut above the water line. Exhaust hoods, powered by HEPA-filtered air handlers, will be positioned at the water surface or, if the cut is performed dry, in close proximity to the cut. These exhaust hoods will capture the majority of the fumes at their source. While cutting of the core barrel above the water line appears to have a schedule advantage over the underwater cutting, it will only be considered if it can be justified by an ALARA review.

If the removal of graphite core components is interrupted due to a shortage of shipping casks, work would commence cutting the core barrel underwater using remotely operated cutting equipment as the core barrel is exposed with the removal of successive layers of graphite core components. This is not expected to affect safety, occupational exposure or cause an undue schedule delay.

With either cutting alternative (i.e., underwater or above the water line), the major activities for removing the core barrel are as follows:

1. Rigging the core barrel sections for removal.
2. Making horizontal and vertical cuts in the core barrel to segment it into sections suitable for handling.

3. Removing the core barrel segments out of the PCRV.
4. Progressively removing the outer keys and thermocouple expansion joint assembly that is between the PCRV liner and the core barrel.

The cutting of the core barrel will be performed with the Work Platform in place. For underwater cutting, a mast or a remotely positioned track-mounted cutting tool will be operated from the Work Platform to make the vertical cuts around the core barrel. When the vertical cuts are complete, rigging will be attached to the core barrel segments prior to making the horizontal cuts. The horizontal cut will then be made and the core barrel segment removed. The jib cranes will be used to lift the segments to awaiting LSA boxes positioned adjacent to the opening on the work platform. The cut pattern will be predetermined based upon the size of LSA containers selected and the features of the remote cutting system. For disposal and cost estimating purposes, it was assumed that the segments were 7.5 feet high X 3.5 feet wide for a 4 foot X 8 foot LSA box and cutting was performed by a sequence of vertical cuts followed by horizontal cuts. However, if it is determined that larger pieces can be packaged, a reduction of time and exposure will be achieved. This process will continue down the entire length of the core barrel until approximately two feet of core barrel remains above the silica blocks. Removal of the lower portion of the core barrel will be coordinated with the removal of the silica insulation that is on top of the CSF. Underwater divers were used to remove most of the insulation on top of the CSF. Additionally, divers removed insulation from the inside of the lower portion of the core barrel, and removed the core barrel bottom remnant. Diving operations in the PCRV are discussed in Section 3.2.12.

The core barrel sections will be surveyed as they break the water to determine exposure rates before being handled. The segments are expected to have a contact dose rate of 40 mR/hr. Loose contamination is expected to be moderate (100,000 - 300,000 dpm/100 cm^2). Loose surface contamination from pieces removed from the water will be controlled by a combination of pressure washing, rinsing with clean water, wet vacuuming and swabbing. These measures will control the spread of contamination and minimize potential for airborne contamination.

2.3.3.10 Removal of the Core Support Floor (CSF)

The radiological conditions expected at this time are based on two sources, the PCRV cavity walls and the CSF. The cavity wall source consists of the fixed contamination on the wall and activated cover plate, insulation, liner plate and concrete. The dose rate from this source is estimated to be 30 mR/hr at any point within the PCRV. The CSF, as a radioactive source, consists of the surface contamination and the activated insulation on the top of the CSF, the activated CSF cladding plate, and the activated concrete. The dose rate contribution from the CSF is expected to be 400 mR/hr on contact with the insulation in place. Removal of the insulation from the top of the CSF, which contains various components and retaining devices made of Inconel, will reduce the exposure rate to approximately 360 mR/hr.

2.3.3.10.1 Removal of CSF Silica Blocks, Cover Plates and Insulation

The insulation on top of the CSF consisted of several layers of silica blocks and a layer of Kaowool (see Figure 3.1-30). The top layer of insulation consisted of dense cast fused silica blocks, approximately 3 inches thick. Under the top layer were two additional layers of lower density silica foam blocks, each layer also approximately 3 inches thick. The bottom silica layer rested on cover plates, under which was a layer of Kaowool insulation, approximately 2 3/4 inches thick. The cover plates were pressed down onto the Kaowool by anchor bolts attached to the 3/4-inch-thick CSF carbon steel casing. The cover plates and anchor bolts were carbon steel, except for a cover plate ring around each of the 12 steam generator inlet ducts, where the cover plates and anchor bolts were Inconel, to accommodate higher temperatures.

Removal of the CSF upper insulation commenced while the core reflector blocks and side spacer blocks were still being removed, before removal of the entire core barrel, in areas above the CSF where all reflector blocks, core support blocks and posts had already been removed. In these areas, insulation could be removed without destabilizing remaining columns of side reflector blocks and/or side spacer blocks. While the insulation was being removed, the PCRV shield water level was near the top of the PCRV liner, to minimize dose rates to personnel on the decommissioning rotary work platform (DRWP) from PCRV internal components. Most of the upper layer of dense fused silica blocks, and some of the silica foam blocks in the second layer, were removed with long handled tools. The remaining silica blocks were removed by divers, which was much more efficient. Diving operations in the PCRV are discussed in Section 3.2.12. The silica blocks around the 12 steam generator inlet ducts were glued together, and these were removed by divers using jackhammers to break up the blocks, as required. The divers loaded the insulation material removed from above the CSF into baskets, underwater, using shovels and vacuum equipment. The baskets were then lifted by the jib cranes to the DRWP and packaged in shipping containers. Divers and remote tooling were used to remove the Inconel sleeves and the alumina pads and dense cast fused silica discs within the sleeves that supported the core support post seats (Figure 3.1-30). Divers also removed the cover plates and bottom layer of Kaowool that was under the silica blocks, exposing the top CSF casing. The bottom several feet of the core barrel was removed in a manner similar to that described in Section 2.3.3.9, with the assistance of divers.

2.3.3.10.2 Removal of the Core Support Floor

The prerequisites necessary to begin removal of the CSF are as follows:

1. All core components, including the core support blocks and posts, have been removed from the PCRV.
2. The core barrel has been removed to within at least a few feet of the CSF.
3. If the PCRV shield water level has been lowered, then loose contamination will have been removed from, or stabilized on, the interior walls (insulation cover plate) of the PCRV, as practical.
4. The DRWP is removed prior to lifting the CSF to the upper PCRV area.

The CSF is a large disc approximately 29 feet in diameter by 5 feet thick and weighing 270 tons. Prior to lifting the CSF, three-inch thick steel plates for shielding were placed on top of the CSF, as discussed below. The total CSF weight, including the steel shield plates, the monorail spider assembly attached to the bottom of the CSF, and water absorbed in the kaowool insulation remaining on the CSF's bottom and sides, was estimated to be slightly less than 340 tons.

As noted in Section 2.2.2, the following items must be disconnected to allow removal of the CSF from the PCRV:

(1) 12 steam generator helium inlet ducts.
(2) 12 steel support columns, located near the CSF periphery, that are welded to the cladding plate and contain an array of cooling tubes and instrumentation tubes.

There is also a monorail spider consisting of twelve heavy structural steel beams on the bottom side of the CSF, that were used to position the steam generators during construction.

Since the existing Reactor Building crane has a capacity limit of 170 tons, the CSF will be jacked-up, sectioned, surveyed and removed in sections that can be lifted by the Reactor Building crane. Due to the tight clearance between the CSF and the PCRV cavity walls, it is necessary to raise the CSF to the upper PCRV region in order to provide access to the sides of the CSF for cutting and sectioning. The major activities that will be performed to cut and remove the CSF include the following:

1. Raising the CSF

Based on the results of radiation surveys, steel shield plates have been placed on top of the CSF while it is underwater to reduce radiation dose rates to acceptable levels when the CSF is lifted out of the water, since personnel will need to access the CSF for diamond wire cutting operations. Gaps between the steel shield plates should enable diamond wire cutting of the CSF to proceed without the need for cutting the steel shield plates. The added weight of these shield plates has been considered in the design of the CSF lifting system.

Prior to lifting the CSF, workers will require access to the areas immediately above and below the CSF inside the PCRV to perform the cutting of the steam generator ducts and CSF columns, and to attach the lifting cables to he CSF. These activities will be accomplished using either diving operations or workers in a man-basket suspended from a crane. The method selected will minimize the time that will be spent in the radiation field and minimize the resultant exposure.

The use of the man-basket will comply with the requirements of 29 CFR 1926.550(g) and will also be coordinated with the containments that will be in place during the various phases of the work. Controls associated with diving operations are described in Section 3.2.12.

Unless dose rates are determined to be significantly below those estimated, it will be necessary to disconnect the steam generator penetrations and the CSF support columns using underwater cutting. Therefore, PCRV water level will be maintained above the top of the CSF to provide adequate shielding during performance of these activities. Underwater cutting, in combination with exhaust hoods and respiratory protection, will provide a suitably safe environment to workers suspended in a man-basket. The DRWP may remain in place during severance of the steam generators and

CSF columns from the CSF. A localized containment may be used to prevent the spread of airborne contamination to other areas or workers.

Stress analysis of the CSF columns will be utilized to determine the number of columns required to support the CSF at this stage of dismantlement. The CSF lifting system will be installed on top of the PCRV, and will be supported by the PCRV. Heavy, wide-flanged beams will be placed across the top of the PCRV, with hydraulic jacks supported by the beams, as shown in Figure 2.3-22. Steel jackrods of the jacking system will be routed through four of the CSF steam generator penetrations and connected to lifting toggle beams positioned horizontally under the CSF. The CSF jacking system will be tested to 110% of the estimated weight of the CSF lift, prior to cutting the last several CSF columns. After the CSF has been cut free from the steel support columns, the CSF will be lifted and supported inside the PCRV. The CSF will be raised to the PCRV top head region using the hydraulic jacking system, which uses multiple steel jackrods attached both to the lifting toggle beams positioned under the CSF and to the jacking stations that have been established on top of the PCRV (see Figure 2.3-22). After raising the CSF, supports will be installed on the PCRV ledge where the PCRV top head was previously cut and removed, and the CSF will then be lowered onto these supports.

2. Segmenting the CSF

With the CSF supported in the top head area, the CSF will be cut in half by means of diamond wire cutting operations. Prior to initiation of cutting activities, a slurry catch pan will be installed below the CSF to minimize slurry entering the shield water. Radiological containments may also be constructed if determined necessary.

After the CSF is cut into two half-moon sections, each of these sections will be transferred by the Reactor Building crane from the PCRV cavity to a CSF segmenting area established on the refueling floor, for further segmenting. Segmenting the CSF will be performed using the diamond wire cutting operation. The primary work area for the segmenting activity will be around the perimeter of the CSF. This will keep the workers away from the top of the CSF which is the significant source of radiation exposure. The diamond wire cutting process is adequate to segment the CSF and the monorail spider located under the CSF, eliminating the need to remove this monorail separately. However, the monorail spider may be removed by divers prior to lifting and segmenting the CSF, if evaluation determines this method is more efficient. Individual segments of the CSF will be packaged for transport either on the refueling floor, or in the Reactor Building truck bay. These CSF segments will be lowered by the Reactor Building crane down the truck bay to a trailer, where the segments will be transferred to a temporary staging area and/or shipped to the disposal site.

2.3.3.11 Disassembling the PCRV Lower Plenum

During prior operations, the helium ducts connecting the CSF to the 12 steam generators were severed and the CSF was removed from the PCRV. Removal of the CSF will make lower plenum components accessible, including the steam generator primary assemblies, the helium diffusers, the CSF support columns, the lower floor, the lower plenum insulation and other miscellaneous components (see Figure 2.2-5). The following prerequisites should be completed prior to beginning removal of the helium circulator diffuser assemblies and the steam generator primary modules:

1. The helium circulator and steam generator secondary assemblies outside the PCRV have been removed.
2. The steam generators have been disconnected from the PCRV penetration flanges.
3. The CSF has been removed from the PCRV.

2.3.3.11.1 Steam Generator Primary Assemblies:

Each steam generator primary module is approximately 6 feet in diameter, 26 feet in height and weighs 65,000 pounds. The radiation source is primarily attributable to plateout contamination with a minor contribution due to activation. The uppermost portion of the primary steam generator nearest the outlet of the reactor is estimated to have a contact dose rate of 700 mR/hr. The lower portion of the primary steam generator is estimated to have a contact dose rate of 50 mR/hr. Localized hot spots on the generators are estimated to be up to 2 R/hr on contact.

1. Disconnecting the Steam Generator Primary Assemblies

The primary modules will have been structurally disconnected from the PCRV penetration flanges during the activity that removed the uncontaminated steam generator secondary assemblies. That task (Section 2.3.3.3.2) will have been accomplished in an uncontaminated environment with a low radiation field. The primary modules remain connected to the PCRV internals by the connection of the steam generator shrouds to the plenum floor in the lower portion of the PCRV and must be disconnected to allow the lifting of the primary modules from the PCRV. The separation of the steam generator primary modules from the plenum floor is the most complex task associated with the primary steam generator removal tasks. The steam generator helium inlet ducts, which were cut from the top and bottom of the CSF, will be removed from atop the steam generator primary assemblies.

An evaluation was performed to assess whether it was necessary to support the steam generator primary modules by rigging to the Reactor Building crane prior to unclamping or making the final severance cut. Each module has 36 closure studs, 1 3/4 inch diameter and 9 inches long, which penetrate the primary closure flange. While the nuts were previously removed from the bottom of the studs, enabling each module to be lifted straight out, these studs would provide substantial resistance to toppling, as would the lower plenum floor, a 1 inch thick steel plate 2 1/2 feet above the bottom mating flange. Calculations demonstrated that even without credit for the closure studs or lower plenum floor, a substantial force would be required at the top of a module to begin to topple it, and it would not continue to topple until the top was displaced more than 4 feet. The evaluation concluded that the modules were adequately supported without being rigged to the Reactor Building crane, and that a module would not topple after being disconnected from the lower plenum floor.

If necessary, the PCRV water level will be maintained above the top of the primary module to reduce radiation levels while it is being separated from the plenum floor. The steam generator will be disconnected from the plenum floor using either remotely-operated cutting equipment or locally-operated cutting equipment if conditions permit, by unclamping or cutting the clamp or lower seal at the connection of the steam generator shroud to the lower floor. Any remaining instrumentation or connections between the steam generators and the lower plenum will be severed remotely, or locally, if conditions permit.

Due to the expected radiation levels associated with the steam generator primary modules and the limited access in the area of the joint between the primary modules and the lower plenum floor, there is no simple means of making the separation. An alternative method for separating the steam generators from the plenum floor requiring similar precautions and effort is to cut the plenum floor around the attachment location, using a thermal cutting means. The method to be used will be based on an evaluation of the performance characteristics of both methods in the limited access in which it will be used.

Either of these methods would be utilized underwater to derive the benefit of the water for shielding the workers from radiation. Fumes from cutting and any potential airborne contamination will be collected by an exhaust hood at the surface of the water. However, if radiological surveys of the primary modules indicate that actual radiation and contamination levels are low, the PCRV water level may be lowered to obtain more direct control of the unclamping or severance cut during the separation of the primary module from the lower plenum floor. Lowering the PCRV water level will only be considered if it can be justified by an ALARA review.

2. Removing the Steam Generator From the PCRV

Following separation from the lower plenum floor, the steam generator will be removed from the PCRV cavity by the Reactor Building crane. Prior to, or as the primary modules are lifted from the PCRV, the outer shroud and tube outer surfaces will be washed down to remove as much contamination and cutting debris as possible, and will be allowed to drain as necessary over the PCRV cavity. Each of the twelve steam generator primary modules will be lowered into a cylindrical steel shipping container, whose inner diameter is slightly greater than the outer diameter of the module. The shipping containers will have side walls approximately 0.5 inch thick, which is calculated to provide adequate shielding. The shipping containers will be positioned on the horizontal ledge of the PCRV, where the decommissioning rotary work platform was previously situated. Each shipping container will be secured, oriented vertically, to ensure that the container remains in place during insertion of a steam generator primary module and during the time that the Reactor Building crane is disconnected from the module. During the movements of the modules, radiation protection personnel will ensure that distance is maintained between the workers and the source to keep exposures ALARA.

After a steam generator primary module has been lowered into its shipping container, absorbent material will be added as necessary, and the shipping container lid will be bolted onto a flange sealing the package. The weight of the steam generator primary module, shipping container and absorbent material is estimated to be less than 90,000 lbs., within the rating of the Reactor Building crane's 50 ton hook that will be used to lower it down the truck bay. Each loaded steam generator shipping container will be placed onto a trailer in the truck bay, then moved to an on-site location for temporary staging and/or shipped by tractor-trailer rig to the disposal site. Postulated drop of a steam generator primary module in the truck bay is evaluated in Section 3.4.10.

2.3.3.11.2 Helium Diffuser and Shutoff Valve Assemblies

The helium diffuser assemblies will have been detached from the PCRV penetrations after removal of components from within each penetration. The helium diffuser and shutoff valve assemblies will be removed using techniques similar to those described above for removal of the steam generator primary assemblies. Radiation levels on the helium diffuser assemblies are expected to be much lower than those experienced on the steam generator primary assemblies. The helium diffuser and shutoff valve assemblies will be disconnected by cutting the clamp at the connection of the diffuser to the lower floor. An alternative method for separating the helium diffuser and shutoff valve assemblies from the lower floor is to cut the lower floor around the attachment locations. The assemblies will be rigged to the Reactor Building crane, removed and transferred to the waste handling area for processing and disposal. Due to the lower radiation levels, no special shipping containers will be required.

2.3.3.11.3 Remaining Components

With the steam generators and helium diffusers and shutoff valves removed, all of the significant radiation sources in the PCRV will have been removed. This will allow the PCRV vessel to be totally drained. The work remaining in the lower plenum includes the removal of the CSF support columns, the lower floor and other miscellaneous lower floor area components, and the insulation and insulation cover plates on the PCRV liner and penetrations. These features will be removed utilizing hands-on tools and will be processed for disposal. The Kaowool insulation removed in this activity will most likely require removal of the absorbed water to assure compliance with shipping and disposal regulations. The removal of the absorbed water will initially be accomplished by pressing or squeezing the wet Kaowool, or other suitable drying techniques as required. All of the remaining components in the lower plenum will be removed and transferred to the waste handling area for processing and disposal.

During these final dismantling activities, the dose rates inside the PCRV lower plenum will be significantly lower than during previous operations since the largest radiation source, the steam generators, will have been removed. It is estimated that the general area radiation level will be low enough to allow activities to be performed in the lower plenum manually, which will increase productivity and still be ALARA acceptable.

2.3.3.12 Final Dismantling, Decontamination, and Cleanup Activities

The following activities are included in this task:

1. Cutting the PCRV sidewall insulation and liner.
2. Cutting and removing the activated concrete in the beltline region of the PCRV (See Figures 2.3-24 and 2.3-25).
3. Removal and/or decontamination of all remaining contaminated concrete.
4. Decontaminating the PCRV lower plenum liner.
5. Performing the final survey of the PCRV.
6. Demobilization and decontamination of the PCRV D/D tools and equipment.
7. Disposal of the PCRV Shield Water System.

The activated concrete will be removed in sectional units from the side walls of the PCRV, with the attached liner and both layers of thermal insulation intact as part of each unit. Diamond wire cutting has been selected as the method to remove the activated concrete sections.

Tendons which must be removed for access of the diamond wire will be detensioned and removed. Other tendons detensioned to relieve compressive stress on the kerf of the diamond wire cut will be left in place. This was discussed in Section 2.3.3.5.

Beltline concrete segments are being removed in two phases. The first phase involved removal of upper segments approximately 43 ft. long and a minimum of 31 inches thick, which included concrete that was at the elevation of the core and exposed to a relatively high neutron flux. Following removal of this beltline concrete around the inner circumference of the PCRV, samples of liner and concrete were taken from the remaining beltline area below the segments removed in phase 1. Analyses of these samples indicated that it would be necessary to remove additional beltline concrete in the lower plenum to achieve compliance with the Final Radiation Survey Plan (Section 4) release criteria. Thus, phase 2 beltline concrete removal will involve removal of concrete segments from the PCRV lower plenum approximately 26 ft. long and a minimum of 27 inches thick, as discussed below. Following removal of the second phase liner and beltline concrete segments, it is expected that activity concentrations and radiation levels from the PCRV liner and concrete that remains in place will be below the release criteria.

Circumferential tendons at the elevations of the horizontal cuts for both phases of beltline removal will be removed to provide a path for the diamond wire. The diamond wire cuts will be made in two steps from opposite directions, making a complete cut underneath the activated belt line concrete as shown in Figure 2.3-24. The horizontal cuts for the phase 1 beltline concrete removal are near the elevation where the bottom of the core support floor (CSF) was located. The phase 1 horizontal cuts were started after the CSF was removed from the PCRV, while the steam generator primary modules, helium circulator diffuser/shutoff valves, and CSF support columns were still in the PCRV. However, controls were in place to assure that the diamond wires did not penetrate the PCRV liner. Following removal of the steam generator primary modules and helium circulator diffuser/shutoff valves, and draining of the PCRV water level below the elevation of the phase 1 horizontal cuts, the cuts proceeded through the PCRV liner to completion. The CSF columns had been shortened and a liner placed on top of the columns to catch the diamond wire cutting slurry in the PCRV, to minimize further contamination of the lower plenum. The slurry was routed through a hole in the side of the PCRV to drums where the slurry was collected.

The inner ring of vertical PCRV tendon tubes, located 32 inches from the PCRV sidewall, are suitably positioned for removing the beltline activated concrete (see Figure 2.3-25). However, in the event that these tendon tubes prove to be unsuitable for the initiation of diamond wire cuts, new vertical holes will be core drilled.

The phase 1 vertical back (circumferential) cuts were made between each of the inner vertical prestressing tendon tubes, with the diamond wire threaded down one tendon tube, along the kerf of the horizontal cut, and up the adjacent tendon tube. This resulted in the phase 1 beltline concrete segments having a depth of at least 31 inches, from the inside of the steel liner. 14 vertical radial cuts were made by threading the diamond wire down an inner vertical tendon tube, along the kerf of the horizontal cut, and up the PCRV interior. The resulting 14 concrete segments were approximately 43 ft. long, 8 ft. wide and 31 to 32 inches thick, weighing approximately 75 tons. These segments were anchored at the top to support them until they could be connected to and lifted by the Reactor Building crane's 170 ton hook. The segments were transferred by the Reactor Building crane to a segmenting tent on the refueling floor where they were down-ended from a vertical to horizontal position. Each phase 1 segment was divided in the segmenting tent into 3 smaller segments using diamond wire cutting, packaged for shipping, lowered down the truck bay onto a trailer with the Reactor Building crane's 50 ton hook, transferred to a temporary staging area, then shipped to the disposal site.

The phase 2 horizontal cuts were made at an elevation approximately 6 ft. above the steel liner on the PCRV bottom head. Prior to making the horizontal cuts the PCRV had been completely drained of shield water, and the CSF columns removed. The vertical back cuts and vertical radial cuts will be made in a similar manner to that described above for the phase 1 cuts, except that the vertical back cuts will be made between every third inner vertical tendon tube rather than between each tube. There will be 14 vertical radial cuts and 14 phase 2 beltline segments, with each segment approximately 26 ft. long, 8 ft. wide and a minimum of 27 inches thick, each weighing approximately 40 tons. The phase 2 beltline segments will be lifted out of the PCRV and transferred to the segmenting tent on the refueling floor using the Reactor Building crane, and down-ended to a horizontal position in the segmenting tent. These segments will be cut approximately in half with diamond wire, and the remaining segments packaged and disposed of in the same manner as the phase 1 beltline segments.

The PCRV Shield Water System will be dismantled and decommissioned similar to balance of plant piping system. The system will be drained and the water processed as liquid waste as discussed in Section 3.3.2.2. The piping and components will be decontaminated, dismantled and packaged for disposal. The demineralizers will be the last items taken out of service. The demineralizer resins will be processed, packaged, and disposed of as radioactive waste. The demineralizers will be leased equipment, and will be decontaminated and packaged as necessary for return to the owner.

Following the removal of the activated beltline concrete, a final cleanup and decontamination of the entire PCRV cavity will be performed. Decontamination methods may include conventional wiping techniques, scabbling, scarifying, vacuum sand blast, or a hydrolaser method, depending on the degree to which the contamination is fixed on the surface. A survey of the PCRV will be conducted to verify that free release criteria has been met.

As dismantlement activities proceed, guardrails, covers, barricades, caps, etc., will be placed as appropriate consistent with industrial safety considerations. Upon completion of PCRV activities, a top head closure along with other appropriate penetration caps and guardrails will be installed in compliance with good industrial safety practices.

2.3.4 Contaminated BOP System Dismantlement and Decontamination

2.3.4.1 Introduction

The decontamination and dismantlement of contaminated or potentially contaminated balance of plant systems will be done by either (1) decontamination in place, (2) removal and decontamination, or (3) removal and disposal as radioactive waste. Systems which are contaminated or potentially contaminated above releasable limits requiring decontamination or dismantlement include the following:

1. System 13 - Fuel Handling Equipment
2. System 14 - Fuel Storage Facility
3. System 16 - Auxiliary Equipment
4. System 21 - Helium Circulator Auxiliary Equipment
5. System 23 - Helium Purification System
6. System 24 - Helium Storage System
7. System 46 - Reactor Plant Cooling Water System
8. System 47 - Purification Cooling Water System
9. System 61 - Decontamination System
10. System 62 - Radioactive Liquid Waste System
11. System 63 - Radioactive Gas Waste System
12. System 72 - Reactor Building Drain System
13. System 73 - Reactor Building Ventilation System
14. System 93 - Instrumentation & Controls

Contaminated balance of plant decommissioning is scheduled to coincide with fluctuations in critical path PCRV activities to level project manpower and to minimize competition for use of plant equipment.

In general, contaminated or potentially contaminated piping, components, structures, walls and ductwork will be dealt with in the following manner. Potentially contaminated items will be surveyed to determine acceptability for unrestricted free release or to determine the cleanup required for release. Verification that plant systems or structures may be released for unrestricted use will be provided by a comprehensive radiological assessment that provides statistically significant confidence levels for all plant systems. Since the plant systems cannot be altered for these detailed radiological surveys until the systems are no longer needed to meet NRC license requirements, the detailed surveys will be conducted during the implementation phase of the decommissioning project.

The results of these radiological assessments will be used to determine the workscope required for final removal of contaminated or potentially contaminated systems and components.

The piping and equipment removal experience gained at the Shippingport Station Decommissioning Project demonstrated that contaminated or potentially contaminated piping and components can be quickly removed by plasma-arc torch without compromising contamination controls when aided by portable HEPA filtered ventilation units. Because of the relatively small volume of contaminated piping at Fort St. Vrain, however, the cost and support requirements of plasma-arc torch operations (setup, torch maintenance, and HEPA-filter changeout) may dictate the use of other methods, such as portable band saws, hydraulic shears, and alternate thermal cutting processes such as oxy-acetylene. Piping will be cut into segments of approximately equal length. As piping is removed, the open ends will be covered and the piping segments will be placed in LSA containers (e.g., a 4-ft X 8-ft X 3-ft box). All piping, instrumentation, valves, and fittings can be packaged in this size of waste containers.

Piping will be removed by following controlled steps in accordance with project procedures and radiation work permits. System tagout procedures will be followed to de-energize pumps and other electrical equipment. Piping dead legs and traps will be drained of residual water. Piping released for removal will be positively marked before being turned over for dismantling. Contamination controls and waste containers will be set up to support dismantling operations. Contamination controls will include saddle tap valves for draining residual water, drip containments to capture metal filings, HEPA vacuums, anti-contamination clothing, and respirators, as identified by the radiation work permits. Contamination control enclosures may be built where necessary to prevent spread of contamination.

Any potentially contaminated piping that is embedded in concrete will be separated from the rest of the piping system near the face of the concrete structure and internally surveyed with a detector probe inserted into the pipe. Embedded pipe that satisfies the release criteria identified in Section 4.2 will be capped, tagged, and abandoned in place. Piping that does not meet the release criteria will be internally decontaminated by scrub brush or pipe-turning tools, such as a boiler tube cleaner, and internally wiped with moist rags until it meets the release criteria. If it is embedded near the surface, the pipe may be removed from the concrete with a concrete coring tool.

2.3.4.2 System 13 - Fuel Handling Equipment

The contaminated fuel handling equipment at Fort St. Vrain includes the fuel handling machine (FHM), five reactor isolation valves (Figure 2.2-11) and two refueling sleeves (Figure 2.2-12). However, the residual radiation and contamination levels for this equipment are low enough to allow manual disassembly on the operating floor.

During decommissioning activities, the following System 13 components will be used:

1. The FHM will be used to remove MCRBs from the PCRV and to place them into shipping containers or an interim storage area (such as the Fuel Storage Wells (FSW)). If placed into an interim storage area, the MCRBs will have to be removed by the FHM and placed directly into shipping containers. It may also be used to remove the Region Constraint Devices from their storage location and to place them into shipping containers. The ATC may be used for removal of helium purification components from the PCRV top head.

2. The Refueling Sleeves (H-1304) are required to guide the FHM arm while it is in the PCRV removing MCRBs, and when unloading MCRBs from the FHM into a shipping container in one port of the Hot Service Facility (HSF).

3. The Reactor Isolation Valves are required to adapt the FHM to the PCRV and to the facility (FSW, HSF, Fuel Loading Port, Regen Pit) into which it will be unloaded.

4. The System 13 Fuel Handling Purge System provides helium to operate internal actuators in the FHM. This helium is supplied from System 24, via System 13 piping and the FHM umbilical. If either System 24 or the System 13 Purge System is inoperable, pressurized air can be used to operate the actuators.

5. The HSF Adapter & Sleeve Assembly (Zook Sleeve) and the Modified Refueling Sleeve (S-1615) will only be used if the FHM is to interface with the HSF for loading shipping containers, and repair, maintenance, and interchange of grapple heads/manipulators.

6. The Fuel Loading Port and associated equipment may be used as an area for unloading MCRBs from the FHM.

7. The Core Servicing Manipulator (H-1603) and Core Service Vacuum Tool Assembly (H-1606) are FHM attachments which may be used for special operations within the PCRV and shipping containers.

8. The spare grapple (H-1301), spare mast camera (H-1601), and spare Z-drive pumps may be used in the event of a failure of the primary component, allowing repair without affecting MCRB removal operations.

9. The Shipping Cask Loading Seal Adapters (S-1604-250) may be used if the FSV shipping liner/casks are to be used.

The FHM will be externally surveyed and any loose contamination removed. It will then be disassembled into its component parts as necessary for decontamination or disposal. Sleeves will be attached as necessary to maintain a contamination envelope. The body of the FHM will be decontaminated and will be left on the refueling deck if release for unrestricted use limits are achieved. If further disassembly is required for release, the lead shot will be removed and the body will be segmented to segregate the contaminated material from the uncontaminated components. The contaminated scrap will be disposed of as described in Section 3.3 of this plan.

The reactor isolation valve exteriors will be surveyed and decontaminated by manual means. The valves will be removed from the operating floor and the lead shot removed. The shot is not expected to be contaminated or activated. The valve bodies will be disposed offsite according to Section 3.3 of this plan.

The refueling sleeves will be surveyed and decontaminated by manual means, then surveyed for release for unrestricted use. If they cannot be decontaminated, they will be disposed of as described in Section 3.3. The purge vacuum system will be removed and disposed of as described in Section 3.3.

2.3.4.3 <u>System 14 - Fuel Storage Facility</u>

The fuel storage facility consists of nine fuel storage wells (FSWs) constructed of carbon steel liners suspended in concrete pits. Each storage well is comprised of an outer cylinder, approximately 47.5 ft. in length and 54 inches diameter, and a steel inner liner. A schematic of the FSWs is shown in Figure 2.2-13. The inner liner is in the shape of a 4-lobed cloverleaf, with 5 cavities (one in the center) that were designed to store 5 columns of fuel and reflector blocks. The space between the inner liner and the outer cylinder was filled with slag, which served as a heat sink.

The FSWs were used for storing new and irradiated fuel during normal plant operation and may be used to temporarily store MCRBs or graphite reflector blocks during decommission. All fuel will have been removed from the Reactor Building prior to initiation of decommissioning activities. The actual contamination levels in the FSWs will be determined after the fuel has been permanently removed.

During decommissioning activities, the System 14 FSWs may be used as an interim storage area for MCRBs when they are removed from the PCRV with the FHM.

It was originally planned to decontaminate the inner liners of the fuel storage wells to below the release criteria, survey, and leave the clean wells in place. However, based on experience gained during the decontamination and survey of the fuel handling machine and other components, it was determined that due to the complex geometry of the FSW inner liners, decontamination in place and survey of the fuel storage wells was not feasible. The fuel storage wells were removed and disposed of as radioactive waste, as described below.

When the FSWs were no longer needed, each of the nine outer cylinders, which contain the inner liners, were removed and disposed of as radioactive waste. Before the wells were removed, lead shot surrounding the tops of the wells and the slag between the inner and outer liners was vacuumed out, surveyed, and disposed of as non-radioactive waste. The cooling water pipe to each of the wells was severed and plugged at the well end. Following removal of the slag, each of the steel wells (outer cylinder and inner liners) weighed approximately 55,000 lbs. The Reactor Building crane 50 ton hook was rigged to each steel well, and the well was lowered down the truck bay in a vertical position, downended to a horizontal orientation, and loaded onto a trailer for transfer to a temporary storage site within the restricted area. The top access plugs were fastened back on top of each of the nine wells. The steel wells were later transported offsite for disposal as radioactive waste, with the outer cylinders and top access plugs serving as the packaging for the contaminated inner liners. A false floor was installed over the vaults that housed the nine fuel storage wells, following their removal. Analysis determined that the effects of a postulated FSW drop accident in the Reactor Building truck bay would be enveloped by the dropping of a steam generator primary module, evaluated in Section 3.4.10, due to the much greater contamination inventory projected for the steam generator primary modules. The water cooling system piping remaining at the bottom of each pit will be surveyed.

2.3.4.4 System 16 - Auxiliary Equipment

The auxiliary equipment consists of the Auxiliary Transfer Cask (ATC, (Figure 2.2-12), ten Equipment Storage Wells (ESWs), (Figure 2.2-14), the Hot Service Facility (HSF), (Figure 2.2-15), and two shielding adapters (Figure 2.2-16).

The ATC was used to transfer the control rod drive assemblies, refueling sleeves and the shield plugs to and from the ESWs. The ten ESWs are carbon steel structures embedded in concrete. They were used to store the control rod drives and the refueling sleeves. The HSF is constructed of concrete and steel shielding and was used for inspection, repair, maintenance, testing and decontamination work.

Figure 2.2-16 is a general layout of the location of the various fuel handling and storage system components and associated auxiliary equipment on the refueling floor.

During decommissioning activities, the following System 16 components will be used:

1. The ATC may be used for removing and installing Refueling Sleeves into the PCRV during MCRB removal. The ATC may also be used for removing shield plugs from the ESWs, removing helium purification components, as well as retrieving and storing the Refueling Sleeves in the ESWs. If Region Constraint Devices (RCDs) or Control Rod Drives and Orificing Assemblies (CRDOAs) are stored in the ESWs at the beginning of decommissioning, the ATC may be used to remove them.

2. The ESWs may be used as a shielded interim storage area for activated/contaminated components, such as Refueling Sleeves, long-handled tools, and core components. The ESWs may contain CRDOAs and/or RCDs at the start of Decommissioning which will have to be removed during Decommissioning.

3. The Shield Adapters are required to adapt the ATC to the ESWs (for removal and storage of the Refueling Sleeves), to the PCRV (for insertion/removal of the Refueling Sleeves), or to the HSF, Regen Pit or Fuel Loading Port (for miscellaneous activities).

4. The HSF may be used as a multi-purpose area. Uses include: a general dismantlement/decontamination area, an area for holding shipping containers as they are loaded by the FHM or other means, and/or a shielded interim storage area for activated/contaminated components.

All the components of the ATC above the top base (32 ft. 11 in. above the operating floor) will be removed using the Reactor Building crane. A containment sleeve will be used to seal the contaminated ports in the cask and the hoist assembly floor as they are separated. The hoist cover and lift extension will then be lowered to the operating floor and disassembled within a contamination control envelope. The components will either be packaged and shipped for burial or to a licensed facility for processing and final disposition, or decontaminated and released for unrestricted use.

The remaining structure of the ATC will be decontaminated on site. The internal bore will be decontaminated using mechanical means such as sand blasting or hydrolasing to the criteria for release for unrestricted use. After internal decontamination, the crane will be used to lay the cask body over onto the floor for disassembly and decontamination of the bottom flange. When all surfaces meet the criteria for release for unrestricted use, it will be lifted by the crane and returned to storage on the operating floor.

The three shielding adapters will be decontaminated manually to the criteria for release for unrestricted use.

The ten ESWs are internally contaminated and will be decontaminated to the criteria for release for unrestricted use and abandoned in place. Contamination levels in the ESWs will be determined when they are no longer needed. After the plugs are removed, the ESWs will be vacuumed using a HEPA vacuum assembly similar to that for the FSWs. After vacuuming, the ESWs will be further cleaned using mechanical methods as necessary to reduce the contamination to the criteria for release for unrestricted use. After decontamination, the wells will be surveyed for release for unrestricted use. The top access plugs will be decontaminated, replaced and sealed.

Following final use of the HSF for decommissioning activities, all equipment(such as the manipulators and service platform sling) will be removed. This equipment may either be decontaminated onsite or packaged and shipped to a licensed facility for processing and final disposition. The walls, floor, ceiling and remaining structural components will then be decontaminated by sandblasting or hydrolasing. HEPA-filtered ventilation will be used to maintain a negative pressure in the HSF during decontaminations.

2.3.4.5 System 21 - Helium Circulator Auxiliaries

The auxiliary equipment for System 21 was used to provide a supply of high-pressure water for the helium circulator bearing lubrication and a supply of purified buffer helium to prevent in-leakage of bearing water into the primary coolant helium. The major equipment items include buffer helium recirculators, heat exchangers, filters, pumps, helium dryers, chemical injection components, containment tanks, and compressors (see Figure 2.2-17).

Following the defueling of the reactor, the helium circulator system will no longer be used. It has no function in the decommissioning of the facility.

Contamination has been detected within the helium circulator auxiliary equipment. Surveys will be performed during disassembly to determine the extent of the contamination.

2.3.4.6 System 23 - Helium Purification Auxiliaries

The helium purification auxiliary equipment consists of two trains and was used to assist in purification of the helium used as the primary reactor coolant. Most of the contaminated major equipment items are located in the PCRV top head and include filters, adsorbers, heat exchangers, dryers and piping (see Figure 2.2-18).

System 23 equipment is located in the top head in eight wells. This equipment will be surveyed and a determination made whether to remove the equipment with the ATC or by manual means. All System 23 equipment located in the PCRV top head will be disposed of as radioactive wastes. After the wells have been emptied, they will be surveyed and decontaminated as necessary.

The remainder of the helium purification system will be surveyed and decontaminated or removed as necessary.

2.3.4.7 System 24 - Helium Storage System

The primary purpose of the helium storage system was to provide for both storage and transfer of helium from the reactor vessel to the storage tanks. In addition, the helium storage system was used in testing the control rod reserve shutdown system and for various FHM purging operations. The primary equipment items include a helium transfer compressor, storage tanks, oil absorber, and high-pressure helium supply tanks (see Figure 2.2-19).

Following the defueling of the reactor, the helium storage system will no longer be used. It has no function in the decommissioning of the facility.

The helium storage compressors have been found to be contaminated. This system, including the 108 helium storage bottles, will be surveyed during disassembly to determine the extent of the contamination. The results of this survey will be used to determine decontamination or disposal requirements for specific components.

2.3.4.8 System 46 - Reactor Plant Cooling Water System

The reactor plant cooling water system (see Figure 2.2-20) provided cooling water for process heat removal from all auxiliary equipment in the reactor plant. Three loops were provided that formed the PCRV circuit (liner cooling tubes), the PCRV auxiliary circuit (closed loop for various systems/components) and the service water circuit (open loop for various systems/components). The major equipment items include surge tanks, pumps, demineralizers, filters, heat exchangers, chemical injection (tank and pump) and recondenser chiller.

Portions of the system external to the PCRV have been found to be contaminated. The system will be surveyed during disassembly to determine the extent of the contamination. Cleanup or disposal requirements will be determined based on survey results.

The reactor plant cooling water system loop supplying the PCRV will not be used for cooling of plant components during decommissioning. It will be disconnected and isolated from the PCRV and from the FSWs before decommissioning of those systems occurs. Fifty percent of the PCRV cooling tubes will be cut and surveyed.

2.3.4.9 System 47 - Purification Cooling Water System

The purification cooling water system (two loops) provided cooling water to the helium purification system heat exchangers. The major components are pumps, expansion tanks, exchangers and associated piping (see Figure 2.2-21).

This cooling water system has been found to be contaminated. The system will be surveyed during disassembly to determine the extent of the contamination. Cleanup or disposal requirements will be determined based on survey results.

The purification cooling water system will be isolated from the helium purification system before it is decommissioned. The purification cooling water system has no other use during the decommissioning.

2.3.4.10 System 61 - Decontamination System

The decontamination system consists of a water heater, a drying air heater, a filter, pumps, a solution tank and a chemical injection system (see Figure 2.2-22).

The decontamination system will be surveyed to determine the extent and location of radioactive contamination following final system use. The decontamination system components are small, and will be removed and packaged in LSA shipping containers along with other contaminated components and piping. The decontamination solution tank may be removed in one piece for shipment, or segmented and packaged in LSA shipping containers.

2.3.4.11 System 62 - Radioactive Liquid Waste System

The major equipment items in the Radioactive Liquid Waste System include a waste sump (1000 gallon tank), pumps, filters, two 3000-gallon receiver tanks, two demineralizers, and a 3000-gallon waste monitor tank (see Figure 2.2-23).

The liquid waste system is expected to be used for its original function during decommissioning operations. Therefore, it will be one of the last systems to be decommissioned.

During decommissioning, System 62 piping and components can be used for collection, monitoring, and dispositioning of liquid effluent generated by decommissioning activities, mainly by the PCRV Shielding Water System. Additionally, fluids may be processed from decommissioning activities which originate from the Helium Regeneration Pit Drains, Liquid Drain Tank (System 63), Reactor Vent and Drain System Standpipe M-5 (System 72), Reactor Building Sump and Sump Pump (System 72), and the Reactor Plant Exhaust Filter housing drains (System 73). The latter sources are those normally encountered during reactor operation and shall be processed by established methods. All sources, including the PCRV shielding water, require the necessary piping and components that will be utilized to remain in service until no longer needed.

In addition, effluent from the PCRV Shielding Water System will require the installation of a connection between the discharge of the PCRV Shielding Water System transfer pump and System 62 piping, and slight modification of valving to utilize System 62 as desired. This will permit pumping of shielding water directly into either of the Liquid Waste Receivers. Also, valves will be installed to permit shielding water to be pumped directly into the Liquid Waste Monitor Tank without travelling through the Liquid Waste Transfer Pumps, the Liquid Waste Demineralizers, and long lengths of piping.

A characterization survey of the radioactive liquid waste system will be performed when the system is no longer needed to determine the extent and location of radioactive contamination.

The contaminated radioactive liquid waste system components are small and include: the two liquid transfer pumps, the two liquid waste sump pumps, the two liquid waste filters, and the two liquid waste demineralizers. The liquid waste monitor tank and the two liquid waste receivers may be decontaminated and abandoned in place, shipped as one piece containers, or segmented and packaged in LSA shipping containers depending on the extent and location of radioactive contamination. The liquid waste sump will be considered for either (1) decontamination to free release levels and abandonment, or (2) segmentation and packaging as LSA waste.

2.3.4.12 System 63 - Radioactive Gas Waste System

The major equipment items in this system include pre-filters, filters, exhaust blowers, tanks (vacuum, surge, and drain), and compressor (see Figure 2.2-24).

Following final use of the system, the radioactive gas waste system will be surveyed to determine the extent and location of radioactive contamination. The large components such as the two gas waste surge tanks and the gas waste vacuum tank may be decontaminated and abandoned in place, shipped as one-piece units, or segmented for packaging and shipping. The other components are small enough to be shipped in LSA shipping containers with other contaminated piping.

Decontamination of these systems will be by manual mechanical methods depending on the levels of contamination found during the characterization survey. The system will not be used in the decommissioning of the plant.

2.3.4.13 System 72 - Reactor Building Drain System

The major equipment items include drain tanks, sump, pumps, piping and filters (see Figure 2.2-25). Two gravity flow drains are provided to direct drainage from the Reactor Building equipment, piping and floor drains to either the radioactive liquid waste sump for potentially contaminated liquids or the Reactor Building sump for all other liquids. The drain system will continue to be used for its original function during much of the decommissioning work and will be one of the last systems to be decommissioned.

When no longer required to remain operational, the system will be surveyed, and a decontamination and decommissioning decision will then be made. Contaminated piping or components will be either removed and shipped in LSA containers, or decontaminated to the criteria for release for unrestricted use and left in place.

The portion of this system that drains to the Reactor Building sump is not expected to be contaminated. The portion of the system that drains to the radioactive liquid waste sump is expected to be contaminated and is included in dismantlement and removal plans.

2.3.4.14 System 73 - Reactor Building Ventilation

The Reactor Building HVAC system services various areas of the Reactor Building with heated or cooled air. All ventilation air, whether outdoor or recirculated, is filtered before distribution. In addition, the reactor plant HVAC system maintains building differential pressure control. As shown in Figure 2.2-26, this system consists of several air handling units and filters. The only part of the system considered to contain possible contamination is the Reactor Building exhaust filters, HSF vent, and the analytical room vent.

The reactor plant exhaust filters are composed of banks of moisture separators and HEPA filters.

During decommissioning, System 73 will be used to maintain the reactor building pressure subatmospheric for selected decommissioning operations as required by the Decommissioning Technical Specifications. The system consists of three trains, one of which will normally be in continuous operation. One train is sufficient to maintain the reactor building subatmospheric. The reactor plant exhaust filters will be periodically monitored and filter media changed, as necessary. Filter change out will be based on excessive pressure drop across the HEPA filters, or excessive leakage, as required by the Decommissioning Technical Specifications.

System 73 will also be used to support the Airborne Contamination Control System (ACCS) which draws air from under the Rotary Work Platform. The ACCS is a temporary addition that will be used during decommissioning and ties into existing System 73 ductwork. The ACCS consists of several ducts that will pull air from under the platform. The ducts tie together forming a single duct that is routed to two roughing filters, in parallel, to remove particulate material. After the roughing filters the ACCS ducting ties into the existing System 73 ducting, upstream of two of the three Reactor Building exhaust fans. Dampers are installed so that one or both of these two Reactor Building exhaust fans can be lined up to take a suction from the plenum under the Rotary Work Platform (ACCS) or from the Reactor Building (normal suction). If one HEPA filter/exhaust fan train in the Reactor Building exhaust system is not available due to changing of the filter media or other maintenance, that train can be isolated and the air flow from the ACCS diverted to the alternate HEPA filter/exhaust fan train that is connected with the ACCS. The ACCS will be removed at the end of decommissioning.

The ventilation system has been found to be contaminated. This system will be maintained during decommissioning to provide ventilation for decommissioning operations. Near the completion of decommissioning activities, surveys will be taken to determine the final disposition of the system.

2.3.4.15 System 93 - Instrumentation and Controls

The instruments and tubing to be removed or decontaminated originate at PCRV penetrations. These include thermometer penetrations, process and moisture instruments, helium circulator instruments, and helium vent piping.

Moisture monitors will be removed during dismantling the PCRV. All other instrument interfaces to contaminated or potentially contaminated systems will be addressed when the respective system is decommissioned. Those System 93 components will be either removed or verified to be below the limits for release for unrestricted use. All systems are scheduled for inclusion in the characterization survey. Contaminated system components will be decontaminated or disposed of as LSA waste.

2.3.5 Decommissioning Schedule

The individual tasks making up the decommissioning effort have been delineated using a work breakdown structure (WBS) approach. Figure 2.3-26 is a schedule of the major decommissioning tasks which includes PCRV and balance of plant system dismantling and decontamination, and site decommissioning. This schedule is used as the top-level view of the project milestones and detailed schedules. Throughout the project, dismantling the PCRV is the critical path activity, with the BOP dismantling activities scheduled to coincide with periods of reduced PCRV efforts as a means of workload leveling. During the planning phase, work will be directed toward characterizing the site, preparing the decommissioning plan, and planning and writing the procedures and specifications for the implementation phase.

The major activities and programs to be developed during the planning phase include:

1. Initial site characterization
2. Decommissioning planning
3. Work specifications and procedures
4. Quality assurance plan
5. Radiation protection program
6. Waste management plan
7. Project performance and control

The schedule depicts the planning phase occurring over an 18 month period, and the actual dismantlement and decontamination activity at the site occurring over a 39 month period.

INTENTIONALLY LEFT BLANK

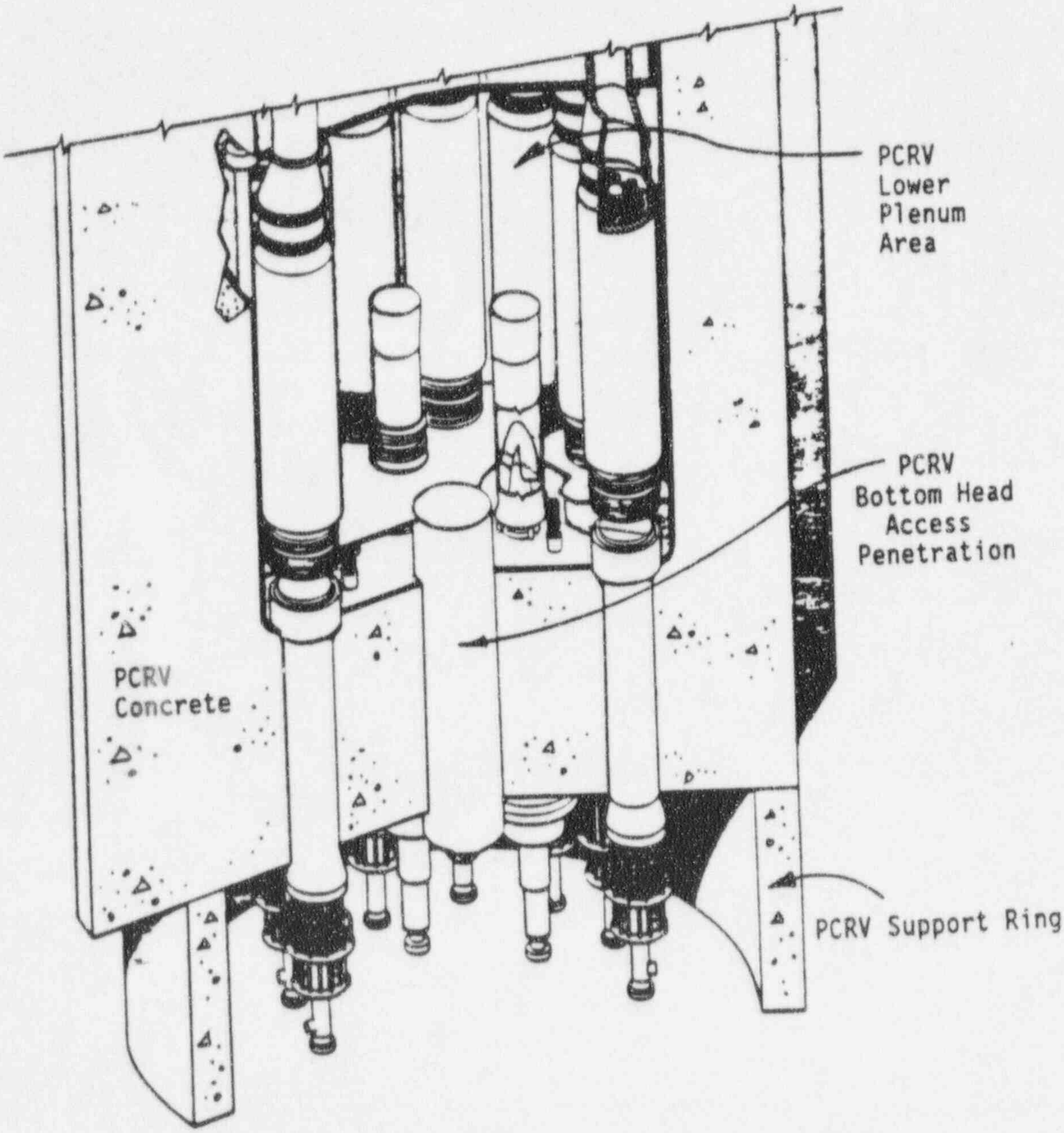

Figure 2.3-3 PCRV Lower Plenum

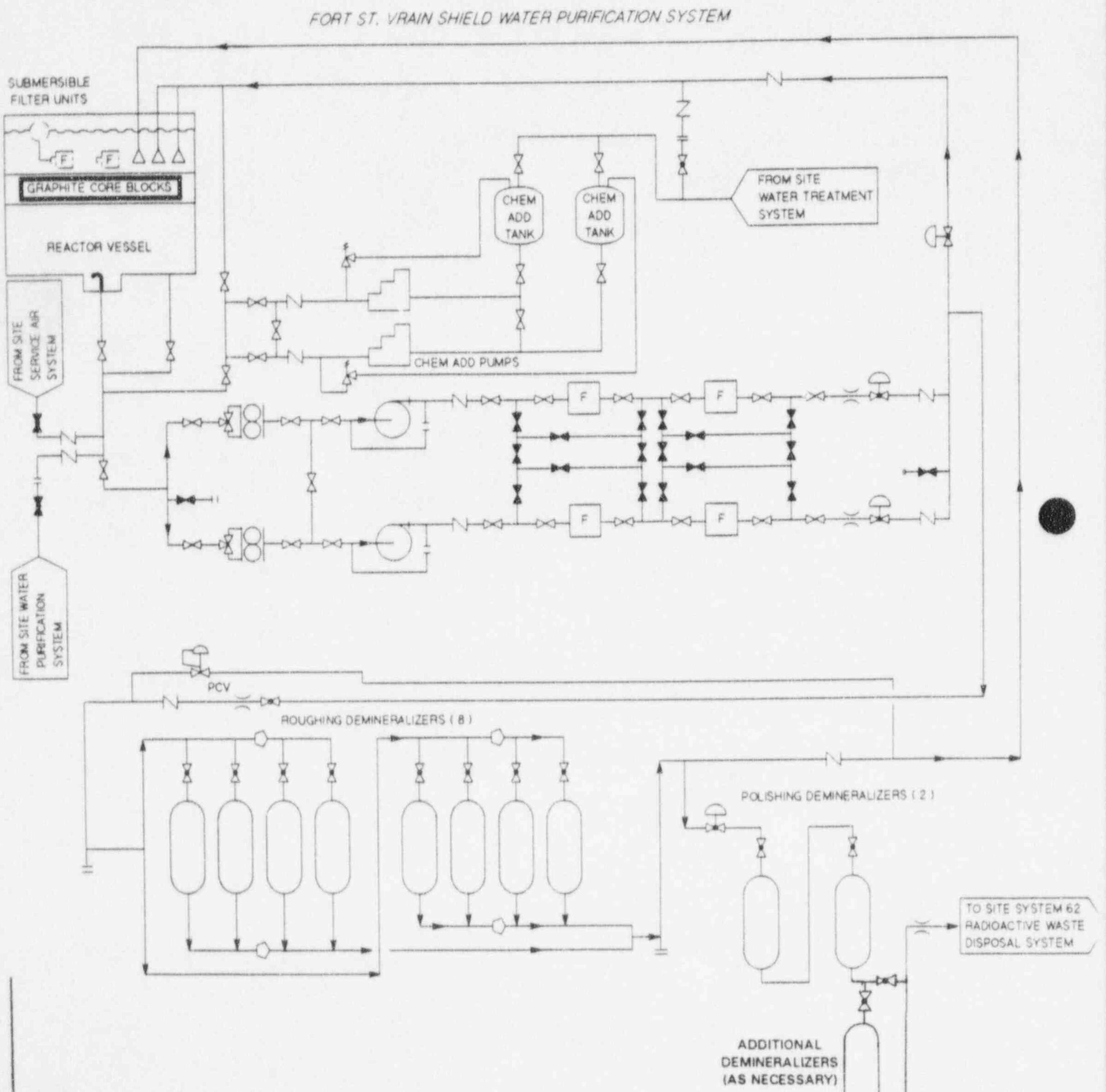

Figure 2.3-4 PCRV Shield Water System

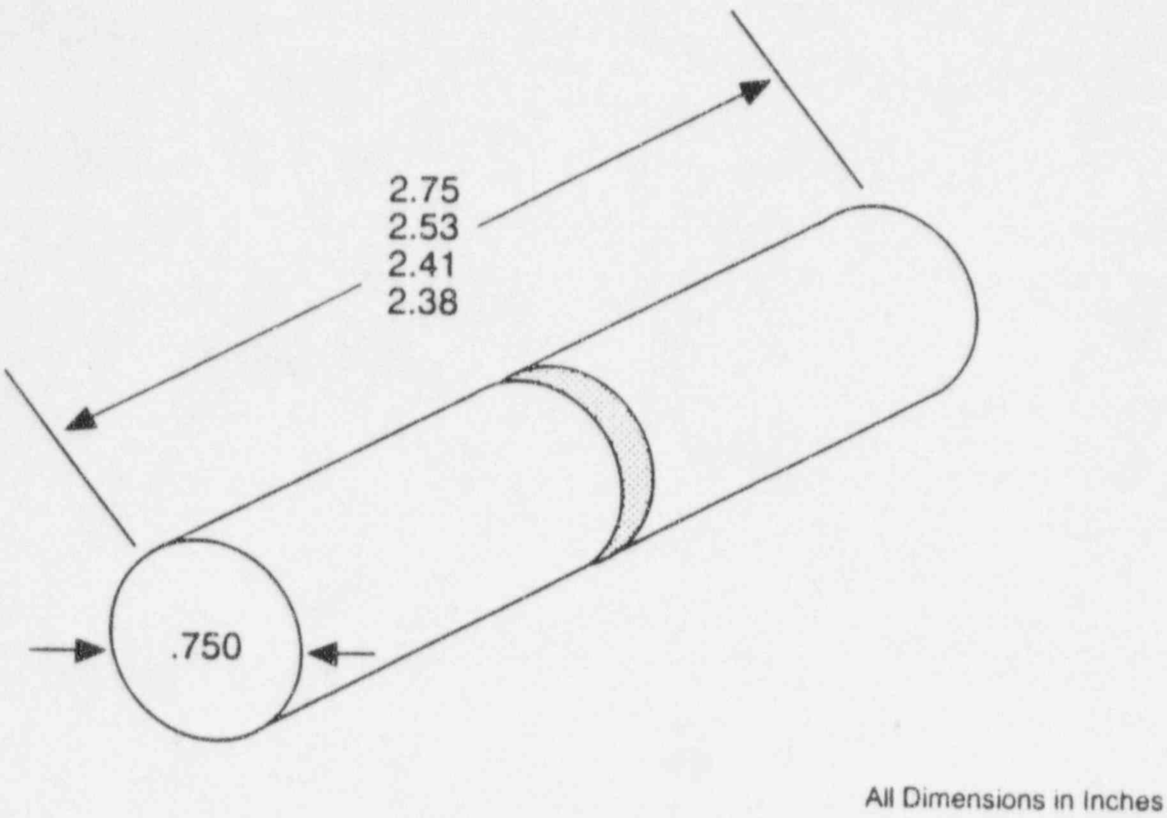

Figure 2.3-21 Boronated Rod

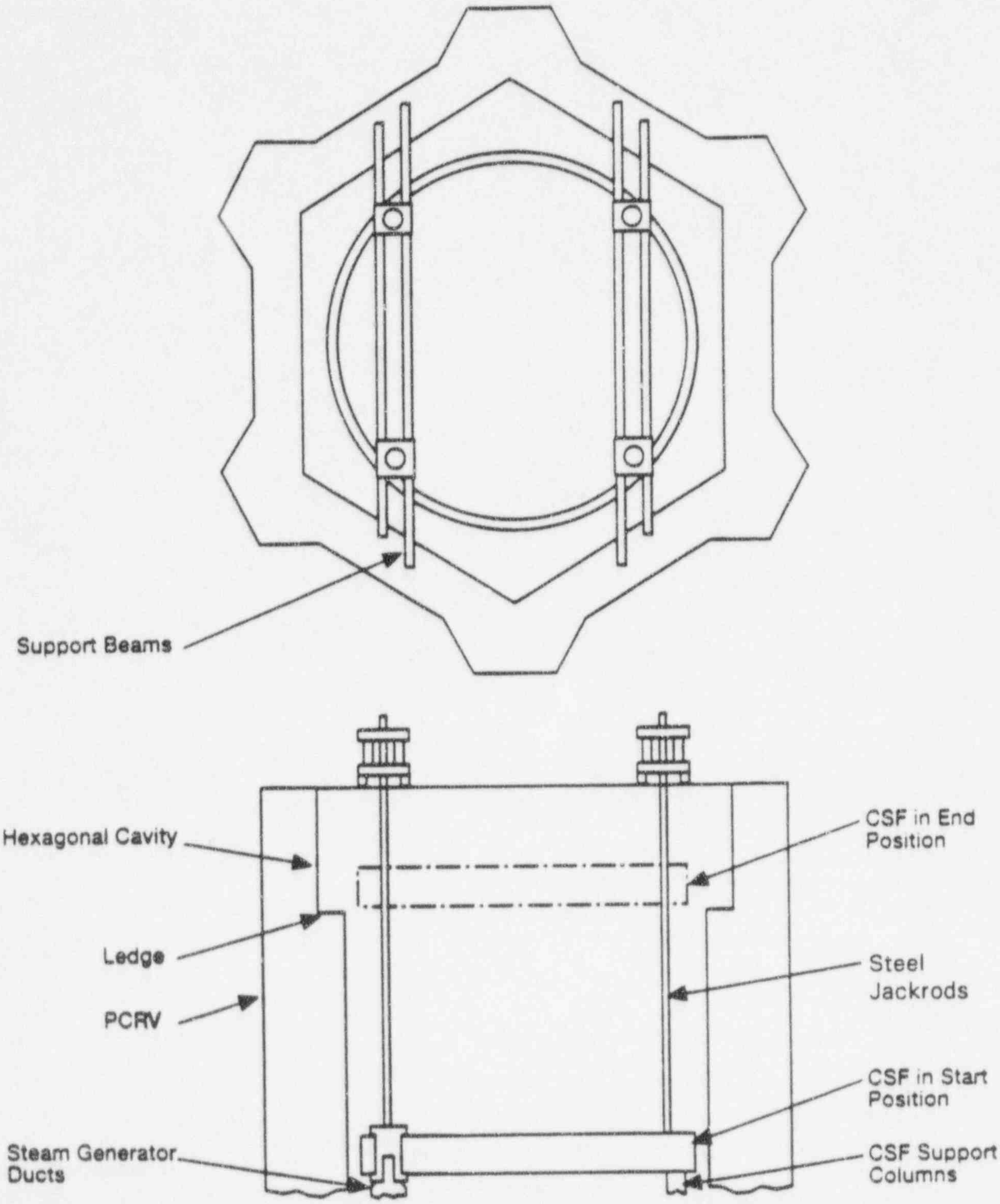

Figure 2.3-22 CSF Four Point Jacking System

2.4 DECOMMISSIONING ORGANIZATION AND RESPONSIBILITIES

2.4.1 PSC Commitment

Public Service Company of Colorado (PSC) is fully committed to compliance with the existing license and applicable regulatory requirements during all phases of the Fort St. Vrain decommissioning. PSC's commitment to the safe decommissioning of the facility will be accomplished with diligence and quality service. Corporate principles, policies, and goals will be followed to ensure performance excellence, management competence, and high standards in every facet of the decommissioning project.

2.4.2 PSC Decommissioning Organization and Functions

The PSC Decommissioning staff for the Fort St. Vrain Nuclear Generating Station and the interface with the Westinghouse team is shown in Figure 2.4-1. The manpower level is approximately 60 people including the key staff members and all performance level people. Overall onsite control and responsibility for all decommissioning activities for both PSC and contractor personnel rests with the PSC Decommissioning Program Director. Within the PSC organization, four main groups report to the Decommissioning Program Director. The groups consist of the Project Assurance Manager, Radiation Protection Manager, Operations Manager, and Decommissioning Engineering Manager. Contractor reporting requirements and lines of authority are identified in Section 2.5. The PSC Decommissioning Program Director interfaces directly with the Westinghouse Project Director for decommissioning activities.

During the decommissioning process, PSC will retain responsibility for the 10 CFR 50 license and therefore will maintain the following responsibilities:

1. Overall management oversight of all decommissioning project activities.
2. Sole point of contact with all regulatory agencies within the Federal, State and local governments.
3. Overall responsibility for all licensing activities.
4. Overall management of those plans and programs required to comply with licensing requirements, including: access control, radiation protection, Quality Assurance, maintenance and operation of existing plant systems, training and configuration management.

The key decommissioning staff members perform the functions described in the following subsections.

2.4.3 PSC Vice-President

The Vice President responsible for nuclear activities, provides leadership and direction at the corporate executive level and has the authority and responsibility to ensure that all activities to carry out decommissioning are performed safely and within applicable regulations.

The Vice President responsible for nuclear activities, shall have a minimum of fifteen years executive experience in waste management, decontamination and decommissioning, and nuclear operations. The Vice President must have a formal education in an engineering or physical science field. Knowledge in the areas of regulation and compliance, decommissioning techniques, and applied radiation protection programs are required. In addition, a background of knowledge with respect to NRC and DOE is desirable.

2.4.4 Decommissioning Program Director

The Decommissioning Program Director is directly responsible to the Vice President responsible for nuclear activities. The Decommissioning Program Director coordinates and oversees all decommissioning activities. This position provides direction to the support groups to ensure radiological and industrial safety, compliance with regulatory requirements, cost-effectiveness, and interfaces for PSC Labor Relations of the decommissioning project. The Westinghouse Team Project Director will report to this position.

The Decommissioning Program Director shall have a minimum of ten years responsible plant experience with formal education in an engineering or physical science field. A significant technical background to have good working knowledge of plant principles of operation, maintenance and engineering principles. Additional knowledge in the areas of regulation and compliance, decommissioning techniques and applied radiation protection programs are required.

2.4.5 Project Assurance Manager

The Project Assurance Manager is responsible for Quality Assurance oversight (including QA reviews, audits and monitoring (surveillance) activities), licensing and regulatory compliance, and overall industrial safety. To ensure the independence of the QA function, this position reports to the Vice President responsible for nuclear activities on quality assurance matters (as indicated by the dotted line in Figure 2.4-1). The Project Assurance Manager reports directly to the Decommissioning Program Director for administrative direction and implementation of the Quality Assurance Plan, coordination and direction for licensing activities, and coordination and direction of the industrial safety program (as indicated by the solid line in Figure 2.4-1).

The Project Assurance Manager shall have a minimum of five years experience in a responsible position that includes coordination, direction and supervision of personnel, a formal education in an engineering or physical science field, and a working knowledge and understanding of nuclear plant design and operation and construction practices is required. A balance of experience in quality assurance related activities and in regulatory/compliance activities is preferred.

2.4.6 Radiation Protection Manager

The Radiation Protection Manager is responsible for Radiation Protection, ALARA, Access Control and Training programs. This position is also responsible for managing support areas of emergency planning and PSC training.

The Radiation Protection Manager has the overall responsibility for the Radiation Protection Program described in Section 3.2 of this plan. The Radiation Protection Manager represents the formal line of communication and authority between Fort St. Vrain and the Westinghouse team for radiation protection matters related to decommissioning. This individual will be responsible for ensuring that the Radiation Protection Program and procedures meet the goals and standards established by PSC management and the governing regulatory agencies. The Radiation Protection Manager will also be directly responsible for the Radiological Environmental Monitoring Program and the Decommissioning Emergency Response Plan. This individual will meet the qualifications contained in NRC Regulatory Guide 1.8 "Qualification and Training of Personnel for Nuclear Power Plants" (Ref. 12). The duties and responsibilities of the Radiation Protection Manager with respect to the Radiation Protection Program are described in further detail in Section 3.2.3 of this plan.

The Radiation Protection Manager shall have a minimum of five years experience in a responsible position that includes coordination, direction and supervision of personnel. A formal education in engineering or the physical sciences or the equivalent experience in a science or engineering subject is preferred. Formal training in radiation protection is required.

2.4.7 Operations Manager

The Operations Manager is responsible for the overall conduct and management of operations and maintenance functions. These responsibilities include system operations, testing and surveillances, system maintenance, lay-up and turnover.

The Operations Manager shall have a minimum of eight years of responsible power plant experience of which five must be nuclear power plant experience, including coordination, direction and supervision of personnel. A thorough working knowledge and understanding of plant design and operation and maintenance functions (including instrumentation and control maintenance activities) are required.

2.4.8 Decommissioning Engineering Manager

The Decommissioning Engineering Manager is responsible for the administrative and technical functions of the decommissioning project. Responsible areas include management of contract work, technical assistance, evaluation and approval of contract changes, reviewing and reporting on decommissioning performance against base line schedules and budgets, legal decommissioning contract administration, recommending changes in decommissioning project direction, developing project cost estimates and issuing regular performance indicator reports. This position is also responsible for the general oversight of field work, preparation of engineering evaluations, coordination with operations, records control and retention, and managing PSC materials and facilities.

The Decommissioning Engineering Manager shall have a minimum of a Bachelor's Degree in engineering or the physical sciences and have a minimum of five years of professional level experience in nuclear services, nuclear plant design and operation, including coordination, direction and supervision of personnel. A working knowledge and understanding of decommissioning techniques, scheduling and contract administration is required.

2.4.9 Decommissioning Safety Review Committee

The DSRC reports to the Vice President responsible for nuclear activities. The function of this committee is to monitor decommissioning operations to ensure that they are being performed safely. This committee will review and audit major decommissioning operations dealing with radioactive material and radiological controls. In addition, they will review work specifications and administrative procedures, reportable occurrences under 10 CFR 20 and 10 CFR 50, and changes made in accordance with 10 CFR 50.59. Specific membership, duties and responsibilities of the DSRC are identified in the Decommissioning Technical Specifications.

INTENTIONALLY LEFT BLANK

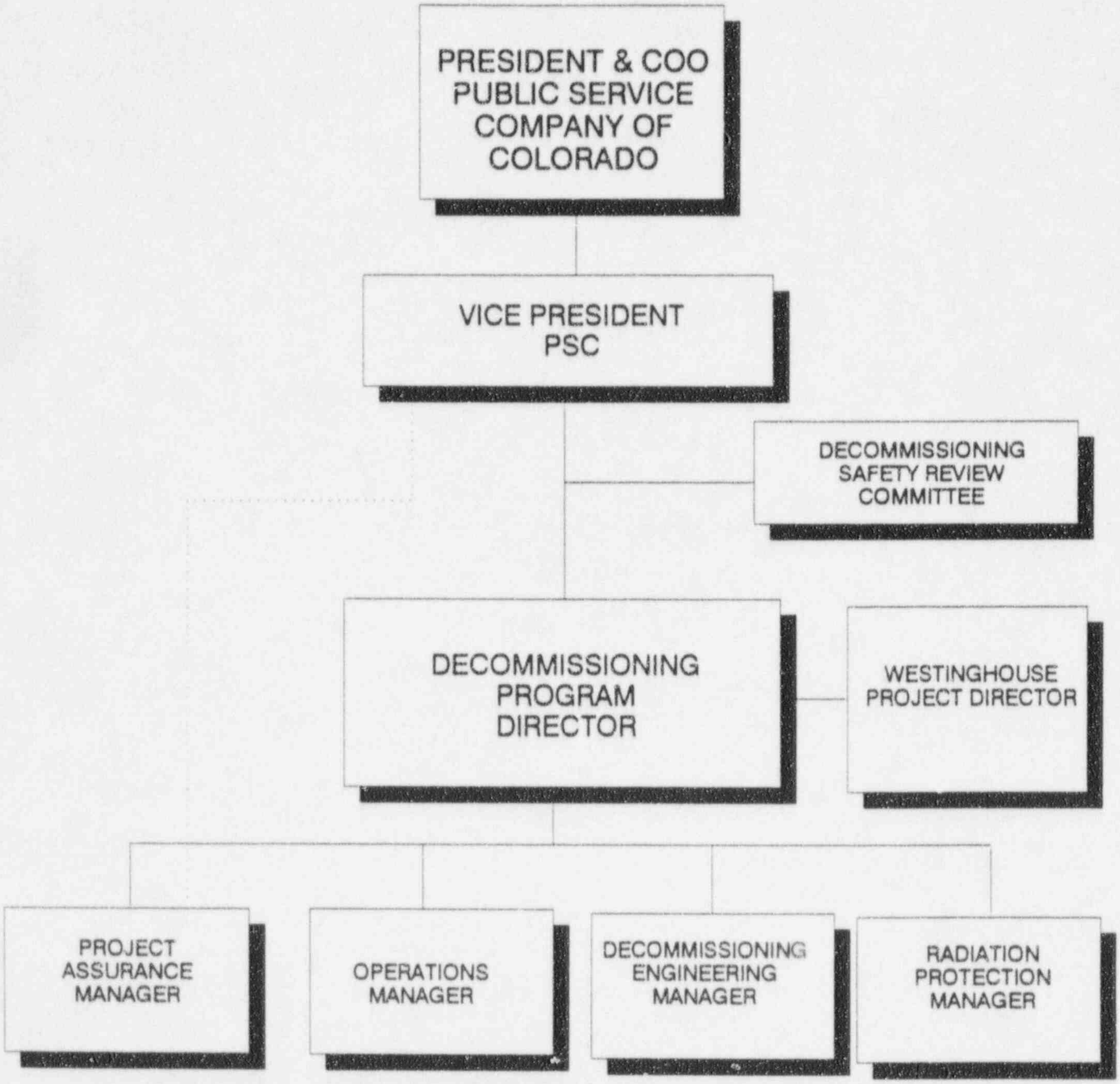

Figure 2.4-1 PSC Decommissioning Organization

INTENTIONALLY LEFT BLANK

3.2 RADIATION PROTECTION PROGRAM

3.2.1 Introduction

This section sets forth the policy and the requirements of the Radiation Protection Program to be implemented during the decommissioning of the Fort St. Vrain facility. This section is the highest tier document of the Radiation Protection Program and provides definitions of the Radiation Protection organization, responsibilities, authorities, administrative policies, program objectives and standards to implement the Radiation Protection Program. This section will also be used as the basis document for all Radiation Protection Program administrative and implementing procedures.

Title 10 Code of Federal Regulations, applicable regulatory guidance documents and industry standards are the basis of the Radiation Protection Program. This section was specifically formatted using Draft NUREG-0761, "Radiation Protection Plans for Nuclear Power Reactor Licensees" (Ref. 15), which provides guidance for the content of a "Radiation Protection Plan". It also incorporates the guidance contained in NRC Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable" (Ref. 16) and NRC Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable" (Ref. 17).

3.2.2 Management Policy

3.2.2.1 Management Policy Statement

PSC and its management are committed to the safe decommissioning of the Fort St. Vrain facility. The primary objective of the Radiation Protection Program is to protect the workers, visitors and the general public from radiological hazards that have the potential of developing during the decommissioning project. PSC and its contractors will provide sufficient qualified staff, facilities and equipment to perform the Fort St. Vrain facility decommissioning in a radiologically safe manner. PSC is committed to strict compliance with regulatory requirements, radiation exposure limits, and limits regarding release of radioactive materials. In addition, PSC will make every reasonable effort to maintain radiation exposures and releases of radioactive materials in effluents to unrestricted areas As Low As Reasonably is Achievable (ALARA) The ALARA philosophy will be incorporated into all decommissioning activities and have full management support.

3.2.2.2 Administration Policy

Activities conducted during the Fort St. Vrain decommissioning project that have the potential for exposure to radiation or radioactive materials will be managed by qualified individuals who will perform program operations according to procedural guidelines. Radiological hazards will be monitored and evaluated on a routine basis to maintain radiation exposures and the release of radioactive materials to unrestricted areas as far below specified limits as is reasonably achievable. All decommissioning project work activities, and each element of the Radiation Protection Program will be specifically defined and implemented using written manuals, procedures and instructions. Radiation protection training will be provided to all occupationally exposed individuals to ensure they understand and accept the responsibility to follow all procedures and to maintain their radiation dose ALARA.

Project management will ensure that work specifications, designs, and work packages involving potential radiation exposure or handling of radioactive materials incorporate effective radiological controls. Project supervisors will include radiation protection considerations in the work activities under their control.

Radiation protection records will be prepared and maintained using high standards of accuracy, traceability and legibility to meet the requirements of regulatory agencies and company procedures.

3.2.2.3 ALARA Policy

All activities at Fort St. Vrain involving radiation and radioactive materials shall be conducted such that radiation exposures to employees, contractors, and the general public are maintained ALARA, taking into account current technology and the economics of radiation exposure reduction in relationship to the benefits to health and safety.

Project management will establish specific goals and objectives for the Fort St. Vrain decommissioning project ALARA program. The ALARA program will be based on the guidance provided in Regulatory Guides 8.8 and 8.10 (References 16 and 17). The ALARA program will incorporate current technology and sound radiation protection practices to maintain exposure to ionizing radiation ALARA.

3.2.2.4 Regulatory Compliance Policy

Project management will maintain the Radiation Protection Program in compliance with the requirements of 10 CFR, 49 CFR and to the extent practical, the information contained in industry standards, Regulatory Guides and other guidance documents referenced in Section 3.2. The decommissioning of Fort St. Vrain is scheduled to occur during the period when the revised 10 CFR 20 regulations become effective. PSC submitted an exemption request in Ref. 80, seeking exemption from the revised 10 CFR 20 requirements. The NRC granted the exemption request in Ref. 81, permitting the decommissioning project to be completed under the existing 10 CFR 20 requirements.

3.2.2.5 Waste Minimization and Disposal Policy

Project management will implement and enforce a program for minimizing the generation of radioactive wastes. Implementing procedures will be developed for the use, classification, treatment, packaging and shipment of radioactive material. These procedures will ensure strict compliance with applicable Federal, State and local regulation and burial site criteria.

Project management will establish waste minimization goals. To ensure these goals are achieved, all decommissioning personnel will receive training in the applicable procedures and practices to minimize the generation of radioactive waste.

3.2.2.6 Respiratory Protection Policy

Project management is committed to minimizing the inhalation of air contaminated with dusts, mists, fumes, gases, vapors and radionuclides. The primary means of achieving this goal will be to prevent or mitigate the hazardous condition at the source. Every reasonable effort will be made to achieve this objective by using engineering controls, including process modification, containment and ventilation techniques. The use of respiratory protection equipment will be consistent with the goal of maintaining the total effective dose to personnel ALARA.

A respiratory protection program will be developed, implemented and maintained in accordance with 10 CFR 20 and using the regulatory guidance found in NRC Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection" (Ref. 18), and the NUREG-0041 "Manual of Respiratory Protection Against Airborne Radioactivity Materials" (Ref. 19).

3.2.3 Radiation Protection Organization and Functions

3.2.3.1 Radiation Protection Organization

The Radiation Protection Organization will ensure that a high level of performance in radiation protection is achieved through effective implementation and control of radiation protection activities. This high level of radiation protection performance will be achieved through the combined efforts of PSC management and the Westinghouse Team.

The PSC management structure that will oversee and control the Radiation Protection Program during the decommissioning project is shown on Figure 2.4-1, "PSC Decommissioning Organization Chart". The PSC organization will provide control, direction and oversight, and ensure the implementation of the Radiation Protection Program.

The PSC Decommissioning Program Director and the Westinghouse Project Director have ultimate responsibility for assuring that an effective Radiation Protection Program is implemented during the Fort St. Vrain decommissioning. This corporate and project interface will ensure a coordinated and effective approach to the minimization of individual and collective dose and the control of radioactive materials during decommissioning.

Reporting directly to the PSC Decommissioning Program Director will be the PSC Radiation Protection Manager. The PSC Radiation Protection Manager has oversight responsibility for the development and implementation of the Radiation Protection Program policies and standards. The PSC Radiation Protection Manager will serve as a member of the Decommissioning Safety Review Committee and will also serve as a Co-chair of the ALARA Committee. The PSC Radiation Protection Manager will ensure that PSC has the proper control and authority over decommissioning activities as they relate to radiation protection. The PSC Radiation Protection Manager represents the formal line of communication and authority between Fort St. Vrain management and the Westinghouse organization for radiation protection matters. The PSC Radiation Protection Manager will be responsible for approval of the Radiation Protection Program manuals and the content of radiation protection training programs. The PSC Radiation Protection Manager will also be directly responsible for the Radiological Environmental Monitoring Program and the Emergency Response Plan. The PSC Radiation Protection Manager will be qualified in accordance with NRC Regulatory Guide 1.8 "Personnel Selection and Training"

(Ref. 20), and ANS/ANSI 3.1 "Selection, Training and Qualification of Personnel for Nuclear Power Plants" (Ref. 21). The staff positions (Senior Health Physicists) reporting to the PSC Radiation Protection Manager will provide review and evaluation functions to ensure that the Radiation Protection Program policies and standards are implemented.

The Westinghouse Project Radiation Protection Organization will be administered by the Project Radiation Protection Manager (PRPM), under the authority of the PSC Radiation Protection Manager. Figure 3.2-1, "Westinghouse Team Radiation Protection Organization Chart" shows the key members of the Radiation Protection Organization. The Project Radiation Protection Manager, under the direction of the PSC Radiation Protection Manager, has the responsibility for the Radiation Protection Program development, implementation and compliance with the applicable regulations. The Project Radiation Protection Manager will be qualified in accordance with NRC Regulatory Guide 1.8 (Ref. 20) and ANS/ANSI 3.1 (Ref. 21).

The Project Radiation Protection Manager will report directly to the Westinghouse Project Director. This reporting chain will ensure sufficient authority and independence to implement an effective Radiation Protection Program. It will also provide a direct line of communication to senior project management.

The Project Radiation Protection Manager will have the authority to stop work whenever activities have the potential to jeopardize the health and safety of workers, visitors or the general public. This authority will not be limited to radiological safety issues. If the activities violate operational parameters, administrative guidelines, safety requirements or Radiation Protection procedures, the Project Radiation Protection Manager will have the authority to stop work. The authority to overrule the Project Radiation Protection Manager's stop work order may only come from the PSC Decommissioning Program Director, PSC Radiation Protection Manager or the Westinghouse Project Director.

The staff positions shown on Figure 3.2-1 will have the primary responsibility for providing technical direction, implementation of the Radiation Protection Program, and supervision of the activities of the Radiation Protection Technicians and support personnel. Designated radiation protection staff members will be qualified in accordance with NRC Regulatory Guide 1.8 (Ref. 20) and ANS/ANSI 3.1 (Ref. 21), and will serve as the qualified substitute for the Project Radiation Protection Manager. The staff positions will have the authority to stop work whenever activities jeopardize the radiological health and safety of workers, visitors or the general public, or, if the activities violate Radiation Protection procedures.

The number and titles of positions shown on Figure 3.2-1 may be modified during the course of decommissioning. This may be necessary during initial project start-up and demobilization. Changes to the Radiation Protection organization will require approval from the PSC Radiation Protection Manager and will be reflected in Ref. 82.

3.2.3.2 Functional Descriptions

The effective implementation of the Radiation Protection Program is the responsibility of all project personnel. Specific responsibilities for the implementation of the Radiation Protection Program are listed below.

The PSC Vice President, responsible for nuclear activities, is responsible for the safe decommissioning of Fort St. Vrain and is the designated Corporate Officer for PSC.

The PSC Decommissioning Program Director is responsible for conducting facility decommissioning in accordance with regulatory requirements including those activities related to radioactive materials and radiation exposure. Major responsibilities related to the Radiation Protection Program include the following:

1. Ensure support for the ALARA program from project personnel, and provide oversight of the ALARA Committee.
2. Participate in the selection of specific radiation protection goals and objectives for the decommissioning.
3. Support the Radiation Protection Manager in implementing the Radiation Protection Program.
4. Ensure periodic status reports on the Radiation Protection Program are distributed to management.

5. Issue or rescind "stop work" orders, as required.
6. Oversee the Decommissioning Emergency Response Plan.

The PSC Radiation Protection Manager reports to the PSC Decommissioning Program Director and is responsible for implementation of laboratory, environmental monitoring, training, access control, emergency response and Radiation Protection Program activities. Major responsibilities related to the Radiation Protection Program include the following:

1. Ensure proper implementation of Fort St. Vrain Radiation Protection policy.
2. Interface with the Project Radiation Protection Manager.
3. Ensure adequate staffing, facilities and equipment are available to perform the functions assigned to Radiation Protection personnel.
4. Responsible for oversight and direction of the Radiation Protection Program.
5. Issue or rescind "stop work" orders, as appropriate.
6. Ensure personnel at Fort St. Vrain have received job specific and general employee training.
7. Serve on the Decommissioning Safety Review Committee.
8. Serve as Co-chair of the ALARA Committee.
9. Ensure proper disposal of radioactive solid, liquid and gaseous wastes.
10. Maintain and implement the Decommissioning Emergency Response Plan.
11. Coordinate revisions to the Radiation Protection Program.
12. Approve Radiation Protection training programs.
13. Ensure implementation of the Radiological Environmental Monitoring Program.
14. Review and approve training programs related to work in radiological areas or involving radioactive material.

The PSC Senior Health Physicists report to the PSC Radiation Protection Manager. Major responsibilities as related to the Radiation Protection Program include the following:

1. Coordinate the annual review of the Radiation Protection Program.
2. Serve as designated alternates to the PSC Radiation Protection Manager.
3. Evaluate Radiation Protection training programs.

4. Monitor collective exposure of various decommissioning activities.
5. Conduct inspections and monitorings of work in progress to evaluate the adequacy of the implementation of the Radiation Protection Program.
6. Evaluate plant contamination control activities.
7. Support the decommissioning ALARA committee.
8. Assist in reviewing radiological occurrences to identify root cause and corrective actions for radiation protection incidents.
9. Issue "stop work" orders, as appropriate.
10. Evaluate the adequacy of the radioactive material disposal and shipping activities.
11. Evaluate the dosimetry and personnel exposure tracking system.
12. Evaluate bioassay and radiochemistry activities.
13. Evaluate the calibration of portable, stationary and laboratory radiation monitoring equipment.
14. Evaluate the adequacy of the Respiratory Protection Program.
15. Evaluate the Radiological Environmental Monitoring Program.
16. Participate in the Decommissioning Emergency Response Plan.

The PSC Project Assurance and Operations Managers' responsibilities as related to the Radiation Protection Program include the following:

1. Ensure personnel under their direction are properly trained in and comply with radiation protection requirements.
2. Support the PSC Radiation Protection Manager in overseeing the implementation of the Radiation Protection Program, including (but not limited to) performance of audits, monitorings, and surveillances of their areas of responsibility, and routine inspections of work areas where their personnel are involved.
3. Participate in the Decommissioning Emergency Response Plan.

The PSC Unit Manager, Operations reports to the PSC Operations Manager. Major responsibilities as related to the Radiation Protection Program include the following:

1. Ensure the safe operation of plant systems.
2. Ensure planned radiological effluent releases are properly performed.
3. Notify the Nuclear Regulatory Commission, as required.
4. Notify Radiation Protection personnel when changes in plant conditions could affect radiological conditions.

5. Ensure personnel under their direction comply with the requirements of the Radiation Protection Program.
6. Participate in the Decommissioning Emergency Response Plan.

The Westinghouse Project Director is responsible for conducting decommissioning operations in accordance with regulatory requirements related to radioactive materials and radiation exposure. Major responsibilities as related to the Radiation Protection Program include the following:

1. Ensure that all project personnel are properly trained in and comply with radiation protection requirements.
2. Ensure support for the ALARA program from all project personnel.
3. Participate in the selection of specific Radiation Protection goals and objectives for the decommissioning.
4. Support the Project Radiation Protection Manager in implementing the Radiation Protection Program.
5. Ensure periodic status reports on the Radiation Protection Program are distributed to project management.
6. Issue or rescind "stop work" orders, as appropriate.
7. Participate in the Decommissioning Emergency Response Plan.

The Westinghouse Team Site Operations Manager, Technical Services Manager and Project Control Manager report directly to the Westinghouse Project Director. Major responsibilities as related to the Radiation Protection Program include the following:

1. Ensure personnel under their direction are properly trained and comply with radiation protection requirements.
2. Support the Project Radiation Protection Manager in the implementation of the Radiation Protection Program.
3. Ensure that exposure and waste reduction techniques are incorporated into work plans and procedures.
4. Ensure radiation protection and ALARA principles are incorporated into project activities.
5. Ensure that requirements, methods, regulations and procedures for waste processing are specified.
6. Ensure that craft labor is provided for operation of the waste processing system, waste segregation and packaging.
7. Participate in the Decommissioning Emergency Response Plan.

The Westinghouse Team Project Radiation Protection Manager reports directly to the Westinghouse Project Director and is responsible for laboratory analysis, radiation protection training, radioactive waste management, dosimetry, respiratory protection, ALARA, radiation protection job coverage and radiological surveys. Major responsibilities related to the Radiation Protection Program include the following:

1. Administer the Radiation Protection Program policies and procedures.
2. Review and approve Radiation Protection procedures, instructions and Technical Basis Documents.
3. Provide an interface for the PSC Radiation Protection Manager and the Westinghouse Project Director.
4. Ensure adequate staffing, facilities and equipment are available to perform the functions assigned to Radiation Protection personnel.
5. Select and approve Radiation Protection staff members.
6. Provide adequate radiation protection personnel to ensure job coverage (minimum established ratio of workers to Radiation Protection Technicians) for project personnel during all working hours.
7. Establish goals and objectives for the Radiation Protection Program.
8. Issue or rescind "stop work" orders, as appropriate.
9. Ensure that locations, operations, and conditions that have the potential for causing significant exposures to radiation are identified and controlled.
10. Review and approve training programs related to work in radiological areas or involving radioactive material.
11. Ensure General Employee Training is provided for personnel working at Fort St. Vrain.
12. Administer the proper disposal of radioactive solid waste, and ensure proper disposal of liquid and gaseous wastes.
13. Provide Radiation Protection input to decommissioning planning.
14. Trend radiation work performance of project personnel including contamination and radiation exposure control.
15. Review Radiological Occurrence Report (ROR) root causes and corrective actions for incidents and deficiencies associated with radiation protection.
16. Ensure an effective ALARA Program is maintained.
17. Ensure development and implementation of the Final Radiological Survey Program.
18. Serve as a Co-chair of the ALARA Committee.
19. Participate in the Decommissioning Emergency Response Plan.

Technical Assistants to the Project Radiation Protection Manager are direct reports. Their major responsibilities as related to the Radiation Protection Program include the following:

1. Provide technical support to the Radiation Protection staff as assigned by the Project Radiation Protection Manager.
2. Review procedures as assigned.
3. Prepare designated reports.
4. Follow designated RP activities to ensure implementation, completion, and close out.
5. Coordinate staffing needs with SEG home office.
6. Track and close commitments.
7. Track the Radiological Occurrence Report Program.

The Westinghouse Team Project Radwaste Supervisor reports to the Project Radiation Protection Manager. Major responsibilities as related to the Radiation Protection Program include the following:

1. Coordinate radioactive waste minimization and disposal activities.
2. Classify radioactive waste material in accordance with 10 CFR 61 in preparation for disposal.
3. Monitor the packaging and preparation of radioactive material for shipment.
4. Schedule and complete shipments of radioactive material in accordance with Department of Transportation (DOT) and NRC regulations and burial site requirements.
5. Make recommendations to the Project Radiation Protection Manager concerning site management issues that affect radwaste operations and shipping.
6. Issue and rescind "stop work" orders, whenever activities have the potential to jeopardize the health and safety of workers, visitors, or the general public.
7. Provide input to training for radioactive waste management.

The Westinghouse Team Final Survey Operations Supervisor reports to the Project Radiation Protection Manager. Major responsibilities as related to the Radiation Protection Program include the following:

1. Ensure surveys are performed and samples are collected to implement the Final Survey Program.
2. Ensure that technicians are trained to handle all phases of final survey work.
3. Review effectiveness of operational aspects of the Final Survey Program.
4. Assist in the review of Radiological Occurrence Reports associated with final survey activities or instrumentation concerns.
5. Interface with the training group.
6. Assist in the development and review of procedures, instructions and Technical Basis Documents that support final survey activities.
7. Perform surveys and collect samples as required by final survey procedures.

The Westinghouse Team ALARA Supervisor reports to the Project Radiation Protection Manager. Major responsibilities of the ALARA Supervisor as related to the Radiation Protection Program include the following:

1. Review of Radiation Work Permits (RWPs) and work packages for ALARA consideration.
2. Establish ALARA person-Rem budgets for project tasks.
3. Coordinate pre-job briefings and mock-up training.
4. Track project exposure, radiological performance indicators, and prepare project person-Rem reports.
5. Serve as ALARA Committee secretary.
6. Implement the ALARA suggestion program.
7. Coordinate the ALARA exposure history records management system.
8. Perform cost benefit analyses, as required.
9. Provide radiation protection and mock-up training, as required.
10. Issue and rescind "stop work" orders, whenever activities have the potential to jeopardize the health and safety of workers, visitors, or the general public.

The Westinghouse Team Radiation Protection Operations Supervisor reports to the Project Radiation Protection Manager. Major responsibilities as related to the Radiation Protection Program include the following:

1. Ensure that policies relating to personnel radiation exposures are established and enforced.
2. Ensure appropriate radiation, contamination and airborne surveys are performed to verify radiation levels and working conditions of the facility.
3. Ensure accountability of all byproduct material.
4. Ensure technicians are trained and qualified to handle all assigned radiation protection work.
5. Review radiological protection operations program effectiveness.
6. Issue and rescind "stop work" orders, whenever activities have the potential to jeopardize the health and safety of workers, visitors, or the general public.
7. Control assigned portable, stationary and laboratory radiation monitoring equipment.
8. Approve and classify RORs and assist in their review to identify root causes.
9. Interface with the Westinghouse training group.
10. Develop, review, and approve procedures for Radiation Protection operations.
11. Ensure implementation of the Radiation Work Permit Program.
12. Participate in the Decommissioning Emergency Response Plan.
13. Evaluate potential radiological hazard situations.
14. Provide direction and oversight of the Respiratory Protection Program.

The Westinghouse Team Radiochemistry/Training Supervisor reports to the Project Radiation Protection Manager. Major responsibilities related to the Radiation Protection Program include the following:

1. Collect appropriated bioassay samples and review results at established intervals.
2. Ensure performance of radiochemical analyses of solid, liquid and gaseous samples from plant systems, radioactive waste and work place.
3. Ensure proper preparation and control of reagents.

4. Ensure adequate laboratory quality control.
5. Prepare radiochemistry operational procedures.
6. Evaluate internal and external dosimetry results and resolve positive or questionable results.
7. Participate in the Decommissioning Emergency Response Plan.
8. Maintain personnel dosimetry exposure records.
9. Provide oversight to the Training Coordinator to ensure adequate training is provided to decommissioning staff and site visitors.

The Westinghouse Team Training Coordinator reports to the Radiochemistry/Training Supervisor. Major responsibilities related to the Radiation Protection Program include the following:

1. Provide site access (General Employee Training) and radiation protection training support.
2. Interface with Westinghouse Team members on training issues.
3. Provide training for radioactive waste management.
4. Implement qualification and training program.
5. Ensure that personnel are evaluated, their qualifications documented, and a training plan is developed as appropriate to ensure they will meet and maintain minimum requirements for independent work when required.
6. Ensure that each responsible supervisor/engineer is knowledgeable about the qualification/certification training and retraining (continuing training) requirements for personnel under their direction.
7. Ensure that the selection, evaluation, training, and qualification/certification process is documented in accordance with applicable implementing procedures.
8. Ensure coordination with the responsible supervisor/engineer to plan and schedule training.
9. Ensure that technicians are trained to handle all phases of Radiation Protection work.
10. Procedure development and review.
11. Participate in the training of decommissioning emergency response personnel.

The Westinghouse Team Technical Projects Supervisor reports to the Project Radiation Protection Manager. Major responsibilities related to the Radiation Protection Program include the following:

1. Provide radiological engineering support for project planning and special projects.
2. Ensure Final Survey Plan is prepared and revised as necessary.
3. Ensure that survey packages are prepared for implementation of surveys and sample collection.
4. Review effectiveness of technical aspects of the Final Survey Program.
5. Ensure adequate data collection and review for final surveys.
6. Supervise the development and review of procedures and Technical Basis Documents that support Final Survey activities.
7. Interface with the training group.
8. Ensure development, control and management of computer data base for final survey.
9. Provide technical interface for outside agencies and QA groups in relation to Final Survey Program.
10. Ensure calibration, maintenance and control of portable radiation monitoring equipment and secondary dosimetry devices.

The Westinghouse Team Final Survey Lead Engineer reports to the Project Radiation Protection Manager. Major responsibilities related to the Radiation Protection Program include the following:

1. Perform oversight of Final Survey Plan.
2. Assist the Technical Projects Supervisor in regulatory interpretation.
3. Interface with PSCo, the NRC, the Oakridge Institute of Science and Education (ORISE) and General Public Utilities (GPU).
4. Resolve regulatory issues and prepare responses.
5. Perform data review and reduction.
6. Perform statistical evaluations.
7. Perform area/unit investigation and reclassification.
8. Perform final survey package closure.
9. Maintain data integrity.
10. Prepare the final report.
11. Review and concur with procedure and Technical Basis Document development and revisions.

The Westinghouse Team Radiation Protection Operations Shift Supervisors report to the Radiation Protection Operations Supervisor. Major responsibilities related to the Radiation Protection Program include the following:

1. Ensure that all required radiological surveys are performed by Radiation Protection Technicians.
2. Ensure that radiation monitoring instrumentation is properly used for radiation protection work.
3. Assign technicians for needed qualification training.
4. Review Radiation Work Permits and radiological posted areas.
5. Review Radiation Protection paperwork for completeness.
6. Participate in the Decommissioning Emergency Response Plan.

The Westinghouse Team Radiation Protection Technicians report to a designated Radiation Protection Supervisor. Major responsibilities related to the Radiation Protection Program include the following:

1. Conduct radiation, contamination and airborne surveys and keep complete and legible records.
2. Perform final radiological surveys and collect samples in accordance with the Final Radiological Survey Plan and implementing procedures.
3. Identify and post radiation, contamination, hot particle, airborne and radioactive material areas.
4. Prepare Radiation Work Permits to control access to and activities in radiologically controlled areas.
5. Monitor work to assure compliance with good radiological work practices.
6. Implement ALARA program requirements.
7. Maintain and calibrate portable monitoring instruments.
8. Issue "stop work" orders whenever activities have the potential to jeopardize the health and safety of workers, visitors, or the general public.
9. Sample various process streams for radiochemical analysis.
10. Verify packaging of radioactive material.
11. Participate in the Decommissioning Emergency Response Plan.

All Project Supervisors have major responsibilities related to the Radiation Protection Program including the following:

1. Ensure that personnel assigned to work with radioactive material attend required training.
2. Ensure personnel under their direction comply with radiation protection requirements.
3. Identify radiation work procedures and practices that need upgrading.
4. Assign tasks and ensure that workers are prepared for tasks in order to maintain doses ALARA.
5. Ensure that employees know the radiological hazards of their duties.
6. Ensure that assigned equipment and facilities are designed, installed and operated to minimize the radiological hazards to personnel.
7. Know the location and the radiological hazards in the work area.
8. Know the exposure status of those for whom they are responsible.
9. Provide information on projected work activities to the Radiation Protection organization.
10. Notify Radiation Protection personnel of any radiological problems encountered.
11. Assign tasks to distribute dose among exposed personnel to minimize the likelihood of overexposure and to maintain individual doses ALARA.
12. Ensure that workers are prepared for tasks with tools, equipment and training to minimize time spent in Radiation Areas.

All Project Workers have major responsibilities related to the Radiation Protection Program including the following:

1. Obey promptly "stop-work" and "evacuate" orders from Radiation Protection personnel.
2. Obey posted, oral and written radiological control instructions and procedures, including instructions on Radiation Work Permits (RWPs).
3. Wear TLDs and self reading/electronic dosimeters where required by postings or as directed by Radiation Protection personnel.
4. Immediately report unexpected exposure and lost or offscale dosimeter to Radiation Protection personnel.
5. Keep track of personal radiation exposure status to ensure that administrative dose limits are not exceeded.
6. Remain in as low a radiation area as practicable to accomplish work.

7. Do not loiter in radiation areas.
8. Do not smoke, eat, drink or chew in radiologically controlled areas.
9. Wear anti-contamination clothing and respiratory protection properly whenever required by postings or by Radiation Protection personnel.
10. Remove anti-contamination clothing and respiratory protection properly to minimize spread of contamination.
11. Monitor for contamination when leaving a contaminated area or a radiologically controlled area and notify Radiation Protection personnel if contamination is found.
12. Minimize the spread of contamination and promptly notify Radiation Protection personnel of any known or potential radioactive spills.
13. Do not unnecessarily touch a contaminated surface or allow clothing, tools or other equipment to do so.
14. Place contaminated tools, equipment and solid waste on disposable surfaces (e.g., sheet plastic) when not in use and inside plastic bags when work is finished.
15. Limit the amount of material that has to be decontaminated or disposed of as radioactive waste.
16. Notify Radiation Protection personnel of faulty or alarming Radiation Protection equipment.
17. Report the presence of open wounds to Radiation Protection personnel prior to working in areas where radioactive contamination exists and exit immediately if a wound occurs while in such an area.
18. Notify Radiation Protection personnel upon returning to the site after medical administration of radiopharmaceuticals.
19. Assure a mentally alert and physically sound condition for performing assigned work.
20. Ensure that work activities do not create radiological problems for others and be alert for possibilities that activities of others may change the radiological conditions to which the individual is exposed.
21. Do not touch or pickup material in RCA's without a prior survey.
22. Comply with the requirements of the Decommissioning Emergency Response Plan.

3.2.3.3 Radiation Protection Organization Staffing

The Radiation Protection organization will provide appropriate personnel and resources to verify a radiologically safe working environment. A sufficient number of Radiation Protection personnel will be present during the various decommissioning activities to ensure compliance with the Radiation Protection Program and implementing procedures. Projected Radiation Protection staffing levels are shown in Figure 3.2-1. The Project Radiation Protection Manager will establish guidelines for adequate Radiation Protection staffing based on radiological parameters and work scope.

Radiation Protection staffing levels will be periodically reviewed by the PSC and Project Radiation Protection Manager and the Radiation Protection Supervisors, as applicable, to ensure that adequate staffing levels are maintained consistent with current and planned decommissioning activities. Specific work packages will be addressed on a case-by-case basis (e.g., by the Radiation Protection Operations Supervisor) to ensure adequate Radiation Protection Technician to worker ratios based on the guidelines provided by the Project Radiation Protection Manager.

Contingencies will be in place to adjust staffing levels during the project for routine functions and unanticipated radiological events. These staffing contingencies will ensure that all work is performed in a radiologically safe and timely manner. Staff adjustments will be implemented when needed, but only after review and recommendation of the PSC and Project Radiation Protection Managers and approval of the Westinghouse Project Director.

Continuous Radiation Protection coverage will be provided for decommissioning work activities that involve significant radiological hazards, such as the removal of unshielded, highly activated/contaminated components from the PCRV. For example, two Radiation Protection technicians would normally be assigned to the PCRV area during component removal and handling. Another example is that additional Radiation Protection technicians may be provided during PCRV concrete cutting operations, which are expected to run 24 hours a day.

Intermittent Radiation Protection coverage will be provided to decommissioning work activities that have a minimal potential for significant personnel exposure. The removal of balance of plant systems is an example of a work activity that will be monitored periodically by Radiation Protection Technicians.

3.2.3.4 Radiation Protection Program Manuals

The Radiation Protection Program will be integrated into all applicable decommissioning work activities. The Radiation Protection Program will be specifically defined and implemented using a program manual consisting of both administrative procedures and specific implementing procedures.

The Radiation Protection Program will incorporate four manuals:

1. Radiation Protection Manual
2. Radioactive Waste Manual
3. Radiation Protection Training Manual
4. Off-Site Dose Calculation Manual

The four (4) manuals will contain the administrative and implementing procedures, which will specify the standards and controls and define corporate and site objectives for the programs described in each manual. The Project Radiation Protection Manager will have the responsibility for the development and implementation of manuals 1-3, following approval by the PSC Radiation Protection Manager. The PSC Radiation Protection Manager will have the responsibility for the development and implementation of Manual 4. The development and control of Radiation Protection procedures will be in accordance with the Quality Assurance Plan (Section 7 of this plan) and will incorporate the following procedural guidelines:

1. Clearly defined scope, applicability, limiting conditions and precautions.
2. Uniform procedure identification and status (titling or numbering, location, and status for page and revision identification).
3. Consistent format (for organization, instruction step format, instruction step designation, caution and note format, and page format.
4. Clearly understood text, using standard grammar, nomenclature and punctuation; concise instruction steps in a logical sequence.
5. "Hold points" for procedures with unique and/or high personnel risk.
6. Effective grouping of procedures and a clear table of contents for the procedure binder or manual to allow easy location of a particular procedure.
7. Review, approval, and issuance of temporary changes and permanent revisions.
8. Periodic review of procedures.

9. Controls to make procedure use as convenient as possible and to ensure that only approved copies are available.

Figure 3.2-2 , "Radiation Protection Program Manual Structure", shows the hierarchy and organization of the Radiation Protection Program manuals and associated program elements.

3.2.3.5 Contracted Radiation Protection Services

Procurement of contracted Radiation Protection services will be provided in accordance with Section 7, the Quality Assurance Plan, the Radiation Protection Program, and bid specifications developed for the decommissioning project.

Westinghouse has been contracted to provide the operational radiation protection, radioactive waste management and final site release survey, either directly or through subcontractors. Examples of subcontracted services include: external dosimetry processing, primary instrument calibration and 10 CFR 61 program sample analyses.

3.2.4 Radiation Protection Training and Qualification

3.2.4.1 General Considerations

All decommissioning project workers will be provided instruction in radiation protection concepts commensurate with the radiological hazards they may encounter during the Fort St. Vrain decommissioning project. This training is recognized as essential in achieving high standards of performance in radiation protection.

The Project Radiation Protection initial training, qualification and retraining programs will be developed using applicable guidance contained in NUREG-0761 (Ref. 15), NRC Regulatory Guide 8.27, "Radiation Protection Training For Personnel At Light-Water-Cooled Nuclear Power Plants", March 1981 (Ref. 22), NRC Regulatory Guide 8.13, "Instruction Concerning Pre-natal Radiation Exposure", (Ref. 23) November 1975 and NRC Regulatory Guide 8.29, "Instruction Concerning Risks From Occupational Radiation Exposure", (Ref. 24) July 1981. Guidance from these documents will be incorporated into a Radiation Protection Training Manual, which will include both administrative and implementing procedures.

Radiation protection training will be provided to three basic work groups:

Non-Radiation Workers, Radiation Workers and Radiation Protection Personnel.

Training for each work group will be organized as follows:

Non-Radiation Workers will receive, as appropriate:

1. Introduction to Radiation Protection
2. Non-radiation worker indoctrination

Radiation Workers will receive, as appropriate:

1. Radiation worker training
2. Specialized ALARA training
3. Respiratory protection training

Radiation Protection Personnel will receive as appropriate:

1. Radiation Protection technician training
2. Radiation Protection support staff training
3. Radiation Protection supervisor training
4. Radiation Protection Manager training
5. Radioactive waste management training

All classroom training will be conducted using lesson plans approved by the Project Radiation Protection Manager and the PSC Radiation Protection Manager. On-the-job training (OJT) will be administered by a qualification card or equivalent which is approved by the Project Radiation Protection Manager. Personnel assigned to perform Radiation Protection training will be qualified as instructors and/or evaluators in accordance with the Radiation Protection Training Manual.

3.2.4.2 Introduction to Radiation Protection

All project personnel, visitors and transients granted unescorted access to the Restricted Area will receive, as a minimum, annual instruction in elementary radiation effects and basic aspects of radiation protection during general employee orientation. This training will meet the requirements of 10 CFR 19.

The Radiation Protection orientation training, which will normally be given during General Employee Training, will include:

1. Instructions to not enter radiologically controlled areas, violate radiological postings or cross radiological boundaries.
2. Discussion of the decommissioning project.
3. Brief explanation of radioactivity and the biological effects of low-level radiation exposure.
4. Emergency response actions.

Personnel will be required to score 80% or above on a written examination. Personnel will be required to complete annual requalification training. The requalification training will give special attention to changes in radiation protection, emergency planning and management policy.

Personnel who require access to the Restricted Area to service and maintain equipment (e.g., vending machines, office equipment, etc.) or who have had unescorted access at other nuclear facilities within the last year, will be allowed to receive expedited project orientation training by:

1. Reading a copy of site specific information covering radiation protection, emergency response and access control procedure.
2. Signing a statement acknowledging their understanding of the information provided.

3.2.4.3 Non-Radiation Worker Indoctrination

All visitors who require access to radiologically controlled areas will receive indoctrination training. Visitor access to radiologically controlled areas will require approval of the Project Radiation Protection Manager or designee. This indoctrination training will include:

1. The requirement that the visitor remain with the escort at all times and follow directions of the escort.
2. A description of the radiological nature and required controls of the area to be entered.
3. The purpose and proper use of dosimeters, including how to read self-reading dosimeters.

4. Potential emergency situations and proper actions to take in such events.

3.2.4.4 Radiation Worker Training

Personnel requiring unescorted access to radiologically controlled areas will be required to attend, as a minimum, Radiation Worker training. This training will provide workers with the knowledge and skills needed to work safely in radiologically controlled areas including; radiation and high radiation areas, airborne radioactivity areas, radioactive material areas and contaminated areas. This training will be consistent with that outlined in NUREG 0761, Appendix A, "Example Qualification Standard for Radiation Work Training" (Ref. 15). Radiation Worker training will include:

1. Biological effects of radiation and the risks associated with radiation exposure.
2. Information needed to comply with Radiation Protection procedures and respond properly to warnings and alarms under both normal and emergency conditions.
3. Information needed to ensure that individuals can maintain their own exposure ALARA and ensure that ALARA considerations are appropriately reflected in decisions which affect the exposure of others.
4. Information needed to comply with Radiation Protection Program procedures.
5. Discussion of worker rights and responsibilities as identified in 10 CFR 19.
6. Discussion of NRC Regulatory Guide 8.13 "Instructions concerning Pre-natal Radiation Exposure" (Ref. 23).
7. Training in emergency response actions.
8. Discussion of radioactive and mixed waste minimization.

In addition to classroom training, each participant will be required to demonstrate their abilities in a practical factors session. This will include:

1. Properly don and remove a complete set of protective clothing (excluding respiratory protection equipment).
2. Read and interpret self reading dosimeters.
3. Read and interpret radiological survey maps.

4. Follow procedures to properly enter and exit a contaminated area, including use of proper frisking techniques.
5. Demonstrate understanding and compliance with a Radiation Work Permit.

Personnel will be required to score 80% or above on a written examination. Personnel who pass the written exam will be required to successfully complete the practical factors section. Personnel who fail exams will be evaluated to determine if additional training is needed, limited duty assignment is appropriate or disqualification is necessary. Radiation workers will be required to complete annual requalification training. The requalification training will give special attention to Radiation Protection Program changes, weaknesses observed in project personnel performance and lessons learned.

Significant changes in Radiation Protection policies, requirements, techniques, procedures and equipment, as determined by the Project Radiation Protection Manager, will be disseminated in a timely manner to affected personnel or organizations through periodic awareness presentations.

Radiation Protection procedures will be developed to allow the Project Radiation Protection Manager, on a case-by-case basis, to exempt personnel from Radiation Worker training. Exemptions will only be granted if the following conditions are met:

1. The individual is escorted by a qualified Radiation Worker or has documented proof of Radiation Worker training at another nuclear facility within the last year.
2. The exemption is valid only for the duration of the specific task for which access was approved.
3. The basis of the exemption and approval is documented.

3.2.4.5 Specialized ALARA Training

In addition to Radiation Worker training, separate and detailed instruction in advanced radiation work practices will be provided to those workers performing tasks that involve potential significant radiation exposure or quantities of radioactive material. This training will typically include workers involved in:

1. Operations which involve handling highly radioactive components that have the potential for creating a significant airborne hazard.

2. Operations which require work in contamination containment devices.
3. Grinding, cutting or similar operations on highly radioactive systems, components or piping.
4. Work activities that require the use of special tools and equipment for reducing exposures.
5. Special complex radiation work which involves skills and training beyond that covered in Radiation Worker training.

Specialized ALARA training may include mock-ups, dry runs, pre-job briefings and other special training classes. This training will normally be attended by all personnel involved with the task, including craft supervision and Radiation Protection Technicians. The need for specialized ALARA training will be identified during ALARA reviews and/or Radiation Work Permit preparation.

3.2.4.6 Respiratory Protection Training

Specialized respiratory protection training will be required for all personnel who use respiratory protection devices. Personnel using respirators in radiologically controlled areas will require Radiation Worker qualification.
Respiratory protection training will be conducted in accordance with 10 CFR 20, NRC Regulatory Guide 8.15 (Ref. 18) and NUREG-0041 (Ref. 19). Topics that will be addressed in respiratory protection training include, but are not limited to, the following:

1. Instructions that individuals are authorized to wear only the type of respirator for which they are fit tested and trained.
2. Discussion of the type of airborne contamination for which the respirators will provide protection.
3. Discussion of construction and limitations of respirator types.
4. Discussion of facial hair policy and use of approved eye wear.
5. Pre-use respirator inspection.
6. Instructions for proper donning and fit.
7. Emergency actions in the event of respirator failure including instructions to leave the area.
8. Practical demonstration of respirator inspection, donning and removal.

In addition to classroom training, each participant will be required to demonstrate their ability in a practical factor session that includes:

1. Inspection of a respirator.
2. Donning a respirator and performing a negative pressure test.
3. Removing a respirator.

Personnel will be required to score 80% or above on a written respiratory protection test. Personnel who pass the written exam and are medically qualified will be required to successfully complete the practical factors and fit test session. Respirator users will be required to complete annual requalification training.

3.2.4.7 Radiation Protection Technician Training and Qualification

Radiation Protection personnel will be selected, trained and qualified to ensure that they have sufficient knowledge and practical abilities to implement the Radiation Protection Program effectively. Qualification criteria and job descriptions will be developed for all positions within the Radiation Protection organization. This qualification criteria will contain the elements outlined in Appendix D of Draft NUREG-0761 (Ref. 15) and will be used to augment Radiation Protection Technician training.

All Radiation Protection Technicians will be required to participate in classroom and specific on-the-job training (OJT). The Radiation Protection Training Manual and implementing procedures will ensure that Radiation Protection personnel, who are selected, trained and qualified, have the knowledge and practical skills necessary to perform their work.

Radiation Protection Technician qualification and training will include the following:

1. Radiation Protection Training procedures that specify Radiation Protection personnel qualification criteria, job descriptions, and responsibilities.
2. Review and verification of resumes by the Project Radiation Protection Manager to ensure that personnel have sufficient education and/or experience in the job functions which they will be assigned. Radiation Protection Technicians will be required to meet the education and experience levels specified in ANSI/ANS 3.1 (Ref. 21).

3. Testing of Radiation Protection Technicians to verify appropriate knowledge level in radiation protection theory, equipment, basic mathematics and recognizing unusual situations involving radioactivity. The test will include representative topics listed in Appendix E of Draft NUREG 0761 (Ref. 15).
4. Training in Radiation Protection procedures, the operation and limitations of survey and count room equipment and methods to ensure proper record documentation and traceability.
5. Training in emergency response duties.
6. Review of major decommissioning work activities and potential radiological hazards that may be encountered.

Radiation protection specialists (not qualified as ANSI/ANS-3.1 Radiation Protection Technician) performing unique radiation protection activities such as dosimetry, respiratory protection, bioassay, control point monitor, etc., will be provided specific task related training. This specialized classroom and/or on-the-job training will be commensurate with assigned duties and approved by the Project Radiation Protection Manager.

Upon completion of required classroom training, Radiation Protection personnel will complete on-the-job training in assigned duties. Successful completion of these duties will be documented by the responsible supervisor on the individual's qualification card. The responsible supervisor will ensure that training has been adequate by observation of on-the-job performance. Annual Radiation Protection Technician refresher training, using a structured program approved by the Project Radiation Protection Manager, will be conducted. This training will be documented and may include a written examination.

Additional training will be provided to Radiation Protection personnel if significant changes occur in Radiation Protection policy, requirements, techniques, procedures or equipment. This information will be disseminated to affected personnel or organizations through periodic awareness presentations and/or required reading.

3.2.4.8 Radioactive Waste Management Training

Personnel assigned to the task of packaging, loading and shipping radioactive materials will be required to attend annual training commensurate with their assigned duties. This training will be in conformance with NRC IE Bulletin 79-19, "Packaging of Low Level Radioactive Waste for Transport and Burial" (Ref. 25).

The training will include instructions in all applicable Federal, State and local regulations and burial site requirements for classification, packaging, loading and shipping of radioactive materials.

3.2.4.9 Radiation Protection Supervisor Training and Qualification

All project Radiation Protection supervisors will be trained and qualified in their positions. Radiation Protection training procedures will include supervisor job descriptions which specify qualification criteria and responsibilities. This qualification and training program will include the following:

1. Review and verification of resumes by the Project Radiation Protection Manager to ensure that the supervisor has sufficient supervisory and technical experience in the area(s) for which they will be responsible. Supervisors designated as the alternate to the Project Radiation Protection Manager will meet the education and experience requirements established in NRC Regulatory Guide 1.8.
2. Training in Radiation Protection and decommissioning procedures associated with their area of responsibility.
3. Training in emergency response duties.
4. Periodic professional Radiation Protection training in the form of refresher courses, retraining, conferences or continuing education which enable Radiation Protection supervisors to keep abreast of current developments in the field.

3.2.4.10 Project Radiation Protection Manager Training and Qualification

The qualification and training program for the Project Radiation Protection Manager will include:

1. Verification of prior education and experience as required by NRC Regulatory Guide 1.8.
2. Orientation on specific decommissioning plans, management organization and decommissioning project procedures.
3. Orientation on the specific design, systems and radiological controls of the facility.
4. Training in emergency response duties.

5. Periodic professional Radiation Protection training in the form of refresher courses, retraining, conferences or continuing education which enable the Project Radiation Protection Manager to keep abreast of current developments in the field.

3.2.4.11 PSC Senior Health Physicist Training and Qualification

All PSC Senior Health Physicists will be trained and qualified in their positions. Radiation Protection training procedures will include job descriptions which specify qualification criteria and responsibilities. This qualification and training program will include the following:

1. Review and verification of resumes by the PSC Radiation Protection Manager to ensure that the Senior Health Physicists have sufficient technical experience in the area(s) for which they will be responsible. Senior Health Physicists designated as the alternate to the PSC Radiation Protection Manager will meet the education and experience requirements established in NRC Regulatory Guide 1.8.
2. Training in Radiation Protection and decommissioning procedures associated with their area of responsibility.
3. Training in emergency response duties.
4. Periodic professional Radiation Protection training in the form of refresher courses, retraining, conferences or continuing education which enable Senior Health Physicists to keep abreast of current developments in the field.

3.2.4.12 PSC Radiation Protection Manager Training and Qualification

The qualification and training program for the PSC Radiation Protection Manager will include:

1. Verification of prior education and experience as required by NRC Regulatory Guide 1.8.
2. Orientation on specific decommissioning plans, management organization and decommissioning project procedures.
3. Training in emergency response duties.

4. Periodic professional Radiation Protection training in the form of refresher courses, retraining, conferences or continuing education which enable the PSC Radiation Protection Manager to keep abreast of current developments in the field.

3.2.4.13 Radiation Protection Training Records

The Radiation Protection Training Manual will specify the types of Radiation Protection training records to be maintained. Records will be maintained in accordance with regulatory requirements and company procedures. These training records will typically include:

1. Final written examination grade.
2. Final practical factors evaluation results.
3. Description of training completed satisfactorily, references to pertinent lesson plans, course outlines, syllabuses and other subject-specific descriptive information.
4. Documents indicating qualification verification (i.e., qualification cards).

3.2.5 Dose Control

Radiation dose control is accomplished by controlling sources of radiation, controlling access to areas containing radioactive materials, measuring radiation exposures of workers, establishing exposure limits for workers and maintenance of an ALARA program. Specific elements of dose control include the following:

- ALARA Program
- Radiation Work Permits
- External dosimetry
- Internal dose control and monitoring
- Administrative dose control
- Area Definitions and Postings
- Respiratory protection

3.2.5.1 ALARA Program

All activities at Fort St. Vrain involving radiation and radioactive material shall be conducted such that radiation dose to employees, contractors, and the general public are maintained ALARA. Project management will establish specific goals and objectives for the Fort St. Vrain decommissioning project ALARA Program. The ALARA Program will be based on the guidance provided in Regulatory Guides 8.8

and 8.10 (References 16 and 17) to the degree applicable for decommissioning and dismantlement.

3.2.5.1.1 <u>ALARA Program Organization and Responsibilities</u>

The PSC Radiation Protection Manager and the Project Radiation Protection Manager are cooperatively responsible to coordinate ALARA Program development and implementation. Specific responsibilities of the PSC Radiation Protection Manager and the Project Radiation Protection Manager, including their ALARA Program responsibilities, are listed in Section 3.2.2.

The actual implementation of specific ALARA actions, as incorporated into daily work activities, will be the responsibility of each individual manager, supervisor and worker.

The ALARA Program will be supported by two levels of management providing oversight and direction. The working level of the ALARA Program organization will be the ALARA Committee and will be comprised of managers and representatives of various crafts at the supervisory level. The ALARA Committee reports to the Decommissioning Program Director. Management oversight of the ALARA Program will be provided by the Decommissioning Safety Review Committee.

The primary responsibilities of the ALARA Committee will be to review decommissioning work activities for effective dose reduction techniques and conformance with the Radiation Protection Program policies and procedures. Additional responsibilities will include reviewing the methods used for decommissioning, providing guidance and solutions for dose reduction and the approval of special equipment and procedures used to reduce and maintain the overall project radiation dose ALARA.

The ALARA Committee will be co-chaired by the PSC Radiation Protection Manager and the Project Radiation Protection Manager. The other members of the committee will be designated managers and/or supervisors involved in the decommissioning project. Members will have the appropriate authority and responsibility necessary to implement an effective ALARA Program. The ALARA Supervisor will implement the ALARA Program, serve on the ALARA Committee as a non-voting member and hold the position of Committee Secretary.

ALARA Committee objectives will be as follows:

1. Ensure that ALARA policy, philosophy, commitments and regulatory requirements are integrated into all appropriate decommissioning work activities.
2. Establish overall ALARA Program goals for the decommissioning project.
3. Review and approve the ALARA budgets for specific project activities/tasks.
4. Review and evaluate individual and collective doses to determine the degree of success being achieved by the ALARA Program.
5. Initiate corrective actions, as necessary, to ensure accomplishment of ALARA Program objectives and goals.
6. Review and evaluate project activities/tasks that have dose estimates above specified action levels.
7. Ensure that necessary resources are provided to achieve the goals and objectives of the ALARA Program.
8. Coordinate efforts of various functional groups (e.g., engineering, operations, technical support and Radiation Protection) to maintain radiation dose ALARA.

The next higher level of management involvement in the ALARA Program will be the Decommissioning Safety Review Committee. This committee will review, evaluate and approve major decommissioning operations dealing with radioactive materials and radiological controls. This committee will also provide overall direction to the ALARA Committee for decisions involving financial solutions, administrative policy and decommissioning methods.

3.2.5.1.2 ALARA Training and Instruction

Commitment to the principles of ALARA will be reflected in all radiation protection training. Training courses will be evaluated by the Project Radiation Protection Manager and approved by the PSC Radiation Protection Manager to ensure that ALARA principles are incorporated into lesson plans.

3.2.5.1.3 Engineering Controls

Engineering controls is the term used for the general class of devices and associated methods used to reduce the exposure of personnel to radiation and radioactive material. Engineering controls typically include temporary shielding, engineering access controls, process instrumentation, control of airborne radiation sources, remote surveillance equipment, control of surface contamination, and other work improvement techniques.

"Temporary Shielding" will be evaluated during the planning phase for activities involving high dose rates, such as core component dismantlement and removal. The use of temporary shielding will continue to be evaluated during the implementation phase. The decision of whether to use temporary shielding will be based on considerations such as:

1. The effectiveness of providing shielding for the component (radiation source) or shielding between the source and the worker (shadow shields).
2. The effectiveness of providing partial shields such as for "radiation streaming", or "high level" sections of piping, drains, sumps, etc.
3. Estimated dose savings by the use of shielding.
4. Estimated dose expended during shielding installation.
5. Projected cost of installing and removing shielding, including the cost due to delay of project, if applicable.

"Engineered Access Control" will be evaluated and used to limit access to High Radiation Areas where it is not practical to provide continuous positive control (e.g., Radiation Protection Technician stationed at the ingress/egress point). Examples of engineered access controls include:

1. Inaccessible barriers
2. Locked gates or doors
3. Barriers with flashing light(s)
4. A combination of the above

"Process Instrumentation" for systems (e.g., the PCRV water filtration system, radwaste processing systems, etc.), will be reviewed by the Project Radiation Protection Staff for instrument location and layout, including such concerns as:

1. General accessibility and associated radiation exposure.
2. Potential radiation exposure due to operation of the system.
3. Potential radiation exposure due to servicing and maintaining the system/process.

"Control of Airborne Radiation Sources" will be considered for work activities that have the potential for producing airborne radioactivity (e.g., cutting and grinding operations). Engineering controls to confine and/or control the source will be evaluated. Examples of engineering controls for airborne radiation sources include:

1. Existing plant ventilation/filtering systems.
2. Auxiliary ventilation/filtering systems for contaminated components and for machining and grinding.
3. Contamination control containments.
4. Purification systems.
5. Decontamination equipment.
6. Wet handling of highly contaminated equipment, such as PCRV components.
7. Air sampling/monitoring instruments located to provide a quick indication of elevated airborne levels.

"Remote Surveillance Equipment" (e.g., TV monitors, audio equipment, "as-installed" photographs and radiation monitors) will be evaluated for use during decommissioning activities that have the potential for producing high radiation or airborne radioactivity areas. Such equipment, when used, will allow personnel to evaluate the radiation or airborne levels, the layout of the area and activities in the area without being exposed to the radiological conditions of the affected area.

"Control of Surface Contamination" will be evaluated and used to control and contain the spread of contamination and prevent the spread of radioactive materials to uncontrolled areas. Examples of methods used to control the spread of contamination include:

- Containments
- Surface decontamination
- Leak control
- Air curtains and plastic coverings
- Glove bags
- Drip pans
- Strippable coatings

Other engineering controls that may promote work efficiency and reduce radiation dose to workers will be evaluated. Examples of other engineering controls include:

- Adequate lighting
- Adequate ventilation

- Adequate working space
- Ease and quickness for installing/dismantling temporary equipment, such as scaffolding & insulation
- Means of accessibility, such as working platforms, cat walks, and fixed ladders
- Removal of components to remote areas, where shielding and special tools are available in low radiation areas

3.2.5.1.4 ALARA Program Goals

Each major task/activity during decommissioning will be assigned an ALARA (person-Rem) goal. Each goal will be established based on the anticipated person-hours, dose rates and dose saving methods employed. The goals will not be an estimate of the dose to be expended, considering the dose rates and person-hours, but will be a goal that requires planning and proper execution of dose-saving methods on the part of all personnel involved in the task.

The estimated dose from major tasks will be combined with minor and/or routine tasks (if necessary) to determine the annual and total dose goals for the Decommissioning Project.

Major activities/tasks will be evaluated during the planning phase. This will involve input from design engineers, the Project Radiation Protection Staff and implementing supervisors to determine an estimated ALARA goal for the task. ALARA goals will be adjusted as additional job reviews, ALARA reviews and task modifications are performed.

Prior to the start of specific activities/tasks, ALARA goals will be reviewed and approved by the ALARA Committee and the Decommissioning Safety Review Committee, as appropriate.

3.2.5.1.5 ALARA Job Reviews

ALARA job reviews will be conducted for each activity/task as an integral part of Radiation Work Permit preparation. (See Section 3.2.5.4) The scope and detail of the ALARA job review, and the individuals designated for review will depend on the complexity of the task, the expected radiological conditions and the estimated doses.

Cost benefit will be considered in ALARA job reviews where significant costs may be incurred to reduce the estimated dose. These costs may be incurred by such factors as temporary shielding, remote tools and/or surveillance, ventilation controls, decontamination, etc.

Pre-job briefings and mock-up training will be conducted for complicated or high dose jobs, as required by implementing procedures and/or the specific Radiation Work Permit.

In-progress ALARA reviews will be required if radiological conditions change significantly, are unexpected or if the projected accumulated dose differs significantly from the estimated value (e.g., greater than twenty-five percent).

Post-job ALARA reviews will be conducted at the completion of activities/tasks for which significant ALARA planning measures were required or accumulated person-Rem exceeded specified action levels.
Post-job reviews will encourage all personnel involved in the project activities to provide input regarding the effectiveness of methods used to perform the work. This input will be evaluated for future work activities to improve work conditions and maintain worker dose ALARA.

3.2.5.1.6 ALARA Work Practices

In order to complete each individual task and minimize personnel exposures, the following work practices, as a minimum, will be implemented:

1. Pre-job briefings will be held with craft and Radiation Protection personnel to assure that ALARA practices have been adequately factored into the work packages for completing tasks.
2. Personnel exposures will be monitored on a regular basis for potentially high exposure tasks to identify any irregularities that may indicate excessive personnel exposures. In the event that an irregularity is found it will be investigated immediately and corrective actions implemented.
3. Tag lines will be attached and used when rigging and lifting high exposure rate components (e.g., steam generators) from the PCRV to keep workers as far from the source as possible.

4. Only essential personnel will be allowed on the refueling floor or work area when high exposure rate components are being removed. Casual observers will not be permitted.
5. Bagging techniques will be developed to minimize exposure.
6. Long-handled wipe tools will be used when appropriate to wipe down wet components removed from the PCRV.
7. Shadow shields (lead blanket curtains or equivalent) will be used, where appropriate, to reduce radiation fields at the work stations.

The radiation levels from activated structures inside the PCRV will be measured as segments of the reactor are dismantled. The measurement of exposure rates at the individual work stations will be performed and compared with calculation prior to commencing each individual task. Adjustments will be made in exposure projections as necessary. Temporary shielding will be used at the work stations to minimize personnel exposures based on actual exposure rate measurements.

3.2.5.1.7 Administrative Controls

During the Project planning phase, specific work packages and the project as a whole, will be reviewed by the Radiation Protection staff to ensure that adequate administrative controls and Radiation Protection hold points are included. The scope and detail of the controls and hold points will be a function of the estimated radiation dose rate levels and the complexity and duration of the work activities.

Work activities that have the potential for high exposure rates (e.g., greater than 500 mR/hr) or high estimated dose for the task (e.g., greater than 10 person-Rem) will be reviewed and approved by the ALARA Committee.

Work activities, that if not performed strictly in accordance with administrative controls could potentially produce personnel exposure in excess of regulatory limits, will require review and approval by the PSC Decommissioning Program Director and/or the Decommissioning Safety Review Committee.

3.2.5.1.8 ALARA Suggestion Program

An ALARA suggestion program will be developed to ensure that all personnel have the opportunity to participate in identifying potential ALARA concerns or recommendations to support the project's dose reduction efforts. All suggestions will receive review and response by the Radiation Protection staff to maintain open communications on ALARA issues.

ALARA Suggestion Program implementing procedures will address: submitting ALARA suggestions; reviewing, evaluating and approving ALARA suggestions; and implementing and tracking ALARA suggestions.

3.2.5.1.9 ALARA Program Evaluation and Appraisal

ALARA Program effectiveness will be monitored and evaluated on a continuing basis to determine appropriateness and effectiveness. A variety of feedback mechanisms will be in effect to provide information for these evaluations. These mechanisms will include pre-job, on-going and post-job reviews, and trending of Radiation Protection Performance Indicators.

ALARA Performance Indicators will be used to monitor and trend a variety of indicators to identify those areas where the ALARA program is performing effectively and also where problems may be occurring in the Radiation Protection Program. The need for additional ALARA Performance Indicators will be identified through the review of ALARA suggestions, radiological occurrences, radiological incidents, etc. Typical ALARA Performance Indicators include, but are not limited to:

- Respirator usage as compared to the number of entries to controlled areas
- Personnel contaminations
- Collective and individual doses
- Number of positive bioassay results

3.2.5.2 Occupational Exposure Estimate

The total cumulative occupational exposure for the entire decommissioning project is estimated to be 433 person-Rem, due almost entirely to PCRV dismantlement and associated waste handling activities. The estimated cumulative radiation exposure for each major activity where the potential for worker exposure exists is provided in Table 3.2-1. The estimate of Occupational Radiation Exposure (ORE) for the decommissioning of Fort St. Vrain was based on the tasks outlined in PDP Section 2.3.

The 433 person-Rem total exposure estimate will be used for planning purposes only and is not considered to be a restricting upper limit. Actual exposures will be controlled in accordance with ALARA principles (see Section 3.2.5.1). If projections indicate that the 433 person-Rem estimate may be exceeded during the project, written notification will be provided to the Decommissioning Safety Review Committee (See Section 2.4.9) for assessment.

3.2.5.2.1 Assumptions Used to Determine ORE

The scheduled task durations identified in PDP Section 2.3.5 formed the basis for estimating worker time spent in the radiation environments. The estimates of the radiation levels were based on calculated activities for each activated component inside the PCRV and on estimated plateout activities for contaminated components. In determining personnel exposures for dismantling the PCRV, it was assumed that approximately 50% of the task duration time would require workers to be in the radiation fields. In addition, the radiation fields are expected to vary for tasks performed during dismantlement of the PCRV after the water is removed from the PCRV. Consequently, "crew averaged" radiation fields were determined by distributing the total exposure estimated to complete the task uniformly among the crew.

The general area background dose rates associated with the BOP systems are expected to be less than 1 mR/hr. However, it is anticipated that individual components such as filter housing, valves and piping that process fluids from the PCRV may have radiation levels that exceed this dose rate. Therefore, the exposures estimated for the individual systems are based on the removal of some components that exceed 1 mR/hr and are expected to result in the exposures estimated for the individual systems as listed.

The radiation environments for packaging and shipping radioactive waste are expected to vary over the course of decommissioning operations. The use of shielded transfer containers, the HSF, long-handled tools and tag lines for handling radioactive materials and components will be among methods used to minimize worker exposures. In addition, shielded shipping containers will be used for packaging radioactive waste that exceeds dose limits for shipping. It is expected that worker exposures recorded for the packaging and shipping of radioactive waste will result from working extended periods in low radiation backgrounds. Consequently, personnel exposure estimates for BOP and radioactive waste processing are based on experience from similar nuclear industry projects rather than direct dose calculations.

Actual measurements of radiation levels at individual work stations will be performed prior to commencing each individual task. These measurements will determine the actual radiation environment and the ALARA practices required to complete the task. The design of the PCRV Shield Water System and the design of tooling will provide flexibility in radiation protection. The adjustment of the PCRV water level and the use of underwater manual and semi-remote operating tooling will minimize personnel exposures.

3.2.5.2.2 <u>ORE Calculational Methodology</u>

The detailed breakdown of each major activity to the WBS level and the associated projected exposures are also provided in Table 3.2-1. The methods for calculating the ORE consisted of:

1. For PCRV operations only, the time workers spend at the work station radiation environment was assumed to be 50% of the time scheduled to complete the task. The total scheduled time was used in the BOP and radwaste handling areas since radiation levels are expected to be low in comparison to the PCRV internals.
2. The "crew averaged" radiation levels were determined by assuming the total exposures estimated for completing a task would be uniformly distributed among the crew.
3. The majority of graphite reflector blocks will be removed without the use of shielded transfer casks.
4. The highly activated components, such as boronated pins and large side reflector blocks, will be loaded into shielded containers under water and transferred to the HSF for processing.
5. The PCRV water level will be maintained such that the general area exposure rate on the work platform will typically be less than 2 mR/hr.

6. Personnel exposures for Health Physics (HP) and Quality Assurance (QA) coverage for PCRV decommissioning operations were assumed to be 10% (for HP) and 1% (for QA) of craft personnel exposures for a total of 11%. An estimate of 10% (for HP) was assumed for BOP and radioactive waste packaging operations. Radiation exposure for QA coverage of BOP and radioactive waste operations was assumed to be minimal.
7. The "crew averaged" dose rate for BOP systems decommissioning is expected to be less than 1 mR/hr, resulting in minimal exposures. The exposures listed for each system are estimates based on the potential for worker exposure at work stations in proximity of the PCRV shield/pipe penetrations and contaminated components.
8. The estimate of exposures for radwaste handling were based on experience gained from packaging and shipping radwaste in other operating nuclear plants. The dose rates are expected to vary. Sources will be shielded to acceptable levels to meet shipping requirements.
9. The workers will be trained in ALARA principles to minimize occupational exposures.

3.2.5.3 Administrative Dose Control

Administrative radiation dose controls will be implemented to ensure personnel do not exceed regulatory dose limits, to ensure an equitable distribution of dose among project workers with similar jobs and to ensure that the collective dose to workers is ALARA.

The Radiation Protection Manual implementing procedures will detail administrative dose control requirements and activities. These procedures will include, but not be limited to, the following elements:

1. An approval system by various levels of supervision and management which controls both planned and actual doses to individuals as they progressively (incrementally) approach regulatory or established administrative limits.
2. Permission to exceed the lower administrative limits (e.g., 500 mRem/quarter and/or 1000 mRem/year whole body dose for adults) will require approval of the individual's supervision and the Project Radiation Protection Manager.

3. Permission to exceed the higher administrative limits (e.g., 2000 mRem/year whole body dose for adults) will require approval of the individual's supervision, the Project Radiation Protection Manager, the PSC Radiation Protection Manager, and the PSC Decommissioning Program Director.
4. The Project Radiation Protection Manager's approval to exceed an administrative limit will be based on a determination that the dose to be received by the individual is ALARA.
5. Administrative controls will be in place and all personnel will be instructed and trained in these controls to ensure that any activity can be and will be stopped, if necessary, to re-evaluate the evolution and ensure no excessive doses are incurred.
6. Specific administrative limits and guidelines for exposure to the unborn, visitors, minors, etc.
7. Guidelines & policies governing emergency dose authorization and methods for handling overexposures.

3.2.5.4 Radiation Work Permits

Radiation Work Permits (RWPs) will be used for the administrative control of personnel entering or working in areas that have, or potentially have, radiological hazards present. RWPs will summarize the Radiation Protection controls established as part of job planning and will be detailed enough to deal with changing (or potentially changing) radiological conditions expected during the course of the work. RWPs will use current radiological survey information when establishing dose control measures and will specify any special survey requirements prior to, during and after the work activity.

An RWP will be required for the following:

1. Entry into or work in a radiologically controlled area.
2. Entry into Radiation, High Radiation, or Radioactive Materials Areas located outside a radiologically controlled area.
3. Activities involving equipment, controls, or instrumentation containing or suspected of containing radioactive material which are located outside a radiologically controlled area.
4. When determined by Radiation Protection that radiological controls in the form of an RWP are appropriate.

The RWP process will provide a systematic method to evaluate radiological conditions under which decommissioning work activities will be accomplished, specify radiation protection requirements and ensure that required worker briefings are given. Acceptable radiation work practices will be described and sufficient Radiation Protection Technician coverage assigned to assure worker protection and ensure that worker dose is maintained ALARA. RWPs will also provide a method to record doses for each individual by major job or task. The recorded RWP doses will also allow dose trend analysis and will frequently provide workers with their current dose status. The specific methods used for dose accountability and trending will be prescribed in the Radiation Protection Manual implementing procedures. Appropriate management approvals for RWPs involving significant projected total dose will be established in conjunction with the ALARA Program. RWP preparation and approval will be specified in Radiation Protection implementing procedures.

RWPs will typically provide the following information:

1. Description of job or activity to be performed.
2. Anticipated radiological conditions including, as applicable, contamination levels, radiation levels, airborne radioactivity levels.
3. Reference to, or a copy of, dose rate and contamination level survey maps.
4. Number and identification of personnel assigned to the job or activity (if appropriate).
5. Monitoring requirements during the job, such as constant radiation protection coverage, intermittent coverage, air monitoring, etc..
6. Special instructions and equipment to minimize exposure to radiation and contamination.
7. Protective clothing and equipment requirements.
8. Personnel dosimetry requirements (e.g., whole body, extremity).
9. Estimated exposure time and dose (person-Rem) to complete the task.
10. Actual exposure time, dose and other information obtained during the task.
11. ALARA pre-job briefing elements.

RWPs will normally be initiated by the group responsible for the job to be performed. The RWP request will typically include the following information:

1. Job description, work package number and if appropriate the purpose of the task.

2. Location(s) of work.
3. Estimated time to complete the job including, if appropriate, crew size and work location.
4. Exposure estimate.

RWPs will be documented in a legible and easy to comprehend format, and will be readily accessible for workers' review. All personnel assigned to a RWP will be required to read and sign the RWP verifying they will comply with its requirements. It will be expected that radiation workers be in strict compliance with RWP requirements. Willful or habitual disregard of RWP instructions will be cause for disciplinary action.

RWPs will be classified as Standing or Special, as determined by Radiation Protection Supervision. Standing RWPs will be used for the performance of routine activities such as observation, inspection, operator rounds or laundry operations where radiological conditions are stable. The Project Radiation Protection Manager will approve all Standing RWPs. Job specific RWPs will be used for the performance of a defined activity in specific locations.

Radiation Protection Manual implementing procedures will specify how RWPs will be maintained and retained. These records will typically include copies of RWPs, RWP sign-in logs and ALARA review documentation.

3.2.5.5 Area Definitions and Postings

Radiological postings will be provided at the entrance and boundaries of radiological areas to advise workers of radiological hazards. Methods will also be provided to clearly distinguish radioactively contaminated systems and indicate any special precautions required for work on such systems. Informational postings may also be used to provide additional radiological instructions to workers. It is the responsibility of each worker to observe the radiological postings and comply with the indicated requirements.

3.2.5.5.1 Radiological Postings

A Radiation Area is any area, accessible to personnel, in which there exists radiation, originating in whole or in part within licensed material, at such levels that a major portion of the body could receive in any one hour a dose in excess of 5 mRem, or in any 5 consecutive days, a dose in excess of 100 mRem. Radiation Areas will be posted "CAUTION -RADIATION AREA". Radiation area boundaries will be designated by the use of barriers, walls, ropes, markings and/or signs.

A <u>High Radiation Area</u>, as defined in the Decommissioning Technical Specification 5.8.1, is an area where whole body dose rates exceed 100 mRem/hr at 45 cm (18 inches) from the radiation source. These areas will be posted "CAUTION - HIGH RADIATION AREA". High Radiation Areas will be barricaded and conspicuously posted and entrances will be controlled by requiring a Radiation Work Permit. Personnel permitted to enter High Radiation Areas will be provided with or accompanied by one or more or the following:

1. A survey instrument which continuously indicates the area dose rate.
2. A radiation monitoring instrument which continuously integrates the radiation dose rate in the area and alarms when a preset cumulative dose is received (e.g., Digital Alarming Dosimeters (DADs)). Entry is permitted only after the area dose rate has been made known to personnel.
3. A Radiation Protection Technician equipped with a survey meter. The Radiation Protection Technician will be responsible for performing radiological monitoring at the frequency specified on the Radiation Work Permit and for providing positive control over the activities within the area.

As specified in the Decommissioning Technical Specifications, radiation areas accessible to personnel with radiation levels greater than 1000 mR/hr at 45 cm (18 inches) from the radiation source or from any surface that the radiation penetrates, in addition to the requirements for High Radiation Areas identified above, will be provided with locked enclosures to prevent unauthorized entry into such areas. Keys to such areas will be under the administrative control of Radiation Protection supervision. High radiation enclosures will remain locked except during periods of access by personnel under an approved RWP which will specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in the area. In lieu of the stay time specification of the RWP, continuous Radiation Protection Technician surveillance (either direct observation or remote by use of closed circuit TV cameras) will be used to provide positive control over the activities within the area.

Certain areas accessible to personnel with radiation levels of greater than 1000 mR/hr may be located within larger areas where no enclosure exists or can be reasonably constructed for purposes of locking individual areas. In this case, the area will be roped off, conspicuously posted, and a flashing light will be activated as a warning device.

A Contamination Area is an area accessible to personnel with loose surface beta-gamma radioactivity in excess of 1000 dpm/100 cm^2 or loose surface alpha radioactivity exceeds 20 dpm/100 cm^2. Contamination Areas will be conspicuously posted with the radiation symbol and the words "CAUTION - CONTAMINATION AREA". Contamination Area boundaries will be designated by use of barriers, walls, ropes, markings, signs and/or step off pads.

An Airborne Radioactivity Area is an area where airborne radioactivity is present, at concentrations greater than MPC, or 10 MPC-hours over 40 hours. Airborne Radioactivity Areas will be conspicuously posted at the entrance with "CAUTION - AIRBORNE RADIOACTIVITY AREA". Airborne Radioactivity Area boundaries will be designated by use of barriers, walls, ropes, markings and/or signs.

A Radioactive Materials Area is an area or room in which radioactive material is used or stored in an amount exceeding 10 times (100 times for uranium and thorium) the quantity specified in 10 CFR 20. Radioactive Material Areas will be posted "CAUTION -RADIOACTIVE MATERIALS AREA". Radioactive Materials Area boundaries will be designated by the use of barriers, walls, ropes, markings and/or signs.

A Radiologically Controlled Area (RCA) is defined as any of the above defined areas, including a Radiation Area, High Radiation Area, Contamination Area, Airborne Radioactivity Area, or Radioactive Materials Area. RCA boundaries will be defined by the use of barriers, walls, walls, ropes, and markings. The boundaries will be clearly marked with posted signs that identify the type(s) of RCA within the boundaries, and entrance and exit points will normally be posted.

3.2.5.5.2 Informational Postings

A Hot Particle Area will be established to identify areas where discrete particles with high specific activity are located. Hot particle areas will be contained and posted within a contaminated area as "HOT PARTICLE AREA".

A Hot Spot is where localized dose rates near an item are much greater (e.g., five times) than the general area whole body dose rates. These locations are posted "CAUTION - HOT SPOT".

A Restricted Area is any area to which access is controlled for purposes of protection of individuals from exposure to radiation and radioactive material.

An Unrestricted Area is any area to which access is not controlled for purposes of protection of individuals from exposure to radiation and radioactive materials. Except as authorized per 10 CFR 20, radiation levels in "Unrestricted Areas" will not exceed levels that, if an individual were continuously present, would result in an individual receiving in excess of 2 mRem in any one hour or 100 mRem in any seven consecutive days.

A Low Dose Waiting Area (LDWA) is established to identify areas where dose rates are lower than other locations within the work area. When practical, workers should be directed to remain in a LDWA unless they are actually needed at the work location. LDWAs will be posted; "LOW DOSE WAITING AREA".

3.2.5.6 External Dosimetry

3.2.5.6.1 General Considerations

External radiation dose monitoring will be accomplished through the use of thermoluminescent dosimeters (TLDs) and self-reading dosimeters (SRDs) or digital alarming dosimeters (DADs). The official record of external dose to beta and gamma radiations will normally be obtained from TLDs. SRDs or DADs will be used as a means for tracking dose between TLD processing periods and may also be used as a back-up to the TLD. TLDs will be processed at a frequency to ensure personnel dose limits are not exceeded.

A contract dosimetry service will supply TLDs during the decommissioning project. The dosimetry laboratory will be accredited by the National Voluntary Laboratory Accreditation Program (NAVLAP) for the radiation type and energy expected to be monitored. The laboratory shall participate in a testing program as described in ANSI N13.11, "Criteria for Testing Personnel Dosimetry", 1983 (Ref. 26). Self-reading dosimeters will meet the requirements specified in NRC Regulatory Guide 8.4, "Direct-Reading and Indirect-Reading Pocket Dosimeters", (Ref. 27) February 1973.

3.2.5.6.2 Monitoring Whole Body Dose

All project workers will be required to wear external radiation monitoring devices whenever they enter radiologically controlled areas. SRDs or DADs will be read prior to their use and periodically thereafter by the wearer. If a SRD is off-scale or lost under conditions such that a high dose was possible, the individual's TLD will be promptly processed and the individual will be denied access to radiologically controlled areas. The TLD and SRD or DAD will normally be worn on the trunk of the body between the neck and waist in close proximity to each other. Under certain conditions, where the chest or trunk may not be the location of highest whole body dose, dosimetry devices may be relocated. Radiation Protection Manual implementing procedures will specify criteria for relocating whole body dosimetry.

The use of multiple whole body dosimetry will be evaluated whenever work is to be performed in a non-uniform radiation field and that portion of the body which will receive the highest dose is not easily determined. In these cases, multiple sets of dosimeters will be worn on those parts of the body expected to receive the highest dose. Guidance for conducting the evaluation and criteria for determining when multiple dosimetry is required will be provided in Radiation Protection Manual implementing procedures. RWPs will be used to communicate dosimetry requirements to the workers.

3.2.5.6.3 Monitoring Extremity Dose

Extremity monitoring devices will be used whenever is likely to receive a dose in excess of 25% of the quarterly extremity dose limit of 18.75 Rem. Guidance for evaluating the need for extremity monitoring will be provided in Radiation Protection Manual implementing procedures.

3.2.5.6.4 Monitoring Skin Dose

Monitoring of the skin of the whole body will normally be accomplished utilizing the whole body TLD. Calculation of skin dose due to contamination or "hot particles" will be performed in accordance with acceptable models and equations identified in NRC Bulletins, Regulatory Guides and in published technical literature. The methods for calculating and documenting skin dose due to contamination and "hot particles" will be provided in Radiation Protection Manual implementing procedures.

3.2.5.6.5 Dosimetry Quality Control

Periodic quality assurance checks of vendor supplied dosimetry will be conducted. These checks will be addressed in Radiation Protection Manual implementing procedures. In addition, SRD results will be compared to TLD results. Each discrepancy greater than 25% for quarterly doses over 300 mRem will be evaluated. The evaluation will include consideration of factors such as energy dependence of the device used, survey results, exposure times, doses of other personnel performing similar work, location of devices worn on the body and clerical errors.

3.2.5.7 Internal Dosimetry Control and Monitoring

3.2.5.7.1 General Considerations

Internal radiation dose is inherently more difficult to measure than external radiation dose, but is generally much easier to prevent. Therefore, the major emphasis will be placed on preventing internal radiation dose, provided it is consistent with the goal of keeping total effective dose ALARA.

The primary methods for controlling intake of radioactive material into the body will be identifying and minimizing sources of airborne radioactivity and applying engineering controls to reduce airborne radioactivity concentrations. The use of respiratory protection will serve as a secondary method of control. Administrative controls and limits will be established to minimize intakes of radioactive materials.

Radiation Protection Manual implementing procedures will be developed to conduct a routine bioassay program including criteria for the performance of bioassay, dose tracking and methods for data analysis and interpretation. The bioassay program will be based on NRC Regulatory Guide 8.26, "Application of Bioassay for Fission and Activation Products", (Ref. 28) September 1980, NRC Regulatory Guide 8.9, "Acceptable Concepts, Models, Equations and Assumptions for a Bioassay Program", (Ref. 29) September 1973 and NRC Regulatory Guide 8.32 "Criteria for Establishing a Tritium Bioassay Program", (Ref. 30) July 1988.

3.2.5.7.2 Whole Body Counting

Whole body counting will be the primary method used to determine the identity and quantity of gamma emitting isotopes in the body at any given time. Radiation workers will receive, as a minimum, a baseline and annual whole body count. In addition, personnel will receive a whole body count after a suspected intake of radioactive materials. Radiation Protection Manual implementing procedures will provide guidance on whole body counter operation, calibration and quality control.

3.2.5.7.3 Indirect Bioassay

Indirect bioassay (in-vitro) measurements will be made, as necessary, to monitor for alpha and beta emitting radioisotopes. This method of bioassay will typically be used only for isotopes which cannot be determined by whole body counting or when additional information on an intake is required. Urinalysis will be used to assess personnel intakes of tritium. Radiation Protection Manual implementing procedures will include criteria for indirect bioassay and methods for data analysis and interpretation.

3.2.5.8 Respiratory Protection Program

The Respiratory Protection Program will be established in accordance with 10 CFR 20 and consistent with the guidance of NUREG-0041 (Ref. 19). The primary objectives of the Respiratory Protection Program are personnel safety and limiting the inhalation of airborne radioactive materials. Engineering controls will be applied to minimize concentrations of radioactive materials whenever practicable. When engineering controls are not practicable, other controls such as increased surveillance, limitations of working times or use of respiratory protection equipment may be appropriate.

The Respiratory Protection Program will include the following elements:

1. A written policy statement and standard operating procedures.
2. Guidance on proper selection of equipment, based on the hazard.
3. Proper training and instruction to users.
4. Proper fitting, use, cleaning, storage, inspection, quality assurance and maintenance of equipment.
5. Appropriate surveillance of work conditions.

6. Regular inspection and evaluation to determine continued program effectiveness.
7. Program responsibility vested in one qualified individual.
8. An adequate medical surveillance program for respirator users.
9. Use of only Bureau of Mines/NIOSH - certified or NRC authorized equipment.
10. Maintenance of a bioassay program.

3.2.5.8.1 Program Administration

The Respiratory Protection Program will be administered by a Radiation Protection staff member designated by the Project Radiation Protection Manager. Qualifications of this individual will satisfy the requirements of ANS/ANSI 3.1. Program administrative responsibilities will typically include the following:

1. Provide overall program development, technical direction and the evaluation of program effectiveness.
2. Provide technical guidance for the control of airborne radiological contaminants.
3. Develop procedures, training materials and directives related to the program.
4. Conduct routine overviews of the program for compliance with policy, procedures and regulations.

3.2.5.8.2 Respirator User Qualification

Respirator user qualification criteria will include:

1. Annual examination by a physician to establish physical and psychological capabilities necessary to perform tasks using a respirator. A medical re-evaluation will be performed annually.
2. Successful completion of respiratory protection training as described in Section 3.2.4, "Radiation Protection Training and Qualification."
3. Successful quantitative fit test prior to the use of respirators requiring a facepiece-to-face seal on an annual basis.
4. No facial hair between the face and the sealing surface of the respirator and no facial hair interfering with valve function of the respirator.

3.2.5.8.3 Bioassay

Bioassay techniques will be used to determine the amount and type of radionuclides in the body as an evaluation of the effectiveness of the Respiratory Protection Program.

3.2.5.8.4 Respiratory Protection Equipment Description and Selection

The selection of respiratory protection equipment will be based on work area survey data and/or expected airborne contamination levels. The need for respiratory protection will normally be determined and prescribed by Radiation Work Permits. All work tasks in contaminated areas will be evaluated for respiratory protection requirements. Special attention will be given to the requirement for respiratory protection when the work activity/task involves any of the following operations:

- Thermal cutting
- Welding
- Concrete demolition
- Concrete scabbling
- Grinding

Respiratory protection equipment will be selected to provide a protection factor greater than that required for the expected peak concentration of airborne radioactive materials in the work area. Assigned protection factors will not exceed those specified in 10 CFR 20. If the selection of a respiratory protection device is inconsistent with the goal of keeping total effective dose ALARA, consideration will be given to alternative controls or respiratory protection equipment with a lower protection factor.

3.2.5.8.5 Supplied Air Respiratory Equipment

Breathing air may be supplied to respirators from compressed air cylinders, air compressors or the plant breathing air system. All sources of compressed breathing air will meet the requirements for Grade D breathing air as specified in ANSI/CGA G-7.1, "Commodity Specification for Air", (Ref. 31) 1989.

3.2.5.8.6 Equipment Inspection and Maintenance

Requirements and techniques for inspection and maintenance of respiratory protection equipment will be contained in Radiation Protection Manual implementing procedures. Inspection and maintenance will be performed in accordance with manufacturers' and regulatory requirements.

Respirators will be maintained and issued in a NIOSH certified configuration. The certification for a respirator will be automatically voided if the respirator is not the same in all respects as certified by NIOSH or if the respirator is not maintained in a certified condition.

Acceptable methods of cleaning respiratory equipment will be performed in accordance with manufacturers' specifications and Radiation Protection Manual implementing procedures.

3.2.5.8.7 Quality Assurance (QA)

Periodically, respirators will be randomly selected to verify they have been properly cleaned and inspected. Respirator users will also be randomly selected for whole body counts to verify program effectiveness. In addition, if there is an indication of equipment failure or improper use (e.g., positive nasal smear), the respirator user will be whole body counted.

3.2.5.8.8 MPC-Hour Tracking

Administrative and engineering controls will limit the intake of radioactive materials to levels that are ALARA. An administrative limit of 2 MPC-hours in any one day and 10 MPC-hours in any one week will be established. Radiation Protection Manual implementing procedures will provide guidance on MPC-hour determination and tracking.

3.2.6 Radioactive Material Controls

Radioactive material controls will be established to provide for control of radioactive material, prevent inadvertent release of radioactive materials to uncontrolled areas, ensure personnel are not unknowingly exposed to radiation from lost or misplaced radioactive material and minimize the amount of radioactive waste material generated during the decommissioning. Radioactive material is defined as material activated or contaminated by the operation or decommissioning of Fort St. Vrain and licensed material procured and used to support the operation or decommissioning of Fort St. Vrain (e.g., calibration sources, check sources and radiography sources).

The Radwaste Supervisor and the Radiation Protection Operations Supervisor will share the responsibility for the radioactive material controls.

Detailed radioactive material controls will be described and implemented by the Radioactive Waste and Radiation Protection Manual implementing procedures. Specific radioactive material controls include the following:

1. Receipt of radioactive material
2. Identification of radioactive material
3. Control and movement of radioactive material
4. Storage of radioactive material
5. Accountability and inventory of radioactive sources
6. Release of materials for unrestricted use
7. Control of materials entering radiologically controlled areas
8. Preparation of radioactive materials for shipment
9. Radioactive liquid and gaseous release

3.2.6.1 Receipt of Radioactive Material

Implementing procedures for the purchase and receipt of radioactive material will include, but not be limited to the following requirements:

1. Requests for radioactive materials will be submitted to the Project Radiation Protection Manager for license verification and inventory check.
2. Personnel who initiate purchase orders for radioactive materials will be required to specify that shipments be marked "Attention: Project Radiation Protection Manager".

3. Picking up, receiving and opening packages identified as containing radioactive material will be performed in accordance with 10 CFR 20. The Project Radiation Protection Manager, or designee, will be notified as soon as possible after receipt of any package. The external surface of the package will be inspected and surveyed. Shipping damages and other discrepancies will be documented and Radiation Protection Supervision notified.

3.2.6.2 Identification of Radioactive Material

Implementing procedures will be developed to specifically address the identification of radioactive material. Each container in which radioactive material is transported, stored, or used in quantities exceeding those given in 10 CFR 20, will bear a durable, clearly visible label, identifying the radioactive contents. This label will contain, as a minimum, the radiation caution symbol and the words; "CAUTION OR DANGER - RADIOACTIVE MATERIALS". The label will provide sufficient information to permit individuals handling or using the material, or working nearby, to take precautions to avoid or limit their exposure. Requirements for radioactive material labeling may be excepted if:

1. The material is uniquely identified for use as radiological protection equipment (e.g., respirators, protective clothing, etc.).
2. The material is under the direct control of personnel trained as radiation workers who are aware of the contents and the associated radiological hazards.
3. The material consists of radiological samples or sampling equipment in the custody of Radiation Protection personnel.
4. The material is packaged and labeled in accordance with DOT regulations while awaiting transport.
5. The material is contained in permanently installed equipment and/or potentially contaminated systems. Radiation level posting requirements shall remain applicable.

Radioactive material storage areas will be posted as "CAUTION - RADIOACTIVE MATERIAL AREA", in addition to postings required for radiation and contamination, as indicated in Section 3.2.5.5.

3.2.6.3 Control and Movement of Radioactive Material in the Restricted Area

Implementing procedures will be developed to specifically address the control and movement of radioactive materials within the Restricted Area. Radioactive material removed from contaminated areas will be contained, surveyed and labeled to allow appropriate control of radioactive material. Radioactive liquid samples or sources will be properly contained and will be transported by, or escorted by, Radiation Protection qualified personnel.

Materials which are to be prepared for shipment or storage will be packaged in containers suitable for shipping or storage, as applicable. The materials and packages will be surveyed for radiation and contamination levels and the package appropriately labeled to reflect those levels. External surfaces of containers or wrappings containing radioactive material will be surveyed to ensure that loose surface contamination levels meet the unconditional release criteria, unless specifically exempted by Radiation Protection Supervision.

Whenever material or equipment is transferred from one location to another location, it will meet the radiation and contamination limits of each area through which it passes or be under the control of Radiation Protection personnel.

Procedures for the control and movement of radioactive material will include, but not be limited to, the following provisions:

1. Guidelines for monitoring and handling radioactive material (e.g., quantities, geometries and estimated concentrations will be used to estimate radiation levels and container requirements).
2. Unique features will be used (e.g., yellow plastic bags, yellow and magenta tags, etc.) to clearly identify the physical and radiological parameters of the material.
3. Uncontaminated materials will be separated from contaminated materials and be labeled, as necessary.
4. Lifting and rigging equipment used for radioactive materials will be commensurate with the physical and radiological parameters of the material to ensure safe movement and ALARA practices.
5. Portable tools and equipment that are contaminated will be designated as such and will be stored in an area separate from uncontaminated tools.

6. Contaminated tools and equipment will be decontaminated and/or discarded when they reach designated fixed and loose contamination levels.

3.2.6.4 Control and Movement of Radioactive Material Outside the Restricted Area

Implementing procedures will be developed to specifically address the control and movement of radioactive materials outside the Restricted Area.

Radioactive material being transferred from a radiologically controlled area to another radiologically controlled area will be properly surveyed and contained and will be escorted by Radiation Protection personnel.

Radioactive materials removed from the Restricted Area to be shipped offsite will be packaged and labeled for shipment in accordance with DOT regulations. Radioactive materials removed from the Restricted Area that are not designated to be shipped will be:

1. Properly contained to minimize radiation levels and/or prevent spread of contamination.
2. Surveyed to determine radiation and contamination levels.
3. Labeled in accordance with 10 CFR 20 as radioactive materials, indicating the radiation and contamination levels. Adequate features and markings will be used to clarify the physical and radiological parameters of the material.
4. Inventoried (e.g., listed in a log indicating person responsible for material(s) and where material(s) will be located) prior to leaving the radiologically controlled area.
5. Controlled by Radiation Protection personnel (e.g., stored in a locked area and not used or moved without Radiation Protection personnel's permission and/or presence).
6. Radioactive material will not be stored outside of the Restricted Area unless specifically approved by the Project Radiation Protection Manager.

Radioactive material found uncontrolled and outside the Restricted Area will be brought to the immediate attention of Radiation Protection supervision. Action will be taken to ensure that the radioactive material is surveyed, labeled and properly dispositioned (e.g., returned to the Restricted Area). The incident will be investigated by the Project Radiation Protection Manager and corrective action(s) initiated.

3.2.6.5 Storage of Radioactive Material

Implementing procedures will be developed to specifically address storage of radioactive materials. Interim radioactive material storage and liquid processing areas will require a safety evaluation to ensure compliance with NRC Generic Letter 81-38.

Access to radioactive material storage areas located inside the Restricted Area will be controlled by Radiation Protection personnel. Storage and processing of materials will be consistent with analyzed activities and radiation levels. Storage areas will be regularly surveyed and inventoried.

Temporary radioactive material storage areas within the Restricted Area will be designated by Radiation Protection personnel and posted in accordance with Section 3.2.5.5. Consideration for establishing temporary radioactive material storage areas will include:

1. The number of temporary storage areas and length of storage time in these areas will be minimized.
2. Radioactive material storage areas and surrounding areas will be surveyed on a routine basis.
3. Radioactive material will not be stored outside except for short periods during transit, or if packaged in accordance with DOT requirements while awaiting shipment.

3.2.6.6 Accountability and Inventory of Radioactive Material

Implementing procedures will specifically address the accountability of radioactive materials. Radioactive material storage areas will be controlled and periodically surveyed. The status of radioactive material storage areas will be periodically reviewed and include:

1. A list of storage areas to evaluate the continued need for the storage areas and/or the materials and the types and radiological parameters of the areas (e.g., high radiation material, contaminated material, activated material, etc.).

2. Inspection of materials stored, to evaluate the status of materials and area controls (e.g., physical condition of containers, access control, posting, etc.).

Radioactive sources used for calibration, standardization and instrument checks will be received as indicated in Section 3.2.6.1. Implementing procedures will specifically address inventory and accountability requirements. Sources will be inventoried at least semiannually, taking into consideration: radioactive decay; comparison to Certificates of Calibration; and use and disposal of liquid and gaseous sources. The results will be documented and reported to Radiation Protection supervision.

Sealed sources will be controlled in accordance with 10 CFR 30 through 34. Sealed sources will be leak tested with the frequencies and criteria for results based on activity level and type of source. The test will be documented and will be capable of detecting a minimum of 0.005 microCurie of contamination. If leakage is detected based on established criteria, the Project Radiation Protection Manager will be notified and reports will be submitted as indicated in Section 3.2.10.

Sources will be maintained, locked or otherwise controlled when not in use. A source usage log will be maintained. If a source can not be accounted for during a periodic inventory or at any other time, actions will be taken to include the following:

1. Evaluate the physical and radiological characteristics of the missing source and evaluate potential hazards to radiation workers and the public.
2. Report the missing source as required by applicable implementing procedures and State and Federal regulations.
3. Initiate investigations to locate the source and/or to determine the reason for loss.
4. Prepare Radiological Occurrence report(s).

3.2.6.7 Special Controls

Special circumstances may arise during the decommissioning of Fort St. Vrain that will require special handling considerations for radioactive materials. Examples of radioactive materials requiring special handling are radiography sources and highly radioactive waste (e.g., greater than 10 R/hr gamma radiation levels).

Receipt of "special" types of materials/sources require the following actions:

1. Notification of the Project Radiation Protection Manager prior to bringing the radioactive material/source inside the Restricted Area.
2. Development and approval by the Project Radiation Protection Manager of a plan of action for control, safe storage and release of the radioactive material/source (e.g., use existing procedures, develop temporary procedures, etc.) prior to bringing it inside the Restricted Area.
3. Notification of the PSC Shift Supervisor prior to use of a source (e.g., radiography).

Special nuclear material, fissile material, and highly radioactive waste will be addressed on a case-by-case basis in accordance with administrative procedures and applicable regulations. Specific control, accountability and storage, if applicable, will be specified by the Project Radiation Protection Manager with the cognizance of the PSC Radiation Protection Manager. These details will be addressed by the Radiation Work Permit Program and/or temporary procedures and action plans, and will include the following considerations:

1. Use of special containers and/or shielding.
2. Use of special rigging and lifting devices for moving the radioactive material/source including any containers as applicable.
3. Temporary evacuation of non-essential personnel and suspension of activities in the area.
4. Temporary implementation of additional security measures.
5. Other special considerations for ALARA purposes as indicated in Section 3.5.1.

3.2.6.8 Release of Materials for Unrestricted Use

Procedures for the release of materials and equipment from radiologically controlled areas will be developed. Material will not be released for unrestricted use if it contains detectable amounts of radioactive material. Instrumentation, counting times and survey techniques will be selected such that detection sensitivities are consistent with the applicable guidance of:

1. NRC Circular 81-07, "Control of Radioactively Contaminated Materials," (Ref. 32) for decontaminated items (e.g., tools and

equipment) and scrap materials.

2. NRC IE Notice 85-92, "Surveys of Wastes Before Disposal from Nuclear Reactor Facilities" (Ref. 33) for monitoring of segregated dry active waste (DAW) prior to disposal to a sanitary landfill.
3. NRC Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors" (Ref. 34), for the loose and fixed surface contamination levels of items left on the site after decommissioning.
4. Colorado Department of Health, Radiation Controls Division, "Soil Contamination Guidance Policy," (Ref. 35).
5. 10 CFR 20 and the Decommissioning Technical Specifications for bulk liquids such as oil, glycol, etc..

3.2.6.9 Control of Materials Entering Radiologically Controlled Areas

Materials entering radiologically controlled areas will require the following controls:

1. Chemicals that may cause the generation of "mixed wastes" (e.g., chemicals that can potentially produce hazardous waste) will be controlled to prevent contamination with radioactive material.
2. Packing material and unnecessary items will not be allowed to accumulate in radiologically controlled areas.

3.2.6.10 Preparation of Radioactive Materials for Shipment

Packages prepared for shipment, which upon inspection do not meet DOT requirements, will be evaluated for repackaging and/or storage. Received packages which do not meet DOT requirements will be reported to the NRC as indicated in Section 3.2.10 and in accordance with DOT regulations.

Radioactive material will be packaged and transferred to, and received from, other facilities in accordance with applicable DOT requirements and implementing procedures. Radiological surveys will be conducted prior to shipping and when receiving radioactive materials to verify that the radiation and removable contamination levels are in compliance with regulatory requirements and are described on shipping labels and documentation. Examples of surveys that will be performed are:

1. Contamination surveys for packages received and packages shipped to ensure that they meet the DOT requirements for smearable contamination levels (e.g., external beta-gamma and alpha, and internal beta-gamma and alpha).

2. Radiation surveys (e.g., package contact, vehicle contact, specified distances from the package and the vehicle, and normally occupied positions in the vehicle cab) for the material and package and for the transport vehicle depending on the type of shipment (e.g., LSA, Exclusive Use LSA, etc.).

Additional elements described in the Radioactive Waste Manual implementing procedures will include, but not be limited to:

1. Sorting and segregation of materials and processing to an acceptable form.
2. Classification of the material in accordance with 10 CFR 61.
3. Receipt survey of vehicles used to transport radioactive waste.
4. Packaging, labeling and marking of material in accordance with 10 CFR, 49 CFR and Disposal Site Criteria.
5. Shipment of material in accordance with 49 CFR and 10 CFR
6. Disposal and off-site volume reduction arrangements.

Additional details on radioactive waste management can be found in Section 3.3.

3.2.6.11 Radioactive Liquid and Gaseous Release Control

Liquid and gaseous effluent releases will be monitored and controlled using installed plant equipment and guidance provided in the Decommissioning Technical Specifications and Offsite Dose Calculation Manual. Typical process controls will include:

- Sample and analyze the waste stream
- Calibrate and test instrumentation
- Monitor the waste stream
- Prepare a release permit
- Complete the release
- Record types and quantities of material(s) released

3.2.7 Surveillance

Radiological surveillance will be conducted routinely to identify radiation sources, determine radiological conditions, and comply with the requirements of 10 CFR 20. Special surveys will be scheduled by Radiation Protection Supervision as needed to evaluate radiological conditions in support of decommissioning activities. Elements of radiological surveillance program will consist of the following:

1. Routine surveys-general
2. Dose rate surveys
3. Surface contamination surveys
4. Airborne radioactivity surveys
5. Environmental sampling and analysis
6. Personnel contamination monitoring
7. Survey documentation and review

Detailed radiological surveillance requirements and activities will be described in the Radiation Protection Manual administrative and implementing procedures. These procedures will specify the types of instrumentation, survey methods, and actions required when abnormal radiological conditions are discovered.

Calibrated instrumentation will be available for the detection and measurement of alpha, beta and gamma radiation. Air sampling equipment will be available for general area and breathing zone air samples.

Survey and monitoring information will be used by the Radiation Protection staff and other support groups for:

1. Procedure development, as applicable
2. Decommissioning work packages development
3. Decommissioning engineering design criteria
4. RWP preparation
5. Radiation, contamination and airborne radioactivity trend analysis
6. Pre-job ALARA planning
7. Pre and post job briefings

Surveillance frequencies will be specified in implementing procedures with consideration given to hazards which may be encountered, potential for changing radiological conditions and frequency of occupation. Examples of surveys and associated frequencies include the following:

1. Active work areas where radiological conditions may change as a result of work being performed will normally be surveyed for radiation and contamination at least once per shift or more frequently if radiological conditions could change (e.g., upon opening a radioactive system).
2. Active exit points from contaminated areas will be surveyed for contamination at least daily and once per shift during frequent use.
3. Eating areas used by individuals who have worked in radiologically controlled areas will be surveyed for contamination at least weekly.
4. Storage areas for solid radioactive waste and irradiated/contaminated components and equipment will be surveyed weekly when material and/or personnel have entered the area.
5. All personnel and equipment exiting radiologically controlled areas will be monitored for contamination.
6. RWP's will normally specify the frequency of Radiation Protection technician coverage and surveys required (e.g., continuous, intermittent).

3.2.7.1 Routine Surveys-General

Routine radiation, contamination and airborne radioactivity surveys will be performed to evaluate radiological conditions and verify radioactive materials are being adequately controlled. Survey data will be used for job evaluations, trend analysis, ALARA pre-planning and informing personnel of radiological conditions. Action levels and associated responses will be established for abnormally high or unusual survey results. Survey results will be made available to workers entering radiologically controlled areas. Locations where routine surveys and monitoring will be conducted include, but are not limited to:

1. Active work areas where conditions may change as a result of the work being performed.
2. Entry and exit points of contaminated areas (e.g. step off pads).

3. Offices, trailers, shops and trash receptacles outside radiologically controlled areas.
4. Major pathways inside radiologically controlled areas
5. Radiation Area boundaries for verification of posting adequacy.
6. Storage areas for radioactive wastes.
7. Entrances to locked High Radiation Areas.
8. Radiation Protection and Radiochemistry laboratories.
9. Eating and break areas used by personnel who have been working in radiologically controlled areas.
10. Selected unrestricted areas, if appropriate.

Radiation Protection supervision will routinely review surveys with regard to necessity and frequency consistent with good radiological protection practices and regulatory requirements.

Routine surveys will not normally be conducted in High Radiation Areas except as directed by Radiation Protection supervision. These surveys will be coupled with, or prior to, planned work activities in those areas in order to maintain personnel exposure ALARA.

3.2.7.2 Dose Rate Surveys

Dose rate surveys will be performed to provide specific radiological information on beta and gamma radiation dose rates. These surveys will normally be performed with portable, hand held survey instruments. Dose rate information may also be obtained through the use of fixed radiation monitors. Dose rate surveys will be performed to:

1. Assess changing radiological conditions in radiologically controlled areas which are frequently occupied.
2. Identify localized hot spots.
3. Provide data for pre-job ALARA planning and RWP preparation.
4. Establish "Low Dose Waiting Areas".
5. Monitor the receipt of radioactive materials.
6. Support the packaging/shipping of radioactive waste.
7. Ensure proper release of materials and equipment for unrestricted use.
8. Monitor unanticipated spills or spread of radioactive materials.
9. Assess radiological conditions during decommissioning work (e.g., breaching of a radioactive system, PCRV dismantlement etc.).
10. Provide data for environmental monitoring.

11. Support emergency response activities.
12. Establish and verify radiation area boundaries and postings.
13. Monitor laundered/decontaminated personnel protective equipment (e.g., protective clothing, respirators) prior to reuse.

Dose rate radiation survey instruments will be calibrated to the radiation (beta, gamma) being detected to assure an accurate, consistent, reliable and predictable response to radiation levels.

3.2.7.3 Surface Contamination Surveys

Contamination surveys will be performed to provide specific radiological data on the levels of beta-gamma and alpha contamination, and will be performed to:

1. Monitor personnel and equipment exiting radiologically controlled areas to prevent the inadvertent release of radioactive material.
2. Establish boundaries of contaminated areas.
3. Support the radioactive source accountability program (e.g., source leak tests).
4. Monitor the receipt of radioactive materials.
5. Support the radwaste packaging/shipping program.
6. Ensure the proper release of materials and equipment for unrestricted use.
7. Provide data for pre-job ALARA planning and RWP preparation.
8. Determine radiological conditions during coverage of jobs with changing radiological conditions (e.g., welding, grinding radioactive system opening).
9. Provide data for environmental monitoring.
10. Support decommissioning emergency response.
11. Assess conditions following the discovery of a spill or spread of radioactive materials.
12. Survey TLD's prior to processing.
13. Detect and control "Hot Particles".
14. Monitor decontaminated personal protective equipment (e.g., respirators) prior to reuse.
15. Monitor applicable areas, such as clean waste dumps and landfills, salvage areas, warehouses, tool storage areas and contractor buildings.

16. Assist in personnel decontamination by monitoring for adequate decontamination techniques.

3.2.7.4 Airborne Radioactivity Surveys

Airborne radioactivity will be measured in areas where personnel may be exposed to airborne particulates and tritium. Representative air sampling will be performed to provide measurements during work which has the potential for the generation of airborne radioactivity. Continuous air monitors, breathing zone air samples and grab air samplers will be used for obtaining air samples. Airborne radioactivity surveys will typically be performed:

1. During work operations known or suspected to cause airborne radioactivity (e.g., grinding, welding, burning, cutting, hydrolazing, vacuuming, sweeping or use of compacting equipment).
2. During work that involves the breach of a radioactive system.
3. Upon initial entry and periodically thereafter into any area known or suspected to contain airborne radioactivity concentrations in excess of 25 percent of MPC.
4. Immediately following the discovery of a significant spill or spread of radioactive materials or whenever airborne radioactivity levels are suspected to have changed.
5. Periodically in radiologically controlled areas where the potential for airborne radioactivity exists.
6. Any time respiratory protection devices or MPC-hr accounting are used to control internal radiation exposure.
7. When continuous air monitoring is performed, high volume grab samples or breathing zone air samples will be periodically taken to verify that continuous air monitoring of the work area is representative of the breathing zone (This surveillance will also be performed for Tritium).
8. Periodically to verify the effectiveness of the Respiratory Protection Program.

Air sample counting equipment will be available to measure alpha, beta and gamma emitting radionuclides. Collection efficiencies for air sampling media will be determined. The minimum detectable activity (MDA) for air sampling equipment will be at least one (1) MPC in one (1) hour for the radioisotopes most likely to be present. Air sampling equipment will be calibrated using guidance in NRC Regulatory Guide 8.25 "Calibration and Error Limits of Air Sampling Instruments for Total Volume of Air Sampled", (Ref. 36) August 1980.

Periodic samples will be collected and analyzed to verify the adequacy of engineering controls that are used to minimize airborne radioactivity. Typical controls that will be verified are:

- Room ventilation systems
- Portable ventilation systems
- Air locks
- Ventilation hoods
- Containment devices (e.g., tents, glove boxes, etc.)

3.2.7.5 <u>Environmental Sampling and Analysis</u>

Environmental sampling and analysis will be conducted during decommissioning. The current PSC Radiological Environmental Monitoring Program (REMP) will be continued, in part, specifically tailored to determine the effect on radiological conditions of the environment due to decommissioning activities.

In addition, the Offsite Dose Calculation Manual (ODCM) will provide the methodologies to assure compliance with Fort St. Vrain Decommissioning Technical Specifications related to liquid and gaseous radioactive effluents. This program will demonstrate compliance with 10 CFR 20, 10 CFR 50 Appendix A (GDC 64) and Appendix I and 40 CFR 190.

Specific sample types and locations will be addressed in the REMP and ODCM. Typical environmental monitoring techniques that will be utilized, include:

- Area TLDs
- Soil sample analysis
- Water sample analysis
- Vegetation sample analysis

3.2.7.6 Personnel Contamination Monitoring

Adequate personnel contamination monitoring instrumentation will be available to control the spread of contamination and hot particles. Radiation Protection Manual implementing procedures will require monitoring upon exiting radiologically controlled areas and will establish acceptable methods for performing personnel frisking. Whole-body contamination monitors (e.g., Eberline PCM-1B) will be used at radiological controlled area exits, as appropriate. Sensitivities of these instruments will be set to detect contamination levels equal to or better than conventional hand frisking methods (e.g., 5000 dpm/100 cm^2).

In areas where whole-body contamination monitors are not available, a hand-held frisker will be used to monitor personnel contamination. Typical frisking techniques that will be required include:

1. Maintain frisking speed of less than 2 inches per second.
2. Maintain detector to body distance at less than 1/2 inch.
3. Pause (approximately 5 seconds) at the nose and mouth area to check for indications of inhalation/ingestion of radioactive material.
4. Pay particular attention to feet (shoes), elbows, knees or other areas with a high potential for contamination..
5. Ensure a total frisking time of greater than 2 minutes to cover at least 10% of the body.
6. Maintain background for frisking at less than 300 cpm.

When background levels are unacceptable (e.g., greater than 300 cpm) for personnel frisking, actions will be taken that include one or more of the following:

1. Move the whole body frisker and/or the hand-held frisker (and the contamination control point) to an area that has an acceptable background level (e.g., around the corner, behind a column, etc.).
2. Shield the frisking area and equipment to reduce background.
3. Frisk for gross contamination levels in the high background area, but locate the equipment for final frisking at a remote area, and provide contamination control for the passage to the remote frisking location.

3.2.7.7 Survey Documentation and Review

Radiation Protection supervision will review completed survey documentation to ensure appropriate, adequate and complete information is recorded. The supervisor reviewing the survey will ensure that the recorded results are legible, in accordance with Radiation Protection Manual implementing procedures and consistent with anticipated levels and will determine the reason for any variances. Information that will typically be included on survey maps or forms is:

- Date and time of survey
- Location of survey
- A sketch or description of the area or component surveyed
- Instrument type and serial numbers
- Instrument calibration due date
- Name and signature of surveyor

The results of evaluations will be documented on approved survey forms which will be made available to personnel entering radiologically controlled areas. Survey data will contain enough detail to provide personnel with adequate information concerning radiological conditions existing in the area surveyed. Survey maps will include, as applicable:

1. Contact and general areas dose rates
2. Contamination levels
3. Airborne radioactivity levels, if applicable
4. Identification of specific hazards (i.e., hot spots)
5. Location of radiological boundaries

Personnel contamination detected on hair or skin will be promptly removed under the supervision of trained Radiation Protection personnel. Personnel skin and clothing contaminations will be documented and evaluated to help improve contamination controls.

Personnel contamination forms will include such items as:

1. Names of individuals involved
2. Survey results
3. Decontamination methods

4. Results of decontamination
5. Body locations where contaminated
6. Areas worked
7. Radiation Work Permit number
8. Corrective action to help prevent recurrence of contamination

Survey records will be filed and maintained so that previous radiological conditions can be determined. All original survey records will be maintained and retained in accordance with 10 CFR 20 and Fort St. Vrain Decommissioning Technical Specifications.

3.2.8 Instrumentation

A sufficient inventory and variety of operable and calibrated portable, semi-portable and fixed radiological instrumentation will be maintained to allow for effective measurement and control of radiation exposure and radioactive material and to provide back-up capability for inoperable equipment. Equipment will be appropriate to enable the assessment of sources of gamma, beta, and alpha radiation including the capability to measure the range of dose rates and radioactivity concentrations expected. Installed process and effluent monitors will be set up and operated using Offsite Dose Calculation Manual implementing procedures. Calibration procedures for process and effluent monitors will be implemented and maintained.

Accuracy requirements, remote read-out utilization, alarm set points and conditions, and types of surveying or monitoring to be performed will be specified in Radiation Protection Manual implementing procedures. Remote and special monitoring equipment will be obtained and calibrated per approved procedures if required during the decommissioning.

Counting instrumentation is located in multiple laboratory facilities providing back-up capability. Methods to perform manual calculations as a backup to computerized systems will be contained in implementing procedures.

The Radiation Protection Manual will contain administrative and implementing procedures for the following activities:

1. Instrument inventory and control
2. Instrument calibration
3. Instrument operating procedures

3.2.8.1 Instrument Inventory and Control

Radiation Protection Manual implementing procedures will ensure that instruments are calibrated at the required frequency, functioning properly, issued to appropriate personnel and returned when necessary. Special use or dedicated instruments will be marked to ensure they are not used for other purposes. Adequate instruments will be available for radiation surveillance and associated radiation protection measurements, taking into consideration:

1. Number of personnel and numbers of separate work areas requiring surveillance.
2. Frequency and types of surveys or measurements required to support decommissioning activities.
3. Allowance for repair and calibrations.
4. Efforts to minimize delays in personnel access and egress from radiologically controlled areas.
5. Dedicated instruments (if any) that will be required for emergency response.

A minimum instrument inventory level will be established to ensure that decommissioning activities will not be limited due to inadequate survey capability. Table 3.2-2, "Typical Fort St. Vrain Decommissioning Radiation Monitoring Instruments" lists typical equipment which will be available.

Instruments that are broken or require calibration will be tagged out of service by Radiation Protection personnel. The out-of-service instruments will be separated from operable instruments and placed in a designated location until they can be repaired and/or calibrated.

A whole body counter will be maintained onsite and will be capable of identifying approximately 10% of the maximum permissible organ or body burden from those gamma emitting isotopes likely to be encountered (e.g., Co-60, Cs-137).

3.2.8.2 Instrument Calibration

Procedures for calibration and response checks of radiation monitoring equipment and air sampling equipment will be prepared consistent with guidance provided in the following documents:

1. NRC Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)", (Ref. 37).
2. NRC Regulatory Guide 8.25 "Calibration and Error Limits of Air Sampling Instruments for Total Volume of Air Sampled", (Ref. 36).
3. ANSI N13.1-1969, "American National Standard Guide to Sampling Airborne Radioactive Material in Nuclear Facilities", (Ref. 38).
4. ANSI N42.14-1978, "Calibration and Usage of Germanium Detectors for Measurement of Gamma-Ray Emission of Radionuclides", (Ref. 39).
5. ANSI N42.3-1969, "American National Standard and IEEE Standard Test Procedure for Geiger-Mueller Counters", (Ref. 40).
6. ANSI N320-1979, "Performance Specifications for Reactor Emergency Radiological Monitoring Instrumentation", (Ref. 41).
7. ANSI N323-1978, "Radiation Protection Instrumentation Test and Calibration", (Ref. 42).
8. ANSI/IEEE Std 325-1986, "IEEE Standard Test Procedures for Germanium Gamma-Ray Detectors", (Ref. 43).

The primary calibration frequency for commonly used portable radiation monitoring instruments and portable air sampling equipment will be every 6 months, after repairs or modifications or when malfunctions are suspected. Semi-portable and fixed instrumentation will be calibrated at least annually, after repairs or when malfunctions are suspected. Instrument performance checks (source checks) will be conducted in accordance with ANSI N323 as prescribed in Radiation Protection Manual implementing procedures. At least annually, a review of historical maintenance and calibration trends will be performed for each instrument type. The review will evaluate instrument performance, and the adequacy of calibration frequencies.

Calibration procedures will typically include the following:

1. Instrument specification and limitations
2. Frequency of calibration
3. Description of operating settings/parameters
4. Environmental limitation (if appropriate)
5. References (e.g., instruction manuals, other related procedures, regulatory guidance, etc.)
6. Required equipment for calibration (e.g., sources, tools, jigs, test equipment, etc.)
7. Applicable drawings and schematics

8. Calibration data forms (including as-found/as-left settings, instrument and source identification, charts, etc.)

Laboratory analysis equipment will be calibrated using National Institute of Standards and Technology traceable sources of appropriate geometries and energies. The radiochemistry laboratory will participate in an interlaboratory cross-check program. Documentation of calibrations and cross-checks will be maintained as quality records in accordance with the Fort St. Vrain Quality Assurance Program. Audits of radiation monitoring and air sampling equipment will also be performed in accordance with PDP Section 7, the Fort St. Vrain Quality Assurance Plan.

3.2.8.3 Instrument Operating Procedures

An operating procedure will be prepared for each type of instrument in use, including emergency and special use instruments. Different models of equipment from the same manufacturer with similar features and performance characteristics may be combined into a single procedure, if the operating characteristics are essentially the same, (e.g., Eberline RO-2 and RO-2A). Functional checks of portable radiation monitoring equipment will normally be performed daily or prior to use on the scale(s) expected to be used. Each scale not function checked will be clearly labeled to prevent its use.

Operating procedures will typically include:

1. User responsibilities
2. Instrument and detector description
3. User instructions (including battery check, meter zero, range identification, etc.)
4. Precautions and limitations
5. Identification of proper check sources and associated jigs
6. Performance of source check and/or operational checks

3.2.9 Review and Audit

To ensure the Radiation Protection Program is effectively implemented and maintained, an organized system of review and audits will be implemented in accordance with the Quality Assurance Plan as defined in Section 7.0, "Decommissioning Quality Assurance Plan". Reviews and audits will be conducted by various project organizations and include the following components:

1. Radiation Protection Self Assessments and Reviews
2. Radiation Protection Corporate Oversight and Reviews
3. Quality Assurance Group Audits

3.2.9.1 Radiation Protection Self Assessment and Review

The Project Radiation Protection Manager will be responsible for the quality of work performed by Radiation Protection personnel. The Project Radiation Protection Operations Supervisor will review for adequacy and approve completed radiation survey documentation on a day-to-day basis.

In order to further assure the quality of the Radiation Protection program, Radiation Protection Supervisory Reviews will be planned and conducted by all Project Radiation Protection Supervisors (including the PSC and Project Radiation Protection Manager) on a routine basis. These self assessment/reviews will include in-plant walk downs to directly observe the effectiveness of the Radiation Protection Program including, but not limited to, the following:

1. Radiation protection staff effectiveness
2. Facilities and equipment allocation and use
3. Worker radiological work practices
4. Compliance with Radiation Protection procedures, policies and specifications
5. Compliance with Radiation Work Permit and ALARA programs
6. Conformance with project goals such as person-Rem dose, radioactive waste minimization, etc.

Deficiencies and other findings will be documented and addressed in accordance with Section 3.2.10, Radiation Protection Performance Analysis.

3.2.9.2 Radiation Protection Corporate Oversight and Review

| The PSC Radiation Protection Manager and the PSC Senior Health Physicists are responsible for overseeing the Radiation Protection Program to ensure proper implementation.

Periodic reviews, audits and monitoring of the Radiation Protection Program will be performed to ensure the following:

1. The Radiation Protection Manual and implementing procedures are adequate to meet the Radiation Protection Program as described in Decommissioning Plan.
2. The Radiation Protection Manual and implementing procedures are being followed.
3. The Radiation Protection Manual and implementing procedures are adequate to meet the applicable regulations.
4. The Radiation Protection Program objectives are met.
5. The Radiation Protection Program is being effectively implemented and maintained.
6. The work completed is in accordance with the Fort St. Vrain Project Quality Plan.

3.2.9.3 Quality Assurance Group Audits

The PSC Quality Assurance group and the Westinghouse Quality Assurance organizations will conduct planned audits, reviews and assessments of the Radiation Protection Program in accordance with the Fort St. Vrain Project Quality Plan and in compliance with applicable items of PDP Section 7.

3.2.10 Radiation Protection Performance Analysis

The Radiation Protection staff will establish methods to identify radiological incidents and radiological deficiencies in order to determine root causes and correct errors that cause radiological performance problems. Detailed performance monitoring requirements and activities will be described in and implemented by the Radiation Protection Manual which will consist of administrative and implementing procedures.

3.2.10.1 Radiological Occurrence Reports

Radiological Occurrence Reports will be classified as either a "Deficiency" or an "Incident". Principle elements of the reporting program will include, but not be limited to:

1. A Radiological Occurrence Report may be completed by anyone identifying a radiological occurrence. The report will include pertinent information relating to the occurrence (e.g., date, time, individual reporting occurrence, location, observations, etc.).
2. Radiological Occurrence Reports will be submitted to Radiation Protection Supervision for review, and classification.
3. The Project Radiation Protection Manager and the PSC Radiation Protection Manager will approve corrective actions and implement disciplinary actions, when applicable for designated classes of occurrences.
4. A root cause evaluation for designated types and levels of occurrences will be used,as applicable, to determine the circumstances and causes of the event and to develop short-and long-term corrective actions to prevent recurrence. The evaluations will be conducted by the cognizant first line supervisors and managers with assistance from the Radiation Protection staff.
5. A tracking system for Radiological Occurrence Reports and corrective actions will be implemented. The reports will be trended and evaluated periodically to integrate lessons learned, licensee experience and experience from others into Radiation Protection program improvement. Records will be maintained in accordance with regulatory requirements and company procedures.

3.2.10.2 Radiological Deficiencies

Radiological "Deficiencies" are occurrences involving poor radiological work practices with relatively minor consequences, but require supervisory action for proper resolution. Examples of occurrences attributable to Radiological Deficiencies include, but are not limited to, the following:

1. Failure to comply with radiological posting.
2. Failure to comply with Radiation Protection procedures.
3. Lost dosimetry due to poor work practices, or involving excessive dose.
4. Improper frisking.
5. Personnel contamination instances above a designated level.
6. Poor radiological work practices.
7. Improper use of, or problems with, respiratory protection equipment.

8. Eating, drinking, smoking, or chewing in the radiologically controlled area.
9. Failure to comply with Radiation Work Permit requirements.
10. Unnecessary generation of radioactive or mixed waste.
11. Operation and maintenance of equipment in a radiologically unsafe manner.
12. Minor spills of radioactive materials.

These types of occurrences will be evaluated for possible deficiencies in areas such as training, procedures, equipment and human performance. Appropriate corrective action and follow up will be required by the Project Radiation Protection Manager or designee.

3.2.10.3 Radiological Incidents

Radiological "Incidents" are occurrences that have, or could have the potential for, violating Federal Regulations and Fort Saint Vrain Decommissioning Technical Specifications or involve a serious breakdown in the effectiveness of the Radiation Protection Program. Examples of occurrences that would be considered Radiological Incidents include, but are not limited to the following:

1. Radiation exposures exceeding federal limits.
2. Radiation exposures exceeding administrative limits without previous authorization.
3. Radioactive body burdens in excess of 25% Maximum Permissible Organ Burden.
4. Unplanned exposures of individuals to airborne radioactivity in excess of 2 MPC-hours per day or 10 MPC-hours in any seven consecutive days.
5. Significant spills or spread of radioactive materials that affect decommissioning activities.
6. Lost radioactive materials or radioactive materials found in uncontrolled areas.
7. Flagrant violations of dosimetry procedures, such as intentional or willful loss of or damage to dosimetry, or failure to wear or improperly wearing required dosimetry.
8. Lack of or improper access control for High Radiation Areas.
9. Improperly posted areas, especially high radiation areas.
10. Failure to follow instructions and "stop work" orders.

11. Damaged or leaking radioactive material shipments.
12. Flagrant violation of Radiation Protection Procedures.

Incidents that are required to be reported in accordance with NRC Regulations will be addressed in accordance with the PSC Licensee Event Report Program. Radiological Occurrence Reports involving Radiological Incidents will be investigated and critiqued by a team assigned by the PSC Radiation Protection Manager or the Project Radiation Protection Manager. The team will normally consist of a PSC Senior Health Physicist, a Project Radiation Protection Staff Member, and the affected organization. Appropriate corrective action will be required to prevent recurrence. Completed Radiological Occurrence Reports and investigations will be reviewed by the ALARA committee and the Decommissioning Safety Review Committee, as appropriate, depending on the type of occurrence and the severity.

3.2.11 Radiation Work Practices

Standardized radiation work practices and engineering controls will be located in the Radiation Protection Manual administrative and implementing procedures. Radiation work practices and engineering controls will also be incorporated into work specifications, engineering designs and work packages involving radiation exposure or handling of radioactive materials as indicated in Section 3.2.5.1. Typical radiation work practices that will be addressed, include:

1. Radiation exposure reduction methods
2. Radiation work performance methods
3. Use of temporary shielding
4. Contamination control equipment
5. Work area ventilation
6. Decontamination processes
7. Liquid and solid waste processing
8. Control system for contaminated tools
9. Use of protective clothing

3.2.11.1 Radiation Exposure Reduction Methods

Specific work activities/tasks with the potential for moderate or high radiation exposure will require that radiological controls be incorporated into planning and scheduling development, written instructions be prepared, and pre-job briefings be conducted prior to commencing work and post-job debriefings be conducted for lessons learned. Radiological conditions that will be considered include:

1. Presence of radioactive liquids.
2. Potential for high activity sources or contamination (PCRV disassembly).
3. Potential for creating Hot Particles (PCRV pool work).
4. Transient high radiation levels (e.g. PCRV disassembly).
5. Airborne radioactivity (e.g. opening a system).
6. Specific radiological evaluations upon occurrence of unusual events (e.g., notify Radiation Protection for radiation monitor alarms).
7. Reminder of the need to follow the Radiation Work Permit which may have special controls associated and therefore longer lead time for Radiation Protection preparation.
8. Precautions for work when systems are in unusual status or configuration.
9. Changing of system status which may affect radiological conditions (e.g., starting and stopping the PCRV water purification system).

Engineering Controls

Radiation Protection engineering controls considered during ALARA reviews and RWP preparation. Examples of engineering controls will include, but will not be limited to, the following:

- Temporary shielding
- Specialty and remote handling tools
- Contamination control containments
- HEPA ventilation systems
- Decontamination equipment and techniques
- Remote surveillance systems

3.2.11.2 Radiation Work Performance Methods

Planning for radiological work is an essential element in assuring an efficient use of resources, maintaining control of radioactive material, and keeping worker exposure ALARA. An objective of planning will be to provide adequate time for each affected department to prepare for radiological work. Planning will typically include the following considerations:

1. Provide adequate time for work area preparation, including removal of hazards, in addition to radiological hazards, to provide a safe working environment.
2. Decontaminate work areas to increase worker efficiency.
3. Install systems to contain radioactive materials.
4. Provide workers with specialized training and other identified ALARA requirements.
5. Incorporate previous experience on similar jobs.

3.2.11.3 Use of Temporary Shielding

Temporary shielding will be used to reduce dose rate levels near "hot spots" and in the general area where work is to be performed. Determination as to the type and amount will be evaluated by Radiation Protection personnel. An implementing procedure will be used for the control and use of temporary shielding. Additional details on temporary shielding is covered in Section 3.2.5, Dose Controls.

3.2.11.4 Contamination Control Equipment

Contamination will be controlled by employing a variety of engineering controls including HEPA ventilation, enclosures, strippable paint, and area/component decontamination. Examples of contamination control methods that will be used, include:

1. The PCRV will be filled with water to control radioactive particulates that would normally be released when handled in air.
2. Containment or enclosures of appropriate size, equipped with HEPA ventilation, will be used as necessary to prevent the spread of contamination while contaminated graphite blocks and other components are being removed from the PCRV or otherwise handled.

3. A work platform will be installed on the PCRV after the PCRV head has been removed. The platform will be equipped with a HEPA-filtered ventilation system that will exhaust air from beneath the work platform. This airflow will minimize the spread of contamination.
4. A debris collection system will be used in concrete cutting operations to minimize the spread of contamination.
5. Strippable paint or other suitable enclosures will be applied to some radiologically clean components or areas to prevent cross-contamination.

Additional contamination control methods will be considered during job planning and work package review. Isolation containments may be used to minimize the spread of contamination if the surrounding work area is uncontaminated or is much cleaner than the work area.

Radiation Protection Manual implementing procedures will provide guidance on the application and use of contamination control equipment. Examples of equipment include, but are not limited to:

- HEPA ventilation
- Containments
- Glove bags
- HEPA vacuums
- Strippable paint
- Sheeting (e.g. plastic, herculite)

3.2.11.5 Work Area Ventilation

Portable HEPA ventilation units will be used in work areas for the control of airborne contaminants during work activities that have the potential for airborne contamination generation (e.g., burning, welding, grinding). Periodic tests will be conducted to assure air flows are from areas of low potential airborne contamination to areas of higher potential contamination. Containments or tents may also be used in conjunction with HEPA ventilation to control airborne contamination. Radiation protection implementing procedures will provide instruction on the use and control of portable HEPA ventilation units.

3.2.11.6 Decontamination Processes

Various decontamination processes will be employed during decommissioning to prevent the spread of contamination, minimize the potential for internal uptake and reduce the associated radiation levels for ALARA purposes. Examples of decontamination processes which will be used, include:

- Scabbling
- Hydrolazing
- Strippable coatings
- Chemical cleaning
- Abrasive blasting
- Ultrasonic cleaning
- HEPA vacuuming
- Hands on decontamination

3.2.11.7 Liquid and Solid Waste Processing

Process controls will be applied to the identification, collection, processing, packaging and disposal of radioactive waste to ensure compliance with state and federal applicable regulations. Radioactive waste processing and controls are addressed in Section 3.3.

3.2.11.8 Control System For Contaminated Tools

Storage areas and hot tool cribs will be identified for the project, and will be used for the storage of reusable contaminated tools, components, equipment and materials. This designated storage will help to prevent the spread of contamination and maintain radiation doses ALARA. Implementing procedures will include the control and use of contaminated tools and equipment.

3.2.11.9 Area Posting

Supplemental postings will be used to provide additional information to workers. These postings will be used in conjunction with the required posting and may contain the following types of information/instructions.

1. Contact Radiological Protection for Entry
2. Hot Particle Controls Area
3. Frisk Hands and Feet Prior to Exiting
4. Potentially Contaminated Area (e.g., in overheads)
5. Internal Contamination (e.g., inside electrical panels)
6. "Keep Out" Radiography in Progress

7. Radiation Work Permit Required for Entry

3.2.11.10 Description and Functions of Protective Clothing

Protective clothing will be provided for personnel working in contaminated areas and will be required as specified on RWPs. Selection and use of protective clothing will be based on known and expected contamination levels in the work area, as well as the expected working conditions. Protective clothing requirements will be specified on Radiation Work Permits. To ensure the proper control and use of radiological protective clothing, they will be used as indicated by Radiation Work Permits. Instructions for the proper donning and removal of protective clothing will be addressed as part of Radiation Worker Training.

Containers for the disposal of used protective clothing will normally be placed at the exits of contaminated areas. Used protective clothing will be treated as potentially contaminated and handled as such.

Implementing procedures will be developed for the control and laundering of contaminated protective clothing. Reusable protective clothing will be used whenever possible. Laundry will be monitored for acceptable residual contamination levels prior to reuse.

3.2.12 Diving Operations in the PCRV

Underwater divers are being used to perform decommissioning tasks in the PCRV. It was determined that diving operations are preferable to use of remote tooling for a number of underwater operations, including removal of insulation from the top of the CSF and from the inside of the lower portion of the core barrel, removal of the core barrel bottom remnant, and separation of the core support floor from its support columns and the steam generators. Since divers can work closer to the components to be removed from the PCRV, the work can be performed much more efficiently than with remote tooling, especially considering the depth of water in the PCRV and limited visibility through the shield water. The ability to segregate radioactive wastes by activation/contamination levels is also enhanced by the use of diving operations, increasing the effectiveness of radioactive waste packaging.

Diving operations have been used extensively in the nuclear industry, especially for maintenance of spent fuel storage pools, with radiation levels much higher than those that will be encountered in the PCRV. The NRC has issued guidance for diving operations in high and very high radiation areas, contained in NRC Regulatory Guide 8.38 (Reference 83). Regulatory Guide 8.38 states: "The use of proper underwater work techniques can result in substantial savings of time and reductions in radiation doses." Applicable guidance from this Regulatory Guide, especially that provided in Appendix A of the Regulatory Guide, has been adopted in the FSV procedures that govern diving operations in the PCRV.

Following are prerequisites to diving operations at FSV:

- Divers have been trained in diving procedures, which include industrial and radiation safety measures.
- Divers have received training on the use of underwater survey meters and survey techniques.
- A staging area has been established for moving the diver in and out of the water.
- Telemetric remote dosimetry monitoring systems are operational.
- Communication systems, which permit two-way communications between the diver and personnel topside, are installed and operational.
- A camera system is installed and operational, when necessary, so personnel topside can visually observe actions of the divers.
- Water clarity is suitable for operations to be performed.

Following is a discussion of safety concerns associated with the diving operations, and means of addressing the concerns.

1. <u>Leakage of Water into the Diving Suit</u>

The divers will be wearing "dry suits" with full body coverage that will not permit their skin to come in contact with the PCRV shield water. Full helmets are used with integral face plate and communications equipment. Divers are required to exit the water if water leakage is detected or suspected.

Calculations were performed to assess doses that could potentially occur in the event of a leak in a diving suit. It was assumed that 100% of a diver's skin was exposed to water for 10 minutes while the diver inhales air above the surface of the water, and ingests 10 ml of water. It was conservatively assumed that tritium and Co-60 concentrations were much higher than those that actually exist in the PCRV. The whole body dose which could result from a diving suit leak was calculated to be less than 2 mrem.

Internal surfaces of the diving suits are routinely surveyed for loose contamination if there is indication of leakage. The divers provide in-vitro bioassay samples as directed by Radiation Protection procedures, that are used to detect and calculate any internal exposures, along with whole body counts (in-vivo) prior to an individual's first dive and after the last dive at FSV.

2. Mitigating Dose Rates in the PCRV During Diving Operations

Some of the equipment in the PCRV is highly activated and there is the potential for significant exposure of divers. With the water shielding, dose rates in the PCRV can change dramatically over short distances, and as activated materials are moved or removed from the PCRV. A detailed survey was performed inside the PCRV prior to the start of diving operations, using two radiation detection devices. TLDs were also positioned in various locations in the PCRV to allow comparison between survey instrument and TLD response. Areas or equipment items with dose rates greater than 100 mrem/hr are identified on survey forms or maps provided for diving information. For areas or items with dose rates greater than 1,000 mrem/hr, strict controls are maintained by Radiation Protection personnel to prevent inadvertent diver access. During diving operations, verification surveys of the work area and divers' travel route are performed at appropriate frequencies determined by Radiation Protection. Typically, these routine surveys are performed once a shift while diving operations are underway. The sources of higher-than-average or unexpectedly high readings are investigated. In the event that a diver enters an area inaccessible to survey instruments, the diver is provided with a remote readout radiation detector whose readout is continuously monitored by a radiation protection technician, with an alarming dosimeter set to alarm on both an integrated dose and dose rate.

Since dose rates to different parts of a diver's body can vary considerably at any given time, divers are required to wear special dosimetry on several different body areas. Typically, divers are supplied with the following whole body TLDs: head, chest, middle of back, each upper arm and each thigh. Divers are also typically supplied with extremity TLDs on each foot and finger rings, as appropriate. In addition to TLDs, secondary dosimetry and a telemetric system are used to continuously monitor divers' exposure by Radiation Protection personnel stationed topside.

Continuous voice contact is maintained between divers and topside personnel, by means of a radio communication system. A safety line is attached to each diver, and standard line signals will be used for communications in the event of radio failure. The dive will be terminated if radio communications cannot be quickly restored.

Radiation Protection personnel meet with the divers prior to commencement of diving activities on a shift. Radiation Protection personnel provide ALARA briefings to divers and support personnel, as required, stressing the presence of hot spots. By monitoring diver exposure and work progress, Radiation Protection personnel assure the divers do not exceed exposure goals or limits. An underwater camera will be used to monitor the diver when line-of-sight coverage is not possible, as required by Radiation Protection.

3. Preventing the Spread of Contamination and Dealing with Hot Particles

Contamination may be removed from the PCRV and spread if proper precautions are not taken. The divers and diving equipment and tools are rinsed off and wiped down upon exiting the PCRV shield water. External surfaces of the diving suits are surveyed for contamination, including hot particles, after each dive. Radiation surveys are performed on PCRV components removed from the water. Contamination surveys are performed on tools and non-trash items removed from the PCRV. Contamination surveys are routinely performed in the staging area while work is being performed in the PCRV. In addition, air samples are routinely taken in the vicinity of the diver staging area when diving operations are in progress.

Hot particles can be produced by some underwater operations. A vacuum cleaner will be available for use in cleaning underwater work area surfaces prior to commencement of work in areas with a high potential for the presence of hot particles. In addition, a vacuum cleaner will be available to capture debris during work which has a high potential for creating hot particles, such as mechanical cutting of highly irradiated materials. When divers exit the PCRV shield water, the divers and their equipment/tools are rinsed off with water. Radiation Protection personnel survey the divers for hot particles, perform a detailed hot particle survey of the dry suit after removal, survey diving equipment/tools, and survey rags used to wipe down the diver and equipment/tools. A hot particle survey of the staging area is routinely performed when diving operations have occurred or materials have been removed from the PCRV.

4. Industrial Safety of Divers

Industrial safety of the divers is established by OSHA, U.S. Navy Diving Manual Dive Tables, and the diving contractor's safe practices manual. The diving supervisor has control over any diving related problems or emergencies and safety determinations. Plant emergencies are handled in accordance with approved plant emergency procedures. For emergencies requiring evacuation of the Reactor Building or the diver staging area, divers will be immediately evacuated from the PCRV.

The divers' air supply is in accordance with OSHA Part 1910, Subpart "T" requirements, with testing of the primary air compressor required every 6 months. The air supply for diving operations is from the plant's breathable air system. The two air compressors that supply this system are inspected on a daily basis while diving operations are underway, including system pressures, filter checks, oil levels/pressures, all tags and locks in place, high temperature alarm and low pressure alarm, and appropriate maintenance logs completed. A 25-foot vehicle restriction zone from the air-compressor intake is established and maintained during all diving operations to help assure acceptable breathing air quality. A backup air supply is available in the event the primary air supply is interrupted. When diving under the core support floor, divers are required to wear a "bail-out bottle". Lighting required for diving operations is equipped with ground fault interrupters.

INTENTIONALLY LEFT BLANK

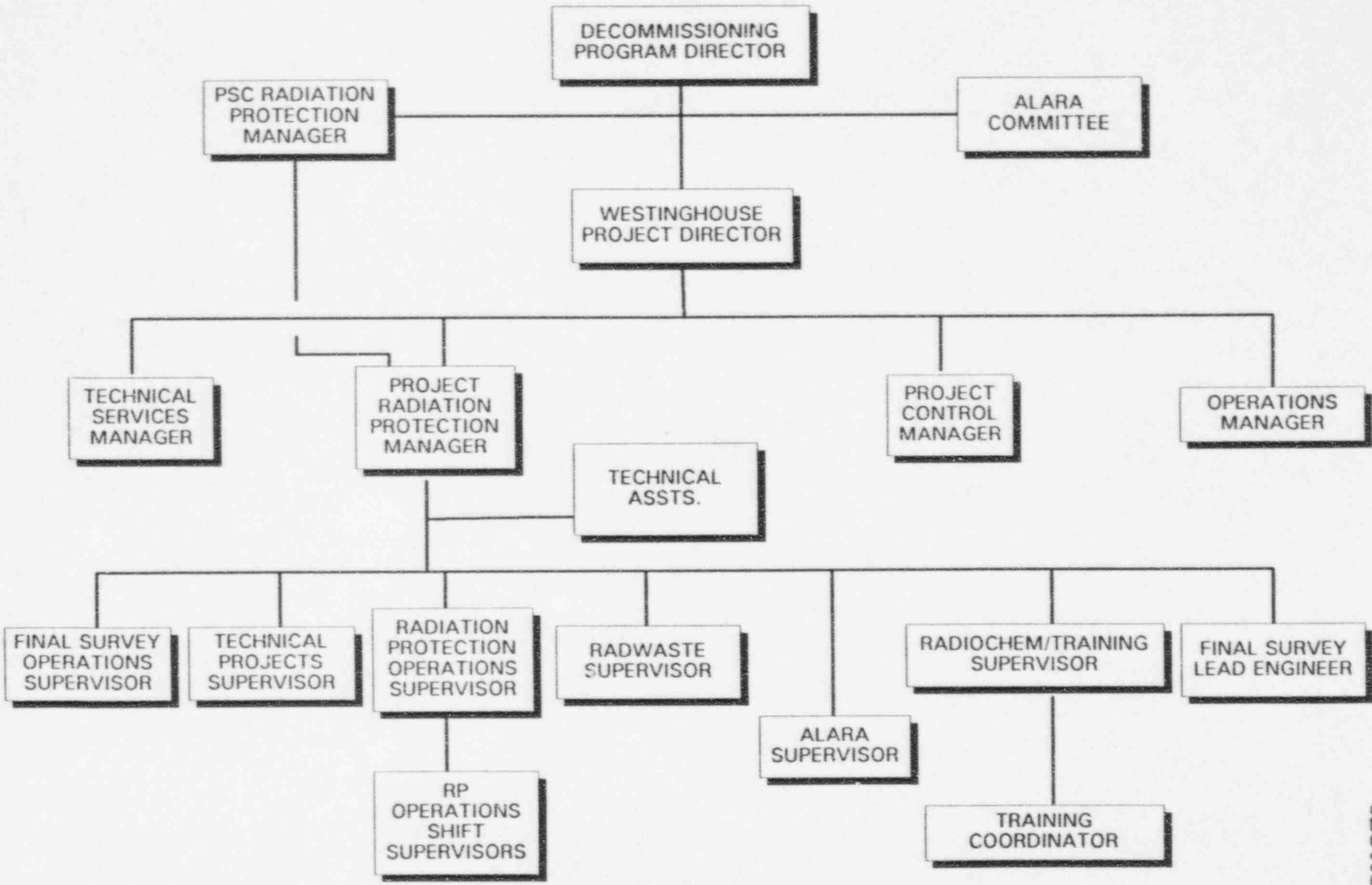

Figure 3.2-1 Westinghouse Team Radiation Protection Organization Chart

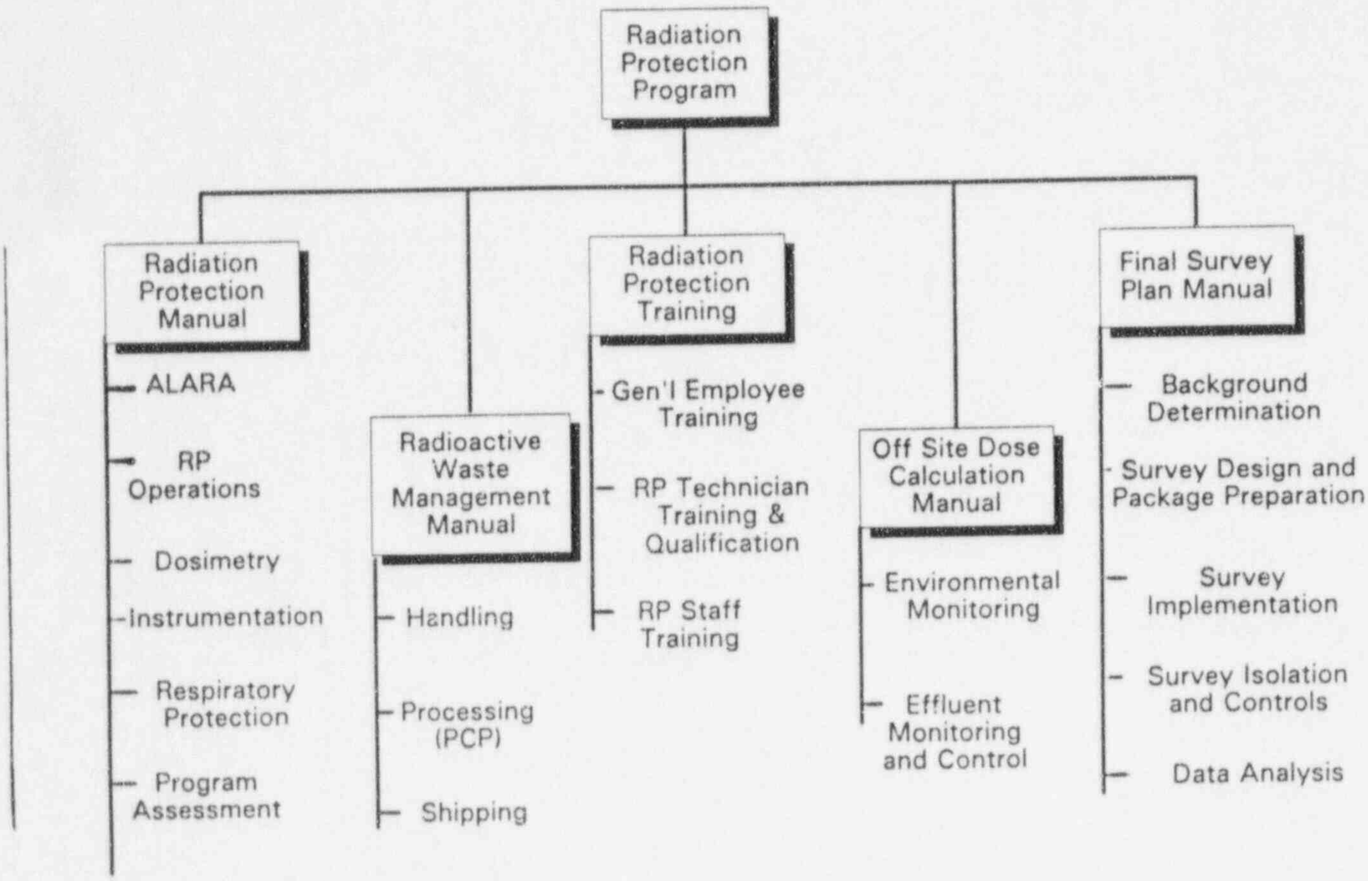

Figure 3.2-2 Radiation Protection
Program Manuals Structure

3.3 RADIOACTIVE WASTE MANAGEMENT

This section addresses the technologies, equipment, and procedures to be implemented for the management of radioactive waste during the Fort St. Vrain (FSV) decommissioning project. These technical approaches are based upon experience and address facets of planning, decontamination, packaging, storage, transportation, volume reduction or beneficial reuse, and final disposition of the waste materials, while minimizing secondary wastes.

In developing the Radioactive Waste Management Program, many elements were considered, including the following:

1. End use of the facility
2. Location and availability of disposal facilities
3. Potential for offsite release during D/D operations
4. Preventing contamination of uncontaminated areas
5. Use of existing buildings to support the waste packaging operations
6. Methods of approach related to waste type, waste class, and impact on safety
7. Cost effectiveness
8. Logical approach to the D/D operations
9. Ensuring that the occupational exposures are maintained as low as reasonably achievable (ALARA)
10. Minimizing the impact on the health and safety of the general public
11. Maintaining flexibility for waste management to allow for unexpected wastes and changes in available technology
12. Effective implementation of a Process Control Program for radioactive wastes

This section contains a description of the following activities associated with the radioactive waste management program:

- Spent fuel disposal (3.3.1)
- Radioactive Waste processing (3.3.2)
- Radioactive waste disposal (3.3.3)
- Disposal of non-radioactive wastes (3.3.4)

3.3.1 Defueling to the Independent Spent Fuel Storage Installation

Although not related to proposed decommissioning plans, the following information is provided on the ultimate disposition of the Fort St. Vrain spent fuel.

Due to the uncertainty of shipping of spent fuel to Idaho or other DOE facilities, PSC pursued an alternate plan to license, construct and operate an Independent Spent Fuel Storage Installation (ISFSI) in accordance with 10 CFR 72. The NRC issued PSC an Environmental Assessment of No Significant Impact for the ISFSI on February 1, 1991 (Ref. 44), and PSC commenced construction on the same date.

The ISFSI facility is located immediately adjacent to the current site. The actual location is outside the plant's existing restricted area, approximately 1500 feet northeast of the Reactor Building. The ISFSI, using the Modular Vault Dry Store (MVDS) System, is designed to store up to 1482 fuel elements, up to 37 metal clad reflector blocks (MCRB's) and up to 6 neutron sources.

Following review of PSC's application, the NRC granted PSC a 10 CFR 72 license on November 4, 1991 (Ref. 45). PSC commenced defueling to the ISFSI on December 26, 1991. Defueling to the ISFSI is on schedule to allow completion of defueling by mid 1992, which will allow decommissioning to commence not later than mid 1992.

3.3.2 Radioactive Waste Processing

3.3.2.1 Program Description

For materials that may contain licensed radioactive material, radiological surveys will be performed to determine the extent of the contamination or activation. Based on these results, options for decontamination or disposal, packaging, and processing will be determined.

Onsite packaging or processing of radioactive waste prior to transportation will be performed in areas appropriate for these activities. Examples of such areas are the Hot Service Facility (HSF), the gas waste compressor rooms (Reactor Building level 1, el. 4740'), the Compactor Building, and the Fuel Storage Building.

Items not considered for decontamination or items that, following decontamination, are considered to have too high a specific activity for offsite volume reduction, will be packaged and shipped directly for disposal at a licensed burial facility. Greater than Class C (GTCC) wastes, if any, will be packaged for onsite storage and subsequent shipment to a designated storage or disposal facility.

Radioactive wastes are expected to be categorized as follows:

1. Potentially contaminated or requiring minor spot decontamination: These include potentially contaminated materials that: 1) appear to be uncontaminated; 2) all surfaces are easily accessible; and 3) have a small surface area-to-weight ratio will be surveyed to determine if the material can be released for unrestricted use without decontamination or with minor decontamination effort. For example, a small surface area with only spot and/or smearable contamination can easily be decontaminated by such means as wiping, grinding, or removing the hot spot.
2. General contamination with accessible surfaces and a low area-to-weight ratio: Materials with readily accessible surfaces for purposes of surveying and decontamination, and that possess a low surface area-to-weight ratio may be shipped directly to a licensed offsite processing facility for decontamination of the surfaces and final disposition.
3. General contamination/inaccessible surfaces/high surface area-to-weight ratio: Smaller metallic scrap or metals with inaccessible surfaces for performing surveys (e.g., previously sheared material) will be assumed to be contaminated and be packaged for shipment for further processing at a licensed facility or shipped directly to burial.
4. Activated: Activated materials and high specific activity materials (primarily concrete, metals and graphite components), will either be packaged and shipped direct for disposal or to a licensed facility for further processing and volume reduction.

Radioactive materials as categorized above will be evaluated to determine the optimum method for release, decontamination, or shipment offsite for further processing or for burial. The following onsite and offsite methods will be considered.

1. Onsite processing of liquid wastes.

2. Onsite processing, release and disposal options for tritiated liquid wastes.
3. Onsite filtration of airborne wastes.
4. Onsite decontamination.
5. Onsite waste volume reduction.
6. Onsite packaging.
7. Offsite decontamination.
8. Offsite volume reduction.
9. Offsite repackaging/consolidation for disposal.

3.3.2.2 Onsite Processing of Liquid Wastes

During the Fort St. Vrain decommissioning project, contaminated water will be generated through several processes (such as diamond wire cutting, flooding of the PCRV, rinsing of contaminated components removed from the PCRV) and through decontamination operations. Flooding the PCRV will put into solution radionuclides that exist in the PCRV as a result of activation and plateout. Of primary concern are tritium and the gamma-emitting isotopes Cs-137 and Co-60. Expected releases of tritium from graphite components into the PCRV shield water is discussed in PDP Section 3.1.5. Tritium processing, release and disposal options are discussed in the following section (Section 3.3.2.3).

Releases of PCRV Shield Water will be processed through the PCRV Shield Water System where filters and demineralizers will substantially reduce the concentration of radionuclides, with the exception of tritium. As shield water is released, it will be replaced with clean water as necessary to maintain the desired PCRV water level.

Water to be released from the PCRV will be transferred through the PCRV shield water system demineralizers, either to a liquid waste holdup tank in the existing Radioactive Liquid Waste System (System 62) or to the Reactor Building Sump (RBS), for sampling and analysis. Releases from System 62 and the RBS are described in Section 2.2.3.10. Samples are analyzed for gross alpha and beta activity, principal gamma emitters, tritium and other radioisotopes of concern. Based on the radionuclide concentrations, dilution factors are calculated per the Fort St. Vrain Offsite Dose Calculation Manual (ODCM, Ref. 48) to assure that the concentrations released to the environment will not exceed the values specified in 10 CFR 20 or the EPA Safe Drinking Water Standards in 40 CFR 141 (for tritium), and doses to the general public will not exceed the 10 CFR 50 Appendix I limitations, nor those doses evaluated in Section 4.2 of the Environmental Report Supplement for

FSV Decommissioning (Ref. 84). Maximum allowable flow rates from System 62 or the RBS are stipulated, along with minimum allowable cooling tower blowdown flow, to assure adequate dilution. If further processing of the radioactive liquid is desired prior to release, water from a System 62 holdup tank can be circulated through the System 62 demineralizers and returned to the holdup tank,or transferred to the RBS/keyway. Water in the RBS or liner in the RBS keyway can be circulated through the PCRV shield water system polishing demineralizers, and/or additional demineralizers installed in the polishing demineralizers' drain line, and returned to the RBS or liner in the RBS keyway. Automatic protective features function to isolate releases from System 62 or the RBS in the event of either high effluent gamma activity levels or low cooling tower blowdown flow.

It is also permissible to discharge directly from the PCRV, provided the water in the PCRV has been processed to the extent that the concentrations of Co-60, Cs-137 and Fe-55 in the entire PCRV water volume are less than approximately 1.0% of the 10 CFR 20 MPC limits and tritium concentration is less than its 10 CFR 20 MPC limit. In this mode the water will be sent directly from the PCRV to the discharge line, where it will be diluted with blowdown flow and released. At that time, the entire PCRV will be considered a process tank and releases will be made directly to the dilution point, as long as no activities are in progress inside the PCRV that could stir up additional contaminants.

It was not possible to fully drain water from the 17 PCRV bottom head penetrations by means of the PCRV shield water system, since the shield water system took suction on the center access penetration, with an alternate suction on a helium circulator penetration. Therefore, the PCRV lower plenum and penetration drain system was installed after the water level was below the elevation of the PCRV lower plenum floor (less than 50,000 gallons remaining in the vessel). This system permitted the shield water system to be isolated and decommissioned, while providing a path for draining the remaining water from the PCRV and its bottom head penetrations. Following isolation of the shield water system, water was drained from each of the PCRV bottom head penetrations, through a sock filter into the System 62 radioactive liquid waste sump. Water was pumped from the System 62 sump through the additional demineralizers that were installed in the shield water system polishing demineralizers' drain line to the liner in the keyway section of the RBS. The capability existed to further process water from either the keyway liner or the RBS with the additional demineralizers. Following necessary processing, water was released from the RBS as described above.

Operating simplicity of this system will minimize the radwaste movement, handling and personnel exposure. Spent resins and filter media requiring stabilization will be processed in accordance with the Process Control Program (PCP). The PCP shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, 49 CFR 100, State regulations, disposal site burial requirements, and other requirements governing the disposal of solid radioactive waste. When possible, this will be done inside the disposal package or liner to minimize additional waste handling prior to disposal.

3.3.2.3 Tritium Processing, Release and Disposal Options

A. Tritium Release Alternatives

Since tritium cannot be removed from the water by processing, it must either be diluted to releasable levels or disposed of as radioactive waste. The release option was chosen for the low tritium concentrations expected in the PCRV Shield Water System. The PCRV Shield Water System, discharging to the existing Fort St. Vrain plant liquid effluent stream, was selected as the best possible release path for tritium in the PCRV shield water. In addition, occupational and public doses from effluent discharge operations are the lowest of all the alternatives evaluated. This effluent discharge path was fully analyzed in the Fort St. Vrain FSAR and the original Supplement to the Environmental Report (Ref. 46), and the impacts during normal plant operations were determined to be acceptable to the NRC. Moreover, the PSC environmental monitoring program, which complies with the Regulatory Guide 1.21 (Ref. 47), has confirmed no significant impacts to the environment due to discharges of tritiated water over the last fifteen years of operation.

The current tritium discharge pathway is modeled by the Fort St. Vrain Offsite Dose Calculation Manual (ODCM) from this release pathway. This pathway currently has adequate measuring and monitoring capabilities for the anticipated discharge level (both water quantity and curie content). This pathway provides adequate water to dilute the anticipated quantity of tritium to below 10 CFR 20 limits.

In summary, the discharge of tritium to the existing Fort St. Vrain liquid effluent stream provides the most advantageous method for tritium release during decommissioning. In addition, it is an accepted and demonstrated safe method that will minimize both occupational and public doses.

B. Solidification of Highly Tritiated Water

In the unlikely event that the amount of tritium entering the water greatly exceeds the expected levels, and the effluent discharge release method cannot be used, alternate disposal methods are available. In this case, after tritium pickup by the water is complete and suitable containers are in place, a feasible contingency plan is to remove the water from the PCRV in its entirety and solidify it for disposal. An acceptable solidification process would be to use Aquaset, which has a solidification efficiency of 45 gallons in a 55 gallon drum and requires about 1 hour per drum.

Appropriate radiological controls would be implemented during the solidification of the tritiated water to maintain external and internal radiation exposures ALARA. Because of the increased costs and inability to continuously improve water quality, the solidification method would not be used unless the tritium level greatly exceeds expected levels. Solidification of highly tritiated water is discussed here to demonstrate that suitable technology is currently available, and to establish a bounding cost for disposal should the preferred method not be suitable.

C. Tritium Release and Monitoring

The PCRV will be filled with approximately 325,000 gallons of water. No discharge will be made until the trend of tritium concentration is determined. The initial concentration of tritium in the PCRV (approximately 5 days after fill) is estimated to be less than 0.40 μCi/ml, based on 500 Ci of tritium diluted in 325,000 gallons of water.

The Decommissioning Technical Specifications require that the PCRV water be sampled and analyzed daily for tritium concentrations during the initial fill of the PCRV. Sample frequency may be reduced to weekly after the tritium concentration has decreased to less than 0.1 μCi/ml. Limits have been established in the Decommissioning Technical Specifications to assure that tritium activity concentrations in the PCRV Shield Water System will not exceed those postulated in the decommissioning accident analyses.

When it is desired to transfer water from the PCRV for subsequent release, water from the PCRV will be processed through the PCRV Shield Water System and a side stream will be transferred to either a liquid waste holdup tank in the existing plant Radioactive Liquid Waste System (System 62) or to the Reactor Building Sump (RBS). After the PCRV water level was below the PCRV lower plenum floor, the remaining water was processed by means of the PCRV lower plenum and penetration drain system (described in Sections 2.2.3.10 and 3.3.2.2). PCRV shield water transferred to the RBS enters a liner that has been installed in the rectangular keyway section of the RBS to reduce the potential for contamination of the concrete. Water is directed from the liner to the suction of the RBS pumps. The holdup tank/RBS will be sampled for tritium and other principal radionuclides. Based on sample results and the limits prescribed in the Fort St. Vrain Offsite Dose Calculation Manual (ODCM) (Ref. 48), an allowable release rate will be determined. This method will ensure that the desired discharge concentration (less than MPC) is not exceeded. Radionuclide concentrations in liquid effluent will be released within the following limits:

1. Tritium concentrations will comply with EPA Safe Drinking Water Standards in 40 CFR 141 (20,000 pCi/l average concentration), at the downstream sampling location located approximately 5 miles downstream of the effluent discharge location.
2. 10 CFR 50 Appendix I limitations on doses to individual members of the public (1.5 mRem whole body per quarter, 3 mRem whole body per year).
3. 10 CFR 20 MPC limits on concentrations in effluents released to unrestricted areas.

Administrative controls will be implemented to ensure that the above limits are met. The above standards also ensure compliance with the 40 CFR 190 EPA public dose limits of 25 mRem per year. In addition to the above limits, PSC intends to ensure that PCRV shield water is processed to the extent that shield water releases will not result in doses to offsite personnel greater than those calculated in Section 4.2 of the Environmental Report Supplement for Decommissioning (ERS, Ref. 84). Calculations were performed in August 1994 (Ref. 85), in accordance with methodology in the ODCM, which assessed offsite doses that could result from the release of the PCRV shield water inventory. It was conservatively assumed that 500,000 gallons of water was released via the radioactive liquid effluent pathway at 100 gpm. Radionuclide concentrations were projected based on testing to determine activity in the shield water after it had passed through the shield water system demineralizers. The resultant offsite dose was calculated to be a factor of approximately 30% below the dose presented in Section 4.2 of the ERS (0.165 mrem/year vs. 0.545 mrem/year).

After sampling, liquid in the System 62 liquid waste holdup tank will be discharged at a rate up to 10 gpm and diluted by the cooling tower blowdown flow prior to release to the surrounding surface water. The minimum cooling tower blowdown flow of 1100 gpm defined in the ODCM will ensure a dilution factor of more than 100 for releases from the System 62 holdup tank. The maximum capacity of a RBS pump is approximately 60 gpm. Assuming one RBS pump discharges liquid effluent from the RBS at its capacity and there is a minimum blowdown flow of 1100 gpm, a dilution factor of approximately 18 would be achieved. Valves in the RBS pumps' discharge path are throttled to reduce the RBS discharge flow rate, increasing the dilution factor as necessary to comply with ODCM requirements.

Figure 3.3-1 shows a representative decrease in PCRV tritium concentration assuming approximately 500 Curies of tritium leaches into the PCRV shield water shortly after the initial fill, with a discharge of up to 10.9 Curies per day (2000 gpm cooling tower blowdown discharge) until tritium concentrations drop below levels where this rate can be maintained without requiring dilution to meet 10 CFR 20 limits. Tritium concentration will continue to be reduced, and after approximately 3 months of effluent discharge operations, the PCRV water tritium concentrations will be low enough to allow direct discharge to the environment (i.e., less than 10 CFR 20 MPC limits). Discharge at a slower rate (1100 gpm, resulting in a release rate of 6 Curies/day or less) would extend the time to decrease below the 10 CFR 20 concentration limits.

Water consumption requirements to support the dilution of the vessel water may range from the minimum flow rate of 1100 gallons per minute to a flow rate of more than 2000 gallons per minute. Existing site capacity can accommodate these requirements and no additional water sources are required. Makeup water is obtained from ditch water diverted from the South Platte River and St. Vrain Creek and is supplemented by water from a system of six shallow wells.

The cooling water blowdown line (normal discharge path) flows to a diversion box from which the flow can be directed to the South Platte River via the continuation of Goosequill irrigation ditch or to the St. Vrain Creek via a slough. Further downstream from the plant, the Goosequill irrigation ditch flows into the Jay Thomas irrigation ditch and the combined stream flows into a 25 acre farm pond. The overflow from this pond flows into the South Platte River close to its confluence with the St. Vrain Creek. The drainage path via Goosequill ditch and the pond is normally used.

3.3.2.4 Release of Airborne Contamination

Plant gaseous effluent filtration and monitoring systems will be operated and maintained as described in the Decommissioning Technical Specifications and the Offsite Dose Calculation Manual.

The HEPA filter penetration and bypass acceptance limits in the Technical Specification surveillances are applicable based upon a HEPA filter efficiency of 99%. The HEPA filter bank will be tested using the test procedure guidance in Regulatory Position C.5.a and C.5.c of Regulatory Guide 1.52 (Ref. 49), with a flow rate of at least 17,100 cfm to verify that the filter penetration and bypass leakage test acceptance criteria of less than 0.05% is met.

The replacement frequency of the HEPA filters in the existing Reactor Building ventilation exhaust system is also identified in Decommissioning Technical Specifications, and is based upon either high exhaust radiation readings (or alarm) in the ventilation exhaust duct, or upon exceeding the maximum allowable pressure differential (which indicates that the filters are filled with dust).

Effluent monitoring of the reactor building exhaust will be accomplished and reported using installed plant equipment and established procedures. Supplemental effluent air monitoring in the form of air samples for areas or operations remote from the Reactor Building with air discharge capabilities will be maintained. Monitoring capabilities include beta/gamma radiation measurement of samples.

3.3.2.5 Onsite Decontamination Techniques

Onsite decontamination techniques will normally be used for processing and volume reduction of solid wastes. Solid wastes will be processed in accordance with written procedures. A general plan for solid waste processing is to initially identify the waste at the point of generation as to the type of material and exposure rate, and segregate the material to allow for decontamination onsite or packaged for shipment to an offsite vendor for volume reduction or to an approved disposal site.

Standard industry decontamination techniques will be used and may include the following:

1. Strippable Coatings: Strippable coatings may be used to lift particulates from contaminated surfaces. A strippable coating is applied "wet" to a surface in a manner similar to painting a surface. Additives in the coating are designed to attract and combine chemically with radioactive contaminants. Once the coating is dry, the contaminant is locked in the dried coating. The dried coating is easily "peeled" to allow stripping of the film containing the contamination. The stripped film can then be packaged and buried as a solid waste. Strippable coating may also be used to protect surfaces from becoming contaminated.
2. Chemical or Solvent Decontamination: Chemical decontamination is utilized principally for batches and is best used on a production basis for large volumes of similar materials, but may result in a hazardous radiologically contaminated mixed waste. Chemicals used for decontamination will be evaluated for hazardous constituents using 40 CFR and Material Safety Data Sheets (MSDS). Decontamination chemical wastes could possibly include acids, caustics, detergents and non-hazardous solvents. The specific chemical for a particular application will depend on the material to be decontaminated. Acids or bases may be neutralized and solidified or used for water chemistry control in the PCRV water clean-up system. Detergents and other water based solvents will generally be associated with damp rags or wipes. If a mixed waste stream is identified, a treatability study will be performed to determine if it can be made non-hazardous. At this time, no mixed wastes have been identified. Furthermore, no processes are planned to be used during decommissioning that will create a non-treatable mixed waste.
3. Dry Abrasive Impingement: Dry abrasive impingement (e.g., sandblasting) is effective for removing heavy or tightly adhering oxide films.
4. Fixatives: The application of fixatives may be used to fix transferable contamination prior to cutting or packaging.
5. Vacuum Cleaning: HEPA filtered vacuum cleaners may be used in areas of high gross transferable contamination.

3.3.2.6 Onsite Radioactive Waste Volume Minimization

Project management and performance level personnel will incorporate radioactive waste minimization practices into work procedures. Performance indicators will be developed to track total radioactive waste generated during decommissioning. The actual volume of waste generated for an evolution will be compared to the pre-job estimate for that task. Radioactive waste volume reduction and minimization techniques discussed below will be used as appropriate.

1. Personnel Training. Affected personnel will receive Radiation Worker training. This training will identify work techniques to prevent unnecessary contamination of areas and equipment, practices for reuse of materials, and policies to prevent the unnecessary generation of mixed or radioactive wastes.
2. Prevention of Waste. Unnecessary generation of radioactive and mixed wastes will be controlled by procedures established to evaluate and control chemicals brought onsite, and prevent unnecessary packaging, tools and equipment from entering radiologically controlled areas.
3. Reuse of Materials. Typical materials reused during the decommissioning include contaminated tools, equipment and clothing. Contaminated tool and equipment storage and issue areas will be maintained. Protective clothing and collection bags will be laundered, repaired and made available for reuse.
4. Segregation and Packaging. Waste material after collection will be identified at the point of generation as to type of material, exposure rate and contamination levels, if known. At the segregation/packaging facilities, the waste will be further segregated as to form and expected end process. Liquid wastes will be separated from solid wastes.

3.3.2.7 Onsite Waste Packaging

Radioactive waste packaging at Fort St. Vrain will be performed in areas that minimize radiation exposure to personnel, control the spread of contamination, and are adequate for packaging activities. Examples of potential onsite waste packaging areas are:

- Reactor Building refueling floor
- Hot Service Facility

- Compressor rooms (Reactor Bldg., El. 4740')
- Fuel Storage Building
- Temporary facilities designated for waste packaging

Waste packages will be selected for each waste stream that meet the requirements for transportation and disposal. Examples of the waste containers that may be used are drums (52-gallon, 55-gallon), boxes (2'x4'x6', 4'x4'x6'), liners, high integrity containers (HIC's), sea/land containers, shielded casks, and other specialty containers. The capacity and weight limitations of each container are governed by the activity levels, form and classification of the enclosed materials. The waste container to be used will be determined by the size, weight, classification, and activity level of the material to be packaged. Guidance for selection of appropriate packaging will be provided in radioactive waste procedures. In all cases, packaging selected will comply with requirements specified by 49 CFR, 10 CFR 71, and the Disposal Facility Site Criteria, as applicable.

To the maximum extent practicable, voids in disposal containers will be filled with other decommissioning debris. This will reduce the total volume of waste for disposal. Therefore, since voids in packages are filled with wastes that would otherwise be packaged separately for burial, a superior waste form is produced, efficiency is maximized, and project cost, disposal site allocation usage, and transportation risk are minimized. Alternatively, the onsite use of a mobile super compactor may be a cost effective means of volume reduction. After appropriate waste segregation and packaging have occurred, the waste will be transported directly for disposal or transported to an offsite licensed facility for further processing and final disposition.

3.3.2.8 Offsite Shipments of Radioactive Materials for Further Processing

Cost benefit analyses will be performed to determine if it is more cost efficient to process certain radioactive materials at an offsite facility specializing in the treatment of these materials. Based on the results of these analyses, a significant amount of radioactive material generated during the decommissioning project may be shipped to a licensed volume reduction facility.

Methods described below are examples of volume reduction processes that may be employed.

1. Incinerable material may be transferred to a licensed incinerator facility for burning. This may include such materials as paper, certain plastics, lubricating oils and solvents. When required by regulations, EPA characteristic tests (or other analyses) will be performed to verify acceptability of a material for incineration.
2. Low specific activity metals may be transferred to suitably licensed facilities for either melting and consolidation, or decontamination and release. A variety of decontamination options exist including abrasive (grit blasting), chemical and ultrasonic cleaning methods.
3. Volume reduction by compacting or super-compacting.

Waste packages sent to offsite facilities will primarily be sea/land containers selected to meet the requirements of transportation and receipt at the offsite processing facility. Voids in transport containers are not a critical concern. However, efficient management of transportation resources will be an important consideration to minimize project costs and reduce the total number of shipments made. Only radioactive materials that are acceptable according to the individual license(s) of the receiving facility will be transported to that offsite processing facility.

Radioactive material control and accountability procedures to accurately track material originating from Fort St. Vrain during receipt, sorting, processing, and packaging for disposal will be developed and implemented. Only offsite processing facilities which provide adequate radioactive material control and accountability procedures will be selected to perform decontamination, volume reduction or waste processing services.

3.3.3 Radioactive Waste Disposal

3.3.3.1 Program Description

The radioactive waste disposal program will follow 10 CFR 20 and 10 CFR 61, the disposal site criteria, and other applicable Federal and State regulations. Radioactive waste processing, packaging, and shipping activities at Fort St. Vrain will be performed in accordance with written procedures. Similar operations performed at offsite facilities will be controlled as directed by local requirements and specific facility licenses. Radioactive waste may be stored onsite, subject to approved safety evaluations, storage and separation criteria established in Section 3.4 and 10, and applicable State or NRC guidance.

Prior to January 1993, low level radioactive waste (LLRW) generated by FSV decommissioning was transported to the Beatty, Nevada facility for disposal. Beginning in January 1993, the Rocky Mountain Compact was permitted access to the existing Northwest Compact disposal facility, and LLRW from FSV decommissioning was transported to the Richland, Washington facility. It is anticipated that the agreement between the Rocky Mountain Compact Board and the Northwest Compact Board will remain in effect through the duration of decommissioning.

GTCC waste, if any, will be stored in the adjacent ISFSI or in a structure which meets the design requirements to handle GTCC waste. The waste will be stored until such time as it can be transported to a facility licensed to accept the GTCC waste.

3.3.3.2 Projected Radioactive Waste Generation

Tables 3.3-1 and 3.3-2 identify the radioactive wastes that may be shipped for further processing. The pre-volume reduction totals and estimated number of waste containers are delineated on Tables 3.3-3 and 3.3-4.

The initial estimate of the processed and volume reduced radioactively contaminated waste for disposal is 100,072 cubic feet, with 99,219 cubic feet from the PCRV and associated operations, and 853 cubic feet from the balance of plant (BOP). The waste from the PCRV consists of activated concrete, graphite blocks, other activated components, miscellaneous equipment and piping, and concrete rubble. PCRV waste is contaminated principally with Fe-55, tritium, and Co-60. The waste from the BOP consists of tanks, pumps, HVAC filters, and miscellaneous equipment and piping. There may also be radioactively contaminated asbestos. After processing and volume reduction, it is estimated that the volume of radioactive waste will be segregated into the following categories:

CLASS	VOLUME (CUBIC FEET)
A	79,157
B	20,279
C	636

Due to uncertainties in the analysis, the potential exists that some Class C wastes may be reclassified as GTCC, though this is considered unlikely. Waste volume estimates will change as ongoing planning and decommissioning operations proceed.

3.3.3.3 Classification of Radioactive Wastes

Classification of radioactive waste is required by 10 CFR 20, 10 CFR 61, and disposal site requirements.

A Waste classification compliance program will be developed and implemented to assure proper classification of waste for disposal. This program will ensure that a realistic representation of the distribution of radionuclides in waste is known and that waste classification is performed in a consistent manner. Any of the following basic methods, used individually or in combination, will be used to achieve this goal: materials accountability (including process knowledge and activation analysis), classification by source, gross radioactivity measurements, and measurement of specific radionuclides.

3.3.3.4 Transportation Plan

Packages and packaging for radioactive materials and waste will meet all applicable regulations and requirements.

Before packaging waste for shipment from Fort St. Vrain, each package will be inspected to ensure it meets all applicable design and/or certification requirements and that it is not damaged or impaired. A bar code capable of being read by computerized scanners will typically be affixed to the container and the corresponding lid as an aid to inventory and to track individual containers.

The majority of radioactive material and waste shipments performed during the decommissioning project will be done by truck. In some cases, approved shielded casks will be employed due to radiation levels or limits for quantities of radioactivity in a package. Tables 3.3-3 and 3.3-4 provide preliminary estimates for the number of packages anticipated during the decommissioning project. Tables 3.3-5 and 3.3-6 provide preliminary estimates for the types of packages anticipated during the decommissioning project.

Transportation surveys and documents will be prepared prior to any shipment offsite. To determine isotopic inventory and concentration for classification, onsite personnel will assess each loaded shipping container prior to transport.

The actual routing of shipments may vary with weather and highway conditions. Additionally, local and state restrictions pertaining to radioactive material transport may affect some route selections, particularly in congested metropolitan areas. The carrier is responsible for selecting the appropriate route, which must conform to applicable federal, state, and local shipping, in accordance with DOT and NRC regulations.

3.3.3.5 Mixed Waste Contingency Plan

Except for lead shielding, no sources of mixed waste are known to exist onsite. No chemicals or other substances are anticipated to be used during decommissioning operations that may become hazardous wastes. It will be necessary for project management to authorize the use of any chemical or other substance that may become hazardous waste. If mixed waste is identified, it will be classified and stored onsite until regulations allow declassification or disposition.

If mixed wastes are generated, they will be managed according to Subtitle C of RCRA to the extent it is not inconsistent with NRC handling, storage and transportation regulations.

If technology, resources and approved processes are available, PSC and the Westinghouse Team will evaluate the processes for rendering mixed waste "non-hazardous" to determine its adaptability to Fort St. Vrain decommissioning activities. PSC does not intend to petition the EPA to delist any mixed waste. However, if PSC determines it is necessary to delist any mixed waste, the procedures outlined in 40 CFR 260.20 and 260.22 will be used to exclude that waste form from regulations.

3.3.3.6 Waste Storage Facilities

Waste storage facilities planned for use during decommissioning activities include:

1. The Independent Spent Fuel Storage Installation (ISFSI) may be used for greater than Class C wastes (GTCC), if any, pending approval of an appropriate disposal site. (No GTCC wastes are currently expected.)
2. The Fuel Storage Building may be used as a processing and storage area for dry low level wastes.
3. The Compactor Building may be used as a processing and storage area for dry and dewatered low level wastes.
4. The Reactor Building may be used for the storage of liquid and solid wastes.
5. Trailers and sea/land containers may be stored and used onsite to house dry and solidified low level waste.
6. Selected yard areas may be used for short term storage of packaged waste staged for transport.

The activity levels of wastes stored in these areas will be controlled to levels as evaluated in an accident analysis.

Safety evaluations have been performed that assess and permit storage of low level radioactive waste on the Fort St. Vrain site consistent with the guidelines of NRC Generic Letter 81-38 (Ref. 50) and the Standard Review Plan (NUREG-0800), Appendix 11.4-A (Ref. 51). The Fort St. Vrain Technical Specifications permit possession and use of byproduct, source, and special nuclear material in quantities as required pursuant to 10 CFR 30, 40 and 70.

Due to the building seismicity and other drainage and collection requirements for the storage of wet radwaste, PSC does not intend to store wet/liquid radwaste outside the Reactor Building. The Reactor Building was designed and built with drainage systems that route spillage to collection points/sumps that are monitored for radioactivity and properly processed. Other forms of radwaste may also be stored in the Reactor Building without significant concern, due to the building's additional features relative to fire detection and suppression, and its filtered ventilation system.

The Compactor Building is a steel building constructed on a concrete foundation, with its own "wet pipe" fire suppression and fire detection systems. This building has two concrete basins that may be used to store barrels of dewatered wastes, consistent with the recommendation of NRC Generic Letter 81-38 (Ref. 50). Other dry and solidified wastes may be stored in this building in amounts consistent with limitations of the decommissioning accident analyses. A radwaste compactor, with a self-contained HEPA-filtered ventilation system, is also housed in this building.

The Fuel Storage Building may also be used to store packaged dry and solidified low-level radwaste. A safety evaluation has determined that no increase in an accident probability will result from radwaste storage in this location. As stated in the decommissioning accident analysis, a fire detection system will be provided before combustible radwaste can be stored in the Fuel Storage Building.

Trailers and sea/land containers have been evaluated to house dry and solidified radwaste. Accident scenarios have been postulated and the total allowable activity levels for storage are controlled accordingly. Yard fire hydrants are available for use if necessary.

Certain large radioactive components (such as helium circulators packaged for shipment) may be stored outside within the restricted area while awaiting shipment offsite. Tie-down systems will be considered for components stored outside, and will be installed when needed. Steps will be taken to protect containers from external corrosion as required.

3.3.4 Disposal of Non-Radioactive Waste

Non-radioactive wastes will be disposed of by release to appropriate disposal facilities such as land fills, scrap yards and scrap recovery facilities. Materials that are inappropriate for surface surveys, such as resin fines, will be sampled and appropriately analyzed. Materials found to be non-contaminated will be disposed of as non-radioactive waste.

INTENTIONALLY LEFT BLANK

SECTION 4
FINAL RADIATION SURVEY PLAN

4.1 DISCUSSION OF FINAL RADIATION SURVEY PLAN

The purpose of the final radiation survey will be to demonstrate the effectiveness of the decommissioning and to provide documentation that contaminated materials, structures, areas and components have been successfully removed/decontaminated to acceptable levels to permit release for unrestricted use. The final radiation survey to release the Fort St. Vrain site, facilities and installed equipment for unrestricted use will be performed following the completion of the decontamination and dismantlement activities. Materials and equipment determined to be free of radioactive contamination will be unconditionally released on an on-going basis.

All radiological surveys will be conducted in accordance with approved procedures using techniques that determine the effectiveness of a particular dismantlement and/or decontamination effort. These surveys will indicate when no further decontamination is needed and indicate that the equipment, area or structure has been prepared for unrestricted release.

A Final Radiation Survey Plan was originally included in this Section 4 of the Decommissioning Plan. However, the earlier plan has been superseded by the NRC-approved Final Survey Plan, as described below. The NRC-approved Final Survey Plan is referenced by the DP, though no longer included in its entirety.

The NRC requested changes to the Final Radiation Survey Plan that was originally included in the DP to bring it in line with current regulatory guidance and industry experience. In February 1994, PSCo submitted a revised FSV Final Survey Plan for Site Release (Reference 1 - FSP). This FSP reflected industry experience, NRC discussions, modified total surface contamination release criteria for tritium and iron-55 (Reference 2), and the more recent regulatory guidance in Draft NUREG/CR-5849 (Reference 3), in lieu of the guidance in NUREG/CR-2082 (Reference 4). The NRC submitted several requests for additional information (RAI) regarding this FSP in References 5, 6 and 7, to which PSCo responded in References 8, 9 and 10. The NRC approved the Reference 1 FSP, as supplemented by PSCo's responses to the NRC's RAIs, in January 1995 (Reference 11). At the NRC's request, PSCo incorporated applicable information from the RAI responses into the FSP, and submitted the Updated FSP in Reference 12. In addition to RAI responses, the Updated FSP reflected modified release criteria of removable surface contamination for tritium and iron-55, approved by the NRC in Reference 13.

The approved FSP is controlled and maintained in accordance with the document control system, with revisions approved by the Decommissioning Safety Review Committee. As stated in the Reference 12 approved FSP, revisions to the FSP may be implemented without prior NRC approval, provided the changes do not:

- Involve an unreviewed safety question as defined in 10 CFR 50.59 and do not require a change in the Decommissioning Technical Specifications,
- Reduce the required survey frequency for the classification of the survey unit,
- Increase the action levels for conducting investigation and followup surveys, or
- Affect the statistical treatment of survey data in a manner which could reduce the confidence that the site meets the criteria for unrestricted use.

Proposed changes to the FSP that do not meet the above criteria are submitted to the NRC for approval prior to implementation.

The FSP describes the proposed methodology and criteria that will be used in performing the final surveys. This includes definition of residual radioactivity limits (including background evaluation), radiation survey methods, material release criteria and site release criteria, instrumentation and documentation.

4.2 REFERENCES FOR SECTION 4

1. PSC Letter, Warembourg to Austin, dated February 17, 1994, "Final Survey Plan for Site Release," (P-94019).

2. NRC letter, Pittiglio to Crawford, dated June 15, 1994, "Approval of Modification of Facility Release Criteria for Tritium and Iron-55 Surface Contamination at Fort St. Vrain Nuclear Generating Station," (G-94113).

3. NUREG/CR-5849: "Manual for Conducting Radiological Surveys in Support of License Termination," ORAU-92/C57, Draft Report, June 1992.

4. NUREG/CR-2082: "Monitoring for Compliance with Decommissioning Termination Survey Criteria." (ORNL/HASRD-95). June, 1981.

5. NRC Letter, Pittiglio to Crawford, received June 8, 1994, "Review of the Final Survey Plan for Sit Release for Fort St. Vrain Nuclear Generating Station," (G-94100).

6. NRC Letter, Pittiglio to Crawford, dated December 7, 1994, "Response to NRC Comments on the Final Survey Plan for Site Release for Fort St. Vrain Nuclear Generating Station," (G-94201).

7. NRC Letter, Pittiglio to Crawford, dated January 17, 1995, "Response to NRC Comments on the Final Survey Plan for Site Release for Fort St. Vrain Nuclear Generating Station," (G-95005).

8. PSC Letter, Warembourg to Austin, dated September 21, 1994, "Response to Comments Regarding the Final Survey Plan for Site Release," (P-94080).

9. PSC Letter, Fisher to Austin, dated January 11, 1995, "Response to Comments Regarding the Final Survey Plan for Site Release," (P-95005).

10. PSC Letter, Fisher to Austin, dated January 20, 1995, "Response to Comments Regarding the Final Survey Plan for Site Release," (P-95011).

11. NRC Letter, Pittiglio to Crawford, dated January 26, 1995, "Response to NRC Comments on the Final Survey Plan for Site Release for Fort St. Vrain Nuclear Generating Station," (G-95020).

12. PSC Letter, Fisher to Weber, dated May 25, 1995, "Updated Fort St. Vrain Final Survey Plan for Site Release," (P-95050).

13. NRC letter, Pittiglio to Crawford, dated January 18, 1995, "Response to Proposed Modification of Removable Surface Contamination Release Criteria of Removable Surface Contamination for Tritium and Iron-55 at Fort St. Vrain Nuclear Generating Station," (G-95019).

SECTION 8
DECOMMISSIONING ACCESS CONTROL PLAN

8.1 BASIS FOR ACCESS CONTROL PROGRAM

The Fort St. Vrain Decommissioning Access Control Plan is based on 10 CFR 20.105 and NRC Regulatory Guide 1.86 (Ref. 1) which requires isolation of radioactive materials remaining on site from public access to preclude inadvertent exposure to hazardous levels of radiation and administrative procedures for notification and reporting abnormal occurrences.

The Fort St. Vrain Decommissioning Access Control Plan provides adequate industrial security measures to assure public health and safety are not endangered by inadvertent access to the decommissioning site, provides support to implement the Radiation Protection Plan and the Decommissioning Emergency Response Plan, minimizes corporate liability during decommissioning, and establishes controls and procedures related to access to the decommissioning site. Movements by personnel within the decommissioning site and access control requirements for radiologically controlled areas are addressed in Section 3.2, Radiation Protection Program.

8.2 SITE ACCESS CONTROL ORGANIZATION

The PSC Radiation Protection Manager is the designated site representative responsible for implementing the FSV Access Control Program. The PSC Radiation Protection Manager is also the designated site representative responsible for coordination with local law enforcement authorities. Familiarization briefings on access procedures, plant layout, and emergency arrangements will be made as requested by the local law enforcement agency.

Access control personnel will be properly trained and demonstrate understanding of decommissioning area access control requirements and responsibilities. Access control personnel will be unarmed and equipped for continuous onsite and offsite communications.

8.3 ACCESS CONTROL PHYSICAL SECURITY MEASURES

The decommissioning area will be surrounded by a physical barrier to inhibit unauthorized access to restricted areas. This barrier shall be inspected at least quarterly for deterioration or breaches and that applied locks and locking apparatus are intact. Repairs to physical barriers and equipment will be accomplished in a timely manner. All portals and gates associated with decommissioning area physical barriers will be kept locked or continuously monitored by access control personnel and all locks and keys will be controlled by access control personnel.

Personnel access point will be located at the main plant entry for the decommissioning area. Even though other access points may be established as needed, this gate will generally serve as the primary personnel access portal.

Vehicle access gates will be established as needed to accommodate decommissioning activities.

Access for decommissioning workers will be authorized by project management. A list of individuals granted authorized access will be prepared and maintained. Access control personnel will verify authorized access for each individual before granting passage through the physical barrier surrounding the decommissioning area. Visitor access to the decommissioning area will be controlled in a manner consistent with the Radiation Protection Plan and Emergency Response Plan.

Personnel accountability, including accountability for visitors, in the event of an emergency will be maintained through the Access Control Program in support of Emergency Response Plan requirements.

Access to the restricted area does not guarantee access to radiological controlled areas. The radiation protection staff will administer the radiological controlled area access control program. Specific requirements that must be met prior to accessing radiologically controlled areas are identified in Section 3.2.

8.4 COMMUNICATIONS

Telephone service will be available for access control personnel to contact local law enforcement authorities in the event of an abnormal occurrence and to make necessary notifications to decommissioning site management. Radio communications between the FSV control room and access control personnel will also be provided to supplement telephone service, especially during emergencies.

8.5 PROCEDURES

Written procedures will be prepared and implemented to provide the access control personnel guidance for routine occurrences. Examples include:

1. Personnel access control
2. Vehicle access control
3. Communications equipment and routine testing requirements
4. Surveillance/inspection of decommissioning area physical barriers

Written procedures will be prepared and implemented to provide the access control personnel guidance for abnormal occurrences. Examples include:

1. Personnel disturbance
2. Acts or perceived threat of sabotage
3. Civil disturbance
4. Suspected or confirmed sabotage or intrusion attempt
5. Breached security area barrier
6. Unidentified person in security area
7. Site evacuation

The content of abnormal occurrence procedures will include (1) criteria for identifying abnormal conditions within the decommissioning area; (2) access control personnel actions; and (3) required notifications. Guidance to respond to other abnormal occurrences such as fire, explosion, site radiological emergencies, and medical emergencies is contained in emergency plan procedures.

8.6 REFERENCES FOR SECTION 8

1. Regulatory Guide 1.86: "Termination of Operating License for Nuclear Reactors". June 1974.

INTENTIONALLY LEFT BLANK